Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	003 K4.07	
	Importance Rating	3.2	

Knowledge of RCPS design feature(s) and/or interlock(s) which provide for the following: Minimizing RCS leakage (mechanical seals)

Proposed Question: Common 1

Which ONE (1) of the following components acts to limit Reactor Coolant System leakage when the vapor seal fails on a Reactor Coolant Pump?

(Assume the lower, middle and upper seals are intact and functioning properly.)

A. Rotating baffle and shaft hydrostatic bushing.

B. Shaft hydrostatic bushing and O-rings.

C. Seal breakdown coil and rotating seal face.

D. Controlled bleedoff excess flow check valve and stationary carbon rings.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because both components act to limit seal flow, however, given the conditions described (normal seals intact) these components manage gross seal leakage via multiple seal failures.

B. Incorrect. Plausible because both components act to limit seal flow, however, given the conditions described (normal seals intact) the O-rings only act as seals for individual seal package components.

C. Correct. Given the conditions of having the normal seals intact, these components limit flow out the vapor seal (seal breakdown coil is ~1 gpm and seal face is negligible but sufficient to maintain cooling).

D. Incorrect. Plausible because with a check valve failure the flow out a vapor seal could increase RCS leakage, however, the stationary carbon rings only partially act to limit flow through a seal face.

Technical Reference(s) SD-SO23-360, page 22 (Attach if not previously provided)
SO23-13-6, Attachment 2

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 94463 (As available)

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 3

Comments:

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

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

Proposed Question: Common 2

Which ONE (1) of the following sets of conditions describes the automatic operation of the Reactor Coolant Pump Lube Oil Pumps?

A. The Normal Oil Lift Pump starts when RCP speed is less than 90%.
The Normal Anti-Reverse Rotation Device Pump starts when RCP speed is less than 90%.

B. The Normal Oil Lift Pump starts 15 seconds after the RCP is secured.
The Normal Anti-Reverse Rotation Device Pump starts when the RCP reaches zero speed.

C. The Normal Oil Lift Pump starts when RCP speed is less than 90%.
The Normal Anti-Reverse Rotation Device Pump starts when the RCP reaches zero speed.

D. The Normal Oil Lift Pump starts 15 seconds after the RCP is secured.
The Normal Anti-Reverse Rotation Device Pump starts when RCP speed is less than 90%.

Proposed Answer: A

Explanation (Optional):

A. Correct.

B. Incorrect. Plausible because the Standby Oil Lift Pump will start in this condition, however, the ARRD Pump would start before zero speed is reached. Note that the ARRD is not used until zero speed is achieved.

C. Incorrect. Plausible because the Oil Lift Pump portion is correct, however, the ARRD Pump would start before zero speed is reached. Note that the ARRD is not used until zero speed is achieved.

D. Incorrect. Plausible because the ARRD Pump portion is correct, however, only the Standby Oil Lift Pump will start in this condition.

Technical Reference(s) SD-SO23-360, pages 29 & 31 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 94468 (As available)

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	004 K3.02	
	Importance Rating	3.7	

Knowledge of the effect that a loss or malfunction of the CVCS will have on the following: PZR LCS

Proposed Question: Common 3

Given the following conditions:

- Unit 3 is in MODE 1 at 50% power at MOC.
- 3TV-0223, Letdown Heat Exchanger Temperature Control Valve fails closed.
- All control systems are in Automatic.
- Assume NO operator action.
- 3TV-0224B, Ion Exchanger Bypass Valve fails to go to BYPASS.

Which ONE (1) of the following describes the initial effect on the Pressurizer Level Control System?

Actual Pressurizer level __________ and Pressurizer level setpoint __________.

A. remains the same; remains the same

B. lowers; remains the same

C. remains the same; lowers

D. lowers; lowers

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible if failure to recognize actions associated with rising LD temperature.

B. Incorrect. Plausible and partially correct, however, the lowering of PZR level is a direct result of the changing level setpoint due to temperature.

C. Incorrect. Plausible and partially correct, however, with no auto actions RCS boron will continue to increase causing temp to lower and eventually driving down PZR level.

D. Correct. Letdown temp rises → RCS boron rises → RCS Tavg lowers → PZR level setpoint lowers → PZR level lowers.

Technical Reference(s) SD-SO23-390, page 42 (Attach if not previously provided)
SO23-3-2.1, L & S 3.2

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 56419 & 52742 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	005 A4.02	
	Importance Rating	3.4	

Ability to manually operate and/or monitor in the control room: Heat exchanger bypass flow control

Proposed Question: Common 4

Given the following conditions:

- Unit 2 is in MODE 5 on Shutdown Cooling (SDC) in the normal alignment.
- RCS temperature is being maintained constant at 150ºF.
- It is determined that SDC Pump (P-015) flow is too high for the current condition.

Which ONE (1) of the following is the preferred alignment for reducing the flowrate through the pump?

A. Ensure HV-8161, SDC HX Bypass Normal Block Valve is closed and then throttle closed HV-8160, SDC HX Bypass Normal Flow Control Valve.

B. Ensure HV-0396, SDC HX Bypass Standby Flow Control Valve is closed and then close HV-8161, SDC HX Bypass Normal Block Valve.

C. Ensure HV-0396, SDC HX Bypass Standby Flow Control Valve is closed and then throttle closed HV-8160, SDC HX Bypass Normal Flow Control Valve.

D. Ensure HV-8160, SDC HX Normal Bypass Flow Control Valve is closed and then throttle closed HV-0396, SDC HX Bypass Standby Flow Control Valve.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because HV-8160 is throttled closed; however, flow would be totally blocked given valve configuration.

B. Incorrect. Plausible because HV-0396 is closed, however, when HV-8161 is closed there is no HX bypass flow. Candidate must know that these valves are not in the same line.

C. Correct. This is the guidance outlined in SO23-3-2.6, SDC Operations.

D. Incorrect. Plausible because this configuration would limit flow, however, HV-8160 is electrically blocked 10% open and cannot be fully closed and only one of the valves is normally in service.

Technical Reference(s) SO23-3-2.6, Step 6.2.7 (Attach if not previously provided)
SD-SO23-740, Figure 1
SO23-3-2.6, Attachment 1

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 53010 (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 3,7,10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	006 A1.05	
	Importance Rating	2.9	

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ECCS controls including: CCW flow (establish flow to RHR heat exchanger prior to placing in service)

Proposed Question: Common 5

Given the following conditions:

- Unit 3 is in MODE 5 with Shutdown Cooling (SDC) in service.
- RCS temperature is 190ºF.
- Train A SDC Heat Exchanger (HX) is in service and Train B SDC HX is in standby.
- Train B CCW is in standby.

Which ONE (1) of the following identifies the impact to the standby HX?

The Train B SDC HX…

A. will heat up to RCS temperature due to slow leakby through the SDC HX outlet valve.

B. maintains ambient conditions of SDC HX Room until the SDC HX outlet valve is opened.

C. SDC inlet valve must remain closed to prevent overpressurizing the HX.

D. may undergo potential tube damage due to ambient cooldown while isolated.

Proposed Answer: A

Explanation (Optional):

A. Correct. See SO23-3-2.6, L & S 3.3.

B. Incorrect. Plausible because when the outlet valve is opened the temp will rise, however, ambient conditions will not be maintained due to leak by of the outlet valve.

C. Incorrect. Plausible if it is not known that the inlet valve is normally open, however, in this condition the HX cannot over pressurize because the CCW inlet is open.

D. Incorrect. Plausible because this could occur if proper alignment is unknown.

Technical Reference(s) SO23-3-2.6, L & S 3.3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 52692 (As available)

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 7, 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	006 K2.04	
	Importance Rating	3.6	

Knowledge of bus power supplies to the following: ESFAS-operated valves

Proposed Question: Common 6

Unit 3 has experienced a Loss of Offsite Power and a LOCA.

Which ONE (1) of the following Motor Control Center combinations will be available to power their respective ESFAS Valves?

A. 3BRA; 3BRB; 3BRC

B. 3BDX; 3BHX; 3BMX

C. 3BJ; 3BY;3BZ

D. 3BD; 3BE; 3BF

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible as 3BRA and 3BRB are powered from 1E Buses 3B04 and 3B06 respectively, however, 3BRC is powered from Bus 3B18 which is powered from Non-1E Bus 3A07.

B. Incorrect. Plausible as 3BDX and 3BHX are located in the Emergency Diesel Rooms for 3G002 and 3G003 respectively, however, 3BHX is powered from 3BDX which is powered from Non-1E Bus 3B14. 3BMX is powered from Non-1E Bus 3B13.

C. Correct. 3BJ and 3BZ are powered from 3B06 while 3BY is powered from 3B04.

D. Incorrect. Plausible as 3BD and 3BE are powered from 1E Bus 3B04, however, 3BF is powered from Bus 3B03 which is powered from Non-1E Bus 3A03.

Technical Reference(s) LP 2XE102 Handout (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 79744 (As available)

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

Question History: Last NRC Exam ____________

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

10 CFR Part 55 Content: 55.41 8

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	007 K5.02	
	Importance Rating	3.1	

Knowledge of the operational implications of the following concepts as the apply to PRTS: Method of forming a steam bubble in the PZR

Proposed Question: Common 7

Which ONE (1) of the following sets of conditions must be established within the Quench Tank prior to drawing a Pressurizer bubble?

A. Oxygen concentration must be maintained < 1% to prevent an explosive mixture from forming in the Quench Tank.

B. Oxygen concentration must be maintained < 5% in the event the Pressurizer or Reactor Head Vent Valves are opened.

C. Nitrogen pressure must be maintained > 1 psig to ensure a motive force for fluid to the Reactor Coolant Drain Tank.

D. Nitrogen pressure must be maintained > 5 psig to allow for purging of the Quench Tank.

Proposed Answer: A

Explanation (Optional):

A. Correct. This condition must be established in the QT prior to bubble formation.

B. Incorrect. Plausible because the valves could be opened during the RCS Fill and Vent procedure, however, the oxygen concentration is too high.

C. Incorrect. Plausible since the Quench Tank does gravity drain to the RCDT, however, the motive force is established by level (75 to 80%) and not pressure.

D. Incorrect. Plausible as purging is required if O_2 exceeds 1%, however, there is no established limit for N_2 pressure when drawing a bubble.

Technical Reference(s) SO23-3-1.4, Attachment 5, Step 2.6 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 94469 (As available)

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

Question History: Last NRC Exam ____________

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

10 CFR Part 55 Content: 55.41 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	008 K1.05	
	Importance Rating	3.0	

Knowledge of the physical connections and/or cause-effect relationships between the CCWS and the following systems: Sources of makeup water

Proposed Question: Common 8

Given the following conditions:

- Unit 3 is in MODE 1 with Train A Component Cooling Water System in service.
- Normal makeup was initiated by the RO in response to a low level in the Train A Surge Tank.
- The Non-Critical Loop Isolation valves automatically closed during the makeup evolution.

Which ONE (1) of the following caused the Non-Critical Loop Isolation?

A. CCW Rad Monitor RE-7819 detected radiation in the Non-Critical Loop.

B. The Train A Component Cooling Water Surge Tank was overfilled.

C. A Safety Injection Actuation Signal has occurred.

D. Low suction pressure trip on the operating CCW Pump.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because this detector does monitor the radiation levels in the Non-Critical loop (NCL), however, it is alarm and indication only.
B. Correct. Overfilling of the surge tank fills the reference leg and creates a low-low level condition that shuts the NCL valves.
C. Incorrect. Plausible because the NCL valves do respond to ESFAS signals, however, it is a CIAS that will close these valves.
D. Incorrect. Plausible because this feature was once incorporated in the CCW System.

Technical Reference(s)	SO23-13-7, Attachment 1	(Attach if not previously provided)
	SD-SO23-690, page 8	

Proposed references to be provided to applicants during examination: NONE

Learning Objective:	81028	(As available)

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

Question History:	Last NRC Exam	

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

10 CFR Part 55 Content:	55.41	7, 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	008 A3.05	
	Importance Rating	3.0	

Ability to monitor automatic operation of the CCWS, including: Control of the electrically operated, automatic isolation valves in the CCWS

Proposed Question: Common 9

Given the following conditions:

- Unit 2 has tripped from 100% power.
- Containment pressure is 3.6 psig and slowly rising.
- NO operator actions have been taken.

Which ONE (1) of the following identifies the automatic actions of the Component Cooling Water System?

A. SWC Outlet Valves from the standby CCW Heat Exchanger close.

B. CCW Supply and Return Lines to the Containment Normal Coolers open.

C. CCW Non-Critical Loop Isolation Valves close.

D. SDC Heat Exchangers CCW Outlet Valves open.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because the standby loops CCW and SWC Pumps will be off, however, a SIAS will start the CCW Pumps in both trains which will open the SWC valves.

B. Incorrect. Plausible because a high Containment pressure opens the Emergency Cooling Unit (ECU) CCW valves, however, the Containment Normal Coolers are cooled by chilled water which is cooled by TPCW.

C. Correct. A CIAS signal will close the NCL Valves.

D. Incorrect. Plausible because a high Containment pressure will open the valves, however, current pressure is insufficient to create a CSAS signal.

Technical Reference(s) SD-SO23-400, Section 2.1.5.2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 81029 (As available)

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	010 K1.06	
	Importance Rating	2.9	

Knowledge of the physical connections and/or cause-effect relationships between the PZR PCS and the following systems: CVCS

Proposed Question: Common 10

Given the following conditions:

- Unit 2 is in MODE 5.
- Drawing a Pressurizer bubble is in progress.
- RCS temperature is 150ºF.
- The crew has just verified formation of a Pressurizer bubble.
- PV-0201A and PV-0201B, Letdown Backpressure Control Valves are fully open.
- RCS pressure is continuing to rise.

Which ONE (1) of the following will act to mitigate the RCS pressure rise?

A. Stop all but one Charging Pump to prevent a loss of Letdown flow.

B. Secure Charging Pumps and/or Pressurizer heaters to maintain pressure.

C. Throttle Letdown Flow Control Valves LV-0110A and LV-0110B open to allow PV-0201A and PV-0201B, Backpressure Control Valves to pass more flow.

D. Align LV-0227A, Volume Control Tank Inlet Valve to RADWASTE position to minimize backpressure on PV-0201A and PV-0201B.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because this action will offset the pressure rise, however, with the current RCS temperature Letdown flow would not be interrupted with a loss of Charging.
B. Correct. This action will stop or slow the pressure increase.
C. Incorrect. Plausible because throttling the Flow Control Valves may allow the Backpressure Control Valves to open, however, it will not be effective in limiting the RCS pressure rise.
D. Incorrect. Plausible because a lower backpressure will allow the valves to pass more flow but will not be effective in limiting the RCS pressure rise.

Technical Reference(s) SO23-3-1.4, Attachment 4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 94469 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5, 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	012 A3.02	
	Importance Rating	3.6	

Ability to monitor automatic operation of the RPS, including: Bistables

Proposed Question: Common 11

Which of the following Reactor Protection System bistables will automatically change state (trip) upon loss of power to an individual Core Protection Calculator (CPC)?

A. High LPD and low DNBR

B. Low DNBR and high linear power

C. High LPD and high linear power

D. High log power and low DNBR

Proposed Answer: A

Explanation (Optional):

A. Correct.
B. Incorrect. Plausible as low DNBR is correct.
C. Incorrect. Plausible as high LPD is correct.
D. Incorrect. Plausible as low DNBR is correct.

Technical Reference(s) SD-SO23-710, pages 5, 8 and 11 (Attach if not previously provided)
Tech Spec 3.3.1

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 56628 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 2, 6

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	012 K4.04	
	Importance Rating	3.1	

Knowledge of RPS design feature(s) and/or interlock(s) which provide for the following: Redundancy

Proposed Question: Common 12

Which ONE (1) of the following statements describes the response of the Reactor Protection System to a loss of 120 VAC Vital Bus Y-03?

A. 2 of 8 Reactor trip breakers will open because their corresponding trip path deenergizes and the Reactor WILL trip.

B. 2 of 8 Reactor trip breakers will open because they share the instrument AC power supply and the Reactor WILL NOT trip.

C. 4 of 8 Reactor trip breakers will open because 2 trip paths will deenergize and the Reactor WILL NOT trip.

D. 4 of 8 Reactor trip breakers will open because 2 trip paths will deenergize and the Reactor WILL trip.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because two trip paths will deenergize upon loss of the vital bus, however this is accompanied by the opening of 4 RTBs.

B. Incorrect. Plausible because with 4 vital buses and eight RTBs it is implied that two breakers receive power from each bus, however, it is the trip path de-energization that opens 4 breakers. Instrument AC power does not impact this question.

C. Correct.

D. Incorrect. Plausible at it is partially correct, however, two trip paths deenergize.

Technical Reference(s) SO23-13-18, Attachment 4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 56627 (As available)

Question Source:	Bank #	____	
	Modified Bank #	____	(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam	____	
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis	____	
10 CFR Part 55 Content:	55.41	6	
		____	

Comments:

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	013 K3.01	
Importance Rating	4.4	

Knowledge of the effect that a loss or malfunction of the ESFAS will have on the following: Fuel

Proposed Question: Common 13

Given the following conditions:

- A large break LOCA has occurred with a loss of offsite power.
- The rupture occurs on a Loop Cold Leg where SIT T-008 is connected to the RCS.
- RCS pressure immediately lowered to Containment pressure.
- Train A Emergency Diesel Generator has failed to start on an ESFAS signal.
- Train B Emergency Diesel Generator has just received its ESFAS signal to start.
- Safety Injection Tank pressures at the start of the transient were 600 psia.
- Reactor Vessel Plenum Level is 0%.

If this condition continues, which ONE (1) of the following describes the effect on the fuel assemblies?

A. Fuel failure will not occur. SIT injection pressure is insufficient to reflood the core.

B. Fuel failure will not occur. The large break LOCA analysis assumes one SIT does not inject.

C. Fuel failure may occur. SIT injection pressure is insufficient to reflood the core.

D. Fuel failure may occur. The large break LOCA analysis assumes one SIT does not inject.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because SIT pressures are 600 psia at the start of the event, however, this does not meet the Tech Spec minimum.

B. Incorrect. Plausible because the SIT design basis assumes one (1) SIT is lost due to the location of the break, however, this assumes that minimum SIT pressures are met.

C. Correct. With no power to the Safeguards buses due to an ESFAS signal failure on Train A and a delay in actuation of ESFAS on Train B a limited flow from the SITs is available due to the low initial SIT pressure.

D. Incorrect. Plausible because fuel failure may occur, the SIT design basis assumes one (1) SIT is lost due to the location of the break, however, this assumes that minimum SIT pressures are met.

Technical Reference(s) SD-SO23-740, pages 29 & 30 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 53791 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	013 G2.2.22	
	Importance Rating	3.4	

Equipment Control Knowledge of limiting conditions for operations and safety limits.

Proposed Question: Common 14

With Unit 2 operating at 100% power, which ONE (1) of the following conditions would require entry into a Technical Specification one (1) hour Limiting Condition of Operation?

A. A loss of three (3) Matrix Logic channels due to a common power source failure.

B. One (1) High Pressurizer Pressure trip channel is INOPERABLE.

C. A LPSI Pump is declared INOPERABLE following flow testing.

D. RWST temperature drops to 38ºF during a cold snap.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because of ESFAS components affected, however, this is a 48 hour ACTION (TS 3.3.6.A).

B. Correct. Place the channel in bypass or trip in 1 hour (TS 3.3.1.A).

C. Incorrect. Plausible as this is a 7 day ACTION for LPSI but a 1 hour ACTION for HPSI (TS 3.5.2.A).

D. Incorrect. Plausible because the surveillance must be performed with ambient air temperature <40ºF, however, this is an 8 hour ACTION (TS 3.5.4.A).

Technical Reference(s) Tech Spec 3.3.1.A (Attach if not previously provided)
Tech Spec Table 3.3.1-1

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 56636 (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	10	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	022 A2.04	
	Importance Rating	2.9	

Ability to (a) predict the impacts of the following malfunctions or operations on the CCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of service water

Proposed Question: Common 15

Given the following conditions:

- Unit 3 in MODE 2.
- Turbine Plant Cooling Water flow to the running Containment Chiller is blocked.

Which ONE (1) of the following:

a.) Identifies the impact on the Containment Cooling System?

b.) What action should be taken to mitigate the situation?

A. a.) Containment Chiller trips on low TPCW Supply pressure.
b.) Place the Containment Emergency Cooling Units in service when Containment average air temperature exceeds 100ºF per SO23-1-4, Containment Normal Heat Removal.

B. a.) Containment Chill Water Pump trips on Chiller high condenser pressure.
b.) Place the Containment Emergency Cooling Units in service when Containment average air temperature exceeds 105ºF per SO23-1-4.1, Containment Emergency Cooling.

C. a.) Containment Chiller trips on low TPCW Supply pressure.
b.) Place the Standby Containment Chiller in service when Containment average air temperature exceeds 100ºF per SO23-1-4, Containment Normal Heat Removal.

D. a.) Containment Chill Water Pump trips on low suction pressure.
b.) Place the Standby Containment Chiller in service when Containment average air temperature exceeds 105ºF per SO23-1-4.1, Containment Emergency Cooling.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible as the reason for tripping is correct; however, the ECUs are placed in service at 105ºF per SO23-1-4.1.
B. Incorrect. Plausible as the Containment ECUs are directed to be placed in service at 105ºF and the Chiller will trip on high condenser pressure of 161 psig due to a loss of cooling, however, the trips associated with the Chill Water Pumps are low suction pressure, low discharge flow, or a CW Containment Isolation Valve closing.
C. Correct. For the conditions given this is the correct trip and desired action.
D. Incorrect. Plausible as the CW pump will trip on low suction pressure but this parameter is not impacted by the loss of TPCW. Placing the ECUs in service at 105ºF per SO23-1-4.1 is correct.

Technical Reference(s) SO23-1-4, Step 6.15 (Attach if not previously provided)
SD-SO23-770, page 33

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 81638 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 9, 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	026 K2.02	
	Importance Rating	2.7	

Knowledge of bus power supplies to the following: MOVs

Proposed Question: Common 16

Which ONE (1) of the following is the power supply to 2HV-9367, Train A Containment Spray Header Isolation Valve?

A. 2B09

B. 2BE

C. 2B08

D. 2BJ

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible since 2B09 is located in the 63' Penetration Building, however it is powered from Non-1E Bus 2A09.

B. Correct. 2BE is located in the 50' Control Building and powered from 2B04.

C. Incorrect. Plausible since 2B08 is located in the 63' Penetration Building, however it is powered from Non-1E Bus 2A09.

D. Incorrect. Plausible since 2BJ is located in the 50' Control Building, however, it is powered from 1E Bus 2B06.

Technical Reference(s) SD-SO23-740, page 63 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 79744 (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	8	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	039 K3.06	
	Importance Rating	2.8	

Knowledge of the effect that a loss or malfunction of the MRSS will have on the following: SDS

Proposed Question: Common 17

Given the following conditions:

- Unit 2 is at 100% power.
- 2FT-1021, Steam Generator E-088 Main Steam Flow Transmitter input to Steam Bypass Control System (SBCS) fails **low**.

Which ONE (1) of the following identifies the effect on the SBCS?

The Master Controller Remote Setpoint (black and white pen)…

A. rises to 1000 psia.

B. rises to 900 psia.

C. remains at 830 psia.

D. lowers to 650 psia.

Proposed Answer: A

Explanation (Optional):

A. Correct. This is the expected response at 100% power or anytime a steam flow instrument fails low.

B. Incorrect. Plausible because the Remote Setpoint is calculated as a function of steam flow **and** Pressurizer pressure. Because PZR pressure does not change it could be construed that the steam flow failure is minimally impacted.

C. Incorrect. Plausible because this is the Remote Setpoint at 100% power. If a steam flow transmitter fails high, there is no effect on the Master Controller Remote Setpoint because the setpoint is dictated by the highest steam pressure.

D. Incorrect. Plausible as this is the correct response for a steam pressure instrument failing low; however, this indication is on the red pen.

Technical Reference(s) SD-SO23-175, Figures 9 & 13 (Attach if not previously provided)
Lesson Plan 2XIR05 Slide 72

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 54350 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 4, 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	059 G2.4.6	
	Importance Rating	3.1	

Emergency Procedures / Plan Knowledge symptom based EOP mitigation strategies.

Proposed Question: Common 18

Which ONE (1) of the following describes the mitigation strategy for SO23-12-6, Loss of Feedwater?

A. Trip all RCPs; Attempt feedwater restoration; Control Pressure, Inventory and Heat Removal in the Functional Recovery Guidelines; Cooldown as required.

B. Trip one (1) RCP in each loop; Attempt feedwater restoration; Depressurize Steam Generators; Control Pressure, Inventory and Heat Removal.

C. Trip all RCPs; Minimize SG inventory loss; Attempt feedwater restoration; Control Pressure, Inventory and Heat Removal; Cooldown as required.

D. Trip one (1) RCP in each loop; Minimize SG inventory loss; Initiate cooldown at maximum rate; Control Pressure, Inventory and Heat Removal.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because all steps are appropriate with the exception of entering the FRGs for Safety Function control.
B. Incorrect. Plausible because the trip 2 RCPs / leave 2 RCPs running is used throughout the ERGs. Depressurizing the SGs is appropriate if a Condensate Pump is available for service.
C. Correct.
D. Incorrect. Plausible because it contains correct components of the mitigation strategy, however, cooldown is only required if necessary and all RCPs must be tripped.

Technical Reference(s) SO23-14-6, Attachment 1 (Attach if not previously provided)
SO23-14-6, Section 3

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 52745 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	061 K5.03	
	Importance Rating	2.6	

Knowledge of the operational implications of the following concepts as the apply to the AFW: Pump head effects when control valve is shut

Proposed Question: Common 19

Given the following conditions:

- Unit 2 was in MODE 2 at 1% power when a Reactor trip occurred.
- Steam Generator E-089 is at 20% narrow range level and 730 psia
- Steam Generator E-088 is at 19% narrow range level and 720 psia.

Which ONE (1) of the following identifies the discharge pressure of each Steam Generators Auxiliary Feedwater Pump?

AFW Pump P-141 is at __________. AFW Pump P-504 is at __________.

A. ~940 psig; ~930 psig

B. ~1350 psig; ~930 psig.

C. ~940 psig; ~1350 psig.

D. ~1350 psig; ~1350 psig.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible as discharge pressure selected corresponds to full AFW Pump flow with the given SG Pressure.
B. Incorrect. Plausible as discharge pressure selected corresponds to full AFW Pump flow with the given SG Pressure.
C. Incorrect. Plausible as discharge pressure selected corresponds to full AFW Pump flow with the given SG Pressure.
D. Correct. With the conditions stated, both AFW Pumps are at their respective shutoff heads.

Technical Reference(s) SD-SO23-780, page11 (Attach if not previously provided)
SD-SO23-780, Figure 13

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 52374 & 53283 (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 4, 7, 8

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	061 A3.05	
	Importance Rating	2.5	

Ability to monitor automatic operation of the AFW, including: Recognition of leakage, using sump level changes

Proposed Question: Common 20

Which ONE (1) of the following sets of sump level alarms could be used to identify Auxiliary Feedwater System leakage?

A. Penetration, Component Cooling Water

B. Storage Tank, Containment

C. Safety Equipment Building, Control Building

D. Storage Tank, Penetration

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Penetration Building is on the other side of Containment from the AFW penetrations.
B. Correct. System leakage could occur in these areas. AFW penetrates Containment at penetrations #75 and #78. See SO23-3-3.10, Attachment 5.
C. Incorrect. Safety Equipment Building is adjacent to the AFW penetrations; however, the piping does not pass through there.
D. Incorrect. Penetration Building is on the other side of Containment from the AFW penetrations.

Technical Reference(s) SO23-3-3.10, Attachment 5 (Attach if not previously provided)
SD-SO23-780, Figure 1
SD-SO23-670, Figure 2

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 52374 (As available)

Question Source: Bank # ____________

Modified Bank # ____________ (Note changes or attach parent)

New X

Question History: Last NRC Exam ____________

Question Cognitive Level: Memory or Fundamental Knowledge ____________

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 4, 9

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	062 K4.06	
	Importance Rating	2.9	

Knowledge of ac distribution system design feature(s) and/or interlock(s) which provide for the following: One-line diagram of 6.9kV distribution, including sources of normal and alternative power

Proposed Question: Common 21

Which ONE (1) of the following sets of conditions must be met in order for a 6.9 kV bus to automatically transfer from the Unit 2 Unit Auxiliary Transformer to the Unit 2 Reserve Auxiliary Transformer?

A. 2XR3 energized with 6.9 kV supply breaker in AUTO.
2XR3 is not supplying 3A02.
2XU2 and 2XR3 lockout relays are reset.

B. 2XR2 energized with 6.9 kV supply breaker in AUTO.
2XR2 is not supplying 3A01.
2XU2 and 2XR2 lockout relays are reset.

C. 2XR1 energized with 6.9 kV supply breaker in AUTO.
2XR1 is not supplying 3A01.
2XU1 and 2XR1 lockout relays are reset.

D. 2XR3 energized with 6.9 kV supply breaker in AUTO.
2XR3 is not supplying 3A02.
2XU1 and 2XR3 lockout relays are reset.

Proposed Answer: A

Explanation (Optional):

A. Correct.

B. Incorrect. Plausible because the general conditions are met, however, 2XR2 is not available for cross tying to 2XU2.

C. Incorrect. Plausible because the general conditions are met, however, 2XR1 and 2XU1 are not the source of power for the RCP buses.

D. Incorrect. Plausible because the general conditions are met, however, 2XU1 is not the source of power for the RCP buses

Technical Reference(s) SD-SO23-120, Figures 1 & 3 (Attach if not previously provided)
SD-SO23-120, page 26

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 94147 (As available)

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	063 G2.1.12	
	Importance Rating	2.9	

Ability to apply technical specifications for a system.

Proposed Question: Common 22

Train A, Train B, Train C, and Train D 1E DC electrical power subsystems shall be OPERABLE per Technical Specification 3.8.4 to support plant safety systems in which MODES?

A. Only MODES 1 and 2.

B. Only MODES 1, 2 and 3.

C. Only MODES 1, 2, 3 and 4.

D. Only MODES 1, 2, 3, 4, and 5.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because these are 2 of 4 MODES required being OPERABLE.
B. Incorrect. Plausible because these are 3 of 4 MODES required being OPERABLE.
C. Correct. Per Tech Spec 3.8.4.
D. Incorrect. Plausible because MODES 1 to 4 are correct and Tech Spec 3.8.10 requires DC power necessary to support equipment required to be OPERABLE, however, the question does not specify what is going to be performed. Tech Spec 3.8.5, DC Sources - Shutdown only requires MODES 5 & 6 when moving irradiated assemblies.

Technical Reference(s) Tech Spec 3.8.4 (Attach if not previously provided)
Tech Spec 3.8.5
Tech Spec 3.8.10

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 56649 (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 10

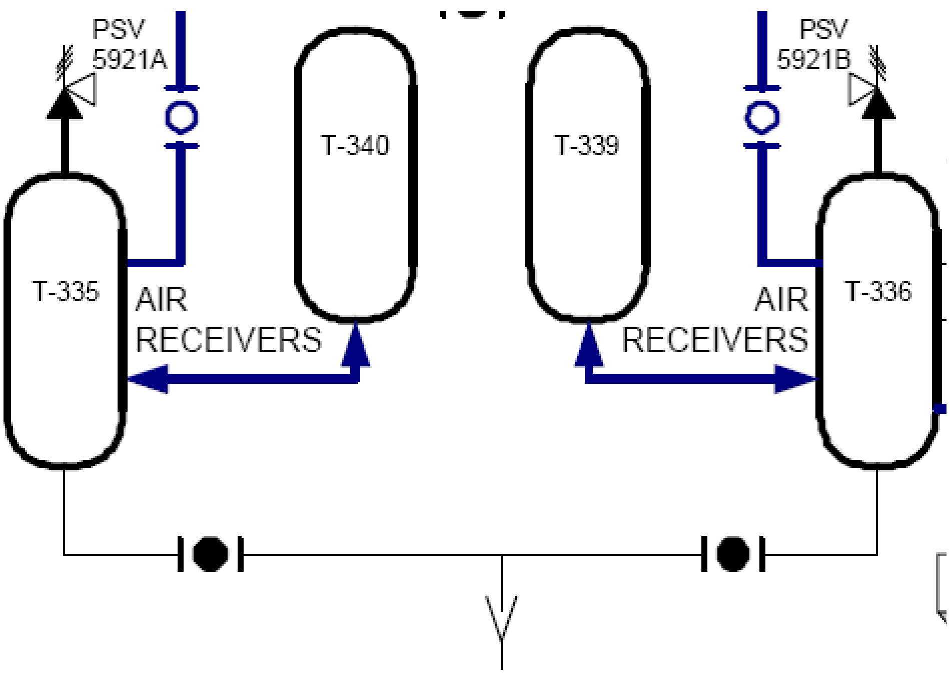
Comments:

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

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

Proposed Question: Common 23

Which ONE (1) of the following identifies the impact to an Emergency Diesel Generator's Starting Air System when the outlet valve from air receivers T-335 and T-340 is isolated?



A. One Air Start Motor on each diesel continues to function.

B. Both sets of Air Start Motors on one diesel continue to function.

C. One set of Air Start Motors on each diesel continue to function.

D. Both sets of Air Start Motors on both diesels continue to function.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because there are 4 Air Start Motors per diesel, however, their air supply is in series and they operate as a pair.

B. Incorrect. Plausible because the EDGs are coupled and when one starts they should both turnover, however, the starting air system is split between EDGs.

C. Correct.

D. Incorrect. Plausible because a cross connect valve is available, however, this valve is normally closed and each pair of receivers supplies a set of air start motors of each diesel.

Technical Reference(s) SD-SO23-750, Figure V-1 (Attach if not previously provided)
SD-SO23-750, page 108

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 55462 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 8

Comments:

Drawing is included to prevent confusion as to what is meant by isolating a “pair of air receivers.” In the EDG Room T-335 is on the 30’ level along with T-336. T-339 and T-340 were added later to ensure sufficient starting air, however, these tanks are located on the 45’ level.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	073 K4.01	
	Importance Rating	4.0	

Knowledge of PRM system design feature(s) and/or interlocks which provide for the following: Release termination when radiation exceeds setpoint

Proposed Question: Common 24

Which ONE (1) of the following is the effect of a radiation signal that exceeds the setpoint to RE-7865, Plant Vent Stack Wide Range Radiation Monitor, when it is aligned to the Containment Purge Stack?

A radiation signal that exceeds the setpoint to RE-7865 would...

A. secure Continuous Exhaust Fans A-310, A-311 and A-312.

B. initiate a Containment Purge Isolation Signal (CPIS).

C. close the Outside Containment Purge Isolation Valves.

D. close FV-7202, Waste Gas Discharge Flow Control Valve.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because this action is related to the closure of the Waste Gas Discharge valve, however, RE-7865 is aligned to the Primary Vent Stack.

B. Incorrect. Plausible because this rad monitor will isolate the outside Containment Purge Valves, however, it does not input into the CPIS circuitry.

C. Correct. With this rad monitor aligned to the Containment Purge Stack it will isolate these valves on a high radiation signal.

D. Incorrect. Plausible because this action is performed when RE-7865 is aligned to the Primary Vent Stack.

Technical Reference(s) SD-SO23-690, page 9 (Attach if not previously provided)
SO23-8-15, L & S 4.5

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 52747 (As available)

Question Source:	Bank #		
	Modified Bank #	N3139	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	11	

Comments:
What is the effect of a high radiation detected by RE-7865 PVS, WRGM when it's aligned to the Plant Vent Stack?

A. Closes HV-7202, Waste Gas Discharge Isolation Valve
B. Closes only the Outside Containment Purge Isolation valves
C. Initiate a Containment Purge Isolation Signal (CPIS)
D. Turns off fans A310, A311 and A312

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	076 K2.08	
	Importance Rating	3.1	

Knowledge of bus power supplies to the following: ESF-actuated MOVs

Proposed Question: Common 25

Which ONE (1) of the following describes the power supply arrangement for the Salt Water Cooling Valves?

CCW Heat Exchanger SWC Normal Outlet Valves (HV-6495 and HV-6497) are powered from __________ and Saltwater Emergency Discharge Valves (HV-6494 and HV-6496) are powered from __________.

A. 50’ Control Building MCCs BY and BZ; 7’ Turbine Building MCC BK

B. 7’ Turbine Building MCC BK ; 50’ Control Building MCCs BY and BZ

C. 7’ Turbine Building MCC BM; 7’ Turbine Building MCC BK

D. 50’ Control Building MCC BY; 50’ Control Building MCC BZ

Proposed Answer: A

Explanation (Optional):

A. Correct. Despite the “emergency” nomenclature, HV-6494 and HV-6496 are powered from a Non-1E MCC.

B. Incorrect. Plausible given the nomenclature used for the valves, however, the normal valves are 1E powered and the emergency valves are Non-1E powered.

C. Incorrect. Plausible as this partially correct and the mnemonic for MCC BM is “by the mountains” which is the next closest MCC to valves HV-6495 and HV-6497.

D. Incorrect. Plausible because both of these MCCs are 1E powered.

Technical Reference(s) SD-SO23-410, page 18 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 60304 (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	4, 7	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	076 A2.01	
	Importance Rating	3.5	

Ability to (a) predict the impacts of the following malfunctions or operations on the SWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of SWS

Proposed Question: Common 26

Given the following conditions:

- Unit 2 is in MODE 1.
- Train A Component Cooling Water (CCW) Pump P-025 is in service.
- Train A Salt Water Cooling (SWC) Pump P-112 is in service.
- CCW temperatures are rising.

Subsequently the following annunciator alarms are received:

- 64A35 - CCW HX TRAIN A DIFF PRESS HI.
- 64A55 - SWC TRAIN A FLOW TROUBLE.

Which ONE (1) of the following:

a.) Identifies the status of the Salt Water Cooling System?

b.) What action should be taken to mitigate the situation?

A. a.) Marine fouling is occurring.
b.) Start SWC Pump P-307 per SO23-2-8, SWC System Operation.

B. a.) SWC Pump P-112 performance is degrading.
b.) Transfer CCW to Train B per SO23-13-7, Loss of CCW/SWC.

C. a.) Marine fouling is occurring.
b.) Transfer CCW to Train B per SO23-13-7, Loss of CCW/SWC.

D. a.) SWC Pump P-112 performance is degrading.
b.) Start SWC Pump P-307 per SO23-2-8, SWC System Operation.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because marine fouling is occurring, however, starting P-307 is hampered due to Kirk Key interlock and the HX problem is not rectified.
B. Incorrect. Plausible because the flow trouble alarm is in, however, the high HX DP identifies marine fouling.
C. Correct. This is the correct action for the situation.
D. Incorrect. Plausible because the flow trouble alarm is in, however, the high HX DP identifies marine fouling and starting P-307 is hampered due to Kirk Key interlock.

Technical Reference(s) SD-SO23-410, Figure 1 (Attach if not previously provided)
SO23-13-7, Step 14
SO23-15-64A55, page 133
SO23-15-64A35, page 89

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 60306 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 4, 7, 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	078 K3.02	
	Importance Rating	3.4	

Knowledge of the effect that a loss or malfunction of the IAS will have on the following: Systems having pneumatic valves and controls

Proposed Question: Common 27

Given the following conditions:

- A Unit trip has occurred due to a loss of Instrument Air.
- The Atmospheric Dump Valves were placed in AUTO / MODULATE and set at 1000 psia post trip.
- No other operator actions are taken.

Which ONE (1) of the following describes the condition of plant systems following the trip?

A. Letdown flow is isolated.
Main Steam Safety Valves are maintaining SG pressure due to loss of the Atmospheric Dump Valves.

B. Letdown flow is at a minimum.
Atmospheric Dump Valves are controlling SG pressure.

C. Letdown flow is isolated.
Atmospheric Dump Valves are controlling SG pressure.

D. Letdown flow is at a minimum.
Main Steam Safety Valves are maintaining SG pressure due to loss of the Atmospheric Dump Valves.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because Letdown is isolated, however, ADVs are controlling SG pressure using backup nitrogen.
B. Incorrect. Plausible because the ADVs are controlling SG pressure using backup nitrogen, however, Letdown is isolated.
C. Correct. Letdown is isolated; ADVs are controlling SG pressure using backup nitrogen.
D. Incorrect. Plausible because MSSVs would maintain pressure if the ADVs did not function.

Technical Reference(s) SO23-13-5, Attachment 2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 55261 (As available)

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 7, 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	103 A3.01	
	Importance Rating	3.9	

Ability to monitor automatic operation of the containment system, including: Containment isolation

Proposed Question: Common 28

A Containment Isolation Actuation Signal has occurred.

Which ONE (1) of the following sets of valve "CLOSE" indications could be used to make that determination?

A. HV-9920, Containment Normal Cooling System Supply Isolation Valve.
HV-9304, Containment Emergency Sump Outlet Isolation Valve.
HV-8205, Steam Generator E-088 Main Steam Isolation Valve.

B. HV-9823, Containment Mini-Purge Supply Isolation Valve.
HV-5437, Nitrogen to Containment Isolation Valve.
HV-6211, Non-Critical Loop Containment Isolation Valve.

C. HV-9825, Containment Mini-Purge Exhaust Isolation Valve.
HV-4714, Steam Generator E-088 AFW Containment Isolation Valve.
HV-9971, Containment Normal Cooling System Return Isolation Valve.

D. HV-9336, SDC to LPSI Pump Suction Containment Isolation Valve.
HV-9205, Letdown Containment Isolation Valve.
HV-5803, Containment Sump Radwaste Isolation Valve.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible as 2 of 3 indications are correct, however, HV-9304 is normally closed and opens on a RAS signal.
B. Correct.
C. Incorrect. Plausible as 2 of 3 indications are correct, however, HV-4714 closes on a MSIS signal.
D. Incorrect. Plausible as 2 of 3 indications are correct, however, HV-9336 does not receive a signal from any ESFAS related isolation.

Technical Reference(s) SO23-3-2.22, pages 71 and 72 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 81447 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 9

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	001 A1.11	
	Importance Rating	3.7	

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CRDS controls including: Required primary system subcooling during shutdown; location of indication

Proposed Question: Common 29

Given the following conditions:

- Unit 2 is shutting down from 100% power.
- CEAs are being inserted for Axial Shape Index control.
- Annunciator 56B45 - RCS SUBCOOLED MARGIN LO has just alarmed.
- The Subcooled Margin Monitor (SCM) Display on CR-56 is not available.

Which ONE (1) of the following would be used to determine the most conservative subcooled margin?

A. Observe “HEAD” and verify SCM on the DLMS.

B. Observe “CET” and verify SCM on the QSPDS.

C. Observe “HEAD” and verify SCM on the QSPDS.

D. Observe “CET” and verify SCM on the DLMS.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because “HEAD” is a selection for SCM, however, the DLMS cannot be use to determine SCM.

B. Correct. Given the initial at power conditions, the CETs would be used to determine the SCM. Because of the bypass flow from Tcold into the head region, the SCM for HEAD would read an SCM associated with Tave.

C. Incorrect. Plausible because “HEAD” is a selection for SCM, however, using HEAD will not deliver the most limiting SCM.

D. Incorrect. Plausible because “CET” is a selection for SCM, however, the DLMS cannot be use to determine SCM.

Technical Reference(s)	SD-SO23-820, page 84 SO23-15-56.B-45 SO23-3-2.32, Attachment 1 Sat Margin Computer Printout SD-SO23-360, page 45	(Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 54394 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 7, 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	015 K6.04	
	Importance Rating	3.1	

Knowledge of the effect of a loss or malfunction on the following will have on the NIS: Bistables and logic circuits

Proposed Question: Common 30

Which ONE (1) of the following inputs is used to enable the 55% Loss of Load bistable?

A. Linear Power.

B. Hydraulic oil pressure from the Main Turbine HP Stop Valves.

C. Log Channel Power.

D. Valve position indication of the Main Turbine HP Stop Valves.

Proposed Answer: A

Explanation (Optional):

A. Correct. Refer to Figure 1 to see the interface between the NIS and loss of load bistable.
B. Incorrect. Plausible because this signal generates the loss of load trip.
C. Incorrect. Plausible because this signal comes from the safety channel, however, it is used for low power applications.
D. Incorrect. Plausible because this is the valve used to generate the loss of load trip signal.

Technical Reference(s) SD-SO23-470, Figure 1 (Attach if not previously provided)
SD-SO23-470, page 14
SD-SO23-710, page 19
SD-SO23-180, page 68

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 56473 (As available)

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

Question History: Last NRC Exam ____________________

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

10 CFR Part 55 Content: 55.41 6, 7
____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	017 K3.01	
	Importance Rating	3.5	

Knowledge of the effect that a loss or malfunction of the ITM system will have on the following: Natural circulation indications

Proposed Question: Common 31

Given the following conditions:

- Representative Core Exit Thermocouple (REP CET) value is 560ºF.
- A CET that was previously indicating 560ºF has an open circuit.

Which ONE (1) of the following identifies the impact that the CET open circuit failure has on the REP CET value used for verification of Natural Circulation?

The REP CET calculated value…

A. does not change, the failed CET input is NOT used by QSPDS.

B. indicates higher, the failed CET input is NOT used by QSPDS.

C. indicates lower, the failed CET input is flagged but used by QSPDS.

D. does not change, the failed CET input is flagged but used by QSPDS.

Proposed Answer: A

Explanation (Optional):

A. Correct. The input is flagged as invalid and discarded.
B. Incorrect. Plausible because an open could be construed as a maximum voltage.
C. Incorrect. Plausible because an open will create a low CET temperature, however, the input is flagged as invalid and not used by QSPDS.
D. Incorrect. Plausible because an open could be construed as a maximum voltage, however, it is flagged as invalid and not used by QSPDS.

Technical Reference(s) SD-SO23-820, page 87 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 54386 & 54394 (As available)

Question Source: Bank #
Modified Bank # N75652 (Note changes or attach parent)
New

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 7

Comments:
Given the following conditions:
· Unit 2 operating at 100% full power
· One Core Exit Thermocouple (CET) failed to 0ºF output
What is the response of the Qualified Safety Parameter Display System (QSPDS), Representative CET (REPCET) reading to an input failing low?
A. Does not change, input not used in calculation.
B. Indicates lower, input used in calculation.
C. Does not change, input used in calculation.
D. Indicates lower, flagged as invalid

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	041 K3.04	
	Importance Rating	3.5	

Knowledge of the effect that a loss or malfunction of the SDS will have on the following: Reactor power

Proposed Question: Common 32

A loss of Non-1E Instrument Bus Q0612 has occurred on Unit 3 while at full power. The RO is directed to place the Steam Bypass Control System (SBCS) Master Controller in MANUAL per SO23-13-19, Loss of Non-1E Instrument Buses.

Which ONE (1) of the following is the reason for placing the Steam Bypass Control System in MANUAL?

A. Minimize load across 3VS612, Instrument Bus Transfer Switch, when taken to the EMERGENCY position.

B. Prevent the SBCS Valves from inadvertently opening when Q0612 is reenergized.

C. To avoid defeating the single failure design of the SBCS.

D. Prevent a power excursion as the valve permissives are in MANUAL.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because this action is taken to restore power to Q0612 from its emergency source.

B. Correct. The Master Controller will "load" when power is restored and with the controller in AUTO the SBCS Valves will rapidly open with a resulting power excursion.

C. Incorrect. Plausible because this statement is true when the valve permissives are in MANUAL and the Master Controller is in REMOTE, however, this is not the condition in this situation.

D. Incorrect. Plausible because with the valve permissives in MANUAL the valves bypass the "permission" portion of the SBCS "permission before modulation" scheme, however, it is the re-energization of the SBCS that is the concern.

Technical Reference(s) SO23-13-19, Step 1 (Attach if not previously provided)
SO23-3-2.18, Step 14.6
SO23-3-2.18, L & S 3.4

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 56100 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 4, 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	029 A1.02	
	Importance Rating	3.4	

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: Common 33

Given the following conditions:

- A Containment Purge is in progress.
- RE-7865, Plant Vent Stack Radiation Monitor is aligned to the Purge Stack.
- Radiography in the vicinity of RE-7807, Containment Airborne Radiation Monitor has placed it in alarm.

Which ONE (1) of the following identifies the result of RE-7807 going into alarm?

A. Initiates a Containment Purge Isolation Signal.

B. Closes only the outside Containment Purge Isolation Valves.

C. Secures Continuous Exhaust Fans A-310, A-311 and A-312.

D. Closes only the inside Containment Purge Isolation Valves.

Proposed Answer: A

Explanation (Optional):

A. Correct. RE-7807 will initiate a CPIS.

B. Incorrect. Plausible because high radiation on RE-7865 will initiate these closures in its current alignment.

C. Incorrect. Plausible because this action is associated with radioactive release, however, the valve is FV-7202, Waste Gas Release Valve which cannot open if these fans are secured.

D. Incorrect. Plausible because high radiation on RE-7865 will initiate these closures in its current alignment, however, only the outside Containment Valves are affected.

Technical Reference(s) SD-SO23-770, pages 8 & 44 (Attach if not previously provided)
SD-SO23-770, Figure 1

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 81639 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 9

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	035 K5.03	
	Importance Rating	2.8	

Knowledge of operational implications of the following concepts as the apply to the S/GS: Shrink and swell concept

Proposed Question: Common 34

Which ONE (1) of the following responses is expected immediately following the opening of a Steam Bypass Control Valve at 50% power?

Steam Generator Narrow Range level…

A. swells due to an increase in SG downcomer mass.

B. shrinks due to lowering SG pressure.

C. swells due to an increase in density of the SG liquid-vapor mixture.

D. shrinks due to collapse of bubbles in the tube bundle region.

Proposed Answer: A

Explanation (Optional):

A. Correct. SG level swells due to formation of bubbles in the tube bundle area and an increase in SG downcomer mass.
B. Incorrect. Plausible because SG pressure will lower, however, this causes bubbles in the TBA and resultant swell.
C. Incorrect. Plausible because SG level will swell, however, it is caused by a decrease in density of the SG liquid-vapor mixture.
D. Incorrect. Plausible because there is a change in bubble formation in the tube bundle area, however, they increase due to lower SG pressure.

Technical Reference(s) SD-SO23-250, page 86 (Attach if not previously provided)
Steam Tables

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

Learning Objective: 54724 (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	5	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	045 K1.20	
	Importance Rating	3.4	

Knowledge of the physical connections and/or cause-effect relationships between the MT/G system and the following systems: Protection system

Proposed Question: Common 35

Which ONE (1) of the following is the setpoint of the overspeed trip on the Main Turbine?

A. 1890 rpm

B. 1926 rpm

C. 1980 rpm

D. 2034 rpm

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because this represents the preemptive 105% setpoint used in the old Turbine Control System to secure steam to the LP Turbines.

B. Correct. The electronic trip for overspeed is 1926 rpm. This answer is based on the new Digital Control System that was installed and the change to setpoints for the Main Turbine.

C. Incorrect. Plausible because this was the original 110% setpoint before Turbine Control System modifications were incorporated.

D. Incorrect. Plausible because this is the speed at which the Turbine should be manually tripped if the overspeed trip fails to function per SO23-10-4.

Technical Reference(s) SD-SO23-180, page 59 (Attach if not previously provided)
SO23-15-99A, 99A35
SO23-10-4, L & S 1.1

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 83808 (As available)

Question Source:	Bank #	____	
	Modified Bank #	____	(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam	____	
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis	____	
10 CFR Part 55 Content:	55.41	7	
		____	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	068 A2.02	
	Importance Rating	2.7	

Ability to (a) predict the impacts of the following malfunctions or operations on the Liquid Radwaste System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Lack of tank recirculation prior to release

Proposed Question: Common 36

Given the following conditions:

- A liquid release of T-057, Radwaste Secondary Tank has been in progress for five (5) minutes using release valve 2/3HV-7641, Radwaste Discharge.
- The release was automatically terminated due to high radiation as sensed by 2/3 RE-7813, Radwaste Discharge Line Radiation Monitor.
- Review of the Release Permit has identified an insufficient recirculation of T-057, Radwaste Secondary Tank prior to release.

Which ONE (1) of the following actions should be taken to correct the situation?

A. Perform a one (1) volume recirculation of T-057.
Re-sample and finish releasing the tank per the current in use attachment.

B. Perform a four (4) volume recirculation of T-057.
Complete the release attachment, re-sample and initiate a new attachment to finish releasing the tank.

C. Perform a one (1) volume recirculation of T-057.
Complete the release attachment, re-sample and initiate a new attachment to finish releasing the tank.

D. Perform a four (4) volume recirculation of T-057.
Re-sample and finish releasing the tank per the current in use attachment.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because a one volume recirc is required, however, SO23-8-7 directs the use of a new attachment.

B. Incorrect. Plausible because SO23-8-7 directs the use of a new attachment, however, only a one volume recirc is required.

C. Correct. Per SO23-8-7, Attachment 1 and associated L&S, a one volume recirc is required and if a high radiation secures the release then perform actions as noted.

D. Incorrect. Plausible because a recirc of the tank volume is required, however, a one volume recirc is required and SO23-8-7 directs the use of a new attachment.

Technical Reference(s) SO23-8-7, L&S 2.5 (Attach if not previously provided)
SO23-8-7, Attachment 1

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 53393 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 10, 13

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	072 A3.01	
	Importance Rating	2.9	

Ability to monitor automatic operation of the ARM system, including: Changes in ventilation alignment

Proposed Question: Common 37

Which ONE (1) of the following identifies the change in the Control Room ventilation alignment when a high radiation signal is received on 2/3RE-7824G1 & 2/3RE-7825G2, Control Room Intake Air Train A and B Radiation Monitors?

A. 2/3 A-206 and 2/3 A-207, Emergency Ventilation Supply Fans are started.
94% of the Control Room Complex air is recycled.
2/3 E-335 and 2/3 E-336, Emergency Chiller Units are started.

B. 2/3 A-206 and 2/3 A-207, Emergency Ventilation Supply Fans are secured.
100% of the Control Room Complex air is recycled.
2/3 E-335 and 2/3 E-336, Emergency Chiller Units are started

C. 2/3 A-206 and 2/3 A-207, Emergency Ventilation Supply Fans are secured.
94% of the Control Room Complex air is recycled.
2/3 A-201 and 2/3 A-202, Control Room Complex Exhaust Fans are secured

D. 2/3 A-206 and 2/3 A-207, Emergency Ventilation Supply Fans are started.
100% of the Control Room Complex air is recycled.
2/3 A-201 and 2/3 A-202, Control Room Complex Exhaust Fans are secured.

Proposed Answer: A

Explanation (Optional):

A. Correct. The Control Room monitors act as ARMs due to their unique “flying wing” design. This particular design acts like an Area Radiation Monitor vice a Process Radiation Monitor in that it does not use an isokinetic nozzle and draw off a sample to a unique sampling location via filters and pumps.
B. Incorrect. Plausible because these actions are correct for a TGIS, but not a CRIS.
C. Incorrect. Plausible because these actions are partially correct for a TGIS and a CRIS.
D. Incorrect. Plausible because emergency fans are started and CR Complex Exhaust Fans are secured, however, recycled air setpoint is 94%.

Technical Reference(s) SD-SO23-690, page 19 (Attach if not previously provided)
SD-SO23-690, pages 82 & 86

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 81367 & 81366 (As available)

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 7, 8

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	079 A4.01	
	Importance Rating	2.7	

Ability to manually operate and/or monitor in the control room: Cross-tie valves with IAS

Proposed Question: Common 38

Instrument air pressure has been steadily decreasing from 105 psig as read on CR-61 2/3 PI-5344A. Pressure continues to decrease until it stabilizes at 86 psig.

Which ONE (1) of the following actions occurred that caused pressure to stabilize?The LAG 1 Instrument Air Compressor started and is fully loaded.

B. PCV-5354, Respiratory and Service Air cross-connect valve has opened.

C. PCV-5458, Nitrogen Backup to Instrument Air supply valve has opened.

D. The LEAD 1 Instrument Air Compressor started and is fully loaded.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because this compressor is available and will automatically start at 98 psig, however, the setpoint for full loading is ~94 psig.
B. Correct. This valve opens at 88 psig and will attempt to maintain pressure >84 psig.
C. Incorrect. Plausible because this valve is available and will automatically open, however, the setpoint for PCV-5458 is 83 psig.
D. Incorrect. Plausible because this compressor is available and will automatically start at 106 psig, however, the setpoint for full loading is ~102 psig.

Technical Reference(s) SO23-13-5, page 9 (Attach if not previously provided)
SO23-1-1, Attachment 22

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 55261 (As available)

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

Question History: Last NRC Exam ____________

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

10 CFR Part 55 Content: 55.41 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	007 G2.4.50	
	Importance Rating	3.3	

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

Proposed Question: Common 39

Given the following conditions:

- Unit 2 is performing the SO23-12-7, Loss of Forced Circulation / Loss of Offsite Power following a loss of offsite power.
- The following Annunciators alarm in the Control Room:
  - 63A33 - 2D2 125 VDC BUS TROUBLE
  - 63A43 - 2D2 CHARGER TROUBLE
- The PEO has verified that 2D2 Battery Breaker is open.
- DC Bus 2D2 indicates 129 VDC.

Which ONE (1) of the following actions is required?

A. Declare DC Bus 2D2 INOPERABLE due to low battery voltage.

B. Perform one re-close attempt of the battery breaker (breaker remains open), and then leave the Y-002 Inverter in service.

C. Verify proper operation of Y-002 Inverter in its current configuration.

D. Check 2D2 for a battery ground and remove Y-002 Inverter from service if an Inverter DC side ground is identified.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because this is the Tech Spec minimum voltage.

B. Incorrect. Plausible because one re-close attempt is the standard practice at SONGS.

C. Incorrect. Plausible, however, a battery charger and inverter should not be in service with the battery breaker open.

D. Correct. This action is required when a ground condition is confirmed with the ground LED light solidly illuminated.

Technical Reference(s) SO23-6-33, Step 6.4 (Attach if not previously provided)
SO23-6-33, L & S 1.2
SO23-15-63.A43, Step 3.1 Caution
SO23-15-63.A33, Footnote

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 80606 & 80607 (As available)

Question Source: Bank # ______
Modified Bank # N127271 (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 7, 10

Comments:
Given the following conditions:

- Unit 2 is performing the SO23-12-2, Reactor Trip Recovery following a loss of offsite power.
- Annunciator 63A43 - 2D2 BATTERY BKR OPEN alarms in the Control Room.
- The PEO has verified that 2D2 Battery Breaker is open.
- DC Bus 2D2 indicates 129 VDC.

Which ONE (1) of the following remedies is required?

A. Attempt to re-close the battery breaker or remove the Y-002 Inverter from service.

B. Perform one re-close attempt of the battery breaker (breaker remains open), and then leave the Y-002 Inverter in service.

C. Verify proper operation of Y-002 Inverter in its current configuration.

D. Check 2D2 for a battery ground and remove Y-002 Inverter from service if the ground LED light is extinguished.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	008 AA2.26	
	Importance Rating	3.1	

Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: Probable PZR steam space leakage paths other than PORV or code safety

Proposed Question: Common 40

Given the following conditions:

- Pressurizer pressure is slowly lowering.
- Pressurizer level is slowly rising.
- Containment humidity is slowly rising.
- Pressurizer vapor space temperature is slowly lowering.
- Subcooled Margin is slowly lowering.
- No automatic actions have occurred.

Which ONE (1) of the following is the reason for the conditions listed?

A Pressurizer…

A. safety valve has minor seat leakage.

B. Spray valve is open.

C. level condensing pot is leaking.

D. heater weld is leaking.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because all the conditions lead to this problem except Containment humidity is rising, this would not be a symptom.

B. Incorrect. Plausible because all the conditions lead to this problem except Containment humidity is rising, this would not be a symptom.

C. Correct. A leak from the condensing pot would cause level to rise due to reference leg flashing.

D. Incorrect. Plausible because Containment humidity is rising, however, PZR level would be lowering.

Technical Reference(s) SD-SO23-360, Figure III-1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 94467 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 3, 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	009 EK3.07	
	Importance Rating	3.3	

Knowledge of the reasons for the following responses as the apply to the small break LOCA: Increasing indication on CCWS process monitor: indicates in-leakage of radioactive liquids

Proposed Question: Common 41

Given the following conditions:

- A small break LOCA is in progress on Unit 3 in MODE 1.
- Both Trains of CCW are in service.

Due to equipment malfunctions the following alignment occurs:

- The CCW Letdown Heat Exchanger is aligned to Train A.
- The CCW Non-Critical Loop is aligned to Train B.
- RE-7819, CCW Process Radiation Monitor goes into alarm.

Which ONE (1) of the following is the location of the leak?

A. Letdown Heat Exchanger.

B. RCS Hot Leg Sample Cooler.

C. Shutdown Cooling Heat Exchanger.

D. RCP Seal Cooler.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because the LD HX is cooled by CCW, however, in the current alignment a LD to CCW leak would not be detected because RE-7819 is supplied from a flow venturi on the NCL.

B. Incorrect. Plausible because this could be the source of the leak, however, this component is cooled by chill water.

C. Incorrect. Plausible because the SDC HX is cooled by CCW, however, in the current alignment CCW pressure is higher than SDC pressure.

D. Correct. Given the current CCW Train alignment, this is the source.

Technical Reference(s) SD-SO23-400, Figures 2A & 4 (Attach if not previously provided)
SD-SO23-400, page 25
SD-SO23-420, Figure 1

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 54381 (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 4, 7, 11

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	011 EK2.02	
	Importance Rating	2.6	

Knowledge of the interrelations between the and the following Large Break LOCA: Pumps

Proposed Question: Common 42

Given the following Unit 2 conditions:

- SIAS actuated due to a LOCA.
- Both HPSI Pumps are tripped.
- RCS pressure is 450 psia.
- RCS temperature is 430°F.
- Containment pressure is 2.5 psig.
- All other equipment is running per design.
- The crew is performing actions of SO23-12-1, Standard Post Trip Actions.

Which ONE (1) of the following describes the required action and associated reason regarding operation of the RCPs?

A. Stop two RCPs and leave two RCPs running to provide forced cooling flow of the RCS.

B. Stop two RCPs and leave two RCPs running to minimize fluid mass loss out of the break.

C. Stop all RCPs to prevent phase separation of RCS liquid.

D. Stop all RCPs due to loss of Net Positive Suction Head.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because HPSI Pumps are secured, however, RCP NPSH is lost.

B. Incorrect. Plausible because the Trip 2/Leave 2 criteria when <1430 psia is based on minimizing inventory loss, however, this condition applies to SBLOCAs not a LBLOCA.

C. Incorrect. Plausible because phase separation may occur in SBLOCAs after RCPs are tripped, leading to core uncovery.

D. Correct. RCPs are tripped due to a loss of NPSH.

Technical Reference(s): SO23-12-1, Step 6 RNO (Attach if not previously provided)
SO23-12-1, Attachment 3

Proposed references to be provided to applicants during exam: SO23-12-1, Attachment 3

Learning Objective: 56252 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	015 AK3.03	
	Importance Rating	3.7	

Knowledge of the reasons for the following responses as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow) : Sequence of events for manually tripping reactor and RCP as a result of an RCP malfunction

Proposed Question: Common 43

Given the following conditions for Reactor Coolant Pump (RCP) P-004:

- Lower Seal pressure is 2235 psia.
- Middle and Upper Seal pressures are oscillating between 75 and 2235 psia.
- Vapor Seal Cavity pressure is 75 psia.
- Controlled Bleed-Off flow is lost.

Which ONE (1) of the following is the required action and reason for the given set of conditions?

A. Immediately trip the Reactor.
When CEAs have been inserted for 5 seconds, trip RCP P-004 to prevent exceeding DNBR.

B. Initiate a plant shutdown.
Trip the Reactor and trip RCP P-004 to preserve the Vapor Seal and prevent a small break LOCA.

C. Immediately trip the Reactor.
After 5 seconds trip RCP P-004 to prevent exceeding local power density (LPD) setpoints.

D. Initiate a plant shutdown.
When the Reactor is tripped and CEAs have been inserted for 5 seconds, trip RCP P-004 to prevent exceeding local power density (LPD) setpoints.

Proposed Answer: A

Explanation (Optional):

A. Correct. This is the required action and reason for the given conditions.

B. Incorrect. Plausible because a trip is required, however, procedure requires waiting 5 seconds after the CEAs are inserted and trip should occur immediately.

C. Incorrect. Plausible because if not for the reason, this action would be correct.

D. Incorrect. Plausible because if not for the CBO flow problem and reason, this action would be correct.

Technical Reference(s) SO23-13-6, Step 2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 55452 (As available)

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 3, 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	022 AK3.04	
	Importance Rating	3.2	

Knowledge of the reasons for the following responses as they apply to Loss of Reactor Coolant Makeup: Isolating letdown

Proposed Question: Common 44

With the Unit at 100% power, which ONE (1) of the following would cause a loss of Letdown flow?

A. LV-0227A, Volume Control Tank Inlet Valve fails to RADWASTE position.

B. E-063, Regen Heat Exchanger outlet temperature of 450ºF

C. E-062, Letdown Heat Exchanger outlet temperature of 150ºF.

D. 80 GPM Letdown flow with only one Charging Pump running.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because when LV-0227A aligns to Radwaste the VCT level will drop, however, when level reaches 6% in the VCT the RWST to Charging Valve will open and restore Charging flow.

B. Correct. TV-0221 and TV-9267 both close when the Regen HX outlet temp reaches 428ºF.

C. Incorrect. Plausible because this valve closes when the Letdown Regenerative Heat Exchanger when temperature exceeds 140ºF Letdown flow is diverted from the Letdown Ion Exchangers, however, Letdown flow is not disturbed.

D. Incorrect. Plausible because the difference between Charging and Letdown could initiate a Regen HX outlet high temp, however, plant design dictates that max Letdown and min Charging will not cause Letdown to isolate.

Technical Reference(s) SD-SO23-390, page 11 (Attach if not previously provided)
SD-SO23-390, pages 53 & 173
SD-SO23-390, Figure 1

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 53388 (As available)

Question Source: Bank #
Modified Bank # N56704 (Note changes or attach parent)
New

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 7

Comments:
Unit 3 is operating in a normal full power alignment.
What could result in a loss of Letdown flow?
A. Volume Control Tank Outlet Valve (LV -0227B) fails closed.
B. Temperature of 150°F leaving the Letdown Heat Exchanger.
C. Loss of instrument air to the Boronometer Flow Control Valve (FV-0203).
D. 80 gpm Letdown flow with only one Charging Pump running.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	025 AA1.01	
	Importance Rating	3.6	

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

Proposed Question: Common 45

Given the following conditions:

- Unit 2 was cooling down to MODE 5 with Train A Shutdown Cooling (SDC) in service.
- RCS cooldown rate was 20ºF per hour when a loss of Train A SDC Pump occurred.
- RCS temperature is 300ºF.
- When the Train B SDC Pump was started the cooldown rate rose to 50ºF per hour.

Which ONE (1) of the following describes the actions necessary to continue with the RCS cooldown per SO23-13-15, Loss of Shutdown Cooling?

A. Throttle closed the SDC HX Bypass Flow Control Valve to establish a minimum flowrate of 2750 gpm.
Throttle closed the SDC Heat Exchanger Outlet Valve to establish the required cooldown rate.

B. Throttle open the SDC HX Bypass Flow Control Valve to establish a minimum flowrate of 2500 gpm.
Throttle closed the SDC Heat Exchanger Outlet Valve to establish the required cooldown rate.

C. Throttle closed the SDC HX Bypass Flow Control Valve to establish a minimum flowrate of 2750 gpm.
Throttle open the SDC Heat Exchanger Outlet Valve to establish the required cooldown rate.

D. Throttle open the SDC HX Bypass Flow Control Valve to establish a minimum flowrate of 2500 gpm.
Throttle open the SDC Heat Exchanger Outlet Valve to establish the required cooldown rate.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Partially correct as throttling closed the HX outlet valve will control the cooldown rate; however, the bypass valve would be opened.

B. Correct. The bypass valve must be throttled open to establish minimum flow and the HX closed to control the cooldown rate.

C. Incorrect. Partially correct for minimum flowrate, however, the bypass valve would be opened and the HX outlet valve closed.

D. Incorrect. Partially correct as throttling open the bypass valve will control the cooldown rate, however, the bypass valve would be opened.

Technical Reference(s) SO23-13-15, Steps 5 & 6 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 55323 (As available)

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 5, 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	026 G2.1.2	
	Importance Rating	3.0	

Conduct of Operations: Knowledge of operator responsibilities during all modes of plant operation for loss of CCW.

Proposed Question: Common 46

Given the following plant conditions:

- Unit 2 is in MODE 5 on Shutdown Cooling (SDC).
- A total loss of Component Cooling Water (CCW) occurs on Unit 2.

Which ONE (1) of the following actions is required to restore cooling to Unit 2 Reactor Coolant System?

A. Allow the RCS to heatup and use the Steam Generators for heat removal.

B. Cross connect CCW through the Instrument Air System.

C. Cross connect CCW through the Radwaste System Supply and Return Headers.

D. Align the fire main to the SDC Heat Exchangers.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because this action is an alternative form of cooling for a loss of SDC but not a loss of CCW.
B. Incorrect. Plausible because there is a cross connect, however, the system is TPCW not CCW.
C. Correct. This is the action outlined in SO23-13-7 when the Unit is in MODE 5 or 6.
D. Incorrect. Plausible because fire main can be used throughout the plant, however, there are no provisions for the SDC HX.

Technical Reference(s) SO23-13-7, Attachment 4 (Attach if not previously provided)
SD-SO23-400, Figure 1

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 55542 (As available)

Question Source:	Bank #	______	
	Modified Bank #	______	(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam	______	
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis	______	
10 CFR Part 55 Content:	55.41	10	
		______	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	027 AA1.04	
	Importance Rating	3.9	

Ability to operate and / or monitor the following as they apply to the Pressurizer Pressure Control Malfunctions: Pressure recovery, using emergency-only heaters

Proposed Question: Common 47

Given the following conditions:

- Pressurizer Pressure Channel Y is in service.
- Pressurizer Pressure indication PT-0100Y has failed high.
- HS-0100A, Pressurizer Pressure Channel Selector is in the Channel Y position.
- An inadvertent SIAS has occurred.
- Pressurizer Pressure is 2150 psia and slowly lowering.

Which ONE (1) of the following is required to restore operation of the 1E Pressurizer heaters?

A. Transfer PZR Pressure Control to Channel X.
Restore 1E PZR Heaters by going to OVERRIDE, then OFF, then AUTO.

B. Reset the SIAS signal.
Restore 1E PZR Heaters by going to OFF, then ON, then AUTO.

C. Transfer PZR Pressure Control to Channel X.
Restore 1E PZR Heaters by going to OFF, then ON, then AUTO.

D. Reset the SIAS signal.
Restore 1E PZR Heaters by going to OVERRIDE, then OFF, then AUTO.

Proposed Answer: A

Explanation (Optional):

A. Correct. This is the desired action to restore the 1E PZR heaters in their current configuration.

B. Incorrect. Plausible because resetting the SIAS would remove the SIAS trip signal, however, leaving Channel Y in service continues to send a trip signal to the heaters.

C. Incorrect. Plausible because this is the desired action to restore the Non-1E PZR heaters, however, with SIAS present heaters must be overridden.

D. Incorrect. Plausible because had the Non-1E heaters not tripped this would place them in operation. Plausible because resetting the SIAS would remove the SIAS trip signal, however, leaving Channel Y in service continues to send a trip signal to the heaters.

Technical Reference(s) SD-SO23-360, page 98 (Attach if not previously provided)
SO23-13-27, Attachment 1

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 55220 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	038 G2.4.6	
	Importance Rating	3.1	

Emergency Procedures / Plan Knowledge symptom based EOP mitigation strategies.

Proposed Question: Common 48

Which ONE (1) of the following describes the conditions necessary to perform an asymmetric Steam Generator natural circulation cooldown following a Steam Generator Tube Rupture?

A. Limit the cooldown rate of the steaming SG.
Maintain a stable non-divergent ΔT between RCS loops.

B. Maximize the cooldown rate of the steaming SG.
Maintain a stable divergent ΔT between RCS loops.

C. Limit the cooldown rate of the steaming SG.
Maintain a stable divergent ΔT between RCS loops.

D. Maximize the cooldown rate of the steaming SG.
Maintain a stable non-divergent ΔT between RCS loops.

Proposed Answer: A

Explanation (Optional):

A. Correct.

B. Incorrect. Plausible because you are maximizing the cooldown rate of the steaming SG, however, a divergent ΔT will cause natural circulation to cease in the non-steaming loop.

C. Incorrect. Plausible because cooldown conditions for the steaming SG is correct, however, a divergent ΔT will cause natural circulation to cease in the non-steaming loop.

D. Incorrect. Plausible because you are maximizing the cooldown rate from the steaming SG, however, this would not act to maintain a non-divergent ΔT.

Technical Reference(s) SO23-14-11, pages 80 & 81 (Attach if not previously provided)
SO23-12-11, page 92

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 55339 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	055 G2.1.32	
	Importance Rating	3.4	

Conduct of Operations: Ability to explain and apply all system limits and precautions.

Proposed Question: Common 49

Given the following conditions:

- A Station Blackout is in progress on both Units.
- SO23-12-11, EOI Supporting Attachments, Attachment 9, Control Building Ventilation Emergency Actions must be performed.

Which ONE (1) of the following is the time associated with performing Attachment 9 and the reason for this action?

Open the Control Room cabinet doors within…

A. 15 minutes of loss of Control Room Ventilation to prevent spurious actuation of relays.

B. 30 minutes of loss of Control Room Ventilation to prevent damage to equipment due to overheating.

C. 45 minutes of loss of Control Room Ventilation to prevent spurious actuation of relays.

D. 60 minutes of loss of Control Room Ventilation to prevent damage to equipment due to overheating.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because this time is associated with establishing natural circulation following trip of RCPs during a Station Blackout.
B. Correct. This is the guidance outlined in the Caution of Attachment 9.
C. Incorrect. Plausible because this is the time associated with reducing Battery D5 loads.
D. Incorrect. Plausible because this is the time associated with restoring emergency chillers and emergency HVAC.

Technical Reference(s) SO23-12-11, Attachment 9 (Attach if not previously provided)
SO23-12-8, page 24

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 55268 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	056 AK3.02	
	Importance Rating	4.4	

Knowledge of the reasons for the following responses as they apply to the Loss of Offsite Power: Actions contained in EOP for loss of offsite power

Proposed Question: Common 50

Given the following conditions:

- Unit 2 is at 100% when a Loss of Offsite Power occurs.
- 4160 Volt Buses 2A03 and 2A07 become de-energized.
- The Main Steam Isolation Valves and Steam Generator Blowdown Valves must be closed.
- EFAS-1 and EFAS-2 have actuated.

Which ONE (1) of the following states the reason for closing these valves manually vice initiating a Main Steam Isolation Signal?

A. Auxiliary Feedwater flow would be interrupted when SG levels went below 21%.

B. Steam Generator Blowdown Valves could not be re-opened.

C. Main Feedwater flow would be interrupted until it is overridden.

D. Atmospheric Dump Valves would be isolated until they are overridden.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because AFW flow would be interrupted if level was above 26%.
B. Incorrect. Plausible because the valves are closed by an MSIS, however, they can be overridden.
C. Incorrect. Plausible because this condition is identified in the bases, however, given the status of 2A03 and 2A07 Main Feedwater would not be in service.
D. Correct. It is more desirable to isolate manually then to have all valves go closed because the MSIS will eventually be reset later in the procedure.

Technical Reference(s) SO23-12-7, Step 4 Note (Attach if not previously provided)
SO23-14-7, Step 4 Note

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 53005 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 7, 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	057 AA2.03	
	Importance Rating	3.7	

Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: RPS panel alarm annunciators and trip indicators

Proposed Question: Common 51

Given the following conditions:

- Unit 3 is at 100% power
- The following Annunciators are in alarm:
  - 56A02 - LOG POWER LEVEL HI CHANNEL TRIP
  - 56A03 - LOCAL POWER DENSITY HI CHANNEL TRIP
  - 56A04 - DNBR LO CHANNEL TRIP
- Pressurizer pressure is aligned to Channel Y and is 2235 psia and steady.
- Pressurizer level is aligned to Channel Y and is 53% and steady.
- Reactor Trip Path 3 and 4 lights are **extinguished**.
- Plant Protection System bistables on Channels A and C ROMs are **not tripped**.

Which ONE (1) of the following Vital AC Buses has been lost?

A. Vital Bus 1.

B. Vital Bus 2.

C. Vital Bus 3.

D. Vital Bus 4.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because Reactor Trip Paths 1 & 2 are actuated, however, in this condition the lights are extinguished, not illuminated. Also, neither channel of PZR level and pressure were affected.

B. Incorrect. Plausible because the Channel A & C ROMs are not tripped, however, a loss of VB #2 removes all trip path indication from the Control Room. Also, neither channel of PZR level and pressure were affected.

C. Incorrect. Plausible because PZR level and pressure were not affected and trip paths 1 & 2 are lit, however, in this condition, PPS B & D ROMs are not tripped.

D. Correct. Given the trip path and PPS ROM indications, VB #4 was lost.

Technical Reference(s) SO23-13-18, Attachment 4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 55180 (As available)

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	058 G2.4.6	
	Importance Rating	3.1	

Emergency Procedures / Plan Knowledge symptom based EOP mitigation strategies.

Proposed Question: Common 52

Given the following conditions:

- Unit 2 is in MODE 3 and the following alarms have been received:
  - 63A32 - 2D1 125 VDC BUS TROUBLE
  - 63A52 - 2D1 CHARGER TROUBLE
- It was determined that the 2D1 Battery Charger has malfunctioned.

Which ONE (1) of the following actions is required to maintain power to the DC bus?

A. Open the 2D1 Battery Breaker and cross-tie DC Bus 2 with DC Bus 1.

B. Minimize DC loads and place the Spare Battery in service.

C. Minimize DC loads and place the Spare Battery Charger in service.

D. Open the 2D1 Battery Breaker and cross-tie DC Bus 3 with DC Bus 1.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible as this action would be allowed during a Station Blackout or in MODES 5 or 6, however, DC Bus 2 must be cross connected with DC Bus 4.

B. Incorrect. Plausible as this action is possible, however, this does not resolve the issue with charger trouble alarms.

C. Correct. This action is performed per SO23-6-15.

D. Incorrect. Plausible as this action would be allowed in MODES 5 or 6, however, not in MODES 1-4.

Technical Reference(s)	SO23-15-63.A, 63A52	(Attach if not previously provided)
	SD-SO23-140, page 21	
	SO23-6-15, Step 6.1	
	SO23-6-15, Attachment 6	

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 80606 (As available)

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 8, 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	062 AK3.03	
	Importance Rating	4.0	

Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water Guidance actions contained in EOP for Loss of nuclear service water

Proposed Question: Common 53

SO23-13-7, Loss of CCW/SWC contains guidance to start the affected train CCW Pump prior to the SWC Pump start while recovering from the loss of a SWC Pump.

Which ONE (1) of the following is the reason for this action?

A. Ensures the CCW radiation monitor is in service in the event of CCW leakage into SWC.

B. Prevent thermal shock of the CCW Heat Exchanger.

C. CCW System provides cooling for the CCW Pump seal.

D. Prevent saltwater from entering the CCW System in the event of a tube leak.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible when referring to the Letdown HX and the Non-Critical Loop.
B. Incorrect. Plausible because thermal shock is a concern.
C. Incorrect. Plausible since cooling/sealing water is provided to the CCW Pump from CCW.
D. Correct. This is the guidance contained in SO23-13-7.

Technical Reference(s) SO23-13-7, Attachment 12 (Attach if not previously provided)
SO23-2-8, L & S 1.4

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 55542 (As available)

Question Source: Bank # ______
Modified Bank # N56529 (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 10

Comments:
Operating Instruction S023-2-8, Saltwater Cooling System Operation, cautions against running a Saltwater Cooling (SWC) Pump without its associated Component Cooling Water loop pressurized.
What is the reason for this caution?

A. Prevent saltwater from entering the CCW system in the event of a tube leak.

B. Prevent thermal shock of the CCW heat exchanger.

C. CCW system provides cooling for the SWC pump motor.

D. Ensures the CCW radiation monitor is in service in the event of CCW leakage into the SWC system.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	029 EK1.03	
	Importance Rating	3.6	

Knowledge of the operational implications of the following concepts as they apply to the ATWS: Boron effects on reactivity

Proposed Question: Common 54

Given the following conditions:

- Power level is 25%.
- Pressurizer pressure is rising.
- An ATWS is in progress.
- All attempts to trip the Reactor have failed.
- Emergency Boration is being initiated.
- The BAMU Pumps will NOT start.

Which ONE (1) of the following sources of boron will have the greatest impact on core reactivity and why?

A. BAMU Tanks because of the higher flowrate.

B. BAMU Tanks because of the higher boron concentration.

C. RWST because of the higher head of the RWST Tank.

D. RWST because of the closer connection to the Charging Pump suction.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because BAMU Tanks are closer to the Charging Pumps, however, flow is limited by the positive displacement Charging Pumps.

B. Correct.

C. Incorrect. Plausible because the top of the tank is at a higher level than the BAMU Tanks, however, flow is limited by the positive displacement Charging Pumps.

D. Incorrect. Plausible because RWST connection is at the Charging Pump suction but the BAMU is also (physically different connection).

Technical Reference(s) SD-SO23-390, page 100 (Attach if not previously provided)
LCS Figure 3.1.104-1

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 52658 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	E05 EK2.2	
	Importance Rating	3.7	

Knowledge of the interrelations between the (Excess Steam Demand) 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: Common 55

Given the following conditions:

- Steam Generator (SG) E-089 Excess Steam Demand Event (ESDE) has occurred inside Containment.
- SG E-089 wide range level is 0% (dryout).
- SG E-088 narrow range level is 34%.
- EFAS-2 was actuated post-trip while EFAS-1 was not.
- The crew is preparing to reset ESFAS Functions per SO23-12-5, ESDE.

Based on the Caution for resetting a Main Steam Isolation Signal, what actions should be taken and why?

A. MSIS can be reset. Limit the flowrate to SG E-088 to avoid collapsing the feed ring.

B. MSIS should **not** be reset. AFW flow may initiate to SG E-089 and result in a Steam Generator tube rupture.

C. MSIS can be reset. Limit the flowrate to SG E-088 to avoid Pressurized Thermal Shock concerns.

D. MSIS should **not** be reset. AFW flow may initiate to SG E-089 and collapse the feed ring.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because MSIS could be reset if both EFAS 1 & 2 had actuated, however, both SGs are not >21%. The reason is correct because SG level is <40%, the point at which feed ring collapse is a concern.

B. Correct. Because EFAS-1 was not actuated AFW flow could initiate and result in a SGTR.

C. Incorrect. Plausible because PTS is a concern, however, it should have been addressed by this point in the procedure.

D. Incorrect. Plausible because the MSIS status is correct, however, the reason is not.

Technical Reference(s): SO23-14-5, Step 18d (Attach if not previously provided)
SO23-12-5, Step 18d

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 54790 (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	E06 EA1.1	
	Importance Rating	4.0	

Ability to operate and / or monitor the following as they apply to the (Loss of Feedwater) Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Proposed Question: Common 56

Given the following conditions:

- Unit 2 is recovering from a loss of feedwater.
- Plant temperature is stable at 545ºF.
- Main Feedwater will be used to feed the Steam Generators.
- There is no Main Feedwater Pump running at this time.

Which ONE (1) of the following identifies the actions required to start a Main Feedwater Pump?

A. Ensure the Motor Speed Controller (MSC) and Electric Automatic Positioner (EAP) are at the low speed stop.
Reset the MFWPT and verify the MFWPT HP and LP Stop Valves are open.

B. Ensure the Motor Speed Controller (MSC) and Electric Automatic Positioner (EAP) are at the high speed stop.
Reset the MFWPT and verify the MFWPT HP and LP Stop Valves are closed.

C. Ensure the Motor Speed Controller (MSC) and Electric Automatic Positioner (EAP) are at the high speed stop.
Reset the MFWPT and verify the MFWPT HP and LP Stop Valves are open.

D. Ensure the Motor Speed Controller (MSC) and Electric Automatic Positioner (EAP) are at the low speed stop.
Reset the MFWPT and verify the MFWPT HP and LP Stop Valves are closed.

Proposed Answer: A

Explanation (Optional):

A. Correct. These are the required actions per SO23-12-6.

B. Incorrect. Plausible because these actions must be performed, however, the EAP/MSC must be at the low speed stop and the HP/LP Stop Valves will be open, the HP/LP Governor Valves will be closed.

C. Incorrect. Plausible because the HP/LP Stop Valves will be open, however, the EAP/MSC must be at the low speed stop.

D. Incorrect. Plausible because the EAP/MSC must be at the low speed stop, however, the HP/LP Stop Valves will be open; the HP/LP Governor Valves will be closed.

Technical Reference(s) SO23-12-6, Step 9h (Attach if not previously provided)
SO23-2-1, Attachment 1

Proposed references to be provided to applicants during examination: ____________

Learning Objective: 53911 (As available)

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

Question History: Last NRC Exam ____________

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

10 CFR Part 55 Content: 55.41 4, 7, 10

Comments:

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	001 AA1.05	
Importance Rating	4.3	

Ability to operate and/or monitor the following as applied to a Continuous Rod Withdrawal: Reactor trip switches

Proposed Question: Common 57

Given the following conditions:

- Unit 2 is at 65% power.
- A power ascension is in progress.
- The RO is withdrawing Group 6 CEAs when a CEDMCS malfunction causes continuous withdrawal of Group 6 CEAs.

Assuming no other operator actions are taken (no manipulations of any controls / systems), which ONE (1) of the following actions is required?

A. Place the CEDMCS Selector Switch in OFF.
If CEA withdrawal continues, reposition the Bank Selector Switch to any other CEA Group than Group 6.

B. Depress Reactor Trip Switches 2HS-9132-1 through 2HS-9132-4 to open all Reactor Trip Breakers.
If Reactor fails to trip, open MG Set Output Breakers by deenergizing Load Centers 2B16 and 2B17.

C. Place the CEDMCS Selector Switch in OFF.
If CEA withdrawal continues, depress Reactor Trip Switches 2HS-9132-1 through 2HS-9132-4 to open all Reactor Trip Breakers.

D. Depress Reactor Trip Switches 2HS-9132-1 through 2HS-9132-4 to open all Reactor Trip Breakers.
If Reactor fails to trip, open MG Set Output Breakers by deenergizing Load Centers 2B14 and 2B15.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because this action could stop the rod withdrawal, however, there is no confirming indication that the withdrawal has stopped.

B. Incorrect. Plausible because the Reactor trip breakers will open, however, if Reactor failed to trip only one MG Set Breaker would open.

C. Correct. This is the correct action based on the annunciator and the EOI.

D. Incorrect. Plausible because the Reactor will trip, however, if Reactor failed to trip only one MG Set Breaker would open.

Technical Reference(s) SO23-12-1, Step 2 (Attach if not previously provided)
SO23-3-2.19, Section 6.11
SD-SO23-510, Figure 13

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 56252 & 81786 (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 6, 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	024 AK2.03	
	Importance Rating	2.6	

Knowledge of the interrelations between the Emergency Boration and the following: Controllers and positioners

Proposed Question: Common 58

Given the following conditions:

- An ATWS has occurred at 90% power.
- The feeder breaker to Bus B04 has opened.
- An emergency boration is started in accordance with SO23-13-11, Emergency Boration of the RCS/Inadvertent Dilution or Boration.
- SIAS has NOT actuated.

Which ONE (1) of the following is the reason for the listed valve position during an emergency boration?

A. LV-0227B, VCT Outlet Valve is in MANUAL and CLOSED to prevent VCT pressure from stopping gravity feed flow.

B. LV-0227B, VCT Outlet Valve is in MANUAL and CLOSED to allow the BAMU Pump head to reach the Charging Pump suction.

C. FV-9253, Blended Makeup to VCT Isolation Valve is in MANUAL and CLOSED to prevent bypass flow from the blend tee to the VCT.

D. FV-9253, Blended Makeup to VCT Isolation Valve is in MANUAL and CLOSED to allow borated water to flow from the RWST when BAMU tanks empty.

Proposed Answer: A

Explanation (Optional):

A. Correct. With B04 unavailable, the flowpath is via gravity feed.

B. Incorrect. Plausible as this is the correct valve position, however, with Bus 2B04 (via MCC BY) power lost the BAMU Pumps are not available.

C. Incorrect. Plausible as this is the correct valve position, however, once the gravity feed path is chosen this valve position is inconsequential.

D. Incorrect. Plausible as this is the correct valve position and it is true that when the BAMU Tanks empty the RWST will be aligned, however, once the gravity feed path is chosen this valve position is inconsequential.

Technical Reference(s) SO23-13-11, Step 2d (Attach if not previously provided)
SD-SO23-390, page 124

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 55510 (As available)

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

Question History: Last NRC Exam _____

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

10 CFR Part 55 Content: 55.41 6, 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	028 AK1.01	
	Importance Rating	2.8	

Knowledge of the operational implications of the following concepts as they apply to the pressurizer level control system malfunctions : Pressurizer reference leg abnormalities

Proposed Question: Common 59

Given the following conditions:

- Pressurizer Level Channel X is in service.
- A leak develops on the Pressurizer Channel X reference leg.

Which ONE (1) of the following identifies the difference between indicated and actual Pressurizer level?

A. Indicated level rises; Actual level lowers.

B. Indicated level rises; Actual level rises.

C. Indicated level lowers; Actual level lowers.

D. Indicated level lowers; Actual level rises.

Proposed Answer: A

Explanation (Optional):

A. Correct. As the reference leg empties the indicated Pressurizer level will rise, the Level Control System will respond to this malfunction by opening the Letdown Control Valves.

B. Incorrect. Plausible because indicated level will rise, however, the Letdown Control Valves open rather than close.

C. Incorrect. Plausible because of misconception of effects of reference leg level to indicated level.

D. Incorrect. Plausible because of misconception of effects of reference leg level to indicated level.

Technical Reference(s) OFD122, Chapter 2, page 18 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 55219 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 5, 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	036 AA2.01	
	Importance Rating	3.2	

Ability to determine and interpret the following as they apply to the Fuel Handling Incidents: ARM system indications

Proposed Question: Common 60

Given the following conditions:

- The Unit is in MODE 6 with refueling in progress.
- The Fuel Transfer Tube is open.
- Both Spent Fuel Pool Gates are open.

Which ONE (1) of the following identifies the Radiation Monitors which would be the first to indicate a Fuel Handling Accident has occurred inside Containment?

A. RE-7845, Containment Personnel Lock Area Radiation Monitor
RE-7822, Fuel Handling Building Airborne Radiation Monitor

B. RE-7845, Containment Personnel Lock Area Radiation Monitor
RE-7848, Containment Building 30’ Area Radiation Monitor

C. RE-7848, Containment Building 30’ Area Radiation Monitor
RE-7823, Fuel Handling Building Airborne Radiation Monitor

D. RE-7822, Fuel Handling Building Airborne Radiation Monitor
RE-7823, Fuel Handling Building Airborne Radiation Monitor

Proposed Answer: B

Explanation (Optional):

A. Incorrect Plausible because the Containment and Fuel Handling Building are connected.
B. Correct.
C. Incorrect. Plausible because the Containment and Fuel Handling Building are connected.
D. Incorrect. Plausible because the Containment and Fuel Handling Building are connected.

Technical Reference(s) SO23-13-20, Step 2 (Attach if not previously provided)
SD-SO23-690, pages 29 & 43

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 52821 (As available)

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 11

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	037 AK3.05	
	Importance Rating	3.7	

Knowledge of the reasons for the following responses as they apply to the Steam Generator Tube Leak: Actions contained in procedures for radiation monitoring, RCS water inventory balance, S/G tube failure, and plant shutdown

Proposed Question: Common 61

Given the following conditions:

- SO23-13-14, Reactor Coolant Leak was entered for a Steam Generator tube leak on SG E-089.
- While performing steps in SO23-13-14, several tubes failed and the crew transitioned to SO23-12-4, Steam Generator Tube Rupture.
- Actions of SO23-12-4, Steam Generator Tube Rupture are in effect with the crew preparing to place the Shutdown Cooling System in service.
- RCS pressure is 330 psia and slowly lowering.
- RCS Thot is 360ºF and slowly rising.
- Reactor Vessel Plenum level is 80%.

Which ONE (1) of the following actions is required and what is the reason for that action?

A. Continue to bleed steam from Steam Generator E-089 as SDC entry conditions are not met.

B. Restore Reactor Vessel Plenum level to greater than or equal to 100% to ensure the hot legs are full and SDC will operate unimpeded.

C. Lower Core Exit Saturation Margin and maintain as close to 20ºF as possible to minimize primary to secondary leakage.

D. Restore Reactor Vessel Head level to 100% to ensure adequate reserve volume in the RCS during the cooldown.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because RCS Thot is slowly rising, however, nominal SDC entry conditions are met even with Thot as high as 386ºF.
B. Correct. This is required otherwise SDC or natural circulation would be impeded.
C. Incorrect. Plausible because the saturation margin could be lowered as it is currently about 64ºF and the desired SCM is ≥ 20ºF, however, at this stage of the cooldown the level in the SG should be below the tube bundle and SCM is less of a concern.
D. Incorrect. Plausible because the RCS will shrink when the cooldown is initiated, however, maintaining the hot legs full is the priority.

Technical Reference(s) SO23-14-4, Step 25 (Attach if not previously provided)
SO23-14-4, Step 23

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 53000 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	059 G2.1.2	
	Importance Rating	3.0	

Conduct of Operations: Knowledge of operator responsibilities during all modes of plant operation.

Proposed Question: Common 62

Given the following:

- The plant is at 100% power.
- NO radioactive releases are in progress.
- P-168, Radwaste Primary Pump has developed a leak while pumping 2/3 T-066, Radwaste Primary Tank contents through the Radwaste Secondary Ion Exchanger.

Which ONE (1) of the following radiation monitors will be the first to indicate the leak?

A. RE-7813, Radwaste Discharge Line Monitor.

B. RE-7865, Plant Vent Stack Wide Range Monitor.

C. RE-7828, Containment Purge Stack Monitor.

D. RE-7838, Sample Lab Isolation Monitor.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because this radiation monitor is normally used for radwaste discharge but it will not sense the leak from T-066.

B. Correct.

C. Incorrect. Plausible because individual may think that this monitor can be aligned to the Plant Vent Stack or the Containment Purge Stack like RE-7865.

D. Incorrect. Plausible because this monitor could detect the leak since it is in close proximity to the source of the leak, however, this monitor is physically isolated from the leak source.

Technical Reference(s) SD-SO23-690, page 65 (Attach if not previously provided)
SD-SO23-622, page 57

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 54381 & 81525 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 11

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	068 AA1.26	
	Importance Rating	3.6	

Ability to operate and / or monitor the following as they apply to the Control Room Evacuation: Unlocking of switches and operation of AFW valves

Proposed Question: Common 63

Given the following Unit 3 conditions:

- The Control Room was evacuated 15 minutes ago due to dense smoke.
- SO23-13-2, Shutdown from Outside the Control Room is in progress.
- Attachment 9, 33 Duties is being performed to establish AFW flow to the Steam Generator E-088.
- 3MP-141, Auxiliary Feedwater Pump is the only available AFW source.
- SG 3ME-088 is at 10% narrow range level.
- SG 3ME-089 is at 65% narrow range level.

Which ONE (1) of the following sets of actions will align flow to Steam Generator 3ME-088?

A. UNLOCK and OPEN S31305MU634, 3MP-504/3MP-141 Cross-connect.
UNLOCK and OPEN S31305MU635, 3MP-141/3MP-504 Cross-connect.

B. UNLOCK and OPEN S31305MU635, 3MP-141/3MP-504 Cross-connect.
Ensure S31305MU634, 3MP-504/3MP-141 Cross-connect is CLOSED.

C. UNLOCK and OPEN S31305MU634, 3MP-504/3MP-141 Cross-connect.
Ensure S31305MU635, 3MP-141/3MP-504 Cross-connect is CLOSED.

D. OPEN 3HV-4705, 3MP-140 Turbine Pump Discharge to SG E-088.
OPEN 3HV-4706, 3MP-140 Turbine Pump Discharge to SG E-089.

Proposed Answer: A

Explanation (Optional):

A. Correct.

B. Incorrect. Plausible because this action is procedurally correct, however, both cross-connect valves must be open for flow to occur.

C. Incorrect. Plausible because this action is procedurally correct, however, both cross-connect valves must be open for flow to occur.

D. Incorrect. Plausible because these valves would allow cross connecting the AFW Pumps, however, there are check valves in the lines.

Technical Reference(s): SO23-13-2, Attachment 9 (Attach if not previously provided)
SD-SO23-780, Figure 1
P & ID 40160A AFW

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 52579 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 4, 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	076 AK3.05	
	Importance Rating	2.9	

Knowledge of the reasons for the following responses as they apply to the High Reactor Coolant Activity: Corrective actions as a result of high fission-product radioactivity level in the RCS

Proposed Question: Common 64

Given the following:

- Unit 2 is at 90% power and is scheduled for a Refueling shutdown.
- Chemistry reports that gross gamma activity and fission product gases have increased in the Reactor Coolant.

Which ONE (1) of the following sets of actions will have the greatest effect at reducing fission product gases in the Reactor Coolant System?

A. Force Pressurizer Spray using Proportional Heaters.
Align PZR Degas to the VCT to maximize degas effect.

B. Force Pressurizer Spray using Proportional Heaters.
Align PZR Degas to the Radwaste header to maximize degas effect.

C. Force Pressurizer Spray using Proportional and Backup Heaters.
Align PZR Degas to the VCT to maximize degas effect.

D. Force Pressurizer Spray using Proportional and Backup Heaters.
Align PZR Degas to the Radwaste header to maximize degas effect.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because both of these actions will reduce RCS gas levels, however, aligning to the VCT does not maximize the effect.

B. Incorrect. Plausible because aligning to Radwaste header is correct, however, both sets of heaters will maximize spray flow.

C. Incorrect. Plausible because these actions will reduce RCS gas levels, however, aligning to the VCT does not maximize the effect.

D. Correct. This is the correct combination of actions to maximize removal of fission product gases.

Technical Reference(s) SO23-3-2.1, L & S 4.1 and 4.9 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 52378 & 56421 (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 5

Comments:

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	E09 EK1.2	
Importance Rating	3.2	

Knowledge of operational implications of the following concepts as they apply to the (Functional Recovery) Normal, abnormal and emergency operating procedures associated with (Functional Recovery).

Proposed Question: Common 65

Given the following conditions:

- The crew has entered SO23-12-9, Functional Recovery.
- Natural Circulation conditions exist in the RCS.
- Preparations are being made to cooldown on Natural Circulation to Shutdown Cooling entry conditions.
- No RAS has occurred.

Which ONE (1) of the following is the operational implication of performing a Natural Circulation cooldown in this condition?

A. Voiding in the head is NOT expected to occur.
Collapse any voids when Reactor Vessel Head level is less than 100% to prevent gas binding of the SDC Pumps when placed in service.

B. Voiding in the head is expected to occur.
Collapse the void when Reactor Vessel Head level is less than 100% to prevent disruption of Natural Circulation.

C. Voiding in the head is NOT expected to occur.
Collapse any voids when Plenum level is less than 100% to prevent disruption of Natural Circulation.

D. Voiding in the head is expected to occur.
Collapse the void when Plenum level is less than 100% to prevent gas binding of the SDC Pumps when placed in service.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because voiding is expected to occur, however, collapsing any voids that initially form in the head will impact the time to get on SDC.

B. Incorrect. Plausible because voiding is expected to occur, however, NC flow would not be disrupted until level is low in the Plenum.

C. Incorrect. Plausible because head voiding under normal circumstances is an undesirable situation, however, it will occur given the plant conditions.

D. Correct. This is the expected response when cooling down on NC while in the FR procedures.

Technical Reference(s) SO23-12-9, Step 13 (Attach if not previously provided)
SO23-14-9, Step 13 Bases

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 55217 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 5, 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	G2.1.25	
	Importance Rating	2.8	

Ability to obtain and interpret station reference materials such as graphs, monographs, and tables which contain performance data.

Proposed Question: Common 66

Unit 2 is in MODE 3 with plant status as follows:

- Main Feedwater is secured.
- Auxiliary Feedwater Pumps P-141 and P-504 are running to provide feedwater to the Steam Generators.
- Fire in a cable tray located in 45' Penetration Building has disabled 2LI-3204-2 (Condensate Storage Tank T-121 level indication) on CR-53.
- Unit 2 CRS has entered SO23-13-21, Fire.
- The CRS has directed you to proceed with SO23-13-21, Attachment 3 for local monitoring of Condensate Storage Tank T-121 level.
- Condensate Storage Tanks T-121 and T-120 are NOT cross-connected.

  - 2PI-3394L = 9.0 psig
  - 2PI-4701 = 8.0 psig
  - 2PI-4708 = 6.5 psig
  - 2PI-4734 = 7.0 psig

Which ONE (1) of the following is the quantity of makeup water that is available per Attachment 3?

A. 87,303 gallons

B. 93,755 gallons

C. 105,950 gallons

D. 118,201 gallons

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because this number can be determined by using incorrect pressure gauge (PI-4734 and PI-4708 interpolated value) on T-121.

B. Incorrect. Plausible because this number can be determined by using incorrect pressure gauge (PI-4734) on T-121.

C. Correct. This number is determined by applying PI-4701 on T-121.

D. Incorrect. Plausible because this number can be determined by using PI-3394L pressure on T-121.

Technical Reference(s) SO23-13-21, Attachment 3 (Attach if not previously provided)

Proposed references to be provided to applicants during exam: SO23-13-21, Attachment 3

Learning Objective: 53413 (As available)

Question Source: Bank # ____
Modified Bank # N127476 (Note changes or attach parent)
New ____

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 10

Comments:
Unit 2 is in MODE 3 with plant status as follows:
Main Feedwater secured. Auxiliary Feedwater Pumps P-140 and P-504 are running to provide feedwater to the Steam Generators. Fire in a cable tray located in 45' Penetration Building has disabled 2LI-3204-2 (T-121 level indication) on CR 53.Unit 2 CRS has entered SO23-13-21, Fire.
The CRS has directed you to proceed with SO23-13-21, Attachment 3 for local monitoring of T-121 level. (Assume that T-121 and T-120 **are** cross-connected.)
2PI-4701 = 6.5 psig 2PI-4708 = 8.0 psig 2PI-4734 = 7.0 psig
Which ONE (1) of the following will be the fluid volume given the AFW suction pressures?
A. 404,733 gallons
B. 289,110 gallons
C. 298,783 gallons
D. 358,915 gallons

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	G2.1.28	
	Importance Rating	3.2	

Knowledge of the purpose and function of major system components and controls.

Proposed Question: Common 67

Which ONE (1) of the following are the minimum plant conditions required to have a Diverse Emergency Feedwater System (DEFAS) actuation?

(Assume NO EFAS or MSIS has actuated.)

A. 2 out of 4 Steam Generator levels less than 16% narrow range.

B. 1 out of 2 Steam Generator levels less than 16% wide range.

C. 2 out of 4 Steam Generator levels less than 21% narrow range.

D. 1 out of 2 Steam Generator levels less than 21% wide range.

Proposed Answer: A

Explanation (Optional):

A. Correct. This is the correct logic and level for a DEFAS.

B. Incorrect. Plausible because SG level of 16% is the initiating condition, however, the logic is 2 out of 4 and SG narrow range levels are used.

C. Incorrect. Plausible because the logic and narrow range level is correct, however, 21% is the upper band to the DEFAS cycling relays, not the actuation setpoint.

D. Incorrect. Plausible because SG level is the initiating condition, however, the logic is 2 out of 4 SG narrow range levels and the actuating level is 16% not 21%.

Technical Reference(s) SD-SO23-720, pages 23, 26 & 28 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 56621 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 6, 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	G2.1.33	
	Importance Rating	3.4	

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

Proposed Question: Common 68

Which ONE (1) of the following conditions requires entry into a Technical Specification ACTION statement while Unit 2 is at 80% power?

A. Diesel Generator 2G002 Fuel Oil Day Tank level is 29 inches.

B. Fuel Oil Storage Tank 2T-035 level is 45,850 gallons.

C. Reactor Coolant System Tcold is 541ºF.

D. Charging Pump 2P-191 is tagged out for repairs.

Proposed Answer: A

Explanation (Optional):

A. Correct. Day Tank minimum Tech Spec level is ≥ 31.5 inches.
B. Incorrect. Fuel Oil Storage Tank minimum Tech Spec level is 41,800 gallons.
C. Incorrect. Minimum Tech Spec Tcold is 535ºF if > 30% Rated Thermal Power.
D. Incorrect. If 2P-191 is OOS, would only be in a Tech Spec Action Statement if other Charging Pump was OOS.

Technical Reference(s) Tech Spec SR 3.8.1.4 (Attach if not previously provided)
Tech Spec 3.8.3

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 55301 (As available)

Question Source: Bank #
Modified Bank # N126491 (Note changes or attach parent)
New

Question History: Last NRC Exam ____________________

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

10 CFR Part 55 Content: 55.41 10

Comments:
Which ONE (1) of the following conditions requires entry into a Technical Specification action statement while Unit 2 is at normal operating temperature and pressure?

A. Fuel Oil Storage Tank 2T035 level is 45,200 gallons.

B. Diesel Generator 2G002 Fuel Oil Day Tank level is 32 inches.

C. Tcold is 544°F

D. Charging Pump 2P191 is under clearance.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	G2.2.12	
	Importance Rating	3.0	

Knowledge of surveillance procedures.

Proposed Question: Common 69

Given the following:

- A surveillance test on HPSI Pump P-017 was being performed in MODE 4.
- Plant conditions require Return-to-Service of HPSI Pump P-017 prior to completion of the surveillance.
- The surveillance will be completed next week outside its specified time frame.

Which ONE (1) of the following describes the notification requirements for the missed surveillance?

A. Immediately notify the Work Process Supervisor.

B. Notify the Work Process Supervisor within one (1) hour.

C. Immediately notify the SRO Operations Supervisor.

D. Notify the SRO Operations Supervisor within one (1) hour.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because the WPS would be involved in work scheduling, however, they are not the responsible individual.
B. Incorrect. Plausible because the WPS would be involved in work scheduling, however, they are not the responsible individual.
C. Correct. Per Step 6.5.9 of SO23-3-3, Operations Surveillance Program Requirements.
D. Incorrect. Plausible because the SRO Ops Supervisor must be notified, however, the notification must be immediate.

Technical Reference(s) SO23-3-3, Steps 6.5.9 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 54907 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	G2.2.27	
	Importance Rating	2.6	

Knowledge of the refueling process.

Proposed Question: Common 70

Which ONE (1) of the following describes the MINIMUM requirement for Source Range Nuclear Instrumentation prior to commencing core off-load in MODE 6?

	Visual in CR	Audible in CR	Audible in CTMT
A.	2	0	1
B.	2	1	1
C.	1	1	1
D.	2	2	2

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because Source Range is correct.
B. Correct. Tech Specs requires 2 OPERABLE Source Range in CR, and audible in both CR and CTMT.
C. Incorrect. Plausible because minimum is met with one SR INOPERABLE.
D. Incorrect. Plausible because Source Range is correct.

Technical Reference(s) SO23-5-1.8, L & S 4.2 (Attach if not previously provided)
Tech Spec 3.9.2 and Bases

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 56319 (As available)

Question Source: Bank #
Modified Bank # N127204 (Note changes or attach parent)
New

Question History: Last NRC Exam ________________

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

10 CFR Part 55 Content: 55.41 __10__
____

Comments:
Prior to commencing core offload, a minimum of _______ audible Source Range NI channel(s) must be operable in the control room and a minimum of ________ audible Source Range NI channel(s) must be operable in containment.

A. 1 1
B. 1 2
C. 2 1
D. 2 2

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	G2.3.4	
	Importance Rating	2.5	

Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.

Proposed Question: Common 71

A new SONGS employee has the following radiation exposure in 2007:

- US Navy – 1825 mrem.
- PG&E at Diablo Canyon – 90 mrem.

<u>Without</u> Health Physics Manager authorization, which ONE (1) of the following is the MAXIMUM dose this employee may receive at SONGS in 2007?

A. 85 mrem

B. 175 mrem

C. 1085 mrem

D. 1910 mrem

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Represents the 2000 mrem limit, but this number is for total exposure at all facilities (3000 allowed).
B. Incorrect. Represents 2000 mrem limit, but using other facility exposure.
C. Correct. 1915 accumulated so far, the employee may receive up to 3000 mrem from all facilities combined for the year prior to requiring authorization for dose extension.
D. Incorrect. This would be correct for SONGS exposure alone, but employee would have exceeded the 3000 total threshold.

Technical Reference(s) SO123-VII-20.5, Section 6.1.4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective:	54709		(As available)
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	12	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	G2.3.1	
	Importance Rating	2.6	

Knowledge of 10 CFR: 20 and related facility radiation control requirements

Proposed Question: Common 72

An accessible area with a dose rate of 96 mrem per hour and contamination level of 32,550 DPM/100 CM^2 beta/gamma will be posted as a…

A. Contamination area and Radiation area

B. Contamination area and High Radiation area

C. High Contamination area and Radiation area

D. High Contamination area and High Radiation area

Proposed Answer: A

Explanation (Optional):

A. Correct. Greater than 5 but less than 100 mr/hr is radiation area, greater than 1,000/100 cm2 is contaminated area.

B. Incorrect. Plausible but must exceed 100 mr/hr to be called a High Radiation Area.

C. Incorrect. Plausible but must exceed 150,000 DPM/100 cm2 to be High Contamination Area.

D. Incorrect. Plausible but must exceed 150,000 DPM/100 cm2 to be High Contamination Area and 100 mr/hr to be called a High Radiation Area.

Technical Reference(s) SO123-VII-20, Attachment 1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 53334 (As available)

Question Source: Bank #

Modified Bank # N127113 (Note changes or attach parent)

New

Question History: Last NRC Exam ____________

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

10 CFR Part 55 Content: 55.41 12

Comments:
While reviewing an REP, an operator on tour of the plant notes one accessible
area that may be entered has a dose rate of 1073 mrem/hour and contamination levels of 31,162 dpm/100 cm2 beta/gamma.
This area should be posted as a:
A. Contamination area and high radiation area.
B. Contamination area and radiation area.
C. High contamination area and radiation area.
D. High contamination area and high radiation area.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	G2.4.25	
	Importance Rating	2.9	

Knowledge of fire protection procedures.

Proposed Question: Common 73

Which ONE (1) of the following correctly describes a "valid fire" per SO23-13-21, Fire?

A. Annunciator 61A15 - FIRE DETECTED has alarmed in the Control Room.

B. A PEO reports that a Local Fire Detection Panel is in alarm.

C. Any FIRE PUMP RUNNING indication is annunciated in the Control Room.

D. Verbal confirmation of the fire is reported to the Control Room.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because this is the one of the indications of a fire per SO23-13-21, however, this alarm could be due to a high energy line break.
B. Incorrect. Plausible because it is an indication of a fire per SO23-13-21, however, verbal confirmation must be obtained of the actual fire.
C. Incorrect. Plausible because all are indications of a fire per SO23-13-21, however, verbal confirmation must be obtained.
D. Correct. This is the criteria defined in SO23-13-21.

Technical Reference(s) SO23-13-21, Step 2.0 (Attach if not previously provided)
SO23-15-61.A1, Alarm 61A15

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 53413 (As available)

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

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 10

______

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	G2.4.7	
	Importance Rating	3.1	

Knowledge of event based EOP mitigation strategies.

Proposed Question: Common 74

Given the following:

- A Steam Generator Tube Rupture has occurred.
- The crew is performing SO23-12-4, Steam Generator Tube Rupture.
- RCS cooldown is in progress.

Which ONE (1) of the following describes the strategy for maintaining RCS pressure and temperature during the cooldown?

Maintain subcooling in the…

A. lower end of the Pressure/Temperature limits to allow backflow of ruptured SG into the RCS to prevent lifting SG safety valves.

B. lower end of the Pressure/Temperature limits to minimize RCS leakage.

C. higher end of the Pressure/Temperature limits to ensure continued RCP operation.

D. higher end of the Pressure/Temperature limits to prevent steam bubble formation in the Reactor vessel head.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. During the RCS cooldown, allowing backflow would likely be violating the P/T limits. Backflow is not initiated until cooldown and depressurization are complete.

B. Correct.

C. Incorrect. RCP operation is maintained at any point within the limits of the curve

D. Incorrect. As long as you are within the limits, saturation should not be an issue under the head if cooldown rate is maintained and RCPs are operating.

Technical Reference(s) SO23-14-4, Step 12 Bases (Attach if not previously provided)
SO23-12-4, Step 12a

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 53000 (As available)

Question Source: Bank #
Modified Bank # N11606 (Note changes or attach parent)
New

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 10

Comments:
SO23-12-4, Steam Generator Tube Rupture cautions the operators to maintain Post Accident Pressurizer Pressure and Reactor Coolant System Temperature within cooldown limits.
Which of the following describes why you want to maintain Post Accident Pressure and Temperature Limits at the lower end of the limits?
A. Reduces the amount of leakage from the Reactor Coolant System to the affected Steam Generator.
B. Reduces the amount of feed water inventory required for the affected Steam Generator.
C. Reduces the amount of feed water inventory required for the unaffected Steam Generator.
D. Reduces the amount of leakage from the affected Steam Generator to the Reactor Coolant System.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #	4	
K/A #	G2.4.10	
Importance Rating	3.0	

Knowledge of annunciator response procedures.

Proposed Question: Common 75

Which ONE (1) of the following annunciator window colors describes an 'equipment priority' alarm where a degradation of equipment functional capability has occurred?

A. RED

B. AMBER

C. WHITE

D. BLUE

Proposed Answer: B

Explanation (Optional):

A. Incorrect. System priority alarm.
B. Correct.
C. Incorrect. Control Room assessment alarm.
D. Incorrect. Delegated assessment alarm.

Technical Reference(s) ARPs (generic info) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 55177 (As available)

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	007 EA2.04	
	Importance Rating		4.6

Ability to determine or interpret the following as they apply to a reactor trip: If reactor should have tripped but has not done so, manually trip the reactor and carry out actions in ATWS EOP

Proposed Question: SRO 76

Given the following:

- A manual Reactor trip has been attempted.
- All Full-length CEAs remain out.
- An operator has been dispatched to open the Reactor Trip circuit breakers locally.
- The RO is attempting to establish RCS boration. Flow has NOT been verified.
- Reactor power is stable at 56%.

What are the proper actions by the operating crew in response to this event?

Continue attempts to emergency borate the RCS…

A. and immediately transition to SO23-13-11, Emergency Boration of the RCS / Inadvertent Dilution or Boration.

B. and immediately transition to SO23-12-9, Functional Recovery.

C. and complete the Standard Post Trip Actions then diagnose a Reactor Trip Recovery event.

D. and complete the Standard Post Trip Actions then diagnose a Functional Recovery entry.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Reactivity control is satisfied if a boration is in progress, but in this case it is not. Immediate transition not called for in SPTAs.

B. Incorrect. Immediate transition not required, and Reactivity Control is not satisfied. Complete SPTAs.

C. Incorrect. Would be correct if Reactivity Control was satisfied but it is not.

D. Correct. Reactivity Control is NOT satisfied.

Technical Reference(s) SO23-12-1, Attachment 1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 56252 (As available)

Question Source: Bank #
Modified Bank # N3927 (Note changes or attach parent)
New

Question History: Last NRC Exam SONGS 2006

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

10 CFR Part 55 Content: 55.41
55.43 5

Comments:
The Reactor has tripped and four (4) Full-length CEAs are stuck out. After opening the Reactor Trip circuit breakers locally, two (2) CEAs fall in. Reactor power is lowering and startup rate is negative.
What are the proper actions by the operating crew in response to this event?
A. Emergency borate the RCS, and immediately go to SO23-12-9, Functional Recovery.
B. Emergency borate the RCS, and immediately go to the SO23-12-2, Reactor Trip Recovery.
C. Emergency borate the RCS, finish the Standard Post Trip Actions, and diagnose a Functional Recovery entry.

D. Emergency borate the RCS, finish the Standard Post Trip Actions, and diagnose a Reactor Trip Recovery event.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	015 AA2.07	
	Importance Rating		2.9

Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): Calculation of expected values of flow in the loop with RCP secured

Proposed Question: SRO 77

Given the following:

- Unit 2 is at 100% power.
- RCP 2P-003 must be removed from service due to Controlled Bleedoff leakage into Containment of approximately 6 GPM.

Which ONE (1) of the following describes the procedure flowpath that will be required to remove 2P-003 RCP and what will be the RCS flow in the affected loops Steam Generator when the RCP is tripped?

A. Trip the Reactor and enter SO23-12-1, SPTAs; Flow through the affected Steam Generator is reduced by approximately half.

B. Initiate a controlled plant shutdown in accordance with SO23-5-1.7, Power Operation; Flow through the affected Steam Generator is reduced by approximately half.

C. Trip the Reactor and enter SO23-12-1, SPTAs; Flow through the affected Steam Generator is reversed.

D. Initiate a controlled plant shutdown in accordance with SO23-5-1.7, Power Operation; Flow through the affected Steam Generator is reversed.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Trip required for leakage >10 GPM.
B. Correct. Tripping 1 of 2 RCPs in the loop, SG flow will be approximately half.
C. Incorrect. Wrong procedure and no reverse flow with 1 RCP running.
D. Incorrect. No reverse flow with 1 RCP running.

Technical Reference(s)	SO23-13-6, Step 3		(Attach if not previously provided)
Proposed references to be provided to applicants during examination:	NONE		
Learning Objective:	55452		(As available)
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41		
	55.43	5	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	027 G2.1.2	
	Importance Rating		4.0

Conduct of Operations: Knowledge of operator responsibilities during all modes of plant operation.

Proposed Question: SRO 78

Given the following conditions:

- Unit 3 is at 100% power.
- PT-0100X, Pressurizer Control Channel X is the controlling PZR pressure channel.
- The following alarms are received:
  - o 50A04, PZR PRESSURE DEVIATION HI/LO
  - o 50A14, PZR PRESSURE HI/LO
- PT-0100X indicates 2200 psia and trending down.
- PT-0100Y indicates 2285 psia and trending up.

Which ONE (1) of the following describes the procedure required to mitigate the event, and the Technical Specification action required, if any?

A. Enter SO23-13-27, Pressurizer Pressure and Level Malfunction.
Technical Specification ACTION is not currently required.

B. Enter SO23-13-27, Pressurizer Pressure and Level Malfunction.
Restore Pressurizer pressure within 2 hours.

C. Enter SO23-3-1.10, Pressurizer Pressure and Level Control, Attachment for Foxboro Alarm Response and Foxboro Controller Page Data.
Technical Specification ACTION is not currently required.

D. Enter SO23-3-1.10, Pressurizer Pressure and Level Control, Attachment for Foxboro Alarm Response and Foxboro Controller Page Data.
Restore Pressurizer pressure within 2 hours.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Indication of a controlling pressure transmitter failure. Actual PZR pressure is outside of TS limits (2225-2275).(TS 3.4.1)
B. Correct.
C. Incorrect. AOI gives option of referring to this procedure if necessary but not the procedure required to mitigate the event.
D. Incorrect. Action is required, but AOI gives option of referring to this procedure if necessary but not the procedure required to mitigate the event.

Technical Reference(s) SO23-13-27, Step 3 (Attach if not previously provided)
Tech Spec Section 3.4.1

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 55213 & 56422 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 ____
55.43 2, 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	038 G2.1.20	
	Importance Rating		4.2

Conduct of Operations: Ability to execute procedure steps for SGTR

Proposed Question: SRO 79

The following conditions exist on Unit 2 after a seismic event:

- Steam Generator E-088 wide range level is 75% and rising with all feedwater secured.
- Steam Generator E-089 wide range level is 80% and rising with all feedwater secured.
- SG E-088 Main Steam Line Radiation monitor is in alarm and rising.
- SG E-089 Main Steam Line Radiation monitor is in alarm and rising.
- SIAS/CCAS/CRIS/CIAS/MSIS have actuated.
- Both Steam Generators are available for cooldown.

Which ONE (1) of the following actions is required and what will the crew implement to mitigate the event in progress?

A. Commence a cooldown using SG E-088.
Isolate the SG with the highest activity when Thot is less than 530ºF.

B. Commence a cooldown using SG E-088.
Isolate the SG with the highest level when Thot is less than 530ºF.

C. Commence a cooldown using SG E-088 and E-089.
Isolate the SG with the highest activity when Thot is less than 530ºF.

D. Commence a cooldown using SG E-088 and E-089.
Isolate the SG with the highest level when Thot is less than 530ºF.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because SG E-088 has a lower level and therefore an implication that the leak is greater in SG E-089, however, both SGs should be cooled down to prevent lifting a Main Steam Safety Valve on the isolated SG.

B. Incorrect. Plausible because SG E-088 has a lower level and therefore an implication that the leak is greater in SG E-089, however, both SGs should be cooled down to prevent lifting a Main Steam Safety Valve on the isolated SG.

C. Correct. Given the conditions listed it is desirable to cooldown using both SGs and then isolate the SG with the highest activity once identified. This strategy is determined by the SRO based on evaluation of SG activity and level.

D. Incorrect. Plausible because given the conditions listed it is desirable to cooldown using both SGs, however, the least affected SG may have a higher activity and therefore a greater potential for radiation release.

Technical Reference(s) SO23-12-4, Steps 4 & 7 (Attach if not previously provided)
SO23-14-4, Step 7

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 56252 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 ____
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	058 AA2.02	
	Importance Rating		3.6

Ability to determine and interpret the following as they apply to the Loss of DC Power: 125V dc bus voltage, low/critical low, alarm

Proposed Question: SRO 80

Given the following conditions:

- Unit 2 is at 100% power.
- Pressurizer Pressure and Level Control are both selected to Channel X.
- Annunciator 63A32 - 2D1 125 VDC BUS TROUBLE alarms in the Control Room.
- DC Bus 2D1 indicates 118 VDC.
- The dispatched PEO has NOT reported conditions in the 2D1 Battery Charger Room.

Given the information provided, which ONE (1) of the following is the impact on plant operations and what are the procedural implications?

A. Unit 3 RCP 3P-001 cannot be tripped from the Control Room. Refer to SO23-3-1.7, Reactor Coolant Pump Operation to transfer control power from Unit 2 to Unit 3.

B. Bus 2D1 is grounded. Refer to SO23-6-33, Ground Isolation to isolate the ground.

C. Pressurizer pressure and level rise. Stop Charging Pumps and heaters to control Pressurizer level and pressure per SO23-13-27, Pressurizer Pressure and Level Malfunction.

D. DC Control power to Bus 2A04 is degraded. Refer to SO23-6-15, Operation of 125 VDC Systems to initiate maintenance and place the battery on an equalizing charge if appropriate.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because the Unit 3 RCPs are normally supplied from the Unit 2 DC Bus and vice versa, however, DC Bus D5 is the source of control power.

B. Incorrect. Plausible because this annunciator can be an indication of a ground, however, there is insufficient information to make this determination until the PEO reports the status of the ground indication at DC Bus 2D1.

C. Incorrect. Plausible because Channel X is in service and Vital Bus Y-001 could be affected, however, there is no indication that VAC is lost.

D. Correct.

Technical Reference(s) SO23-6-15, Section 6.6 (Attach if not previously provided)
SO23-3-1.7, Attachment 8
SO23-15-63.A, 63A32

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 52762 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 ____
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	E06 G2.1.23	
	Importance Rating		4.0

Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Proposed Question: SRO 81

Given the following conditions:

- Unit 2 has tripped following a Bus 2A07 overcurrent trip and loss of vacuum.
- Auxiliary Feedwater Pump P-141 breaker will not close and P-504 was cleared for Boundary of the Week.
- Auxiliary Feedwater Pump P-140 is running but cannot develop sufficient discharge head.
- The CRS has completed SO23-12-1, Standard Post Trip Actions and has transitioned to SO23-12-6, Loss of Feedwater.

Which ONE (1) of the following actions is required?

A. Open Atmospheric Dump Valves to lower SG pressure to P-140 discharge pressure.
Transition to SO23-12-9, Functional Recovery, Attachment FR-5, Recovery - Heat Removal success path HR-1.

B. Transition to Step 10, Establish Condensate Pump flow to Available SGs.
Initiate SIAS and CCAS.
Align a Condensate Pump and depressurize the Steam Generators to 500 psig.

C. Initiate SIAS and CCAS.
Transition to Step 10, Establish Condensate Pump flow to Available SGs.
Open Atmospheric Dump Valves to lower SG pressure to P-140 discharge pressure.

D. Transition to SO23-12-9, Functional Recovery, Attachment FR-5, Recovery - Heat Removal success path HR-1.
Align a Condensate Pump and depressurize the Steam Generators to 500 psig.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because P-140 is running, however, since the discharge is not stated one cannot determine what the pressure is.

B. Correct. Transition to Step 10 then initiate SIAS and CCAS per SO23-12-6.

C. Incorrect. Plausible because these are the necessary recovery actions, however, they are being performed out of order.

D. Incorrect. Plausible because these are actions directed by the Loss of Feedwater procedure, however, at this point there is no reason to transition since Bus 2A03 is available and the depressurization can occur in SO23-12-6.

Technical Reference(s) SO23-12-6, Step 10 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 53001 (As available)

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	024 AA2.01	
	Importance Rating		4.1

Ability to determine and interpret the following as they apply to the Emergency Boration: Whether boron flow and/or MOVs are malfunctioning, from plant conditions

Proposed Question: SRO 82

Given the following:

- Unit 2 is in MODE 6.
- Refueling activities are in progress.
- Chemistry sample indicates Refueling Cavity boron concentration is below the Technical Specification limit.

Which ONE (1) of the following describes the action required, assuming all equipment operates as required?

A. Initiate boration at greater than 80 GPM using SO23-3-2.2, Makeup Operations. Boration flow may be verified by BAMU tank level trend.

B. Initiate boration at greater than 40 GPM using SO23-3-2.2, Makeup Operations. Boration flow may be verified by RWST level trend.

C. Initiate boration at greater than 80 GPM using SO23-13-11, Emergency Boration of the RCS/Inadvertent Dilution or Boration. Boration flow may be verified by RWST level trend.

D. Initiate boration at greater than 40 GPM using SO23-13-11, Emergency Boration of the RCS/Inadvertent Dilution or Boration. Boration flow may be verified by BAMU Tank level trend.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Wrong procedure and 80 GPM is flow from 2 Charging Pumps. Only 1 required.
B. Incorrect. Wrong procedure and wrong source of water.
C. Incorrect. Wrong source of water and only 1 Charging Pump required if all equipment is operating properly.
D. Correct. Enter SO23-13-11 for boron less than 2600 ppm during Refueling. If all equipment operates normally, 40 GPM from BAMU tank will provide adequate boration.

Technical Reference(s) SO23-13-11, Step 2b & 2j (Attach if not previously provided)
SO23-13-11, Entry Conditions

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 55510 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 ____
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	003 G2.1.12	
	Importance Rating		4.0

Ability to apply technical specifications for a system

Proposed Question: SRO 83

Given the following:

- Unit 2 is at 100% power with all CEAs fully withdrawn.
- A Group 6 CEA drops into the core.

Which ONE (1) of the following is the Technical Specification ACTION?

Reduce power level to _____ within _____ hour(s).

A. 90% one (1)

B. 98% one (1)

C. 90% two (2)

D. 95% two (2)

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because power must be reduced to 90% but this is the 120 minute power reduction requirement.
B. Incorrect. Plausible because this is the correct power reduction and time for a PLCEA.
C. Correct. This is the Tech Spec required action per the COLR and Tech Spec bases.
D. Incorrect. Plausible because this is the power reduction requirement for a Group 6 CEA, however, the time requirement is 60 minutes.

Technical Reference(s) Tech Spec 3.1.5 and Bases (Attach if not previously provided)
SO23-13-13, Step 2

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 54879 & 54876 (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		
	55.43	2	

Comments:

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		2
K/A #	069 AA2.01	
Importance Rating		4.3

Ability to determine and interpret the following as they apply to the Loss of Containment Integrity: Loss of containment integrity

Proposed Question: SRO 84

During a refueling outage with fuel movement in progress, which ONE (1) of the following would be considered a <u>loss</u> of Containment Closure?

A. Containment equipment hatch held in place by 4 equally spaced bolts.

B. Both Containment Personnel Airlock doors simultaneously open.

C. Containment Purge valves are open with a Purge in progress.

D. SG Secondary side manways off and a Main Steam Safety Valve is removed.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Equipment Hatch requires a minimum of 4 equally spaced bolts.
B. Incorrect. Airlock doors may be open in MODE 6 as long as capable of being closed and other restrictions met (Cavity level).
C. Incorrect. Purge may be in progress as long as rad monitors remain capable of isolating it if necessary.
D. Correct. Provides for a direct path to atmosphere that cannot be immediately isolated. This was documented in INPO OE Event # 362-950826-1 for SONGS.

Technical Reference(s): Tech Spec 3.9.3 (Attach if not previously provided)
Tech Spec 3.9.3 Bases
INPO OE Event # 362-950826-1

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 81449 (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		
	55.43	2	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	A13 G2.1.32	
	Importance Rating		3.8

Conduct of Operations: Ability to explain and apply all system limits and precautions.

Proposed Question: SRO 85

Given the following:

- Unit 2 was operating at 100% power when a complete loss of Component Cooling Water occurred.
- The crew has performed all actions required by SO23-13-7, Loss of CCW/SWC.
- The Standard Post Trip Actions have been completed.

Which ONE (1) of the following describes the procedure transition from SPTAs and the procedure and strategy required for a plant cooldown?

A. SO23-12-2, Reactor Trip Recovery.
Cooldown will be performed using SO23-5-1.4, Plant Shutdown to Hot Standby.

B. SO23-12-2, Reactor Trip Recovery.
Cooldown will be performed using SO23-12-11, Attachment 3, Cooldown / Depressurization.

C. SO23-12-7, Loss of Off-Site Power/Loss of Forced Circulation.
Cooldown will be performed using SO23-12-11, Attachment 3, Cooldown / Depressurization. Minimizing Reactor vessel upper head voids takes priority over RCS P/T limits during the cooldown.

D. SO23-12-7, Loss of Off-Site Power/Loss of Forced Circulation.
Cooldown will be performed using SO23-12-11, Attachment 3, Cooldown / Depressurization. RCS P/T limits take priority over minimizing Reactor vessel head voids during the cooldown.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Wrong procedure for RCPs tripped. SO23-5-1.4 would be correct if SO23-12-2 was used (See reference SO23-12-2, Step 9), however, SO23-12-1 directs you to SO23-12-7, Loss of Forced Circulation if RCPs are tripped.

B. Incorrect. Wrong procedure for RCPs tripped. SO23-12-2 would direct crew to SO23-5-1.4 (See reference SO23-12-2, Step 9), however, SO23-12-1 directs you to SO23-12-7, Loss of Forced Circulation if RCPs are tripped.

C. Incorrect. Correct procedure but wrong priority (See Caution before Step 14).

D. Correct. With a loss of CCW, all RCPs will be tripped, therefore a natural circulation cooldown will be performed and void formation would interrupt NC flow.

Technical Reference(s) SO23-12-7, Step 14 (Attach if not previously provided)
SO23-12-2, Steps 9

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 52560 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 ____
55.43 5

Comments:

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #	004 G2.1.23	
Importance Rating		4.0

Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Proposed Question: SRO 86

Given the following:

- Unit 2 is at 100% power.
- Pressurizer level control is selected to Channel X.
- Pressurizer level is slowly lowering.
- Letdown flow is slowly lowering, and currently indicates approximately 32 GPM.
- VCT automatic makeup is in progress.

Which ONE (1) of the following describes the event in progress, and the next action required?

A. RCS Leak; Enter SO23-13-14, RCS Leak and start additional Charging Pumps.

B. RCS Leak; Enter SO23-13-14, RCS Leak and isolate Letdown.

C. PZR Level Control System Malfunction; Enter SO23-13-27, Pressurizer Pressure and Level Malfunction and switch control to Channel Y.

D. PZR Level Control System Malfunction; Enter SO23-13-27, Pressurizer Pressure and Level Malfunction and place LIC-0110, PZR Level Controller in MANUAL.

Proposed Answer: A

Explanation (Optional):

A. Correct. Indications are of an RCS leak because VCT Makeup indicates that a loss of inventory is occurring as well as the Letdown flow at minimum

B. Incorrect. Correct procedure, but would only isolate letdown if all Charging pumps were running and level not stable

C. Incorrect. Wrong failure. If VCT level was stable, this could be chosen, but controller would be placed in manual first.

D. Incorrect. Plausible because the symptoms lead to a control channel failure with the exception of VCT AND letdown simultaneously being abnormal.

Technical Reference(s) SO23-13-14, Steps 1 & 2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 54932 (As available)

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41
55.43 5

Comments:

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #	034 G2.1.12	
Importance Rating		4.0

Ability to apply technical specifications for a system

Proposed Question: SRO 87

Given the following:

- Unit 3 is in MODE 5.
- Both Train A and B Fuel Handling Isolation System (FHIS) Radiation Monitors (RE-7822 & 7823) are OPERABLE.
- Train A Emergency Diesel Generator is in Maintenance Lockout.
- Train A and B Fuel Handling Building Post-Accident Cleanup Units (PACU) are available.
- Fuel Handling Building Normal Ventilation is currently in service.
- Train B CREACUS is in service.

Refueling Engineers have requested they be allowed to shuffle irradiated fuel and install a new rack (1500 lbs.) in the Spent Fuel Pool.

Which ONE (1) of the following describes when these operations should be allowed?

Spent Fuel Pool operations...

A. may begin as long as the Train B PACU Unit remains OPERABLE.

B. may NOT begin until the INOPERABLE Train A PACU Unit has been returned to OPERABLE status.

C. may begin as long as the Train B PACU Unit remains OPERABLE and is placed in service in the ISOLATE mode.

D. may NOT begin until the Train A PACU Unit has been placed in service in the ISOLATE mode.

Proposed Answer: A

Explanation (Optional):

A. Correct. Only one train of PACU is required to be OPERABLE per SO23-3-2.11.

B. Incorrect. Plausible because when this was a Tech Spec (old LCO 3.7.14), two trains were required to be OPERABLE. Not required per LCS 3.7.118, only one Train is required to be OPERABLE.

C. Incorrect. Plausible because in some conditions PACU must be OPERABLE, however, it is not required to be in ISOLATE.

D. Incorrect. Plausible because in some conditions PACU must be in ISOLATE, however, not for the conditions listed.

Technical Reference(s) LCS 3.7.118 (Attach if not previously provided)
SO23-3-2.11, Attachment 16, Steps 2.1 & 2.2

Proposed references to be provided to applicants during examination: LCS 3.7.118
SO23-3-2.11, Attachment 16

Learning Objective: 54863 (As available)

Question Source: Bank # ____
Modified Bank # N57915 (Note changes or attach parent)
New ____

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 ____
55.43 1, 2

Comments:
The following conditions exist:
· Both Train A and B Fuel Handling Isolation System (FHIS) radiation monitors (RE-7822 & 7823) are OPERABLE.
· Train A Fuel Handling Building Post-Accident Cleanup Unit (PACU) is *inoperable*.
· Train B PACU Unit is OPERABLE and OFF.
· Fuel Handling Building Normal Ventilation is in service.
Refueling Engineers have requested they be allowed to move irradiated fuel in the Spent Fuel Pool.
When should the movement of irradiated fuel be allowed?
A. Movement of irradiated fuel may NOT begin until the OPERABLE Train B PACU unit has been placed in service.
B. Movement of irradiated fuel may NOT begin until the *inoperable* Train A PACU unit has been returned to OPERABLE status.
C. Fuel movement MAY begin immediately, but in 7 days the OPERABLE Train B PACU unit must be placed in service in the ISOLATE mode.
D. Fuel movement MAY begin immediately, but in 7 days the OPERABLE Train B PACU unit must be placed in service in the PARALLEL mode.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #	006 G2.1.20	
Importance Rating		4.2

Conduct of Operations: Ability to execute procedure steps.

Proposed Question: SRO 88

Given the following conditions:

- Unit 2 has tripped after a LOCA and loss of Offsite Power.
- SIAS has actuated and all equipment is operating as designed.
- The crew is performing the actions of SO23-12-3, Loss of Coolant Accident.
- While evaluating FS-7, Verify SI Throttle/Stop Criteria, the following parameters are observed:
  - SG pressures are 1000 psia.
  - SG narrow range levels are approximately 22% and trending upwards.
  - Pressurizer Level is 65% and slowly rising.
  - Core Exit Saturation Margin is 9ºF.
  - Reactor Vessel Level is 100% Plenum.
  - Containment pressure is 1.2 psig and rising SLOWLY.

Which ONE (1) of the following describes the action required?

A. Initiate FS-32, Monitor RCS Solid Operation, and FS-30, Establish CVCS Letdown.

B. Secure one train of HPSI to limit the rise in Pressurizer level to avoid a Pressurized Thermal Shock transient.

C. Throttle or stop HPSI one train at a time per FS-7, Verify SI Throttle / Stop Criteria.

D. Maintain current conditions until all of the criteria are met per FS-7, Verify SI Throttle / Stop Criteria.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because with levels increasing plant could be solid in a short period of time, however, this action would not be required until > 80% per FS-7.

B. Incorrect. Plausible because some criteria are met, however, cannot secure HPSI Pumps until all criteria are met.

C. Incorrect. Plausible because some criteria are met, however, cannot secure HPSI Pumps until all criteria are met.

D. Correct.

Technical Reference(s) SO23-12-11, FS-7 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 55279 (As available)

Question Source: Bank #
Modified Bank # N126589 (Note changes or attach parent)
New

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41
55.43 5

Comments:

Given the following conditions:

- Unit 2 has tripped after Offsite Power was lost. SIAS has actuated and all equipment is operating as designed, including ECCS. The crew is performing the actions of SO23-12-3, Loss of Coolant Accident.
- While evaluating FS-7, Verify SI Throttle/Stop Criteria, the following parameters are observed:
  - SG pressures are 1000 psia.
  - SG narrow range levels are approximately 22% and trending upwards.
  - Pressurizer Level is 55% and rising.
  - Core Exit Saturation Margin is 9ºF.
  - Reactor Vessel Level is 100% Plenum.
  - Containment pressure is 1.2 psig and rising SLOWLY.

Which ONE (1) of the following describes the action required?

A. Continue monitoring SI Throttle/Stop criteria. Throttle or stop HPSI one train at a time when all criteria are met.

B. Stop Charging and LPSI pumps. Reset SIAS and ensure CIAS, CCAS, and CRIS are actuated. Throttle or stop HPSI one train at a time when all of the criteria are met.

C. Stop HPSI pumps to limit the rise in pressurizer level to avoid a pressurized thermal shock transient. Initiate FS-30, Establish CVCS Letdown Flow, and start all charging pumps.

D. Initiate FS-32, Monitor RCS Solid Operation, and FS-30, Establish CVCS Letdown flow.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	063 G2.2.25	
	Importance Rating		3.7

Equipment control knowledge of bases in technical specifications for limiting conditions for operations and safety limits.

Proposed Question: SRO 89

Given the following conditions:

- Unit 2 is in MODE 5 preparing to enter MODE 6.
- Train A is the Protected Train.
- Train B 125 VDC Bus D2 out-of-service for battery replacement.
- 125 VDC Bus D1 sustains a fault and is de-energized.

Which ONE (1) of the following describes the reason that Technical Specifications prevents entry into MODE 6?

A. The failure of DC Bus D1 also makes 120 VAC Bus Y-003 INOPERABLE.

B. Failure of protected train DC power raises the Shutdown Risk level to an unacceptable RED status.

C. The plant no longer meets the initial conditions assumed in the safety analysis of a redundant set of AC and DC power sources OPERABLE during an assumed loss of offsite AC power and single failure of one other AC source.

D. There is insufficient instrumentation and control power available to recover from a Fuel Handling Accident.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because loss of the DC Bus does render its associated AC Bus INOPERABLE, however, the wrong Bus is identified.
B. Incorrect. Plausible as it may be a true statement, but Shutdown Risk and TS are not interdependent.
C. Incorrect. Plausible because this is the basis for OPERABILITY in MODES 1 - 4, however, Unit is in MODE 5.
D. Correct. Per the Tech Spec Bases MODE 6 entry would not be allowed.

Technical Reference(s): Tech Spec 3.8.5 Bases (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 54863 (As available)

Question Source: Bank #
Modified Bank # 127030 (Note changes or attach parent)
New

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41
55.43 2

Comments:
Given the following conditions:
Unit 2 is in Mode 6.Refueling is in progress with an irradiated fuel assembly movement in progress in containment. Train "A" is the Protected Train. Train "B" 125V DC Bus D2 is out-of-service for battery replacement. 125 VDC Bus D1 sustains a fault and is de-energized. The Refueling crew is ordered to complete the move in progress and then suspend refueling operations.
Which ONE (1) of the following describes the reason that Technical Specifications requires suspending fuel movement?

A. There is insufficient instrumentation and control power available to recover from a postulated event, such as a Fuel Handling Accident.

B. The failure of DC Bus D1 also makes 120V AC distribution inoperable.

C. Failure of protected train DC power raises the Shutdown Risk level to an unacceptable status.

D. The plant no longer meets the initial conditions assumed in the safety analysis of a redundant set of AC and DC power sources operable during an assumed loss of off-site AC power and single failure of 1 other AC source.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	062 A2.12	
	Importance Rating		3.6

Ability to (a) predict the impacts of the following malfunctions or operations on the ac distribution system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Restoration of power to a system with a fault on it

Proposed Question: SRO 90

Given the following:

- A Station Blackout has occurred.
- The crew is performing SO23-12-8, Blackout.
- SO23-12-11, Attachment 8, Restoration of Off-Site Power, is in progress.
- While performing Attachment 8, the following alarm is identified:
    - o 63C21 - RES XFMR XR2 PROTECTION TRIP

Which ONE (1) of the following actions is appropriate for this condition?

A. Verify the System Separation Alarm is reset, then reset 63C21 and continue in Attachment 8.

B. Initiate SO23-6-6, Reserve Auxiliary Transformer Operation; continue actions for Emergency Faulted Reserve Auxiliary Transformer Operations.

C. Initiate SO23-6-6, Reserve Auxiliary Transformer Operation, and remove Generator Iso-Phase Bus disconnects to allow use of the Main Transformers.

D. Stop performance of Attachment 8, initiate repairs to XR2, and continue attempting restoration of EDGs in accordance with SO23-12-8, Station Blackout.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Actions are performed in procedure, but will not mitigate the transformer trip because 63C21 would not be reset in this condition.

B. Correct.

C. Incorrect. Wrong procedure use for correct alternate action.

D. Incorrect. May continue if you can make use of SO23-6-6. No need to stop Attachment 8.

Technical Reference(s) SO23-12-11, Attachment 8 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 55279 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 ____
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	071 A2.02	
	Importance Rating		3.6

Ability to (a) predict the impacts of the following malfunctions or operations on the Waste Gas Disposal System; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Use of waste gas release monitors, radiation, gas flow rate and totalizer

Proposed Question: SRO 91

Given the following conditions:

- A Waste Gas release is in progress on T-085, Waste Gas Decay Tank.
- 2/3 FV-7202 is open and releasing at 30 SCFM.
- Continuous Exhaust Fans A-311 and A-312 are in service.
- Radiation Monitor 2/3RT-7808G is aligned to the Continuous Exhaust Plenum.
- Radiation Monitor 3RT-7865-1 is aligned to the Unit 3 Containment Purge Stack. The purge was completed last shift.

During the release the following occurs:

- Radiation Monitor 2/3RT-7808G has failed high.
- Continuous Exhaust Fan A-311 trips on overcurrent.
- 2/3 FV-7202, Waste Gas Decay Tank Header Vent Valve has closed.

Which ONE (1) of the following:

a.) Identifies the cause of the closure of 2/3 FV-7202, Waste Gas Decay Tank Header Vent Valve?

b.) Identifies the action(s) required to continue the release?

A. a.) Tripping of Continuous Exhaust Fan A-311.
b.) Ensure 2RT-7865-1 is aligned to the Plant Vent Stack per SO23-8-14, Radwaste Gas Collection System Operation.

B. a.) Radiation Monitor 2/3RT-7808G failing high.
b.) Ensure 2RT-7865-1 is aligned to the Plant Vent Stack per SO23-8-14, Radwaste Gas Collection System Operation.

C. a.) Tripping of Continuous Exhaust Fan A-311.
b.) Bypass 2/3RT-7808G and align 3RT-7865-1 to the Plant Vent Stack to monitor the release per SO23-8-15, Radwaste Gas Discharge.

D. a.) Radiation Monitor 2/3RT-7808G failing high.
b.) Bypass 2/3RT-7808G and align 3RT-7865-1 to the Plant Vent Stack to monitor the release per SO23-8-15, Radwaste Gas Discharge.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because RT-7865 is used, however, it must be from Unit 3 and not Unit 2.
B. Incorrect. Plausible because the cause is correct, however, wrong procedure in use.
C. Incorrect. Plausible because the correct procedure is reference, however, both A-311 & A-312 must trip before FV-7202 will close.
D. Correct. With 3RT-7865 aligned and 2/3RT-7808G bypassed, the ODCM and procedure requirements are met.

Technical Reference(s): SO23-8-15, L & S 1.3 & 4.2 (Attach if not previously provided)
SO23-8-15, L & S 4.5, & 4.6

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 54022 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content:
55.41 ____
55.43 2

Comments:

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		2
K/A #	041 G2.4.30	
Importance Rating		3.6

Emergency Procedures / Plan Knowledge of which events related to system operations/status should be reported to outside agencies.

Proposed Question: SRO 92

Given the following conditions:

- The Unit was at 100% power when a failure of the Steam Bypass Control System (SBCS) caused the SBCS Valves to open.
- RCS temperature decreased to 528ºF.
- RCS pressure decreased to 2020 psig.
- Pressurizer level decreased to 15%.
- The Reactor was automatically tripped.
- SBCS was isolated during the performance of SPTAs.

Which ONE (1) of the following describes the earliest report required to the NRC?

A. 1 hour

B. 4 hours

C. 8 hours

D. 24 hours

Proposed Answer: B

Explanation (Optional):

A. Incorrect. 1 hour report not required unless event classification required.
B. Correct. Unplanned trip with auto RPS actuation.
C. Incorrect. An 8 hour report may also be required depending on plant conditions.
D. Incorrect. Unplanned trip requires a 4 hour report.

Technical Reference(s) SO123-0-A7, Attachment 1 (Attach if not previously provided)
SO123-0-A7, Attachment 5
SO123-0-A7, Step 6.3

Proposed references to be provided to applicants during examination:		SO123-0-A7, Attachment 1, Event Index
Learning Objective:	56187	(As available)
Question Source:	Bank #	
	Modified Bank #	(Note changes or attach parent)
	New X	
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43 5	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	056 G2.1.23	
	Importance Rating		4.0

Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Proposed Question: SRO 93

Given the following conditions:

- Unit 3 is at 100% power.
- All four Circulating Water Pumps are operating.
- Full Flow Condensate Polishing Demineralizers (FFCPD) are in service.
- The following Annunciators are in alarm:
    - 52A32 - CONDENSATE CATION CONDUCTIVITY HIGH
    - 52A34 - FEEDWATER CATION CONDUCTIVITY HI
    - 52A42 - CONDENSER NE HOTWELL CONDUCTIVITY HI
- Chemistry reports Condensate Cation Conductivity > 10 μS/cm.
- LV-3245, Condensate Drawoff Valve was placed in DISABLE.

Which ONE (1) of the following responses and associated procedure should be performed?

A. Perform a Rapid Power Reduction per SO23-5-1.7, Power Operations to preserve Main Feedwater Pump NPSH while overboarding.
Secure NE Condenser Circulating Water Pump after the power reduction to comply with NPDES limits.

B. Initiate a Reactor and Turbine trip then refer to SO23-12-1, Standard Post Trip Actions to preserve the Full Flow Condensate Polishing Demineralizer.
Stop the NE Condenser Circulating Water Pump to minimize introduction of contaminants.

C. Secure the NE Condenser Circulating Water Pump to comply with NPDES limits.
Initiate a Reactor and Turbine trip then refer to SO23-12-1, Standard Post Trip Actions to minimize chloride buildup in the Steam Generators.

D. Secure the NE Condenser Circulating Water Pump to minimize introduction of contaminants.
Perform a Rapid Power Reduction per SO23-5-1.7, Power Operations to preserve Main Feedwater Pump NPSH while overboarding.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because actions are required, however, Drawoff valve will prevent contamination and CW Pump should be immediately secured.

B. Incorrect. Plausible because actions would minimize contamination, however, with the adjacent CW Pump operating a Reactor trip is not required.

C. Incorrect. Plausible because tripping would minimize contamination of SG, however, FFCPD is in service and with the adjacent CW Pump operating a Reactor trip is not required.

D. Correct. These are the correct actions based on the conditions.

Technical Reference(s) SO23-13-9, Steps 1 & 2 (Attach if not previously provided)
SO23-13-9, L & S 1.4

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 54867 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 ____
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	G2.1.12	
	Importance Rating		4.0

Ability to apply technical specifications for a system

Proposed Question: SRO 94

Given the following plant conditions:

- Unit 2 is currently in MODE 3.
- At 1200 today, it is discovered that a Technical Specification required routine 24 hour surveillance was last performed at 0500 on the previous day.

Which ONE (1) of the following is the LATEST time the surveillance may be completed in accordance with Technical Specification requirements prior to declaring the associated equipment INOPERABLE?

A. 1300 today

B. 1800 today

C. 0500 tomorrow

D. 1200 tomorrow

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because the surveillance may be construed to be performed within one hour.
B. Incorrect. This time would represent the surveillance interval times 1.25 from the time of discovery of the missed surveillance.
C. Incorrect. This time indicates the 24 hour extension granted by TS SR 3.0.3 applied from the time the surveillance was missed, however, it is from the time of discovery.
D. Correct. The surveillance requirements are satisfied if the surveillance is completed by 1200 tomorrow, because of the 24 hour extension in TS SR 3.0.3.

Technical Reference(s) Tech Spec SR 3.0.3 Bases (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 56437 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 ____
55.43 2

Comments:

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Group #		2
K/A #	G2.2.10	
Importance Rating		3.3

Knowledge of the process for determining if the margin of safety, as defined in the basis of any technical specification is reduced by a proposed change, test or experiment.

Proposed Question: SRO 95

In accordance with SO123-0-A4, Configuration Control, when is a 10CFR50.59 Safety Evaluation required?

A. In support of Maintenance activities any time equipment is placed in an alternate alignment and must be documented on a Status Control Form.

B. In support of Maintenance activities when an alternate alignment will be in place for greater than or equal to 90 days.

C. When alternate alignments are directed by plant procedures with the exception of Annunciator Response Procedures and the equipment will remain in the off-normal position upon completion of the procedure.

D. When alternate alignments are directed by any plant procedures and the equipment will remain in the off-normal position upon completion of the procedure.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Only if alignment off normal for >90 days.
B. Correct.
C. Incorrect. Requires use of Status Control Form, but not 10CFR50.59 review.
D. Incorrect. Requires use of Status Control Form.

Technical Reference(s) SO123-0-A4, Step 6.5.1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 56338 (As available)

Question Source:	Bank #	______	
	Modified Bank #	______	(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam	______	
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis	______	
10 CFR Part 55 Content:	55.41	______	
	55.43	4	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		2
	K/A #	G2.2.19	
	Importance Rating		3.1

Knowledge of maintenance work order requirements.

Proposed Question: SRO 96

Given the following:

- Unit 2 is in MODE 5.
- Reduced Inventory operations are in progress.
- Work is being planned on equipment that MAY affect Shutdown Cooling System operation.

Which ONE (1) of the following describes the restriction on performance of this work in accordance with SO123-XX-5, Work Authorizations?

A. The effect must be annotated in the Capability Limitation section of the WAR; The WAR must be approved by the Manager, Plant Operations or his designee.

B. The effect must be annotated in the Capability Limitation section of the WAR; The WAR must be approved by the Control Room Supervisor.

C. This work is NOT allowed while Reduced Inventory Operations are in progress.

D. This work is NOT allowed while Reduced Inventory Operations are in progress UNLESS a 10CFR50.59 Safety Evaluation is completed and approved.

Proposed Answer: A

Explanation (Optional):

A. Correct.
B. Incorrect. CRS may not approve, authority is higher for this condition.
C. Incorrect. The work may be performed with restrictions.
D. Incorrect. 10CFR50.59 not required for Maintenance activities unless mods will be made to Systems, Structures and Components.

Technical Reference(s) SO123-XX-5, Part A, Step 6.3.5.3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 55428 (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 ____
55.43 4

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		3
	K/A #	G2.3.8	
	Importance Rating		3.2

Knowledge of the process for performing a planned gaseous radioactive release.

Proposed Question: SRO 97

Given the following conditions:

- A Gaseous Waste release is planned.
- Wind direction is blowing from the east.

Which ONE (1) of the following describes the status of the planned release in accordance with SO23-8-15, Gaseous Effluent Release?

The release…

A. CANNOT be initiated until wind direction changes.

B. may be initiated **ONLY IF** wind speed is below the minimum required by a calculation.

C. may be initiated **ONLY IF** wind speed is above the minimum required by a calculation.

D. is DESIRABLE and may commence without restriction.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. From the east is blowing toward the ocean. Conditions are desirable.
B. Incorrect. Requires higher wind speed if wind blowing toward land.
C. Incorrect. Higher wind speed is higher dispersion factor, and would be true if wind was blowing toward land.
D. Correct. Desirable if wind is blowing towards the ocean.

Technical Reference(s) SO23-8-15, Attachment 4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 53393 (As available)

Question Source: Bank #
Modified Bank # N127642 (Note changes or attach parent)
New

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41
55.43 4, 5

Comments:
Given the following conditions:

- A Gaseous Waste release is planned for Unit 2.
- Wind direction is from the ocean.

Which ONE (1) of the following describes the status of the planned release in accordance with SO23-8-15, Gaseous Effluent Release? The release…

A. CANNOT be initiated until wind direction changes.
B. may be initiated **ONLY IF** wind speed is below minimum required by a calculation.
C. may be initiated ONLY IF wind speed is above the minimum required by a calculation.
D. is DESIRABLE and may commence without restriction.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Group #		3
K/A #	G2.3.2	
Importance Rating		2.9

Knowledge of facility ALARA program.

Proposed Question: SRO 98

Which ONE (1) of the following states the Emergency Exposure Guideline (EPA-400) limit for Protecting Valuable Property at SONGS?

A. 5 REM TEDE.

B. 10 REM TEDE.

C. 25 REM TEDE.

D. 50 REM TEDE.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because this is the limit for annual exposure.
B. Correct. Per the Emergency Exposure Authorization Form.
C. Incorrect. Plausible because this is the old limit for protecting valuable property.
D. Incorrect. Plausible because this is the old limit for life saving activities.

Technical Reference(s) EPIP Form EP (123) 3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 55369 (As available)

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content:	55.41	
	55.43	4

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #	G2.4.40	
	Importance Rating		4.0

Knowledge of the SRO's responsibilities in emergency plan implementation.

Proposed Question: SRO 99

Given the following conditions:

- A declared emergency has been in progress for 35 minutes.
- An Alert has been declared.
- The Emergency Coordinator determines there is a need to reclassify the event as a Site Area Emergency.

Which ONE (1) of the following states the effect this change in the event classification will have on NOTIFICATIONS made to offsite agencies?

Changing the event classification requires...

A. performing Emergency Recall activation.

B. a four (4) hour notification to the NRC.

C. a new set of notifications.

D. a new set of notifications if any Protective Action Recommendations are also changed.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Requires both verbal and hard copy (printed) notifications
B. Incorrect. Requires both verbal and hard (printed) copy notifications.
C. Correct.
D. Incorrect. Not in accordance with procedure. Reclassification or change in PAR requires a new set of notifications.

Technical Reference(s) SO123-VIII-10, Steps 6.8 & 6.3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 55369 (As available)

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #	G2.4.38	
	Importance Rating		4.0

Ability to take actions called for in the facility emergency plan, including (if required) supporting or acting as emergency coordinator.

Proposed Question: SRO 100

Given the following:

Time	Event
0803	RCS leakage requires entry to AOI.
0807	Leakage quantified to be 36 GPM.
0818	Reactor Trip due to excessive RCS leakage.
0819	SIAS actuated.
0821	Alert declared.
0823	Site Area Emergency declared.

Which ONE (1) of the following describes the LATEST time that you must notify the state and NRC?

	State	NRC
A.	0822	0822
B.	0822	0907
C.	0836	0836
D.	0836	0921

Proposed Answer: D

Explanation (Optional):

A. Incorrect. 15 minutes from declaration. This is 15 minutes from TS LCO exceeded.
B. Incorrect. 15 minutes for TS LCO for state. 1 hour for NRC.
C. Incorrect. 15 minutes for state is correct, but NRC is 1 hour.
D. Correct.

Technical Reference(s) SO123-VIII-10, Step 6.3.1 (Attach if not previously provided)
SO123-VIII-10, Precaution 4.1

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 55369 (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 ______
55.43 5

Comments:

SONGS June 2007 NRC Written Exam Worksheet References

1. From SO23-13-6, Attachment 2

RCP NORMAL OPERATING AND MAXIMUM PARAMETERS

PARAMETER	NORMAL	MAXIMUM	NOTES
Controlled Bleed-off Temperature	< 170°F	240°F	[1] [2] [3]
CCW from RCP Seal Heat Exchanger Temperature	90 to 105°F	120°F	[1] [2]
CCW Temperature to RCPs	70 to 95°F	115°F	
Controlled Bleed-off Flow	1.25 to 1.75 gpm	3.5 gpm	[4]

RCP CAVITIES	AT 350 psia (NOMINAL)	2250 psia	MAXIMUM RANGES AT 2250 psia [5]
Vapor Seal Cavity Pressure (psi)	50	40-60	25-265
Upper Seal Cavity Pressure (psi)	150	795-895	560-980
Middle Seal Cavity Pressure (psi)	250	1475-1575	1305-1715

[1] Temperatures may not reach the lower range until RCS temperature and pressure reach normal power operation conditions and/or the RCP has been operating for 1 hour.

[2] CBO temperature can be expected to generally run 50-70°F hotter than RCP seal CCW return temperatures. Temperatures above these maximum values can result in deterioration of the vapor seal carbon element and may lead to eventual seal failure. Seal elastomers are Ethylene Propylene Rubber and are rated for temperatures up to 300°F without degradation.

[3] CBO temperature ≥ 170°F following a sudden unexplained, large, and sustained temperature rise, is indicative of rotating Baffle Bolt Failures. Under this condition, there may be no appreciable change in seal staging pressure, and no appreciable change in CBO flow. Experience has shown that elevated CBO temperatures will persist for 24 to 48 hours prior to total baffle failure and resultant loss of CBO flow. If CBO temperature is greater than 200°F for 2 hours, then it is anticipated that Maintenance Engineering will recommend a plant shutdown. (Ref. 5.4.3)

[4] At an indicated flow of 3.5 gpm, 1 gpm is flowing through the breakdown coils and the remaining 2.5 gpm flows across the seal faces. As the indicated flow approaches 3 gpm, which corresponds to 2 gpm across the seal face, the sealing face of the stationary carbon ring begins to erode causing further seal degradation.

[5] Pressures outside of the ranges indicate the possibility of one or more failed seals. Attachment 1 is used to determine if a seal(s) has failed.

1. From SD-SO23-360, page 22

.3 **Seal cartridge** (Figures I-9, 11, and 12) consists of four face type mechanical seals; three full pressure seals mounted in tandem, and a fourth low pressure vapor seal designed to withstand system operating pressure when the pump is not operating. A controlled bleedoff flow of ~ 1.5 gpm through the seal is used to cool the seals and to equalize the pressure drop across each seal. The bleedoff flow is collected in the Volume Control Tank (VCT) of the CVCS as shown in Figure I-13. Leakage past the vapor seal is collected in the containment sump.

The seal cartridge assembly is cooled by allowing controlled leakage from the RCS to flow through a heat exchanger (Figure I-14) integral with the pump case and then past the seal cartridge assembly. The seals are designed for operation for up to thirty minutes with no cooling water with no seal damage. Seal design accommodates full RCS operating pressure; however, the first three seals of the cartridge assembly normally operate with a differential pressure of one-third of system pressure, with a very small differential pressure across the vapor seal. The seal rotors are tungsten carbide operating against a graphite stator.

2. From SD-SO23-360, page 29

2.2.5 **Reactor Coolant Pumps (P001, P002, P003, and P004)** (Continued)

.27 **RCP oil lift pumps** (P-260, P-261, P-262, P-263, P-264, P-265, P-266, P-267), are used to lift the rotating assembly during startup and shutdown of the RCPs. Each RCP has two oil lift pumps associated with it. Each oil lift pump has an associated backlit switchlight module control located on CR56. Each oil lift pump also has a control switch on the emergency shutdown panel L-42. The following list gives the oil lift pumps and their associated controls for each RCP.

RCP P-001	L-42	CR56
P-260	HS-9108B	HS-9108A
P-261	HS-9109B	HS-9109A

RCP P-002	L-42	CR56
P-262	HS-9117B	HS-9117A
P-263	HS-9118B	HS-9118A

RCP P-003	L-42	CR56
P-264	HS-9111B	HS-9111A
P-265	HS-9112B	HS-9112A

RCP P-004	L-42	CR56
P-266	HS-9114B	HS-9114A
P-267	HS-9115B	HS-9115A

The controls on CR56 for the oil lift pumps have four pushbuttons with positions for START, STOP, NORMAL and STANDBY. The normal and standby pushbuttons determine in which mode of control the oil lift pumps operate. In the "normal" mode, the oil lift pump will automatically start when the RCP is < 90% of normal speed. The control room operator would manually stop the oil lift pump after he received the RCP zero speed indication. In the "standby" mode, the oil lift pump will start automatically 15 seconds after the RCP is < 90% of normal speed provided that the redundant pump's discharge pressure was less than normal.

2. From SD-SO23-360, page 31

2.2.5 **Reactor Coolant Pumps (P001, P002, P003, and P004)** (Continued)

.30 **RCP anti-reverse rotation device pumps** (P-399, P-400, P-401, P-402, P-403, P-404, P-405, P-406) are used to circulate lube oil from ARRD to the lube oil cooler. Each RCP has two ARRD pumps associated with it. Each ARRD pump has an associated backlit switchlight module control located on CR56 and a control switch on the emergency shutdown panel L-42. The following list gives the ARRD pumps and their control for each RCP.

<u>RCP P-001</u>	L-42	CR56
P-399	HS-9166B	HS-9166
P-400	HS-9167B	HS-9167
<u>RCP P-002</u>	L-42	CR56
P-401	HS-9196B	HS-9196
P-402	HS-9197B	HS-9197
<u>RCP P-003</u>	L-42	CR56
P-403	HS-9176B	HS-9176
P-404	HS-9177B	HS-9177
<u>RCP P-004</u>	L-42	CR56
P-405	HS-9186B	HS-9186
P-406	HS-9187B	HS-9187

The controls on CR56 for the ARRD pumps have four pushbuttons with positions for START, STOP, NORMAL and STANDBY. The normal and standby pushbuttons determine which mode of control the ARRD pumps operate. In the "normal" mode, the ARRD pump will automatically start when the RCP is < 90% of normal speed. The control room operator would stop the ARRD pump after he received the RCP zero speed indication. In the standby mode, the ARRD pump will start automatically 15 seconds after the RCP is < 90% of normal speed if there is a low discharge flow on the alternate pump.

3. From SD-SO23-390, page 42

2.2 Components (Continued)

2.2.17 **Ion Exchanger Bypass Valve, 2(3)TV-0224B** (Figure I-1)

PURPOSE:	Bypass letdown flow around the Purification Ion Exchangers
DESIGN PRESS:	100 psig
SIZE:	3"
OPERATOR:	Air
TYPE:	3-Way
FAIL POSITION:	TO VCT (deenergized)
SETPOINT:	140°F

.1 Ion Exchanger Bypass Valve, 2(3) TV-0224B, will BYPASS Letdown Flow around the Purification Ion Exchangers on a high Letdown Temperature of 140°F

.1.1 This is accomplished by a 3-way Temperature Control Valve, located upstream of the Purification Ion Exchangers.

.1.2 It is designed to divert Letdown Flow directly to the VCT.

3. From SO23-3-2.1, L & S 3.2

3.0 **ION EXCHANGERS**

3.1 An unused Deborating Ion Exchanger has the capacity to lower RCS Boron concentration by approximately 60 ppm at any time during core life. Following this, no further reduction in Boron concentration is possible using the Deborating Ion Exchanger.

3.2 Changes in letdown temperature to the Purification IXs will cause changes in RCS Boron concentration. (Higher temperature will raise Boron, lower temperatures will lower concentration.) [Ref. 2.2.1]

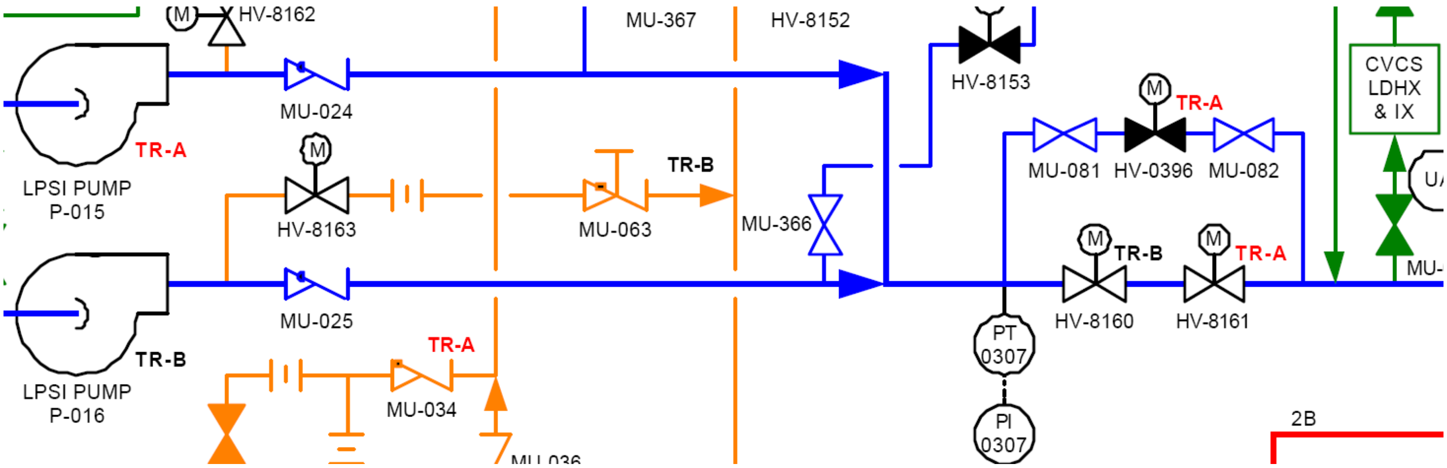
4. From SO23-3-2.6, Step 6.2.7

6.2.7 If SDC flow adjustments are necessary, then perform the following: (LS-2.0)

.1 Throttle the in-service LPSI Header Injection Valves (HV-9322, HV-9325, HV-9328, HV-9331), and/or in-service SDC HX Bypass Flow Valve (HV-8160, HV-0396), as required.

.2 Adjust the SDC Hi/Lo Flow alarm based on SDC Flow per Step 6.2.2 (CFMS Page 314).

4. From SD-SO23-740, Figure 1



4. From SO23-3-2.6, Attachment 1

2.7 Bypass Flow Valve Alignment

2.7.1 Requisite step completed: Step 2.4 ________

2.7.2 ENSURE CLOSED 2(3)HV-0396, SDC HX Bypass Standby Flow Valve. ________

2.7.3 CYCLE 2(3)HV-8160, SDC HX Bypass Normal Flow Valve, CLOSED then OPEN while verifying proper valve response. (Ref. 2.2.15) ________

✔	HV-8160 RESULTS	MARK N/A
	SAT	Steps 2.7.4.1 through 2.7.4.4
	UNSAT	

2.7.4 Place 2(3)HV-0396, SDC HX Bypass Standby Flow Valve in service, as follows:

STEP	NUMBER OF COMPONENT	NOUN NAME	REQUIRED POSITION	INITIALS PERF	IND VER
.1	2(3)HV-8160	SDC HX Bypass Normal Flow Valve	CLOSED	______	N/A
.2	2(3)BRB-15	HV-8160 SDC HX Bypass Normal Flow Valve Breaker	OPEN	______	______
.3	2(3)BRA-13	HV-0396 SDC HX Bypass Standby Flow Valve Breaker	CLOSED	______	______
.4	Initiate a LCOAR/EDMR, and indicate action taken in the Comments section.			______	N/A

5. From SO23-3-2.6, L & S 3.3

3.3 Due to slow leak-by of the Standby SDC HX outlet valve, <u>when</u> the RCS is > 140°F, <u>then</u> CCW flow should be directed through both SDC HXs. Otherwise, the Standby SDCHX will heat up to RCS temperature.

6. From Lesson Plan 2XE102 Handout

BUS & MCC LOCATIONS AND POWER SUPPLIES

BUS	LOCATION	NORMAL	ALTERNATE	EMERGENCY
2A01	45' PEN	2XU2	2XR3	3XR3
2A02	63' PEN	2XU2	2XR3	3XR3
2A03	30' TBSG	2XU1	2XR1	
2A04	*50' CB*	*2XR1*	*3A04/2XU1*	*DG G002*
2A06	*50' CB*	*2XR2*	*3A06/2XU1*	*DG G003*
2A07	30' TBSG	2XU1	2XR2	
2A08	85' CB	2XU1	2XR1	
2A09	85' CB	2XU1	2XR2	

BUS	LOCATION	SUPPLY
2B01	45' PEN	2A08
2B02	45' PEN	2A08
2B03	30' TBSG	2A03
2B04	*50' CB*	*2A04*
2B06	*50' CB*	*2A06*
2B07	30' TBSG	2A07
2B08	63' PEN	2A09
2B09	63' PEN	2A09
B10	85' CB (U2)	2(3)A08
2B11	30' TBSG	2A07
2B12	30' TBSG	2A03
2B13	30' TBSG	2A03
2B14	30' TBSG	2A07
2B15	85' CB	2A09
2B16	85' CB	2A08
B17	30' AWS	2A03*
2B18	56' TB	2A07
B19	HFMUD	SDG&E

*2A03 only/Alternate SDG&E

MCC	LOCATION	SUPPLY
2BA	45' PEN	2B01
2BB	7' TB	2B03
2BC	34' TB	2B13
2BD	*2G002*	*2B04*
2BDX	2G002	2B14
2BE	*50' CB*	*2B04*
2BF	30' AFW	2B03
BG	30' CB (U2)	2(3)B15
2BH	*2G003*	*2B06*
2BHX	2G003	2BDX
2BI	34' TB	2B11
2BJ	*50' CB*	*2B06*
2BK	7' TB intake	2B07
2BL	30' TB	2B07
2BLX	30' TB	2B12
2BM	7' TB	2B11
2BMX	30' TB	2B13
2BN	63' PEN	2B09
BO	70' CB closet	B10
BP	50' CB	B10
BQ	*50' CB*	*2(3)B04*
2BRA	*50' CB*	*2B04*
2BRB	*50' CB*	*2B06*
2BRC	34' TB	2B18
BRD	HFMUD	B19
BRE	HFMUD	B19
BS	*50' CB*	*2(3)B06*
BT	50' RW	2(3)B16
BU	50' RW	2(3)B15
2BV	34' TB	2B12
2BW	7' TB	2B14
2BX	50' CB	2B16
2BY	*50' CB*	*2B04*
2BZ	*50' CB*	*2B06*

NOTE: 1E items are *shaded & in italics*

Load Center DM from 2B11 is located at the OCC

(Located in NDMS file 2XE102HO2 / Revised on 09/12/03)

7. From SO23-3-1.4, Step 2.6

2.6 Establishing a Pressurizer Bubble (LS-12.14)

GUIDELINES

1. Sections 2.6 and 2.7 may be performed concurrently.
2. When bubble formation occurs, then the auto setpoints of PIC-0201A and B, Letdown Backpressure Controller, will need to be lowered in response to the RCS pressure rise.
3. Annunciator UA58A22, LETDOWN FLOW HI, is expected during this evolution.

2.6.1 Ensure RCS Pressure is at 350 ± 10 psia. ________

2.6.2 INITIATE heatup of the Pressurizer for steam bubble formation, as follows:

.1 ENSURE RCS ≥ 125°F. (LS-12.10) ________

.2 ENSURE Quench Tank in service: ________

- Level 75 to 80%.
- Oxygen < 1% (Step 2.3.13)
- Nitrogen blanket established

.3 INITIATE plotting Pressurizer temperature every 30 minutes using PMS point T101 per SO23-5-1.3, Attachment for Heatup Log. ________

8. From SO23-13-7, Attachment 1

COMPONENT COOLING WATER EMERGENCY MAKEUP

CONTINUOUS USE

OBJECTIVE

To provide direction for using the CCW Emergency Makeup System to maintain level in the CCW Surge Tanks. This Attachment should be used when the normal nitrogen supply system for the CCW Surge Tanks is in service.

1.0 PROCEDURE

PERF. BY INITIALS

1.1 Determine the performance requirements of this attachment: (Leave unused Sections blank) ________

✓	EVOLUTION	PERFORM
	Makeup to CCW Train A	Section 1.2
	Makeup to CCW Train B	Section 1.3
	Fire Water System Makeup to CCW	Section 1.4

NOTES

1. T-056 (Unit 2) or T-055 (Unit 3), Primary Plant Makeup Storage Tank, is the CCW Makeup source for P-1018 (Train A) and P-1019 (Train B). These pumps discharge into piping downstream of the Letdown HX Return Valves at a maximum flow rate of =45 gpm.
2. A portable Oxygen monitor and flashlight may be required for entry into the Primary Make-up Tank room following an earthquake or fire.
3. To prevent overfilling the CCW Surge Tank, level must be monitored continuously. CCW Makeup Pumps DO NOT have a high level auto-stop feature.
4. If HV-6569 or HV-6570, CCW Makeup Pump Discharge Valves, fail to start opening within 5 seconds after pump start, then the pump will trip.
5. Overfilling the CCW Surge Tank may cause the level instrument to fail low and automatically isolate the CCW Noncritical Loop.

From SD-SO23-690, page 8

.5.1 Component Cooling Water (CCW) Non Critical Loop Radiation

.5.1.1 2(3)RE-7819 consists of a detector mounted on the Non-Critical Loop CCW return line at the 30 foot elevation Penetration Area near the west wall. It is used to indicate a leak of a radioactive component into the CCW system (measuring gross gamma activity, predominantly Cs-137). The function of this monitor is alarm and indication only.

9. From SD-SO23-400, page 9

.5 The Component Cooling Water System is designed to respond automatically to Engineered Safety Feature Actuation Signals as follows:

.5.1 **SIAS** (Safety Injection Actuation Signal) ensures one component cooling water pump on each loop is started.

.5.2 **CIAS** (Containment Isolation Actuation Signal) isolates the Non-Critical Loop by closing the supply and return valves to/from both loops, and isolates the cooling supply and return lines to the cooled components located within the Containment. The affected components in the Containment are served by the Non-Critical Loop and consist of the reactor coolant pump motor and bearing coolers, and the Control Element Drive Mechanism (CEDM) coolers.

.5.3 **CSAS** (Containment Spray Actuation Signal) opens the return line stop valves allowing cooling water flow through the Shutdown Heat Exchangers.

.5.4 **CCAS** (Containment Cooling Actuation Signal) opens the Component Cooling Water Supply and Return Lines to the four Containment Emergency Cooling Units (ECU).

10. From SO23-3-1.4, Attachment 4, Step 2.5.6

2.5.6 Verify Pressurizer bubble formation occurred: ________

.1 Pressurizer water temperature at 432°F with RCS pressure at 350 psia.

.2 Letdown flow greater than Charging flow.

.3 Pressurizer level decreasing with pressure remaining constant or increasing.

.4 If PV-0201A and B, Letdown Backpressure Control Valves, reach full open, and RCS pressure is still increasing, then Stop Charging Pumps and/or turn Off PZR Heaters to maintain pressure.

11. From SD-SO23-710, pages 5, 8 and 11

.3 The Departure from Nucleate Boiling Ratio and Local Power Density trips are generated by the Core Protection Calculators (CPC) using inputs from Reactor Coolant Pump Speed, Hot Leg and Cold Leg Temperatures, Control Element Assembly position, Pressurizer Pressure and Excore Nuclear Instrumentation Flux Power.

INITIATING DEVICE: Core Protection Calculator

.3.1 The Local Power Density High Trip is provided to prevent the Linear Heat Rate (kW/ft) in the limiting fuel rod in the core from exceeding the fuel design limit in the event of an Anticipated Operational Occurrence.

.4 **LOW DEPARTURE FROM NUCLEATE BOILING RATIO (DNBR) TRIP** (Continued)

.4.1 The Departure from Nucleate Boiling Ratio Low Trip is provided to prevent the Departure from Nucleate Boiling Ratio in the limiting coolant channel in the core from exceeding the fuel design limit in the event of Anticipated Operational Occurrences. The Departure from Nucleate Boiling Ratio is calculated in the Core Protection Calculator.

.4.2 The trip variable is calculated by the Core Protection Calculator, incorporates various uncertainties and dynamic compensation routines to assure a trip is initiated prior to violation of fuel design limits. These uncertainties and dynamic compensations ensure that a reactor trip will occur prior to violating the Departure from Nucleate Boiling Ratio Limiting Safety System Settings.

11. From Tech Spec 3.3.1 Bases

The RPS is segmented into four interconnected modules. These modules are:

- Measurement channels;
- Bistable trip units;
- RPS Logic:
  - Matrix Logic
  - Initiation Logic (trip paths)
- Reactor trip circuit breakers (RTCBs).

This LCO addresses measurement channels and bistable trip units. It also addresses the automatic bypass removal feature for those trips with operating bypasses. The RPS Logic and RTCBs are addressed in LCO 3.3.4, "Reactor Protective System (RPS) Logic and Trip Initiation." The CEACs are addressed in LCO 3.3.3, "Control Element Assembly Calculators (CEACs)."

<u>Measurement Channels</u>

Measurement channels, consisting of field transmitters or process sensors and associated instrumentation, provide a measurable electronic signal based upon the physical characteristics of the parameter being measured.

The excore nuclear instrumentation, the core protection calculators (CPCs), and the CEACs, though complex, are considered components in the measurement channels of the Linear Power Level — High, Logarithmic Power Level — High, DNBR — Low, and Local Power Density (LPD) — High trips.

Bistable Trip Units

Bistable trip units, mounted in the Plant Protection System (PPS) cabinet, receive an analog input from the measurement channels. They compare the analog input to trip setpoints and provide contact output to the Matrix Logic. They also provide local trip indication and remote annunciation.

There are four channels of bistables, designated A, B, C, and D, for each RPS parameter, one for each measurement channel. Bistables de-energize when a trip occurs, in turn de-energizing bistable relays mounted in the PPS relay card racks.

The contacts from these bistable relays are arranged into six coincidence matrices, comprising the Matrix Logic. If bistables monitoring the same parameter in at least two channels trip, the Matrix Logic will generate a reactor trip (two-out-of-four logic).

12. From SO23-13-18, Attachment 4

.6	RX Trip Paths 3 and 4 Actuated	• RTCBs 3, 4, 7, and 8 Open. ☐ VERIFY RX Trip Path 1 and 2 lights LIT. ☐ VERIFY RTCBs 1, 2, 5 and 6 are CLOSED. ☐ VERIFY RX Trip Path 3 and 4 indicating lights EXTINGUISHED.

13. From SD-SO23-740, pages 29 & 30

2.2.7 Safety Injection Tanks, 2(3)T007, 008, 009 & 010

TYPE:	Vertical, right cylindrical
DESIGN PRESSURE:	700 psig
DESIGN TEMPERATURE:	200°F
INTERNAL VOLUME:	2250 ft^3
NOMINAL LIQUID VOLUME:	1743 ft^3
PRESSURIZING GAS:	Nitrogen
FLUID:	Borated water (1.25% wt. boric acid)
NORMAL OPERATING PRESSURE:	625 psia
NORMAL OPERATING TEMP.:	120°F
OPERATING LIMITS (MODES 1-3)	
LIQUID VOLUME:	1680-1807 ft^3 (77.9 - 84.1%)
BORON CONCENTRATION:	2200-2800 ppm
NITROGEN PRESSURE:	615-655 psia
LOCATION:	45' and 90' Containment

The four Safety Injection Tanks (SIT) provide a means to flood the core with borated water following depressurization due to a large break LOCA, and keep it covered until flow from the safety injection pumps becomes available. For purposes of safety analysis, it is assumed that the inventory of one SIT is lost through spillage.

13. From SD-SO23-740, pages 29 & 30

2.2.7 **Safety Injection Tanks, 2(3)T007, 008, 009 & 010** (Continued)

During normal plant operation, each SIT is isolated from the Reactor Coolant System by two check valves in series. The SITs automatically discharge into the core through each of the cold legs if Reactor Coolant System pressure decreases below SIT pressure during reactor operation.

Small break loss of coolant accident analysis indicates that for the smallest break requiring the SIT, the RCS pressure remains above the SIT pressure for approximately 45 minutes. As such, SIT availability would be required for at least 45 minutes into a loss of coolant accident. This small break loss of coolant accident scenario was the limiting condition for SIT availability until analyzing for a Station Blackout event. Station blackout analysis in response to 10CFR 50.63 concludes SI tanks initial discharge occurs at 2.2 hours and continues to 4 hours. At 4 hours, electrical power is assumed to be restored and normal makeup is resumed (Reference 4.7.1).

Design Basis Events other than LOCA may cause Reactor Coolant System (RCS) depressurization to the SIT Pressure and result in partial discharge of SIT volume. These include a Main Steam Line Break (SLB) and Station Blackout (SBO).

13. From SD-SO23-740, pages 29 & 30

Loss of Coolant Accident (LOCA)

For Large Break LOCAs, RCS depressurization is rapid and the SITs discharge immediately to assist in refilling the lower plenum, assist in reflooding the core, and limit the clad temperature (PCT) to below 2200°F. For Small Break LOCAs, RCS depressurization is gradual and may or may not reach the SIT discharge pressure (615 psia) depending on break size.

For break sizes larger than 0.05 ft^2, the SITs discharge before the maximum PCT occurs (UFSAR Table 15.6-12) which indicates the SITs are required to assist in mitigating and prevent fuel damage. For break sizes equal to or smaller than 0.05 ft^2, the SIT discharge after the maximum PCT occurs or SIT discharge may not occur at all. For 0.05 ft^2 break analyzed in the UFSAR, PCT occurs at 35 minutes and SIT discharge occurs at 36 minutes. The PCT is turned around by HPSI/Charging Flow prior to SIT discharge hence, the SITs aid in core recovery but are not required to prevent fuel damage.

14. From TS 3.3.1.A

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one automatic RPS trip channel inoperable.	A.1 Place Channel in bypass or trip.	1 hour
	AND	
	A.2 Restore channel to OPERABLE status.	Prior to entering MODE 2 following next MODE 5 entry

14. From TS Table 3.3.1-1

3. Pressurizer Pressure — High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 2385 psia

14. From TS 3.3.6.A

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- This action also applies when three Matrix Logic channels are inoperable due to a common power source failure de-energizing three matrix power supplies. ---------------------- One or more Functions with one Matrix Logic channel inoperable.	A.1 Restore channel to OPERABLE status.	48 hours

14. From TS 3.5.2.A

3.5.2 ECCS — Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with pressurizer pressure $\geq$ 400 psia.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One LPSI subtrain inoperable.	A.1 Restore subtrain to OPERABLE status.	7 days

14. From TS 3.5.4.A

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RWST boron concentration not within limits. OR RWST borated water temperature not within limits.	A.1 Restore RWST to OPERABLE status.	8 hours

15. From SD-SO23-770, page 33

2.2.8 **Chiller Unit, 2(3)E-201 and E-202** (Figures 7 & 7a) (Continued)

.20.2 **Alarms**

.20.2.1 Chiller Alarms are Chiller TRIP parameters which protect the Chiller by TRIPPING it OFF.

.20.2.2 Cabinet Alarms are used to monitor the Cabinet's internal health and do not TRIP the Chiller.

.20.2.3 Red border on "ALARM" button on all screens

.20.2.3.1 Operator should depress "ALARM" (on local Panel) to display Alarm Screen

.20.2.3.2 Will cause a "CONTAINMENT CHILLER E201 PROCESS PROTECTION" alarm on 2(3)L-154

.20.2.3.3 Will cause a "CONTAINMENT CHILLED WATER SUPPLY TROUBLE" alarm on 2/3CR-60A04

.21 **Shutdown Alarm Parameters & Screen:**

.21.1 Cooler Refrigerant Liquid Temperature, <31.5ºF

.21.2 Entering Condenser TPCW Pressure, <30 psig

.21.3 Condenser Pressure, >161 psig

.21.4 Compressor Refrigerant Gas Discharge Temperature, >220ºF

.21.5 Oil Pump ΔP, <13 psig

.21.6 Bearing Oil Temperature, >220ºF

.21.7 Leaving Chill Water Temperature, <38ºF (Low Load Recycle)

.21.8 Chill Water Flow <850 gpm/52a contact OPENS (Chill Water Loss of Flow)

15. From SO23-1-4, Step 6.15

6.15 **Containment Temperature Control**

INFORMATION USE

1. If average air temperature exceeds 100 F, then PLACE the Standby Containment Normal Cooler in service. (Ref. 2.2.4)

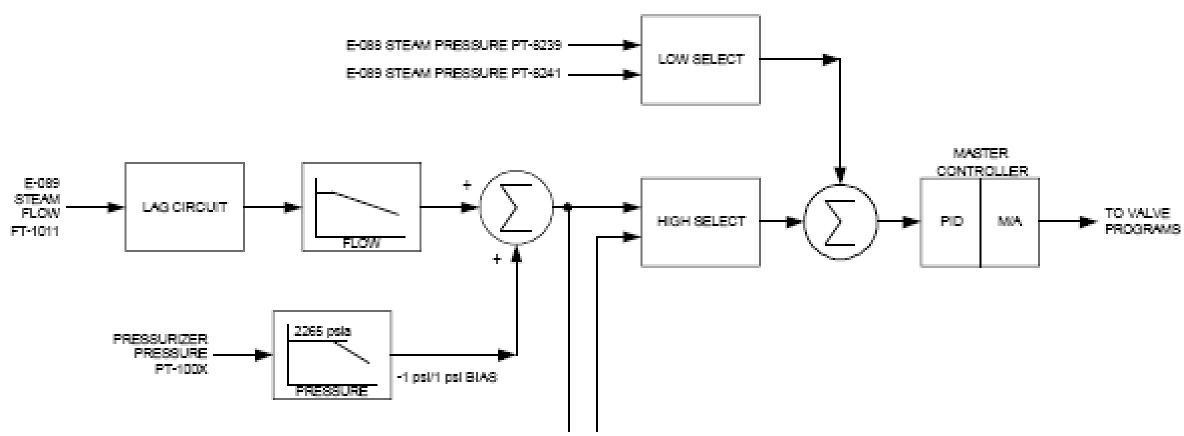
2. If average air temperature exceeds 105 F, then refer to SO23-1-4.1. (Ref. 2.2.4)

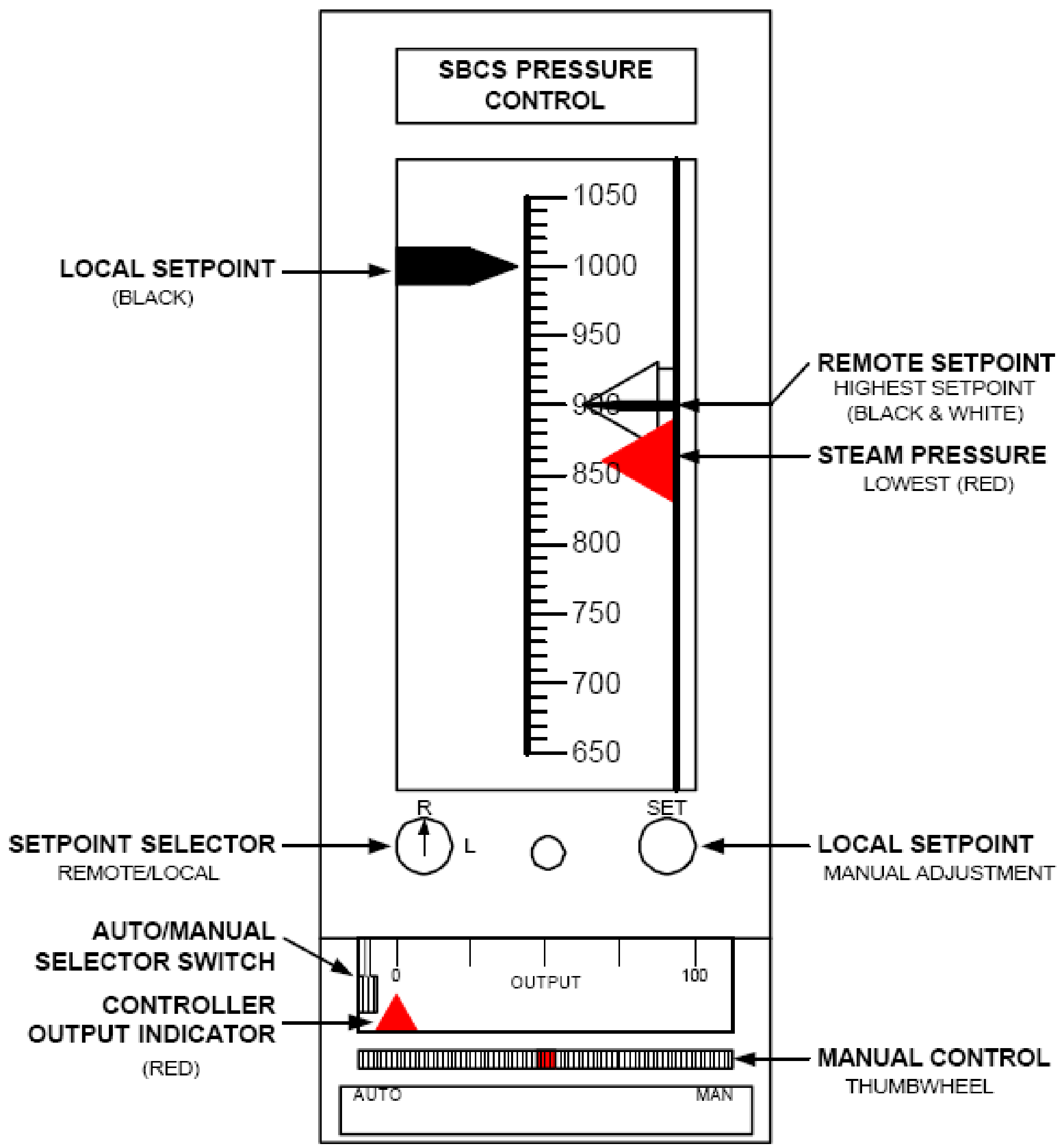
16. From SD-SO23-740, page 63

COMPONENT	TRAIN	TYPE	UNIT 2	UNIT 3
Safety Injection Miniflow Isolation Valve HV-9306	A	480 VAC	2BY09	3BY09
Safety Injection Miniflow Isolation Valve HV-9307	A	480 VAC	2BY13	3BY13
Safety Injection Miniflow Isolation Valve HV-9347	B	480 VAC	2BJ09	3BJ09
Safety Injection Miniflow Isolation Valve HV-9348	B	480 VAC	2BJ13	3BJ13
Containment Spray Header Isolation Valve HV-9367	A	480 VAC	2BE29	3BE29
Containment Spray Header Isolation Valve HV-9368	B	480 VAC	2BJ25	3BJ25

17. From SD-SO23-175, Figures 9 & 13

FIGURE 9: MODULATE AND PERMISSIVE PROGRAM





17. From Lesson Plan 2XIR05 SBCS SH SHW, Slide 72

Failure Mechanisms

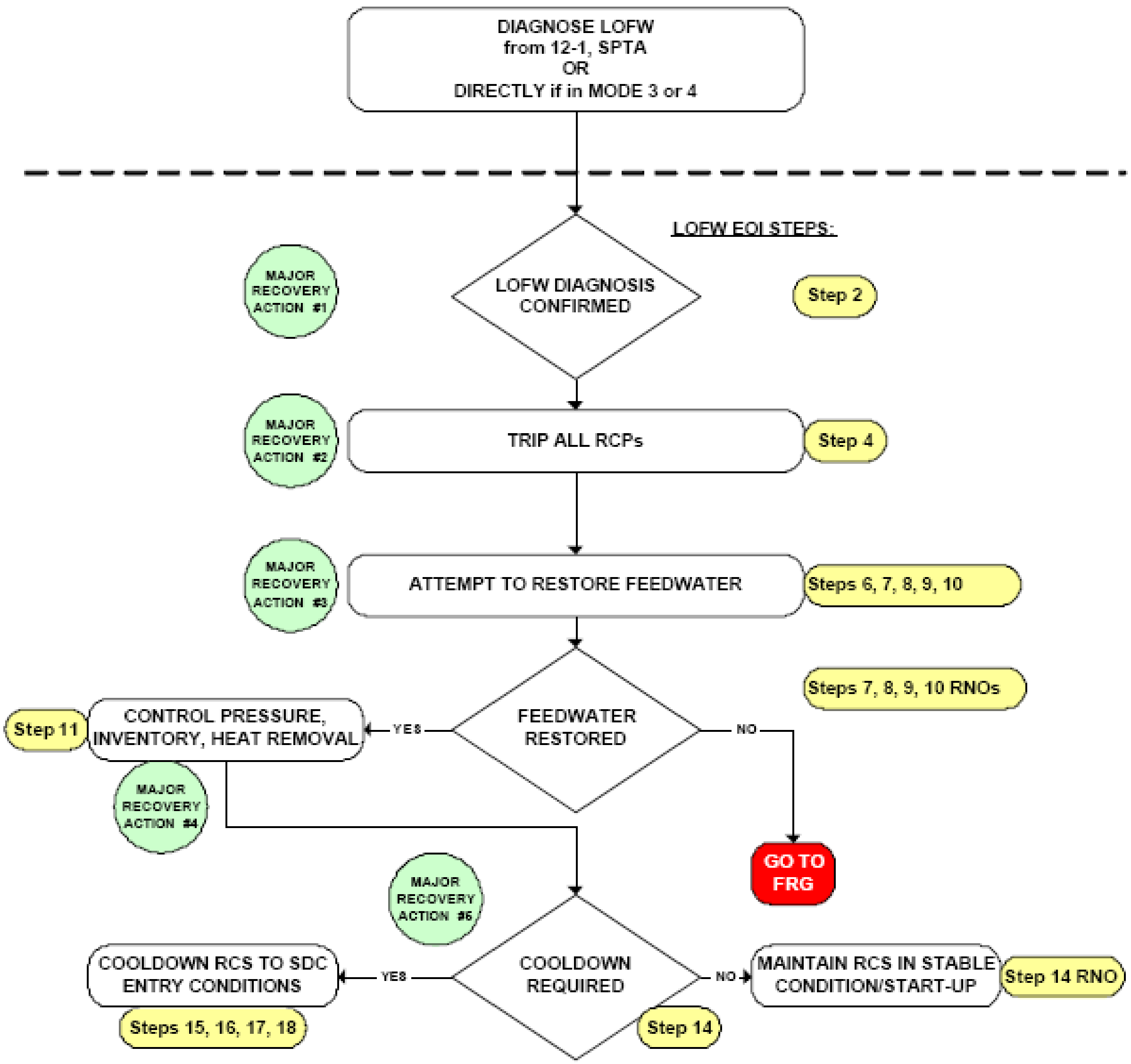
- Steam Flow fails high:
  - No visible impact to SBCS.
  - Prevents QO due to loss of rate-of-change input.
- Steam Flow fails low:
  - Drives remote setpoint (b&w pen) on Master Controller high.
  - Prevents QO due to loss of rate-of-change input.
- Steam Pressure fails high:
  - SBCS Permissive lights on.
- Steam Pressure fails low:
  - Red pen (steam pressure input) fails low on Master Controller.

18. From SO23-14-6, Attachment 1

LOSS OF FEEDWATER BASES AND DEVIATIONS JUSTIFICATION

EOI STEP BASES

Figure 1, LOFW Major Recover Actions



18. From SO23-14-6, Section 3

3.0 RECOVERY TECHNIQUE

The Recovery Actions are directed toward determining the cause of the interruption in adequate feedwater flow, regaining adequate feedwater flow from AFW or MFW, and recovering the plant to a stable condition. Also, during the event, actions are taken to ensure adequate RCS inventory and heat removal.

The recovery actions are directed toward conserving the available S/G water inventory, thereby maximizing the time to S/G dryout. Stopping the RCPs minimizes heat input to the RCS and closing S/G blowdown and sample valves minimizes water losses from the S/Gs. Actions are taken to restore MFW or AFW to the S/Gs. Direction is provided to reinitiate MFW or AFW with reduced flow to reduce thermal shock to the S/G. This action minimizes the possibility of damage to the S/Gs that could result in losing the S/G capability to provide cooling or damage to the S/G tubes (i.e., a S/G tube leak or rupture).

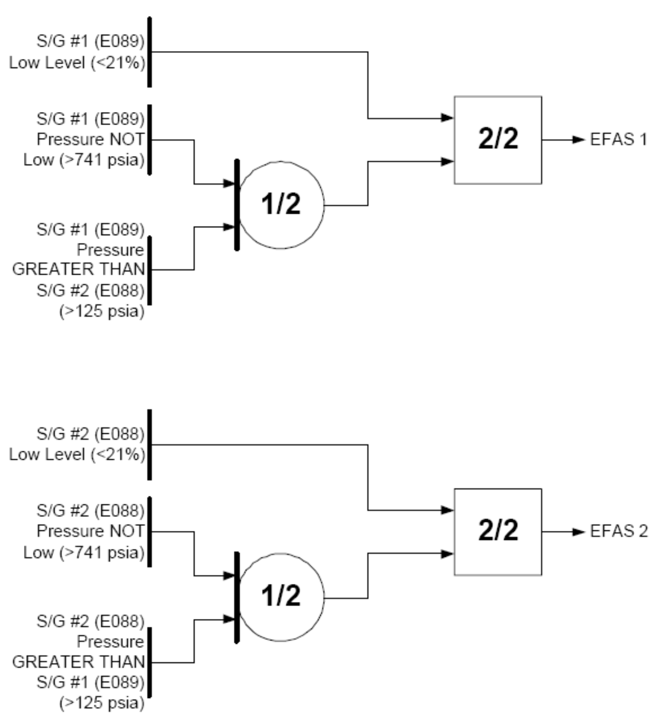
19. From SD-SO23-780, page 11 and Figure 13

2.2.1 **Motor-Driven Auxiliary Feedwater Pumps, 2(3)P-141 & 504**

PUMP TYPE:		Byron Jackson, 4" x 6" x 90", horizontal 8-stage DVMX, centrifugal
PRIME MOVER:	Unit 2:	Allis-Chalmers type AZ, 4160 VAC, 3-phase, 3600 rpm, induction motor
	Unit 3:	Siemens-Allis type AZ 4160VAC, 3 phase, 3600 rpm, induction motor

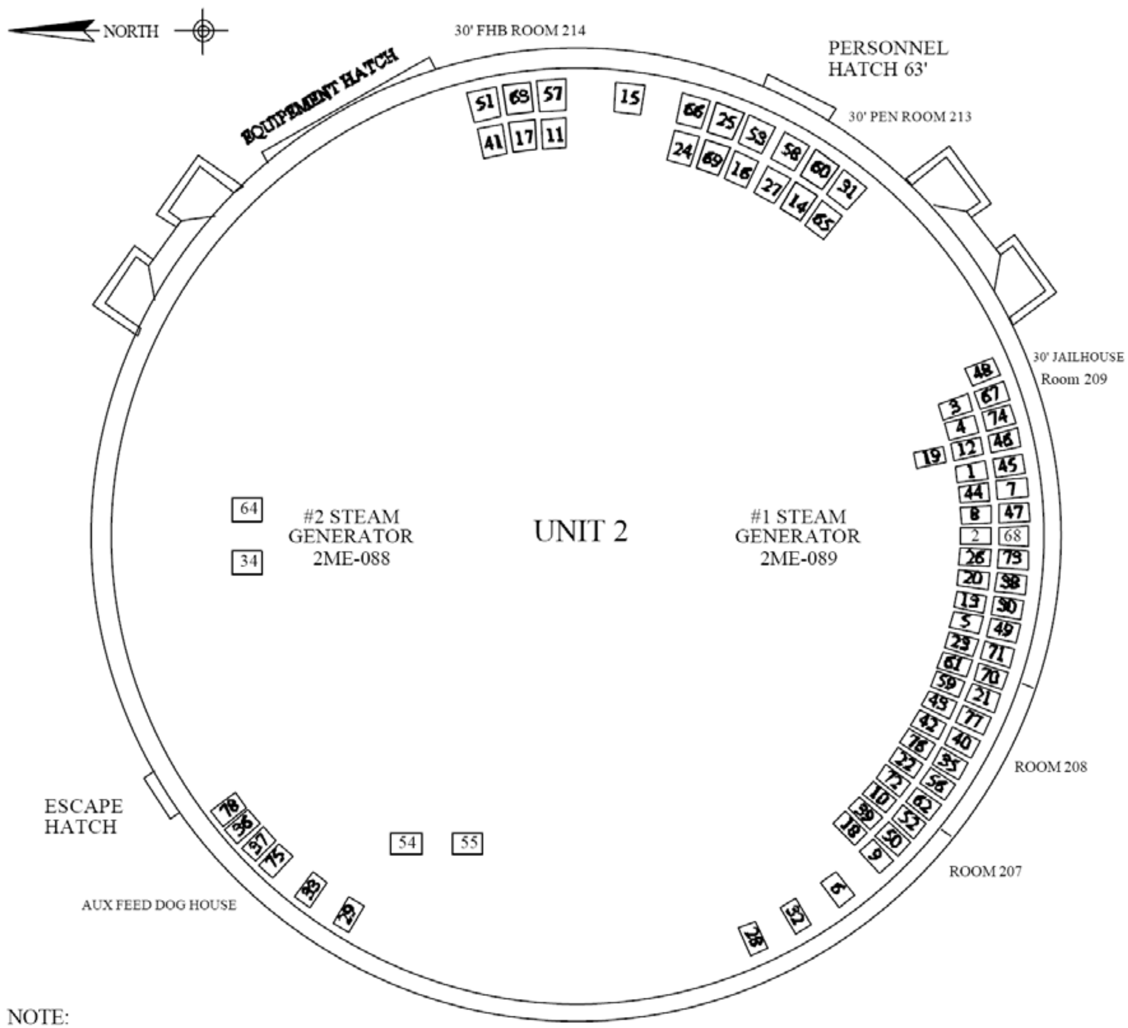
PUMP:

FLUID:	Condensate
DESIGN FLOW RATE:	860 gpm
NET FLOW RATE:	760 gpm (design minus recirculation)
DESIGN HEAD:	2842 ft. (1230 psia)
DESIGN TEMPERATURE	100°F
DESIGN NPSH REQUIREMENT:	18 ft. (7.8 psia)
DESIGN RPM:	3570 rpm
DESIGN SHUT-OFF HEAD:	~3170 ft. (1372 psia)



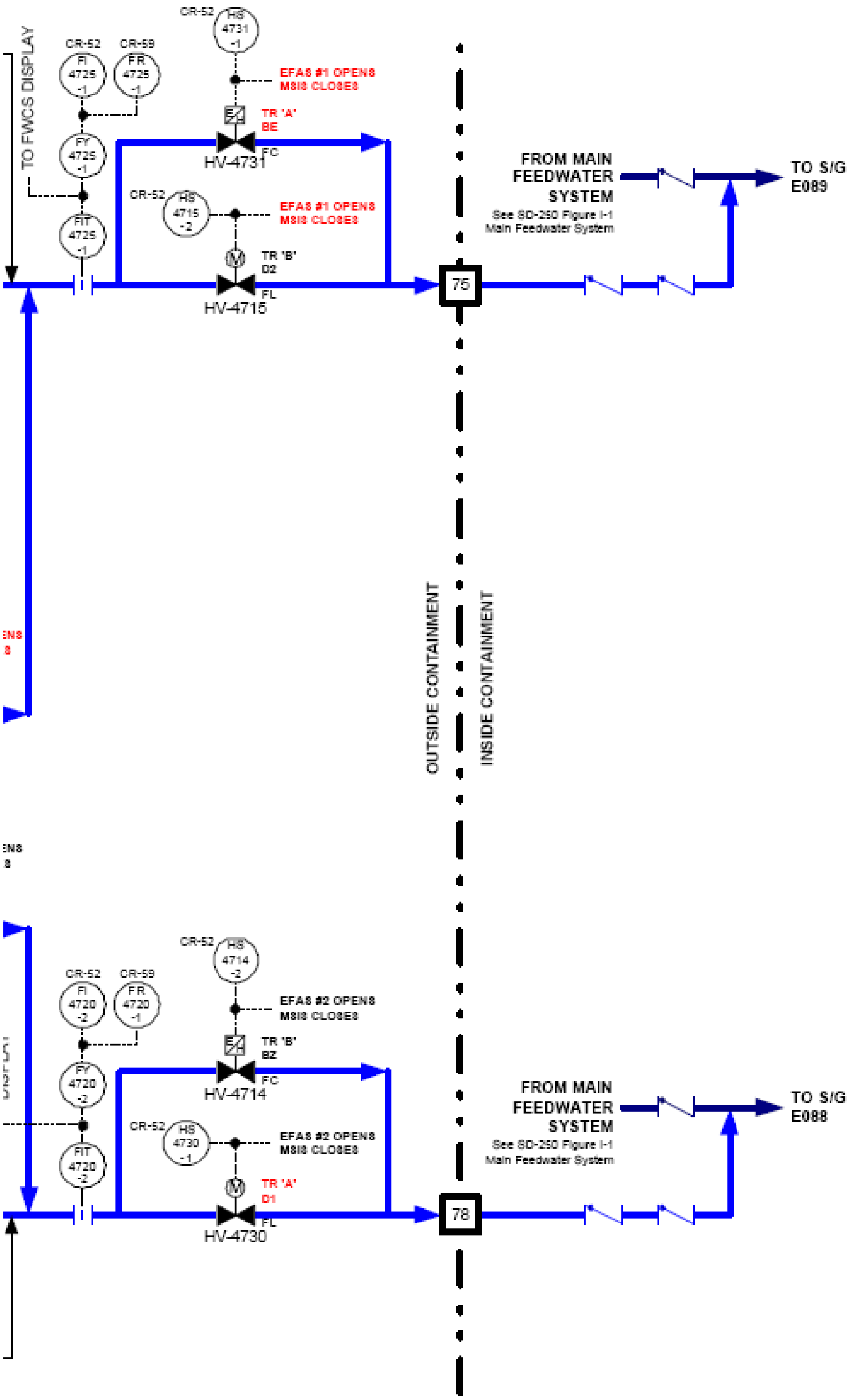
20. From SO23-3-3.10, Attachment 5

PIPING PENETRATIONS UNIT 2 CONTAINMENT



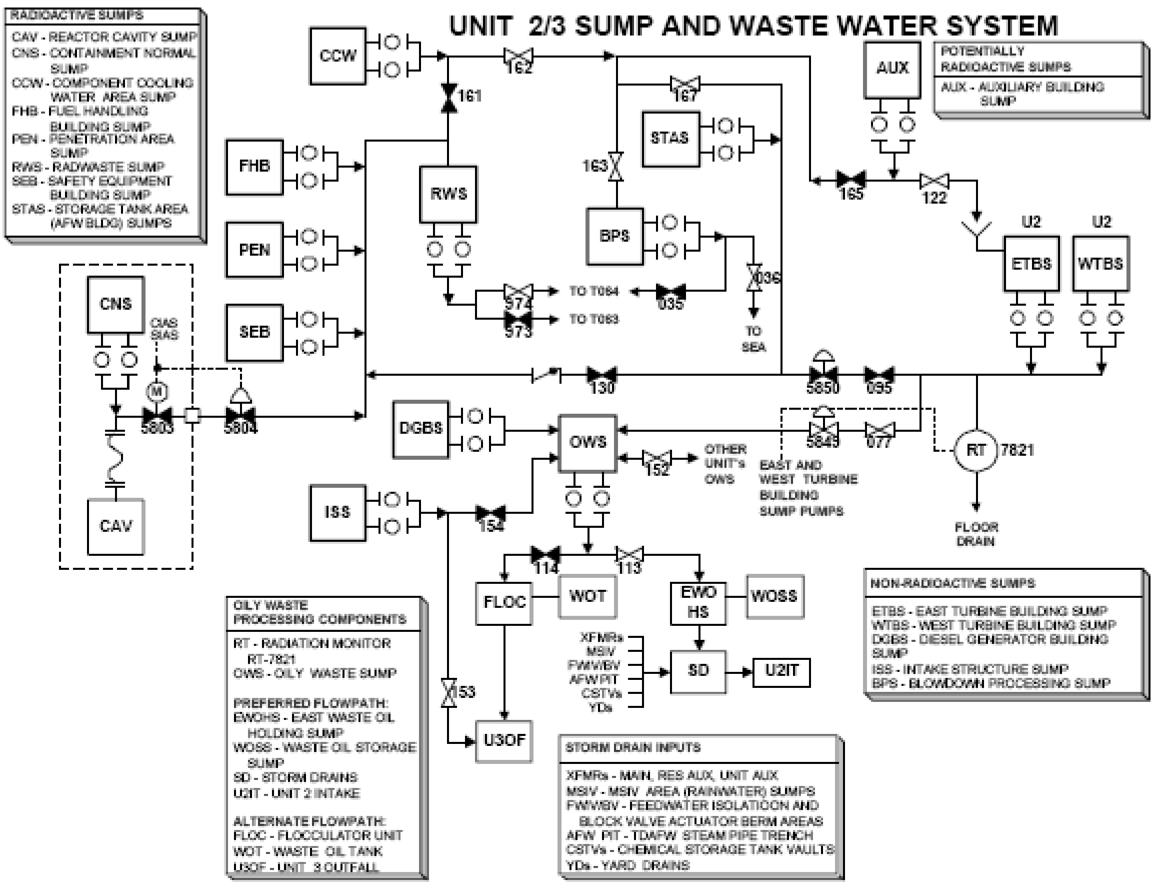
NOTE:
PENETRATIONS NOS. 10, 16, 23, 27, 30, 40, & 73 ARE MULTIPLE PENETRATIONS

20. From SD-SO23-780, Figure 1

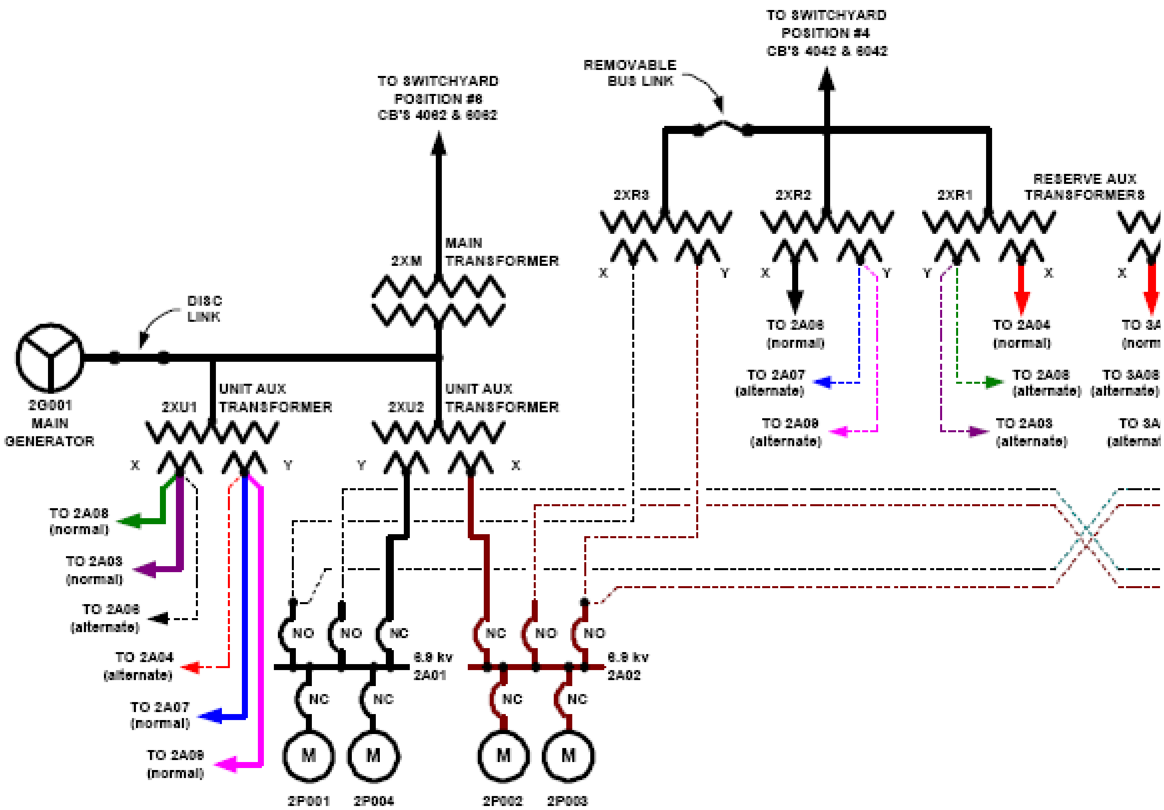


20. From SD-SO23-670, Figure 2

FIGURE 2: U2 & 3 RADIATION & NON-RADIATION SUMPS



21. From SD-SO23-120, Figures 1



21. From SD-SO23-120, page 26

4. AUTOMATIC TRANSFER FROM THE UNIT AUXILIARY TRANSFORMER TO THE SAME UNIT'S RESERVE AUXILIARY TRANSFORMER (Figure I-3)

4.1 Permissives that must be satisfied before an automatic transfer can take place are listed as follows:

- The "Auto" pushbutton for the Reserve Auxiliary Transformer supply breaker must be depressed.
- Normal Voltage at the Reserve Auxiliary Transformer secondary windings must be available. This prevents closing the bus onto a dead transformer.
- The Unit's Reserve Auxiliary Transformer is not supplying power to the same (companion) 6.9 kV bus on the other Unit. This is monitored by either: contact 252b (indicating that the companion bus crosstie breaker is open); contact 233 (indicating that the companion bus crosstie breaker is either racked out or in the test position). The interlock prevents overloading the Reserve Auxiliary Transformer.
- The Unit Auxiliary Transformer supply breaker lockout relay must be reset (prevents closing onto a faulted bus).
- The 6.9 kV crosstie supply breaker lockout relays are reset (prevents closing onto a faulted bus).
- The Reserve Auxiliary Transformer supply breaker lockout relays are reset.
- Reserve Auxiliary Transformer trips or 22O kV Switchyard Circuit Breaker Failure Backup protection trips (at the position which supplies the respective Reserve Auxiliary Transformer) are not present. This interlock prevents closing the Reserve Auxiliary Transformer supply breaker, if the Reserve Auxiliary Transformer is required to be isolated by electrical protection devices.

22. From Tech Spec 3.8.4

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources — Operating

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

APPLICABILITY: MODES 1, 2, 3, and 4.

22. From Tech Spec 3.8.5

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources — Shutdown

LCO 3.8.5 DC electrical power subsystem shall be OPERABLE to support the DC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems — Shutdown."

APPLICABILITY: MODES 5 and 6,

During movement of irradiated fuel assemblies.

22. From Tech Spec 3.8.10

3.8 ELECTRICAL POWER SYSTEMS

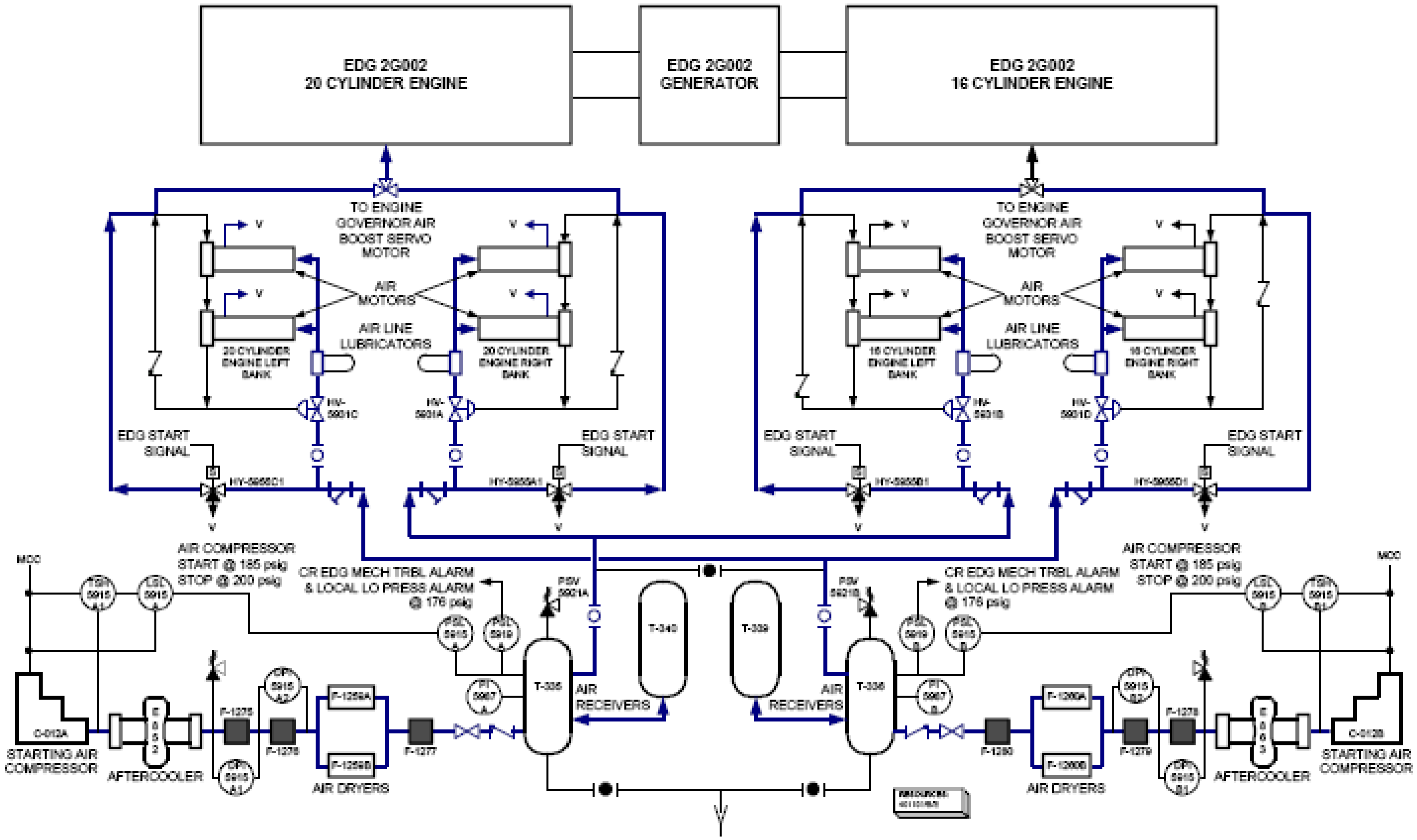
3.8.10 Distribution Systems — Shutdown

LCO 3.8.10 The necessary portion of AC, DC, and AC vital bus electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

23. From SD-SO23-750, Figure V-1

FIGURE V-1: EDG 2G002 STARTING AIR SYSTEM



23. From SD-SO23-750, page 108 (refers to “set” of air start motors)

2.1.2 **General Control Scheme**

.1 When the Diesel Generator Lockout Relay is RESET, starting of the Diesel can be initiated by a local start signal, Control Room start signal, Loss of Voltage Signal or Safety Injection Actuation Signal. Each Diesel Air Start System is rated at 100% capacity, meaning either set is capable of starting the Diesel such that it is at rated speed and voltage within 9.4 seconds upon receipt of a start signal.

24. From SD-SO23-690, page 9

.6.4 Plant Vent Stack/Containment Purge Wide Range Radiation

.6.4.1 2(3)RE-7865A1, B1, C1 is switchable between the Plant Vent Stack and Containment Purge Stack.

.6.4.2 If aligned to the Plant Vent Stack it generates an alarm and initiates closure of the waste gas discharge header flow control valve on high radiation levels or loss of power.

.6.4.3 If aligned to the Containment Purge Stack generates an alarm and initiates closure of the outside containment purge valves on high radiation levels, instrument failure or loss of power.

24. From SO23-8-15, L & S 4.5

4.5 High radiation signal from 2/3RE-7808G or 3RE-7865-1 will cause automatic closure of 2/3FV-7202 **and** termination of the Waste Gas Release.

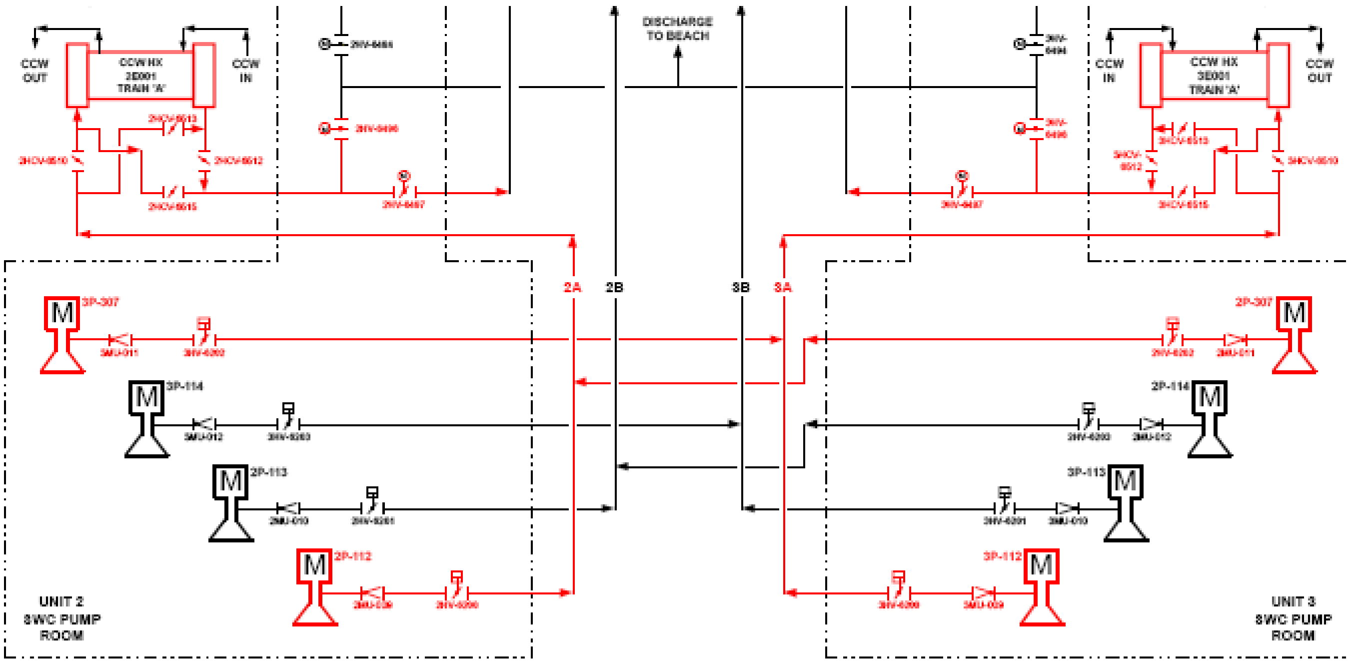
4.5.1 3RE-7865-1 *High* Alarm will actuate an alarm in the State Offices of Emergency Services. However, the release should terminate when the *Alert* setpoint is reached.

25. From SD-SO23-410, page 18

Component Cooling Water Heat Exchanger Saltwater Outlet Valves:		
HV-6496	2BK 23 (3BK 22)	
HV-6497	BY 35	
HV-6494		2BK 27 (3BK 18)
HV-6495		BZ 31

Each train has separate power supplies. Loss of one 4160 VAC Bus makes one train inoperable and does not affect the status of the other train.

26. From SD-SO23-410, Figure 1



26. From SO23-13-7, Step 14

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
14 **Loss of a Single SWC Pump**	
☐ a. ENSURE CCW/SWC on the unaffected loop - IN SERVICE.	☐ a. INITIATE PLACING the standby SWC Pump for the affected loop - IN SERVICE.
☐ 1) TRANSFER Noncritical loop to the unaffected loop.	1) MAINTAIN RUNNING affected Train CCW Pump during the SWC Pump transfer.
☐ 2) TRANSFER the Letdown HX to the unaffected loop.	2) SECURE unnecessary loads on the affected Train.

26. From SO23-15-64A55, page 133

64A55 SWC TRAIN A FLOW TROUBLE

APPLICABILITY	PRIORITY	REFLASH	ASSOCIATED WINDOWS
Modes ALL	WHITE	NO	NONE

INITIATING DEVICE	NOUN NAME	SETPOINT	VALIDATION INSTRUMENT	PMS ID	LINK # U2/U3
2(3)PSL-6476	Saltwater from 2(3)ME-001 Pressure Switch Low	7 psig	2(3)FI-6398	NONE	1375/1375
2(3)PDSHL-6534	CCW Heat Exchanger 2(3)ME-001 ΔP SW LO	2 psid			1376/1376

1.0 **<u>REQUIRED ACTIONS</u>**:

1.1 <u>If</u> Loss of Saltwater is suspected, <u>then</u> GO TO SO23-13-7, Loss of Component Cooling Water (CCW)/Saltwater Cooling (SWC).

2.0 <u>CORRECTIVE ACTIONS</u>:

SPECIFIC CAUSES	SPECIFIC CORRECTIVE ACTIONS
2.1 Marine fouling of the Saltwater side of the CCW Heat Exchanger	2.1 <u>If</u> a high ΔP Saltwater side exists <u>or</u> Saltwater flow is out of the acceptable range of SO23-2-8, Attachment for Saltwater Injection Temperature vs. Minimum Saltwater Flow, <u>then</u> perform SO23-2-8.1, Section for Infrequent and/or Abnormal Operation of SWC System.

26. From SO23-15-64A35, page 89

64A35 CCW HX TRAIN A DIFF PRESS HI

APPLICABILITY	PRIORITY	REFLASH	ASSOCIATED WINDOWS
Modes ALL	WHITE	N/A	NONE

INITIATING DEVICE	NOUN NAME	SETPOINT	VALIDATION INSTRUMENT	PMS ID	LINK # U2/U3
2(3)PDSHL-6534	CCW Heat Exch. 2(3)ME-001 ΔP SW HI	12 psid (±0.2 psid)	2(3)PDIT-6484 2(3)PI-6474 2(3)PI-6204	NONE	1351/1351

1.0 **<u>REQUIRED ACTIONS</u>:**

1.1 <u>If</u> the high differential pressure is unexplained <u>or</u> the loss of the heat exchanger is imminent, <u>then</u> GO TO SO23-13-7, Loss of Component Cooling Water (CCW)/Saltwater Cooling (SWC).

27. From SO23-1-1, Attachment 2

AFFECTED UNIT EQUIPMENT RESPONSE

INFORMATION USE

NOTE
Loss of Instrument Air will cause all air operated CIVs to Fail-Closed ***except*** for the Steam Supply Valves to MP-140 (Steam Driven Aux Feed Pump) and Charging to the Regenerative HX, which Fail-Open.

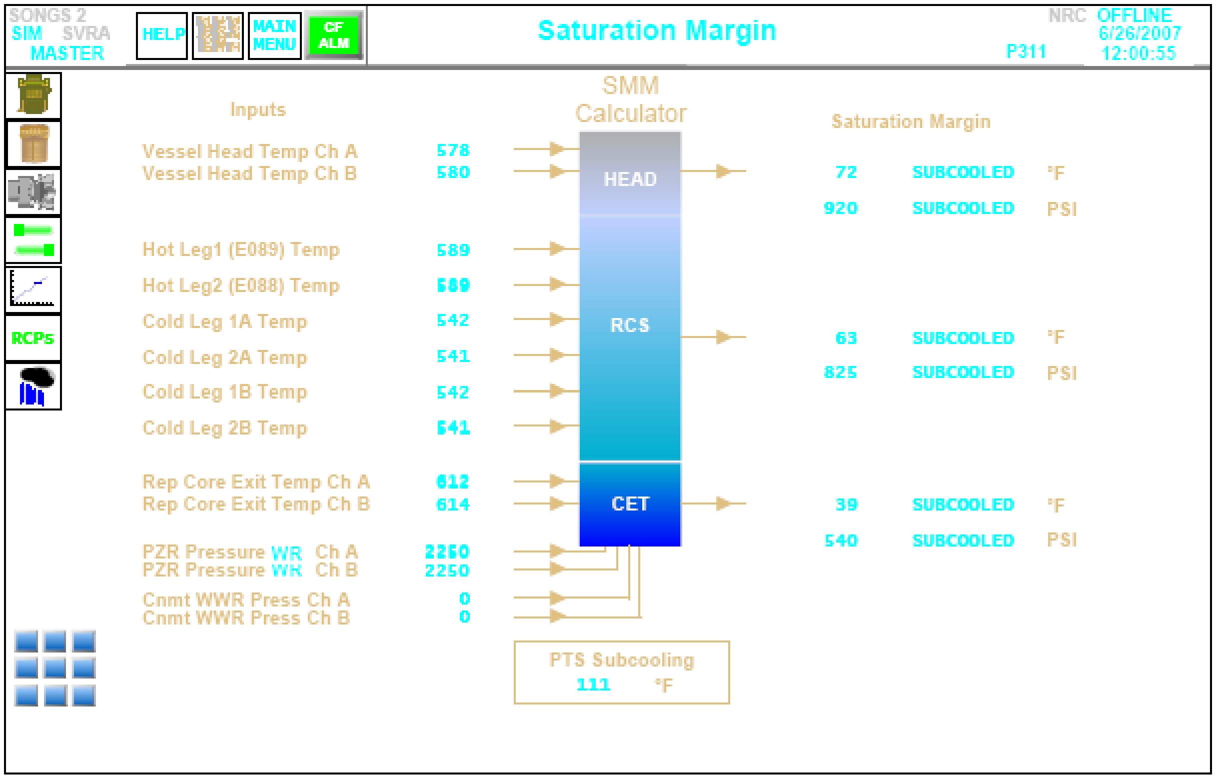
AFFECTED HEADER	IMPACTED EQUIPMENT TO ISOLATED HEADER
Containment (Excess flow check valve HV-5343 closes at 200 scfm.) Isolation: HV-5388 (IA to Containment Isolation; Pen #22)	In Modes 1-4: • TV-0221 & HV-9204, Letdown Isolation Valves Fail-Closed • SIT Fill & Drain Valves Fail-Closed • PZR Spray Valves Fail-Closed • SIT Fill Line Isolation Valves Fail-Closed • Hot Leg Injection Drain Valves Fail-Closed In Modes 5-6: • Reactor Vessel Pool Seal Ring System will rely on Backup Nitrogen • Steam Generator Nozzle Dams will rely on Backup Nitrogen
Saltwater Pump Room Isolation: • SA2417MU109 (20' Unit 2 in 7' Turbine Building over BK) • SA2417MU112 (20' Unit 3 in 7' Turbine Building over BK)	• Saltwater Pump Discharge Valves Fail-Open (If the valves are not manually failed open per SO23-2-8.1, Attachment for Failing and Return to Service of a SWC Pump Discharge Valve, then loss of the Back Up Accumulator pressure will cause the valves to drift.) • Saltwater Pump Seal Water Supply Valves Fail-Open • Flush Water Supply Valves to the Rakes & Screens Fail-Closed

Turbine Building
 	
Isolation:
• S22417MU028 (Unit 2, 7' Turbine by Instrument Air Filters)
• S32417MU028 (Unit 3, 30' Turbine behind 3BRC)
	• ADVs lose air, will rely on Backup Nitrogen
	• HV-8200 and HV-8201, P-140 Main Steam ISO VLVs, Fail-Open
	• Steam Bypass Valves Fail-Closed
	• HV-1105 and HV-1106, Feedwater bypass Valves, Fail-Closed.
	• FV-1111 and FV-1121, Feedwater Regulating Valves, Fail-As-Is.
	• Condenser Vacuum breakers Fail-Closed
	• CREACUS dampers Fail-Isolate Mode
	• Turbine Plant Cooler/Heat Exchanger Outlet TPCW Cooling Control Valves Fail-Open
	• Normal and High Level Feedwater Shell side Control Valves Fail-Open
	• FFCPD Full [2(3)HV-4902A], and Partial [2(3)HV-4902B], Bypasses Fail-Open
	• **On Unit 3 FFCPD only,** (On Unit 3 FFCPD Air Isolation S32417MU135 is downstream of S32417MU028)
	  • FFCPD Service Vessel Valves (CPs and MBPs main flowpath valves) Fail as-is
	  • Regeneration Tank Valves will Drift Open (any in-progress regeneration and transfer operations should be stopped)

28. From SO23-3-2.22, pages 71 and 72

2.2.26	HV-9823	Containment Mini Purge Supply Isolation		CLOSED	________
2.3.14	HV-5437	Nitrogen to Containment Isolation	[1][2]	CLOSED	________
2.3.3	HV-6211	CCW NCL Containment Supply Isolation	[1]	CLOSED	________

29. From Saturation Margin Computer Printout



29. From SD-SO23-820, page 84

.8 An RIME dataviewer generated SCM display is provided on control board 2(3)CR056 located above the Channel D instruments. This monitor provides the operator with information on the subcooled margin, cold leg temperature and pressurizer pressure in large clearly visible numbers as shown in Figure I-13. The unit operator's CFMS terminal user dialogue page 130 is used for controlling the SCM display as shown in Figure I-14. The following options are available to the operator:

.8.1 Channel selection allows the operator to select channel A or B, or combined channels of the two channels of QSPDS.

.8.2 SCM selection allows the operator to select CET, Head, RCS or worst case subcooled margins from the selected channel(s) of QSPDS.

.9 The first line on the SCM monitor displays the subcooled margin as selected by the option keyed in by the operator on the CFMS terminal. The SCM display provides the operator with the subcooled or superheated margin value, the channel and also the region (i.e. CET or Head or RCS or worst case).

29. From SO23-15-56.B-45

56B45 RCS SUBCOOLED MARGIN LO

APPLICABILITY	PRIORITY	REFLASH	ASSOCIATED WINDOWS
Modes 1-4	RED	NO	NONE

INITIATING DEVICE	NOUN NAME	SETPOINT	VALIDATION INSTRUMENT	PMS ID	LINK # U2/U3
N/A	QSPDS (Page 611)	Saturation Margin Less Than 20°F	CFMS (Page 311)	NONE	489/516

1.0 **REQUIRED ACTIONS**:

1.1 If the reactor has tripped, then GO TO SO23-12-1, Standard Post Trip Actions.

1.2 If the reactor has Not tripped, and both channels indicate <20°F Saturation Margin, then Evaluate for localized DNB. Consider reducing Reactor Power per SO23-5-1.7, to restore Margin.

4.0 COMPENSATORY ACTIONS:

DEVICE NUMBER	SPECIFIC COMPENSATORY ACTIONS
4.1 CFMS In Service	4.1 Monitor Operable QSPDS Channel Subcooled Margin twice per shift by Selecting to Subcooled Margin Monitor Display per SO23-3-2.32, Section for Operation of the Subcooled Margin Monitor Display.
4.2 CFMS OOS	4.2 Monitor Subcooled Margin twice per shift on Operable QSPDS Channel.

29. From SO23-3-2.32, Attachment 1

2.0 PROCEDURE (continued)

2.16 Subcooled Margin Monitor (SCM)

NOTES

1. The preferred method for programming the Subcooled Margin Monitor (SCM) is to use the highest TCold and the lowest Pressurizer Pressure to calculate and display Reactor Vessel Head Saturation margin in degrees Fahrenheit (°F) subcooled or superheated (as applicable).
2. When Pressurizer Pressure decreases to ≤720 psia, then the SCMM is from Low Range Transmitters, 2(3)PT-0103-1 and/or 2(3)PT-0104-2. When Pressurizer Pressure increases to ≥745 psia, then the SCMM is from High Range Transmitters 2(3)PT-0102-1 and/or 2(3)PT-0102-2.
3. CFMS page 311 will indicate the worst case temperature Subcooled Margin of the two channels supplied from the QSPDS.
4. When Containment Wide Range Pressure is ≥5 psig, then the Subcooled Margin Monitor is compensated for Containment Pressure.

2.16.1 Access the SCM as follows:

.1 From the Main Menu, select REMOTE DISPLAYS, then select SCM.

.2 From the SCM page, select a channel.

2.16.2 To program the display of the Subcooled Margin Monitor on CR-56 perform the following:

.1 SELECT one of the following QSPDS Pressurizer Pressure instruments:

- Channel A (QSPDS A Composed Pressurizer Pressure; KPZRA)
- Channel B (QSPDS B Composed Pressurizer Pressure; KPZRB)
- Combined Channels (Average Composed Pressurizer Pressure)

.2 SELECT one of the following QSPDS Temperature instruments:

- Head SCM
- RCS SCM
- CET SCM
- Worst Case SCM

2.16.3 Select SEND.

29. From SD-SO23-360, page 45

2.1.2 **Additional Flow Paths** (Figure II-1)

Bypass flow passes ~2.6% of total reactor coolant flow through various portions of the internals package. The bypass flow does not contribute to cooling the fuel assemblies and is restricted to less than a design maximum of 3.0% of total flow to maximize heat removal from the fuel. However, bypass flow does cool internal components heated by gamma absorption reactions and prevents chemical stratification. Total bypass flow measured in percent of total reactor coolant flow is comprised of the following:

.1 0.6% through outlet nozzle clearance.

.2 0.1% through the alignment keyways of the internals package and reactor vessel.

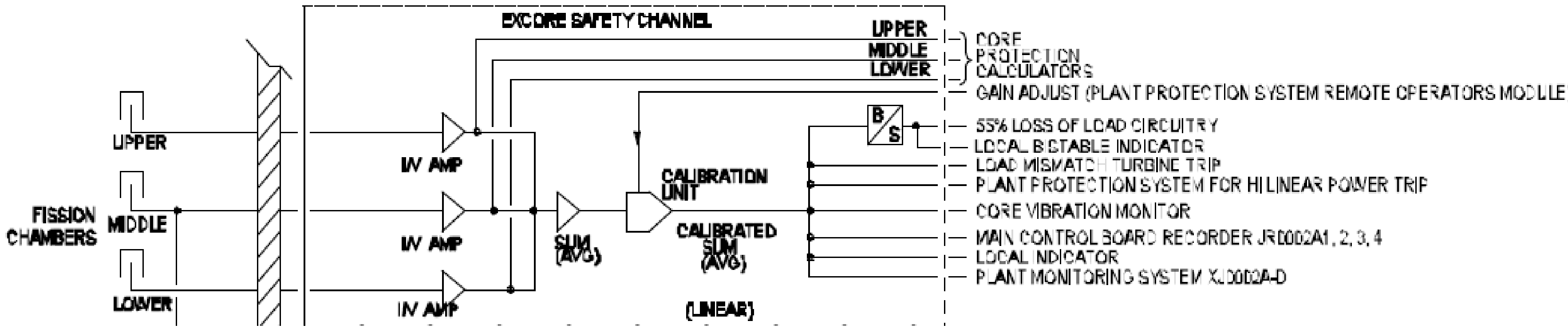
.3 0.3% through support cylinder holes.

.4 0.3% through the core shroud clearances.

.5 1.3% through empty guide tubes, rodded guide tubes, and instrumented center guide tubes.

30. From SD-SO23-470, Figure 1

FIGURE 1: EXCORE NUCLEAR INSTRUMENTATION



30. From SD-SO23-470, page 14

.6 Controls and Interlocks from the Linear Channel are as follows:

The **55% Loss of Load Bistable** is provided to trip the reactor if the turbine trips and reactor power exceeds the capacity of the Steam Bypass Control System.

30. From SD-SO23-710, page 19

2.1 System Overview (Continued)

.12 **LOSS OF LOAD TRIP**

PURPOSE:	To trip the reactor when the Turbine is tripped.
INPUTS:	Turbine trip signal.
SETPOINTS:	Turbine HP Stop Valves ≤ 100 psig on Unitized Actuators. Setpoints are Tech. Spec. limits. For actual trip setpoints, contact Instrument and Control.

30. From SD-SO23-180, page 68

.20 **TURBINE TRIP - REACTOR TRIP INTERLOCK**

.20.1 The Turbine Protection System is interlocked with the Reactor Protection System so that a Turbine Trip INITIATES a Reactor Trip if reactor power is greater 55%.

.20.1.1 A Turbine Trip CLOSES the Loss-of-load contact in the Turbine Protection Cubicle causing a Turbine Trip Signal to be sent to the Reactor Protection System's Loss-of-load Bistable Relay.

31. From SD-SO23-820, page 87

2.3.7 Core Exit Thermocouple Temperature

.1 The QSPDS displays individual core exit temperatures with a core map, highest and next highest core exit temperature in each quadrant and a representative core exit temperature.

.2 The Representative Core Exit Thermocouple (REP CET) temperature is calculated as follows based on the statistical analysis with practical checks from other inputs.

.2.1 First, the out-of-range CET inputs are flagged and discarded. Second, the mean CET temperature is calculated from the remaining CET inputs. Then, CET inputs are checked with statistical bands (standard deviation) about the mean CET temperature.

.2.2 Those falling outside the bands are flagged as suspicious inputs and discarded from the calculation. The mean CET temperature is recalculated from the remaining CET inputs. This flagging process goes on until no more CETs are flagged. Then, the CET inputs are considered stable and the representative is calculated.

32. From SO23-13-19, Step 1

OPERATOR ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1 Determine Non-1E Instrument Bus Power Loss:	
a. VERIFY Instrument Bus #1 (Q065) ENERGIZED: ☐ Annunciator 63B24 - Q065 INST BUS 1 POWER SUPPLY FAILURE - NOT alarming.	☐ a. 1) SELECT 2(3)VS65, 2(3)Q065 Instrument Bus #1 Transfer Switch, to EMERGENCY. [Room 307A(B)] ☐ 2) GO TO Step 2
b. VERIFY Instrument Bus #2 (Q0612) ENERGIZED: ☐ Annunciator 63B34 - Q0612 INST BUS 2 POWER SUPPLY FAILURE - NOT alarming.	☐ b. 1) PLACE SBCS in Manual. 2) SELECT 2(3)VS612, 2(3)Q0612 Instrument Bus #2 Instrument Bus Transfer Switch, to EMERGENCY. [Room 307A(B)] ☐ 2) GO TO Step 2

32. From SO23-3-2.18, Step 14.6

14.6 Returning SBCS to Service After a Loss of Power

REFERENCE USE

14.6.1 ENSURE the Master Controller is placed in MANUAL. (LS-3.4)

14.6.2 If SBCS restarts automatically (52A09, SBCS TROUBLE, clears within 2 minutes of power restoration), then perform one the following:

.1 If SBCS System exhibits normal system characteristics, then automatic control may be resumed by placing the MASTER CONTROLLER to AUTO.

.2 If SBCS System is acting erratically, then Ensure SBCS Master Controller remains in MANUAL, and Request I&C to investigate.

32. From SO23-3-2.18, L & S 3.4

3.0 **SBCS MANUAL OPERATIONS**

3.1 To avoid defeating the single-failure design of the SBCS, operations that include the Master Controller in REMOTE and Valve Permissive(s) in MANUAL should be avoided.

3.2 Operations that include Master Controller in LOCAL require that Valve Permissive(s) be placed in MANUAL.

3.3 The SBCS Master Controller, PIC-8431, should be placed in MANUAL prior to changing the Remote/Local setpoint selector switch.

3.4 Placing the Master Controller in MANUAL after a loss of power to the SBCS will eliminate system transients when power resumes and SBCS automatically restarts.

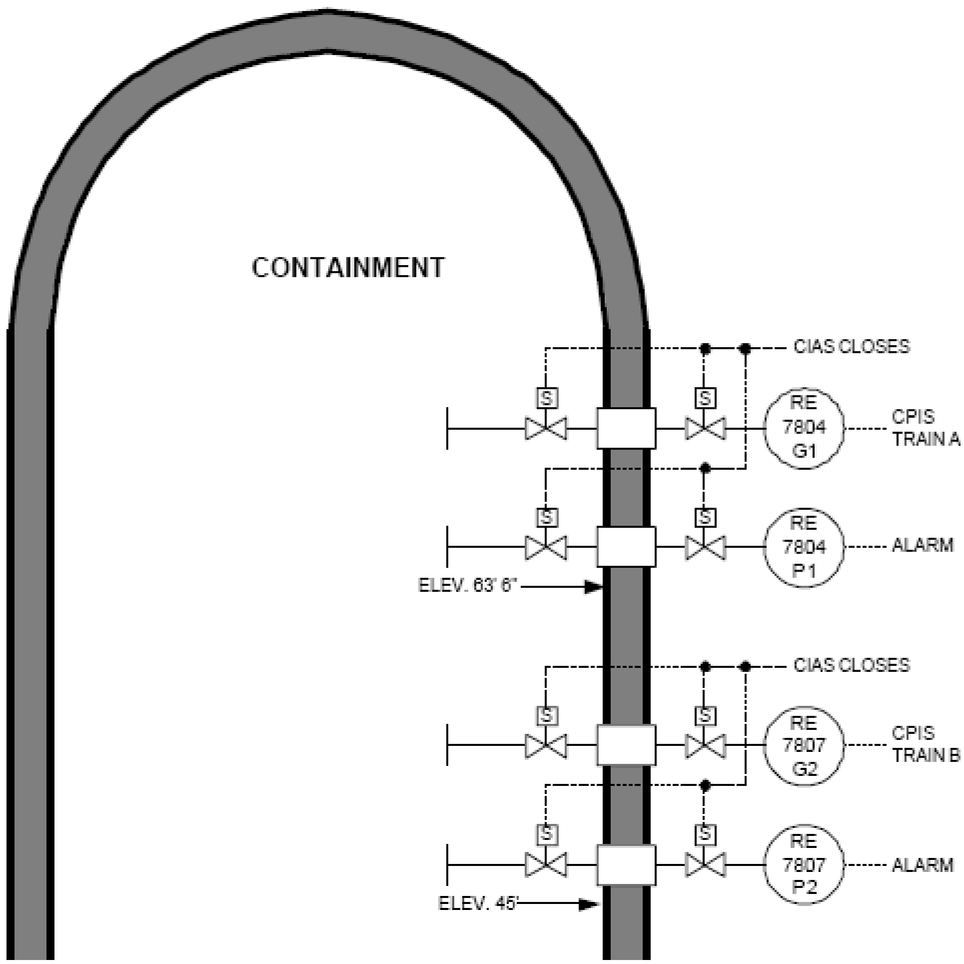
32. From SD-SO23-175, page 14

.5 Steam Bypass Demand

The SBCS is designed for both normal operations, such as Turbine synchronization, and off-normal events up to and including a 55% load rejection. Matching secondary load to reactor power, particularly due to large mismatches, is less stressful on the nuclear fuel and less likely to result in a reactor trip. This ability of the SBCS to quickly match reactor power, prevents the unwanted, large primary power swing that might otherwise occur after a main turbine load rejection. bypass demand which provides modulation or quick opening of SBCS valves is generated by the combined demand signals from the Master Controller and the Permissive Controller.

33. From SD-SO23-770, Figure 1

FIGURE 1: CONTAINMENT TRAIN A(B) AIRBORNE RADIATION MONITORS
2(3)RE-7804G1,P1 & 2(3)RE-7807G2,P2



33. From SD-SO23-770, page 8

.4.3 Containment Train A(B) Airborne Radiation

.4.3.1 2(3)RE-7804G1, P1 (Train "A") and 2(3)RE-7807G2,P2 (Train "B") generate alarms and initiate a Containment Purge Isolation Signal (CPIS) on high radiation levels or loss of power.

33. From SD-SO23-770, page 44

2.2.21 Normal Purge Inlet Isolation Valves, 2(3)HV-9948 and 9949 (Figure 5)

NORMAL PURGE INLET ISOLATION VALVE, 2(3)HV-9948

SIZE:	42"
OPERATOR:	Air operated
TYPE:	Butterfly
FAIL POSITION:	CLOSED
INTERLOCKS:	CPIS or High Radiation from RE-7828 (RE-7865-1) CLOSES

NORMAL PURGE INLET ISOLATION VALVE, 2(3)HV-9949

SIZE:	42"
OPERATOR:	3⌀, 480 VAC, 3.2 hp motor
TYPE:	Butterfly
FAIL POSITION:	AS IS
INTERLOCKS:	CPIS CLOSES

.1 The Normal Purge Supply Unit is provided with two inlet isolation valves to isolate the Containment in accident conditions.

.1.1 These valves must be open to allow the Normal Purge Supply Unit to operate. These valves are normally sealed shut (blind flange installed) during power operations.

.1.2 A CPIS CLOSES both valves, high radiation from 2(3)RE-7828 (or 2(3)RE-7865-1 when aligned to Purge Stack) closes the valve outside Containment. Closure of either valve TRIPS the Normal Purge Supply Unit to secure purging.

34. From SD-SO23-250, page 86

.1 For an up-power transient, assume an initial power level of 50% with S/G Downcomer Level at 68% and Steam Flow equal to Feedwater flow.

.1.1 A 10% step increase in Turbine load occurs with the following two major effects.

.1.2 1st an initial increase in steam demand results in a S/G Downcomer Level increase due to swell.

.1.2.1 This higher level tells the Feedwater Control System to decrease Feedwater flow.

35. From SD-SO23-180, page 59

2.2 **General Control Scheme** (Continued)

2.2.4 **TURBINE PROTECTION SYSTEM** (Figures II-2A & 2B) (Continued)

.6 **Turbine Trip and Alarm Setpoints**

Turbine Trip Setpoints		
Description	**Norm Operation**	**Trip Setpoint**
Electronic Overspeed Trip	1800 rpm	≥1926 rpm

35. From SO23-15-99A, 99A35

99A35 OVERSPEED TURBINE TRIP

APPLICABILITY	PRIORITY	REFLASH	ASSOCIATED WINDOWS
Modes 1,2	RED	YES	NONE

INITIATING DEVICE	NOUN NAME	SETPOINT	VALIDATION INSTRUMENT	PMS ID	LINK # U2/U3
OA OVERSPD ETS OVERSPD	Turbine DCS OA Overspeed Trip ETS Overspeed Trip	1926	NONE	SY8219	1117/1117 1118/1118 1119/1119

35. From SO23-10-4, L & S 1.1

TURBINE PROTECTIVE DEVICE TESTING LIMITATIONS AND SPECIFICATIONS

OBJECTIVE: To provide a list of system/component limitations and specific operational details related to the steps in this procedure. Although the information presented here is not necessary to perform an evolution, it does provide supplementary information to enhance understanding and increase awareness. Some of this information may also be considered for Pre-job Brief subjects. Appropriate steps in this procedure will reference this attachment, for example ***(LS-2.2)*** for *Limitations and Specifics Item No. 2.2.*

Verify this document is current by checking a controlled copy or by using the method described in SO123-VI-0.9.

1.0 GENERAL

1.1 LIMIT: Turbine Speed of 2034 rpm shall not be exceeded. If 2034 rpm is approached, then the Turbine should be manually tripped.

36. From SO23-8-7, L&S 2.5

2.5 If a release automatically terminates on high radiation, then the release Attachment should be completed, and a new Attachment initiated to finish releasing the tank.

36. From SO23-8-7, Attachment 1

NUCLEAR ORGANIZATION UNITS 2 AND 3	OPERATING INSTRUCTION REVISION 14 TCN 14-1 ATTACHMENT 1	SO23-8-7 PAGE 4 OF 129

PLACING RADWASTE SECONDARY TANK T-057 ON ONE-VOLUME RECIRC

CONTINUOUS USE

OBJECTIVE

Place Radwaste Secondary Tank T-057 on one volume recirc using MP-170. Request a release permit from Chemistry. Determine operability of 2/3FI-7643, 2/3RT-7813, DAS for alarm monitoring, and the Unit to be used for the release. Ensure Chemistry meets the NPDES requirement for monitoring oil, grease, and TSS for the release flowpath. Determine which Attachment to use for the release.

37. From SD-SO23-690, page 19

.5 Gaseous In-duct Sampling

.5.1 The in-duct monitors consist of two "flying wings" located in the control room intake duct. Each wing houses a single channel gamma sensitive gas monitor. The wings comprise A and B train control room isolation. Both trains are located on the normal HVAC system to the control room.

.5.2 The single channel in-duct monitors are:

.5.2.1 2/3RE-7824G1 and 2/3RE-7825G2 Control Room Intake Air Train A(B) Radiation.

37. From SD-SO23-690, page 82

Control Room Emergency Vent Supply Units, 2/3A-206/207 and Inlet Dampers, 2/3HV-9761/9742	Train "A" 30' CB Room 219 Train "B" 30' CB Room 236	MCR 2/3CR-60	MCR U2 VCP 30' CB Rooms 219/236	To provide outside air (6%) to the Emergency Air Conditioning Unit during CRIS conditions.	Vertical draw-through fan, 2050 cfm Prefilter - eff. 55% HEPA Filter - eff. 99.95% Charcoal filter - eff. 95% Charcoal Beds - protected by a MANUAL fire water valve. Electro-hydraulic operated Inlet Damper, fail "AS-IS" **NORMALLY: Fan - OFF; Damper - CLOSED** **CRIS: FAN - ON; DAMPER - OPEN** **TGIS: FAN - OFF; DAMPER CLOSED** **CRIS: 2050 cfm; filter ?P's - various**	MANUAL START CRIS/TGIS Train "A"/"B" (AUTO or MANUAL Actuation) 2/3HV-9742/9761 AUTO MODULATE when 2/3E-418/ 419 receives START Signal. Fire Isolation Switch on Panel 2/3L-412, with 2nd-Point-of-Control Switch on 2(3)BQ-09 (2(3)BS-09):

37. From SD-SO23-624, page 86

COMPONENT	LOCATION			FUNCTIONS	DESIGN DATA	INTERLOCKS	PROCEDURE/ P&ID/ELEM
	COMPONENT	CONTROL	INSTRUMENT		CAPACITY/TEMP/PRESS		
Control Room Isolation Signal (CRIS) Actuation Radiation Monitors, 2/3RT-7824/7825	Train "A" 30' CB Room 230 Train "B" 9' CB Room 107	30' CB Room 228 2/3L-104 2/3CR-60	30' CB U2 hall	To isolate the outside air supply, start the emergency filtration unit <u>and</u> initiate the Emergency Control Room HVAC System upon detection of a radiation hazard.	Two independent sensors designed to detect Gaseous and Particulate/Iodine radioactivity. 2/3RT-7824 is Train "A" 2/3RT-7825 is Train "B" **NORMALLY: both Radiation Monitors are in service continuously**	Actuation of <u>either</u> radmonitor subsystem will cause its Train of CRIS to actuate. Components actuated during a Train "A"/"B" CRIS: * Emergency Ventilation Supply Fans, 2/3A-206/207 - START * Emergency A/C Units, 2/3E-418/419 - START * Cabinet Area Emergency A/C Units, 2E-423/2E-424/3E-426/ 3E-427 - START * Emergency Chilled Water Pump, 2/3P-160/162 - START * Emergency Chiller Unit, 2/3E-335/336 - START * Emergency Inlet and Isolation Dampers, 2/3HV9732/9733/ 9738/9739/9742/9761/9778 - OPEN * Computer Room 2(3)HV-9734 Dampers, require MANUAL OPENING for a Train "A" **only** CRIS. Computer Room 2(3)HV-9715 Dampers, require MANUAL CLOSING for a Train "A" **only** CRIS. * Normal Ventilation Dampers, 2/3HV-9702/9703/9711/ 9712/9717/9756/9757/9758 / 9769/9779 - CLOSE * Normal A/C Unit, 2/3E-295 - TRIP OFF * Normal Exhaust Fans, 2/3A-201/202 - TRIP OFF * Smoke Exhaust System TRIPS OFF	SO23-15-60.B 40173A 31394/31395

38. From SO23-1-1, Attachment 22

1.6 Instrument Air System Response to Lowering Air Pressure:

INSTRUMENT AIR RECEIVER/AIR DRYER INLET PRESSURE	
PRESSURE	**ACTION**
≤ 106 psig	Lead Compressor will Start, after 15 seconds, the Lead Compressor will 50% Load.
≤ 102 psig	Lead Compressor will 100% Load.
≤ 98 psig	LAG 1 Compressor will Start, after 15 seconds, the LAG 1 Compressor will 50% Load.
≤ 94 psig	LAG 1 Compressor will 100% Load.
≤ 90 psig	LAG 2 Compressor will Start, after 15 seconds, the LAG 2 Compressor will 50% Load.
=88 psig[1]	PCV-5354, RSAS Backup to Instrument Air will Open to maintain Instrument Air System pressure >84 psig.
≤ 86 psig	LAG 2 Compressor will 100% Load. A leak downstream of the air dryers will not cause the LAG 2 compressor to start due to the high pressure drop across the dryer at high flows.
=83 psig	PCV-5448, N2 Backup to Instrument Air will Open to maintain Instrument Air System pressure > 70 psig <u>and</u> ANN 61B38, N2 SUPPLY TO INST AIR HEADER ON will annunciate. (UFSAR 9.3.1.2.3)
[1] This is the normal setpoint for PCV-5354, Setpoint may be changed per Section 6.12 (main body).	

39. From SO23-6-33, Step 6.4

6.4 Inverter DC Side Grounds (LS-1.2)

REFERENCE USE

6.4.1 Remove the associated 120 VAC 1E Inverter from service per SO23-6-17, Attachment for Removing an Inverter from Service, or the PMS Inverter per SO23-6-17.2, Section for Removing Plant Computer Inverter Y005 from Service.

6.4.2 CHECK the affected bus 125 VDC BUS TROUBLE (CR 63A) Control Room alarm reset.

6.4.3 If the ground is still present, then return the 120 VAC 1E Inverter per SO23-6-17, Attachment for Returning an Inverter to Service, or the PMS Inverter per SO23-6-17.2, Section for Returning Plant Computer Inverter Y005 to Service.

39. From SO23-6-33, L & S 1.2

1.2 To prevent inverter damage, a battery charger and inverter should not be in service with the battery breaker open. The battery charger is unable to regulate and filter DC to the inverter without the capacitance effect of the battery.

39. From SO23-15-63.A, Alarm 63A43

63A43 2D2 BATTERY BKR OPEN

APPLICABILITY	PRIORITY	REFLASH	ASSOCIATED WINDOWS
Modes ALL	AMBER	N/A	63A33

INITIATING DEVICE	NOUN NAME	SETPOINT	VALIDATION INSTRUMENT	PMS ID	LINK #
2D201, 52a contact	2D2, Battery Breaker Open	N/A	LOCAL	NONE	1932

1.0 **REQUIRED ACTIONS**:

1.1 Dispatch an Operator to the 50' Control Building.

1.1.1 Monitor 2D2 bus voltage, battery ground alarms and perform local inspection of batteries.

2.0 CORRECTIVE ACTIONS:

SPECIFIC CAUSES	SPECIFIC CORRECTIVE ACTIONS
2.1 2D2 Battery Breaker, Overcurrent Trip	2.1 Contact Electrical Test Maintenance to assist in identifying and correcting the cause of the overcurrent. 2.1.1 After the fault or overcurrent condition has been corrected, then RECLOSE the battery breaker.

3.0 ASSOCIATED RESPONSES:

CAUTION

To prevent Inverter damage, a battery charger and inverter should not be in service with the battery breaker open. The battery charger is unable to regulate and filter DC to the inverter without the capacitance effect of the battery.

3.1 If the battery breaker can not be Reclosed, then remove the affected inverter from service per SO23-6-17, Section for Removing an Inverter from Service.

39. From SO23-15-63.A, Alarm 63A33

63A33 2D2 125 VDC BUS TROUBLE

APPLICABILITY	PRIORITY	REFLASH	ASSOCIATED WINDOWS
Modes ALL	AMBER	YES	63A53

INITIATING DEVICE	NOUN NAME	SETPOINT	VALIDATION INSTRUMENT	PMS ID	LINK #
59 Relay	Bus Overvoltage	147.2 VDC	NONE	EY8192	1928 1929 1930
27 Relay	Bus Undervoltage	118.2 VDC			
64 Relay	Bus Ground	25 ± 10K OHMS [1]			

1.0 **<u>REQUIRED ACTIONS</u>**:

1.1 Dispatch an Operator to the 2D2 Battery Charger Room.

2.0 <u>CORRECTIVE ACTIONS</u>:

SPECIFIC CAUSES	SPECIFIC CORRECTIVE ACTIONS
2.1 Battery Charger Malfunction	2.1 Refer to SO23-6-15, Section for Abnormal Operation.
2.2 DC Ground	2.2 Refer to SO23-6-33, Section for Ground Isolation.

3.0 <u>ASSOCIATED RESPONSES</u>:

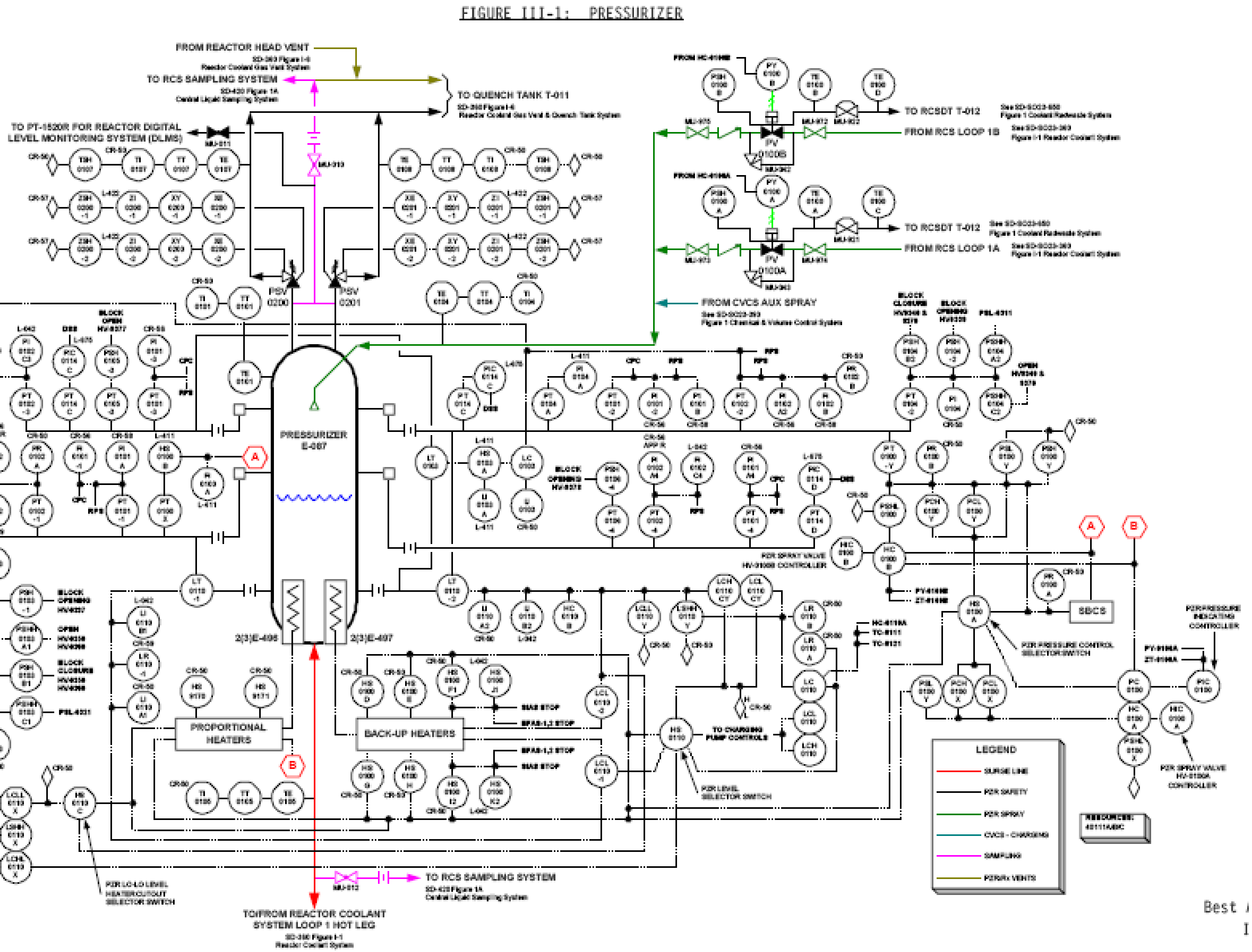
3.1 Notify the CRS/SM and the STA to review Tech. Specs. LCO 3.8.4, LCO 3.8.5, LCO 3.8.9, LCO 3.8.10 and initiate an EDMR/LCOAR, as required.

4.0 <u>COMPENSATORY ACTIONS</u>:

DEVICE NUMBER	SPECIFIC COMPENSATORY ACTIONS
4.1 2D2 bus voltage and % ground	4.1 Monitor 2D2 bus voltage and ground condition at least twice per shift.

[1] Ground detector is located on the DC Bus panel. A ground condition exists when the positive or negative ground LED light is solidly ILLUMINATED.

40. From SD-SO23-360, Figure III-3



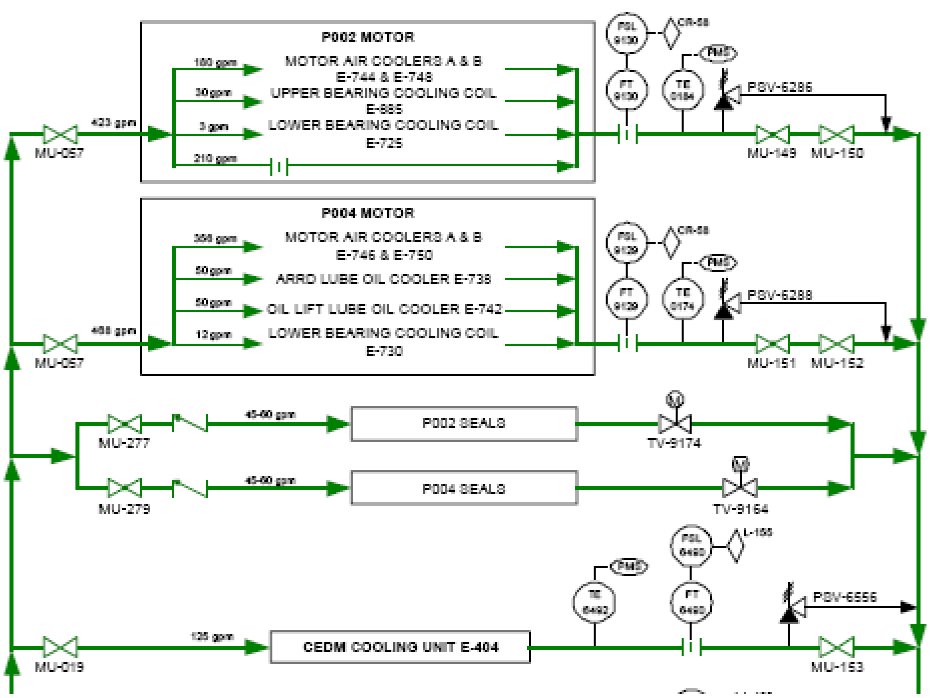
41. From SD-SO23-400, page

2.2.14 **Process Radiation Monitor**

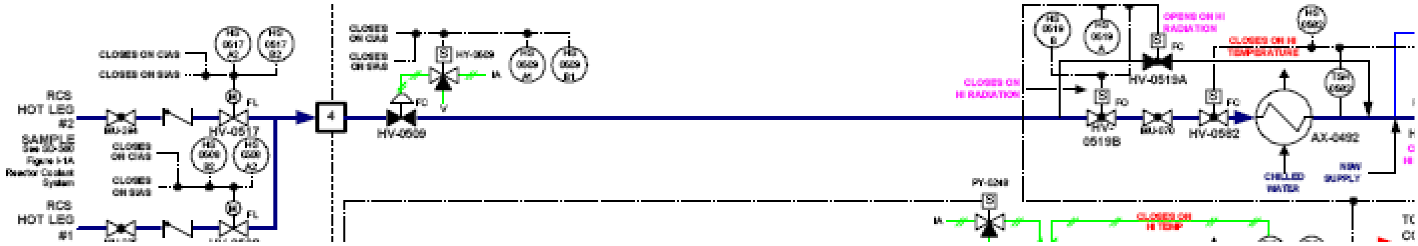
.1 A process radiation monitor 2(3)RE7819 is installed to monitor for radioactive leakage into the component cooling water. The radiation level is read on DAS panel L-103 and 2/3L104, located outside the control room entry. The range of indication is 5×10^{6} to 5×10^{-1} uci/cc.

41. From SD-SO23-400, Figure 4

FIGURE-4: COMPONENT COOLING WATER TO CONTAINMENT

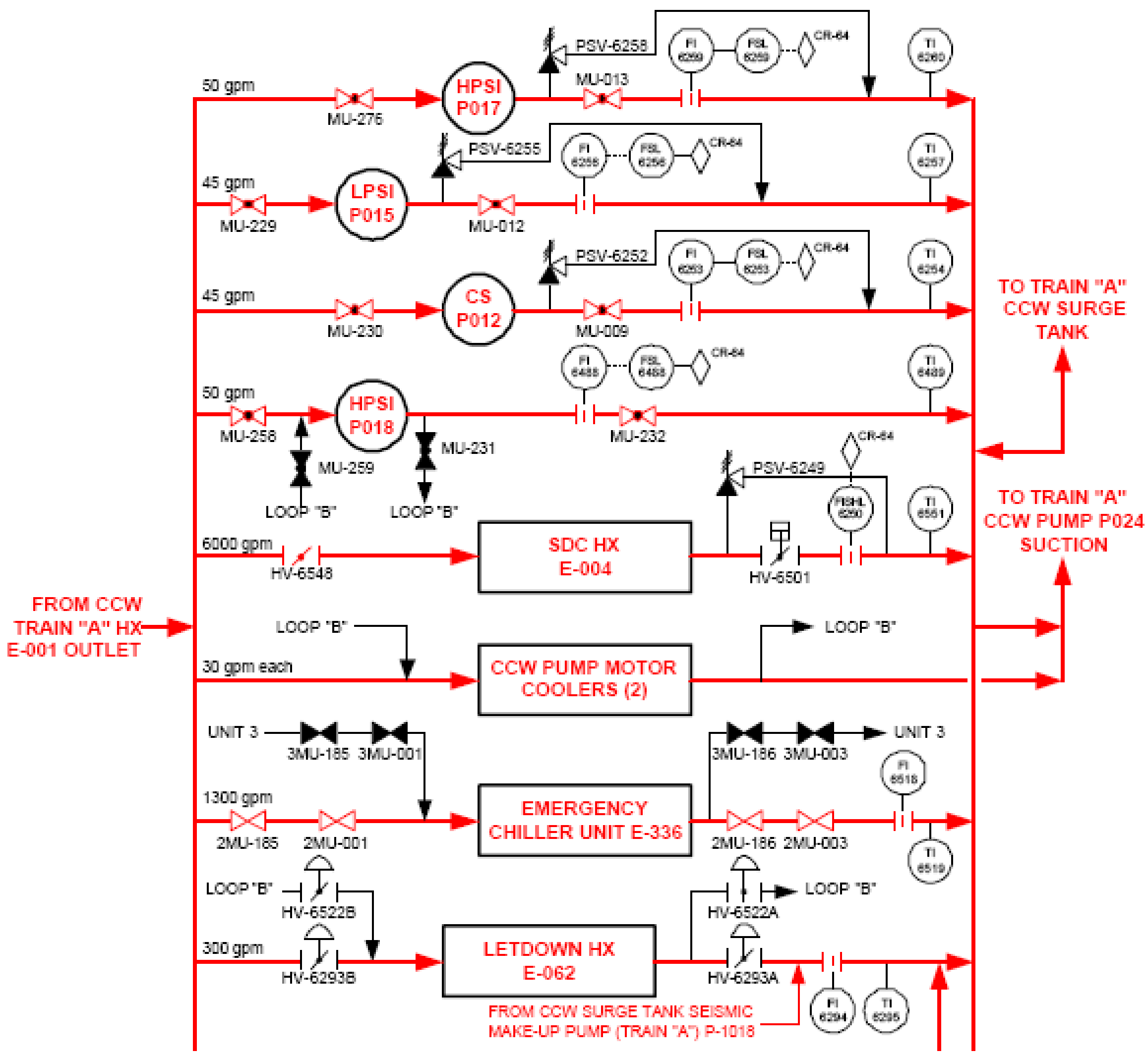


41. From SD-SO23-420, Figure 1



41. From SD-SO23-400, Figure 2A

FIGURE-2A: CRITICAL LOOP COMPONENTS -TRAIN A



42. From SO23-14-1, Step 10

RCP Trip Strategy

The RCP trip strategy is to trip one RCP in each loop for events other than Reactor Trip Recovery. The second two RCPs (i.e., all RCPs) are stopped if PZR pressure lowers to less than the minimum RCP NPSH requirements of the Post-Accident Pressure/Temperature Limits. Additionally, if any RCP does not satisfy the operating requirements of Floating Step *MONITOR RCP Operating Limits* (e.g., temperatures, seal flow, oil pressures, motor amperage), that pump would be stopped. This strategy provides the operators with maximum flexibility for plant control while still ensuring a conservative approach to event recovery.

The RCP trip strategy gives as much time as possible to make a diagnosis of the event in progress while minimizing the severity of any resultant Core uncovery. Tripping two RCPs initially (during the SPTA) reduces the inventory loss rate out the break if a LOCA is in progress, and extends the amount of time for diagnosis.

The Trip 2 / Leave 2 strategy has two main goals: ***First***, it maintains forced RCS circulation for non-LOCA depressurization events, and for LOCA events in which the rate of RCS inventory loss is not unduly exacerbated by leaving two RCPs in service. ***Second***, it ensures that all four RCPs are tripped for LOCAs in which the RCS leak rate may challenge RCS heat removal capability if forced circulation is continued. These actions to stop the RCPs are primarily intended to limit RCS inventory loss under certain large LOCAs. The principal intent is not RCP protective actions.

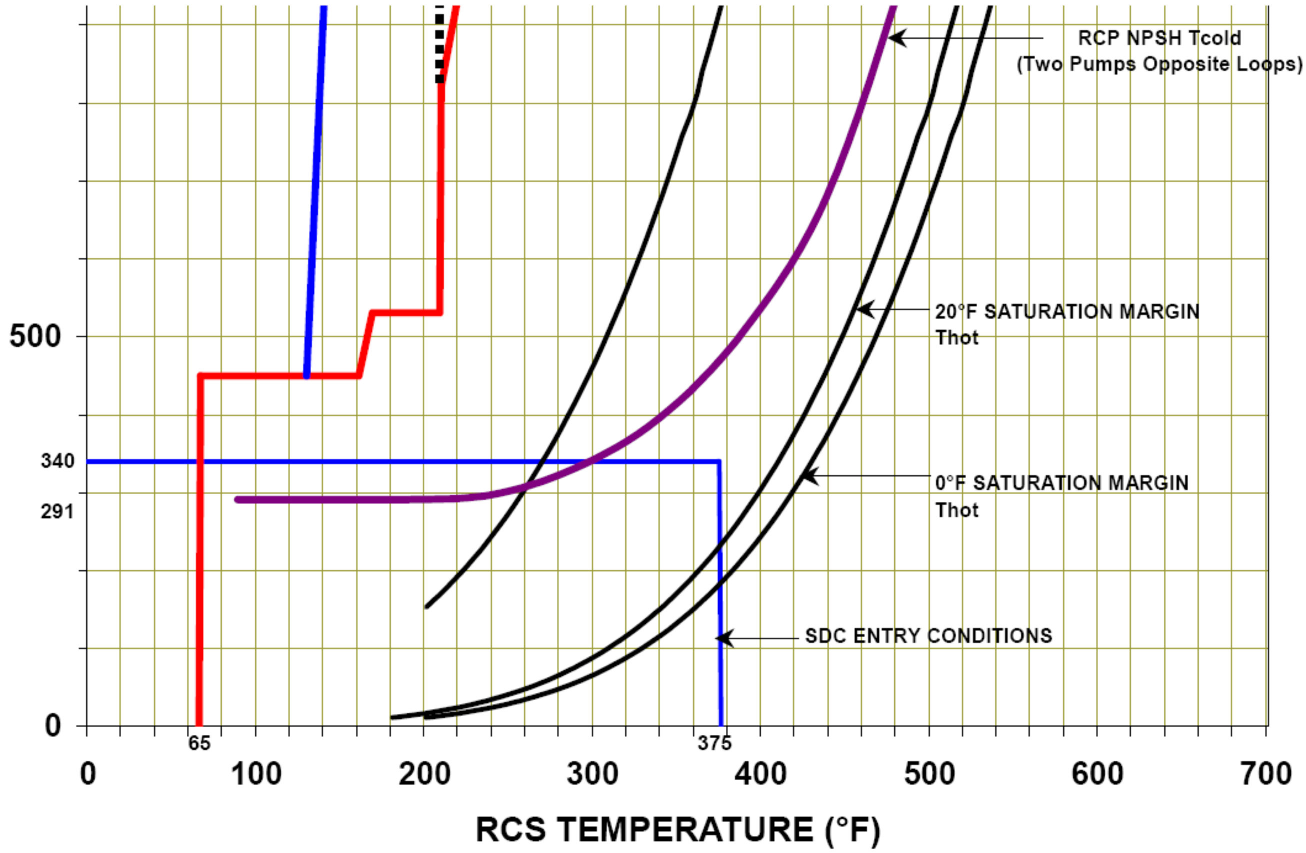
The choice of which two pumps to trip first is up to the Control Room operators, but it is preferred to use two RCPs, located in diametrically opposed loops. The basic T2/L2 strategy does not differentiate between the various pump combinations. When choosing a pump combination, consideration should be given to maintaining the Pressurizer main spray capability.

If the *SO23-12-3, Loss of Coolant Accident* or *SO23-12-9, Functional Recovery* procedures are implemented when no charging pumps are available and no significant RCS leak exists, stopping the second two RCPs with Pressurizer pressure at the [maximum pressure plateau for SBLOCA] would severely inhibit the operator's ability to reduce RCS pressure to the point where HPSI pumps could restore the RCS inventory, since neither main nor auxiliary spray would be available. Conversely, continuing to run two RCPs for LOCAs in which the RCS Inventory loss is great enough to challenge the subcooling margin acceptance criteria carries a separate risk.

Under these conditions, forced circulation tends to increase the total RCS inventory loss. Shortly after NPSH is lost, it is likely that the remaining RCPs will have to be tripped, due to loss of subcooling margin, at a time when the RCS inventory is less than it would have been, had the RCPs been turned off earlier in the transient.

The technical justification for the RCP Trip 2 / Leave 2 strategy found in CEN-268 suggests that a more appropriate point to stop the second two RCPs is when the existence of a LOCA has been conclusively demonstrated by maximum subcooling being less than the 20°F value or RCS pressure is less than the minimum RCP NPSH requirements, whichever is most limiting; for SONGS this is the RCP NPSH requirements. This Trip 2 / Leave 2 has been deemed acceptable by the NRC.

SO23-12-11, Attachment 29



43. From SO23-13-6, Step 2

REACTOR COOLANT PUMP SEAL FAILURE

OPERATOR ACTIONS

2 **Immediate Diagnosis/actions:**

<table>
<tr><th>✓</th><th>AFFECTED PUMP CONDITIONS</th><th>ACTIONS</th></tr>
<tr><td>☞</td><td>3 Seal stages have failed.</td><td rowspan="2">☐ 1) Immediately TRIP the Reactor.
☐ 2) AFTER the CEAs have been inserted for 5 seconds, THEN TRIP the affected RCP(s).
☐ 3) GO TO SO23-12-1.</td></tr>
<tr><td>☞</td><td>Complete loss of CBO flow AND abnormal trends on multiple seal parameters.</td></tr>
<tr><td></td><td>2 Seal stages have failed.</td><td rowspan="3">☐ 1) INITIATE a plant shutdown per SO23-5-1.7.
☐ 2) AFTER the Reactor is tripped AND CEAs have been inserted for 5 seconds, THEN TRIP the affected RCP(s).</td></tr>
<tr><td></td><td>Pressure drop across any single seal stage indicates ≥ 1500 psid.</td></tr>
<tr><td></td><td>Vapor Seal Cavity pressure indicates ≥ 265 psig.</td></tr>
<tr><td></td><td>1 Seal stage has failed AND CBO flow is > 0.6 gpm.</td><td>☐ 1) Contact Maintenance Engineering for evaluation.
☐ 2) GO TO Step 4.</td></tr>
<tr><td></td><td>1 Seal stage has failed AND CBO flow is ≤ 0.6 gpm.</td><td rowspan="3">☐ 1) CONTACT Maintenance Engineering for evaluation.
☐ 2) With Shift Manager Approval, INITIATE a plant shutdown per SO23-5-1.7.
☐ 3) AFTER the Reactor is tripped AND CEAs have been inserted for 5 seconds, THEN TRIP the affected RCP(s).</td></tr>
<tr><td></td><td>Seal staging pressure indicates a change of > 500 psig in one or more stages AND CBO flow is off-scale high (3.5 gpm) AND CBO temperature is > 210°F and not lowering.</td></tr>
<tr><td></td><td>CBO temperature > 170°F following an unexplained, large, sustained temperature rise AND No appreciable change in CBO flow.</td></tr>
</table>

44. From SD-SO23-390, page 11

2.2.1 **Letdown Temperature Isolation Valve, 2(3)TV-0221** (Figure I-1)

SIZE:	2 inches
DESIGN PRESSURE:	2500 psia
OPERATOR:	Air
TYPE:	Globe
FAIL POSITION:	CLOSED
INTERLOCKS:	SIAS CLOSES Letdown Regenerative Heat Exchanger Outlet High Temperature CLOSES

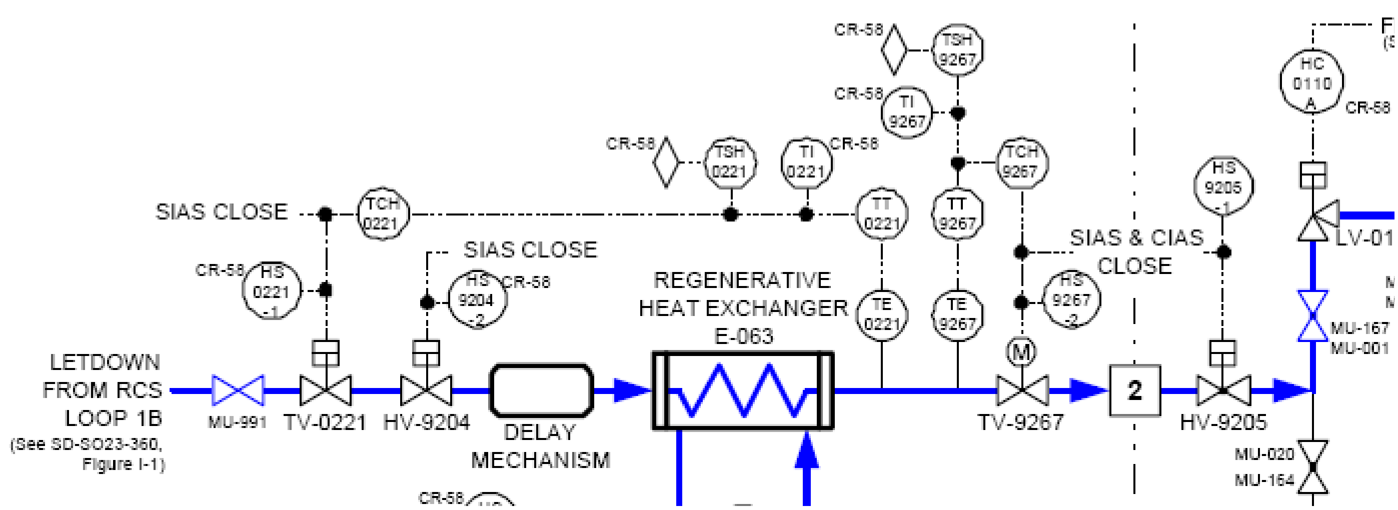
.1 2(3)TV-0221, a safety related valve, is located upstream of the Letdown Delay Mechanism inside the biological shield on Containment 17' elevation.

.1.1 2(3)TV-0221 is manually controlled from an OPEN-CLOSE-OVERRIDE Switchlight Module on 2(3)CR-58 and automatically CLOSES, isolating Letdown upon receipt of one of the following:

.1.1.1 Safety Injection Actuation Signal (SIAS)

.1.1.2 High temperature, @428°F, which is the saturation temperature for 320 psig, which corresponds to the Letdown Flow Control Valves outlet pressure.

44. From SD-SO23-390, Figure 1



44. From SD-SO23-390, page 53

2.2 Components (Continued)

2.2.21 **Volume Control Tank (VCT), 2(3)T-077** (Figures I-1 & 13) (Continued)

.10 Differential Pressure Level Instruments provide VCT, 2(3)T-077 Level Indication, 2(3)LI-0226A, Level Recorder, 2(3)LR-0226 on 2(3)CR-58, and controls for Automatic Makeup System.

.10.1 2(3)L1-0226A is also provided on Evacuation Shutdown Panel 2(3)L-042.

.10.2 Differential Pressure Level Instrument, 2(3)LT-0227, provides a signal to operate VCT Valves 2(3)LV-0227A, 0227B, and 0227C.

.11 Annunciation and recording are provided by the PMS/CFMS.

.11.1 Alarms and actions on decreasing level:

.11.1.1 75% VCT Letdown Flow returns to the VCT.

.11.1.2 73% VCT HIGH LEVEL alarm clears.

.11.1.3 Normal VCT level varies between 37% and 60%.

.11.1.4 35% VCT LOW LEVEL alarm (58A04).

.11.1.5 32% VCT AUTOMATIC Makeup.

.11.1.6 22% VCT LOW LOW LEVEL alarm (58A05).

.11.1.7 6% RWST Outlet Valve, 2(3)LV-0227C OPENS.

.11.1.8 6% VCT Outlet Valve, 2(3)LV-0227B CLOSES.

44. From SD-SO23-390, page 173

COMPONENT	LOCATION			FUNCTIONS	DESIGN DATA	CONTROLS/ INTERLOCKS
	COMPONENT	CONTROL	INSTRUMENT		CAPACITY/TEMP/PRESS	
Boronometer 2(3)AE-0203 and Boronomer Flow Control Valve, 2(3)FV-0203	24' RWB	MCR CR-51 L-090 L-042	MCR CR-51 L-090 L-042	To provide the operator with trends on RCS Boron (B-10) Concentration C_B which supplements normal chemical analysis. To maintain Boronometer flow at 13 gpm.	Neutron Absorption Technique using a 1 curie source of Americium Beryllium (Am-Be) The detectors are Boron Triflouride (BF3) Proportional Counters with an accuracy of ±1.5% plus 5 ppm of actual C_B Design data: 150 °F, 13.0 gpm, 200 psig Boronometer reading is equal to the chemical analysis + 7 ppm. FV-0203: is a 3", air operated, globe valve The spring loaded check valve in parallel with FV-0203 is designed to OPEN on a high ΔP of 35 psid and allow Letdown Flow to continue when FV-0203 CLOSES. NORMALLY - IN SERVICE	Following a step change in C_B equilibrium boron response time is about 10 minutes. Factors in determining Boronometer response time are, RCS to instrument transportation, instrument mixing, and processing times. Normally, Letdown Flowrate is 38 gpm and Boronometer flowrate is 13 gpm. With normal flowrates RCS to instrument response time is ~ 6 minutes, Boronometer mixing time constant is ~ 1.8 minutes and instrument electronic processing time is ~1 minute. FV-0203 fails OPEN on loss of air or power.

45. From SO23-13-15, Step 5

5 **RECOVER SDC Flow:** (Continued)

CAUTION

SFP Cooling from a CS Pump through the same SDC HX may divert SDC flow from the RCS to the SFP.

☐ c. VERIFY SDCS flow - ≥ 2500 gpm. (≥ 2400 gpm on CFMS) [14]

AND

LESS THAN the maximum flow allowed in Attachment 13 (Page 78).

c. 1) ☐ THROTTLE the in-service SDC HX Bypass Flow Valve HV-8160 (preferred) or HV-0396 (alternate), to establish required flowrate.

2) IF standby SDC Pump reverse flow - indicated,

☐ THEN, CLOSE standby SDC Pump Discharge Valve.

3) IF required flowrate cannot be established,

☐ THEN, ENSURE SDCS flowpath alignment - correct.

4) IF CS to SFP Cooling is flowing through the same SDC HX,

45. From SO23-13-15, Step 6

6 **RECOVER RX Core exit temperature:**

CAUTION
While the RCS is in the Midloop Condition, loss of SDCS flow renders all SDCS and RCS loop temperature indicators invalid for monitoring Reactor Core conditions. The <u>only</u> valid indicators are the CETs and/or HJTCs TU8A/B and TU7A/B.

☐ a. VERIFY Core exit temperature [4] - greater than 200°F (350°F if initially in Mode 4) OR - rising.	a. ☐ GO TO Step 7.
☐ b. VERIFY in-service SDC HX CCW flow - greater than 5800 gpm, AND Inlet temperature - less than 95°F.	b. ☐ INITIATE SO23-13-7, Loss of Component Cooling Water (CCW)/Saltwater Cooling (SWC)
☐ c. ESTABLISH required heat removal rate by throttling the in-service SDC HX Outlet Valve (HV-8150 and/or HV-8151). AND	c. 1) ☐ PERFORM SDCS valve alignment check per Attachment 8. 2) <u>IF</u> SDCS valves are properly aligned,

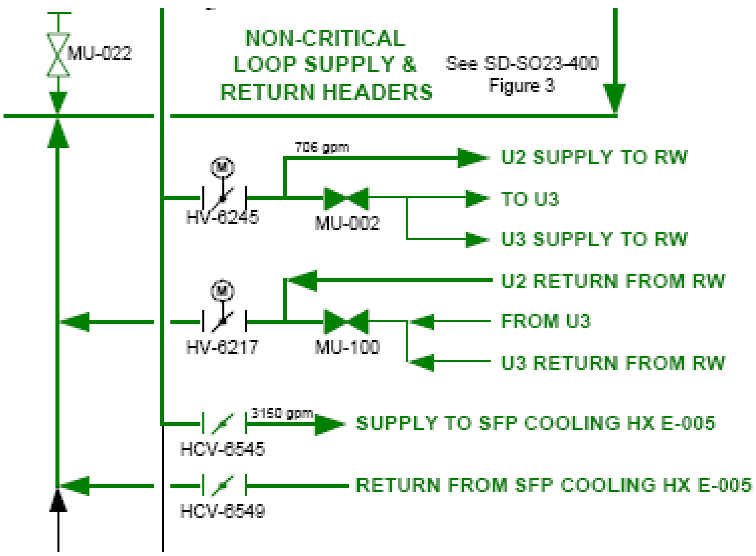
46. From SO23-13-7, Attachment 4

SUPPLYING UNIT 2 CCW SYSTEM FROM UNIT 3 TRAIN A .5.0.0.0.1CCW SYSTEM

CONTINUOUS USE

OBJECTIVE Supply the Unit 2 CCW System from the Unit 3 CCW System by cross connecting through the Radwaste CCW Supply and Return Headers when Unit 2 is in Mode 5 or 6. This attachment would only be used when Unit 2 has lost all Saltwater Cooling or Component Cooling Water. **This attachment invokes 10CFR50.54.X on Unit 2 only.**

46. From SD-SO23-400, Figure 1



47. From SD-SO23-360, page 98

3.3.2 **Pressurizer Pressure Control System Malfunction**

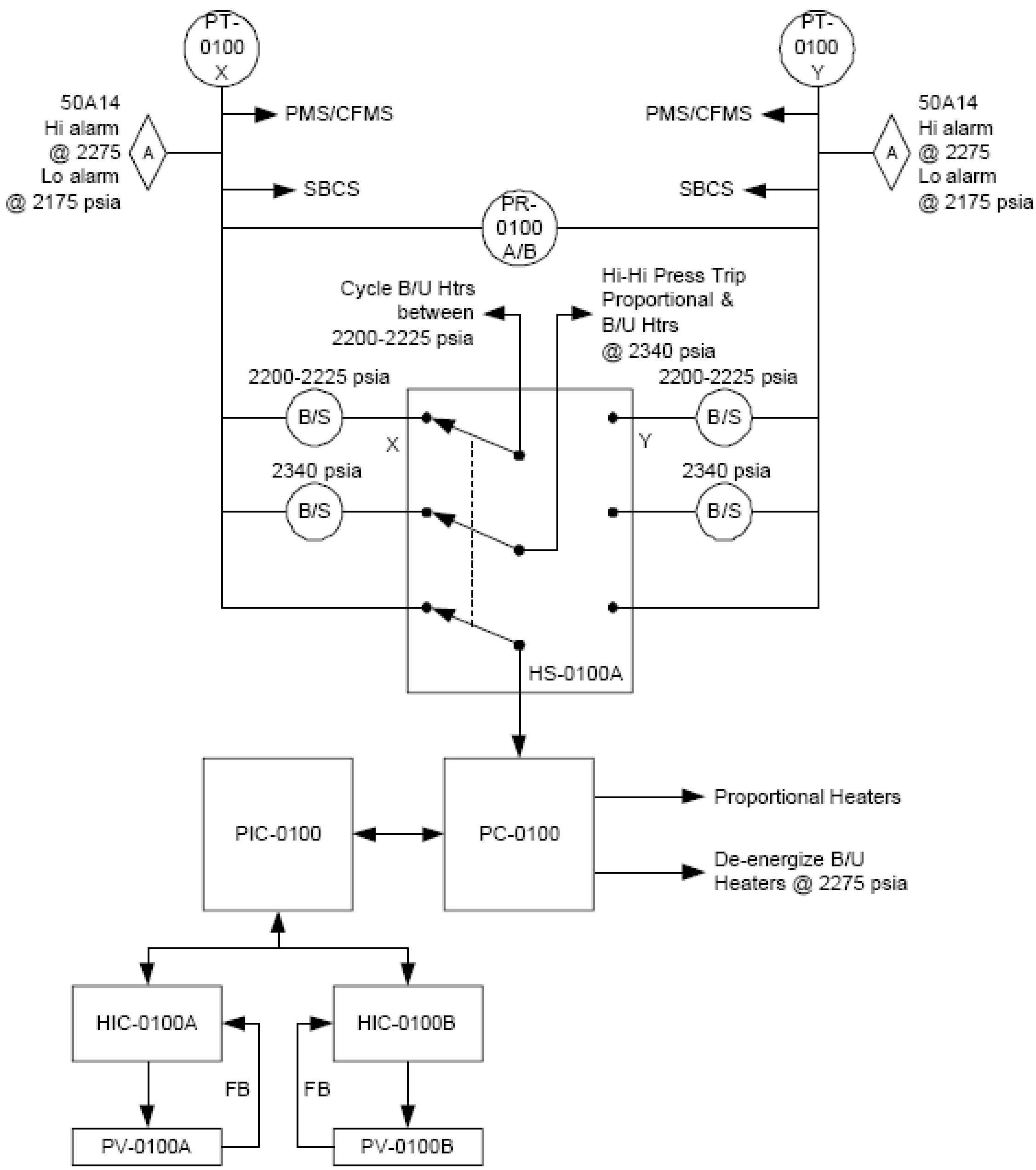
The symptoms of a pressurizer pressure control system malfunction are alarms and displays indicating a deviation between the normal setpoint and actual pressure. These conditions may cause the pressurizer heaters to energize or de-energize if the malfunction is a low or high failure. Spray valves may also open if pressure fails high.

Subsequent actions may require placing controls in manual but this should not be done unless misoperation of automatic control is apparent. Pressure should be restored to within its normal control band using manual control as necessary. If the malfunction is from a failed control channel, pressurizer pressure control is transferred to the operable channel and return control to automatic.

If SIAS and/or EFAS have occurred it will be necessary to use the override feature to control the 1E backup heaters. Pressure is maintained as dictated by emergency conditions and operating instructions.

47. From SO23-13-27, Attachment 1

PRESSURIZER PRESSURE CONTROL SYSTEM BLOCK DIAGRAM



48. From SO23-14-11, page 80

Asymmetric Steam Generator Cooldown

Reactor Coolant System cooldown following an emergency event is normally desired to be accomplished under forced circulation utilizing both Steam Generators to remove heat. Under these optimum conditions, the RCS cools in a generally uniform manner as do both S/Gs.

Specific circumstances of an event however, may require the RCS cooldown to proceed utilizing only one S/G. Additional circumstances may also require the cooldown to be accomplished under natural circulation. It is important to recognize and understand certain differences in the heat transfer mechanisms for this type of cooldown. These mechanisms can result in a significantly different response of the RCS to the cooldown.

Utilizing only one S/G places all the heat removal on the operating S/G. The heat load on the operating S/G might be expected to double for the same RCS cooldown rate. The actual RCS response is one that produces more than a doubling of the heat load. The extra heat load is the result of partial cooling of the secondary side liquid taking place in the non-steaming S/G as RCS liquid passes through its tubes. This extra heat removed from the non-steaming S/G is rejected by the operating S/G. The term ***non-steaming*** S/G is used here rather than ***isolated S/G*** to emphasize that the S/G does not need to be isolated, only that it is not steamed.

Overall this is referred to as ***reverse heat transfer*** in the non-steaming S/G. The heat flow direction in that S/G is from the secondary side liquid to the primary side liquid, which is the reverse direction in which the heat is normally transferred. This should not be confused with reverse RCS flow. In this discussion, RCS flow remains in the normal direction.

RCS liquid passing though the non-steaming S/G increases in temperature as it picks up this heat. RCS T_{COLD} in that loop can actually be greater than RCS T_{HOT} in that same loop. This is frequently called an ***inverted ?T***. Under forced circulation conditions, the increase in temperature is small since the RCS flow is so high. Under natural circulation conditions however, the increase in temperature across this non-steaming S/G is much greater and easy to detect.

Natural circulation itself in the RCS is also affected by the non-steaming S/G. The best way to see this is to first look at natural circulation when both S/Gs are used for the RCS cooldown.

Natural circulation is normally driven by two thermally induced pressure differentials that are complementary to each other. One hydraulic pressure differential is across the Core and is caused by the coolant temperature difference. The Core differential pressure causes the major force that acts to move the coolant in the normal direction through the Core and RCS loops.

48. From SO23-14-11, page 81

The second hydraulic differential pressure is across the S/G and is caused by a similar coolant temperature difference. The S/G differential pressure also causes a force that compliments the Core differential pressure, though smaller. This acts to move the coolant in the normal direction through the RCS loops and eventually the Core.

If one S/G is not steamed and RCS cooldown is performed on the other S/G, a different effect is seen. Reverse heat transfer in the non-steaming S/G produces a hydraulic differential pressure in the opposite direction for that S/G. Rather than being complementary to the Core differential pressure, it actually opposes it.

As long as this opposing differential pressure across the non steaming S/G is small enough, natural circulation will continue in that loop, albeit lower flow than in the loop with the operating S/G. Should the opposing differential pressure increase to a large enough value it can actually stop the natural circulation flow in the loop with the non-steaming S/G. It should be recognized the natural circulation in the loop with the operating S/G continues, generally unaffected by this occurrence, and Core cooling is continued.

Probably the most important item to recognize is that the operating crew is normally in control of the magnitude of this opposing differential pressure. Thus they have the ability to maintain natural circulation flow in the loop associated with the non-steaming S/G.

The parameter most affecting the RCS flow in this loop is the rate of RCS cooldown. The greater the RCS cooldown rate, the greater the reverse heat transfer and the greater the opposing differential pressure. With a faster RCS cooldown rate of 90°F/HR for example, only the RCS loop associated with the operating S/G should be expected to maintain natural circulation flow. The natural circulation flow in the loop associated with the non-steaming S/G should be expected to slow and eventually stagnate. At this point the RCS loops are uncoupled. With RCS loops uncoupled, cooling in one loop will be continued as Core temperatures are lowered. Cooling in the other loop will essentially stop since flow in that loop has stagnated.

A slower RCS cooldown rate on the other hand is not expected to stop natural circulation in the loop associated with the non-steaming S/G. There will still be reverse heat transfer and an opposing differential pressure although it will not be great enough to stop the flow in that loop. Cooling in both RCS loops will be retained and the loops will not become uncoupled.

Overall then it can be seen that not all of the RCS may participate in the cooldown unless both S/Gs are steamed or the RCS cooldown rate is sufficiently low. There are advantages to a faster cooldown rate but there are also some disadvantages if the loops become uncoupled as described earlier.

48. From SO23-12-11, page 92

1 **INITIATE RCS Cooldown and Depressurization:** (Continued)

NOTE

IF Natural Circulation conditions exist and an Asymmetric Cooldown must be performed, THEN slower RCS cooldown rates are needed to maintain RCS flow in both loops. Normally a guideline rate of 35-40°F/HR may be established initially, then slowly increased further provided T_C ΔT does not continuously diverge. For low decay heat conditions, an initial guideline rate of 10-20°F/HR should be used.

49. From SO23-12-11, Attachment 9

CONTROL BUILDING VENTILATION EMERGENCY ACTIONS

ACTION/EXPECTED RESPONSE — RESPONSE NOT OBTAINED

NOTE

Control Room and ESF Switchgear Room Emergency HVAC Train specific equipment is as follows:

	Train A	Train B
CR Emergency AC Units	E-418	E-419
CR Emergency Ventilation Supply Unit	A-207	A-206
CR Emergency Chillers	E-336	E-335
CR Chiller Auxiliaries MCC	BQ	BS
ESF Switchgear Room Emergency Cooling Units	E-255	E-257
CR Cabinet Emergency AC Units	E-424/427	E-423/426

CAUTION

Station Blackout Analysis assumes that:

1) Opening CR Cabinet Doors within thirty (30) minutes of loss of CR ventilation,
2) Restoration of Control Building Emergency Chillers, CR Emergency HVAC, and ESF Switchgear Room Emergency Cooling within one (1) hour of loss of normal HVAC.

These actions prevent damage to equipment in those areas as a result of overheating.

From SO23-12-8, page 24

STATION BLACKOUT

TIME DEPENDENT STEPS

UNIT ________ DATE ________________ Time of entry into **SBO** ____________
Time of **LOOP** ____________

TIME DEPENDENT STEPS		STEP INITIATED	STEP COMPLETED
Step 4c	**30 minutes from losing HVAC.**................ HVAC actions for Control Room and ESF switchgear room.	❑ (Pg. 4)	❑
Step 4e	**30 minutes from SBO**................ *SO23-12-11, Attachment 20, CLASS 1E BATTERY LOAD REDUCTION* must be completed.	❑ (Pg. 4)	❑
Step 4d	**45 minutes from Loss of Offsite Power.**................ D5 loads need to be reduced within 45 minutes. D6 loads need to be reduced as soon as possible if loss of power is expected to extend past 90 minutes. D7 load is reduced after the Main Generator shaft has come to rest if loss of power is expected to extend past 120 minutes.	❑ (Pg. 4)	❑

50. From SO23-12-7, Step 4 Note

4 VERIFY Electrical Power Distribution:

a. VERIFY Reserve AUX Transformers to unit – energized.	a. INITIATE SO23-12-11, Attachment 8, RESTORATION OF OFFSITE POWER.

NOTE

Closing MSIVs using the Control Room MSIV handswitches is preferred over closure through MSIS actuation.

b. VERIFY all Non-1E 4kV buses – energized.	b. 1) IF Non-1E 4kV buses A03 and A07 – NOT energized, THEN a) ENSURE MSIVs – closed: b) ENSURE S/G Blowdown – closed. c) INITIATE FS-19, MONITOR Secondary Plant Equipment.

50. From SO23-14-7, Step 4 Note

4.4.4 STEP 4 VERIFY Electrical Power Distribution

Intent
The intent of this step is to verify the status of the Non-1E 4kV buses; and take the appropriate actions if the buses are de-energized, i.e., restore offsite power, establish secondary plant protection.

.1 NOTE prior to step 4b.

Informs the operator that it is preferred to close the MSIVs using the handswitches. This will minimize the equipment operated and aid in the recovery process. A MSIS actuation initiates closure of valves to provide secondary isolation and S/G blowdown isolation. However, MSIS is less preferred method as it will be reset later and it interrupts existing AFW/FW flow and isolates the ADVs until overridden.

51. From SO23-13-18, Attachment 4

2.2 **EFFECTS AND ACTIONS ON LOSS OF VITAL BUS Y04.**

2.2.1 Perform the following:

AFFECTED EQUIPMENT	**INDICATIONS AND ASSOCIATED ACTIONS**
.1 PPS D status lights extinguished	☐ VERIFY protection system bistables NOT TRIPPED on PPS Channels A and C ROMs.
.2 Channels 2 & 4 Red ESFAS Function lights along the bottom of the ROM extinguished	☐ VERIFY all ESFAS function lights ILLUMINATED on PPS Channels A and C ROMs.
.3 Channel D Lumigraphs on CR56 extinguished	☐ VERIFY Safety Channel indications providing input to PPS Channels A, B, and C <u>do not</u> indicate that a Plant Protection Trip Setpoint has been exceeded.
.4 Vital Bus Inverter Y004 de-energized	☐ ENSURE SO23-6-17, Attachment for Re-energizing Vital Bus Y04 from the Alternate Source, in progress. (Tech. Spec. LCO 3.8.7 and LCO 3.8.9)
.5 EFAS Trip Paths 2 & 4 Valves: HV-4712 HV-4705 HV-4731 HV-4715	• Valves Open. • The affected Unit is in a *4 hour* Action Statement (Tech. Spec. LCO 3.7.5) since these valves will not close on a MSIS signal.
.6 RX Trip Paths 3 and 4 Actuated	• RTCBs 3, 4, 7, and 8 Open. ☐ VERIFY RX Trip Path 1 and 2 lights LIT. ☐ VERIFY RTCBs 1, 2, 5 and 6 are CLOSED. ☐ VERIFY RX Trip Path 3 and 4 indicating lights EXTINGUISHED.
.7 Channel D CPC	• Tripped.
.8 PPS HI Log Power	• Tripped.

52. From SD-SO23-140, page 21

3.3 Abnormal Operations - new: post 2004

3.3.1 During periods when it is necessary to isolate the DC Bus from its associated Inverter, such as maintenance or for ground isolation, the Vital Bus can be transferred to its alternate source through the Manual Transfer Switch.

3.3.2 A Swing Battery Charger is available should one of its Train associated Class 1E Battery Chargers fail or be taken out of service. Connection of the Spare Charger allows the Battery and DC Bus to remain operable even though the Class 1E Charger is out of service. The Swing Battery Charger is identical to the normal Battery Chargers.

.1 Post 2004: Channels A or C and B or D DC Buses can be supplied by their respective Swing Battery Chargers, B021 for Train A and B022 for Train B. The Swing Chargers can only supply one 1E DC Bus at a time via Kirk-Key interlock.

.1.1 Additionally, the Swing Battery Charger B022 for Train B, can supply DC power to the Non-1E D5 Bus. The Train B Swing Charger has an added Kirk-Key interlock to preclude powering more than one DC Bus at a time.

3.3.3 During a Station Blackout event:

.1 Channel A Bus D1 can be cross-connected with Channel C Bus D3, and

.2 Channel B Bus D2 can be cross-connected with Channel D Bus D4.

52. From SO23-15-63.A52, page 107

SPECIFIC CAUSES	SPECIFIC CORRECTIVE ACTIONS
2.1 Battery charger, 2B001 malfunction	2.1 Monitor battery charger operation per SO23-6-15, Sections for 125 VDC Battery and Charger Operation and Abnormal Operation. 2.1.1 If the Charger has Shutdown on High Voltage (HVSD Relay Actuated), then Depress the white HVSD Relay Reset Pushbutton (inside right hand cabinet door, relay card number MCB-2920-E), and perform the following: .1 If Charger Restarts, then check for normal operating parameters, and Notify Electrical Test to Evaluate Charger operation following Restart. .2 If Charger Does Not Restart, then perform Step 2.1.2, and contact Electrical Test to investigate cause of Charger Shutdown. 2.1.2 If the battery charger trouble condition can not be corrected, then perform the following: .1 Unload all unnecessary equipment from 2D1 125 VDC battery. .2 Remove 2B001, battery charger from service per SO23-6-15, Section for 125 VDC Battery and Charger Operation.

52. From SO23-6-15, Step 6.1

6.1 125 VDC Battery Charger Operation

INFORMATION USE

6.1.1 **Removing a Charger from Service**

.1 Remove a Charger (B001, B002, B003, B004, or B005) from service per Attachment 2.

6.1.2 **Returning a Charger to Service**

.1 Return a Charger (B001, B002, B003, B004, or B005) to service per Attachment 3.

6.1.3 **Spare Charger Operation**

.1 Place the Spare Charger S2(3)1806EB017 in service per Attachment 4.

.2 Remove the Spare Charger S2(3)1806EB017 from service per Attachment 5.

.3 If a loss of B017 normal power occurs while supplying D1 or D2 Bus in Modes 1-4, then PROVIDE 1E Power to B017 per Attachment 9.

52. From SO23-6-15, Attachment 6 (reference for Distractor D)

D3 SUPPLY TO D1

CONTINUOUS USE

OBJECTIVE
To allow D1 to remain Operable to support the opposite Unit's AC Sources when in Modes 1-4. D1 Operability is maintained by energizing D1 from D3 Battery and either B001 or B003 Battery Charger while in Mode 5 and 6.

UNIT_______ MODE ________ (Mode 5 or 6) DATE ______________ TIME ________

53. From SO23-13-7, Attachment 12

NUCLEAR ORGANIZATION UNITS 2 AND 3 | ABNORMAL OPERATING INSTRUCTION REVISION 11 ATTACHMENT 12 | SO23-13-7 PAGE 106 OF 109

TRANSFERRING SALTWATER COOLING PUMPS

CONTINUOUS USE

OBJECTIVE

Provide direction for the expeditious transfer of Saltwater Cooling (SWC) Pumps in the event a running SWC Pump trips and the opposite Train is unavailable. This Attachment performs the minimum required actions to start the standby SWC Pump. After the standby SWC Pump is in service, then the normal post-start actions are performed.

CAUTION

DO NOT close the DC Control Power for the SWC Pump to be started until the running CCW Pump has been stopped.
The SWC Pump breaker receives an autostart signal from the running CCW Pump. In the event of a breaker failure, severe injury or death could occur.

2.8 CLOSE the DC Control Power for the SWC Pump to be started. ________

- ☐ A04-10 for P-112 ☐ A06-10 for P-113
- ☐ A04-11 for P-307 ☐ A06-11 for P-114

2.9 START the CCW Pump(s) stopped in Step 2.7. ________

- ☐ MP-024 ☐ MP-026
- ☐ MP-025 (Train A) ☐ MP-025 (Train B)

2.10 Verify the SWC Pump starts. (Mark N/A if the SWC Pump fails to start.) ________

2.10.1 If the SWC Pump fails to start, then GO TO Step 18 (Main Body, Page 25). (Mark N/A if the SWC Pump started.) ________

53. From SO23-2-8, L & S 1.4

1.4 When starting a SWC Pump, then the associated CCW Train should be pressurized to ensure that there is no saltwater leakage into the CCW System.

54. From SD-SO23-390, page 100

2.2.2 **Boric Acid Makeup Tanks, 2(3)T071 & 072** (Figure II-1)

TYPE:	Vertical, Cylindrical
VOLUME:	11,800 gal
DESIGN PRESSURE:	
INTERNAL:	15 psig
EXTERNAL:	Atmospheric
DESIGN TEMPERATURE:	200°F
NORMAL OPERATING	
PRESSURE:	Atmospheric
TEMPERATURE:	=80°F
TYPE HEATER:	Electrical Strap-On
HEATER CAPACITY:	2.25 kW each (2 Banks of 3 Each)
FLUID:	2.25 wt% to 3.5 wt% Boric Acid

.1 Both Boric Acid Makeup (BAMU) Tanks are located on Radwaste Building 24' elevation.

.1.1 BAMU Tanks provide a source of concentrated boric acid solution, with a minimum Technical Specification Limit of 2.3 wt%, for RCS Injection during normal and emergency conditions.

.1.2 Both BAMU Tanks are vented to the Vent Gas Collection Header.

.2 BAMU Tanks contents and volumes are required to be maintained in accordance with Figure LCS 3.1.104 of Licensee Controlled Specifications (LCS).

54. From LCS Figure 3.1.104-1

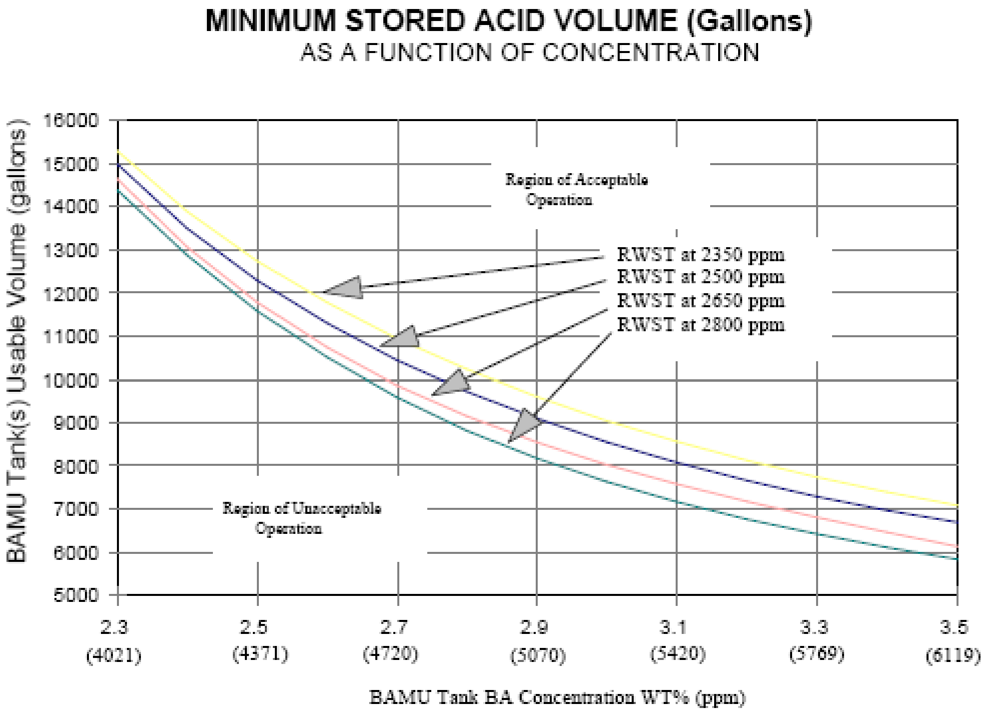


Figure 3.1.104-1

55. From SO23-12-5, Step 18d

18 RESET ESFAS functions: (Continued)

d. VERIFY:

1) Both EFAS-1 and EFAS-2
   – actuated

OR

2) Most affected S/G level

   – greater than 21% NR

   AND

   – NOT lowering.

d. GO TO step h.

CAUTION
MSIS reset may result in AFW Flow to the most affected S/G and possible S/G Tube Rupture.

e. RESET MSIS per *SO23-3-2.22, ESFAS OPERATION*.

55. From SO23-14-5, Step 18d

Step d. & e.: To prevent feeding a hot/dry S/G, MSIS is not reset if either **1)** both EFAS signals have not been actuated or, **2)** *most affected* S/G level is less than or equal to 21% NR or level is lowering. It would be acceptable to reset MSIS if both EFAS-1 and EFAS-2 signals are actuated since the EFAS signal will have been overridden and feed secured to the affected S/G per the S/G isolation step.

Following a Reactor trip from higher power levels, S/G levels typically decrease below 21% NR, resulting in an EFAS actuation. A MSIS concurrent with a ?P less than 125 PSID between the S/Gs prevents automatically feeding AFW to both S/Gs. This condition can result from a steam line break downstream of the MSIVs. In this case, the MSIS signal must be reset to allow EFAS to automatically restore AFW flow to (both) S/Gs. Otherwise, only manual actions (e.g., overriding AFW valves) will restore AFW flow to S/Gs.

Resetting MSIS if *most affected* S/G level is above 21% NR and level not lowering is also allowed since that S/G will not receive an EFAS subsequent to MSIS being reset. (Level is above EFAS actuation setpoint and the S/G is isolated).

56. From SO23-12-6, Step 9h

9 **ESTABLISH MFW Flow to at least one S/G:** (Continued)

h. ESTABLISH at least one MFW Pump discharge pressure greater than S/G pressure:

1) ENSURE MFW Pump discharge valve
– open.

2) VERIFY MFW Pump pretrip alarms
– NOT alarming.

3) ENSURE MSC and EAP at low speed stop.

4) RESET MFW Pump turbine and verify the following:

a) MFW Pump turbine HP and LP stop valves open.

b) MFW Pump miniflow valve open.

5) OPERATE the MSC or EAP to establish MFW Pump discharge pressure greater than S/G pressure.

6) VERIFY proper MFW Pump operation per *SO23-2-1, MAIN FEEDWATER PUMP AND TURBINE OPERATION.*

h. GO TO step 10.

56. From SO23-2-1, Attachment 1

2.16.10 RESET MK-005 Main Feedwater Pump Turbine. (LS-4.7) ________

.1 VERIFY OPEN 2(3)HV-8206 and 2(3)HV-8614, H.P. and L.P. Stop Valves (ZL-8611, CR-53). ________

.2 VERIFY MFWP Turbine speed does not increase by > 200 rpm. (Excessive control valve leak-by.) ________

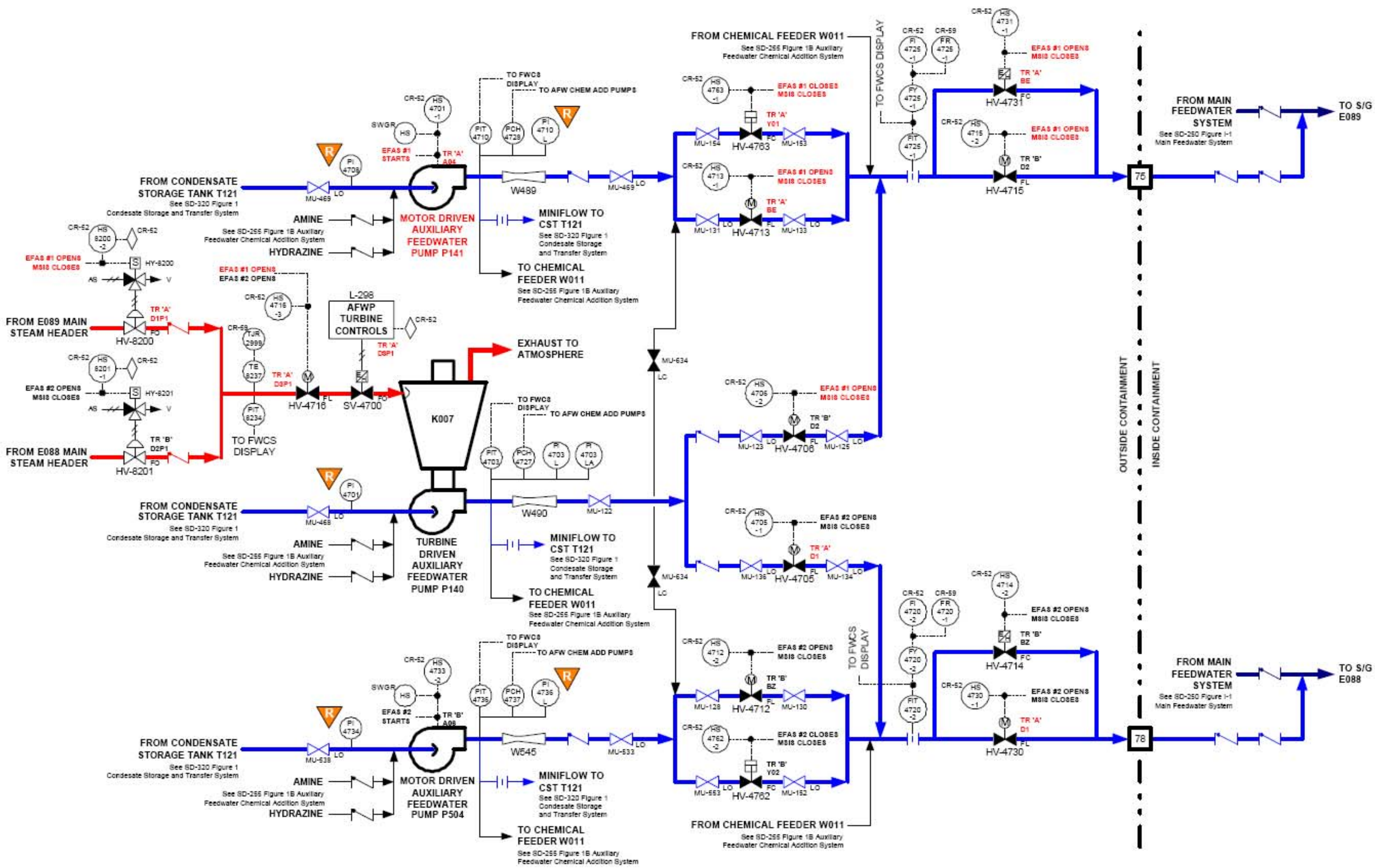
.3 ENSURE 2(3)FV-3433, MK-005 Miniflow Control Valve remains OPEN. ________

.4 STOP the Standby Lube Oil Pump: (LS-8.9, LS-8.11) ________

☐ MP-121 ☐ MP-123

56. From SD-SO23-780, Figure 1

FIGURE 1: AUXILIARY FEEDWATER SYSTEM



57. From SO23-12-1, Step 2

2 VERIFY Reactivity Control criteria satisfied:

a. VERIFY Reactor Trip Circuit Breakers (8) – open.

a. 1) TRIP the Reactor.

2) IF Reactor Trip Circuit Breakers (8) – NOT open,

THEN

ENSURE at least one of the following:

- BOTH M/G set output contactors – open
- 480V Load Centers B15 and B16 – de-energized
- ALL RTCBs – locally opened.

57. From SO23-3-2.19, Section 6.11 (provided as reference for which RTBs open with which trip switch)

6.11.1 _When_ plant conditions require the RTCBs to be opened (e.g., maintenance activities), _then_ perform the following:

.1 Depress HS-9132-1, Reactor Trip 1 pushbutton and verify TCB-1 and TCB-5 OPEN indication ILLUMINATED on Control Room Status Panel _or_ on PPS Reactor Trip Status Panel (above L-032).

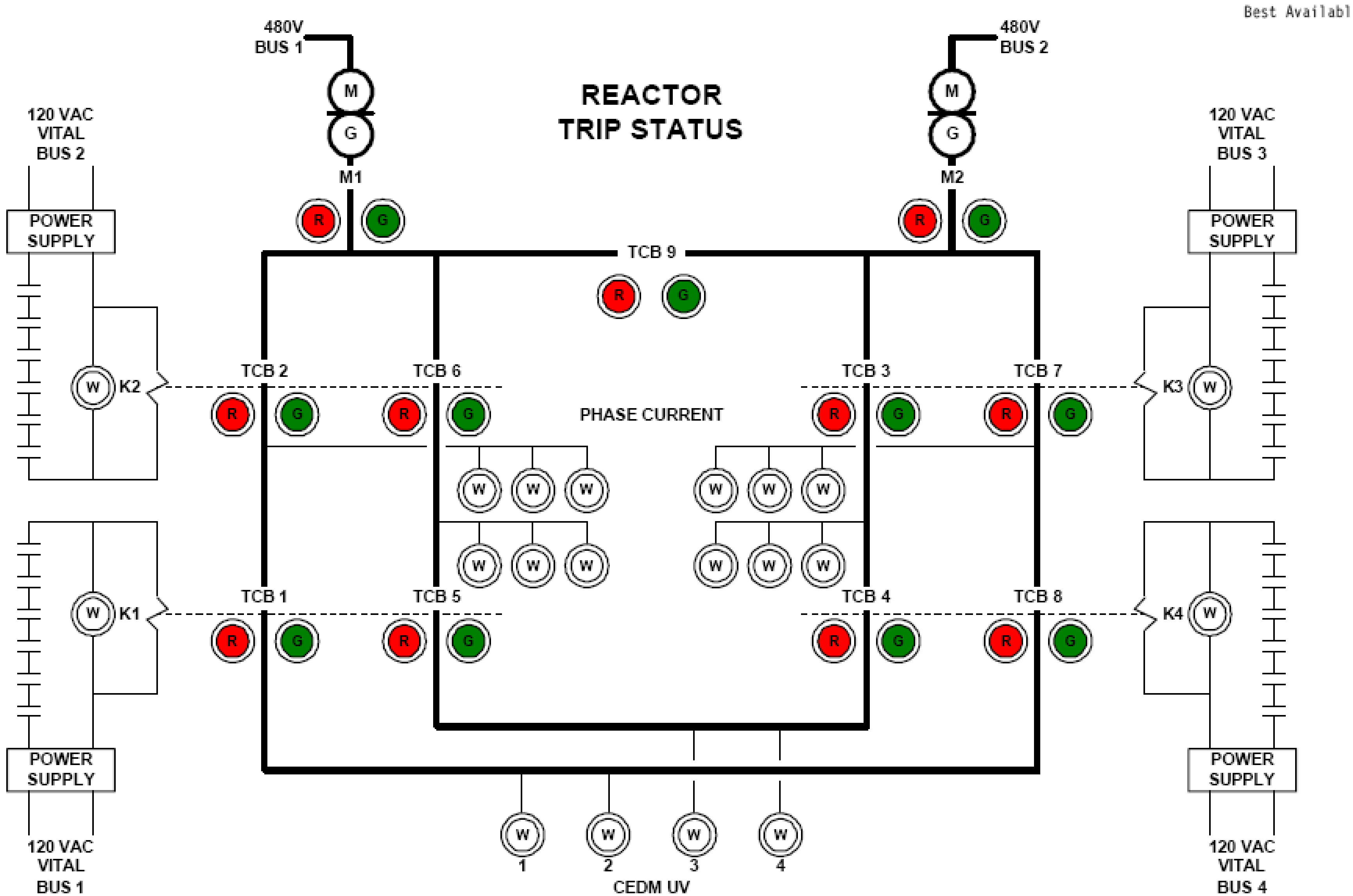
.2 Depress HS-9132-4, Reactor Trip 4 pushbutton and verify TCB-4 and TCB-8 OPEN indication ILLUMINATED on Control Room Status Panel _or_ on PPS Reactor Trip Status Panel (above L-032).

.3 Depress HS-9132-2, Reactor Trip 2 pushbutton and verify TCB-2 and TCB-6 OPEN indication ILLUMINATED on Control Room Status Panel _or_ on PPS Reactor Trip Status Panel (above L-032).

.4 Depress HS-9132-3, Reactor Trip 3 pushbutton and verify TCB-3 and TCB-7 OPEN indication ILLUMINATED on Control Room Status Panel _or_ on PPS Reactor Trip Status Panel (above L-032).

SONGS June 2007 NRC Written Exam Worksheet References

57. From SD-SO23-510, Figure 13



58.From SO23-13-11, Step 2d

2 **Emergency Boration Actions:**

☐a. VERIFY Refueling NOT in progress.	☐ a. ENSURE operations involving core alterations or positive reactivity changes are suspended.
☐b. VERIFY at least one Charging Pump is available.	☐ b. ENSURE RCS Pressure is <1450 psia, AND ☐ INITIATE Boration at >40 gpm using the Operable Boration Flowpath, (RWSTs and HPSI Pump). ☐ 1) GO TO Step 2j.
☐c. OPEN 2(3)HV-9247, Emergency Boration Block Valve.	
☐d. START either BAMU Pump. 2(3)MP-174 2(3)MP-175	d. INITIATE Emergency Boration using Gravity Feed: ☐ 1) CLOSE 2(3)HV-9247, Emerg. Boration Block Valve. ☐ 2) OPEN 2(3)HV-9240, BAMU Tank MT-071 to Charging Pump Gravity Feed Valve. ☐ 3) OPEN 2(3)HV-9235, BAMU Tank MT-072 to Charging Pump Gravity Feed Valve. ☐ 4) ENSURE IN MANUAL AND CLOSE 2(3)LV-0227B, VCT MT-077 Outlet Valve ☐ 5) GO TO Step 2h.
☐e. CLOSE 2(3)HV-9236, BAMU Pump 2(3)MP-174 Recirculation Valve.	
☐f. CLOSE 2(3)HV-9231, BAMU Pump 2(3)MP-175 Recirculation Valve.	
☐g. CLOSE 2(3)FV-9253, Blended Makeup to VCT Isolation, in MANUAL.	
☐h. ENSURE charging flow >40 gpm (FI-0212).	
☐i. START additional Charging Pumps, as necessary, to increase flow rate.	

58. SD-SO23-390, page 124

2.4 Power Supplies

COMPONENT	POWER SUPPLY	UNIT 2	UNIT 3
Boric Acid Makeup Pump, 2(3)P-174	MCC	BY14	BY14
Boric Acid Makeup Pump, 2(3)P-175	MCC	BY15	BY15

59. From 0FD122, Chapter 2, page 18.

Reference Leg Partially/Completely Drains:

If the reference leg level were to drop due to a leak or evaporation, then reference leg height (z) decreases and

$$P_R = ?_{liquid_R} z + P_{gas} \quad \text{decreases,}$$

Therefore,

$$\text{Output} = P_R - P_V \quad \text{decreases}$$

Output decreases means IL increases

AL did not change, but IL increases means that **IL > AL**

A partially or completely drained reference leg produces an indicated level that is greater than actual tank level. Unless wet reference legs are maintained full and properly monitored on a periodic basis, the level detection system will indicate a greater tank inventory than there actually is.

Wet Reference Leg Level Detection: Saturated System

The SONGS Pressurizer and the shell-side of SONGS Steam Generators are referred to as ***saturated systems***. Any system where water and steam coexist at the same temperature and pressure fits this classification.

For a saturated system, the "gas" above the water level is actually water vapor at the pressure at which the system is being maintained (ignoring non-condensable gases in the steam space). ***Figures 9*** and ***10*** show that a wet reference leg level detection system is used for the Pressurizer and Steam Generators.

60. From SO23-13-20, Step 2

2 Fuel Handling Accident with High Radiation actions: (Continued)

☐ c. VERIFY Containment Area Radiation Monitors - NOT alarming <u>or</u> trending to alarm.	c. PERFORM the following: ☐ 1) INITIATE CPIS.

60. From SD-SO23-690, page 29

RAD MONITORS	DESCRIPTION
2/3 RE-7844	Radwaste High Radioactive Storage Area Radiation
2(3)RE-7845	Containment Personnel Lock Area Radiation
2(3)RE-7847	Safety Equipment Building Area Radiation
2(3)RE-7848	Containment Building 30 Ft. Area Radiation
2(3)RE-7850	FHB Spent Fuel Cask Area Radiation

60. From SD-SO23-690, page 43

Radmonitor	Description
2(3)RIC-7804G1,P1	Containment Train A Airborne Radiation
2(3)RIC-7807G2,P2	Containment Train B Airborne Radiation
2(3)RIC-7822G1	Fuel Handling Building Train A Airborne Radiation
2(3)RIC-7823G2	Fuel Handling Building Train B Airborne Radiation

61. From SO23-14-4, Step 25

4.4.25 STEP 25 VERIFY SDC Entry Conditions

Intent
The intent of this step is to verify the Shutdown Cooling entry conditions that were previously identified have been established.

.1 NOTE prior to Step 25a.

Sensor #4 of the RVLMS is located approximately five inches below the bottom of the Upper Guide Support Plate. If sensor #4 is covered, then the indicated Plenum level will be 100%. An indicated level of 100% in the Reactor Plenum level indicates the Plenum is free of voids. Keeping the Plenum free of voids was selected to be conservative. Maintaining these limits provides assurance that gas binding of the SDC Pumps will not occur once SDC is initiated.

Method
Per *SO23-3-2.6, Shutdown Cooling System Operation*, post-accident SDC entry values are RCS T_{HOT} of 375°F and PZR pressure of 340 PSIA. Entry into shutdown cooling may be initiated when the following plant conditions exists:

1) Reactor Vessel Plenum level greater than or equal to 100%,
2) PZR pressure less than 340 PSIA,
3) Core Exit Saturation Margin greater than or equal to 20°F,
4) SM/OL evaluates that RCS activity is within limits,
5) SO23-12-11 attachment for Cooldown/ Depressurization is complete, and
6) RCS T_{HOT} is less than 375°F (or 386°F if using multiple indications).

Step a.: Verifies RCS Inventory. The Reactor Vessel level (Plenum) value of 100% indicates that the hot leg nozzles are covered (HJTC #4 is covered with water and it is located at the top of the plenum area). If the RCS cannot be depressurized, then voiding may be causing RCS pressure to remain high. If Reactor Vessel level (Plenum) is less than 100%, then the Floating Step, *ELIMINATE Voids*, is initiated. An indicated plenum level of 100% is required to ensure the hot legs are full and therefore, ensures sufficient inventory to sustain Natural Circulation or SDC. During natural circulation cooldown (without CEDM fans), the Reactor Vessel Head does not receive significant cooling, and its temperature remains higher than the loop temperature. This results in an expected void forming in the head as the cooldown progresses. Eliminating voiding may take considerable time; during this time, the SM/OL may evaluate additional measures to minimize the release of contaminants, review the SO23-5-1.5/EOI interface to identify facilitating actions, or review alternate procedures[1] as required to place the plant on SDC.

61. From SO23-14-4, Step 23

4.4.23 STEP 23 ESTABLISH SDC Entry Conditions

Intent
The intent of this step is to establish the principal conditions that must be achieved until the normal process for placing LTOP in service per *INITIATE SDC Operation* is accomplished. A subsequent verification performed later will verify these along with additional secondary type conditions needed for SDC entry.

.1 CAUTION prior to Step 23b.

As discussed in earlier steps, the Caution warns that isolated S/G pressure response may inhibit establishing SDC conditions. Isolated S/G cooldown steps are intended to prevent this.

Method
The RCS cooldown is continued to establish the RCS temperature (T_{HOT}) and pressure conditions for entry to the SDC system. The values of RCS T_{HOT} less than 375°F and PZR pressure less than 340 PSIA are the desired values for post-accident SDC. Selection of these values ensures the design parameters of the SDC System are not exceeded including nominal instrument inaccuracies.

Steps a.: If the value of 375°F cannot be reached then the value of 386°F, based upon multiple indications is allowed. The value of 386°F is within the piping design temperature of 400°F when using a multiple instrument uncertainty of 14°F. Because the value of 386°F is based on averaging multiple indications, it should only be used if the normal value of 375°F cannot be obtained. Per ABB analysis ABB-A-SG-FE-0090, one ADV does not have enough capacity to cool the plant down to below RCS T_{HOT} of 375°F under 10CFR50 Appendix K assumptions: 1) Decay heat of 120% and 2) minimum ADV flow capacity). However, one ADV does have sufficient capacity to cool the plant to below 386°F under Appendix K conditions.

Steps b.: Establishes PZR pressure required for over-pressure protection of the RCS during low-temperature conditions via the SDC suction line relief valve, PSV-9349. The maximum pressure for SDC operation is selected to provide a conservative pressure margin below the SDC relief valve setpoint of 406 PSIG. A PZR pressure of less than 340 PSIA minimizes the potential for lifting PSV-9349. The static head from the relief valve to the PZR has been factored into value for the PZR pressure established for SDC entry conditions. The overpressure protection is provided because of the lower pressure rating of the SDC suction piping versus the pressure rating of the RCS piping. The lower pressure rating of the SDC suction piping is shown as a piping code break. Only the low range pressurizer pressure indicators (either LI0103/0104 on CR50, QSPDS page 611, or CFMS page 311) have the acceptable TLU[1] to verify SDC entry pressure.

61. From SO23-14-11, FS-29 Bases

4.5.29 FS-29, COOLDOWN Isolated S/G (Continued)

Depressurization of the RCS below saturation pressure of the isolated S/G could void large portions of the isolated RCS loop, which could cause the isolated S/G to act as a pressurizer and delay depressurization to SDC entry conditions. Thus, an isolated S/G should be cooled down along with the RCS.

Several methods are available for cooling the isolated S/G. In each method, radiological releases to the environment from the affected S/G must be minimized.

The preferred option is to backflow to the S/G into the RCS until a small portion of the U-tubes is exposed. The next option is to backflow S/G into the RCS then refill the S/G using feedwater. These are described in more detail in the associated attachments.

The third option is to steam the isolated S/G to the Main Condenser. Short term steaming to the Main Condenser provides depressurization while minimizing radiological releases to the environment. Exhaust of non-condensable gases from the Main Condenser is directed through a filtration unit.

If the Main Condenser is not available, then draining the S/G to the Radwaste System is evaluated. Draining the S/G to Radwaste serves to contain the contaminated water in the affected S/G. This method does not, however, provide cooling of the S/G and may not be possible if the contents of the affected S/G are too hot. To facilitate cooling the Shift Manager/Operations Leader may direct feeding of the isolated S/G. This will provide a cooling medium to transfer heat prior to steaming.

Ambient cooling of the isolated S/G occurs with any RCS cooldown. Exclusive use of ambient cooling could take well over 24 hours. If S/G level control can be maintained during this period, this may be considered since no radiological releases occur after the S/G is isolated. This method is typically not used exclusively, however, due to the time required to complete the cooldown.

If other options cannot be implemented, then short duration steaming to the atmosphere via the Atmospheric Dump Valves (ADVs) is evaluated. Radiological effects of the steaming are considered. Radiological effects of the steaming normally make this the least preferred option.

62. SD-SO23-690, page 65

2.3.4 **Gaseous Effluent Radiation Monitoring System** (Continued)

.4 Plant Vent Stack/Containment Purge Wide Range Radiation Monitors, 2(3)RE-7865A1, B1, C1 (See Figures 21B and 15A)

.4.1 2(3)RE-7865A1, B1, C1 is a wide range effluent monitor with the capability of monitoring either the Plant Vent Stack or the Containment Purge Stack (switchable). It covers 12 decades of noble gas activity from 1 E-7 to 1 E+5 μCi/cc. The monitor itself is identical to 2(3)RE-7870A1,B1,C1

.4.2 Aligned to the Plant Vent Stack (its normal alignment), upon high radiation, instrument failure or a loss of power, 2(3)RE-7865A1, B1, C1 CLOSES the Waste Gas Discharge Header Isolation Valve, 2/3FV-7202.

.4.3 The design of the Continuous Exhaust Plenum is such that the monitoring of one Plant Vent Stack indicates one-half the total plant release. The function of this monitor is to supplement the capability of the Plant Vent Stack Wide Range Radiation Monitor, 2/3RE-7808 by providing high range capability for measuring noble gas concentrations.

62. From SD-SO23-622, page 57

2.2.3 Plant Vent Stack Airborne Radiation Monitor, RE-7808 and Plant Vent Stack and Containment Purge Stack Effluent Radiation Monitors, 2RE-7865-1 and 3RE-7865-1

Three Radiation monitors are provided to monitor gaseous radiation of the Continuous Exhaust System. A monitor, common to both units, monitors the discharge of the exhaust fans for gaseous radiation. Each Plant Vent Stack has a wide range gas monitor that can also be used to monitor the Unit's Containment Purge Stack. The monitors provide indication (2/3L-104) and alarms in the Control Room, for more information see System Description SD-SO23-690, Radiation Monitoring System. A high radiation alarm from the monitors closes the Waste Gas Header Isolation Valve (see System Description SD-SO23-600, Gaseous Radwaste System), to prevent Waste Gas discharge into the Continuous Exhaust System.

63. From SO23-13-2, Attachment 9

3.11 Ensure AFW flow to 3ME-089 as follows:

3.11.1 VERIFY 3MP-141 Running. ________

3.11.2 OPEN S31305MR976, AFW Pump MP-141 discharge Pressure gauge 3PI-4710L root valve. ________

3.11.3 Coordinate with 32 as follows:

.1 <u>If</u> 3ME-089 level is less than 30% NR, <u>then</u> 3HV-4713 should be fully Opened.

.2 Calculate $\geq$ 400 gpm of AFW Flow as follows:

Discharge Pressure: 3PI-4710L (A) ___________

Suction Pressure: 3PI-4708 (B) ___________

MP-141 TDH = (A) - (B) = __________ psig ________

.3 Verify Total Discharge Head <1230 psig ($\geq$400 gpm). ________

63. From SO23-13-2, Attachment 9

4.0 Align AFW to 3ME-088, as follows:

4.1 Verify 3MP-141 Running. ________

4.2 Open access cover on top of oil shroud, and DEPRESS 3HV-4716 Manual Trip Lever to Stop 3MP-140. ________

4.3 UNLOCK (GMK) and OPEN S31305MU634, 3MP-504/3MP-141 Cross-connect (located at southwest corner of 3MP-141). ________

4.4 UNLOCK (GMK) and OPEN S31305MU635, 3MP-141/3MP-504 Cross-connect (located south of 3MP-141, below piping). ________

4.5 BLOCK OPEN the AFW Pump Room WEST door with the wedge provided in the SSD KIT, to allow ventilation for AFW Pumps. ________

4.6 In the AFW Penetration Doghouse:

4.6.1 MANUALLY CLOSE 3HV-4730, 3ME-088 AFW Penetration Isolation. ________

4.7 In the West RWST Vault: (Key 222)

4.7.1 MANUALLY OPEN 3HV-4712, 3MP-504 Discharge to 3ME-088. ________

5.0 In the East RWST Vault:

5.1 CLOSE S31414MU092, 3MT-120 MUD Hdr. Isolation to prevent Condensate Storage inventory loss. (Located at bottom of ladder, under missile shield.) ________

6.0 Connect Headset to CKT No. 1 jack on AFW Area west fence, and establish communications with the 31. ________

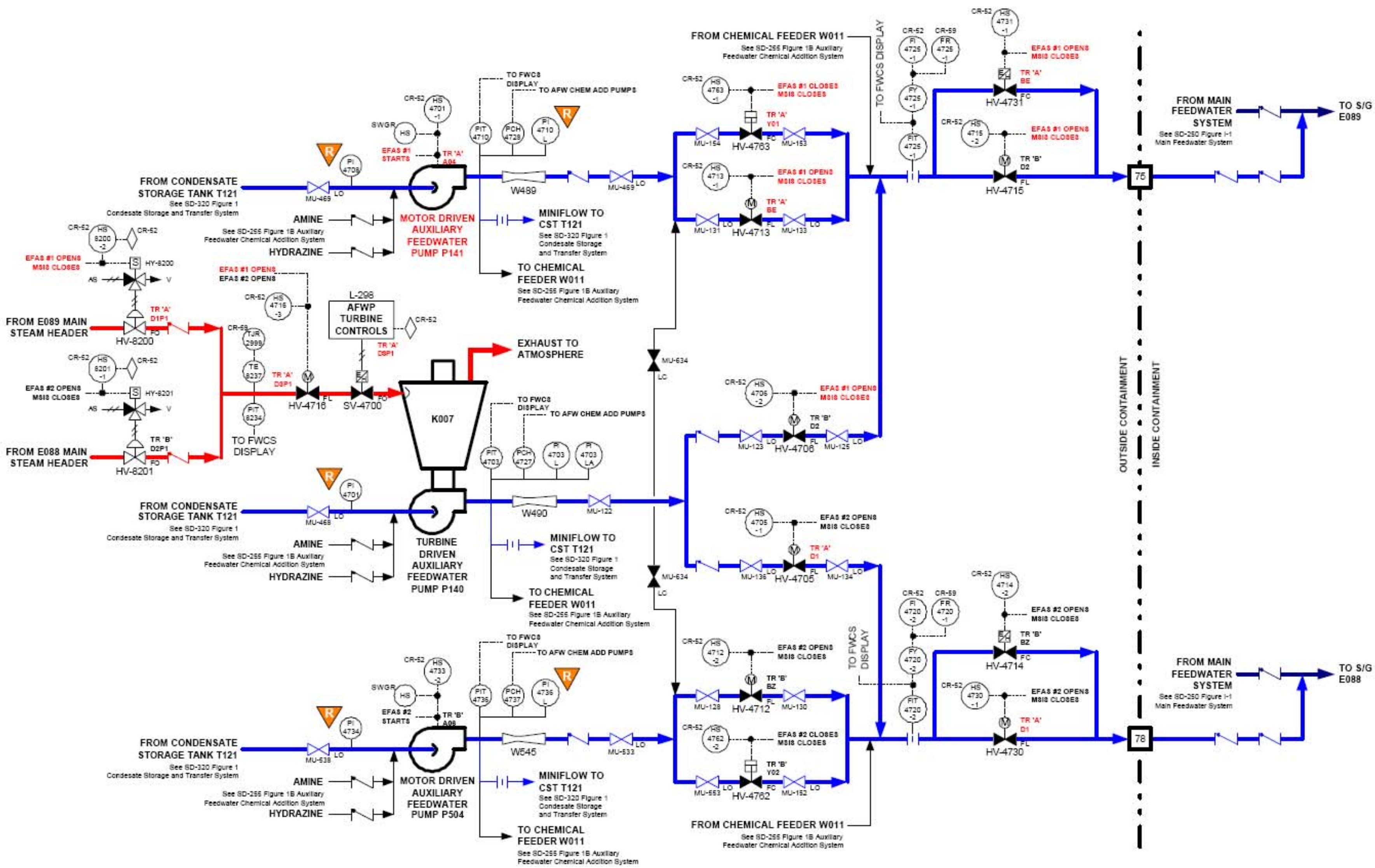
7.0 Manually Control 3HV-4730, 3ME-088 AFW Penetration Isolation, as directed by the 31 as follows:

7.1 <u>If</u> 3ME-088 level is less than 30% NR, <u>then</u> 3HV-4730 should be fully Opened. ________

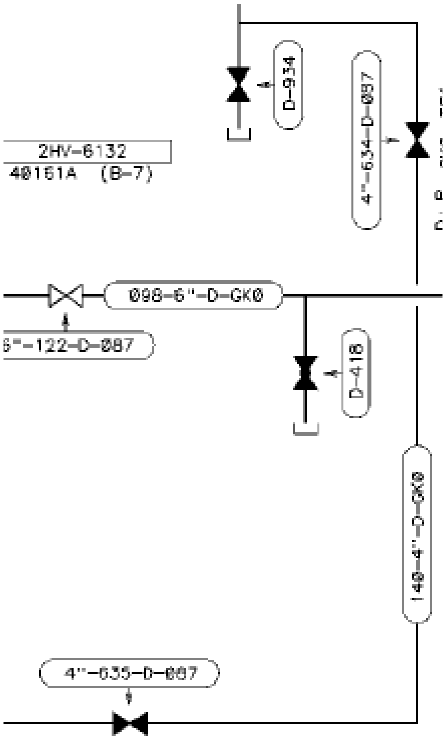
7.2 Coordinate with the 31 to Maintain S/G level between 55% and 90% NR by throttling 3HV-4730. ________

63. From SD-SO23-780, Figure 1

FIGURE 1: AUXILIARY FEEDWATER SYSTEM



63. From P & ID 40160A AFW (Provided to show that the cross connect valves are labeled 634 & 635. There is a typo on SD-SO23-780, Figure 1.)



64. From SO23-3-2.1, L & S 4.1 and 4.9

4.0 **PZR DEGAS INFORMATION**

4.1 Pressurizer degasification provides for the removal of radioactive gases from the RCS. This is normally done when offline or when preparing to go offline, and it may be enhanced by forcing PZR Sprays and/or energizing PZR heaters. The flowpath is from the PZR Sampling System line through Containment Isolation Valves HV-0510 and HV-511 to PV-0248 (in Pen. Rm. 209). This reduces the sample line pressure to ≈80 psig. The fluid is cooled to < 137°F by passing through a sample cooler, and then directed to either the VCT or the Coolant Radwaste System upstream of HV-7823, Radwaste Primary Ion Exchanger Inlet Valve.

4.9 As part of the decision of where to align PZR Degas, the following OE should be considered. During a shutdown while burping the VCT, RCS dissolved H2 dropped, while VCT gas space remained steady. This unexpected condition was due to PZR Degas being aligned to the VCT rather than to Radwaste. Lesson learned to allow timely transition to Shutdown Cooling is that PZR Degas should be aligned to Radwaste during plant shutdown. (AR 060400334)

65. From SO23-12-9, Step 13

13 VERIFY SDC entry conditions:

NOTE
During a Natural Circulation cooldown, voiding in the Head is expected to occur when depressurizing to go on SDC. The strategy is to collapse the void when Plenum level is less than 100% and RAS has not actuated.

a. VERIFY RAS – NOT actuated.	a. 1) IF Reactor Vessel level – greater than or equal to 61% (Plenum): QSPDS page 622 CFMS page 312 SO23-12-11, Attachment 4. THEN GO TO step 13f.

65. From SO23-14-9, Step 13 Bases

4.4.13 STEP 13 VERIFY SDC Entry Conditions

Intent
The intent of this step is to verify the Shutdown Cooling entry conditions that were previously identified have been established.

1. NOTE in step 13:

Sensor #4 of the RVLMS is located approximately five inches below the bottom of the Upper Guide Support Plate. If sensor #4 is covered, then the indicated Plenum level will be 100%. An indicated level of 100% in the Reactor Plenum indicates the Plenum is free of voids. Keeping the Plenum free of voids was selected to be conservative. Maintaining these limits provides assurance that gas binding of the SDC Pumps will not occur once SDC is initiated.

Method
Per *SO23-3-2.6, Shutdown Cooling System Operation*, post-accident SDC entry values are RCS T_{HOT} of 375°F and PZR pressure of 340 PSIA. The following minimum conditions should be met prior to establishing SDC. It is desirable that all of the entry conditions are stable or trending to further within the entry conditions. Entry into shutdown cooling may be initiated when the following plant conditions exists:

1) Reactor Vessel Plenum level greater than or equal to 100% if RAS NOT actuated, or greater than 61% (plenum) if RAS has actuated.
2) RCS T_{HOT} is less than 375°F (or 385°F using multiple indications if only one ADV available); or REP CET is less than 375°F (or 385°F if only one ADV available).
3) PZR pressure less than 340 PSIA (or adjusted for Containment pressure if Containment pressure is greater than 3 PSIG,
4) Core Exit Saturation Margin greater than or equal to 20°F,
5) SM/OL evaluates that RCS activity is within limits, and
6) SO23-12-11 attachment for *Cooldown/ Depressurization* is complete.

66. From SO23-13-21, Attachment 3

FIGURE LVL-CST

NOTE

There must be no flow going past the AFW Pump Suction Pressure Gauges when determining Condensate Storage Tank levels.

SUCTION PRESSURE GAUGE READING (PSI)	PI-4701 (P-140) PI-4708 (P-141) PI-4734 (P-504)		PI-3394L (P-049)		
	%	GAL T-121	%	GAL T-120 ONLY	GAL T-120/T-121 CROSS CONNECTED
12.35	---	----	100.0	446,558	------
12	---	----	97.3	434,680	------
11.5	100.0	148,668	93.5	417,515	566,183
11	96.0	142,649	89.7	400,635	543,284
10	87.7	130,451	82.1	366,714	497,165
9	79.5	118,201	74.5	332,786	450,987
8	71.3	105,950	66.9	298,783	404,733
7	63.1	93,755	59.4	265,160	358,915
6	54.4	80,851	51.7	230,990	311,841
5	46.6	69,330	44.1	197,050	266,380
4	38.4	57,146	36.5	163,185	220,331
3	30.2	44,908	28.9	129,182	174,090
2	22.0	32,725	21.3	95,262	127,987
1	13.8	20,540	13.4	60,270	80,810
0	5.6	8,269	6.3	27,956	36,225

67. From SD-SO23-720, page 23

INPUTS & SETPOINTS:	Emergency Feedwater Actuation Signal (EFAS)	
		Low Steam Generator Level @21%, and no rupture (S/G Pressure >741 psia)
		Low Steam Generator Level @21%, and a rupture (S/G Pressure <741 psia), and Steam Generator differential pressure @ ≥125 psid
	Diverse Emergency Feedwater System (DEFAS)	
		Low Steam Generator Level @16%, concurrent with a Diverse Scram System (DSS) and the absence of a MSIS and EFAS, with same Main Steam and Differential Pressure conditions as EFAS. DEFAS resets (turns off at 21% increasing).
INITIATING DEVICES:		Steam Generator Levels 2(3)LT-1113-1, -2, -3, -4 2(3)LT-1123-1, -2, -3, -4
		Steam Generator Pressure and Differential Pressure (ΔP) 2(3)PT-1013-1, -2, -3, -4 2(3)PT-1023-1, -2, -3, -4
LOGIC:		EFAS & DEFAS: 2/4 coincidence

67. From SD-SO23-720, page 26

.6.8 The Cycling Relays do not lock in when initiated, nor is there EFAS-1 or EFAS-2 RESET at the Actuation Reset Panel like other ESFAS functions.

.6.8.1 Steam Generators might overfill if the EFAS signal locked in.

.6.8.1.1 This allows the Steam Generator Levels to cycle between 21% - 26% for EFAS and 16% - 21% for DEFAS.

67. From SD-SO23-720, page 28

.6.10.3 The signals are processed for a 2/4 logic in a Diverse Logic Cabinet.

.6.10.4 With no EFAS and MSIS signals and a DSS Permissive Signal present, the DEFAS signal actuates the same sub group relays as EFAS.

68. From Tech Spec SR 3.8.1.4

SR 3.8.1.4	Verify each day tank contains ≥ 31.5 inches of fuel oil.	31 days

68. From Tech Spec 3.8.3

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air

LCO 3.8.3 The stored diesel fuel oil, lube oil, and starting air subsystem shall be within limits for each required diesel generator (DG).

APPLICABILITY: When associated DG is required to be OPERABLE.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each DG.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more DGs with fuel volume < 48,400 gallons and > 41,800 gallons in storage tank during MODE 1,2,3 or 4.	A.1 Restore fuel oil level to within limits.	48 hours

69. From SO23-3-3, Step 6.5.9

6.5.9 If any part of a surveillance must be completed outside the specified time frame, then immediate notification of the SRO Ops. Supv. is required. In addition, this fact should be documented in the SOI Comments section with the reason and time the surveillance was completed.

70. From SO23-5-1.8, L & S 4.2

4.0 **REACTIVITY AND BORON**

4.1 Boron concentration difference between the RCS and Pressurizer should be maintained at < 50 ppm.

4.2 LIMIT: While in Mode 6, at least 2 Source Range Channels shall be operable, and one channel shall provide audible indication in CNTMT and the Control Room. (Ref. 2.3.3 and Tech. Spec. LCO 3.9.2)

71. From SO123-VII-20.5, Section 6.1.4

6.1.4 **Manager, Health Physics** approval is required, if:

- New ADC for on-site TEDE is > 2,000 mrem
- New ADC for on-site + off-site TEDE is > 3,000 mrem
- Extension involves any quantity other than TEDE (e.g., TODE, LDE, SDE/WB, or SDE/ME)

72. From SO123-VII-20, Attachment 1

CONTAMINATION AREA means an accessible area with general area loose surface contamination levels greater than or equal to 1000 dpm/100 cm^2 beta-gamma activity or greater than or equal to 20 dpm/100 cm^2 alpha activity.

HIGH CONTAMINATION AREA means an accessible area with general area loose surface contamination levels greater than or equal to 150,000 dpm/100 cm^2 beta-gamma distributed activity or greater than or equal to 0.1 μCi hot particle activity.

HIGH RADIATION AREA (HRA) means an accessible area in which an individual could receive 100 mRem deep dose equivalent in 1 hour at 30 centimeters from the source (10CFR20.1003).

RADIATION AREA means an accessible area in which an individual could receive 5 mRem deep dose equivalent in 1 hour at 30 centimeters from the source (10CFR20.1003).

VERY HIGH RADIATION AREA means an accessible area in which an individual could receive 500 rad absorbed deep dose in 1 hour at 1 meter from the source (10CFR20.1003).

73. From SO23-13-21, Step 2.0

2.0 ENTRY CONDITIONS

NOTES
1. A valid fire exists when verbal confirmation of fire is reported to the Control Room.
2. Entry conditions for this AOI are when a valid fire exists within the Protected Area or Switchyard.

2.1 Alarm 61A15 "FIRE DETECTED"

2.2 Alarm 61A11 "FIRE PUMP P-220(E) RUNNING"

2.3 Alarm 61A12 "FIRE PUMP P-221(C) RUNNING"

2.4 Alarm 61A13 "FIRE PUMP P-222(W) RUNNING"

2.5 Local Detection Panel alarm.

2.6 Verbal report of fire or smoke.

73. From SO23-15-61.A1, Annunciator 61A15

61A15 FIRE DETECTED

APPLICABILITY	PRIORITY	REFLASH	ASSOCIATED WINDOWS
Modes ALL	RED	N/A	NONE

INITIATING DEVICE	NOUN NAME	SETPOINT	VALIDATION INSTRUMENT	PMS ID	LINK #
Primary CPU and/or Backup CPU	Fire Computer Central Processing Unit	Smoke or High Temperature Detected	NONE	NONE	1456 [1]

1.0 **REQUIRED ACTIONS**:

1.1 Check the following to verify the fire condition:

- Fire Pump Start
- Fire location on Fire Computer
- Deluge Valves Actuated

73. From INPO OE Event # 362-010203-1

Event Title:	Circuit Breaker Fault Results in Fire, Loss of Off-site Power, Reactor Scram, and Severe Turbine Damage
Event Summary:	On February 3, 2001, San Onofre Nuclear Generating Station Unit 3 was at 39 percent power and was increasing power following a just completed refueling outage when a circuit breaker fault caused a fire, a partial loss of off-site power, and a reactor scram. A subsequent failure of a DC breaker to function properly resulted in the unavailability of the turbine emergency DC lubricating (lube) oil pump to start resulting in extensive turbine-generator damage. The station was in the process of transferring nonsafety-related buses from the reserve auxiliary transformer (RAT) to the unit auxiliary transformer (UAT). At 3:13 p.m., the unit auxiliary feeder breaker (3A0712) from the UAT was closed onto nonsafety-related 4 kV bus 3A07, and the RAT feeder breaker (3A0714) to bus 3A07 automatically opened as designed. Approximately one minute later, the UAT feeder breaker tripped open on overcurrent, and three seconds later a differential relay associated with the UAT tripped, resulting in a generator/turbine trip. Nonsafety-related loads, with the exception of 3A07, on the UAT, including all reactor coolant pumps (RCP), successfully fast transferred back to the RAT. At about the same time, the RAT tripped on phase differential causing the switchyard breakers to open (as designed) resulting in a loss of off-site power to Unit 3. The 6.9 kV RCPs slow transferred to Unit 2, the 4 kV safety-related electrical buses transferred to Unit 2 as designed, and the emergency diesel generators started but were not required to load. The remaining nonsafety-related 4 kV AC loads lost power. The reactor automatically scrammed when the core protection calculator detected low RCP pump speed (all four RCPs slowed to below 95 percent rated speed) and generated a flow-adjusted DNBR signal. The time that elapsed from the beginning of the bus transfer to the scram was approximately 78 seconds. The unit was stabilized in hot standby with the RCPs running, auxiliary feedwater supplying the steam generators, and decay heat being removed through the atmospheric dump valves. Switchgear Fire At 3:15 p.m., the control room received a fire monitoring alarm, and a field report of smoke and flames indicating there was a fire at the 30-foot elevation switchgear room of the turbine building. **The on-site San Onofre fire department (SOFD) responded to the fire and initially used fire extinguishers that were ineffective on the 3A07 switchgear fire. The fire was ultimately extinguished at 6:11 p.m. following the application of water to bus 3A07. The delay in using water to extinguish the fire was due to control room staff concerns that the buses were still energized with 125 VDC and low voltage AC power.** An off-site fire department responded to the site and assisted SOFD. The delay in the use of water had no impact on the consequences of the event. The fire was fully contained within the nonsafety-related cubicle (3A0712), and damage to the plant occurred within the first six minutes of the event. An unusual event was declared at 3:27 p.m. and terminated at 4:20 p.m. Because of communications errors, the control room was informed that the fire was extinguished at 3:44 p.m. Switchgear Circuit Breaker Faults Breaker 3A0712 The apparent cause of the UAT feeder breaker 3A0712 (model 5HK350 manufactured by Brown Boveri) fault was that phase C did not fully close when the breaker was closed onto the bus. This caused overheating that led to arcing and a fire with thick, dark ionized smoke. The arcing damage prevented the breaker from opening and clearing the fault. The station postulated that when the breaker tripped, phases A and B opened but phase C remained partially engaged. The fire consumed many of the breaker's non-metallic parts and caused substantial melting of current carrying components. Consequently, the exact cause of the breaker fault could not be conclusively determined

74. From SO23-14-4, Step 12 Bases

4.0 BASES DESCRIPTION (Continued)

4.4.12 STEP 12 INITIATE Lowering PZR Pressure

Intent

The intent of this step is to establish control of RCS pressure. The general goals associated with RCS pressure control are:

1) Providing subcooling to support the core heat removal process,
2) Minimizing the pressure differential between the S/G and the RCS to minimize the leakage,
3) Deliberately creating a primary-to-secondary differential pressure to establish backflow to control S/G level rise or reduce S/G pressure/temperature, and
4) Controlling RCS pressure below the Main Steam Safety Valve (MSSV) lift pressure to prevent uncontrolled release of radioactivity to the environment.

74. From SO23-12-4, Step 12a

12 INITIATE Lowering PZR Pressure:

NOTE

SGTR depressurization strategy should be to reduce RCS pressure while maintaining RCP NPSH T_C requirements. This strategy should continue until RCS pressure is within 50 PSI of the ruptured S/G pressure or S/G level is not rising.

CAUTION

Keeping RCS pressure higher than S/G pressure is preferred to minimize RCS dilution due to backflow unless backflow is intended.

CAUTION

IF uncontrolled S/G level rise is occurring, THEN reducing RCS pressure to less than 1000 PSIA takes priority over maintaining RCP NPSH or 20°F Core Exit Saturation Margin. In this case stopping RCPs should be evaluated.

a. MAINTAIN RCS pressure requirements of SO23-12-11, Attachment 29, POST-ACCIDENT PRESSURE / TEMPERATURE LIMITS:

1) ESTABLISH RCS pressure:

– low in allowable band for SGTR (approximately equal to ruptured S/G pressure).

AND

– greater than RCP NPSH curve with RCPs running.

AND

– less than 160°F curve.

a. 1) IF RCP NPSH requirements – NOT satisfied,

THEN

a) STOP all RCPs

b) INITIATE Auxiliary Spray

OR

INITIATE FS-32, ESTABLISH Manual Auxiliary Spray.

2) IF all RCPs stopped,

THEN MAINTAIN RCS Pressure above 20°F Saturation Margin curve of SO23-12-11, Attachment 29, POST-ACCIDENT PRESSURE / TEMPERATURE LIMITS.

75. From any Annunciator Response Procedure (generic info)

6.9 When the EOI or AOI Actions have been implemented and verified, then perform the alarm assessment actions in conjunction with the diagnostic analysis in the order of their priority.

6.9.1 Alarm Color Priority (Listed in order of priority of highest to lowest.)

.1 **RED** - A **system priority alarm**. This alarm demands immediate Operator attention. A degradation of system functional capability has occurred. The magnitude of this alarm condition is sufficient to challenge Reactor safety, continued plant operation, or acceptable performance of a major system.

.2 **AMBER** - An **equipment priority alarm**. This alarm demands immediate Operator attention. A degradation of equipment functional capability has occurred. The magnitude of this alarm condition is sufficient to affect the ability of a major system to perform its function (i.e, the alarm condition has a high potential to cause a system priority alarm).

.3 **WHITE** - A **Control Room assessment alarm**. This alarm requires timely Operator attention. Plant conditions can be determined from associated indicators located in the Control Room. The magnitude of this alarm condition will constrain system capability, but it is not expected to cause degradation of the system process.

.4 **BLUE** - A **delegated assessment alarm** (associated indicators are not available in the Control Room to assess plant conditions). This alarm requires the timely dispatch of an Operator, AND confirmation of local plant conditions within 30 minutes. The magnitude of this alarm condition will constrain system capability, but it is not expected to cause degradation of the system process.

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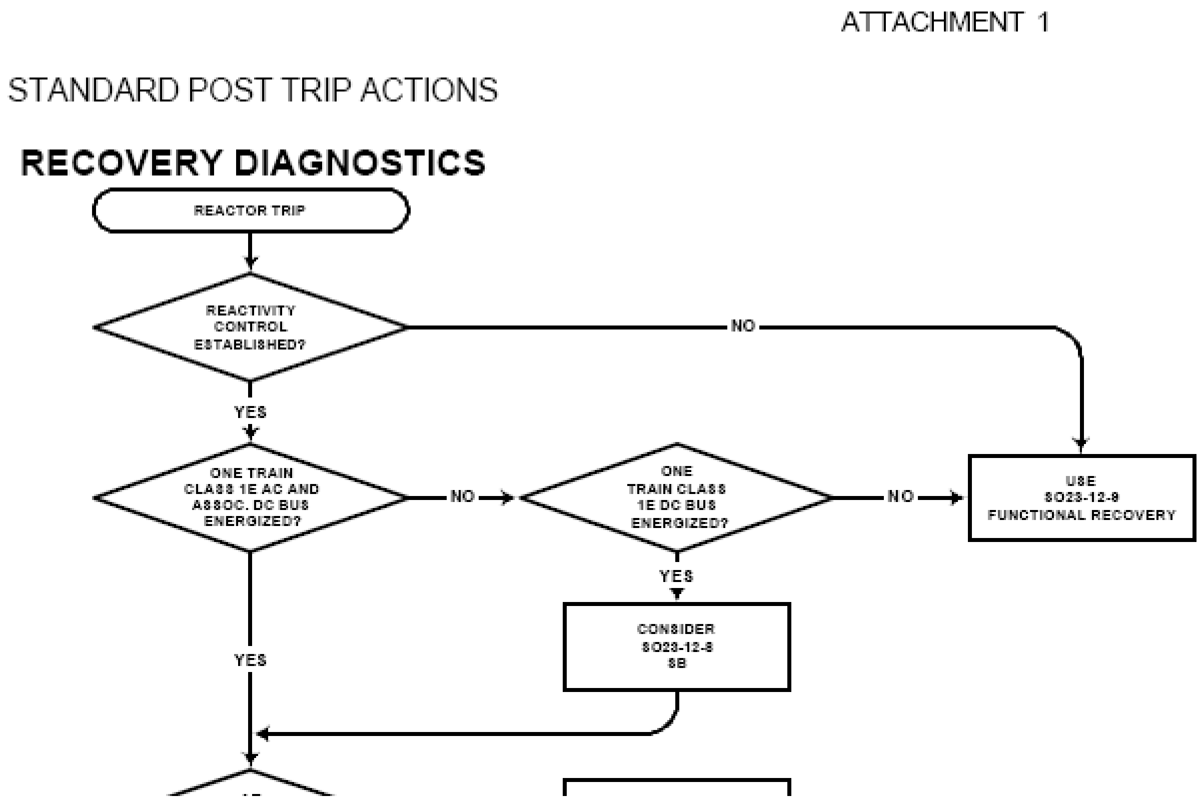
76. From SO23-12-1, Attachment 1



77. From SO23-13-6, Step 3

3 **Subsequent Diagnosis/actions:**

NOTE

If there is no indicated CBO flow, and Vapor Seal Cavity Pressure is low, but other seal parameters are trending normally, then the vapor seal has failed and CBO flow is blowing into Containment. (Tech. Spec. LCO 3.4.13)

☐ a. PERFORM SO23-3-3.37 to determine leakage into Containment.	
☐ b. VERIFY CBO leakage into Containment - ≤ 10 gpm.	☐ b. TRIP the RX. ☐ 1) 5 seconds after CEA rod bottom lights are illuminated, TRIP the affected RCP(s). ☐ 2) GO TO SO23-12-1.
☐ c. VERIFY CBO leakage into Containment - ≤ 4 gpm.	☐ c. INITIATE a controlled Plant Shutdown per SO23-5-1.7. 1) AFTER Reactor is tripped, **AND** CEAs have been inserted 5 seconds, ☐ THEN SECURE the affected RCP(s).

78. From SO23-13-27, Step 3

GUIDELINES:

1) A Pressurizer Pressure signal failure affects the Modulate and Permissive circuits of SBCS in the following way:

   - Channel X or Y high failure could delay the Master Controller response and bring in the permissives early
   - Channel X or Y low failure will delay the response of both controllers

2) See Attachment 1 for the Pressurizer Pressure Control Block Diagram.

3) See Attachment 4 for Pressurizer Pressure Control Diagrams.

4) To diagnose controller alarms, refer to SO23-3-1.10, Attachment for Foxboro Alarm Response and Foxboro Controller Page Data.

5) RCS Reactivity Pressure Coefficient is a positive coefficient and is about one tenth the absolute value of the Moderator Temperature Coefficient.

☐a. VERIFY the selected Pressurizer Pressure channel is between 2225 and 2275 psig and stable.	☐ a. VERIFY the other pressure channel is available by observing PR-0100A or PR-0100B or CFMS page 325. ☐ 1) POSITION HS-0100A, PZR Pressure Channel Select Switch, to the other channel.
☐b. VERIFY Pressurizer Pressure is stable.	b. If Pressurizer pressure is trending **high**, then: ☐ 1) OPERATE Pressurizer Spray in Manual. ☐ 2) SECURE heaters, as necessary. If Pressurizer pressure is trending **low**, then: ☐ 1) START Pressurizer heaters, as necessary. ☐ 2) ENSURE both Pressurizer Spray Valves are closed. ☐ If unable to close affected Spray Valve in manual, then GO TO STEP 3d.
☐c. GO TO Step 3g.	

78. From Tech Spec Section 3.4.1

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS DNB (Pressure, Temperature, and Flow) Limits

LCO 3.4.1 RCS parameters for pressurizer pressure, cold leg temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure $\geq$ 2025 psia and $\leq$ 2275 psia;

b. RCS cold leg temperature (T_c):
   1. For THERMAL POWER less than or equal to 30% RTP, $522°F \leq T_c \leq 558°F$,
   2. For THERMAL POWER greater than 30% RTP, $535°F \leq T_c \leq 558°F$.

c. RCS total flow rate $\geq$ 396,000 gpm.

APPLICABILITY: MODE 1.

79. From SO23-12-4, Step 7

4 INITIATE Lowering RCS T_H to less than 530°F:

NOTE

Lowering RCS T_H below 530°F using BOTH S/Gs is preferred to minimize the possibility of lifting Steam Generator safeties after isolating a Steam Generator.

a. VERIFY both S/Gs available for cooldown.	a. OBTAIN approval of Shift Manager/Operations Leader to allow cooldown using only one S/G.
b. ENSURE one RCP in each loop – stopped.	
c. INITIATE lowering T_H to – less than 530°F using SBCS.	c. INITIATE lowering T_H to – less than 530°F using ADVs.

CAUTION

Failure to reset S/G Low Pressure setpoints during a controlled cooldown will result in MSIS actuation.

d. RESET S/G low pressure setpoint during controlled cooldown.

79. From SO23-12-4, Step 7

7 IDENTIFY Most Affected S/G:

a. EVALUATE S/G radioactive release indications – rising:

1) S/G Blowdown monitors.

2) S/G sample results.

3) Steam line monitors.

b. EVALUATE the following as possible indications of an affected S/G:

1) S/G level – rising when not feeding.

2) S/G feedwater flowrate – significantly mismatched between S/Gs.

3) Steam/feed flow prior to trip – NOT normal.

c. VERIFY most affected S/G – identified.

c. IF both S/Gs affected,

THEN VERIFY S/G with highest activity – identified.

d. NOTIFY Shift Manager/Operations Leader most affected S/G identified.

79. From SO23-14-4, Step 7

4.4.7 STEP 7 IDENTIFY Most Affected S/G

Intent
The intent of this step is to determine which S/G is most affected by a tube rupture.

.1 NOTE prior to Step 7d.

This NOTE explains the general concept of partitioning of a S/G. Establishing S/G level greater than or equal to 40% NR will aid in the reduction of iodine in any effluent until the S/G is isolated. Overall the intent is to keep the tubes covered without violating Technical Specification for cooldown.

Method
Step a.: To minimize radiological releases from the RCS to the environment, the most affected S/G (which has the highest radiological release rates) must be identified before it can be isolated. All relevant indications should be evaluated to determine the most affected S/G. This is done by reviewing the readings of Main Steam Line Monitors, S/G Blowdown Monitors and taking S/G liquid and/or steam samples. Trends both prior to and after the trip should be considered along with known pre-existing leakage.

Step b.: Unexplained changes in S/G feedwater flow rate or S/G level increases can also provide indication of the affected S/G. Automatic feedwater modulation may mask the expected S/G level increase due to a SGTR. (Ref. EPG Supplementary Information, Item 4)

Step c.: This step identifies the *most affected S/G* based upon the information obtained in steps a. and b.

80. From SO23-6-15, Section 6.6

6.6 Abnormal Operation

INFORMATION USE

PROBLEM	ACTION
High Bus Voltage	Ensure associated Battery Charger on FLOAT <u>and</u> request maintenance on the charger.
Low Bus Voltage	Request Maintenance on the associated Battery Charger. If appropriate, then the equalizing circuit may be placed in service.
Loss of the Switchyard Charger	Strip Switchyard DC loads per SO23-6-30, Section for Disconnecting Non-Critical Loads from the 125 VDC Switchyard Battery, <u>and</u> evaluate transferring to the Alternate Supply.
Non 1E power is lost to B005, Battery Charger	Perform Attachment 8.

80. From SO23-3-1.7, Attachment 8

<u>TRANSFERRING 6.9 KV SUPPLY BREAKER CONTROL POWER SUPPLIES BETWEEN UNITS</u>

CONTINUOUS USE

OBJECTIVE:

To transfer DC Control Power for the three 6.9 KV Bus Feeder Breakers (Unit Aux Xfmr XU2, Bus-tie-to-opposite-Unit, and Res Aux Xfmr XR3) between normal source (opposite Unit) and the alternate source (same Unit) depending on power source availability. For Units in Mode 1-4 LCS 3.8.100 will be entered during transfer due to Backup Breaker Protection Devices being solid. [A diagram of the transfer switches is provided in Attachment 9.]

80. From SO23-15-63.A32

63A32 2D1 125 VDC BUS TROUBLE

APPLICABILITY	PRIORITY	REFLASH	ASSOCIATED WINDOWS
Modes ALL	AMBER	YES	63A52

INITIATING DEVICE	NOUN NAME	SETPOINT	VALIDATION INSTRUMENT	PMS ID	LINK #
59 Relay	Bus Overvoltage	147.2 VDC	NONE	EY8191	1903 1904 1905
27 Relay	Bus Undervoltage	118.2 VDC			
64 Relay	Bus Ground	25 ± 10K OHMS [1]			

1.0 **REQUIRED ACTIONS**:

1.1 Dispatch an Operator to the 2D1 Battery Charger Room.

2.0 CORRECTIVE ACTIONS:

SPECIFIC CAUSES	SPECIFIC CORRECTIVE ACTIONS
2.1 Battery Charger Malfunction	2.1 Refer to SO23-6-15, Section for Abnormal Operation.
2.2 DC Ground	2.2 Refer to SO23-6-33, Section for Ground Isolation.

3.0 ASSOCIATED RESPONSES:

3.1 Notify the CRS/SM and the STA to review Tech. Specs. LCO 3.8.4, LCO 3.8.5, LCO 3.8.9, LCO 3.8.10 and initiate an EDMR/LCOAR, as required.

4.0 COMPENSATORY ACTIONS:

DEVICE NUMBER	SPECIFIC COMPENSATORY ACTIONS
4.1 2D1 bus voltage and % ground	4.1 Monitor 2D1 bus voltage and ground condition at least twice per shift.

[1] Ground detector is located on the DC Bus panel. A ground condition exists when the positive or negative ground LED light is solidly ILLUMINATED.

81. From SO23-12-6, Step 10

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10 ESTABLISH Condensate Pump flow to available S/Gs:	
a. VERIFY at least one Condensate Pump from either Unit – available.	a. GO TO SO23-12-9, FUNCTIONAL RECOVERY AND INITIATE SO23-12-9, Attachment FR-5, RECOVERY - HEAT REMOVAL success path HR-1 step 7 immediately.
b. ENSURE Full Flow Condensate Polishing Demineralizers – bypassed: FV-4902A – open HV-4900A – closed HV-4900B – closed.	
c. UNLOCK and INITIATE OPENING 1305MU024, MFW Pump Bypass.	c. OPEN MFW Pump discharge valves.
d. ADJUST FIC-3294, Condensate Pump miniflow controller to – 3000 GPM.	
e. SELECT MFW Regulator Bypass valve controllers to MANUAL.	e. Locally operate MFW Regulator Bypass valves per *SO23-9-6, FEEDWATER CONTROL SYSTEM OPERATION*.
f. ENSURE MFW Block valves – closed. E-088: HV-4047 E-089: HV-4051	
g. ENSURE MFW Regulator Bypass valves – closed. E-088: HV-1106 E-089: HV-1105	

81. From SO23-12-6, Step 10

h. INITIATE the following:

SIAS
CCAS

i. OVERRIDE and operate Charging Pumps as necessary to – maintain PZR level.

j. VERIFY Boration in progress – at greater than or equal to 40 GPM.

j. ENSURE Shutdown Margin established – greater than 5.15% $\Delta K/K$

CAUTION

Steaming the available S/G dry could result in excessive thermal stresses in the tubes and possible tube damage when cool feedwater is added. In the event that both S/Gs do become dry, feed should be restored to only **one** S/G when reinitiating core cooling.

CAUTION

IF S/G dryout occurs, THEN S/G pressure will rapidly drop and MSIS will initiate. Failure to reset S/G low Pressure setpoints during a controlled cooldown will result in MSIS actuation and a loss of the Main Feedwater flowpath.

k. ADJUST available S/G steaming rate to initiate lowering S/G pressure – less than 500 PSIA:

1) RESET MSIS setpoint as controlled cooldown proceeds.

2) MAINTAIN available S/G steaming rates to control RCS temperature within the following limits:

a) Core Exit Saturation Margin – between 20°F and 160°F:

QSPDS page 611
CFMS page 311.

82. From SO23-13-11, Entry Conditions

EMERGENCY BORATION OF THE RCS / INADVERTENT DILUTION OR BORATION

PURPOSE

To provide guidance for Emergency Boration of the RCS, and mitigating the effects of an inadvertent dilution or inadvertent Boration event.

ENTRY CONDITIONS

1. For EMERGENCY BORATION:
   - >1 full length CEA - NOT fully inserted following RX Trip
   - With RX Critical, 1 or more CEA Regulating Group(s) below PDIL
   - SDM <5.15% $\Delta K/K$ when T_{AVE} >200°F
   - SDM < 3.5% $\Delta K/K$ when T_{AVE} ≤200°F (AR 030500877-3)
   - Uncontrolled Cooldown resulting in RCS T_{AVE} more than 25°F below T_{REF}

     OR
   - During refueling:

     K_{eff} >0.95 or
     Boron Conc. <2600 ppm

2. For An INADVERTENT DILUTION EVENT:
   - Unexplained rise in RX Power
   - Unexplained rise in RCS Temperature
   - Unexpected lowering of RCS Boron concentration
   - Unexplained increase in countrate when RX is Shutdown

3. For An INADVERTENT BORATION EVENT:
   - Unexplained lowering of RX Power
   - Unexplained lowering in RCS Temperature
   - Unexpected rise in RCS Boron concentration

82. From SO23-13-11, Steps 2b & 2j

2 **Emergency Boration Actions:**

☐ a. VERIFY Refueling NOT in progress.	☐ a. ENSURE operations involving core alterations or positive reactivity changes are suspended.
☐ b. VERIFY at least one Charging Pump is available.	☐ b. ENSURE RCS Pressure is <1450 psia, AND ☐ INITIATE Boration at >40 gpm using the Operable Boration Flowpath, (RWSTs and HPSI Pump). ☐ 1) GO TO Step 2j.
j. VERIFY Boric Acid delivery to RCS by monitoring: ☐ 1) RCS Temperature lowering when at power. ☐ 2) Boron concentration indicated on Boronometer rising. ☐ 3) BAMU Tank level lowering. LI-0208B (MT-072) LI-0206B (MT-071) ☐ 4) Increased Boron concentration confirmed by RCS sample.	☐ j. 1) <u>IF</u> using Charging Pump(s), <u>THEN</u> TRANSFER suction to the RWSTs. ☐ 2) <u>IF</u> using HPSI Pump, <u>THEN</u> Verify proper Boration Flowpath alignment.

83. From Tech Spec 3.1.5 and Bases

3.1.5 Control Element Assembly (CEA) Alignment

LCO 3.1.5 All full length CEAs shall be OPERABLE and all full and part length CEAs shall be aligned to within 7 inches of all other CEAs in its group.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One regulating CEA trippable and misaligned from its group by > 7 inches.	A.1 Initiate THERMAL POWER reduction in accordance with COLR requirements.	15 minutes
	AND	
	A.2.1 Restore the misaligned CEA(s) to within 7 inches of its group.	2 hours
	OR	

In the case of the full length CEA drop, a prompt decrease in core average power and a distortion in radial power are initially produced, which when conservatively coupled result in local power and heat flux increases, and a decrease in DNBR. For plant operation within the DNBR and local power density (LPD) LCOs, DNBR and LPD trips can normally be avoided on a dropped CEA.

83. From SO23-13-13, Step 2

2 **COMMENCE plant load reduction:**

a. Within 15 minutes of discovery, INITIATE RX power reduction in accordance with the table below. (Ref. LCO 3.1.5 and LCS 3.1.105)

☐ 1) COMMENCE LOWERING Turbine Generator load while maintaining T_{cold} per SO23-5-1.7.

✔	TYPE OF CEA	60 MINUTE POWER REDUCTION REQUIREMENT	120 MINUTE POWER REDUCTION REQUIREMENT
	Non-group 6 Full Length	10%	15%
	Group 6 Full Length	5%	10%
	Part Length Initially ≥ 112.5 Inches Withdrawn	None	None
	Part Length Initially < 112.5 Inches Withdrawn	2%	5%

84. From Tech Spec 3.9.3

3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

LCO 3.9.3 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place by four bolts;

--------------------------NOTE------------------------------

The equipment hatch may be open if all of the following conditions are met:

1) The Containment Structure Equipment Hatch Shield Doors are capable of being closed within 30 minutes,
2) The plant is in Mode 6 with at least 23 feet of water above the reactor vessel flange,
3) A designated crew is available to close the Containment Structure Equipment Hatch Shield Doors,
4) Containment purge is in service, and
5) The reactor has been subcritical for at least 72 hours.

----

b. One door in each air lock closed;

--------------------------NOTE------------------------------

Both doors of the containment personnel airlock may be open provided:

a. one personnel airlock door is OPERABLE, and

b1. the plant is in MODE 6 with 23 feet of water above the fuel in the reactor vessel, or

b2. defueled configuration with fuel in containment (i.e., fuel in refueling machine or upender).

----

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by an OPERABLE Containment Purge System.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

84. From Tech Spec 3.9.3 Bases

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of shutdown when containment

closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed or operable. Operability of the containment personnel airlock door requires that the door is capable of being closed; that the door is unblocked and no cables or hoses are being run through the airlock; and that a designated individual is continuously available to close the airlock door. This individual must be stationed at the outer airlock door.

The use of temporary ramps for equipment access through the containment personnel air lock doors is acceptable during CORE ALTERATIONS or moving of irradiated fuel within containment. These ramps do not impede closure of the containment personnel airlock doors as the ramps are quickly removed by the designated individual stationed at the outer door. Removal of the ramps is a normal function of door closure, and the ability of plant personnel to close the personnel airlock, if needed, is not compromised by the ramps. Similarly, door seal covers may be used, provided they are removed prior to air lock door closure.

84. From INPO OE Event # 362-950826-1

Event Title:	Breach of Containment Integrity during Refueling
Event Summary:	On August 26, 1995, with Unit 3 in a refueling outage, personnel determined that containment integrity had not been maintained while core alterations occurred. A work authorization for steam generator work had been implemented that involved venting a steam generator through an open atmospheric dump valve and opening a steam generator drain line to a sump outside containment. The work authorization was reviewed with a system walkdown by a work control representative to check for any open handholes or manways potentially not addressed by the work authorization. After the work began, other personnel then noted some open drain lines on AFW piping that would have compromised containment integrity. The individual performing the review of the work authorization was provided inadequate instructions and did not check the vent valves. The vent valves were also listed as open on the work authorization. Core alterations occurred while the vent path existed. This event is not significant because the vent lines involved are small diameter, the vent path only existed for 13 minutes, and there was no containment pressurization potential during this time period.
Event Number:	362-950826-1
Event Date:	08/26/1995

85. From SO23-12-7, Step 14

CAUTION

Applicable Pressure/Temperature Limits from SO23-12-11, Attachment 29, POST-ACCIDENT PRESSURE / TEMPERATURE LIMITS have precedence over natural circulation strategies that minimize Reactor Vessel Upper Head void development.

14 INITIATE Plant Cooldown:

a. MAINTAIN CEDM Cooling – operating.	a. GO TO Step b.
b. MAINTAIN Reactor Vessel Head Saturation Margin – greater than 0°F: QSPDS page 611 CFMS page 311.	b. CONTROL PZR pressure to maintain Reactor Vessel level – greater than or equal to 100% (Plenum): QSPDS page 622 CFMS page 312 SO23-12-11, Attachment 4.
c. INITIATE SO23-12-11, Attachment 3, COOLDOWN / DEPRESSURIZATION.	

85. From SO23-12-2, Step 9 (Why Distractor A is wrong)

OPERATOR ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9 VERIFY Desired Plant Status:	
a. INITIATE *SO23-3-2.22, ESFAS OPERATION*, to reset signals and realign plant systems	
b. GO TO *SO23-5-1.4, PLANT SHUTDOWN TO HOT STANDBY.*	

86. From SO23-13-14, Steps 1 & 2

1 **Initial Actions:**

EVALUATE plant conditions against the following to Identify leak location and Procedural Steps to perform:

POSSIBLE PLANT CONDITIONS	LEAK LOCATION	DIRECTION
• Unidentified RCS leakrate ≥ 1 gpm • Identified RCS leakrate ≥ 10 gpm • Charging flow > Letdown flow with plant conditions stable • VCT level lowering • Containment Sump inlet flow ≥ 1 gpm on the CFMS • RCDT inlet flow high alarm on CFMS ≥ 5 gpm • 57C10, CONTAINMENT RADIATION HI, Illuminated • 57C20, RCS LEAKAGE DETECTION ACTIVITY HI, Illuminated • 57C43, RCS LEAKAGE ABNORMAL/RECIRC SYS VV MISALIGNED, Illuminated (Mode 1-4 only)	RCS	☐ **GO TO STEP 2**

2 **RCS Leak**

☐ a. VERIFY Pressurizer level - NOT LOWERING.	☐ a. START Charging Pumps to maintain Pressurizer level.
☐ b. VERIFY Purge is not in service.	☐ b. 1) MANUALLY INITIATE CPIS. AND 2) MANUALLY INITIATE one train of CRIS.

87. From SO23-3-2.11, Attachment 16, Steps 2.1 & 2.2

2.0 PROCEDURE

2.1 **Requirements for Movement of Loads > 125 lbs. and ≤1432 lbs. over the Spent Fuel Racks:**

2.1.1 **Load** to be moved over the fuel assemblies weighs ≤2000 pounds OR is a SFP gate. (LCS 3.9.104)

☐ Yes ☐ No ________

2.1.2 At least **ONE** PACU (ME-370 and/or ME-371) is OPERABLE. (LCS 3.7.118) [LS-4.2]

☐ Yes ☐ No ________

.1 If the Operable PACU Unit is required to be in operation, then ensure it is able to be powered from an operable Diesel Generator. (Mark N/A if PACU is not required to be in operation.) ________

2.2 **Requirements for Movement of Loads > 1432 lbs. Over the Spent Fuel Racks:**

2.2.1 **Load** to be moved over the fuel assemblies weighs ≤2000 pounds OR is a SFP gate. (LCS 3.9.104)

☐ Yes ☐ No ________

2.2.2 **Select** One condition that meets PACUs (ME-370 and ME-371) and Radiation Monitors (RI-7822 and RI-7823) with associated FHIS Logic Operability requirements: (LCS 3.3.112 and 3.7.118) [LS-1.2 and LS-4.2] ________

✔	RAD MON w/FHIS OPERABLE	PACUs OPERABLE	REQUIREMENTS FOR MOVING LOADS >1432 lbs. OVER THE SF RACKS
	2 or 1	1	Minimum Requirement-No Action
	1	1	Components on Opposite Trains PACU IN ISOLATE
	0	1	PACU IN ISOLATE

.1 If the Operable PACU Unit is required to be in operation, then ensure it is able to be powered from an operable Diesel Generator. (Mark N/A if PACU is not required to be in operation.) ________

87. From Plant Systems LCS 3.7.118

3.7 PLANT SYSTEMS

LCS 3.7.118 Fuel Handling Building Post-Accident Cleanup Filter System

One Fuel Handling Building Post-Accident Cleanup Filter System train shall be OPERABLE.

VALIDITY STATEMENT: Revisions 0 and 2, effective 12/05/06, to be implemented within 30 days

APPLICABILITY: During movement of irradiated fuel assemblies in the fuel building or operation of the Spent Fuel Handling Machine with a load, > 125 pounds (includes 35 pounds for weight of hook and block) over the Spent Fuel Pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Two Fuel Handling Building Post-Accident Cleanup Filter System trains inoperable.	A.1 Suspend movement of irradiated fuel assemblies in the fuel building and operation of the Spent Fuel Handling Machine with a load > 125 pounds over the Spent Fuel Pool.	1 hour
B. Required Action and/or associated Completion Time of Condition A not met.	B.1 Perform a Cause Evaluation	Within the time specified by the controlling site procedure

SURVEILLANCE		FREQUENCY
SR 3.7.118.1	Operate each Fuel Handling Building Post-Accident Cleanup Filter System train for ≥ 10 continuous hours with the heaters operating.	31 days
SR 3.7.118.2	Verify each Fuel Handling Building Post-Accident Cleanup Filter System train actuates on an actual or simulated actuation signal.	24 months

88. From SO23-12-11, FS-7

a. VERIFY at least one S/G operating: 1) SBCS – available OR ADV – available. AND 2) Feedwater – available.	a. GO TO *SO23-12-9, FUNCTIONAL RECOVERY* AND INITIATE SO23-12-9, Attachment FR-5, RECOVERY – HEAT REMOVAL.
b. VERIFY PZR level – greater than 30% AND – NOT lowering. c. VERIFY Core Exit Saturation Margin – greater than or equal to 20°F: QSPDS page 611 CFMS page 311. d. VERIFY Reactor Vessel level – greater than or equal to 100% (Plenum): QSPDS page 622 CFMS page 312 Attachment 4.	○ IF any criteria of steps b. through d. – NOT satisfied, THEN • OPERATE Charging and SI systems as necessary to maintain Throttle/Stop criteria – satisfied. • THROTTLE Loop Injection valves – as required. • ENSURE auxiliaries to SI Pumps: a) Electrical power to pumps and valves. b) Proper system alignment. c) CCW flow. d) HVAC.

88. From SO23-12-11, FS-7

e. RCS Cooldown – NOT in progress.	e. MAINTAIN Boration – at least 40 GPM.
f. VERIFY SI Pumps – NOT operating per SO23-12-9, Attachment FR-1, RECOVERY – REACTIVITY CONTROL, to meet RC-3 Success Path.	f. GO to step h.
g. THROTTLE OR STOP SI Pumps as required – one train at a time.	
h. VERIFY Charging Pumps – NOT operating per SO23-12-9, Attachment FR-1, RECOVERY – REACTIVITY CONTROL, to meet RC-2 Success Path	h. GO to step k.
i. VERIFY PZR Level – less than 80%.	i. 1) INITIATE FS-31, ESTABLISH CVCS Letdown Flow. 2) INITIATE FS-33, MONITOR RCS Solid Operation.
j. STOP Charging Pumps as required one at a time.	
k. MAINTAIN criteria of steps a. through e. – satisfied.	

89. From Tech Spec 3.8.5 Bases

B 3.8.5 DC Sources — Shutdown

BASES

BACKGROUND	A description of the DC sources is provided in the Bases for LCO 3.8.4, "DC Sources — Operating."
APPLICABLE SAFETY ANALYSES	The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume that Engineered Safety Feature (ESF) systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the DG control system, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum DC electrical power sources during MODES 5 and 6, ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

The DC sources satisfy Criterion 3 of the NRC Policy Statement.

90. From SO23-12-11, Attachment 8

RESTORATION OF OFFSITE POWER

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1 VERIFY 220kV Switchyard status:	
a. VERIFY annunciators for Reserve Auxiliary Transformers – reset: 63C11 RES XFMR XR1 PROTECTION TRIP 63C21 RES XFMR XR2 PROTECTION TRIP 63C31 RES XFMR XR3 PROTECTION TRIP **Unit 2** ❑ **Unit 3** ❑	a. IF any Reserve Auxiliary Transformer relayed, THEN INITIATE *SO23-6-6, RESERVE AUXILIARY TRANSFORMER OPERATION*, Attachment for Emergency Faulted Reserve Auxiliary Transformer Operations OR INITIATE removing Generator Iso-phase bus manual disconnects to allow use of Unit Auxiliary Transformers per *SO23-6-5, MAIN AND AUXILIARY TRANSFORMER OPERATION*.

91. From SO23-8-15, L & S 1.3

1.3 Failure of all operating Continuous Exhaust Fans will cause 2/3FV-7202, Waste Gas Decay Tank Header Vent Valve, to Close.

1.3.1 If one operating Continuous Exhaust Fan fails (leaving only one in service), then the release will NOT automatically isolate.

91. From SO23-8-15, L & S 4.5 & 4.6

4.5 High radiation signal from 2/3RE-7808G or 3RE-7865-1 will cause automatic closure of 2/3FV-7202 **and** termination of the Waste Gas Release.

4.5.1 3RE-7865-1 *High* Alarm will actuate an alarm in the State Offices of Emergency Services. However, the release should terminate when the *Alert* setpoint is reached.

4.6 A close signal is sent to 2/3FV-7202 when any Waste Gas release monitor fails (2/3RIC-7808G, 2RT-7865-1 or 3RT-7865-1). When a monitor is going to be used for a release, and another monitor is failed, then 2/3FV-7202 will close as soon as it is opened to commence the release. Releases can be made with a failed monitor by the doing the following:

- A release using 2/3RIC-7808G can be made by aligning the failed RT-7865 to the Containment Purge stack, providing a purge is not in progress.
- A release using 3RT-7865-1 can be made by installing a jumper around the contact in 2/3RT-7808G that terminates the release upon failure.

91. From SO23-8-15, L & S 4.2

4.2 Due to inadequate mixing in the PVS plenum, the ODCM only takes credit for 3RT-7865-1 and 2/3RIC-7808G to monitor Waste Gas Tank releases. 2RT-7865-1 is not to be used to monitor Waste Gas Tank Releases.

92. From SO123-0-A7, Attachment 1

EVENT	ATT/STEP(S)/ DOCUMENT	TIME
FOUR HOUR REPORTS		
Unit Trip, Reactor Trip, or Load Change	Att 2, Step 1.2 Att 2, Step 9.1.1 SOB-012, SOB-085	N/A N/A N/A
Reactor Protection System Actuation <u>when the Reactor is Critical</u>	Att 5, Step 1.2 Att 5, Section 3.0	4 HR N/A
Reactor Protection System Actuation <u>when the Reactor is not critical</u>	Att 5, Step 2.1 Att 5, Section 3.0	8 HR N/A
ECCS Injection into the RCS with Valid Signal	Att 5, Step 1.1	4 HR
News Release or Government Agency Notification Required	Att 3, Step 2.2.2 Att 8, Step 1.1	4 HR 4 HR
Loss, Theft, or Missing Licensed Material a. Quantities greater than or equal to 1000 times the quantity specified in 10CFR20 Appendix C *where exposure could result.* b. After 30 days that Licensed Material in quantities greater than 10 times the quantity specified in 10CFR20 Appendix C is still missing.	 Att 7, Step 3.1 Att 7, Step 3.2	4 HR
Subsequent recovery of previously reported Lost, Stolen, or Missing Licensed Material	Att 7, Step 3.3	4 HR
Threatened or Endangered Species found dead or requiring human assistance to leave the Plant side OCA, Parking Lot 2, and/or Parking Lot 3	Att 8, Step 1.1	4 HR
Personnel Injury	Att 2, Step 1.2.12 Att 2, Step 1.4 Att 2, Step 2.1.3 Att 3, Step 2.2.2 Att 3, Step 3.1.3 Att 7, Step 4.1 SO123-XVI-30 SOB-012, SOB-085	N/A N/A N/A 4 HR 8 HR 8 HR 24 HR N/A

END OF 4 HOUR REPORTS

92. From SO123-0-A7, Attachment 5

1.0 FOUR HOUR NOTIFICATIONS

GUIDELINE
The Emergency Plan should be reviewed for possible Emergency Event classification for bracketed [ ] steps.

[1.1] Any event during the past three (3) years that results or should have resulted in Emergency Core Cooling System (ECCS) discharge into the Reactor Coolant System as a result of a VALID SIGNAL except when the actuation results from and is part of a pre-planned sequence during testing or Reactor Operations. [10CFR50.72(b)(2)(iv)(A)]

1.1.1 Safety Injection Tank discharge into the Reactor Coolant System due to an actual low pressure condition in the Reactor Coolant System is reportable. (Ref. 2.1.11)

1.2 Any event or condition that results in actuation of the REACTOR PROTECTIVE SYSTEM (RPS) *when the Reactor is Critical* except when the actuation results from and is part of a pre-planned sequence during testing or Reactor Operation. [10CFR50.72(b)(2)(iv)(B)]

92. From SO123-0-A7, Step 6.3

6.3 NRC Telephone Notification Guidelines

6.3.1 Notify NRC Operations Center as soon as possible, and in all cases, within one, four, eight, or twenty four hours (as applicable) by telephone of the occurrence of any event listed in Attachments 3 through 8. (10CFR50.72)

93. From SO23-13-9, Step 1

1 **Determine requirements for Plant load reduction:**

☐ a. PLACE LV-3245, Condensate Drawoff Valve, to DISABLE.

☐ b. VERIFY FFCPD in service.

1 **Determine requirements for Plant load reduction:** (continued)

☐ c. INITIATE Attachment 1.

☐ d. CONTACT Chemistry to commence monitoring ***feedwater and Steam Generator*** chemistry parameters.

☐ e. VERIFY ***Condensate*** Cation Conductivity **< 1.5 μS/cm.**	☐ e. GO TO Step 1j.

1 **Determine requirements for Plant load reduction:** (continued)

☐ j. VERIFY ***Condensate*** Cation Conductivity **< 5.0 μS/cm.**	☐ j. GO TO Step 1o.

1 **Determine requirements for Plant load reduction:** (continued)

o. VERIFY all of the following:	☐ o. GO TO Step 2.
☐ 1) ***Condensate*** Cation Conductivity **< 10.0 μS/cm**	
☐ 2) Steam Generator sodium concentration **< 250 ppb**	
☐ 3) Steam Generator cation conductivity **< 4 μS/cm**	
☐ p. VERIFY pump adjacent to affected quadrant is running.	p. **Based on the FFCPD Outlet Conductivity and trend,** Shift Manager will direct:

93. From SO23-13-9, Step 2

2 Commence immediate Plant load reduction:

☐ a. VERIFY pump adjacent to affected quadrant is running.	☐ a. 1) TRIP the reactor. ☐ 2) STOP affected quadrant Circulating Water Pump. ☐ 3) PERFORM SO23-12-1 ☐ 4) GO TO Step 2e.
☐ b. VERIFY Heat Treat of the Circulating Water System is NOT in progress.	☐ b. INITIATE FULLY CLOSING Gate 6. 1) WHEN Main Condenser vacuum (HP Zone) is $\leq$ 6" Hg, ☐ THEN GO TO Step 2c.

☐ c. STOP affected quadrant Circulating Water Pump.

☐ d. COMMENCE a rapid downpower to $\leq$ 30% power.

☐ e. CLOSE affected Condenser Quadrant Air Ejector Suction Valve per Attachment 2.

☐ f. COMMENCE Overboarding the affected hotwell per Step 3.

93. From SO23-13-9, L & S 1.4

1.4 The Condensate Pumps divide the required flow of the Condensate System equally to ensure NPSH to the Feedwater Pumps. At full power, this means that each Condensate Pump provides 5000 GPM (four pumps running) or 6600 GPM (three pumps running). If the affected Circulating Water Pump is running, then automatic overboarding starves the Condensate System of the flow of the associated Condensate Pump. The Condenser Makeup System (in AUTO or quickfill) cannot fill the Condenser faster than it is being drained unless the Unit is at low power. The makeup system can provide 3000 GPM maximum to fill the Condenser. If the affected Circulating Water Pump is running and the associated quadrant is being overboarded, then Unit load must be adjusted accordingly to prevent tripping the Feedwater Pumps on low NPSH.

1.5 If the FFCPD is not in service, then the affected quadrant Circulating Water Pump must be stopped expeditiously and the Condenser overboarded.

94. From Tech Spec SR 3.0.3 Bases

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not

SR 3.0.3 (continued)

been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified frequency , whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides an adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10CFR50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, SR 3.0.3 allows for the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

SR 3.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

95. From SO123-0-A4, Step 6.5.1

6.5 Status Control Forms

6.5.1 The purpose of a Status Control Form (Attachment 1) is to provide a method to track temporary changes to alignments of plant equipment, in support of MAINTENANCE ACTIVITIES, or to document performance when using approved procedures, including AOIs or EOIs.

.1 If supporting MAINTENANCE ACTIVITIES or documenting performance when using approved procedures, and the equipment will be in the alternate alignment for ≥ 90 days, then a 10CFR50.59 Review is required to be performed.

96. From SO123-XX-5, Part A, Step 6.3.5.3

6.3.5 **Capability Limitation**:

.1 *IF* the work involves any of the critical components listed in Attachment 6, *THEN* enter *"Critical Component"* in this section if the Unit is online.

.2 Identification of any abnormal alignments that will be required to accomplish the work described on the WAR (*e.g., equipment isolated and bypassed, downstream block valve closed to allow valve stroking, rod control in manual*).

.2.1 Review Defense In Depth Planning Sheets.

.3 **(Units 2/3 Only) During RCS Reduced Inventory Conditions (RIC) or Shut Down Cooling (SDC) operation**, any work that could result in Reactor Coolant System (RCS) perturbations (including makeup capabilities, decay heat removal capabilities, level or temperature indication, or RCS breaches) **SHALL** be annotated in the Capability Limitation Section of the WAR, and the WAR **SHALL** be approved by the Manager, Plant Operations or designee and documented on the WAR.

.4 *If* the MO contains contingency steps (*i.e., IF / THEN statements*) in the work plan, *then* ensure the evaluation addresses the potential impact of contingency step implementation.

97. From SO23-8-15, Attachment 4

DETERMINATION OF CURRENT WEATHER CONDITIONS

CONTINUOUS USE

OBJECTIVE:

Determine which way the wind is blowing. It is preferable to discharge radioactive gases over the ocean, away from populated areas. If the wind is blowing *from* the ocean *towards* the land, then a dispersion factor (χ/Q) is calculated to confirm that it is less than the historical dispersion factor of 4.8E-6. The wind direction and the dispersion factor may be calculated using the Plant Computer or manually. If the wind is blowing towards land, and (χ/Q) is higher than 4.8E-6, then the release should be delayed until weather conditions are favorable. (LS-3.1)

2.1 Determine the following from the Plant Computer Main Menu by selecting USER FUNCTIONS, then WEATHER: ________

✔	WIND DIRECTION, χ/Q VALUE	PERFORM THE FOLLOWING
	Wind direction is DESIRABLE	On the Gaseous Effluent Release Permit, Mark Release Condition **DESIRABLE** and Mark N/A the current χ/Q value. Proceed with the release.
	$\chi/Q \leq$ 4.8E-6 - Conditions are DESIRABLE	Proceed with the release. Record χ/Q value on the Release Permit.
	$\chi/Q >$ 4.8E-6 - Conditions are UNDESIRABLE at this time	When weather conditions improve, then perform this Attachment again.
	$\chi/Q >$ 4.8E-6 - Conditions are UNDESIRABLE but the Shift Manager has determined the release cannot be delayed due to plant conditions	Proceed with the release **AND** State reason on Release Permit. Record χ/Q value on the Release Permit.

98. From EPIP Form EP (123) 3, Emergency Exposure Authorization

EMERGENCY EXPOSURE AUTHORIZATION

During a declared emergency, lifesaving, or plantsaving activities to protect the public health and safety may be required. In these situations, if you are a volunteer to such an activity, regulatory guidance allows an exposure to radiation higher than typical SONGS or NRC limits. If you volunteer please provide the information requested in Section 1 and read Sections 2 and 3.

1. Name ______________________________ Social Security No.

   Date ______________________ Film/TLD ______________________ Age

If the need for an emergency exposure is identified, the following criteria will be considered when selecting a volunteer: 1. the individual must be a volunteer, 2. if you are declared pregnant female others may be selected before you, and 3. persons over the age of 45 may be selected before those under the age of 45.

2. Your exposure may not exceed the following Emergency Exposure Guidelines (EPA-400):

   Protecting Valuable Property - 10 Rem Total Effective Dose Equivalent (TEDE)
   Lifesaving Activities - no upper limit
   Protection of Large Populations - no upper limit

3. Review the following information on effects of acute radiation exposures. If you have any questions, please contact a representative from Health Physics.

0-50 Rem	No apparent effect, except possibly minor blood changes.
80-120 Rem	Vomiting and nausea for about one day in five to ten percent of exposed personnel. Fatigue, but not serious disability.
130-170 Rem	Vomiting and nausea for about one day, followed by other symptoms of radiation sickness in about 25 percent of personnel. In rare instances death may occur.
270-330 Rem	Vomiting and nausea in nearly all personnel on first day, followed by other symptoms of radiation sickness. About 20 percent death rate within two to six weeks; survivors convalescent for about three months.
400-500 Rem	Vomiting and nausea in all personnel on first day, followed by other symptoms of radiation sickness. About 50 percent death rate within one month; survivors convalescent for about six months.

99. From SO123-VIII-10, Step 6.8

6.8 Event Reclassification or Change in PAR

NOTES: (1) Reclassification or change in PAR requires a new set of notifications per Table 2 (Step 6.3.1.)

(2) If an increase in classification occurs within 15 minutes of the previous classification, it is acceptable to provide notification for the second condition only, **provided** that the notification can be initiated within 15 minutes of the initial event.

6.8.1 When conditions indicate the need for a possible reclassification, review all applicable event categories and ensure the event is reclassified to the highest applicable emergency class.

.1 Perform steps in Section 6.1 applicable to the new event classification.

99. From SO123-VIII-10, Step 6.3

6.3 Event Notifications

6.3.1 Complete the notifications for each emergency classification, reclassification, or change in PAR within the time limits specified in Table 2.

TABLE 2 - NOTIFICATION TIME LIMITS

TIME LIMIT	NOTIFICATION	RESPONSIBILITY
EDT + 15 minutes:	Verbal to Local &	NOA
EDT + =20	Verbal to NRC	OPS
EDT + 30 minutes:	ENF to Local & State	NOA
EDT + 90 minutes and every 60 minutes thereafter	ENF Follow-up	NOA

(EDT= Event Declaration Time)

100. From SO123-VIII-10, Precaution 4.1

4.0 PRECAUTIONS

4.1 The EC should ensure the verbal notification to the Nuclear Regulatory Commission (NRC) is made within 20 minutes after declaration, and no later than one hour after declaration.

100. From SO123-VIII-10, Step 6.3.1

6.3 Event Notifications

6.3.1 Complete the notifications for each emergency classification, reclassification, or change in PAR within the time limits specified in Table 2.

TABLE 2 - NOTIFICATION TIME LIMITS

TIME LIMIT	NOTIFICATION	RESPONSIBILITY
EDT + 15 minutes:	Verbal to Local &	NOA
EDT + =20	Verbal to NRC	OPS
EDT + 30 minutes:	ENF to Local & State	NOA
EDT + 90 minutes and every 60 minutes thereafter	ENF Follow-up	NOA

(EDT= Event Declaration Time)

Final Key

Site-Specific Written Examination
SONGS June 2007
Senior Reactor Operator
Answer Key

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

Final Key

USNRC Written Examination
SONGS June 2007
Reactor Operator
Answer Key

1. C
2. A
3. D
4. C
5. A
6. C
7. A
8. B
9. C
10. B
11. A
12. C
"B" only ~~KDC 13. C or D~~
14. B
15. C
16. B
17. A
18. C
19. D
20. B
21. A
22. C
23. C
24. C
25. A
26. C
27. C
28. B
29. B
30. A
31. A
32. B
33. A
34. A
35. B
36. C
37. A
38. B
39. D
40. C
41. D
42. D
43. A
44. B
45. B
46. C
47. A
48. A
49. B
50. D
51. D
52. C
53. D
54. B
55. B
56. A
57. C
58. A
59. A
60. B
61. B
62. B
63. A
64. D
65. D
66. C
67. A
68. A or B
69. C
70. B
71. ~~C~~ KDC A only
72. A
73. D
74. B
75. B