Draft Submittal

MCGUIRE JUNE 2003 EXAM
50-369/2003-301 AND
50-370/2003-301

JUNE 16 - 30, 2003

1. Senior Reactor Operator Written Exam

Bank Question: 1073 **Answer: A**

1 Pt(s) The following conditions exist on Unit 1:

- Reactor power is 100%
- 1A CA Pump is running with 1CA-60A (1A CA Pump Disch to 1A S/G Control) and 1CA-56 (1A CA Pump Disch to 1B S/G Control) closed for post maintenance testing.
- N/R level in 1B S/G increases to 84% due to 1CF-23 (1B S/G Control Valve) failing open.

Which one of the following statements correctly describes the response of the CA system to the above conditions?

A. **1A CA Pump remains running**
1B CA pump auto starts
1CA-60A and 1CA-56A fail open
1CA-44B (1B CA Pump Disch to 1C S/G Control) and 1CA-40B (1B CA Pump to Disch to 1D S/G Control) do not reposition.
Depressing the MD CA Modulating Valve Reset Train 'A' pushbutton will cause 1CA-60A and 1CA-56A to close.

B. **1A CA Pump remains running**
1B CA pump auto starts
1CA-60A and 1CA-56A remain closed.
1CA-44B and 1CA-40B fail closed.
Depressing the MD CA Modulating Valve Reset Train 'A' pushbutton will cause 1CA-60A and 1CA-56A to open

C. **1A CA Pump trips**
1B CA pump remains off
1CA-60A and 1CA-56A remain closed.
1CA-44B and 1CA-40B do not change position.
Depressing the MD CA Modulating Valve Reset Train 'A' pushbutton will cause 1CA-60A and 1CA-56A to open.

D. **1A CA Pump trips**
1B CA Pump remains off
1CA-60A and 1CA-56A fail open
1CA-44B and 1CA-40B fail open.
Depressing the MD CA Modulating Valve Reset Train 'A' pushbutton will cause 1CA-60A and 1CA-56A to close.

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Distracter Analysis:

A. Correct: .
B. Incorrect:
Plausible:.
C. Incorrect:
Plausible
D. Incorrect:
Plausible:.

Level: SRO

KA: APE 054 AA204 (4.2/4.3)

Lesson Plan Objective: CF-CA Obj. # 4,
OP-MC-ECC-ISE Obj. # 13)

Source: New

Level of knowledge: comprehension

References:
1. OP-MC-CF-CA page 13,
2. OP-MC-ECC-ISE page 33

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- 1A CA Pump is running with 1CA-60A (1A CA Pump Disch to 1A S/G Control) and 1CA-56 (1A CA Pump Disch to 1B S/G Control) closed for post maintenance testing.
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Which one of the following statements correctly describes the response of the CA system to the above conditions?

A. **1A CA Pump remains running**
1B CA pump auto starts
1CA-60A and 1CA-56A fail open
1CA-44B (1B CA Pump Disch to 1C S/G Control) and 1CA-40B (1B CA Pump to Disch to 1D S/G Control) do not reposition.
Depressing the MD CA Modulating Valve Reset Train 'A' pushbutton will cause 1CA-60A and 1CA-56A to close.

B. **1A CA Pump remains running**
1B CA pump auto starts
1CA-60A and 1CA-56A remain closed.
1CA-44B and 1CA-40B fail closed.
Depressing the MD CA Modulating Valve Reset Train 'A' pushbutton will cause 1CA-60A and 1CA-56A to open

C. **1A CA Pump trips**
1B CA pump remains off
1CA-60A and 1CA-56A remain closed.
1CA-44B and 1CA-40B do not change position.
Depressing the MD CA Modulating Valve Reset Train 'A' pushbutton will cause 1CA-60A and 1CA-56A to open.

D. **1A CA Pump trips**
1B CA Pump remains off
1CA-60A and 1CA-56A fail open
1CA-44B and 1CA-40B fail open.
Depressing the MD CA Modulating Valve Reset Train 'A' pushbutton will cause 1CA-60A and 1CA-56A to close.

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
3.0	2.0	3.0	3.0	1.5

OBJECTIVES

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	State the purpose of the CA System.	X	X	X	X	
2	Sketch the system drawing (Fig. 7.1) including all major components and valves, show all tie-ins to associated systems.	X	X	X	X	
3	Describe all CA suction supply sources, including venting requirements and actions.	X	X	X	X	
4	Discuss the auto-start of the motor driven and turbine driven auxiliary feedwater pumps, including concurrent BO/S_S signals and BO followed by S_S.	X	X	X	X	X
5	Describe the CA pump minimum flow and pump runout protection.	X	X	X	X	
6	Describe the function of the Auto Start Defeat Switches; include permissives.	X	X	X	X	X
7	Describe the power supplies and steam supplies for the CA pumps.	X	X	X	X	
8	State the flow rates of the CA pumps.	X	X	X	X	
9	Describe the sources of make-up to the Auxiliary Feedwater Storage Tank, include destination of overflow from the Auxiliary Feedwater Storage Tank.	X	X	X	X	X
10	Describe the interlock between the CA motor driven pump and the associated train RN pump. Include why the interlock is required.	X	X	X	X	X
11	Describe the interlock between the CA pump suction pressure and the RN assured makeup valves.	X	X	X	X	X
12	Describe the interlock between the RN assured makeup valves (CA-15, CA-18) and the DG Hx Inlet Valve. Include why the interlock is required.	X	X	X	X	X

1.0 INTRODUCTION

1.1 Purpose

Objective # 1

The auxiliary feedwater system is provided as a backup for the main feedwater system. It is designed as a means to dissipate heat from the Reactor Coolant System when normal systems are not available. The auxiliary feedwater system may also be used in normal plant startup and shutdown, as main feedwater, when the flow is less than 3% maximum design feedwater flow.

1.2 General Description

Objective # 2

Refer to Figure 7.1, 7.2, 7.3, 7.13. The CA system assures required feedwater flow to the steam generators for reactor coolant thermal energy dissipation when the CF system is not available through loss of power or other malfunctions. The CA system is required to operate until normal feedwater flow is restored or until the reactor coolant temperature is lowered to the point where the ND system can be utilized. The CA system is designed to start automatically for any event requiring emergency feedwater. Since the CA system is the only source of makeup water to the steam generators for reactor coolant heat removal when the main feedwater system becomes inoperable, it has been designed with redundancy and diversity. The CA system contains two motor driven pumps and one steam turbine driven pump for each unit.

2.0 COMPONENT DESCRIPTION

2.1 Motor Driven CA Pumps

Objective # 4, 7, 8

The motor driven CA pumps are powered from essential power, ETA (pump A) and ETB (pump B). Each motor driven pump has a design flow rate of 450 gpm and is capable of supplying two steam generators. CA pump "A" supplies steam generators "A" and "B" while CA pump "B" supplies steam generators "C" and "D."

Refer to Figure 7.12. The auto-start signals for the CA Motor Driven pumps are:

- 2/4 detectors low-low level in any one SG (17%)
- Trip of both Main Feedwater pumps
- AMSAC
    - Both Feedwater pumps tripped
    - Loss of flow to 3/4 SGs
- S_S signal
- Blackout signal

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
13	List the setpoints, permissives, and logic required to initiate the following: • Containment Spray (NS) Actuation • Phase "B" Isolation • Main Steam Isolation (MSI) • Main Feedwater Isolation (FWI)		X	X	X	X
14	Explain the relationship between SSPS Testing and the operability of the Systems and functions actuated from the Engineered Safety Features Actuation System.		X	X	X	X
15	Discuss the purpose of the ESF Monitor Lite Panel (BOP Panel).		X	X	X	
16	Concerning AP/1 or 2/A/5500/35, ECCS Actuation During Plant Shutdown. • State the purpose of the AP. • Recognize the symptoms that would require implementation of the AP.		X	X	X	X
17	Concerning the Technical Specifications related to the Engineered Safeguards Actuation System:					
	• Given the LCO title, state the LCO (including any COLR values) and applicability.			X	X	X
	• For any LCO's that have action required within one hour, state the action.			X	X	X
	• Given a set of parameter values or system conditions, determine if any Tech Spec LCO's is (are) not met and any action(s) required within one hour.			X	X	X
	• Given a set of plant parameters or system conditions and the appropriate Tech Specs, determine required actions.				X	X
	• Discuss the bases for a given Tech Spec LCO or Safety Limit. * - SRO Only				X	*

Objective # 13

Main Feedwater Isolation (FWI) is initiated by:

- **Safety Injection (S_S)**
- **Reactor trip and low T-avg (P-4 and 553°F on $^2/_4$ channels)**
- **High High S/G level 83% on $^2/_3$ channels on $^1/_4$ S/G (P-14)**
- **Manually ($^1/_2$ pushbuttons)**

Feedwater Isolation (FWI) Initiating Signal Automatic Actions to CF (Main Feedwater)

S_S *(Safety Injection)*

- FWI (Feedwater Isolation)
- Turbine trip
- Both FWPT's trip

P-4 and Low T-avg

- FWI (Feedwater Isolation)
- P-4 generates turbine trip
- FWPT's rollback hold

P-14

- FWI (Feedwater Isolation)
- Turbine trip
- Both FWPT's trip

Manual

- FWI (Feedwater Isolation)
- FWPT's rollback hold

Valves that close from FWI (Feedwater Isolation) signal

- S/G CF Control Valves (CF-32, 23, 20, 17)
- S/G CF Control Valve Bypasses (CF-104, 105, 106, 107)
- S/G CF Containment Isolations (CF-35, 30, 28, 26)
- CF to CA Nozzle Isolations (CF-126, 127, 128, 129)

The following conditions exist on Unit 1

- RTP is at 100%
- 1A CA Pump is running with 1CA-60A (1A CA Pump Disch to 1A S/G Control) and 1CA-56A (1A CA Pump Disch to 1A S/G Control) closed for post maintenance testing.
- NR level in 1B S/G is at 83% due to 1CF-23 (1B S/G Control Valve) failing open.

Which of the following describe s the response of the CA System to the above conditions?

A. ✓ 1A CA Pump remains running

1B CA pump auto starts.

1CA-60A (1A CA Pump Disch to 1A S/G Control) and 1CA-56A (1A CA Pump Disch to 1A S/G Control) fail open.

1CA-44B (1B CA Pump Disch to 1C S/G Control) and 1CA-40B (1B CA Pump Disch to 1D S/G Control) do not change position.

Depressing the MD CA Modulating Valve Reset A Train Pushbutton will cause 1CA-60A & 1CA-56A to close. ok

B. 1A CA Pump remains running.

1B CA Pump auto starts.

1CA-60A (1A CA Pump Disch to 1A S/G Control) and 1CA-56A (1A CA Pump Disch to 1A S/G Control) remain closed

1CA-44B (1B CA Pump Disch to 1C S/G Control) and 1CA-40B (1B CA Pump Disch to 1D S/G Control) do not change position

Depressing the MD CA Modulating Valve Reset A Train Pushbutton will cause 1CA-60A & 1CA-56A to open.

C. 1A CA Pump trips

1B CA Pump remains off.

1CA-60A (1A CA Pump Disch to 1A S/G Control) and 1CA-56A (1A CA Pump Disch to 1A S/G Control) remain closed

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Replacement for 16.1

1CA-44B (1B CA Pump Disch to 1C S/G Control) and 1CA-40B (1B CA Pump Disch to 1D S/G Control) do not change position

Depressing the MD CA Modulating Valve Reset A Train Pushbutton will cause 1CA-60A & 1CA-56A to open.

D. 1A CA Pump trips

1B CA Pump remains off

1CA-60A (1A CA Pump Disch to 1A S/G Control) and 1CA-56A (1A CA Pump Disch to 1A S/G Control) fail open.

1CA-44B (1B CA Pump Disch to 1C S/G Control) and 1CA-40B (1B CA Pump Disch to 1D S/G Control) do not change position.

Depressing the MD CA Modulating Valve Reset A Train Pushbutton will cause 1CA-60A & 1CA-56A to close.

1 Pt(s)

During a plant shutdown on Unit 1, the operators have blocked the CA auto-start signal by depressing the 'AUTO-START DEFEAT' switch. A subsequent loss of both main feedwater pumps occurred at 0200.

Given the following plant conditions at the times listed:

		Time			
	Condition	0200	0205	0210	0215
1)	NCS temperature (°F)	557	558	558	559
2)	NCS pressure (psig)	1903	1956	1976	1991
3)	NR SG A (%)	16	15	14	13
4)	NR SG B (%)	20	19	18	17
5)	NR SG C (%)	19	18	17	16
6)	NR SG D (%)	19	18	17	16

What time would the motor driven CA pumps restart automatically?

A. **0200**

B. **0205**

C. **0210**

D. **0215**

Bank Question: 16.1 ***Answer: B***

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Given the following plant conditions at the times listed:

	Condition	Time 0200	0205	0210	0215
1)	NCS temperature (°F)	557	558	558	559
2)	NCS pressure (psig)	1903	1956	1976	1991
3)	NR SG A (%)	16	15	14	13
4)	NR SG B (%)	20	19	18	17
5)	NR SG C (%)	19	18	17	16
6)	NR SG D (%)	19	18	17	16

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A. **0200**

B. **0205**

C. **0210**

D. **0215**

Distracter Analysis

A. -- **Incorrect**
Plausible
B. **Correct**
Plausible
C. **Incorrect**
Plausible
D. **Incorrect**
Plausible

LEVEL: SRO

Ques_16.1.doc

SOURCE: Bank

REFERENCES: OP-MC-CF-CA page 13 & 15

LESSON PLAN OBJECTIVE:OP-MC-CF-CA, Obj. 4 & 6

LEVEL OF KNOWLEDGE: Memory

K/A: APE 000054 AA2.04 (4.2/4.3)

APE:	054 Loss of Main Feedwater (MFW)
AA2.	**Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW): (CFR: 43.5 / 45.13)**

AA2.01	Occurrence of reactor and/or turbine trip	4.3	4.4
AA2.02	Differentiation between loss of all MFW and trip of one MFW pump . . .	4.1	4.4
AA2.03	Conditions and reasons for AFW pump startup	4.1	4.2
[redacted]			
AA2.05	Status of MFW pumps, regulating and stop valves	3.5	3.7
AA2.06	AFW adjustments needed to maintain proper T-ave. and S/G level	4.0	4.3
AA2.07	Reactor trip first-out panel indicator	3.4*	3.9
AA2.08	Steam flow-feed trend recorder	2.9	3.3*

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
3.0	2.0	3.0	3.0	1.5

OBJECTIVES

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	State the purpose of the CA System.	X	X	X	X	
2	Sketch the system drawing (Fig. 7.1) including all major components and valves, show all tie-ins to associated systems.	X	X	X	X	
3	Describe all CA suction supply sources, including venting requirements and actions.	X	X	X	X	
4	Discuss the auto-start of the motor driven and turbine driven auxiliary feedwater pumps, including concurrent BO/S_S signals and BO followed by S_S.	X	X	X	X	X
5	Describe the CA pump minimum flow and pump runout protection.	X	X	X	X	
6	Describe the function of the Auto Start Defeat Switches; include permissives.	X	X	X	X	X
7	Describe the power supplies and steam supplies for the CA pumps.	X	X	X	X	
8	State the flow rates of the CA pumps.	X	X	X	X	
9	Describe the sources of make-up to the Auxiliary Feedwater Storage Tank, include destination of overflow from the Auxiliary Feedwater Storage Tank.	X	X	X	X	X
10	Describe the interlock between the CA motor driven pump and the associated train RN pump. Include why the interlock is required.	X	X	X	X	X
11	Describe the interlock between the CA pump suction pressure and the RN assured makeup valves.	X	X	X	X	X
12	Describe the interlock between the RN assured makeup valves (CA-15, CA-18) and the DG Hx Inlet Valve. Include why the interlock is required.	X	X	X	X	X

1.0 INTRODUCTION

1.1 Purpose

Objective # 1

The auxiliary feedwater system is provided as a backup for the main feedwater system. It is designed as a means to dissipate heat from the Reactor Coolant System when normal systems are not available. The auxiliary feedwater system may also be used in normal plant startup and shutdown, as main feedwater, when the flow is less than 3% maximum design feedwater flow.

1.2 General Description

Objective # 2

Refer to Figure 7.1, 7.2, 7.3, 7.13. The CA system assures required feedwater flow to the steam generators for reactor coolant thermal energy dissipation when the CF system is not available through loss of power or other malfunctions. The CA system is required to operate until normal feedwater flow is restored or until the reactor coolant temperature is lowered to the point where the ND system can be utilized. The CA system is designed to start automatically for any event requiring emergency feedwater. *Since the CA system is the only source of makeup water to the steam generators for* reactor coolant heat removal when the main feedwater system becomes inoperable, it has been designed with redundancy and diversity. The CA system contains two motor driven pumps and one steam turbine driven pump for each unit.

2.0 COMPONENT DESCRIPTION

2.1 Motor Driven CA Pumps

Objective # 4, 7, 8

The motor driven CA pumps are powered from essential power, ETA (pump A) and ETB (pump B). Each motor driven pump has a design *flow rate of 450 gpm* and is capable of supplying two steam generators. CA pump "A" supplies steam generators "A" and "B" while CA pump "B" supplies steam generators "C" and "D."

Refer to Figure 7.12. *The auto-start signals for the* CA Motor Driven pumps are:

- 2/4 detectors low-low level in any one SG (17%)
- Trip of both Main Feedwater pumps
- AMSAC
  - Both Feedwater pumps tripped
  - Loss of flow to 3/4 SGs
- S_S signal
- Blackout signal

An <u>Auto-Start Defeat Switch</u> can be used to defeat

- 2/4 low-low level in any SG
- Trip of both Main Feedwater pumps
- AMSAC (Both Main Feedwater Pumps Tripped)

NC System pressure must be below the P-11 setpoint (1955 psig) to enable the Auto-Start Defeat feature. The Auto-Start Defeat feature will "auto unblock" when pressure returns above the P-11 setpoint.

Objective # 10

The train related RN pump will automatically start upon any start (including Manual) of the corresponding CA pump to provide necessary cooling.

2.2 Turbine Driven CA Pump

Objective # 7, 8

Each unit has one Steam Turbine Driven CA pump. The turbine receives steam from "B" and "C" steamlines through two redundant valves. The turbine driven pump has a design flow rate of 900 gpm and supplies all four steam generators. Steam is admitted to the turbine through two piston operated isolation valves, SA-48ABC and SA-49AB. These valves are held closed by control air from normally energized solenoid valves. De-energizing any solenoid valve will bleed-off the control air, open its piston operated valve and admit steam to the turbine.

Objective # 4

Refer to Figure 7.12. The auto-start signals for the CA Turbine Driven pump (which open SA-48ABC and SA-49AB) are:

- 2/4 detectors low-low level in any two SGs (17%)
- Blackout (> 8 seconds)
- 1/1 detector from SSF SG Wide Range Low-Low Level on 2/4 SGs (72%) (only opens SA-48ABC)

NOTE: If a Blackout occurs first followed by a Safety Injection, the Sequencer will reset the start signal to the Turbine Driven CA Pump. If the Turbine Driven CA Pump is running at the time of the Safety Injection, it will continue to run. If the Safety Injection occurs first or coincident with the Blackout, the Safety Injection will BLOCK the Turbine Driven CA Pump start because the sequencer selects the Priority Mode. (This does not affect the Low-Low SG Level auto start signal or the SSF Low-Low Level start signal.)

NOTE: The turbine driven pump will also start on loss of VI or loss of power to the solenoid valves, due to the fail-open design of the valves (not considered an auto-start)

1 Pt. Given the following conditions:

- Unit 1 is in a refueling outage.
- *Fuel movement is in progress.*
- A leak has developed which has caused level to drop in the spent fuel pool.
- The Spent Fuel Pool Level Low computer alarm has actuated.
- Pool was initially at normal level and area radiation at 7 mrem/hr.
- After 20 minutes the pool level has decreased further and area radiation is 18 mrem/hr.

Which one (1) of the following describes the operator response to the current conditions?

A. **Begin makeup to the pool from the Boric Acid Tank. to restore level.**

B. **Move the fuel transfer cart to the reactor side and close 1KF-122 (*Fuel Transfer Tube Block Valve*).**

C. **Move the fuel transfer cart to the spent fuel (pit) side and close 1KF-122 (*Fuel Transfer Tube Block Valve*).**

D. **Place the weir gate in position and inflate the seals.**

Bank Question: 33.1 ***Answer: C***

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D. Place the weir gate in position and inflate the seals.

LEVEL: SRO

SOURCE: BANK

LEVEL OF KNOWLEDGE: Memory

REFERENCES: OP-MC-FH-FC pages 19-25 odd only

LESSON: OP-MC-FH-FC

OBJECTIVE: OP-MC-FH-FC Obj. 6

K/A: 036 AA2.02 (3.4/4/1)

APE 036 Fuel Handling Incidents

K/A NO.	KNOWLEDGE	IMPORTANCE RO	SRO
AK1.	**Knowledge of the operational implications of the following concepts as they apply to Fuel Handling Incidents : CFR 41.8 / 41.10 / 45.3)**		
AK1.01	Radiation exposure hazards	3.5	4.1
AK1.02	SDM	3.4	3.8
AK1.03	Indications of approaching criticality	4.0	4.3
AK2.	**Knowledge of the interrelations between the Fuel Handling Incidents and the following: (CFR 41.7 / 45.7)**		
AK2.01	Fuel handling equipment	2.9	3.5
AK2.02	Radiation monitoring equipment (portable and installed)	3.4	3.9
AK3.	**Knowledge of the reasons for the following responses as they apply to the Fuel Handling Incidents: (CFR 41.5,41.10 / 45.6 / 45.13)**		
AK3.01	Different inputs that will cause a reactor building evacuation	3.1	3.7
AK3.02	Interlocks associated with fuel handling equipment	2.9	3.6
AK3.03	Guidance contained in EOP for fuel handling incident	3.7	4.1
	ABILITY		
AA1.	**Ability to operate and / or monitor the following as they apply to the Fuel Handling Incidents: (CFR 41.7 / 45.5 / 45.6)**		
AA1.01	Reactor building containment purge ventilation system	3.3	3.8
AA1.02	ARM system	3.1	3.5
AA1.03	Reactor building containment evacuation alarm enable switch	3.5	3.9
AA1.04	Fuel handling equipment during an incident	3.1	3.7
AA2.	**Ability to determine and interpret the following as they apply to the Fuel Handling Incidents: (CFR: 43.5 / 45.13)**		
AA2.01	ARM system indications	3.2	3.9
AA2.02	Occurrence of a fuel handling incident	3.4	4.1
AA2.03	Magnitude of potential radioactive release	3.1*	4.2*

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
		1.5	1.5	1.5

OBJECTIVES

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	Describe the roles and responsibilities of Control Room Operators during Fuel Handling operations.			X	X	X
2	Describe the roles and responsibilities of Fuel Handling SRO's during Fuel Handling operations.				X	X
3	Describe how monitoring of core reactivity is accomplished during Fuel Handling.			X	X	X
4	Deleted					
5	Describe the requirements that must be met before **bypassing** a Fuel Handling Interlock.			X	X	X
6	Concerning AP-25, Spent Fuel Damage; AP-40, Loss of Refueling Canal; and AP-41, Loss of Spent Fuel Cooling or Level: • State the purpose of the AP • Given symptoms, state the AP and Case (if applicable)			X	X	X
7	Concerning the Technical Specifications related to the FC System;					
	• Given the LCO title, state the LCO (including any COLR values) and applicability.			X	X	X
	• For any LCO's that have action required within one hour, state the action.			X	X	X
	• Given a set of parameter values or system conditions, determine if any Tech. Spec. is (are) not met and any action(s) required within one hour.			X	X	X
	• Given a set of plant parameters values or system conditions and the appropriate Tech Specs, determine required action(s).				X	*
	• Discuss the basis for a given Tech. Spec. LCO or Safety Limit.					
	* **SRO only**					

The Symptoms include:

- EMF36 UNIT VENT GAS HI RAD alarm
- EMF38 CONTAINMENT PART HI RAD alarm
- EMF39 CONTAINMENT GAS HI RAD alarm
- EMF40 CONTAINMENT IODINE HI RAD alarm
- EMF42 FUEL BLDG VENT HI RAD alarm
- EMF16 CONTAINMENT REFUELING BRIDGE alarm (2 - EMF3 on Unit 2)
- EMF17 SPENT FUEL BLDG REFUEL BRDG alarm (2 - EMF4 on Unit 2)
- Gas bubbles originating from the damaged assemblies
- Visible evidence of damage with the potential of radioactive releases

Operator Actions

CAUTION Damage to the rubber Reactor Vessel Cavity Seal may occur if an assembly is dropped on or near it.

Announce on page. If in containment, **evacuate** containment, assemble in contaminated change room and refer to RP/0/A/5700/11, Conducting a Site Assembly, Site Evacuation, or Containment Evacuation. **Isolate** containment: stop VP fans, ensure VP valves close, stop any VQ release, ensure equipment hatch closed, ensure one airlock door closed, dispatch Operator to move conveyor to Spent Fuel Pool Building, dispatch Operator to close KF-122. If high containment radiation exists, place Aux Carbon Filters in service per OP. Place Refueling Cavity in purification per OP.

If in Spent Fuel Building, **evacuate** Spent Fuel Pool area, assemble in contaminated change room. **Isolate** Spent Fuel Pool area: Check if VF EXH BYP DAMPER closed lite lit, and if not, place it's control switch to "CLOSE", and close the doors to the Spent Fuel Pool area. Ensure KF purification loop in service per OP.

Refer to RP/0/A/5700/00, Classification of Emergency.

3.2.2 AP/1/A/5500/40, LOSS OF REFUELING CANAL LEVEL

The purpose is to provide actions in the event of loss of water in the refueling canal.

The Symptoms include:

- "Spent Fuel Pool Level Low" computer alarm
- Decreasing level in refueling canal
- "Incore Inst Room Sump Hi Level" alarm
- EMF16 CONTAINMENT REFUELING BRIDGE alarm (2 - EMF3 on Unit 2)
- EMF17 SPENT FUEL BLDG REFUEL BRDG alarm (2 - EMF4 on Unit 2)

Operator Actions

NOTE Any available core location may be used when lowering a fuel assembly during emergency conditions.

If fuel movement is in progress: lower any assembly in the reactor building crane to fully down in the core, any assembly in the spent fuel crane to fully down, and any assembly in the upender to fully down. If they won't lower otherwise, manually release the brake and hand crank the hoist down. **NOTE: The sequence for lowering the hoist manually should be to put the emergency handwheel on the end of the hoist motor, hold it steady, while another person screws in the brake release (star shaped knob on a threaded stud) which when threaded in forces the brake disengaged. Care should be taken to remove the handwheel before electric operation of the hoist motor. The upender is similar. The bridge and trolly brake release is a lever, otherwise similar.** Dispatch Operator to locally move fuel transfer cart to the spent fuel (pit) side. Stop FWST Pump and close FW-13, and dispatch Operator to locally close KF-122. If KF-122 cannot be closed, then notify RP to begin surveys, consider installing the weir gate, and isolate the Spent Fuel Building (VF in filter mode and doors closed). Evacuate nonessential personnel from containment and Spent Fuel Building.

Try to identify and correct the cause of decreasing level. Verify seal integrity and air pressure to the Rx Vessel cavity seal and the Rx Vessel nozzle inspection port seals, and if not, reestablish VI to seals. Dispatch an Operator to locally ensure the Refueling Cavity Drains are closed. Check the S/G Nozzle Dams. Refer to AP/19, Loss of ND or ND System Leakage, while continuing with this procedure.

Makeup to the canal per OP/1/A/6200/13. **CAUTION:** Makeup to the SFP could dilute NC system boron concentration.

Monitor the Spent Fuel Pool level. If it gets to minus two feet, stop the KF Pump and turn off the lights. Initiated makeup per OP. If pool level low enough for radiation hazard, makeup from RN.

Ensure Containment Integrity with equipment hatch and airlock doors closed. If time permits, turn off canal underwater lights before they become uncovered. If necessary due to increasing radiation levels, consider using ND or NS to transfer water from the containment sump to the FWST for additional makeup capability.

Refer to RP/0/A/5700/00, Classification of Emergency.

3.2.3 AP/1/A/5500/41, LOSS OF Spent Fuel Pool Cooling or Level

The purpose of this procedure is to provide actions to take in the event of loss of Spent Fuel Cooling for the following cases:

Case 1 Loss of Spent Fuel Pool Cooling

Case 2 Loss of Spent Fuel Pool Level

Case 1 Loss of Spent Fuel Pool Cooling

The Symptoms include:

- "SPENT FUEL POOL TEMP HI" computer alarm
- Both KF Pumps off

Operator Actions

If fuel handling is in progress, lower any fuel assembly in the crane, radioactive component, or fuel assembly in the upender to fully down. Release brake and hand crank hoist or upender, if required.

If KC is aligned to either ND Hx, ensure all KC Pumps are running per KC OP (Encl 4.3, Shifting Trains). Maintain KC flow to less than 4000 gpm per pump in subsequent steps.

Ensure at least one train of KC is aligned to the AB nonessential header. If it's not because of a LOCA, then Enclosure 1 is used to restore cooling during an extended LOCA. Throttle open the appropriate KF Hx Outlet Flow valve for the KF Hx to be placed in service. Check the KF Pump is running, and if not start it. Check the Spent Fuel Pool temperature going down.

If KF Pump can't be started or pool temperature is not going down, place VF in filter mode, close the doors, and notify RP. If boiling occurs, begin makeup per KF OP (Encl 4.4, Spent Fuel Pool Level Control).

Refer to RP/0/A/5700/00, Classification of Emergency.

Case 2 Loss of Spent Fuel Pool Level

The Symptoms include:

- "SPENT FUEL POOL LEVEL LO" computer alarm
- Level in Spent Fuel Pool going down
- EMF17 SPENT FUEL BLDG REFUEL BRDG alarm (2 - EMF4 on Unit 2)

OPERATOR ACTIONS

Announce on Page.

Dispatch an Operator to check the Pool is isolated from the refueling canal (either KF-122 closed, weir gate installed, or transfer tube blind flange installed). If not, go to AP/40, Loss of Refueling Canal Level.

If fuel handling is in progress, lower any fuel assembly in the crane, radioactive component, or fuel assembly in the upender to fully down. Release brake and hand crank hoist or upender, if required.

If the level is lower than minus 2 feet, stop the KF Pump, deenergize the underwater lights, place VF in the filter mode, close the doors, and notify RP. Initiate makeup per KF OP (Encl 4.4, Spent Fuel Pool Level Control).

Dispatch an Operator to locate and isolate the leak. When the leak is isolated and level has been returned to normal, start the KF Pump per the OP.

1 Pt. Unit 1 is in mode 4.

Given the following conditions:

(1) Surveillance testing has been recently completed on the ice condenser
(2) The surveillance test was not satisfactory as described below

Which one (1) of the following situations meets the requirements for a one hour tech spec LCO?

A. **The ice condenser door position monitoring system was declared inoperable when one door did not indicate in the open position during a surveillance test. The door was left in the open position.**

B. **The ice bed was declared inoperable when it was determined that it failed a surveillance test based on total ice weight less than 2,099,790 pounds at a 95% level of confidence.**

C. **The Ice Bed Temperature Monitoring System was declared inoperable when it failed a Tech Spec surveillance test - channel check failure.**

D. **The ice condenser intermediate deck door was declared inoperable when it was discovered to be obstructed from opening by ice and debris.**

Bank Question: 40.1 ***Answer: D***

1 Pt. Unit 1 is in mode 4.

Given the following conditions:

(1) Surveillance testing has been recently completed on the ice condenser
(2) The surveillance test was not satisfactory as described below

Which one (1) of the following situations meets the requirements for a one hour tech spec LCO?

A. The ice condenser door position monitoring system was declared inoperable when one door did not indicate in the open position during a surveillance test. The door was left in the open position.

B. The ice bed was declared inoperable when it was determined that it failed a surveillance test based on total ice weight less than 2,099,790 pounds at a 95% level of confidence.

C. The Ice Bed Temperature Monitoring System was declared inoperable when it failed a Tech Spec surveillance test - channel check failure.

D. The ice condenser intermediate deck door was declared inoperable when it was discovered to be obstructed from opening by ice and debris.

LEVEL: SRO

SOURCE: BANK

REFERENCES: Tech Spec 3.6.13

LESSON: OP-MC-CNT-NF

OBJECTIVE: OP-MC-CNT-NF Obj. 21

K/A: SYS 025 G2.2.22 (3.4/4.1)

LEVEL OF KNOWLEDGE: Memory

2.2 Equipment Control (Continued)

2.2.18 Knowledge of the process for managing maintenance activities during shutdown operations.
(CFR: 43.5 / 45.13)
IMPORTANCE RO 2.3 SRO 3.6

2.2.19 Knowledge of maintenance work order requirements.
(CFR: 43.5 / 45.13)
IMPORTANCE RO 2.1 SRO 3.1

2.2.20 Knowledge of the process for managing troubleshooting activities.
(CFR: 43.5 / 45.13)
IMPORTANCE RO 2.2 SRO 3.3

2.2.21 Knowledge of pre- and post-maintenance operability requirements.
(CFR: 43.2)
IMPORTANCE RO 2.3 SRO 3.5

2.2.22 Knowledge of limiting [redacted]
IMPORTANCE [redacted]

2.2.23 Ability to track limiting conditions for operations.
(CFR: 43.2 / 45.13)
IMPORTANCE RO 2.6 SRO 3.8

2.2.24 Ability to analyze the affect of maintenance activities on LCO status.
(CFR: 43.2 / 45.13)
IMPORTANCE RO 2.6 SRO 3.8

2.2.25 Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.
(CFR: 43.2)
IMPORTANCE RO 2.5 SRO 3.7

2.2.26 Knowledge of refueling administrative requirements.
(CFR: 43.5 / 45.13)
IMPORTANCE RO 2.5 SRO 3.7

2.2.27 Knowledge of the refueling process.
(CFR: 43.6 / 45.13)
IMPORTANCE RO 2.6 SRO 3.5

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
	Concerning the Technical Specifications related to the Ice Condenser					
	• Given the LCO title, state the LCO (including any COLR values) and applicability.			X	X	X
	• For any LCO's that have action required within one hour, state the action.			X	X	X
	• Given a set of parameter values or system conditions, determine if any Tech Spec LCO's is(are) not met and any action(s) required within one hour.			X	X	X
	• Given a set of parameter values or system conditions and the appropriate Tech Spec, determine required action(s).			X	X	X
	• Discuss the bases for a given Tech. Spec. LCO or Safety Limit.				X	*
	* SRO ONLY					

3.6 CONTAINMENT SYSTEMS

3.6.13 Ice Condenser Doors

LCO 3.6.13 The ice condenser inlet doors, intermediate deck doors, and top deck doors shall be OPERABLE and closed.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each ice condenser door.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more ice condenser doors inoperable due to being physically restrained from opening.	A.1 Restore door to OPERABLE status.	1 hour
B. One or more ice condenser doors inoperable for reasons other than Condition A or not closed.	B.1 Verify maximum ice bed temperature is $\leq 27°F$. AND B.2 Restore ice condenser door to OPERABLE status and closed positions.	Once per 4 hours 14 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition B not met.	C.1 Restore ice condenser door to OPERABLE status and closed position.	48 hours
D. Required Action and associated Completion Time of Condition A or C not met.	D.1 Be in MODE 3.	6 hours
	AND	
	D.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.13.1	Verify all inlet doors indicate closed by the Inlet Door Position Monitoring System.	12 hours
SR 3.6.13.2	Verify, by visual inspection, each intermediate deck door is closed and not impaired by ice, frost, or debris.	7 days
SR 3.6.13.3	Verify, by visual inspection, each top deck door: a. Is in place; and b. Has no condensation, frost, or ice formed on the door that would restrict its opening.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.13.4	Verify, by visual inspection, each inlet door is not impaired by ice, frost, or debris.	18 months
SR 3.6.13.5	Verify torque required to cause each inlet door to begin to open is ≤ 675 in-lb.	18 months
SR 3.6.13.6	Perform a torque test on each inlet door.	18 months
SR 3.6.13.7	Verify for each intermediate deck door: a. No visual evidence of structural deterioration; b. Free movement of the vent assemblies; and c. Free movement of the door.	18 months

Bank Question: 1072 ***Answer: A***

1 Pt(s) As an SRO working on a 'Complex Maintenance Plan' you are asked to evaluate four possible work teams who must repair filter housing in a 1500 mRem/hr radiation field.

Which one of the following work teams would maintain station ALARA?

A. **A qualified male worker who has previously performed this task. He can complete this job in 20 minutes. This worker has exceeded his 'Alert' level for exposure and will require a dose extension.**

B. **Two male workers who are qualified to perform the task. Together they can perform the task in 15 minutes. Both workers have already accumulated 325 mRem this year.**

C. **A team of a female worker who is qualified to perform the task and a male worker who needs to qualify to this task. The female is a declared pregnant worker. The team will need 15 minutes to complete the task. The female has no dose and the male worker has 200 mRem for the year.**

D. **A team of a male and female both are qualified to the task but will take 20 minutes to complete the task. Each has less than 100 mRem this year.**

Distracter Analysis:

A. **Correct:** 500 mR total.
B. **Incorrect:** 750 per mrem total
Plausible
C. **Incorrect:** Declared pregnant worker.
Plausible:
D. **Incorrect:** 1000 mrem total

Level: SRO

KA:

Less

Sour

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Post-it® Fax Note 7671	Date	# of pages
To Rick Baldwin	From Charles Sawyer	
Co./Dept.	Co.	
Phone #	Phone #	
Fax #	Fax #	

Bank Question: 1072 ***Answer: A***

1 Pt(s) As an SRO working on a 'Complex Maintenance Plan' you are asked to evaluate four possible work teams who must repair filter housing in a 1500 mrem/hr radiation field.

Which one of the following work teams would maintain station ALARA?

A. **A qualified male worker who has previously performed this task. He can complete this job in 20 minutes. This worker has exceeded his 'Alert' level for exposure and will require a dose extension.**

B. **Two male workers who are qualified to perform the task. Together they can perform the task in 15 minutes.**

C. **A team of a female worker who is qualified to perform the task and a male worker who needs to qualify to this task. Both workers have low dose, but the female is a declared pregnant worker.**

D. **A team of a male and female both are qualified to the task but will take 20 minutes to complete the task. Each has less than 100 mrem this year.**

Distracter Analysis:

A. **Correct:** 500 mR total.
B. **Incorrect:** 750 per mrem total
Plausible
C. **Incorrect:** Declared pregnant worker.
Plausible:
D. **Incorrect:** 1000 mrem total

Replacement for 124.1

Level: SRO

KA: G2.3.2 (2.5/2.9)

Lesson Plan Objective: RAD RP Ojb. 135

Source: New

Level of knowledge: comprehension

References:

1. OP-MC-RAD-RP page 73

1 Pt(s) A team of workers must repack the seals on a pump in a 1500 mrem/hr extra high radiation area.

Which one of the following work teams and estimated repair times would maintain ALARA?

A. **10 people working for 20 minutes**

B. **6 people working for 30 minutes**

C. **4 people working for 1 hour**

D. **2 people working for 2 hours**

Bank Question: 124.1 ***Answer: B***

1 Pt(s) A team of workers must repack the seals on a pump in a 1500 mrem/hr extra high radiation area.

Which one of the following work teams and estimated repair times would maintain ALARA?

A. **10 people working for 20 minutes**

B. **6 people working for 30 minutes**

C. **4 people working for 1 hour**

D. **2 people working for 2 hours**

Distracter Analysis:

A. **Incorrect**: six people can accomplish the job with 4 1/2 Rem.
Plausible: Each individual would have the least exposure.

B. **Correct**:

C. **Incorrect**: six people can accomplish the job with 4 1/2 Rem.
Plausible: fewest individuals not exceeding the admin dose limit.

D. **Incorrect**: six people can accomplish the job with 4 1/2 Rem.
Plausible: Exposes the fewest individuals.

Level: SRO

KA: G2.3.2 (2.5/2.9)

Lesson Plan Objective: RAD-RP Obj. 135

Source: Bank

Level of knowledge: comprehension

References:
1. OP-MC-RAD-RP page 73

2.3 Radiation Control

2.3.1 **Knowledge of 10 CFR: 20 and related facility radiation control requirements.**
(CFR: 41.12 / 43.4. 45.9 / 45.10)
IMPORTANCE RO 2.6 SRO 3.0

2.3.2 **Knowledge of facility ALARA program.**
(CFR: 41.12 / 43.4 / 45.9 / 45.10)
IMPORTANCE RO 2.5 SRO 2.9

2.3.3 **Knowledge of SRO responsibilities for auxiliary systems that are outside the control room (e.g., waste disposal and handling systems).**
(CFR: 43.4 / 45.10)
IMPORTANCE RO 1.8 SRO 2.9

2.3.4 **Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.**
(CFR: 43.4 / 45.10)
IMPORTANCE RO 2.5 SRO 3.1

2.3.5 **Knowledge of use and function of personnel monitoring equipment.**
(CFR: 41.11 / 45.9)
IMPORTANCE RO 2.3 SRO 2.5

2.3.6 **Knowledge of the requirements for reviewing and approving release permits.**
(CFR: 43.4 / 45.10)
IMPORTANCE RO 2.1 SRO 3.1

2.3.7 **Knowledge of the process for preparing a radiation work permit.**
(CFR: 41.10 / 45.12)
IMPORTANCE RO 2.0 SRO 3.3

2.3.8 **Knowledge of the process for performing a planned gaseous radioactive release.**
(CFR: 43.4 / 45.10)
IMPORTANCE RO 2.3 SRO 3.2

2.3.9 **Knowledge of the process for performing a containment purge.**
(CFR: 43.4 / 45.10)
IMPORTANCE RO 2.5 SRO 3.4

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
133	Identify the type of exposures (i.e., Emergency Exposure, Planned Special Exposure, etc.) that are included as part of a radiation workers total allowable exposure.	X	X	X	X	X
134	Discuss the way in which exposure to neutron radiation is considered for the following: • Duke's administrative policy requirements for neutron dosimetry for entering the RCA • How neutron dose accumulation is tracked between the times that the individual's TLD is read	X	X	X	X	X
135	State the goals and efforts of the ALARA Program.	X	X	X	X	X
136	For the following areas: • Define and describe MNS Unrestricted Area • Describe MNS Restricted Area • Describe MNS RCA	X	X	X	X	X
137	State the dose ~~limit~~ associated with an unrestricted area.	X	X	X	X	X
138	List the requirements for wearing dosimetry devices inside and outside the RCA.	X	X	X	X	X
139	State the additional controls placed on entry/access to: • High Radiation Area • Very High Radiation Area • Extra High Radiation Area	X	X	X	X	X
140	Explain when a RWP is required.	X	X	X	X	X
141	Discuss how long a RWP or SRWP is valid.	X	X	X	X	X
142	Correctly interpret the information on the Daily Dose Report.	X		X	X	
143	State the purpose of protective clothing.	X	X	X	X	X

Objective #134

Regulatory Guide 8.14 requires a personnel **neutron** dosimeter if the neutron dose equivalent is likely to exceed 100 mrem in a quarter. Duke Power has an administrative requirement which requires all personnel entering the RCA to wear a TLD (which measures neutron dose equivalent). Estimation of neutron exposure is a method used to temporarily track exposure until the TLD is processed. Estimated neutron exposure tracking for personnel is required if the neutron dose equivalent is likely to exceed 10 mrem per entry or per job if consecutive multiple entries are required. There are two methods used to estimate neutron exposure:

- One method is to measure the neutron dose rate and then calculate the exposure based on stay time.
- The second method is to determine the gamma exposure dose and neutron exposure dose for the given area. If it is determined that the neutron to gamma ratio is essentially constant during the period of personnel exposure, then a gamma/neutron ratio can be utilized. The gamma dose received can be ratioed to find the neutron dose received.

Objective #135

ALARA is a philosophy aimed at the minimizing exposure thru a management commitment. The goals and efforts of the McGuire Nuclear station Program are simple:

- To maintain the annual dose to each individual ALARA
- To maintain the collective dose (total person-Rem) ALARA
- Both points have to be considered simultaneously, as one without the other is not ALARA.

3.5. Radiation Areas and Access Controls

Objective #136

Unrestricted Area - an area, access to which is neither limited nor controlled by the licensee. At McGuire this is the area out side the protected area fence.

Restricted Area - an area, access to which is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. The restricted area at McGuire is enclosed by the protected area fence.

1 Pt. Unit 1 is preparing for a reactor start up following a refueling outage. Given the following conditions:

- T_{avg} = 515 °F
- Plant heatup in progress using NCPs

At *0200, a Station Engineer* reports that a mistake had been made in analyzing the containment Appendix J Leak Rate Test results that were conducted prior to exceeding 200 °F in mode 5. Reanalysis indicated that the combined containment leak rate (Type A) had exceeded 1.0 L_a.

Which one of the following actions are required by Tech Specs in response to this situation?

REFERENCES PROVIDED

A. Commence a plant cooldown to reach mode 5 within 42 hours.

B. Commence a plant cooldown to reach mode 5 within 36 hours.

C. Commence a plant cooldown to reach mode 5 within 37 hours.

D. Commence a plant cooldown to reach mode 5 within 43 hours.

Bank Question: 207.1 **_Answer: C_**

1 Pt. Unit 1 is preparing for a reactor start up following a refueling outage. Given the following conditions:

- T_{avg} = 515 °F
- *Plant heatup in progress using* NCPs

At 0200, a Station Engineer reports that a mistake had been made in analyzing the containment Appendix J Leak Rate Test *results that were* conducted prior to exceeding 200 °F in mode 5. Reanalysis indicated that the combined containment leak rate (Type A) had exceeded 1.0 L_a.

Which one of the following actions are required by Tech Specs in response *to this situation?*

REFERENCES PROVIDED
Tech Spec 3.6.1and Bases

A. Commence a plant cooldown to reach mode 5 within 42 hours.

B. Commence a plant cooldown to reach mode 5 within 36 hours.

C. Commence a plant cooldown to reach mode 5 within 37 hours.

D. Commence a plant cooldown to reach mode 5 within 43 hours.

Distracter Analysis:

A. Incorrect:
Plausible:
B. Incorrect:
Plausible:
C. Correct:
Plausible:
D. Incorrect
Plausible:

SOURCE: BANK

LEVEL: SRO

LEVEL OF KNOWLEDGE: ANALYSIS

KA: SYS 103 A2.01 (2.0*/2.6)

OBJECTIVES: OP-MC-ADM-TS Obj. 2

REFERENCES: Tech Spec 3.6.1 and Bases
OP-MC-ADM-TS pages 29, 31

SYSTEM: 103 Containment System

K6 **Knowledge of the effect of a loss or malfunction on the following will have on the containment system:**
(CFR: 41.7 / 45.7)

K6.01	Valves	2.1*	2.3
K6.02	Controllers and positioners	1.9	2.1*
K6.03	Pumps	1.5	1.6
K6.04	Heat exchangers and condensers	1.5	1.7
K6.05	Breakers, relays, and disconnects	1.5	1.7
K6.06	Sensors and detectors	1.9	2.1

ABILITY

A1 **Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the containment system controls including:**
(CFR: 41.5 / 45.5)

A1.01	Containment pressure, temperature, and humidity	3.7	4.1

A2 **Ability to (a) predict the impacts of the following malfunctions or operations on the containment system- and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations**
(CFR: 41.5 / 43.5 / 45.3 / 45.13)

A2.01	Integrated leak rate test	2.0*	2.6*
A2.02	Necessary plant conditions for work in containment	2.2	3.2*
A2 03	Phase A and B isolation	3.5*	3.8*
A2 04	Containment evacuation (including recognition of the alarm)	3.5*	3.6*
A2.05	Emergency containment entry	2.9	3.9

A3 **Ability to monitor automatic operation of the containment system, including:**
(CFR: 41.7 / 45.5)

A3.01	Containment isolation	3.9	4.2

A4 **Ability to manually operate and/or monitor in the control room:**
(CFR: 41.7 / 45.5 to 45.8)

A4.01	Flow control, pressure control, and temperature control valves, including pneumatic valve controller	3.2*	3.3
A4.02	Excess letdown divert valves to reactor coolant drain tank	2.1*	2.2*
A4.03	ESF slave relays	2.7*	2.7*
A4 04	Phase A and phase B resets	3.5*	3.5*

3.6 CONTAINMENT SYSTEMS

3.6.1 Containment

LCO 3.6.1 Containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment inoperable.	A.1 Restore containment to OPERABLE status.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1	-------------------------------NOTES------------------------------ 1. The space between each dual ply bellows assembly on penetrations between the containment building and annulus shall be vented to the annulus during Type A tests. 2. Following each Type A test, the space between each dual-ply bellows assembly shall be subjected to a low pressure test at 3 to 5 psig to verify no detectable leakage, or the assembly shall be subjected to a leak test with the pressure on the containment side of the assembly at P_a. 3. Type C tests on penetrations M372 and M373 may be performed without draining the glycol-water mixture from the seats of their diaphragm valves if meeting a zero indicated leakage rate (not including instrument error). --- Perform required visual examinations and leakage rate testing except for containment airlock testing, in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

The containment is a free standing steel pressure vessel surrounded by a reinforced concrete reactor building. The containment vessel, including all its penetrations, is a low leakage steel shell designed to contain the radioactive material that may be released from the reactor core following a design basis Loss of Coolant Accident (LOCA). Additionally, the containment vessel and reactor building provide shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment vessel is a vertical cylindrical steel pressure vessel with hemispherical dome and a flat circular base. It is completely enclosed by a reinforced concrete reactor building. An annular space exists between the walls and domes of the steel containment vessel and the concrete reactor building to provide for the collection, mixing, holdup, and controlled release of containment out leakage. Ice condenser containments utilize an outer concrete building for shielding and an inner steel containment for leak tightness.

Containment piping penetration assemblies provide for the passage of process, service, sampling, and instrumentation pipelines into the containment vessel while maintaining containment integrity. The reactor building provides shielding and allows controlled release of the annulus atmosphere under accident conditions, as well as environmental missile protection for the containment vessel and Nuclear Steam Supply System.

The inner steel containment and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J, Option B (Ref. 1), as modified by approved as above exemptions.

The isolation devices for the penetrations in the containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:

a. All penetrations required to be closed during accident conditions are either:

BASES

BACKGROUND (continued)

1. capable of being closed by an OPERABLE automatic containment isolation system, or

2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves";

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks";

c. All equipment hatches are closed and sealed; and

d. The sealing mechanism associated with a penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting Design Basis Accident (DBA) without exceeding the design leakage rates.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA) and a steam line break (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.3% of containment air weight per day (Ref. 3). This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting design basis LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.3% per day in the safety analysis at P_a = 14.8 psig (Ref. 3). Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of 10 CFR 50.36 (Ref. 4).

BASES

LCO

Containment OPERABILITY is maintained by limiting leakage to ≤ 1.0 L_a, except prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test. At this time, the applicable leakage limits must be met.

Compliance with this LCO will ensure a containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates specified for the containment air lock (LCO 3.6.2), purge valves with resilient seals, and reactor building bypass leakage (LCO 3.6.3) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the overall acceptance criteria of 1.0 L_a.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."

ACTIONS

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment OPERABLE during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

B.1 and B.2

If containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within

BASES

ACTIONS (continued)

36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. Failure to meet specific leakage limits for the air lock, secondary containment bypass leakage path, and purge valve with resilient seals (as specified in LCO 3.6.2 and LCO 3.6.3) does not invalidate the acceptability of the overall containment leakage determinations unless the specific leakage contribution to overall Type A, B, and C leakage causes one of these overall leakage limits to be exceeded. As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test is required to be < 0.6 L_a for combined Type B and C leakage, and $\leq$ 0.75 L_a for Option B for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq$ 1.0 L_a. At $\leq$ 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

The Surveillance is modified by three Notes.

Note 1 requires that the space between each duel-ply bellows assembly on containment penetrations between the containment building and the annulus be vented to the annulus during each Type A test.

Note 2 requires that following each Type A test, the space between each dual-ply bellows assembly be subjected to a low pressure leak test with no detectable leakage. Otherwise, the assembly must be tested with the containment side of the bellows assembly pressurized to P_a and meet the requirements of SR 3.6.3.8 (bypass leakage requirements).

Note 3 allows penetrations M372 and M373 to be tested without draining the glycol-water mixture from the associated diaphragm valves (NF-288A, NF-233B and NF-234A) as long as not leakage is indicated. This test may be used in lieu of 10 CFR 50, Appendix J, Option B as defined

BASES

SURVEILLANCE REQUIREMENTS (continued)

in ANSI/ANS 56.8-1994 Section 3.3.5 (Test Medium). The required test pressure and interval are not changed.

All test leakage rates shall be calculated using observed data converted to absolute values. Error analysis shall also be performed to select a balanced integrated leakage measurement system.

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. UFSAR, Chapter 15.
3. UFSAR, Section 6.2.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
1	1	4	4	4

OBJECTIVES

SEQ	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	Explain the following terms, as they apply to Technical Specifications: • Limiting Condition of Operation (LCO) • Applicability • Actions • Condition • Required Actions • Completion Time • Surveillance Requirements Table ADMTS001	X	X	X	X	X
2	Given a Technical Specification, apply the rules of Section 3.0 to determine the appropriate action(s). ADMTS002			X	X	X
3	If given a copy of Technical Specifications be able to determine the meaning of any of the various terms defined therein, or be able to recall from memory, the definition(s) of the following terms: • COLR (Core Operating Limits Report) • Leakage (including Identified, Unidentified, and Pressure Boundary Leakage) • OPERABLE -OPERABILITY ADMTS003	X	X	X	X	X

Objective # 2

d. **The phrase "unless otherwise stated" may require the completion of specific Required Actions even after the affected equipment has been restored to meet the LCO.**

Example:

Specification 3.4.3, "RCS Pressure and Temperature (P/T) Limits.

In this LCO, Condition A contains a NOTE which states that "Required Action A.2 shall be completed when this Condition is entered." Therefore, "Required Action A.2" must be performed even if compliance with the LCO has been restored.

e. **ACTIONS and Completion Times are applicable any time an LCO cannot be met and includes instances when ACTIONS are entered intentionally, such as for surveillance requirement performance. While considered acceptable, the intentional entry into ACTIONS is NOT intended to be used for operational convenience when alternate means are available. The imposition of these requirements will limit the time equipment or parameters can be out of service.**

f. **When a MODE change or specified condition is required to satisfy a Required Action, Completion Times for newly applicable LCO's apply from the time the Specification becomes applicable.**

Objective # 2

3.9.3 LCO 3.0.3

Whenever any of the following conditions exist, a reduction in MODE is necessary to place the plant in a known condition (*within the parameters utilized in our safety analyses*):

- **An LCO is not met and a plant condition exists that is not addressed by the associated Conditions (i.e., an associated ACTION is not provided)**
- **An LCO is not met and the associated Required Actions are not met.**
- **Or, if specifically directed by the Required Actions to place the Unit in a MODE or condition where the LCO is no longer applicable.**

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4. LCO 3.0.3 is not applicable in MODE 5 because:

- **MODE 5 is already the lowest operating MODE, and a change from MODE 5 to MODE 6 may not always be the most conservative action to take.**

LCO 3.0.3 is sometimes referred to as the "Motherhood" Statement. However, use of LCO 3.0.3 in lieu of the specified CONDITIONS and their associated REQUIRED ACTIONS / COMPLETION TIMES is not allowed. LCO 3.0.3 is intended to provide guidance when no other guidance exists.

Objective # 2

When in an LCO 3.0.3 Condition, the plant may be operating outside the boundary of the Safety Analysis and must be placed in a MODE where the subject equipment or parameter is not critical to plant safety. The times specified to reach each lower MODE are based on "time of discovery" and allow for a controlled orderly shutdown, within the maximum cooldown rate limitations, and include a 1-hour preparation period for Unit Shutdown.

If compliance with the original LCO is restored, completion of the LCO 3.0.3 Requirements is not required.

Exceptions to LCO 3.0.3 requirements are stated in the individual Specifications. Usually, where a reduction in MODE may not be conservative or, would not provide proper remedial measures.

Examples:

> **Specification 3.7.13, Spent Fuel Pool Water Level** – If the Spent Fuel Pool Water Level is <23 feet over the top of irradiated fuel assemblies, no safety *benefit is gained by placing the unit in a shutdown condition. However,* immediately suspending movement of irradiated fuel assemblies in the Spent Fuel Pool does provide an added safety benefit.
>
> **Specification 3.7.5, Auxiliary Feedwater System** – If all three Auxiliary Feedwater System Trains are INOPERABLE in MODE 1,2, or 3, no safety benefit is gained by placing the Unit in a shutdown condition. Instead, restoration of one Auxiliary Feedwater System Train to OPERABLE status is required prior to initiating a Unit Shutdown.

The following is a purpose of a Licensing Research Memorandum to Technical Specification (LCO) 3.0.3. Specifically, should the unit begin an actual reactor power reduction, i.e., add negative reactivity, no later than one hour after entering Tech Spec LCO 3.0.3. LCO 3.0.3 Bases states in part "Upon entering LCO 3.0.3, **1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation.** This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid..."

However, the technical specifications do not specify how allowed outage times (AOTs) or shutdown times are to be used; that is, when or how specific actions may be taken within those periods. Circumstances may arise when plant safety is better served by delaying a shutdown action to provide a safer configuration for a shutdown transient or to avoid an unnecessary shutdown transient. <u>If the licensee responsibly concludes that plant shutdown should be delayed or corrective actions can be accomplished so that an unnecessary plant transient can be avoided</u>, we believe that such a decision is permitted as long as the shutdown times specified by the Tech Spec are observed, including the default 3.0.3 provision...

1 Pt Which one of the following statements complies with the requirements of OMP 4-3 (*EP/AP Implementation Guidelines*) regarding the rules of usage for abnormal procedures (APs) when the emergency procedures (EPs) have been implemented?

A. APs may not be implemented when EPs have been entered.

B. Only one AP at a time may be implemented when EPs have been implemented. Concurrent implementation of APs when EPs are in use is not allowed.

C. APs may be implemented concurrently with EPs. However, the APs were written assuming that SI has not actuated and operators must be careful when using APs if SI has occurred.

D. APs may be implemented concurrently with EPs with the exception of events where SI has actuated. APs were written assuming the SI had not occurred and cannot be used if SI has actuated.

✓

Bank Question: 338.1 ***Answer: C***

1 Pt Which one of the following statements complies with the requirements of OMP 4-3 (*EP/AP Implementation Guidelines*) regarding the rules of usage for abnormal procedures (APs) when the emergency procedures (EPs) have been implemented?

A. APs may not be implemented when EPs have been entered.

B. Only one AP at a time may be implemented when EPs have been implemented. Concurrent implementation of APs when EPs are in use is not allowed.

C. APs may be implemented concurrently with EPs. However, the APs were written assuming that SI has not actuated and operators must be careful when using APs if SI has occurred.

D. APs may be implemented concurrently with EPs with the exception of events where SI has actuated. APs were written assuming the SI had not occurred and cannot be used if SI has actuated.

Distracter Analysis:

A. **Incorrect**: APs may be entered after EOPs have been started
Plausible: Many plants have this provision - symptomatic EOPs should address all significant safety challenges without requiring APs

B. **Incorrect**: No limitation on the number of APs
Plausible: Makes sense to limit the number of concurrent procedures in use

C. **Correct answer**

D. **Incorrect**: No explicit prohibition against use of APs when SI has actuated BUT there is a caution and the APs were written for the situation where SI has NOT occurred.
Plausible: APs were written for the situation where SI has NOT occurred.

LEVEL: SRO

SOURCE: BANK

LEVEL OF KNOWLEDGE: Memory

KA: G 2.4.5 (2.9/3.6)

REFERENCE: OMP 4-3 pages 21 & 22

2.4 Emergency Procedures /Plan

2.4.1 **Knowledge of EOP entry conditions and immediate action steps.**
(CFR: 41.10 / 43.5 / 45.13)
IMPORTANCE RO 4.3 SRO 4.6

2.4.2 **Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.**
(CFR: 41.7 / 45.7 / 45.8)
Note: The issue of setpoints and automatic safety features is not specifically covered in the systems sections).
IMPORTANCE RO 3.9 SRO 4.1

2.4.3 **Ability to identify post-accident instrumentation.**
(CFR: 41.6 / 45.4)
IMPORTANCE RO 3.5 SRO 3.8

2.4.4 **Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.**
(CFR 41.10 / 43.2 / 45.6)
IMPORTANCE RO 4.0 SRO 4.3

2.4.5 **Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions.**
(CFR: 41.10 / 43.5 / 45.13)
IMPORTANCE RO 2.9 SRO 3.6

2.4.6 **Knowledge symptom based EOP mitigation strategies.**
(CFR: 41.10 / 43.5 / 45.13)
IMPORTANCE RO 3.1 SRO 4.0

2.4.7 **Knowledge of event based EOP mitigation strategies.**
(CFR: 41.10 / 43.5 / 45.13)
IMPORTANCE RO 3.1 SRO 3.8

2.4.8 **Knowledge of how the event-based emergency/abnormal operating procedures are used in conjunction with the symptom-based EOPs.**
(CFR: 41.10 / 43.5 / 45.13)
IMPORTANCE RO 3.0 SRO 3.7

<u>GO TO</u>	-	discontinue use of present procedure and stay in the referenced procedure. The referenced procedure is always entered at the first step unless otherwise specified.
<u>REFER TO, PER</u>		user is directed to a supplemental procedure/enclosure for actions but will remain in the controlling procedure.
Stable	-	Maintained steady. <u>**IF**</u> a parameter is being controlled within a desired range, or if a slight trend in either direction is occurring, operator judgment may be used to determine if parameter is considered stable.
Evaluate	-	Appraise the situation. Includes taking action based on evaluation.

7.17 Tolerances

Ranges or tolerances are provided if it is important to maintain a parameter within a given band. <u>**WHEN**</u> a range or tolerance (e.g.: 5-15%) is provided, it is understood to mean extra attention should be paid to maintain the parameter within this range.

<u>**WHEN**</u> a single value is given, it is assumed the value is an ideal value. <u>**WHEN**</u> an ideal value (e.g.: at no load or 350 psig) is provided, it is understood to mean attention should be paid to maintain the parameter at the ideal value but <u>**NOT**</u> be overly concerned if the exact value is <u>**NOT**</u> achieved.

7.18 Multiple Use of EPs and APs.

The Control Room SRO will determine how many procedures can be implemented at a time and their priority based on manpower availability and the particular event in progress. More than one EP shall <u>**NOT**</u> be run concurrently unless directed by the procedure. Generally the use of AP's in conjunction with EP's should be avoided. In some instances it would be proper to use an AP concurrently during a major accident which is being addressed by the EP's. An example of this is upon loss of all Nuclear Service Water in the middle of an accident, the operators would need to utilize the AP for Loss of RN also. <u>**IF**</u> an AP is used during an SI event, USE CAUTION. AP's are generally written assuming an SI has <u>**NOT**</u> occurred (exception - AP/35, ECCS Actuation During Plant Shutdown). Evaluate any AP steps in post SI events to ensure the steps do <u>**NOT**</u> conflict with any EP in effect. <u>**NOT**</u> all AP actions would be appropriate if an SI occurred. (Enclosures in EP/G-1 (Generic Enclosures) may be used when reference by EPs or APs).

Use of most APs that have foldouts will likely be terminated when a reactor trip or SI occurs. There are a few APs with foldouts that could potentially be implemented concurrently with an EP though (Loss of VI or Loss of KC for example). Rules of foldout page use as specified in Section 7.13 should be applied in this situation also. Although unlikely, it is possible that the crew may have one EP foldout in effect at the same time as one of the AP foldouts. Implementation and priority of the AP foldouts will be evaluated as discussed in paragraph above.

ROs may be given procedure responsibilities when AP's and EP's or multiple AP's are in effect at the same time.

7.19 The STA/Operations Shift Manager Interface

The STA monitors the Critical Safety Functions (CSF) and otherwise ensures Core Safety through monitoring of activities and parameters. IF any one CSF is other than green, the STA will check whether the CSF non-green status is valid or being caused by an invalid input. IF the non-green CSF is invalid, the STA will notify the operating crew of the invalidity. For Red or Orange path procedures, the STA will immediately notify the operating crew that the condition exists and give the associated functional restoration procedure to the crew to implement as the controlling procedure. For Yellow path procedures, the STA will pull the functional restoration procedure and evaluate whether to implement the procedure, with the Operations Shift Managers concurrence, as time allows. This evaluation should consider whether the optimal recovery procedure is properly addressing the current plant conditions in as timely a manner as the functional restoration procedure.

Once status tree monitoring is initiated, the STA should monitor status trees continuously if an orange or red condition is found to exist. IF no condition more serious than yellow is found, monitoring frequency may be reduced to 10-20 minutes, unless some significant change in plant status occurs. Status tree monitoring may be performed using OAC SPDS display or EP/1(2)/A/5000/F-0 (Critical Safety Function Status Trees). IF the OAC SPDS display is being used, the STA will validate the OAC SPDS status every 10-20 minutes using control board indications.

IF the STA is NOT available, the Operations Shift Manager shall assume the STA responsibilities or delegate the STA responsibilities to another licensed operator.

7.20 Control Room Team Responsibilities During the Use of EP/APs

7.20.1 Operations Shift Manager - Responsibilities

7.20.1.1 Assume role of Emergency Coordinator upon activation of the Emergency Plan until properly relieved by the Station Manager.

1 Pt. Unit 2 is operating at 100% power. 2NI-59 (*Cold Leg Accumulator Check Valve*) begins to leak at 0200. Given the following accumulator indications:

Time	0200	0300	0400	0500
Level (%)	21%	31%	41%	51%
Pressure (psig)	586	613	640	667
Boron (ppm)	2485	2470	2455	2440

When does the accumulator first exceed a limiting condition for operation?

REFERENCES PROVIDED:

A. **0200**

B. **0300**

C. **0400**

D. **0500**

Bank Question: 697.3 ***Answer: B***

1 Pt. Unit 2 is operating at 100% power. 2NI-59 (*Cold Leg Accumulator Check Valve*) begins to leak at 0200. Given the following accumulator indications:

Time	0200	0300	0400	0500
Level (%)	21%	31%	41%	51%
Pressure (psig)	586	613	640	667
Boron (ppm)	2485	2470	2455	2440

When does the accumulator first exceed a limiting condition for operation?

REFERENCES PROVIDED:
Tech Spec 3.5.1
Unit 1 Data Book curve 7.4
Unit 1 Cycle 16 COLR – page 24

A. **0200**

B. **0300**

C. **0400**

D. **0500**

Distracter Analysis: Tech Spec values for CLA parameters are:
Volume ≥ 6870 (12.3%) but ≤ 7342 *gal* (38.7%) – exceeded at 0400 (high)
Pressure ≥ 585 but ≤ 639 psig – exceeded at 0400
Boron concentration ≥ 2475 ppm but ≤2875 ppm – exceeded at 0300

A. **Incorrect Answer**: nothing out of spec.

B. **Correct**: first exceeds Boron concentration (<2475 ppm) at 0300
Plausible:

C. **Incorrect**: the LCO is first exceeded at 0300 on low boron concentration.
Plausible:

D. **Incorrect**: the LCO is first exceeded at 0300 on low boron concentration
Plausible:

Level: SRO

KA: G2.1.25 (2.8/3.1)

Lesson Plan Objective: ECC-CLA Obj. 7

Source: Bank

Level of knowledge: comprehension

References:
1. OP-MC-ECC-CLA page 23
2. Tech Spec 3.5.1 - PROVIDED
3. Unit 1 Data Book Curve 7.4 - PROVIDED
4. Unit 1 Cycle 15 COLR - PROVIDED

SYS 16 NNI

2.1 Conduct of Operations (continued)

2.1.18 **Ability to make accurate, clear and concise logs, records, status boards, and reports.**
(CFR: 45.12 / 45.13)
IMPORTANCE RO 2.9 SRO 3.0

2.1.19 **Ability to use plant computer to obtain and evaluate parametric information on system or component status.**
(CFR: 45.12)
IMPORTANCE RO 3.0 SRO 3.0

2.1.20 **Ability to execute procedure steps.**
(CFR: 41.10 / 43.5 / 45.12)
IMPORTANCE RO 4.3 SRO 4.2

2.1.21 **Ability to obtain and verify controlled procedure copy.**
(CFR: 45.10 / 45.13)
IMPORTANCE RO 3.1 SRO 3.2

2.1.22 **Ability to determine Mode of Operation.**
(CFR: 43.5 / 45.13)
IMPORTANCE RO 2.8 SRO 3.3

2.1.23 **Ability to perform specific system and integrated plant procedures during all modes of plant operation.**
(CFR: 45.2 / 45.6)
IMPORTANCE RO 3.9 SRO 4.0

2.1.24 **Ability to obtain and interpret station electrical and mechanical drawings.**
(CFR: 45.12 / 45.13)
IMPORTANCE RO 2.8 SRO 3.1

2.1.25 **Ability to obtain and interpret station reference materials such as graphs, monographs, and tables which contain performance data.**
(CFR: 41.10 / 43.5 / 45.12)
IMPORTANCE RO 2.8 SRO 3.1

2.1.26 **Knowledge of non-nuclear safety procedures (e.g. rotating equipment, electrical, high temperature, high pressure, caustic, chlorine, oxygen and hydrogen).**
(CFR: 41.10 / 45.12)
IMPORTANCE RO 2.2 SRO 2.6

OBJECTIVES

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
7	Concerning the Technical Specifications related to the CLA System:					
	• Given the LCO title, state the LCO (including any COLR values) and applicability.			X	X	X
	• For any LCO's that have action required within one hour, state the action.			X	X	X
	• Given a set of parameter values or system conditions, determine if any Tech Spec LCO's is (are) not met and any action(s) required within one hour.			X	X	X
	• Discuss the surveillance requirement for tank level change for unknown reasons.			X	X	X
	• Given a set of plant parameters or system conditions and the appropriate Tech Specs, determine the required action.			X	X	X
	• Discuss the bases for a given Tech Spec LCO or Safety Limit.				X	*
	* SRO ONLY					

3.2.2. CLA "A" and "B" nitrogen backup to NC PORV's

Normally all three PORV's are operated by air supplied from VI, however under certain circumstances the nitrogen overpressure from CLA "A" and "B" will provide the motive force for all three PORV's. Anytime the low pressure mode is selected on the NC-32B key switch and WR Loop "C" T_{cold} is < 320 °F, NI-431B will open allowing nitrogen from CLA "B" to provide the motive force for NC-32B and NC-36B if VI pressure fails. Note that only the NC-32B setpoint is reduced to ≤ 400 psig and operates off of WR Loop "C" pressure.

Anytime the low pressure mode is selected on the NC-34A key switch and WR Loop "D" T_{hot} is less than 320 °F, NI-430A will open allowing nitrogen from CLA "A" to provide the motive force for NC-34A if VI, pressure fails (setpoint ≤ 400 psig from WR Loop "D" pressure). NI-430A and NI-431B can also be opened at anytime with the control room manual operator.

NOTE: Minimum CLA Accumulator to maintain LTOP operability is 200 psig.

4.0 TECHNICAL SPECIFICATIONS

Objective #7

4.1 Tech Spec 3.5.1 ECCS Accumulators

Allowed outage times are variable based on boron concentration to ensure that the reactor is shut down following a LOCA and that any problems are corrected in a timely manner. The minimum required to ensure post-LOCA subcriticality is based on nominal accumulator volume conditions and allows additional time since subcriticality is assured when the boron concentration is above this value. The lower limit for 3.5.1.c is based on worst case liquid mass, boron concentration and measurement errors.

The **accumulator isolation valves** are considered to be operating bypasses. Bypasses of a protective function are required to be automatically removed whenever permissive conditions are not met. In addition, as these valves fail to meet single failure criteria, removal of power to the valves is required. The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with the failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be opened immediately, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

4.2 Tech Spec 3.4.14, RCS Pressure Isolation Value (PIV) Leakage

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Accumulators

LCO 3.5.1 Four ECCS accumulators shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS pressure > 1000 psig.

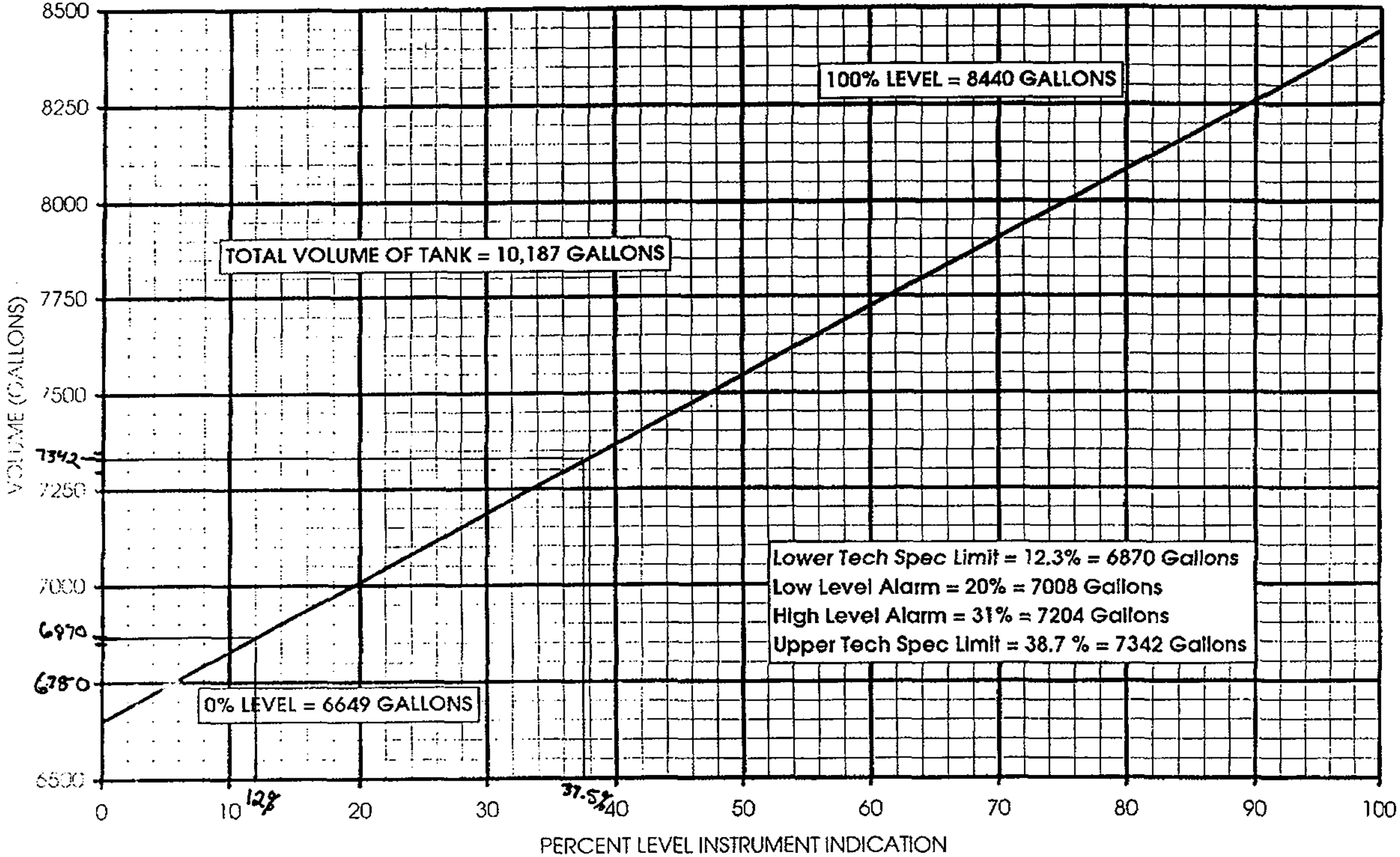
ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One accumulator inoperable due to boron concentration not within limits.	A.1 Restore boron concentration to within limits.	72 hours
B. One accumulator inoperable for reasons other than Condition A.	B.1 Restore accumulator to OPERABLE status.	1 hour
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Reduce RCS pressure to $\leq$ 1000 psig.	6 hours 12 hours
D. Two or more accumulators inoperable.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each accumulator isolation valve is fully open.	12 hours
SR 3.5.1.2	Verify borated water volume in each accumulator is ≥ 6870 gallons and ≤ 7342 gallons.	12 hours
SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is ≥ 585 psig and ≤ 639 psig.	12 hours
SR 3.5.1.4	Verify boron concentration in each accumulator is within the limits specified in the COLR.	31 days AND -------NOTE-------- Only required to be performed for affected accumulators ---------------------- Once within 6 hours after each solution volume increase of ≥ 1% of tank volume that is not the result of addition from the refueling water storage tank
SR 3.5.1.5	Verify power is removed from each accumulator isolation valve operator when RCS pressure is > 1000 psig.	31 days

OP/1/A/6100/22
ENCLOSURE 4.3
CURVE 7.4
COLD LEG ACCUMULATOR
(VOLUME vs. TANK LEVEL)



This data is also available on the OAC.

McGuire 1 Cycle 16 Core Operating Limits Report

2.10 Accumulators (TS 3.5.1)

2.10.1 Boron concentration limits during modes 1 and 2, and mode 3 with RCS pressure >1000 psi:

Parameter	Limit
Cold Leg Accumulator minimum boron concentration.	2,475 ppm
Cold Leg Accumulator maximum boron concentration.	2,875 ppm

2.11 Refueling Water Storage Tank - RWST (TS 3.5.4)

2.11.1 Boron concentration limits during modes 1, 2, 3, and 4:

Parameter	Limit
Refueling Water Storage Tank minimum boron concentration.	2,675 ppm
Refueling Water Storage Tank maximum boron concentration.	2,875 ppm

1 Pt. Unit 1 was operating at 100% when a large break LOCA with loss of offsite power occurs. One diesel generator fails to start. The operators are entering E-1 *(Loss of Reactor or Secondary Coolant).*

Given the following critical safety function status indications:

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

Which one of the following describes the highest priority problem, and the appropriate operator action?

A. **Integrity; Transition to FR-P.1, *(Response to Imminent Pressurized Thermal Shock).***

B. **Core cooling; Transition to FR-C.1, *(Response to Inadequate Core Cooling).***

C. **Heat Sink; Transition to FR-H.1, *(Response to Loss of Secondary Heat Sink).***

D. **Containment; Transition to FR-Z.1, *(Response to High Containment Pressure).***

Bank Question: 776.1 ***Answer: B***

1 Pt. Unit 1 was operating at 100% when a large break LOCA with loss of offsite power occurs. One diesel generator fails to start. The operators are entering E-1 *(Loss of Reactor or Secondary Coolant)*.

Given the following critical safety function status indications:

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

Which one of the following describes the highest priority problem, and the appropriate operator action?

A. **Integrity; Transition to FR-P.1, *(Response to Imminent Pressurized Thermal Shock).***

B. **Core cooling; Transition to FR-C.1, (*Response to Inadequate Core Cooling).***

C. **Heat Sink; Transition to FR-H.1, *(Response to Loss of Secondary Heat Sink).***

D. **Containment; Transition to FR-Z.1, *(Response to High Containment Pressure).***

Distracter Analysis:

A. **Incorrect**: Core Cooling is the highest priority RED
Plausible:
B. **Correct**:
Plausible:
C. **Incorrect**: core cooling is the highest priority RED
D. **Incorrect**: core cooling is the highest priority RED
Plausible:.

Level: SRO

KA: WE 07 EA 2.1 (3.2/4.0)

Lesson Plan Objective: EP-F0 SEQ 2, 3

Source: Bank

Level of knowledge: Memory

References:
1. OP-MC-EP-F0 pages 13, 15
2. OMP 4-3 pages 15-16

EPE: 07 Saturated Core Cooling (Continued)

K/A NO. KNOWLEDGE

EK3.2 Normal, abnormal and emergency operating procedures associated with (Saturated Core Cooling).
IMPORTANCE RO 3.2 SRO 3.7

EK3.3 Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations.
IMPORTANCE RO 3.8 SRO 3.6

EK3.4 RO or SRO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated.
IMPORTANCE RO 3.3 SRO 3.6

ABILITY

EA1. Ability to operate and / or monitor the following as they apply to the (Saturated Core Cooling)
(CFR: 41.7 / 45.5 / 45.6)

EA1.1 Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
IMPORTANCE RO 3.6 SRO 3.6

EA1.2 Operating behavior characteristics of the facility.
IMPORTANCE RO 3.2 SRO 3.7

EA1.3 Desired operating results during abnormal and emergency situations.
IMPORTANCE RO 3.5 SRO 3.9

EA2. Ability to determine and interpret the following as they apply to the (Saturated Core Cooling)
(CFR: 43.5 / 45.13)

EA2.1 Facility conditions and selection of appropriate procedures during abnormal and emergency operations.
IMPORTANCE RO 3.2 SRO 4.0

EA2.2 Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.
IMPORTANCE RO 3.3 SRO 3.9

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
		2	2	2

OBJECTIVES

SEQ	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	State the purpose of each of the six CSF Status Trees. EPF0001			X	X	
2	Explain the priority system associated with the CSF status trees. EPF0002			X	X	X
3	Explain the "Rules of Usage" for Critical Safety Function status trees. EPF0003			X	X	X
4	Explain the bases for all blocks in the six Status Trees. EPF0004			X	X	X

2.0 PROCEDURE SERIES BACKGROUND (continued)

Once the Status Trees are being monitored, the following rules of usage apply:

1. The Status Trees should be continuously monitored in order of Critical Safety Function priority.
2. CSF procedures are not to be implemented prior to transition from E-0, Reactor Trip or Safety Injection. If a CSF path is red or orange while the operating crew is in E-0, but has turned to green upon transition from E-0, the CSF procedure, which was in alarm, shall not be implemented. If the CSF path is yellow, it shall be handled as any other yellow path procedure. If there are any valid red or orange path CSFs on transition from E-0 (unless the transition is to ECA-0 (Loss of All AC Power), the associated CSF procedure shall be implemented.
3. If a valid red or orange path flickers into alarm on SPDS but returns to green prior to the crew validating the condition and implementing the procedure (implementation of procedure being that the SRO either hands out fold-out pages or starts reading from the procedure), the CSF procedure shall not be implemented. If the CSF path is yellow, it shall be handled as any other yellow path procedure. Likewise, if a valid red path or orange path goes into alarm during performance of a higher priority CSF procedure, but returns to green prior to transition from the higher priority CSF path procedure to the lower priority CSF procedure, the associated CSF procedure shall not be implemented. If the CSF path is yellow, it shall be handled as any other yellow path procedure.
4. If a CSF procedure directs the operator to return to the procedure and step in effect, AND the corresponding status tree continues to display the off-normal conditions, THEN the corresponding CSF procedure doesn't have to be implemented again, since all recovery actions have been completed. However, if the same status tree subsequently changes to a valid higher priority condition, (OR if it changes to lower condition and returns to higher priority condition again), THEN the corresponding CSF procedure shall be implemented as required by its priority.
5. Once status tree monitoring is initiated, the STA should monitor status tree continuously if an orange or red path condition exists. If no condition more serious than yellow is found, monitoring frequency may be reduced to 10 – 20 minutes unless some significant change in plant status occurs. Status tree monitoring may be performed using the OAC SPDS display or F-0 (Critical Safety Function Status Trees). If the OAC SPDS display is being used, the STA will validate the OAC SPDS status every 10 – 20 minutes using control board indications. If the STA is not available, the OSM shall assume the STA responsibilities or delegate the STA responsibilities to another licensed operator.

6. Red Path

If any valid red path is encountered during monitoring, the operator is required to immediately implement the corresponding EP. Any recovery EP previously in progress shall be discontinued. If during the performance of any red path procedure, a valid red condition of higher priority arises, the higher priority condition should be addressed first, and the lower priority red path procedure suspended.

7. Orange Path

If any valid orange path is encountered, the operator is expected to scan all of the remaining trees, and then, if no valid red is encountered, promptly implement the corresponding EP. If during the performance of an orange path procedure, any valid red condition or higher priority valid orange condition arises, then the red or higher priority orange condition is to be addressed first, and the original orange path procedure suspended.

Once a procedure is entered due to a red or orange condition, that procedure should be performed to completion, unless preempted by some higher priority condition. It is expected that the actions in the procedure will clear the red or orange condition before all the operator actions are complete. However, these procedures should be performed to the point of the defined transition to a specific procedure or to the "procedure and step in effect" to ensure the condition remains clear. At this point any lower priority red or orange paths currently indicating or previously started but not completed shall be addressed.

7.14.3 The configuration control cards filled out in Steps 7.14.1 and 7.14.2 shall be handled per the following two situations:

- Without operational support center activation

  The configuration control card will be handled by Ops shift per OMP 7-1 (Removal and Restoration (R&R) Requirements).

- With operational support center activation

  WHEN the OSC is activated, Operations will report to the OSC and shall bring with them all configuration control cards that have been filled out. The cards taken to the OSC shall be given to the OPS SRO in the OSC. For handling cards in the OSC, refer to RP/0/A/5700/020 (Activation of the Operations Support Center).

7.15 Usage of Status Trees

There are six different trees, each one evaluating a separate Critical Safety Function of the plant. Color-coding of the status tree end points will be either red, orange, yellow, or green, with green representing a "satisfied" safety status. Each nongreen color represents an action level that should be addressed according to the Rules of Priority as discussed below.

The six Status Trees are always evaluated in the sequence:

- Subcriticality
- Core Cooling
- Heat Sink
- Integrity
- Containment
- Inventory

IF identical color priorities are found on different trees during monitoring, the required action priority is determined by this sequence.

Initial monitoring of the status trees should begin on either of the following conditions:

- As directed by an action step in EP/1,2/A/5000/E-0 (Reactor Trip or Safety Injection).
- WHEN a transfer is made out of the Safety Injection procedure to another EP.

An exception to this is that no status tree monitoring is required during the Loss of All AC Power EP since none of the electrically powered safeguards equipment can be used. **WHEN** power is subsequently restored, EP/1,2/A/5000/ECA-0.1 or 0.2 (Loss of All AC Power Recovery procedures) will direct the operator when monitoring of status trees is required.

7.15.1 Implementing CSF Path Procedures

7.15.1.1 CSF procedures are **NOT** to be implemented prior to transition from EP/1,2/A/5000/E-0 (Reactor Trip or Safety Injection). **IF** a CSF path is red or orange while the operating crew is in EP/1,2/A/5000/E-0, but has turned to green upon transition from E-0, the CSF procedure which was in alarm shall **NOT** be implemented. **IF** the CSF path is yellow, it shall be handled as any other yellow path procedure per Section 7.15.1.7. **IF** there are any valid red or orange path CSF's on transition from E-0 (unless transition is to EP/1,2/A/5000/ECA-0 (Loss of All AC Power), the associated CSF procedure shall be implemented.

7.15.1.2 **IF** a valid red or orange path flickers into alarm on SPDS but returns to green prior to the crew validating the condition and implementing the procedure (implementation of procedure being that the SRO either hands out fold-out pages or starts reading from the procedure), the CSF procedure shall **NOT** be implemented. **IF** the CSF path is yellow, it shall be handled as any other yellow path procedure per Section 7.15.1.7. Likewise, if a valid red path or orange path goes into alarm during performance of a higher priority CSF procedure, but returns to green prior to transition from the higher priority CSF path procedure to the lower priority CSF procedure, the associated CSF procedure shall **NOT** be implemented. **IF** the CSF path is yellow, it shall be handled as any other yellow path procedure per Section 7.15.1.7.

7.15.1.3 **IF** a CSF procedure directs the operator to return to the procedure and step in effect, AND the corresponding status tree continues to display the offnormal conditions, the corresponding CSF procedure doesn't have to be implemented again, since all recovery actions have been completed. However, if the same status tree subsequently changes to a valid higher priority condition, (OR if it changes to lower condition and returns to higher priority condition again), the corresponding CSF procedure shall be implemented as required by its priority.

7.15.1.4 Red Path

IF any valid red path is encountered during monitoring, the operator is required to immediately implement the corresponding EP. Any recovery EP previously in progress shall be discontinued. IF during the performance of any red path procedure, a valid red condition of higher priority arises, the higher priority condition should be addressed first, and the lower priority red path procedure suspended.

7.15.1.5 Orange Path

IF any valid orange path is encountered, the operator is expected to scan all of the remaining trees, and then, if no valid red is encountered, promptly implement the corresponding EP. IF during the performance of an orange path procedure, any valid red condition or higher priority valid orange condition arises, the red or higher priority orange condition is to be addressed first, and the original orange path procedure suspended.

7.15.1.6 Completion of red or orange path procedure

Once procedure is entered due to a red or orange condition, that procedure should be performed to completion, unless preempted by some higher priority condition. It is expected that the actions in the procedure will clear the red or orange condition before all the operator actions are complete. However, these procedures should be performed to the point of the defined transition to a specific procedure or to the "procedure and step in effect" to ensure the condition remains clear. At this point any lower priority red or orange paths currently indicating or previously started but NOT completed shall be addressed.

FR-S.1, P.1 and Z.1 can be entered from either an orange or red path status. IF the color changes from orange to red while you are in one of these EPs, the crew should continue and complete the EP from where they are. Crew does NOT have to backup and restart the EP. IF you exit the orange path, and it subsequently turns red, the EP must be reentered at Step 1.

1 Pt. Unit 1 is operating at 100% power when the OAC registers a low spent fuel pool level alarm. Given the following events and conditions:

- The operators read –2.1 ft SFP level and stable on the main control board.
- The operating KF pump has tripped.
- An NLO reports a large leak in the auxiliary building has stopped.
- Normal SFP makeup is not available.

Which one of the following statements correctly describes the corrective action for this event?

A. **Implement AP/1/A/5500/41 (*Loss of Spent Fuel Cooling or Level*), find and isolate the leak on the KF discharge piping.**

B. **Implement AP/1/A/5500/41 (*Loss of Spent Fuel Cooling or Level*) Find and isolate the leak on the KF suction piping.**

C. **Implement AP/1/A/5500/40 (*Loss of Refueling Canal Level*), and initiate assured makeup due a leak on the discharge piping.**

D. **Implement AP/1/A/5500/40 (*Loss of Refueling Canal Level*), and initiate assured makeup due to a leak on the suction piping.**

Bank Question: 892.3 ***Answer: A***

1 Pt. Unit 1 is operating at 100% power when the OAC registers a low spent fuel pool level alarm. Given the following events and conditions:

- *The operators read –2.1 ft SFP level and stable on the main control board.*
- The operating KF pump has tripped.
- An NLO reports a large leak in the auxiliary building has stopped.
- Normal SFP makeup is not available.

Which one of the following statements correctly describes the corrective action for this event?

A. Implement AP/1/A/5500/41 (*Loss of Spent Fuel Cooling or Level*), find and isolate the leak on the KF discharge piping.

B. Implement AP/1/A/5500/41 (*Loss of Spent Fuel Cooling or Level*) Find and isolate the leak on the KF suction piping.

C. Implement AP/1/A/5500/40 (*Loss of Refueling Canal Level*), and initiate assured makeup due a leak on the discharge piping.

D. Implement AP/1/A/5500/40 (*Loss of Refueling Canal Level*), and initiate assured makeup due to a leak on the suction piping.

Distracter Analysis:

A. Correct:
B. Incorrect: The leak is on the discharge piping.
Plausible: If the candidate confuses the piping immersion depth with the suction pipes, which are at 4 feet.
C. Incorrect:
Plausible:.
D. Incorrect: Do not use the assured source, and the leak is on the discharge piping.
Plausible:.

Level: SRO

KA: SYS 033 A2.02(2.7/3.0)

Lesson Plan Objective: OP-MC-FH-KF Obj. 4/5/14

Source: BANK

Level of knowledge: Comprehension

References:
1. OP-MC-FH-KF pages 19, 23, 49

SYSTEM: 033 Spent Fuel Pool Cooling System (SFPCS)

K5 **Knowledge of the operational implication of the following concepts as they apply to the Spent Fuel Pool Cooling System:**
(CFR: 41.5 / 45.7)

K5.01	Pump theory	1.6	1.9
K5.02	Heat transfer	1.7	1.9
K5.03	D/P detector theory of OPS	1.5	1.6
K5.04	K-eff	2.1	2.3*
K5.05	Decay heat	2.1	2.3
K5.06	Shielding	2.1	2.5

K6 **Knowledge of the effect of a loss or malfunction on the following will have on the Spent Fuel Pool Cooling System:**
(CFR: 41.7 / 45.7)

K6.01	Pumps	1.7	1.9
K6.02	Heat exchangers	1.8	1.9
K6.03	Valves	1.7	1.7
K6.04	Motors	1.7	1.7
K6.05	Pressure and pressure detectors	1.7	1.7
K6.06	Temperature sensors	1.8	1.8
K6.07	Filters and demineralizers	1.7	1.8

<u>ABILITY</u>

A1 **Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with Spent Fuel Pool Cooling System operating the controls including:**
(CFR: 41.5 / 45.5)

A1.01	Spent fuel pool water level	2.7	3.3
A1.02	Radiation monitoring systems	2.8	3.3
A1.03	SFPCS controls and sensors	2.4	2.7

A2 **Ability to (a) predict the impacts of the following malfunctions or operations on the Spent Fuel Pool Cooling System ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:**
(CFR: 41.5 / 43.5 / 45.3 / 45.13)

A2.01	Inadequate SDM	3.0	3.5
[redacted]			
A2.03	Abnormal spent fuel pool water level or loss of water level	3.1	3.5

A3 **Ability to monitor automatic operation of the Spent Fuel Pool Cooling System including:**
(CFR: 41.7 / 45.5)

A3.01	Temperature control valves	2.5*	2.7*
A3.02	Spent fuel leak or rupture	2.9	3.1

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
2	2	2	2	2

OBJECTIVES

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	State the purpose of the Spent Fuel Pool Cooling System.	X	X	X	X	
2	Draw a simplified diagram of the Spent Fuel Pool Cooling System (including all major components) per Training Drawing 7.1, Spent Fuel Pool Cooling System - Simplified.	X	X	X	X	
3	State the flowrates through each of the following flowpaths: • Spent Fuel Pool Cooling Loop • Spent Fuel Pool Purification Loop • Spent Fuel Pool Skimmer Loop	X	X	X	X	
4	List the sources of makeup to the Spent Fuel Pool Cooling System; including the source grade (i.e., borated, non-borated demineralized, and non-borated lake water).	X	X	X	X	
5	Explain the conditions which would require *"assured makeup"*, from the Nuclear Service Water System, to the Spent Fuel Pool Cooling System.	X	X	X	X	X
6	List the power supply for the following Spent Fuel Pool Cooling System Pumps (Unit 1 and Unit 2): • KF Pump(s) • KF Skimmer Pump(s)	X	X	X	X	
7	Describe the controls, indications, and/or alarms, associated with Spent Fuel Pool Cooling System operation, located <u>within</u> the <u>Control Room</u>.		X	X	X	X
8	Describe how the KF Pump motor(s) is cooled during system operation.	X	X	X	X	
9	State the cooling medium for the Spent Fuel Pool Cooling System Heat Exchanger(s).	X	X	X	X	
10	Describe the controls, indications, and/or alarms, associated with Spent Fuel Pool Cooling System operation, located <u>outside</u> the <u>Control Room</u>.	X	X	X	X	

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
11	Given a Limit and/or Precaution, associated with operation of the Spent Fuel Pool Cooling System, discuss it's basis and applicability.		X	X	X	X
12	Describe the Spent Fuel Pool Cooling System response to a Safety Injection Signal and/or a Blackout Signal.		X	X	X	X
13	Concerning AP/1(2)/A/5500/25, Spent Fuel Damage: • State the purpose of AP/5500/25. • Recognize the symptoms that would require implementation of the AP.		X	X	X	X
14	Concerning AP/1(2)/A/5500/41, Loss of Spent Fuel Pool Cooling or Level: • State the purpose of AP/5500/41. • Given a set of symptoms determine which case of AP/5500/41 should be implemented.		X	X	X	X
15	Concerning the Technical Specifications related to the Spent Fuel Pool Cooling System:					
	• Given the LCO title, state the LCO (including any COLR values) and applicability.			X	X	X
	• For any LCO's that have action required within one hour, state the action.			X	X	X
	• Given a set of parameter values or system conditions, determine if any Tech. Spec. LCO's is (are) not met and any action(s) required within one hour.			X	X	X
	• Given a set of plant parameters or system conditions and the appropriate Tech. Specs, determine the required action(s).			X	X	X
	• Discuss the bases for a given Tech. Spec. LCO or Safety Limit.				X	*
	*** SRO ONLY**					

A table within *OP/1(2)/A/6200/05, Spent Fuel Cooling System* provides a conservative addition for makeup to the Spent Fuel Pool using Demineralized Water based upon the last boron sample:

Last Boron Sample	Demin Water Addition to Maintain Boron Concentration >2675 ppm
2700 ppm	2 inches
2725 ppm	5 inches
2750 ppm	7 inches
2775 ppm	10 inches
2800 ppm	12 inches
2825 ppm	15 inches
2850 ppm	18 inches
2875 ppm	20 inches
2900 ppm	23 inches
2925 ppm	25 inches
2950 ppm	28 inches

Borated water, from the Refueling Water Storage Tank (FWST), can be used for makeup and should be considered the *preferred source* of makeup *unless* the *last SFP boron sample indicated >2775 ppm.*

Objective # 4 & 5

Non-borated lake water *(Assured Makeup)*, from the Nuclear Service Water System (RN), can be used for makeup. The *Assured Makeup* **should only be used if borated and demineralized water are not available for makeup** *and* **the Spent Fuel Pool Level is low enough to cause a** *radiation hazard to employees or the public.*

Electrical Power Supply

Objective # 6

Each Spent-Fuel Pool Cooling Pump receives power from its respective *Essential Bus, 1(2)ETA* **(4160V)** *or 1(2)ETB* **(4160V)** *(the same buses that can be powered by the Emergency Diesel Generators).*

Each KF AHU receives power from its respective *Essential Motor Control Center, 1(2) EMXA* **(600V)** *or 1(2) EMXB* **(600V).**

The Fuel Pool Skimmer Pump receives power from one of the normal station buses *1(2) MXK* **(600V)** and can be operated anytime the bus is energized. However, this pump is not required during emergency operations.

Motor-operated valve 1(2)KF-12, Purification Loop Isolation Valve, is also powered from a normal station bus *1(2) MXJ* **(600V)** and can be operated remotely anytime the bus is energized. *Manual operation of this valve can be performed if power is unavailable.*

The Spent Fuel Pool stores fuel assemblies approximately 33 feet 4 inches below the fuel pool operating deck with approximately 25 feet of borated water above the top of each fuel assembly.

Objective # 7

Control Room Indication is provided for Spent Fuel Level and Temperature. *(Refer to Training Drawing 7.3, Spent Fuel Pool Control Room Indication.)*

2.2 Spent Fuel Pool Cooling Pumps

Objective # 7

Two Spent Fuel Pool Cooling Pumps (KF Pumps) are provided for each Unit. The controls and indications, associated with Spent Fuel Pool Cooling Pump operation, located on the Main Control Board (MC-11), consist of the following:

- START / STOP Control Switch

  These momentary START / STOP pushbuttons allow the operator to START and STOP the pump, as desired.

  During a Station Blackout the KF Pump(s) will initially lose power *(load shed)* but receive a *manual start permissive* when Load Group 9 is loaded onto the bus.

  During a Safety Injection Signal, the KF Pump(s) running prior to SI will continue to run. The KF Pump(s) *not running,* prior to SI, will receive a *manual start permissive* when Load Group 9 is loaded onto the bus.

  Any KF Pump(s) running or manually started, while the SI Signal is present, **cannot** be stopped until the *SI Signal is RESET.*

- ON / OFF (Red / Green) Indicating Lights

  These ON / OFF (Red / Green) indicating lights are mounted on the START / STOP Control Switch and provide indication when the KF Pump breaker is CLOSED (ON) or OPEN (OFF).

Each pump is designed for 2310 gpm flow and each takes suction from the Spent Fuel Pool, *four feet below pool level,* and discharge back into the Spent Fuel Pool, *six feet above the fuel assemblies.* Holes drilled into the Spent Fuel Pool Discharge Header act as a vacuum breaker and limit siphon draining to two feet below normal Spent Fuel Pool level.

In addition, each KF Pump is designed to circulate water through the *cooling* and *purification loop* at the same time. Under normal operating conditions only one pump is utilized to supply flow through ***only one*** *cooling loop* and the *purification loop.* During two pump operation each pump will supply flow through a *cooling loop* but one pump will be selected to provide flow for the *purification loop.* One pump can also be aligned to both heat exchangers, with or without supplying flow to the purification loop.

Each pump has mechanical seals provided with leakoff, vent, and drain connections. The internal wetted surfaces of these pumps are made of stainless steel.

Objective # 13

Abnormal Operating Procedure AP/1(2)/A/5500/25, Spent Fuel Damage, is provided to identify operator actions required during a spent fuel damage event. Actions are defined for spent fuel damage inside Containment or within the Spent Fuel Pool. This procedure has only a single Case and the Symptoms are:

- EMF-36, Unit Vent High Gas Radiation Alarm (Process Monitor)
- EMF-38, Containment High Particulate Radiation Alarm (Process Monitor)
- EMF-39, Containment High Gas Radiation Alarm (Process Monitor)
- EMF-40, Containment High Iodine Radiation Alarm (Process Monitor)
- EMF-42, Fuel Handling High Gas Radiation Alarm (Process Monitor)
- EMF-16, Containment Refueling Bridge Alarm (Area Monitor)
- EMF-17, Spent Fuel Building Bridge Alarm (Area Monitor)
- Gas bubbles originating from the damaged assembly(ies).
- Visual evidence of damage with potential of radioactive release(s).

Subsequent operator action(s) will first determine the damaged fuel location. The area affected (Containment or the Spent Fuel Pool) must be evacuated and isolated. Those personnel evacuated must be assembled for accountability while remote action(s) are performed to further secure the event to ON-SITE. In addition, the event must be classified and implementation of the Emergency Plan initiated, if required.

Objective # 14

Abnormal Operating Procedure AP/1(2)/A/5500/41, Loss of Spent Fuel Cooling or Level, is provided to identify operator actions required during a loss of cooling or level event. This procedure has two Cases; Loss of Spent Fuel Cooling and Loss of Spent Fuel Level.

Symptoms for **Case I, Loss of Spent Fuel Cooling** are:

- Spent Fuel Pool Temperature High Alarm on the OAC (Operator Aid Computer)
- Both KF Pumps OFF

Subsequent operator action(s) first determine if *fuel movement* **or** *movement of any radioactive component,* within the Spent Fuel Pool, is taking place. If either of the above is true, then the fuel or component must be placed into a safe position, such as; *lowering any fuel assembly within the Spent Fuel Manipulator Crane to the "fully down" position* **OR** *lowering any radioactive component to "fully down"* **OR** *lowering any fuel assembly within the Upender to the "fully down" position.* Then a systematic check of KC flow (through the KF Heat Exchanger) is performed, including KC alignment and KC Pump operation.

IF BOILING should occur, then makeup in accordance with OP/1(2)/A/6200/05, Spent Fuel Cooling System, Enclosure 4.4:

- Demineralized Water
- Borated Water from the FWST
- Assured Makeup from the RN System

1 Pt. Unit 2 is operating at 100 % power. Given the following events and conditions:

- "B" essential train is in service.
- 2A RN train is in operation for testing.
- The RN trains are split with 2RN-41B *(TRAIN B TO NON-ESS HDR ISOL)* closed.

Which one of the following statements correctly describes the potential consequence if 2RN-190B *(RN TO B KC HX CONTROL)* failed to perform its automatic function?

A. Overheating 2B RN pump.

B. Flashing in the 2B KC heat exchanger.

C. Overheating the running B train KC pumps.

D. 2RN-41B will open to restore flow to the heat exchanger.

Bank Question: 894 .1 **Answer: A**

1 Pt. Unit 2 is operating at 100 % power. Given the following events and conditions:

- "B" essential train is in service.
- 2A RN *train is in operation for testing.*
- The RN trains are split with 2RN-41B *(TRAIN B TO NON-ESS HDR ISOL)* closed.

Which one of the following statements correctly describes the potential consequence if 2RN-190B *(RN TO B KC HX CONTROL)* failed to perform its automatic function?

A. Overheating 2B RN pump.

B. Flashing in the 2B KC heat exchanger.

C. Overheating the running B train KC pumps.

D. 2RN-41B will open to restore flow to the heat exchanger.

Distracter Analysis:

A. **Correct**: Lose mini-flow protection for RN pump 2B.

B. **Incorrect**: no flashing should occur, pressure is not changing.
Plausible: candidate believes that like the letdown regen heat exchanger, flashing on loss of cooling could occur

C. **Incorrect**: *B train pumps cooled by separate supply.*
Plausible: candidate believes heat exchanger and pump cooling come from the same place.

D. **Incorrect**: no auto open signal for RN41B.
Plausible: candidate feels there is some reason for the stated position of 41B in the setup and guesses it can auto open. Valve closes on blackout signal.

Level: SRO

KA: APE 062 AA2.0(2.9/3.6)

Lesson Plan Objective: PSS-RN Obj 7

Source: Bank

Level of knowledge: comprehension

References:
1. OP-MC-PSS-RN pages 23, 41, 73, 85
2. OP-MC-PSS-KC page 39

APE: 062 Loss of Nuclear Service Water

AK1. **Knowledge of the operational implications of the following concepts as they apply to Loss of Nuclear Service Water:**
(CFR 41.8 / 41.10 / 45.3)

None

AK2. **Knowledge of the interrelations between the Loss of Nuclear Service Water and the following:**
(CFR 41.7 / 45.7)

None

AK3. **Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water:**
(CFR 41.4, 41.8 / 45.7)

AK3.01	The conditions that will initiate the automatic opening and closing of the SWS isolation valves to the nuclear service water coolers	3.2*	3.5*
AK3.02	The automatic actions (alignments) within the nuclear service water resulting from the actuation of the ESFAS	3.6	3.9
AK3.03	Guidance actions contained in EOP for Loss of nuclear service water	4.0	4.2
AK3.04	Effect on the nuclear service water discharge flow header of a loss of CCW	3.5	3.7

AA1. **Ability to operate and / or monitor the following as they apply to the Loss of Nuclear Service Water (SWS):**
(CFR 41.7 / 45.5 / 45.6)

AA1.01	Nuclear service water temperature indications	3.1	3.1
AA1.02	Loads on the SWS in the control room	3.2	3.3
AA1.03	SWS as a backup to the CCWS	3.6*	3.6
AA1.04	CRDM high-temperature alarm system	2.7*	2.8
AA1.05	The CCWS surge tank, including level control and level alarms, and radiation alarm	3.1	3.1
AA1.06	Control of flow rates to components cooled by the SWS	2.9	2.9
AA1.07	Flow rates to the components and systems that are serviced by the SWS; interactions among the components	2.9	3.0

AA2. **Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water:**
(CFR: 43.5 / 45.13)

AA2.01	Location of a leak in the SWS	2.9	3.5
AA2.02	The cause of possible SWS loss	2.9	3.6
AA2.03	The valve lineups necessary to restart the SWS while bypassing the portion of the system causing the abnormal condition	2.6	2.9
AA2.04	The normal values and upper limits for the temperatures of the components cooled by SWS	2.5	2.9*
AA2.05	The normal values for SWS-header flow rate and the flow rates to the components cooled by the SWS	2.4*	2.5*
AA2.06	The length of time after the loss of SWS flow to a component before that component may be damaged	2.8*	3.1*

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
2	2	2	2	1

OBJECTIVES

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	State the purpose of the Nuclear Service Water System.	X	X	X	X	
2	State the Ultimate Heat Sink requirements and how these requirements are satisfied at McGuire Nuclear Station.	X	X	X	X	
3	List the power supplies for the RN pumps	X	X	X	X	
4	Relating to the RN pumps:					
	• List the automatic start signals	X	X	X	X	X
	• State the actions required to manually stop the pump(s) following an automatic start.		X	X	X	X
5	Describe the operation of the RN Pump Base Drain, including the sump name to which it is pumped.	X	X	X	X	
6	Concerning the RN Pump Strainers: • State the conditions which will initiate an automatic strainer backwash. • Describe the actions required if a strainer high delta P signal were to occur while a S_S signal was present including the reason for this design feature.	X	X	X	X	X
7	Concerning the RN pump mini-flow protection: • Describe how mini-flow protection is accomplished. • Discuss the operation of the RN mini-flow manual loader and how flow is controlled.	X	X	X	X	

Objective #6

Each RN pump is provided with a **strainer** which will remove any trash and/or debris greater than or equal to 3/16 inch in diameter from its suction line. **(Refer to Drawing 7.2 for strainer configuration in the system).** Each strainer has a strainer motor which is safety related. The RN strainer motors have no manual separate or individual controls. Each strainer motor is interlocked to run when its corresponding RN pump is running. The strainer is also provided with an automatic timer which will backwash the strainer once every 120 hours. If the strainer delta P reaches 1.86 psi (51.47 in.wg), an automatic backwash on Hi Strainer Delta P will be initiated. If the auto backwash actuates on the 1.86 psi signal, the 120 hour timer will be reset. The strainer automatic backwash valves receive a close signal on a S_S signal. If a high Delta P signal were to occur while the S_S signal is present, backwash would have to be performed manually (manual valve operation). This feature is designed to prevent unnecessary loss of water from the system. Water for the backwash is provided by the respective RN pump and discharges to the opposite train RC crossover discharge. Manual backwash can also be performed by an operator by using a local pushbutton. When performed manually, the strainers should be backwashed for 3 to 5 minutes. Manual backwash (electric push button operation) is performed when the automatic backwash is not available and periodically on "some" frequency. The rounds sheet value calls for checking Delta P below 50 in wg.

Objective # 7

Mini-flow protection for the RN pumps is provided by flow through the KC heat Exchanger **(Refer to Drawing 7.4)**. When an RN pump starts, the train related RN to KC inlet isolation valve (RN86A, RN187B) will open (provided their auto/manual selector switch is in auto). These valves also open on a train related S_S or Blackout signal and can be operated by open/close pushbutton on the RN section of MC11. The train related RN to KC heat exchanger flow is controlled by outlet control valve (RN89A, RN190B) manual loaders located on the RN section of MC11. If RN flow falls below 2700 gpm, the auto control feature will override the manual loader and open the valve proportional to flow between 2700 gpm and 0 gpm. Valves RN89A and RN190B will fail open with the aid of springs to open the actuator on loss of air or S_S . Meter indication for the RN to KC A(B)HX flow (0 to 10,000 gpm) is provided on MC9.

The following **alarms on AD12** are provided for the RN pump and strainer

- <u>"A(B) RN PUMP LO SUCTION PRESS"</u>

Setpoint : 2 psig

Origin: Comes off LP side of strainer D/P instrumentation

Probable cause:

1. fouled strainer
2. valve misalignment
3. low level from suction
4. leak
5. excessive flow on A or B RN headers.

Automatic actions: - None

The following are the Train A modulating valves:		Safe Position
• RN-89A	(RN to A KC HX Control)	Open*
• RN-21A	(RN Strainer A Backflush Automatic Supply Isolation)	Close
• RN-22A	(RN Strainer A Backflush Automatic Drain Isolation)	Close
• ND-29	(A ND HX Outlet)	Open
• KC-57A	(A ND HX Return)	Open

The following are the Train B modulating valves:		Safe Position
• RN-190B	(RN to B KC HX Control)	Open*
• RN-25B	(RN Strainer B Backflush Automatic Supply Isolation)	Close
• RN-26B	(RN Strainer B Backflush Automatic Drain Isolation)	Close
• ND-14	(B ND HX Outlet)	Open
• KC-82B	(B ND HX Return)	Open

* Theses valves open to their travel stop position (intermediate)

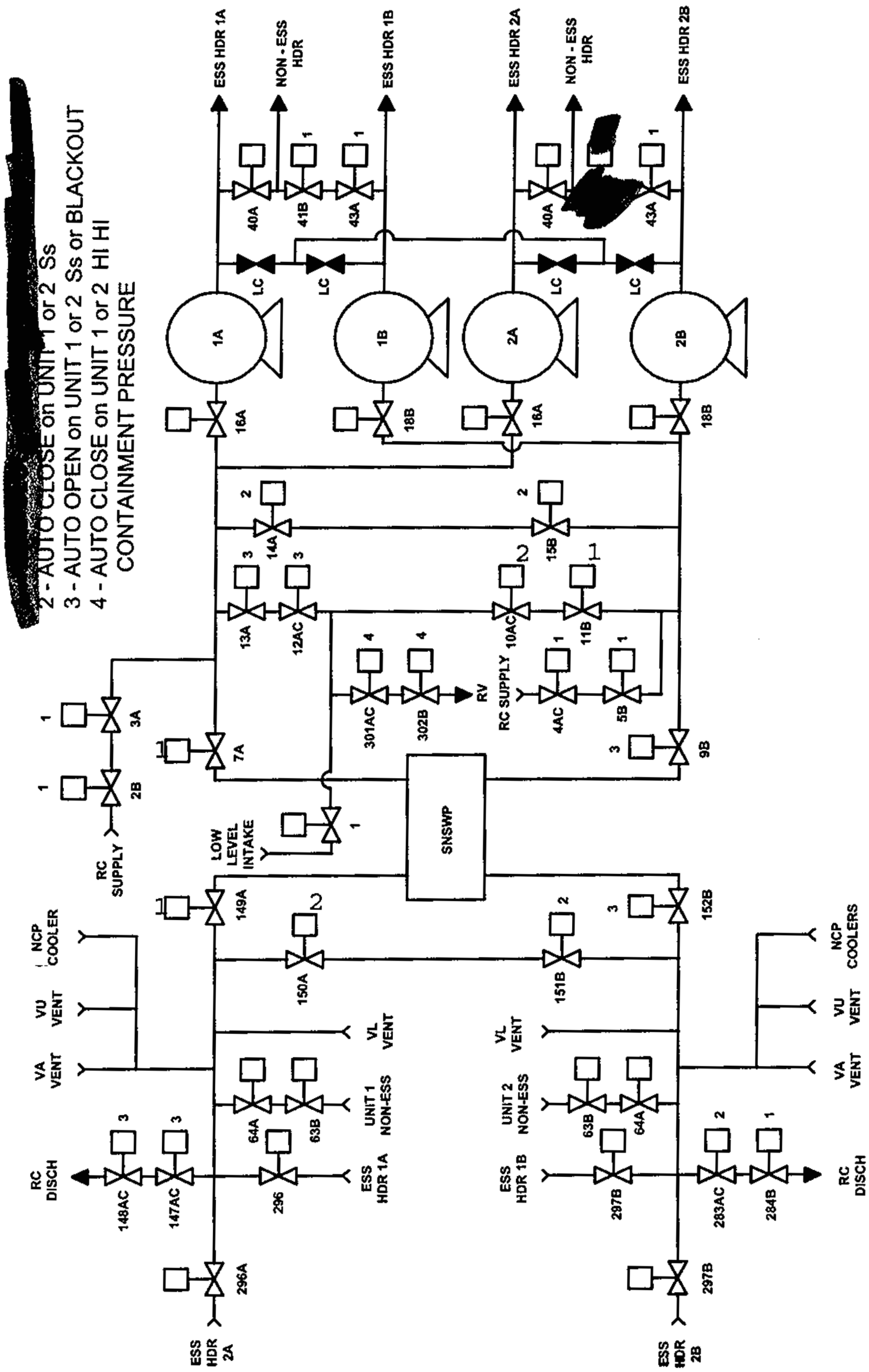
2.7 Instrumentation

Objective # 13

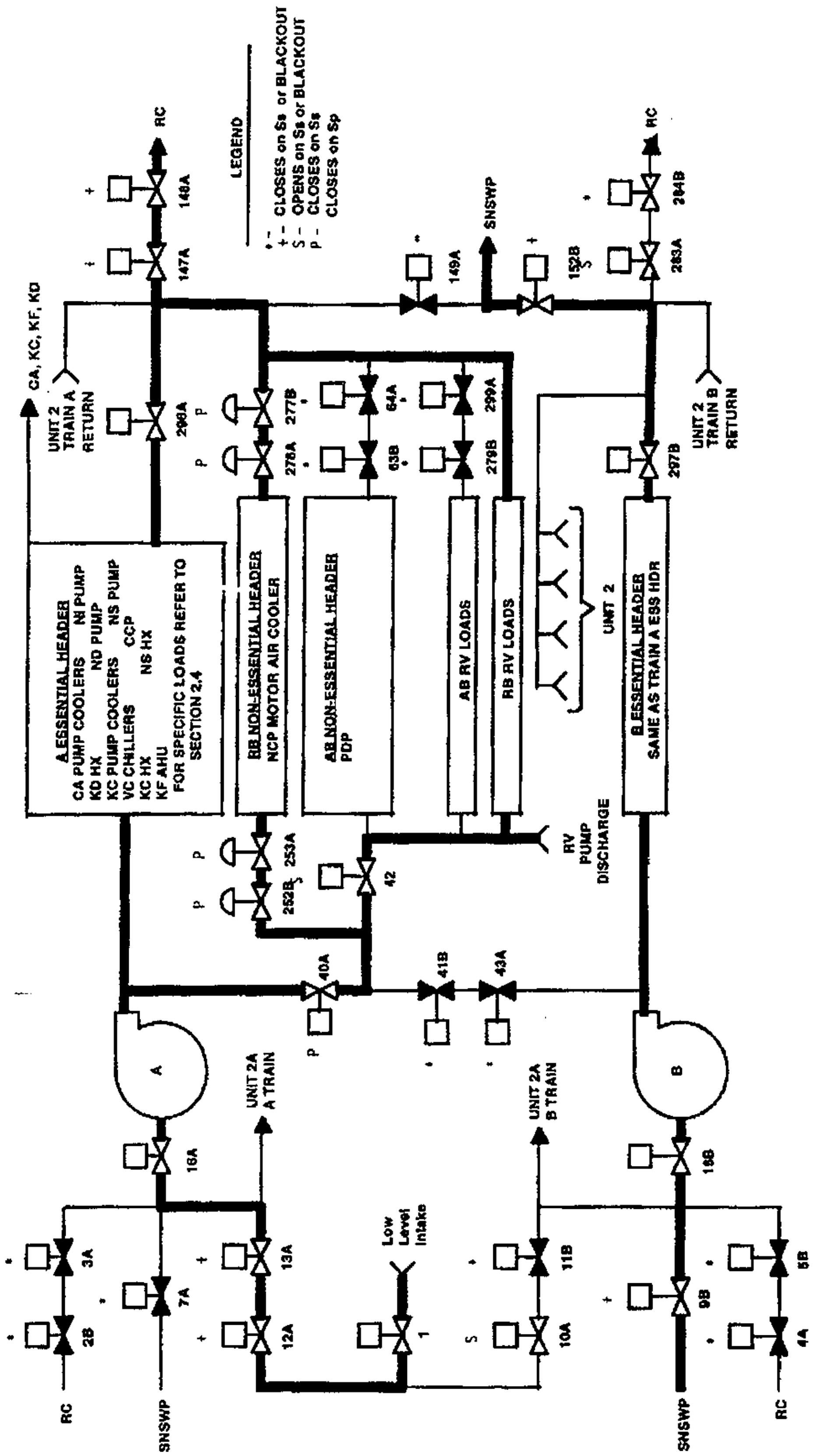
The following parameter indications associated with the RN System are located on MC09:

• A RN Pump Discharge pressure	(0 - 150 psig)
• B RN Pump Discharge pressure	(0 - 150 psig)
• A RN Pump Flow	(0 - 20,000 gpm)
• B RN Pump Flow	(0 - 20,000 gpm)
• RN Non-essential Header Pressure	(0 - 135 psig)
• RN to A KC HX Flow	(0 - 10,000 gpm)
• RN to B KC HX Flow	(0 - 10,000 gpm)
• RN to A NS HX Flow	(0 - 5,000 gpm)
• RN to B NS HX Flow	(0 - 5,000 gpm)
• RN to A D/G HX Flow	(0 - 15,000 gpm)
• RN to B D/G HX Flow	(0 - 15,000 gpm)
• Standby NSW Pond Temp	(30 - 100 ^{0}F)
• Standby NSW Pond Level	(738'11" - 741' elev.)

7.5, RN System Both Units S_S, BO and S_P Valve Logic (7/15/97)

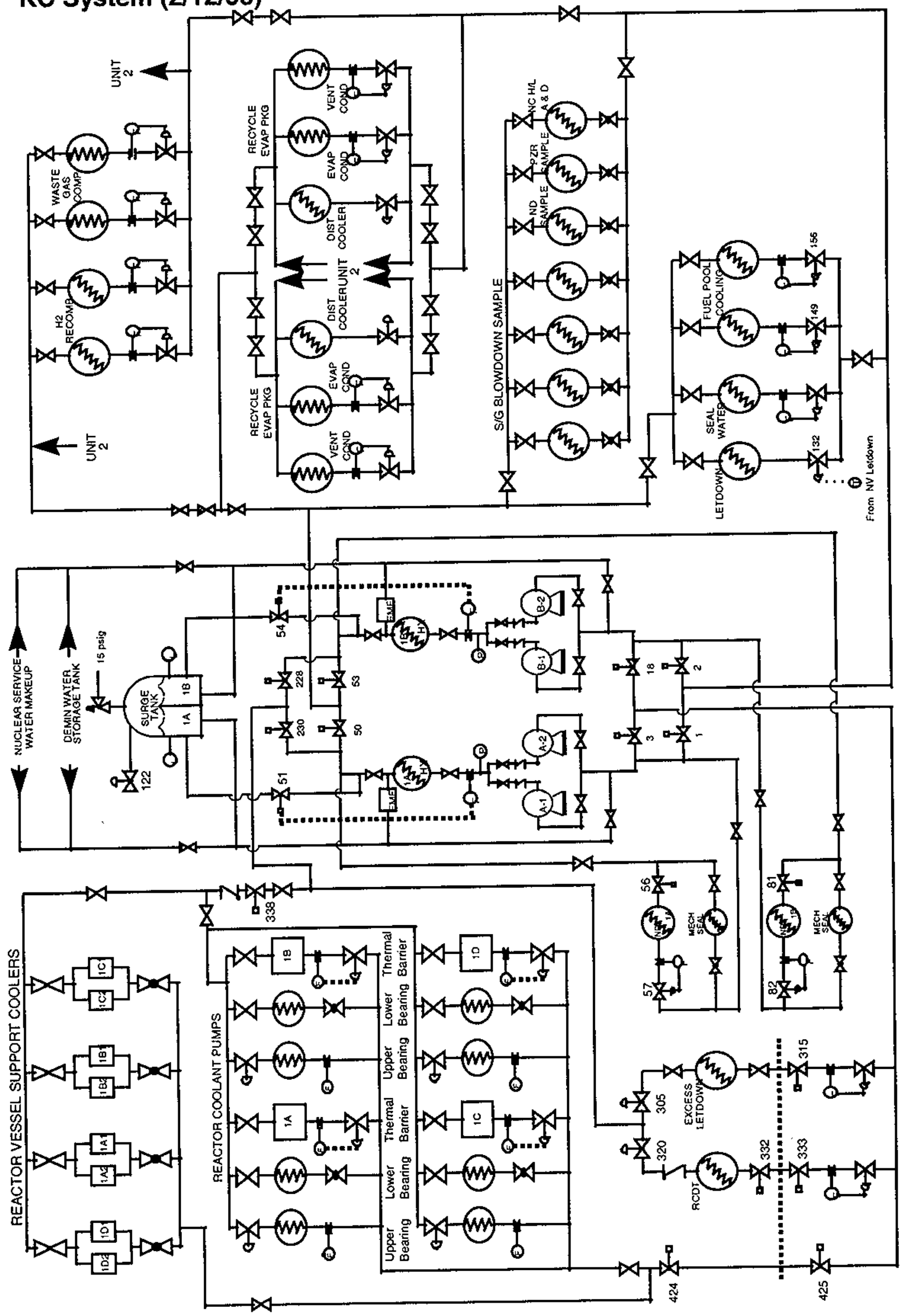


7.11, RN System Unit Blackout Loads and Valve Logic (7/2/98)

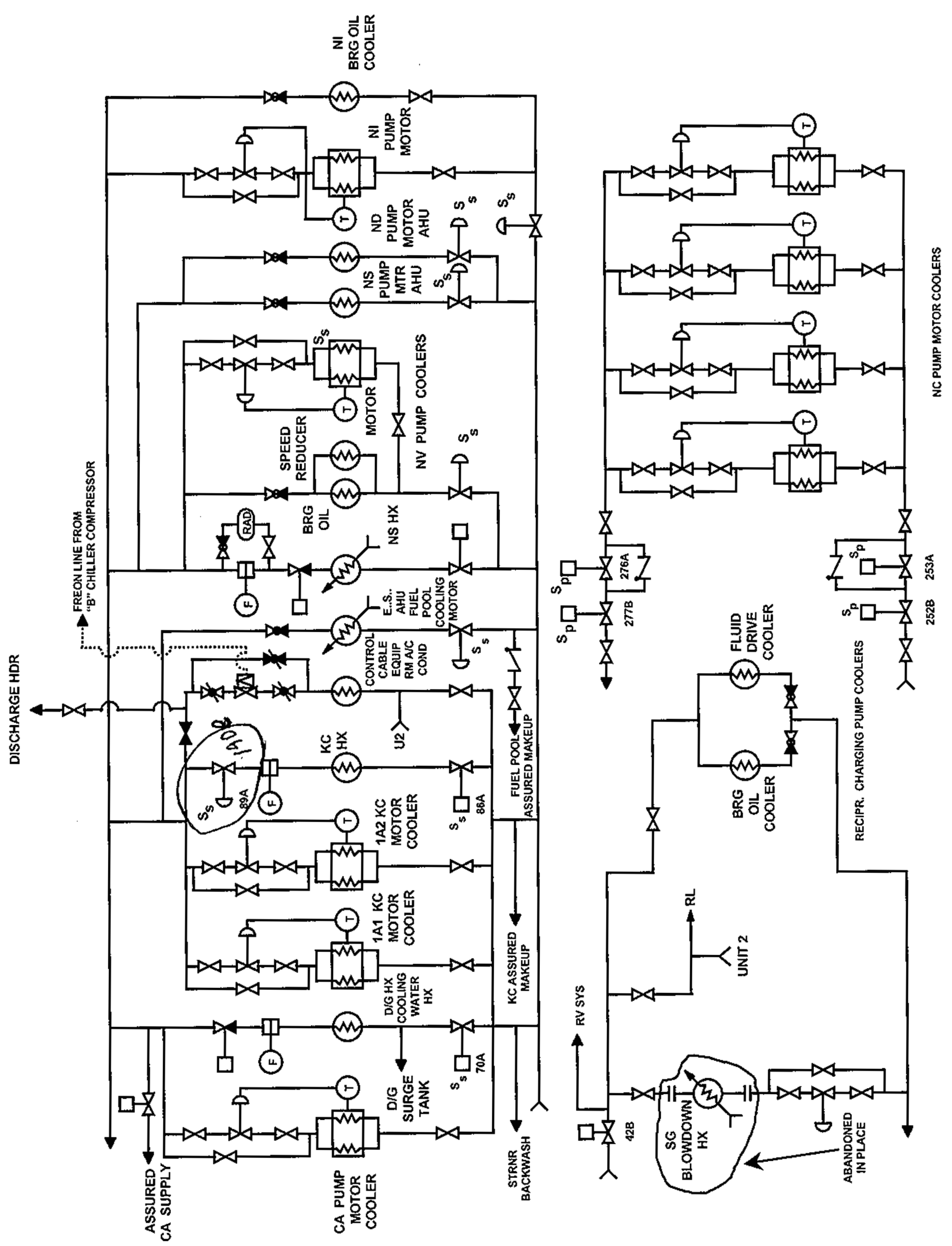


7.0 DRAWINGS

7.1 KC System (2/12/98)



7.4, RN System Component Loads (1/14/00)



Bank Question: 1071 **Answer: C**

1 Pt(s) After Channel 1 7300 Process Control Cabinet Channel Operability Test was completed the Unit 1 Pressurizer level master malfunctions causing it to demand full output while in automatic.

Which one of the following statements correctly describes the basis for the McGuire limit on flow?

A. **Letdown flow rates in excess of 135 gpm are limited to ensure proper demineralizer operation and adhere to the design limits of the letdown piping.**

B. **Letdown flow rates in excess of 120 gpm exceed the design limits of the letdown orifice valves and induce resonance vibration.**

C. **Charging flow rates in excess of 100 gpm during normal operation can induce vibration in the regenerative heat exchanger tubes.**

D. **Charging flow rates between 65 gpm and 100 gpm total charging flow will cause flashing in the regenerative heat exchanger.**

Distracter Analysis:

A. **Inorrect:**
B. **Incorrect: Plausible**
C. **Correct: Plausible**
D. **Incorrect: Plausible:**

Post-it® Fax Note 7671	Date	# of pages ▶
To Rick Baldwin	From C. Sawyer	
Co./Dept.	Co.	
Phone #	Phone #	
Fax #	Fax #	

Level: SRO

KA: APE 028 AA2.09 (2.9/3.2)

Lesson Plan Objective: OP-MC-PS-NV Obj. 13

Source: New

Level of knowledge: comprehension

References:
1. OP-MC-PS-NV pages 81 & 83

- Letdown Orifice Isolation Valves must have a full open indication prior to releasing the valve "Open" switch to prevent closure of the associated valve. (PIP M95-1541)

  Basis: Self-explanatory.

- WHEN the NV Pump shaft driven oil pump is supplying oil pressure, maintain associated NV Lube Oil Pump in the "AUTO" position. (PIP M96-1270)

  Basis: The shaft driven pump provides adequate operating oil pressure. When in ATUO, the auxiliary electric lube oil pump will start if the shaft driven pump pressure drops to 8# to protect the NV Pump bearings.

- When unit shutdown and no NV Pump operating, the NV Lube Oil Pump should be operated in manual for any NV Pump aligned such that a flow path exists through the NV Pump(s)..(PIP M96-1270)

  Basis: Provides lubrication when potential for windmilling exists.

- IF all letdown paths are to be isolated, the gravity drainage paths from the FWST and VCT should be isolated also.

  Basis: This prevents an inadvertent transfer of water to an undesired location.

- The actions of Enclosure 4.7 (Operator Action With NV Aux Spray In Service) (of the NV Letdown procedure) should be performed immediately if a Safety Injection occurs and shall not interfere with the actions of EP/1/A/5000/E-0 (Reactor Trip or Safety Injection). (PIP-0-M96-2396)

  Basis: Satisfies FSAR requirements for Aux Spray isolation.

- IF letdown diverted to the RHT AND unable to maintain VCT level, letdown flow should be reduced.

  Basis: Minimize charging demand and VCT level loss.

- Maximum letdown flows are as follows:

  120 gpm through Normal Letdown
  150 gpm through ND Aux Letdown with single Mixed Bed Demin in service
  185 gpm through ND Aux Letdown with parallel Mixed Bed Demin in service.

  Basis: Flow is limited to ensure effective demineralizer operation and to adhere to design limits of letdown piping.

- Maximum letdown header pressure is 255 psig to avoid lifting 1NV-156 (255 psig setpoint)

  Basis: Self-explanatory.

- If NC pressure is less than 100 psig OR seal injection water is NOT supplied, Seal Bypass Return Isolation Valve and Seal Return Isolation Valves shall be closed. This prevents unfiltered water being supplied to the seals.

  Basis: Self-explanatory.

- WHEN NC System pressure greater than 100 psig OR temperature greater than 150°F, Seal Injection should be maintained to NC Pumps.

  Basis: Ensures pure seal injection water through seals when pressure is high enough to allow seal leakoff (100 psig) or provide cooling to seals and bearing when required (150°F).

- If normal letdown is lost with PZR water temperature > 250 °F, NV Aux Spray is prohibited (prevents thermal shocking spray nozzle).

  Basis: Self-explanatory.

- When NV Aux Spray is used, maximum spray D/T is 320 °F.

  Basis: Avoids thermal shocking spray nozzle.

- Maximum charging flows are as follows: (PIP M96-2513)
  144 gpm through Regen HX during transient / shutdown operation
  100 gpm through Regen HX during normal operation

  Basis: Excessive flow can induce vibrations in the regenerative heat exchanger tubes.

- WHEN idle NV System components placed in service, boron concentration differences can affect reactivity.

  Basis: Reminder to the operator of the potential impact on core reactivity.

- Design flow rate of # 1 PD Pump is 98 gpm.

  Basis: Operator information.

- Placement of PD Pump Speed Control in "AUTO" is prohibited.

  Basis: Auto operation is not maintained by IAE.

- Maintain 1NV-159 locked closed and 1NV-157 locked open due to the potential "gas stripping" of the mini-flow orifice. This alignment routes gases to the VCT instead of the charging pump suction. (PIP 98-0036)

  Basis: This prevents potential gas binding of the ECCS high head pumps.

- Maximum NCP seal injection flow is 12 gpm per pump.

  Basis: Minimize concern with overpressurizing the seal area.

To: Rick Baldwin
From C. Sawyer
40 pages

Bank Question: 1071 **Answer: C**

1 Pt(s)

After Channel 1 7300 Process Control Cabinet ACOTs. The Unit 1 Pressurizer level master malfunctions ~~causing 1NV-238 () to fully open~~ to full output while in automatic. ~~Indicated charging flow has increased to 187 gpm while letdown flow remains constant at 75 gpm.~~

Which one of the following statements correctly describes the basis for the McGuire limit on flow?

A. **Letdown flow rates in excess of 135 gpm are limited to ensure proper demineralizer operation and adhere to the design limits of the letdown piping.**

B. **Letdown flow rates in excess of 120 gpm exceed the design limits of the letdown orifice valves and induce resonance vibration.**

C. **Charging flow rates in excess of 100 gpm during normal operation can induce vibration in the regenerative heat exchanger tubes.**

D. **Charging flow rates ~~in excess of 175~~ < ? gpm gpm total charging flow ~~can exceed design flow limits of the NV pump causing pump runout.~~** causes flashing in the Regenerative Hx.

Distracter Analysis:

A. **Incorrect:**
B. **Incorrect:**
Plausible
C. **Correct:**
Plausible
D. **Incorrect:**
Plausible:

Level: SRO

KA: APE 028 AA2.09 (2.9/3.2)

Lesson Plan Objective:

Source: New

Level of knowledge: comprehension

References:

Replace

Replacement for 902.2

1. OP-MC-

1 Pt. Unit 2 has just begun to shutdown (decreasing 2MWe/min) for refueling. Given the following events and conditions:

- Pressurizer level is at program level and in 'automatic'.
- The controlling pressurizer level transmitter fails at its current output.
- No operator action is taken.

Which one of the following statements correctly describes the system response as plant load is reduced?

A. **Charging flow decreases**
Letdown isolates
Pressurizer heaters turn off

B. **Charging flow increases**
Pressurizer heaters energize
Pressurizer level increase to the trip setpoint

C. **Charging flow decreases**
Letdown will not isolate
Pressurizer level decreases until the pressurizer is empty

D. **Charging flow increases**
Pressurizer heaters will not energize
Pressurizer level increases to the trip setpoint.

Bank Question: 902.2 **_Answer: A_**

1 Pt. Unit 2 has just begun to shutdown (decreasing 2MWe/min) for refueling. Given the following events and conditions:

- Pressurizer level is at program level and in 'automatic'.
- The controlling pressurizer level transmitter fails at its current output.
- No operator action is taken.

Which one of the following statements correctly describes the system response as plant load is reduced?

A. **Charging flow decreases**
Letdown isolates
Pressurizer heaters turn off

B. **Charging flow increases**
Pressurizer heaters energize
Pressurizer level increase to the trip setpoint

C. **Charging flow decreases**
Letdown will not isolate
Pressurizer level decreases until the pressurizer is empty

D. **Charging flow increases**
Pressurizer heaters will not energize
Pressurizer level increases to the trip setpoint.

Distracter Analysis: As load is reduced, Tave will decrease, Program Pressurizer level will decrease. The system will see the controlling channel maintaining a high level and decrease charging in an effort to reduce level. Actual level will decrease. The backup channel will decrease and at 17%, letdown will isolate and heaters will de energize.

A. **Correct**:
B. **Incorrect**: charging flow will decrease.
Plausible: candidate believes charging flow will increase in an effort to maintain the higher level
C. **Incorrect**: letdown will isolate from the bakup channel.
Plausible: if the candidate believes the low level interlock will not be satisfied only from the controlling channel.
D. **Incorrect**: charging flow will decrease.
Plausible: candidate believes charging flow will increase to maintain the higher level

Level: SRO

KA: APE 028 AA2.09 (2.9/3.2)

Lesson Plan Objective: PS-ILE Obj. 12

Source: BANK

Level of knowledge: Comprehension

References:
1. OP-MC-PS-ILE page 33 (Figure 7.2)

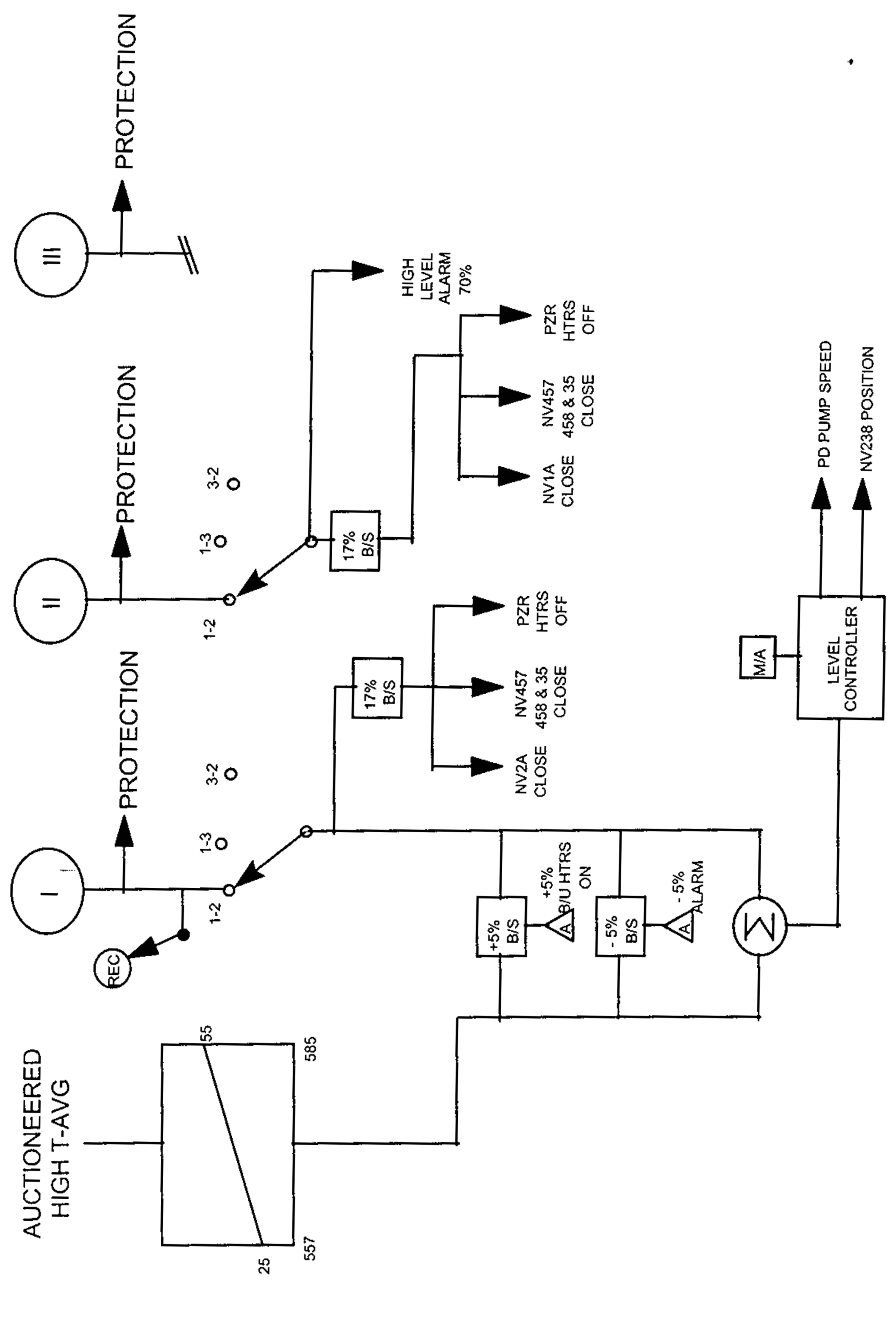
APE:	**028 Pressurizer (PZR) Level Control Malfunction**		
	<u>ABILITY</u>		
AA1.	**Ability to operate and / or monitor the following as they apply to the Pressurizer Level Control Malfunctions: (CFR 41.7 / 45.5 / 45.6)**		
AA1.01	PZR level reactor protection bistables	3.8*	3.9
AA1.02	CVCS	3.4	3.4
AA1.03	RCP and seal water system	2.9	2.9
AA1.04	Regenerative heat exchanger and temperature limits	2.7	2.8
AA1.05	Initiation of excess letdown per the CVCS	2.8	2.9
AA1.06	Checking of RCS leaks	3.3	3.6
AA1.07	Charging pumps maintenance of PZR level (including manual backup)	3.3	3.3
AA1.08	Selection of an alternate PZR level channel if one has failed	3.7	3.6
AA2.	**Ability to determine and interpret the following as they apply to the Pressurizer Level Control Malfunctions: (CFR: 43.5 / 45.13)**		
AA2.01	PZR level indicators and alarms	3.4	3.6
AA2.02	PZR level as a function of power level or T-ave. including interpretation of malfunction	3.4	3.8
AA2.03	Charging subsystem flow indicator and controller	2.8	3.3
AA2.04	Ammeters and running indicators for CVCS charging pumps	2.6	3.1
AA2.05	Flow control valve isolation valve indicator	2.6	2.7
AA2.06	Letdown flow indicator	2.7	2.8
AA2.07	Seal water flow indicator for RCP	2.6	2.9
AA2.08	PZR level as a function of power level	3.1	3.5
[redacted]			
AA2.10	Whether the automatic mode for PZR level control is functioning improperly, necessity of shift to manual modes	3.3	3.4
AA2.11	Leak in PZR	3.2	3.6
AA2.12	Cause for PZR level deviation alarm: controller malfunction or other instrumentation malfunction	3.1	3.5
AA2.13	The actual PZR level, given uncompensated level with an appropriate graph	2.9	3.2
AA2.14	The effect on indicated PZR levels, given a change in ambient pressure and temperature of reflux boiling	2.6	2.8

OBJECTIVES

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
10	Describe the protection (signals, setpoints, permissives) associated with Pressurizer level (logic not required).		X	X	X	X
11	Describe the actions the operator must take to restore Pressurizer heater operation following a low level heater cutoff.			X	X	X
12	For any Pressurizer Level Control System input signal failure, determine the effect and evaluate operator action to be taken.			X	X	X
13	Determine program Pressurizer level for interim power levels between 0% and 100%.		X	X	X	X
14	Concerning the Technical Specifications related to the Pressurizer Level Control System:					
	• Given the LCO title, state the LCO (including any COLR values) and applicability.			X	X	X
	• For any LCO's that have action required within one hour, state the action.			X	X	X
	• Given a set of parameter values or system conditions, determine if any Tech Spec LCO's is (are) not met and any action(s) required within one hour.			X	X	X
	• Given a set of parameters or system conditions and the appropriate Tech Specs, determine required action(s).			X	X	X
	• Discuss the bases for a given Tech Spec LCO or Safety Limit				X	
	*** SRO ONLY**					

7.2. Pressurizer Level Control Circuit (simplified) (8/14/97)

Objective #2



1 Pt. Which one of the following is a correct list of SAFETY LIMITS?

A. **Thermal Power, RCS Highest Loop Tave and Pressurizer Pressure.**

B. **Thermal Power, AFD, Pressurizer Pressure.**

C. **AFD, QPTR and Reactor Power.**

D. **Linear Heat Generation Rate, Thermal Power and QPTR.**

Bank Question: 991.1 ***Answer: A***

1 Pt. Which one of the following is a correct list of SAFETY LIMITS?

A. Thermal Power, RCS Highest Loop Tave and Pressurizer Pressure.

B. Thermal Power, AFD, Pressurizer Pressure.

C. AFD, QPTR and Reactor Power.

D. Linear Heat Generation Rate, Thermal Power and QPTR.

Distracter Analysis:

A. Correct:.
Plausible:
B. Incorrect:
Plausible:
C. Incorrect:
D. Incorrect:.
Plausible:

Level: SRO Only

KA: G 2.1.10 (2.7/3.9)

Lesson Plan Objective: (None)

Source: New

Level of knowledge: memory

References:
1. Tech Spec 2.1.1

2.1 *Conduct of Operations (continued)*

2.1.9 **Ability to direct personnel activities inside the control room.**
(CFR: 45.5 / 45.12 / 45.13)
IMPORTANCE RO 2.5 SRO 4.0

[redacted]

2.1.11 **Knowledge of less than one hour technical specification action statements for systems.**
(CFR: 43.2 / 45.13)
IMPORTANCE RO 3.0 SRO 3.8

2.1.12 **Ability to apply technical specifications for a system.**
(CFR: 43.2 / 43.5 / 45.3)
IMPORTANCE RO 2.9 SRO 4.0

2.1.13 **Knowledge of facility requirements for controlling vital / controlled access.**
(CFR: 41.10 / 43.5 / 45.9 / 45.10)
IMPORTANCE RO 2.0 SRO 2.9

2.1.14 **Knowledge of system status criteria which require the notification of plant personnel.**
(CFR: 43.5 / 45.12)
IMPORTANCE RO 2.5 SRO 3.3

2.1.15 **Ability to manage short-term information such as night and standing orders.**
(CFR: 45.12)
IMPORTANCE RO 2.3 SRO 3.0

2.1.16 **Ability to operate plant phone, paging system, and two-way radio.**
(CFR: 41.10 / 45.12)
IMPORTANCE RO 2.9 SRO 2.8

2.1.17 **Ability to make accurate, clear and concise verbal reports.**
(CFR: 45.12 / 45.13)
IMPORTANCE RO 3.5 SRO 3.6

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the SLs specified in Figure 2.1.1-1 for four loop operation.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained $\leq$ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

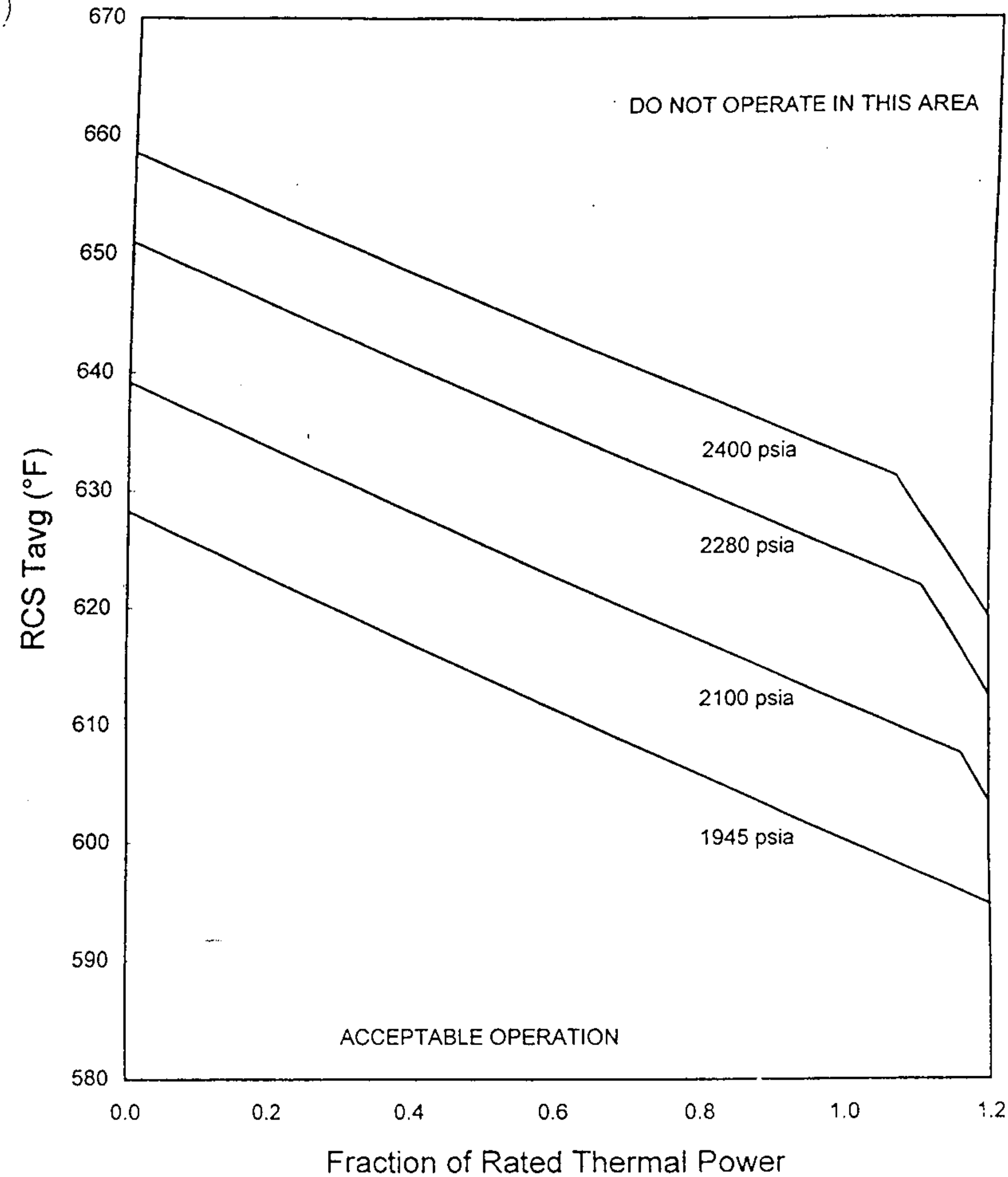


Figure 2.1.1-1

Reactor Core Safety Limits -
Four Loops in Operation

1 Pt. Unit 1 has experienced a 50% load rejection which resulted in Control Bank "D" Group 1 being greater than 12 steps misaligned from its associated step counter. Tech Spec 3.1.4 Rod Control Group Alignment Limits states:

"All shutdown and control rods shall be OPERABLE; with all individual indicated rod positions within12 steps of their group step counter demand position".

Which one of the following is the bases for this Tech Spec?

A. **Ensure SDM limits are maintained and QPTR is maintained within limits.**

B. **Ensure power distribution and SDM limits are preserved.**

C. **Ensure QPTR is maintained within limits and rod alignments are correct.**

D. **Ensure AFD is maintained and limit power distribution.**

Bank Question: 1004.1 ***Answer: B***

1 Pt. Unit 1 has experienced a 50% load rejection which resulted in Control Bank "D" Group 1 being greater than 12 steps misaligned from its associated step counter. Tech Spec 3.1.4 Rod Control Group Alignment Limits states:

"All shutdown and control rods shall be OPERABLE; with all individual indicated rod positions within12 steps of their group step counter demand position".

Which one of the following is the bases for this Tech Spec?

A. Ensure SDM limits are maintained and QPTR is maintained within limits.

B. Ensure power distribution and SDM limits are preserved.

C. Ensure QPTR is maintained within limits and rod alignments are correct.

D. Ensure AFD is maintained and limit power distribution.

Distracter Analysis:

A. Incorrect:.
Plausible:
B. Correct:
C. Incorrect:
Plausible:
D. Incorrect:.
Plausible:

Level: SRO

KA: G2.2.25 (2.5/3.7)

Lesson Plan Objective:OP-MC-IC-IRX Obj. 14

Source: BANK

Level of knowledge: Memory

References:
1.T.S. 3.1.4 Bases

2.2 Equipment Control (Continued)

2.2.18 Knowledge of the process for managing maintenance activities during shutdown operations.
(CFR: 43.5 / 45.13)
IMPORTANCE RO 2.3 SRO 3.6

2.2.19 Knowledge of maintenance work order requirements.
(CFR: 43.5 / 45.13)
IMPORTANCE RO 2.1 SRO 3.1

2.2.20 Knowledge of the process for managing troubleshooting activities.
(CFR: 43.5 / 45.13)
IMPORTANCE RO 2.2 SRO 3.3

2.2.21 Knowledge of pre- and post-maintenance operability requirements.
(CFR: 43.2)
IMPORTANCE RO 2.3 SRO 3.5

2.2.22 Knowledge of limiting conditions for operations and safety limits.
(CFR: 43.2 / 45.2)
IMPORTANCE RO 3.4 SRO 4.1

2.2.23 Ability to track limiting conditions for operations.
(CFR: 43.2 / 45.13)
IMPORTANCE RO 2.6 SRO 3.8

2.2.24 Ability to analyze the affect of maintenance activities on LCO status.
(CFR: 43.2 / 45.13)
IMPORTANCE RO 2.6 SRO 3.8

2.2.25 Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.
(CFR: 43.2)
IMPORTANCE RO 2.5 SRO 3.7

2.2.26 Knowledge of refueling administrative requirements.
(CFR: 43.5 / 45.13)
IMPORTANCE RO 2.5 SRO 3.7

2.2.27 Knowledge of the refueling process.
(CFR: 43.6 / 45.13)
IMPORTANCE RO 2.6 SRO 3.5

OBJECTIVES

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
12	Describe the Control Room controls, indications and alarms, including alarm setpoints.		X	X	X	X
13	Given a limit or precaution associated with the Reactor Control System, discuss its basis and applicability.		X	X	X	X
14	Concerning the Technical Specifications related to the Reactor Control System:					
	• Given the LCO title, state the LCO (including any COLR values) and applicability.			X	X	X
	• For any LCO's that have action required within one hour, state the action.			X	X	X
	• Given a set of parameter values or system conditions, determine if any Tech Spec LCO's is (are) not met and any action(s) required within one hour.			X	X	X
	• Given a set of plant parameters or system conditions and the appropriate Tech Specs, determine required action(s).			X	X	X
	• Discuss the bases for a given Tech Spec LCO or Safety Limit				X	*
	* SRO Only					

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Rod Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (e.g., trippability) of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Protection" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately 5/8 inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. The unit has four control banks and five shutdown banks.

BACKGROUND (continued)

The shutdown banks are maintained either in the fully inserted or fully withdrawn position. The control banks are moved in an overlap pattern as described in the Bases for LCO 3.1.6, "Control Bank Insertion Limits." The control rods are arranged in a radially symmetric pattern, so that control bank motion does not introduce radial asymmetries in the core power distributions.

The axial position of shutdown rods and control rods is indicated by two separate and independent systems, which are the Bank Demand Position Indication System (commonly called group step counters) and the Digital Rod Position Indication (DRPI) System.

The Bank Demand Position Indication System counts the pulses from the rod control system that moves the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± 5/8 inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The DRPI System provides a highly accurate indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube with a center to center distance of 3.75 inches, which is six steps. To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one system fails, the DRPI will go on half-accuracy with an effective coil spacing of 7.5 inches, which is 12 steps. Therefore, the normal indication accuracy of the DRPI System is ± 6 steps (± 3.75 inches), and the maximum uncertainty is ± 12 steps (± 7.5 inches). With an indicated deviation of 12 steps between the group step counter and DRPI, the maximum deviation between actual rod position and the demand position could be 24 steps, or 15 inches.

BASES

APPLICABLE SAFETY ANALYSES

Control rod misalignment accidents are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing control rod inoperability or misalignment are that:

a. There be no violations of:

   1. specified acceptable fuel design limits, or
   2. Reactor Coolant System (RCS) pressure boundary integrity; and

b. The core remains subcritical after accident transients.

Two types of misalignment are distinguished. During movement of a control rod group, one rod may stop moving, while the other rods in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the control rods to meet the SDM requirement, with the maximum worth rod stuck fully withdrawn.

Analyses are performed in regard to static rod misalignment, single rod withdrawal, dropped rod, and dropped group of rods (Ref. 4). With control banks at their insertion limits, one type of analysis considers the case when any one rod is completely inserted into the core. The second type of analysis considers the case of a completely withdrawn single rod from a bank inserted to its insertion limit. Satisfying limits on departure from nucleate boiling ratio in both of these cases bounds the situation when a rod is misaligned from its group by 12 steps. Another type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA also fully withdrawn (Ref. 5).

The Required Actions in this LCO ensure that either deviations from the alignment limits will be corrected or that THERMAL POWER will be adjusted so that excessive local linear heat rates (LHRs) will not occur, and that the requirements on SDM and ejected rod worth are preserved.

Continued operation of the reactor with a misaligned control rod is allowed if the heat flux hot channel factor ($F_Q(X,Y,Z)$) and the nuclear enthalpy hot channel factor ($F^N_{\Delta H}(X,Y)$) are verified to be within their limits in the COLR and the safety analysis is verified to remain valid. When a control rod is misaligned, the assumptions that are used to determine the rod insertion limits, AFD limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the

BASES

APPLICABLE SAFETY ANALYSES (continued)

design peaking factors, and $F_Q(X,Y,Z)$ and $F^N_{\Delta H}(X,Y)$ must be verified directly by incore mapping. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q(X,Y,Z)$ and $F^N_{\Delta H}(X,Y)$ to the operating limits.

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in the safety analyses. Therefore they satisfy Criterion 2 of 10 CFR 50.36 (Ref. 6).

LCO

The requirements on rod OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The limits on shutdown and control rod alignments ensure that the assumptions in the safety analysis will remain valid, and that the RCCAs and banks maintain the correct power distribution and rod alignments.

The requirement to maintain the alignment of any one rod to within plus or minus 12 steps is conservative. The minimum misalignment assumed in safety analysis is 24 steps (15 inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed. Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

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

ACTIONS

A.1.1 and A.1.2

When one or more rods are untrippable, there is a possibility that the required SDM may be adversely affected. Under these conditions, it is

1 Pt. Unit 1 is operating at 100% power when the following occurs:

- LOCA outside containment
- FWST level is 180"
- Containment Sump level is 2.75 feet
- FWST has not ruptured
- ES 1.3 *Transfer to Cold Leg Recirc* being implemented

REFERENCE PROVIDED

Which of the following describes the correct procedure flowpath?

A. Go to ECA 1.1 (*Loss of Emergency Coolant Recirc*)

B Go to ECA 1.2 (*LOCA Outside Containment*)

C Go to ES 1.2 (*Post LOCA Cooldown and Depressurization*)

D. Continue in body of ES 1.3 (*Transfer to Cold Leg Recirc*)

Bank Question: 1033 **_Answer: A_**

1 Pt. Unit 1 is operating at 100% power when the following occurs:

- LOCA outside containment
- FWST level is 180"
- Containment Sump level is 2.75 feet
- FWST has not ruptured
- ES 1.3 *Transfer to Cold Leg Recirc* being implemented

REFERENCE PROVIDED
ES-1.3 AND Enclosure 2

Which of the following describes the correct procedure flowpath?

A. Go to ECA 1.1 (*Loss of Emergency Coolant Recirc*)

B Go to ECA 1.2 (*LOCA Outside Containment*)

C Go to ES 1.2 (*Post LOCA Cooldown and Depressurization*)

D. Continue in body of ES 1.3 (*Transfer to Cold Leg Recirc*)

Distracter Analysis:

A. Correct –

B. Incorrect.

C. Incorrect

D. Incorrect

Level: SRO

KA: EPE W/E11 EA2.1 (3.4/4.2)

Lesson Plan Objective: OP-MC-EP-E1 Obj. 6

Source: New

Level of knowledge: Analysis

Author: CWS

References:
1. OP-MC-EP-E1 page 157

2. ES 1.3 Transfer to Cold Leg Recirc. Provided

EPE: 11 Loss of Emergency Coolant Recirculation (Continued)

K/A NO. KNOWLEDGE

EK3.2 Normal, abnormal and emergency operating procedures associated with (Loss of Emergency Coolant Recirculation).
IMPORTANCE RO 3.5 SRO 4.0

EK3.3 Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations.
IMPORTANCE RO 3.8 SRO 3.8

EK3.4 RO or SRO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated.
IMPORTANCE RO 3.6 SRO 3.8

ABILITY

EA1. Ability to operate and / or monitor the following as they apply to the (Loss of Emergency Coolant Recirculation)
(CFR: 41.7 / 45.5 / 45.6)

EA1.1 Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
IMPORTANCE RO 3.9 SRO 4.0

EA1.2 Operating behavior characteristics of the facility.
IMPORTANCE RO 3.5 SRO 3.8

EA1.3 Desired operating results during abnormal and emergency situations.
IMPORTANCE RO 3.7 SRO 4.2

EA2. Ability to determine and interpret the following as they apply to the (Loss of Emergency Coolant Recirculation)
(CFR: 43.5 / 45.13)

EA2.1 Facility conditions and selection of appropriate procedures during abnormal and emergency operations.
IMPORTANCE RO 3.4 SRO 4.2

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
N/A	N/A	5.0	5.0	4.0

OBJECTIVES

SEQ	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	Explain the purpose for each procedure in the E-1 series. EPE1001			X	X	
2	Discuss the entry and exit guidance for each procedure in the E-1 series. EPE1002			X	X	
3	Discuss the mitigating strategy (major actions) of each procedure in the E-1 series. EPE1003			X	X	X
4	Discuss the basis for any note, caution or step for each procedure in the E-1 series. EPE1004			X	X	X
5	Given the Foldout page discuss the actions included and the basis for these actions. EPE1005			X	X	X
6	Given the appropriate procedure, evaluate a given scenario describing accident events and plant conditions to determine any required action and its basis. EPE1006			X	X	X
7	Discuss the time critical task(s) associated with the E-1 series procedures including the time requirements and the basis for these requirements. EPE1007			X	X	X

STEP 3 Check containment sump level - GREATER THAN 3 FT.

<u>PURPOSE:</u> To ensure there is sufficient level in the sump to support the transfer to Cold Leg Recirc.

<u>BASIS:</u> This step sends the operator to the proper procedure for dealing with a loss of coolant recirculation before the FWST becomes empty if the transfer cannot be performed in order to prevent loss of suction flow to the pumps and potential pump damage. There are three possible causes of inadequate sump level addressed by the RNO to this step.

First, Small LOCAs of certain size and location have been evaluated to cause sump level indication to be less than the required value by the time the FWST reaches the Lo level setpoint. Evaluation of these conditions shows that the proper action would be to transfer to CLR. The evaluation indicated that sufficient sump level would be achieved by the time the FWST reached the Lo Lo level and that vortexing would not be a concern during the period of the realignment prior to reaching Lo Lo level in the tank. Sump level buildup is assured since safety injection has occurred, FWST Lo Level setpoint has been reached and containment Spray has actuated due to high pressure in containment. These conditions indicate that reactor coolant and FWST inventory is being released into containment and will eventually find its way to the sump.

LOCA Outside Containment or FWST Rupture (Due to a Tornado) will result in FWST depletion without the corresponding Sump Buildup. In this case actions are specified in Enclosure 2 (Loss of FWST Inventory outside Containment) to secure injection and implement ECA 1.1 (Loss of Emergency Coolant Recirculation) in the case of LOCA Outside Containment. In the case of an FWST rupture the operator is directed to return to the procedure in effect. A Tornado is not postulated to occur concurrently with a high-energy line break inside containment.

Note that if an Orange or Red path Procedure is in effect upon transition out of ES-1.3, it takes priority over any other procedure including ECA-1.1.

STEP 4 Check KC flow to ND heat exchangers – GREATER THAN 5000 GPM.

<u>PURPOSE:</u> To ensure KC flow to the ND heat exchangers.

<u>BASIS:</u> This step assumes that the ND heat exchangers are used for heat removal during the post accident recirculation phase and that either KC flow has been automatically provided to the heat exchangers or the operator has manually established KC flow prior to the switchover. If KC flow had not previously been established, then it should be established at this time.

1 Pt. Given the following conditions:

- Pressurizer Level Channel 1 is at 28% level
- Pressurizer Level Channel 2 associated bistables are in the tripped condition due to surveillance testing
- Pressurizer Level Channel 3 fails high.
- N-41 is 8%
- N-42 is 10%
- N-43 is 9%
- N-44 is 9%
- Impulse pressure channel 1 is 11%
- Impulse pressure channel 2 is 9%
- No reactor trip has occurred

Which of the following describes the proper operator response?

A. Trip the reactor and enter E-0 (*Reactor Trip or Safety Injection*)

B. Trip the reactor and enter FR-S.1 (*Response Nuclear Power Generation/ATWS*)

C. Do not trip the reactor because thermal power is less than P-7

D. Do not trip the reactor. Initiate unit shutdown.

Bank Question: 1035 **_Answer: D_**

1 Pt. Given the following conditions:

- Pressurizer Level Channel 1 is at 28% level
- Pressurizer Level Channel 2 associated bistables are in the tripped condition due to surveillance testing
- Pressurizer Level Channel 3 fails high.
- N-41 is 8%
- N-42 is 10%
- N-43 is 9%
- N-44 is 9%
- Impulse pressure channel 1 is 11%
- Impulse pressure channel 2 is 9%
- No reactor trip has occurred

Which of the following describes the proper operator response?

A. Trip the reactor and enter E-0 (*Reactor Trip or Safety Injection*)

B. Trip the reactor and enter FR-S.1 (*Response Nuclear Power Generation/ATWS*)

C. Do not trip the reactor because thermal power is less than P-7

D. Do not trip the reactor. Initiate unit shutdown.

Distracter Analysis:

A. Incorrect: – This example is not at valid ATWS event. This is a loss of reactor protection. Therefore you do not trip the reactor.

B. Incorrect: This example is not at valid ATWS event. This is a loss of reactor protection. Therefore you do not trip the reactor.

C. Incorrect: Greater than P-7. Pzr Hi level trip is in effect

D. Correct

Level: SRO

KA: APE 029 EA2.02 (4.2/4.4)

Lesson Plan Objective: OP-MC-IC-IPE Obj. 10 & 11

Source: New

Level of knowledge: Analysis

Author: CWS

References:
1. OP-MC-IC-IPE pages 47, 79 & 81
2. OMP 4-3 page 9

EPE: **029 Anticipated Transient Without Scram (ATWS)**

EA1.09	Manual rod control	4.0	3.6
EA1.10	Rod control function switch	3.6	3.2
EA1.11	Manual opening of the CRDS breakers	3.9*	4.1
EA1.12	M/G set power supply and reactor trip breakers	4.1	4.0
EA1.13	Manual trip of main turbine	4.1	3.9
EA1.14	Driving of control rods into the core	4.2	3.9
EA1.15	AFW system	4.1	3.9

EA2 **Ability to determine or interpret the following as they apply to a ATWS:**
(CFR 43.5 / 45.13)

EA2.01	Reactor nuclear instrumentation	4.4	4.7
[redacted]			
EA2.03	Centrifugal charging pump ammeter	2.9*	3.1*
EA2.04	CVCS centrifugal charging pump operating indication	3.2*	3.3*
EA2.05	System component valve position indications	3.4*	3.4*
EA2.06	Main turbine trip switch position indication	3.8	3.9
EA2.07	Reactor trip breaker indicating lights	4.2	4.3
EA2.08	Rod bank step counters and RPI	3.4	3.5
EA2.09	Occurrence of a main turbine/reactor trip	4.4	4.5
EA2.10	Positive displacement charging pumps	3.1*	3.4*

SEQ	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
8	Describe the function of the First-Out annunciator panel. ICIPE008		X	X	X	
9	Given a Limit and/or Precaution associated with an operating procedure, discuss its basis and applicability. ICIPE009		X	X	X	X
10	List all the Reactor Trip Signals including the setpoints, logic permissives and bases/protection afforded by each. ICIPE010		X	X	X	X
11	List all the protective system permissive ("P" signal) interlocks to include input parameter(s), logic and function. For interlocks which provide Trip block, state the Trips affected and whether Auto or Manual block. ICIPE011		X	X	X	X
12	List all the protection system control ("C" signal) interlocks including logic and functions. ICIPE012		X	X	X	X
13	Briefly describe the incident that occurred at Salem Nuclear Plant and how this event affected McGuire Reactor Trip Breaker operation. ICIPE013		X	X	X	X

Objective # 10

NC Pump Bus Under Frequency (2/4 busses = 56 Hz) - this anticipatory loss of coolant flow trip protects against DNB. *The trip also trips open all four NC pump* breakers to prevent electrical braking of the pump motors during frequency decay. A reduction in pump speed would reduce fly wheel inertia and pump coast down flow capability. This "at-power" trip protection is auto-blocked < 10% power (P-7) and *is automatically reinstated* > P-7.

SG Lo-Lo Level (2/4 channels on 1/4 SGs = 17%) - protects against a loss of heat sink. This protection also causes an auto-start of the CA motor driven pumps (2/4 channels on 1/4 SGs) and the CA turbine driven Pump (2/4 channels on 2/4 SGs).

Single Loop Loss of Flow (2/3 channels in 1/4 loops = 88%) - protects against DNB. This protection is auto-blocked < 48% (P-8) and automatically reinstated > P-8.

Two Loop Loss of Flow (2/3 channels in 2/4 loops = 88%) - protects against DNB. This protection is auto-blocked < 10% (P-7) and automatically reinstated > P-7.

Safety Injection (any SI signal 1/2 Trains) - initiates a reactor trip during LOCA events.

Turbine Trip (2/3 channels ASO < 45psig, 4/4 stop valves closed) - protects against loss of integrity by preventing Pressurizer PORVs from opening on turbine trip at high power.

Objective # 4, 10

General Warning (2/2 Trains) - protects against a loss of both protection trains. Anytime a General Warning is present on both SSPS trains a reactor trip will occur. General Warning is caused by: loose circuit board card; loss of voltage (AC or DC); SSPS train in "Test"; a Reactor Trip By-pass breaker in the Connected position and Closed; a Logic Ground Return fuse blown.

3.1.3 *Protection Permissive Interlocks*

Objective # 11

P-4 (Reactor Trip Breaker and Bypass Breaker Open for a given train) - initiates: Turbine Trip; Feedwater Isolation (coincident with low Tavg of 553 °F); Allows reset of SI signal after one minute time-out; Inputs to Steam Dump Control System for plant trip mode.

P-6 (1/2 IR instruments > 10^{-10} amps) - allows Manual Block of SR reactor trip. On a power reduction, provides automatic reinstatement of SR high voltage and SR reactor trip when 2/2 IR channels < 10^{-10} amps.

P-7 (2/4 PR instruments > 10% or 1/2 Turbine Impulse Pressures > 10%) - Enables (unblocks) the "at power" reactor trips: Pzr Hi-Level, Pzr Lo-Pressure, 2 Loop Loss of Flow, NCP UV, and NCP UF. The above trips are automatically blocked when below P-7, 3/4 PR < 10% and 2/2 Impulse Pressure < 10%.

7.5 Reactor Trips (3/27/01)

REACTOR TRIP	SETPOINT	LOGIC	PERMISSIVES	BASES
MANUAL	Sw. turned 45°	1/2 sw.		operator judgment
S.R. NI HIGH	10^5 CPS	1/2 ch.	P6, P10	uncontrolled rod withdrawal/ startup accidents
I.R. NI HIGH	amps-25% power	1/2 ch.	P10	uncontrolled rod withdrawal/ startup accidents
P.R. NI LOW	25% power	2/4 ch.	P10	reactivity excursion from low powers
P.R. NI HIGH	109% power	2/4 ch.		reactivity excursion from all powers DNB
P.R. POS RATE	+5%/2 sec	2/4 ch.		DNB (rod ejection)
PZR HIGH PRESS	2385 psig	2/4 ch.		coolant system integrity
PZR LOW PRESS	1945 psig	2/4 ch.	P7	DNB
PZR HIGH LEVEL	92%	2/3 ch.	P7	water through safeties (system integrity)
OTΔT	$\Delta T \geq OT\Delta Tsp$	2/4/ ch.		DNB
OPΔT	$\Delta T \geq OP\Delta Tsp$	2/4 ch.		KW/FT
NCP BUS LOW VOLT	74% of normal	2/4 ch.	P7	DNB (anticipatory loss of flow)
NCP BUS LOW FREQ	56 Hz	2/4 ch.	P7	DNB (anticipatory loss of flow)
S/G LO-LO LVL	17%	2/4 in 1/4 s/g		loss of heat sink
1 LOOP LOSS OF FLOW	88%	2/3 in 1/4 loops	P8	DNB
2 LOOP LOSS OF FLOW	88%	2/3 in 2/4 loops	P7	DNB
SAFETY INJECTION	any S/I signal actuated	1/2 S/I trains		trip reactor if trip not generated by trip instrumentation
GENERAL WARNING ALARM	loose card, loss of voltage, train in test, by-pass bkr connected/closed, logic ground return fuse blown	2/2 alarms		loss of protection
TURBINE TRIP	low Auto-stop oil press <45 psig or all 4 stop valves closed	2/3 ASO Press switches 4/4 valves	P8	trip reactor on turbine trip

7.6 Protection Permissive Interlocks (06/15/98)

INTERLOCKS	LOGIC	FUNCTION
P-4	Train A or B Reactor Trip	• Turbine Trip • Feedwater Isolation< Low T_{ave} • Arms condenser dumps • Allows reset of Safety Injection Signal after time delay
P-6	1/2 I.R. > 10^{-10} amps	Allows manual block of S.R. Reactor Trip. De-energizes high voltage to the Source Range detectors. On decreasing power, Source Range Level trips are automatically reactivated and high voltage restored.
P-7	2/4 P.R. > 10% FP (P-10) or 1/2 impulse pressure> 10% (P-13)	On increasing power P-7 automatically enables the following trips: • Pzr High Level • Pzr Low Pressure • Low NC Flow 2/4 Loops • NCP Undervoltage • NCP Underfrequency On decreasing power the above listed trips are automatically blocked.
P-8	2/4 P.R. > 48% FP	On increasing power P-8 enables the 1/4 loop loss of flow Reactor Trip and Reactor Trip on Turbine Trip. On decreasing power, P-8 automatically blocks the above listed trip.

T.S. REFERENCE
MANUAL

The following is a list of automatic safety signals that can be "blocked" from the main control board:

- Feedwater Isolation on Reactor Trip and Lo Tave
- Pressurizer Lo press. SI (Below P-11)
- Low Pressure Steamline Isolation (Below P-11).
- CA Auto Start on Lo Lo S/G Level or both CF pumps tripped (Below P-11)
- S/R Hi Flux Reactor Trip
- I/R Hi Flux Lo Setpoint Reactor Trip
- P/R Hi Flux Lo Setpoint Reactor Trip
- St Interlock Bypass on NC Sample valves: NM-22, NM-25, NM-26

These signals should **NOT** be "blocked" except under the direction of an approved station procedure or to better protect the health and safety of the public or to protect the lives of plant personnel.

7.8 A.T.W.S.

An A.T.W.S. (Anticipated Transient Without Scram) is defined in 10CFR50.62 as an anticipated operational occurrence followed by the failure of the reactor trip portion of the protective system. An anticipated operational occurrence is defined in 10CFR50, Appendix A, as those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are **NOT** limited to loss of power to all NC pumps, tripping of the turbine generator, isolation of the main condenser and loss of all offsite power. Clearly, to have an A.T.W.S. there must be transient followed by a failure of the reactor trip breakers.

Instrument failures, by themselves, are **NOT** necessarily transients. For example, if one channel of Pressurizer Pressure was out of service for preventive maintenance (bistable in tripped condition) and another Pressurizer Pressure channel failed (**NOT** the controlling channel), a reactor trip signal would be generated. **IF** the reactor failed to trip, this would be a failure of the reactor trip breakers and the automatic trip features of the reactor protection system and **NOT** an A.T.W.S. event. Obviously, the control operators would have to recognize and check that the channel failure was indeed a channel failure by checking the other two Pressurizer Pressure channels in this example. This would, however, force Operations to shutdown the affected unit to at least Hot Standby per Tech Specs.

7.9 Adverse Containment Setpoints

Many setpoints in the EP's are presented in a dual format with a second setpoint enclosed in parentheses. This second setpoint is used to account for the additional error in the setpoint due to the containment environment following a high-energy line break. The setpoint in parentheses will be used whenever containment pressure has exceeded 3 psig.

1 Pt. Given the following conditions on Unit 1:

- A steam leak has occurred on the main steam header
- Unit 1 reactor has been tripped and safety injection has actuated
- The MSIVs will not close
- 20 minutes into the event lowest loop NC Tcold is 305 degrees

Based on the above conditions which one of the following is the correct procedure flowpath?

A. **From E-0 *Reactor Trip or Safety Injection* GO TO FR-P.2 *(Response to Anticipated Pressurized Thermal Shock)***

B. **From E-0 go directly to ECA 2.1, *(Uncontrolled Depressurization of all Steam Generators)***

C. **From E-0 GO TO FR-P.1, *(Response to Imminent Pressurized Thermal Shock)***

D. **From E-0 GO TO E-2, *(Faulted Steam Generator Isolation)* and then to ECA 2.1**

Bank Question: 1037 ***Answer: D***

1 Pt. Given the following conditions on Unit 1:

- A steam leak has occurred on the main steam header
- Unit 1 reactor has been tripped and safety injection has actuated
- The MSIVs will not close
- 20 minutes into the event lowest loop NC Tcold is 305 degrees

Based on the above conditions which one of the following is the correct procedure flowpath?

A. **From E-0 *Reactor Trip or Safety Injection* GO TO FR-P.2 (*Response to Anticipated Pressurized Thermal Shock)***

B. **From E-0 go directly to ECA 2.1, (*Uncontrolled Depressurization of all Steam Generators)***

C. **From E-0 GO TO FR-P.1, (*Response to Imminent Pressurized Thermal Shock)***

D. **From E-0 GO TO E-2, (*Faulted Steam Generator Isolation)* and then to ECA 2.1**

Distracter Analysis:

A. **Incorrect:**
Plausible:

B. **Incorrect:**
Plausible:

C. **Incorrect:**
Plausible:

D. **Correct**
Plausible:

LEVEL: SRO

KA: W/E12 EA2.1 (3.2/4.0)

SOURCE: NEW

LEVEL OF KNOWLEDGE: Comprehension

AUTHOR: CWS

LESSON: OP-MC-EP-E2

OBJECTIVES: OP-MC-EP-E2 Obj 2,6

REFERENCES: OP-MC-EP-E2 pages 9, 15 & 23
EP/1/A/5000/F-0 page 7
EP/1/A/5000/E.2 page 2

EPE: 12 **Uncontrolled Depressurization of all Steam Generators (Continued)**

K/A NO. KNOWLEDGE

EA2. Ability to determine and interpret the following as they apply to the (Uncontrolled Depressurization of all Steam Generators)
(CFR: 43.5 / 45.13)

[redacted]

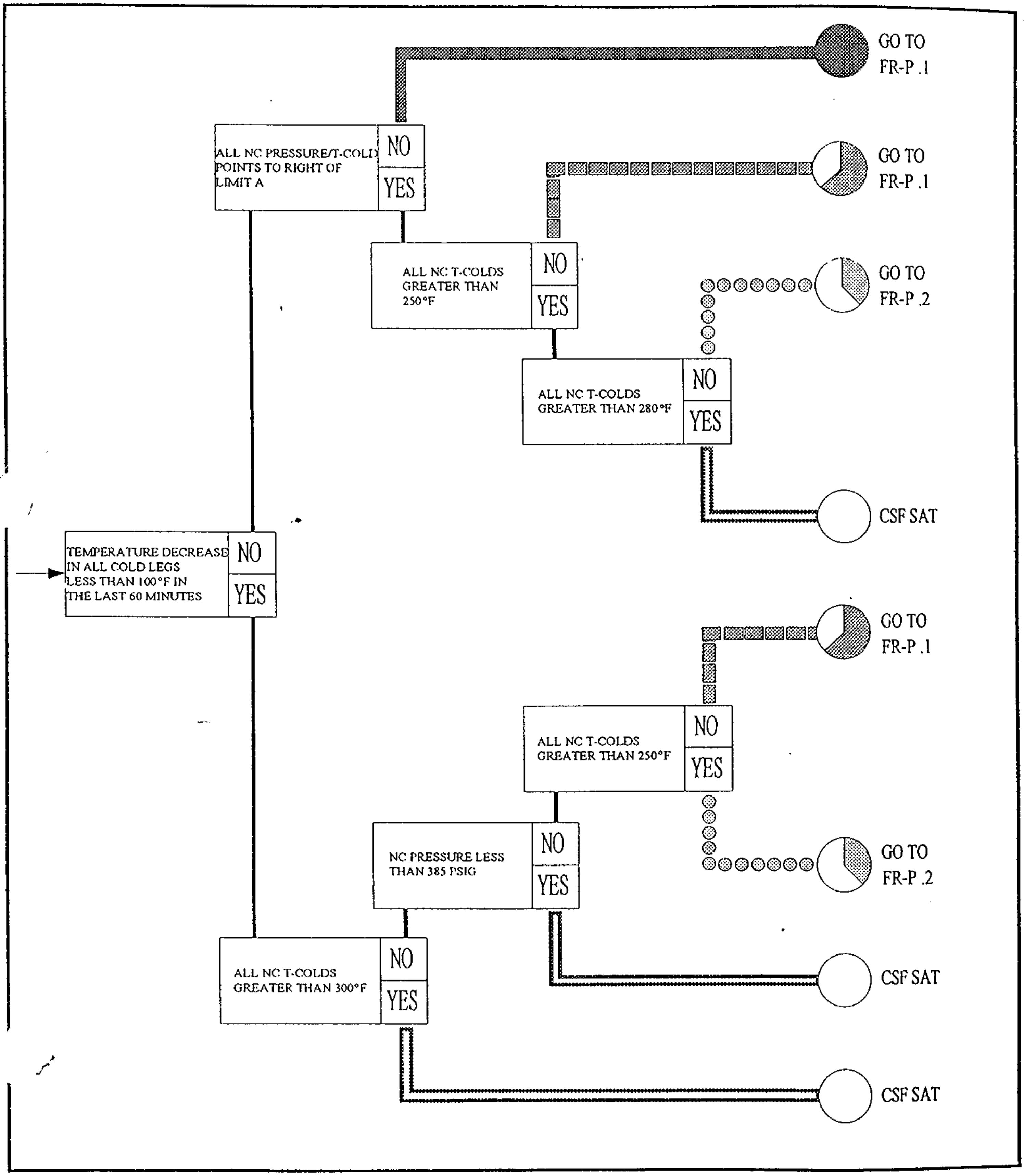
EA2.2 Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.
IMPORTANCE RO 3.4 SRO 3.9

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
N/A	N/A	.75	.75	.75

OBJECTIVES

SEQ	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	Explain the purpose for each procedure in the E-2 series. EPE2001			X	X	
2	Discuss the entry and exit guidance for each procedure in the E-2 series. EPE2002			X	X	
3	Discuss the mitigating strategy (major actions) of each procedure in the E-2 series. EPE2003			X	X	X
4	Discuss the basis for any note, caution or step for each procedure in the E-2 series. EPE2004			X	X	X
5	Given the Foldout page discuss the actions included and the basis for these actions. EPE2005			X	X	X
6	Given the appropriate procedure, evaluate a given scenario describing accident events and plant conditions to determine any required action and its basis. EPE2006			X	X	X



GO TO FR-P.1
ALL NC PRESSURE/T-COLD POINTS TO RIGHT OF LIMIT A
NO
YES
GO TO FR-P.1
ALL NC T-COLDS GREATER THAN 250°F
NO
YES
GO TO FR-P.2
ALL NC T-COLDS GREATER THAN 280°F
NO
YES
CSF SAT
TEMPERATURE DECREASE IN ALL COLD LEGS LESS THAN 100°F IN THE LAST 60 MINUTES
NO
YES
GO TO FR-P.1
ALL NC T-COLDS GREATER THAN 250°F
NO
YES
NC PRESSURE LESS THAN 385 PSIG
NO
YES
GO TO FR-P.2
CSF SAT
ALL NC T-COLDS GREATER THAN 300°F
NO
YES
CSF SAT

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C. Operator Actions	
__ 1. Monitor foldout page.	
__ 2. Maintain at least one S/G available for NC System cooldown in subsequent steps.	
__ 3. Maintain any faulted S/G or secondary break isolated during subsequent recovery actions unless needed for NC System cooldown.	
4. Check the following - CLOSED: __ • All MSIVs __ • All MSIV bypass valves.	__ Close valves.
__ 5. Check at least one S/G pressure - STABLE OR GOING UP.	__ IF all S/Gs faulted, THEN GO TO EP/1/A/5000/ECA-2.1 (Uncontrolled Depressurization Of All Steam Generators).
6. Identify faulted S/G(s): __ • Any S/G pressure - GOING DOWN IN AN UNCONTROLLED MANNER OR __ • Any S/G - DEPRESSURIZED.	Perform the following: a. Dispatch operators to search for initiating break: __ • Main steamlines __ • Main feedlines __ • Other secondary piping. __ b. GO TO Step 11.

1.0 INTRODUCTION

E-2, Faulted Steam Generator Isolation, provides actions to identify and isolate a faulted steam generator (S/G). The procedure is entered from E-0, Reactor Trip or Safety Injection, or E-1, Loss of Reactor or Secondary Coolant, when any S/G pressure goes down in an uncontrolled manner or any S/G is completely depressurizes. Other procedures have a transition to E-2 whenever a faulted S/G is identified and faulted S/G isolation is not confirmed.

After taking the required actions in this procedure, the operator is directed to either E-1, Loss of Reactor or Secondary Coolant, or E-3, Steam Generator Tube Rupture, depending on whether a steam generator tube rupture (SGTR) is identified.

ECA-2.1, Uncontrolled Depressurization of All Steam Generators, provides procedural guidance to recover from an event where all S/Gs are depressurizing in an uncontrolled manner. This procedure is entered from E-2 when an uncontrolled depressurization of all S/Gs occurs. Potential initiating events for this contingency could include steamline breaks, stuck open relief or safety valves, or any combination of conditions that would affect all S/Gs.

ECA-2.1 is exited whenever any S/G pressure boundary is reestablished as indicated by a rise in the associated S/G pressure indication. In this case, the operator transfers to E-2 for further recovery actions.

1.1. Emergency Procedures in this Series

E-2 Faulted Steam Generator Isolation

ECA-2.1 Uncontrolled Depressurization of All Steam Generators

ECA-2.1, Uncontrolled Depressurization of All Steam Generators, provides procedural guidance to recover from an event where all S/Gs are depressurizing in an uncontrolled manner.

An uncontrolled depressurization of all S/Gs initiates from a failure/break in a main steamline, main feedwater line, and/or in any piping system that interconnects with the secondary side pressure boundary. This event results in an extensive cooldown and pressure transient. The consequences vary considerably depending upon system parameters:

1. Size(s) and location(s) of the break(s)
2. Operational safety systems
3. Operational control systems
4. Initial power level
5. Failures which may occur.

It should be noted that this event (with an extensive cooldown and subsequent repressurization) might result in a challenge to the Integrity Critical Safety Function (CSF). In this case the Integrity CSF Status Tree may direct the operator to FR-P.1, Response To Imminent Pressurized Thermal Shock Condition, for further actions.

The following describe the purpose of the two emergency procedures in the E-2 series.

E-2, Faulted Steam Generator Isolation, provides actions to identify and isolate a faulted S/G.

ECA-2.1, Uncontrolled Depressurization of All Steam Generators, provides actions for a loss of secondary coolant that affects all S/Gs.

STEP 5 Check at least one S/G pressure - STABLE OR GOING UP:

PURPOSE: To ensure there is at least one non-faulted S/G.

BASIS: Any cooldown operations that are performed as subsequent recovery actions will require at least one non-faulted S/G. If all S/G pressures are going down in an uncontrolled manner, this indicates a failure affecting all S/Gs. Recovery actions, in this case, should be performed using ECA-2.1, Uncontrolled Depressurization of All Steam Generators, since feedwater flow will be necessary to a faulted S/G and normal level control should not be used.

One faulted S/G may cause the intact S/G pressures to drop just due to the cooldown. *In this case, the operator should continue in E-2.*

STEP 6 Identify faulted S/G(s)

PURPOSE: To identify any faulted S/G.

BASIS: An uncontrolled S/G pressure drop (following MSIV closure and feedwater isolation) or a completely depressurized S/G indicates an unisolable failure of the secondary pressure boundary. The operator is directed to search for the initiating break in main steamlines, feedlines, or other secondary piping such as blowdown lines, sample lines, etc. The operator should also check for stuck open atmospheric steam dump valves and/or safety valves.

1 Pt. Given the following conditions on Unit 1:

- Chemistry had confirmed two leaking fuel rods
- A large break LOCA occurs
- E-0 *Reactor Trip or Safety Injection* is complete
- ES-1.3 *Transfer to Cold Leg Recirc* is complete
- E-1 *Loss of Reactor or Secondary Coolant* is complete
- ES-1.2 *Post LOCA Cooldown and Depressurization* is in effect.
- All Red and Orange Paths have been addressed
- 1EMF 51A is reading 39R/HR
- Pressurizer level is 0%

The SRO is currently considering implementing Yellow Path procedures. Which one of the following describes proper procedure implementation?

A. **Go to FR-I.3, (*Response to Voids in the Reactor Vessel)* and exit ES-1.2**

B. **Stay in ES-1.2 and implement FR-I.3 concurrently**

C. **Go to FR-Z.3, (*Response to High Containment Radiation Level)* and exit ES-1.2**

D. **Stay in ES-1.2 and implement FR-Z.3 concurrently**

Bank Question: 1042 ***Answer: D***

1 Pt. Given the following conditions on Unit 1:

- Chemistry had confirmed two leaking fuel rods
- A large break LOCA occurs
- E-0 *Reactor Trip or Safety Injection* is complete
- ES-1.3 *Transfer to Cold Leg Recirc* is complete
- E-1 *Loss of Reactor or Secondary Coolant* is complete
- ES-1.2 *Post LOCA Cooldown and Depressurization* is in effect.
- All Red and Orange Paths have been addressed
- 1EMF 51A is reading 39R/HR
- Pressurizer level is 0%

The SRO is currently considering implementing Yellow Path procedures. Which one of the following describes proper procedure implementation?

A. **Go to FR-I.3, (*Response to Voids in the Reactor Vessel)* and exit ES-1.2**

B. **Stay in ES-1.2 and implement FR-I.3 concurrently**

C. **Go to FR-Z.3, (*Response to High Containment Radiation Level)* and exit ES-1.2**

D. **Stay in ES-1.2 and implement FR-Z.3 concurrently**

Distracter Analysis:.

A. **Incorrect**: ES1.2 is the controlling procedure and not to be exited
Plausible:
B. **Incorrect**: FR-I-3 is a lower priority than FR-Z-3
Plausible:
C. **Incorrect**: ES1.2 is controlling procedure
Plausible:
D. **Correct**
Plausible:

LEVEL: SRO

KA: W/E16 EA2.1 (4.3/4.4)

SOURCE: NEW

LEVEL OF KNOWLEDGE: Analysis

AUTHOR: CWS

LESSON: OP-MC-EP-F0

OBJECTIVES: OP-MC-EP-F0 Obj 3

REFERENCES: OP-MC-EP-F0 page 17
OMP 4-3 page 19

EPE: 16 **High Containment Radiation (Continued)**

K/A NO. KNOWLEDGE

EK3.2 Normal, abnormal and emergency operating procedures associated with (High Containment Radiation).
IMPORTANCE RO 2.9 SRO 3.3

EK3.3 Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations.
IMPORTANCE RO 3.0 SRO 3.0

| EK3.4 RO or SRO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated.
IMPORTANCE RO 3.0 SRO 3.2

ABILITY

EA1. Ability to operate and / or monitor the following as they apply to the (High Containment Radiation)
(CFR: 41.7 / 45.5 / 45.6)

EA1.1 Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
IMPORTANCE RO 3.1 SRO 3.2

EA1.2 Operating behavior characteristics of the facility.
IMPORTANCE RO 2.9 SRO 3.0

EA1.3 Desired operating results during abnormal and emergency situations.
IMPORTANCE RO 2.9 SRO 3.3

EA2. Ability to determine and interpret the following as they apply to the (High Containment Radiation)
(CFR: 43.5 / 45.13)

EA2.1 Facility conditions and selection of appropriate procedures during abnormal and emergency operations.
IMPORTANCE RO 2.9 SRO 3.3

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
		2	2	2

OBJECTIVES

SEQ	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	State the purpose of each of the six CSF Status Trees. EPF0001			X	X	
2	Explain the priority system associated with the CSF status trees. EPF0002			X	X	X
3	Explain the "Rules of Usage" for Critical Safety Function status trees. EPF0003			X	X	X
4	Explain the bases for all blocks in the six Status Trees. EPF0004			X	X	X

7.15.1.7 Yellow Path

A yellow path does <u>NOT</u> require immediate operator attention. Frequently, it is indicative of an off-normal and/or temporary condition which will be restored to normal status by actions already in progress. In other cases, the yellow status might provide an early indication of a developing red or orange condition. The operator is allowed to decide whether or <u>NOT</u> to implement any yellow path procedure.
Implementation of a yellow path function restoration guideline is based on operator judgment when it is determined that adequate time exists to implement it. In other words, the operator does <u>NOT</u> have to implement a yellow path guideline if a judgment has been made that it is inappropriate based on available time or current plant state; and if an event of higher priority is in progress, the operator should attend to the more important matters prior to implementing a yellow path function restoration guideline. In the prioritization scheme in the EPs, the optimal recovery procedures (including applicable foldout pages) have priority over the yellow path function restoration procedures. The yellow path procedure can be considered as a supplementary set of actions that were provided to address one parameter being in an off-normal state. The controlling guideline in effect is the optimal recovery procedure that the operator is in when he decides that he has enough time to perform the yellow path procedure actions. **While performing the actions of the yellow path, continuous actions or foldout page items of the optimal recovery procedure in effect are still applicable and should be monitored by the operator.** This concurrent procedure usage should <u>NOT</u> cause the operator any difficulties since yellow path procedures are only performed when adequate time exists.

For example, if the operator is in ES-1.1 (SI Termination) and decides to implement FR-H.5 because of low SG level and NC subcooling is lost while in FR-H.5, the operator should terminate FR-H.5 and implement the action of the ES-1.1 foldout page to reinitiate SI flow.

8. Yellow Path

A yellow path does not require immediate operator attention. Frequently, it is indicative of an off-normal and/or temporary condition, which will be restored to normal status by actions already in progress. In other cases, the yellow status might provide an early indication of a developing red or orange condition. The operator is allowed to decide whether or not to implement any yellow path procedure.

Yellow path implementation is a judgement call based on current plant conditions and time available. If a higher priority event is in progress the operators are expected to attend to the most important event. In the prioritization scheme in the EPs, the optimal recovery procedures (including applicable foldout pages) have priority over the yellow path function restoration procedures. The yellow path procedure can be considered as a supplementary set of actions that were provided to address one parameter being in an off-normal state. The controlling procedure in effect is the optimal recovery procedure that the operator is in when he/she decides there is sufficient time to perform the yellow path procedural actions. <u>While performing the actions of the yellow path, continuous actions, or foldout page items of the optimal recovery procedures in effect are still applicable and should be monitored by the operator.</u> This concurrent procedure usage should not cause the operator any difficulties since yellow path procedures are only performed when adequate time exists.

9. SPDS

Normally, the condition of the status trees is continuously monitored and displayed by the OAC. The OAC can be used to validate any off normal alarm and to determine which EP to implement. The entire control room crew is responsible for monitoring the SPDS.

10. How Long to Monitor Status Trees

Monitoring of status trees may be stopped when any of the following are met:

- Cold shutdown or TSC concurrence.

  OR
- Transition to normal recovery procedure (OP).

  OR
- Transition to Severe Accident Management Guideline (SAMG).

1 Pt(s) Given the following conditions on Unit 1:

- Mode 3
- NC System is at 1700 psig and 450 degrees
- In process of cooling down and depressurizing the NC System
- Safety Injection has occurred
- NC Pressure going down in an uncontrolled manner
- Containment pressure going up in uncontrolled manner

Which one of the following describes the proper procedures to mitigate the above?

A. **Enter AP/35 (*ECCS Actuation During Plant Shutdown*) and then go to E-0 (Reactor Trip or Safety Injection).**

B. **Enter E-0 and then go to AP/35**

C. **Enter AP/35 and then go to AP/34 (Shutdown LOCA)**

D. **Enter E-0 and then go to E-1 (*Loss of Reactor or Secondary Coolant*).**

Bank Question: 1053 ***Answer: A***

1 Pt(s) Given the following conditions on Unit 1:

- Mode 3
- NC System is at 1700 psig and 450 degrees
- In process of cooling down and depressurizing the NC System
- Safety Injection has occurred
- NC Pressure going down in an uncontrolled manner
- Containment pressure going up in uncontrolled manner

Which one of the following describes the proper procedures to mitigate the above?

A. Enter AP/35 (*ECCS Actuation During Plant Shutdown*) and then go to E-0 (Reactor Trip or Safety Injection).

B. Enter E-0 and then go to AP/35

C. Enter AP/35 and then go to AP/34 (Shutdown LOCA)

D. Enter E-0 and then go to E-1 (*Loss of Reactor or Secondary Coolant*).

Distracter Analysis:.

A. Correct:
Plausible:
B. Incorrect:
Plausible:
C. Incorrect:
Plausible:
D. Incorrect
Plausible:

LEVEL: SRO

KA: 006 G 2.4.4 (4.0/4.3)

SOURCE: NEW

LEVEL OF KNOWLEDGE: ANALYSIS

AUTHOR: CWS

LESSON: - AP/1/5500/35 Background Document

OBJECTIVES: OP-MC-AP-35 Obj. 1
MC-AP-34 Obj. 1

REFERENCES: AP/1/5500/35 Background Document pages 2-4
AP/1/A/5500/35 page 3
AP/34 Backdground Document page 2

2.4 Emergency Procedures /Plan

2.4.1 **Knowledge of EOP entry conditions and immediate action steps.**
(CFR: 41.10 / 43.5 / 45.13)
IMPORTANCE RO 4.3 SRO 4.6

2.4.2 **Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.**
(CFR: 41.7 / 45.7 / 45.8)
Note: The issue of setpoints and automatic safety features is not specifically covered in the systems sections).
IMPORTANCE RO 3.9 SRO 4.1

2.4.3 **Ability to identify post-accident instrumentation.**
(CFR: 41.6 / 45.4)
IMPORTANCE RO 3.5 SRO 3.8

2.4.4 **Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.**
(CFR 41.10 / 43.2 / 45.6)
IMPORTANCE RO 4.0 SRO 4.3

2.4.5 **Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions.**
(CFR: 41.10 / 43.5 / 45.13)
IMPORTANCE RO 2.9 SRO 3.6

2.4.6 **Knowledge symptom based EOP mitigation strategies.**
(CFR: 41.10 / 43.5 / 45.13)
IMPORTANCE RO 3.1 SRO 4.0

2.4.7 **Knowledge of event based EOP mitigation strategies.**
(CFR: 41.10 / 43.5 / 45.13)
IMPORTANCE RO 3.1 SRO 3.8

2.4.8 **Knowledge of how the event-based emergency/abnormal operating procedures are used in conjunction with the symptom-based EOPs.**
(CFR: 41.10 / 43.5 / 45.13)
IMPORTANCE RO 3.0 SRO 3.7

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
		1.5	1.5	1.5

OBJECTIVES

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	Concerning AP/1(2)/5500/35 (ECCS Actuation During Plant Shutdown): • State the purpose of the AP • Recognize the symptoms that would require implementation of the AP. AP35001			X	X	X
2	Given scenarios describing accident events and plant conditions, evaluate the basis for any caution, note, or step. AP35002			X	X	X

INTRODUCTION

AP/35 covers operator actions for an ECCS actuation from initial plant conditions below P-11 (Safety Injection Block Permissive, less than 1955 psig).

Summary

This procedure provides guidance to the operator in responding to the above abnormal conditions. The actions do not defeat any safety functions or prevent the required operational features of any safety system from performing as required. This AP is for use during inadvertent ECCS actuations while shutdown (below P-11). Scenarios for inadvertent ECCS actuation above P-11 are addressed by direct entry into E-0.

VALID S/I EVENTS: (Since this AP is not for events that require SI flow, early kickouts provide direction to the appropriate procedure)

This AP is not for use involving events that lead to a required ECCS Actuation. There should be no required ECCS actuations in Mode 5. For S/G tube ruptures or steamline breaks from Mode 4 up to P-11, direction is given to go to E-0 since AP/34 is only written for LOCA's on the NC System. For LOCA's from P-11 down to CLA isolation, direction is given to go to E-0. For LOCA's from CLA isolation down through Mode 4, direction is given to go to AP/34. For LOCA's in Mode 5, this AP shouldn't even be entered. These Mode 5 scenarios should use AP/10 and/or AP/19, as appropriate. Note it's likely Mode 3 or 4 LOCA scenarios have symptoms that would lead the crew to AP/34 prior to the ECCS actuation (manual or Containment SI). **Even though both AP/34 and AP/35 would be in effect if an actuation subsequently happened, the crew should stay in AP/34 as a higher priority.** However, even if they went to AP/35, this would not be a concern since AP/35 would quickly kick out to AP/34.

Note "SI" is not manually initiated in this AP. If there is a need for "SI", you should not be in this AP because this AP is not for events requiring ECCS actuation. This AP assumes the event is inadvertent actuation. With the plant shutdown and depressurized, refill of the Pzr will occur much quicker, challenging Pzr PORVs, Pzr Safeties, and ND suction reliefs. This point is emphasized to support a course of action contrary to most operator training. IF ONE TRAIN OF "SI" ACTUATES, DO **NOT** ACTUATE THE SECOND TRAIN. This would needlessly challenge the mitigating strategy of this AP.

Finally, this AP is not written assuming an inadvertent Phase "B" or NS actuation has occurred. Note the title of the AP is specifically ECCS actuation, not ESF actuations in general.

ENTRY CONDITIONS

Entry to this procedure will occur if there is actuation of ECCS equipment. This includes valid or inadvertent actuations. For valid actuations, this AP will quickly provide direction to go to E-0 or AP/34, in the unlikely event that procedure was not already in effect.

STEP DESCRIPTION FOR ECCS ACTUATION DURING PLANT SHUTDOWN

STEP 1:

PURPOSE:

Cue the operator to monitor the foldout page.

DISCUSSION:

The use of a foldout page is unusual for an AP. One was chosen for this AP as a human-factors' consideration. Maintaining critical items on a separate page ensures they are performed in a timely manner. The foldout page contains actions that apply throughout the AP as described in item below:

"NC pump trip criteria". Tripping the NC pumps on low seal D/P or low leakoff ensure the operator is alerted that minimum #1 seal operating conditions may be lost during depressurization or Phase "A" isolation. Phase "A" isolation closes the NCP seal return valves, reducing seal D/P by approximately 100 PSID. The 200 PSID and 0.2 gpm leakoff setpoints are based on the operating limits stated in the NC Pump Tech Manual (MCM 1201-01-193, Controlled Leakage Seal Reactor Coolant Pump). These limits ensure sufficient seal and pump radial bearing lubrication and cooling. If associated with a spray valve, the spray valve is closed to facilitate pressure control now or later. Even though the pump is secured, seal injection flow is needed to supply cool filtered water to the seals.

REFERENCES:
MCM 1201-01-193, Controlled Leakage Seal Reactor Coolant Pump

STEP 2:

PURPOSE:

Direct the operator to the appropriate procedure for real events, since this AP is only for inadvertent ECCS actuation scenarios.

DISCUSSION:

If temperature is below 200°F, the ECCS actuation is very likely to be inadvertent, and so the operator would continue on in the AP to recover from the inadvertent actuation.

If temperature is above 200°F, the actuation could have been caused by a event requiring S/I flow (SGTR, steam leak, or LOCA). There is no procedure guidance in AP/34 for SGTRs or steam leak, so if one of these events has occurred, then no matter what temperature or mode, direction is given to go to E-0. If the event is a LOCA, then direction is given to go to the appropriate procedure, AP/34 if CLAs isolated, E-0 if unisolated.

If above 200°F, but none of the listed events has occurred, then it is an inadvertent actuation, and the operator would continue on with the next step.

The actuation isolates cooling water to the letdown Hx, so letdown from ND is isolated since it doesn't isolate automatically on the actuation. This should prevent the possibility of flashing/water hammer concerns.

REFERENCES:

STEP 3:

PURPOSE:

Ensure NV pumps have a suction source.

DISCUSSION:

The ECCS actuation may be partial. Either VCT outlet closing would isolate the VCT, and even without it isolating, if the NV pump flowrate is high, it wouldn't take long to empty the VCT. It is prudent to ensure the FWST suction is aligned, early in the procedure. The other ECCS pump suctions are not checked since their suctions are normally aligned from the FWST.

REFERENCES:

STEP 4:

PURPOSE:

Ensure proper RN System alignment, especially for partial/one train actuation scenarios.

DISCUSSION:

One possible scenario could be "B" RN Train in operation prior to the event. If only an "A" Train actuation occurred, "B" RN Pump would lose its' suction from LLI and discharge path to RC via the "A" Train valves. Without a "B" Train actuation, it wouldn't align to the SNSWP leaving it without a suction or discharge. Therefor, direction is provided to align "B" Train to the pond.

Another possible scenario is just a "B" Train actuation, in which case the RN non-essential header could lose flow. Therefor, direction is given to ensure both RN pumps are on, so that "A" RN Pump will supply the non-essential headers.

An actuation on Unit 1 will isolate Unit 2's "B" Train from the non-essential header on Unit 2. Direction is given to have Unit 2 start it's "A" RN Pump if running NCPs so they don't lose cooling water.

REFERENCES:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

C. Operator Actions

__ 1. **Monitor foldout page.**

__ 2. **Check NC temperature - LESS THAN 200° F.**

Perform the following:

__ a. Ensure 1NV-121 (ND Letdown Control) is closed.

b. IF S/I was initiated due to known S/G tube rupture or steamline break, THEN:

__ 1) IF CLAs have been isolated, THEN leave them isolated in EPs.

__ 2) GO TO EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

c. Check for symptoms of LOCA:

__ • Pzr level - GOING DOWN IN UNCONTROLLED MANNER

__ • NC pressure - GOING DOWN IN UNCONTROLLED MANNER

__ • Containment pressure - GOING UP IN UNCONTROLLED MANNER.

d. IF LOCA has occurred, THEN:

__ 1) IF prior to isolation of CLAs, THEN GO TO EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

__ 2) IF in mode 3 after CLAs isolated or in mode 4, THEN GO TO AP/1/A/5500/34 (Shutdown LOCA).

3. **Ensure the following valves - OPEN:**

__ • 1NV-221A (NV Pumps Suct From FWST)

__ • 1NV-222B (NV Pumps Suct From FWST).

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
		1.5	1.5	1.5

OBJECTIVES

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	Concerning AP/1(2)/5500/34 (Shutdown LOCA): • State the purpose of the AP • Recognize the symptoms that would require implementation of the AP. AP34001			X	X	X
2	Given scenarios describing accident events and plant conditions, evaluate the basis for any caution, note, or step. AP34002			X	X	X

INTRODUCTION

AP/34 provides the actions for protecting the reactor core in the event of a LOCA that occurs during either Mode 3 after the Cold Leg Accumulators are isolated or Mode 4.

Summary

This procedure provides guidance to the operator in responding to the above abnormal conditions. The actions do not defeat any safety functions or prevent the required operational features of any safety system from performing as required

After entering AP/34, the operator will increase charging flow in an attempt to maintain Pzr level. If both Pzr level and RCS subcooling can be restored with normal charging flow, the operator can terminate AP/34 and return to the appropriate procedure. If normal charging can't maintain both subcooling and Pzr level, the operator will continue with AP/34 to respond to the LOCA.

If the switchover setpoint in the FWST is reached, which could happen at any time during the AP depending on the size of the LOCA, the operator will align the ECCS for cold leg recirculation to maintain coolant flow to the core using Enclosure 2.

After reaching and maintaining cold shutdown conditions (NC less than 200°F), the final step of AP/34 instructs the operators and plant engineering staff to evaluate the long term plant status. At this time, the NC System will be cooled by the ND System or cold leg recirculation. If Pzr level could not be restored and maintained and boron precipitation is a concern, a decision to transfer to hot leg recirculation could be made. Other long term recovery actions can also be determined at this time.

For a LOCA that occurs during shutdown operation, the two largest concerns are **core heat-up** and **cold overpressurization**. An evaluation has been performed to determine that establishing safety injection from one high head SI (NV or NI) pump within ten minutes and flow from a second high head SI pump within 30 minutes WILL successfully mitigate (**core heat-up**) for a small break LOCA (less than 6" in diameter). Hence, the CLA's are not required for a shutdown LOCA event. To successfully mitigate **cold overpressurization** AP/34 only starts SI pumps as required.

ENTRY CONDITIONS

Entry to this procedure will occur if there is an uncontrolled decrease in Pzr level or NC subcooling, or Containment Floor and Equipment Sump level increase, while in Mode 3 after the CLA's are isolated or in Mode 4.

1 Pt Given the following conditions on Unit 1:

- Unit 1 is at 100% power.
- 'A', 'B', and 'C' VL AHU are running
- 'A' and 'C' VL AHUs have tripped and will not restart
- Attempts to start 'D' VL AHU were unsuccessful
- Average temperature in lower containment for past 365 days has been 105 degrees.
- Maintenance indicated it will take two days to repair the VL AHUs.
- Containment lower compartment temperature is 126 degrees and steady.

Which one (1) of the following describes the required Technical Specification actions to address the high containment temperature?

Reference Provided

A. Restore temperature to within limits in 8 hours.

B. Reduce temperature to <125 degrees in 72 hours.

C. No action is required to address high containment temperature.

D. Be in Mode 3 in 14 hours.

Bank Question: 1054 **_Answer: C_**

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- 'A', 'B', and 'C' VL AHU are running
- 'A' and 'C' VL AHUs have tripped and will not restart
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- Maintenance indicated it will take two days to repair the VL AHUs.
- Containment lower compartment temperature is 126 degrees and steady.

Which one (1) of the following describes the required Technical Specification actions to address the high containment temperature?

Reference Provided Tech Spec 3.6.5

A. Restore temperature to within limits in 8 hours.

B. Reduce temperature to <125 degrees in 72 hours.

C. No action is required to address high containment temperature.

D. Be in Mode 3 in 14 hours.

Distracter Analysis:.

A. Incorrect:
Plausible:
B. Incorrect:
Plausible:
C. Correct:
Plausible:
D. Incorrect
Plausible:

LEVEL: SRO

KA: SYS 022 G2.1.12 (2.9/4.0)

SOURCE: NEW

LEVEL OF KNOWLEDGE: Analysis

AUTHOR: CWS

LESSON: OP-MC-CNT-VUL

OBJECTIVES: OP-MC-CNT-VUL Obj. 11

REFERENCES: OP-MC-CNT-VUL pages 15 & 17
Tech Spec 3.6.5

2.1 Conduct of Operations (continued)

2.1.9 **Ability to direct personnel activities inside the control room.**
(CFR: 45.5 / 45.12 / 45.13)
IMPORTANCE RO 2.5 SRO 4.0

2.1.10 **Knowledge of conditions and limitations in the facility license.**
(CFR: 43.1 / 45.13)
IMPORTANCE RO 2.7 SRO 3.9

2.1.11 **Knowledge of less than one hour technical specification action statements for systems.**
(CFR: 43.2 / 45.13)
IMPORTANCE RO 3.0 SRO 3.8

2.1.12 **Ability to apply technical specifications for a system.**
(CFR: 43.2 / 43.5 / 45.3)
IMPORTANCE RO 2.9 SRO 4.0

2.1.13 **Knowledge of facility requirements for controlling vital / controlled access.**
(CFR: 41.10 / 43.5 / 45.9 / 45.10)
IMPORTANCE RO 2.0 SRO 2.9

2.1.14 **Knowledge of system status criteria which require the notification of plant personnel.**
(CFR: 43.5 / 45.12)
IMPORTANCE RO 2.5 SRO 3.3

2.1.15 **Ability to manage short-term information such as night and standing orders.**
(CFR: 45.12)
IMPORTANCE RO 2.3 SRO 3.0

2.1.16 **Ability to operate plant phone, paging system, and two-way radio.**
(CFR: 41.10 / 45.12)
IMPORTANCE RO 2.9 SRO 2.8

2.1.17 **Ability to make accurate, clear and concise verbal reports.**
(CFR: 45.12 / 45.13)
IMPORTANCE RO 3.5 SRO 3.6

OBJECTIVES

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
9.	Given a limit and/or precaution associated with an operating procedure, discuss its basis and applicability.		X	X	X	X
10.	Describe the available indications of containment air temperature.		X	X	X	X
11.	Concerning the Technical Specifications associated with the Containment Ventilation System:					
	• Given the LCO Title, state the LCO (including any COLR values) and applicability.			X	X	X
	• For any LCO's that have action required within one hour, state the action.			X	X	X
	• Given a set of parameter values or system conditions, determine if any Tech Spec LCO(s) is (are) not met and any action(s) required within one hour.			X	X	X
	• Given a set of plant parameters or system conditions and the appropriate Tech Specs, determine required actions.			X	X	X
	• Discuss the basis for a given Tech Spec LCO or Safety Limit.				X	*
	* SRO ONLY					

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature

LCO 3.6.5 Containment average air temperature shall be:

a. ≥ 75°F and ≤ 100°F for the containment upper compartment, and

b. ≥ 100°F and ≤ 120°F for the containment lower compartment.

------NOTES------

1. The minimum containment average air temperature in MODES 2, 3, and 4 may be reduced to 60°F.

2. Containment lower compartment temperature may be between 120°F and 125°F for up to 90 cumulative days per calendar year provided lower compartment temperature average over the previous 365 days is less than 120°F. Within this 90 cumulative day period, lower compartment temperature may be between 125°F and 135°F for 72 cumulative hours.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment average air temperature not within limits.	A.1 Restore containment average air temperature to within limits.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1	Verify containment upper compartment average air temperature is within limits.	24 hours
SR 3.6.5.2	Verify containment lower compartment average air temperature is within limits.	24 hours

These units are shunt-tripped from the essential power system upon receipt of an S_S signal. Once the shunt trip occurs, the fan's HVAC panel "ON – OFF" indication and control power is lost. The motors are overload protected.

The purpose of the upper containment return air fans is to remove air from the dome area of the Containment to prevent stratification. The fans draw the air from the dome and discharge near the suction of an associated upper containment ventilation unit. There are four (4) fans associated with this purpose, two are normally operating. Each fan is interlocked with a corresponding upper containment ventilation unit so that the fans operate in conjunction with their associated upper containment ventilation unit. These are driven by 1HP, 3 phase, non-nuclear safety related motors and perform no emergency functions. The motors are overload protected.

Objective #8

Each Return Air Fan is provided with a selector switch ("AUTO-START-STOP" pushbutton) on the HVAC panel. Each may be manually started or placed in the auto mode. If in auto, the return air fan will start when its corresponding air handling unit is started. Status indication is provided on the HVAC panel.

The normal power supply for these units is a 600VAC Essential Motor Control Center. Following a Blackout, all units will be sequenced on regardless of switch position. Under these conditions, these fans can not be stopped until the sequencer is reset. These units are shunt-tripped from the essential power system upon receipt of an S_S signal and the "ON-OFF" indication on the HVAC panel is lost.

2.2. Lower Containment Ventilation System (VL)

The VL system is regulatory-required per Technical Specification surveillance requirements and performs no safety-related functions. The VL system is designed to maintain a maximum temperature (120°F) inside the lower Containment compartment during normal operation and a minimum temperature (60°F) during shutdown.

Per Tech Spec 3.6.5, the average air temperature for Lower Containment shall be $\geq$ 100°F and $\leq$ 120°F in Modes 1 - 4.

NOTES:

1. The minimum containment average air temperature in MODES 2, 3 and 4 may be reduced to 60°F.
2. Containment lower compartment temperature may be between 120°F and 125°F for up to 90 cumulative days per calendar year provided lower compartment temperature average over the previous 365 days is less than 120°F. Within this 90 cumulative day period, lower compartment temperature may be between 125°F and 135°F for 72 cumulative hours.

This temperature range was determined by incorporating the following temperature limits: (1) the lower Containment compartment temperature assumed in the Containment accident analyses, (2) the equipment qualification temperatures, and (3) temperature requirements for personnel access.

The Lower Containment Weighted Average Temperature (LCWAT) is used by the operator to determine optimum VL/RV/RN operations. The LCWAT program calculates the LCWAT using only the operating VL units inputs (temperatures associated with idle fans are not used).

It is desirable for the VL system to operate during events such as a small isolable LOCA, small main steam break inside Containment, blackout and LOOP to avoid a rise in Containment pressure such that Containment Spray is unnecessarily actuated. Provisions in the design were made such that selected equipment from this system is capable of receiving safety-related 1E power.

The VL system consists of four (4) recirculating ventilation units and their associated cooling coils, fans, and associated ductwork. This equipment is located in the annular concrete chambers around the periphery of the lower Containment compartment (Fan Rooms). The temperature in the annulus between the reactor vessel and the primary shield may exceed the maximum average temperature of lower Containment (This temperature may be allowed to reach 135°F without detrimental effects to the installed instrumentation.)

Objective #8

Each VL AHU has an "OFF-LOW-HIGH" selector switch on the HVAC panel. The VL fan motors are overload protected and status indication is provided on the HVAC panel. Annunciators are provided to indicate mixed speed operation, transfer to emergency power, high speed start and high vibration. Bearing temperatures are monitored by the OAC. The 2A, 2B & 2C VL fan motors are a different design motor which is designed to operate at a higher temperature than the others on Unit 1 & 2. Therefore, these same VL fan motors have higher bearing temperature alarm setpoints on the OAC (see MM-10562). Discharge check damper position is provided on the HVAC panel. Each VL AHU has a suction damper control switch ("AUTO-OPEN-CLOSE" pushbutton) on the HVAC panel. Each VL ventilation unit fan has two-speed capability. At high speed the associated fan operates at 1800 rpm and at low speed the fan operates at 900 rpm.

Objective #2

The cooling water supply is from the Containment Ventilation Cooling Water (RV) system. Nuclear Service Water (RN) through the RV System is the preferred source of cooling water in Modes 1 through 5. In Mode 6 or No Mode, cooling water is not required. Cooling water flow is maintained to the VL ventilation units until the "Phase B" signal is received. Since the cooling water for the VL ventilation units is raw water, fouling of these tubes is a problem. As the fouling of the heat transfer area increases, the efficiency of the cooling coil is decreased, thus increasing the temperature of lower Containment.

Each ventilation unit contains an automatic on-line tube cleaning system. This system incorporates individual brushes that are periodically backwashed through the tubes to remove fouling and silt deposits. The periodic backwash is accomplished with a 4-way reversing valve (1RV433, 1RV434, 1RV435, and 1RV436) and associated controls. Backwashing occurs based on a predetermined frequency and duration. The cleaning cycle is initiated automatically by a cycle timer.

1 Pt. Given the following conditions on Unit 1:

- In Mode 5 cooling down for a refueling outage.
- The '1B' ND pump tripped due to an electrical fault.
- The '1A' ND pump has been started per AP/1/A/5500/19 (*Loss of ND or ND System* Leakage) Encl. 14 (*Startup of ND Pumps*)
- NC temperature before the pump trip was 150 degrees
- NC temperature has increased to 207 degrees.
- AP/1/A/5500/19 (*Loss of ND*) is in effect

The SRO instructs the RO to cooldown to the pre-event temperature.

Which one (1) of the following describes the maximum cooldown rate and minimum flow rate allowed to cooldown?

REFERENCES PROVIDED

A. **Maximum cooldown rate of 50 degrees/hr and minimum flow rate of 1500 gpm.**

B. **Maximum cooldown rate of 75 degrees/hr and minimum flow rate of 1000 gpm**

C. **Maximum cooldown rate of 50 degrees/hr and minimum flow rate of 2000 gpm.**

D. **Maximum cooldown rate of 75 degrees/hr and minimum flow rate of 1500 gpm.**

***Bank Question:* 1055** ***Answer:* D**

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1 Pt. Given the following conditions on Unit 1:

- In Mode 5 cooling down for a refueling outage.
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- NC temperature has increased to 207 degrees.
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Which one (1) of the following describes the maximum cooldown rate and minimum flow rate allowed to cooldown?

REFERENCES PROVIDED
AP/1/A/5500/19 Encl 14
DATA BOOK Encl. 4.3, curve 1.6b

A. Maximum cooldown rate of 50 degrees/hr and minimum flow rate of 1500 gpm.

B. Maximum cooldown rate of 75 degrees/hr and minimum flow rate of 1000 gpm

C. Maximum cooldown rate of 50 degrees/hr and minimum flow rate of 2000 gpm.

D. Maximum cooldown rate of 75 degrees/hr and minimum flow rate of 1500 gpm.

Distracter Analysis:.

A. Incorrect:
Plausible:
B. Incorrect:
Plausible:
C. Incorrect:
Plausible:
D. Correct
Plausible:

LEVEL: SRO

KA: 000025 AA2.05 (3.1*/3.5*)

SOURCE: NEW

LEVEL OF KNOWLEDGE: Analysis

AUTHOR: CWS

LESSON: OP-MC-AP-19

OBJECTIVES: OP-MC-AP-19 Obj 2

REFERENCES: AP-19 Background Document Enclosure 14
AP/1A/5500019 Enclosure 14 Provided
DATA Book Enclosure 4.3 Provided

SUMMARY FOR ENCLOSURE 14, STARTUP OF ND PUMPS

This enclosure attempts to get a ND Pump started under various potential plant conditions. If a loss of VI has occurred, then numerous compensatory actions are needed and rather than complicate this enclosure, a kickout is provided to a separate enclosure for that plant condition.

A couple of system checks are performed prior to starting an ND train. ND-35 is check closed to prevent an inadvertent inventory loss (may have been opened as a makeup option). If open, an operator is dispatched to stand by so it can be closed prior to pump start.

A step is provided to leave ND L/D in service if the NC System is solid. The setpoint for checking if NC solid is 96% Pzr level, which includes 4% instrument error. Note there still may be some volume above just full indicated level (dome of Pzr), but that amount can't be assumed to be available.

If SI has occurred then control of RN modulating valves is reestablished. Then direction is provided to go the section of the enclosure to start the desired ND Pump.

In preparation for starting a ND Pump, the local pump discharge is setup 2 turns open to prevent water hammer concerns. Pump support conditions are established (RN & KC) and ND-35 is closed at this time, if required. ND suction from the loop is aligned, and ND flow bypassing the ND Hx is aligned. The ND recirc valve is de-energized prior to starting pump to prevent any air that may be in the ND Hx from returning to the pump suction.

Several precautions are taken on ND Pump startup addressing voiding concerns. If air entrainment or voiding has occurred, a cue is provided to continue makeup as required, considering void collapse may occur after pump start. Also, a check is made for subcooling. If subcooling can't be restored, FW-27A is aligned open in conjunction with the loop suction valves until after pump start. Once NC System subcooling restored (should happen quickly with the cool FWST water mixed in), FW-27A is closed.

After the ND Pump is started, ND flow is carefully established using ND Pump discharge valve and ND-34, and then flow through the ND Hx is carefully established to maintain NC temperature (considering NC System cooldown limits). A cue is also provided to secure "feed & bleed" when less than 200°F, if it had been established. Finally, a cue is provided to flush the idle ND train if air entrainment may have occurred on it.

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APE:	**025 Loss of Residual Heat Removal System (RHRS)**		
	<u>ABILITY</u>		
AA1.	**Ability to operate and / or monitor the following as they apply to the Loss of Residual Heat Removal System: (CFR 41.7 / 45.5 / 45.6)**		
AA1.01	RCS/RHRS cooldown rate	3.6	3.7
AA1.02	RCS inventory	3.8	3.9
AA1.03	LPI pumps	3.4	3.3
AA1.04	Closed cooling water pumps	2.8*	2.6
AA1.05	Raw water or sea water pumps	2.7	2.6
AA1.06	Not Used	N/A	N/A
AA1.07	Not Used	N/A	N/A
AA1.08	RHR cooler inlet and outlet temperature indicators	2.9*	2.9
AA1.09	LPI pump switches, ammeter, discharge pressure gauge, flow meter, and indicators	3.2	3.1
AA1.10	LPI pump suction valve and discharge valve indicators	3.1*	2.9
AA1.11	Reactor building sump level indicators	2.9	3.0
AA1.12	RCS temperature indicators	3.6	3.5
AA1.13	SWS radiation monitors	2.5	2.6
AA1 14	Waste tank radiation monitors	2.1*	2.1
AA1.15	Waste tank level gauges and recorders	2.1	2.1
AA1.16	Service water pump manual switch, flow gauge, running lights, and ammeters	2.2	2.2
AA1.17	Service water block valve indicators and flow valve controllers	2.1	2.0*
AA1.18	LPI header cross-connect valve controller and indicators	2.6*	2.8*
AA1.19	Block orifice bypass valve controller and indicators	2.6*	2.4*
AA1.20	HPI pump control switch, indicators, ammeter running lights, and flow meter	2.6*	2.5*
AA1.21	Letdown flow indicator	2.3	2.5
AA1.22	Obtaining of water from BWST for LPI system	2.9*	2.8
AA1.23	RHR heat exchangers	2.8	2.9
AA2.	**Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: (CFR: 43.5 / 45.13)**		
AA2.01	Proper amperage of running LPI/decay heat removal/RHR pump(s)	2.7	2.9
AA2.02	*Leakage of reactor coolant from RHR into closed cooling* water system or into reactor building atmosphere	3.4	3.8
AA2.03	Increasing reactor building sump level	3.6	3.8
AA2.04	.Location and isolability of leaks	3.3*	3.6
AA2.05	Limitations on LPI flow and temperature rates of change	3.1*	3.5*
AA2.06	Existence of proper RHR overpressure protection	3.2*	3.4*
AA2.07	Pump cavitation	3.4	3.7

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
		1.0	1.0	1.0

OBJECTIVES

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	Concerning AP/1(2)/5500/19 (Loss of ND OR ND SYSTEM LEAKAGE): • State the purpose of the AP • Recognize the symptoms that would require implementation of the AP. AP19001			X	X	X
2	Given scenarios describing accident events and plant conditions, evaluate the basis for any caution, note, or step. AP19002			X	X	X

SUMMARY FOR ENCLOSURE 14, STARTUP OF ND PUMPS

This enclosure attempts to get a ND Pump started under various potential plant conditions. If a loss of VI has occurred, then numerous compensatory actions are needed and rather than complicate this enclosure, a kickout is provided to a separate enclosure for that plant condition.

A couple of system checks are performed prior to starting an ND train. ND-35 is check closed to prevent an inadvertent inventory loss (may have been opened as a makeup option). If open, an operator is dispatched to stand by so it can be closed prior to pump start.

A step is provided to leave ND L/D in service if the NC System is solid. The setpoint for checking if NC solid is 96% Pzr level, which includes 4% instrument error. Note there still may be some volume above just full indicated level (dome of Pzr), but that amount can't be assumed to be available.

If SI has occurred then control of RN modulating valves is reestablished. Then direction is provided to go the section of the enclosure to start the desired ND Pump.

In preparation for starting a ND Pump, the local pump discharge is setup 2 turns open to prevent water hammer concerns. Pump support conditions are established (RN & KC) and ND-35 is closed at this time, if required. ND suction from the loop is aligned, and ND flow bypassing the ND Hx is aligned. The ND recirc valve is de-energized prior to starting pump to prevent any air that may be in the ND Hx from returning to the pump suction.

Several precautions are taken on ND Pump startup addressing voiding concerns. If air entrainment or voiding has occurred, a cue is provided to continue makeup as required, considering void collapse may occur after pump start. Also, a check is made for subcooling. If subcooling can't be restored, FW-27A is aligned open in conjunction with the loop suction valves until after pump start. Once NC System subcooling restored (should happen quickly with the cool FWST water mixed in), FW-27A is closed.

After the ND Pump is started, ND flow is carefully established using ND Pump discharge valve and ND-34, and then flow through the ND Hx is carefully established to maintain NC temperature (considering NC System cooldown limits). A cue is also provided to secure "feed & bleed" when less than 200°F, if it had been established. Finally, a cue is provided to flush the idle ND train if air entrainment may have occurred on it.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 36. **Slowly throttle close 1ND-34 (A & B ND Hx Bypass) until a drop in ND flow is observed.**	
__ 37. **Have operators open, backseat, and lock 1ND-24 (A ND Pump Discharge Isol).**	
38. **Throttle the following as necessary to maintain stable NC temperature:** __ • 1ND-29 (A ND Hx Outlet) __ • 1ND-34 (A & B ND Hx Bypass).	
__ 39. **Check** NC temperature based on **core exit T/C's - LESS THAN 200° F.**	__ **<u>GO</u> <u>TO</u> Step 41.**
__ 40. **Reduce KC flow to 1A ND HX as required to control NC temperature.**	
__ 41. **<u>IF</u> <u>AT</u> <u>ANY</u> <u>TIME</u> cooldown is required, <u>THEN</u> <u>REFER</u> <u>TO</u> Unit 1 Data Book Curve 1.6 b (Heatup and Cooldown Limits for LTOP).**	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

42. **Check feed and bleed cooling - INITIATED.** — **GO TO Step 43.**

__ a. Initiate cooldown to 200° F on core exit T/C's.

b. WHEN NC temperature is less than 200° F on core exit T/C's, THEN perform the following:

CAUTION **Failure to stop makeup to NC System prior to closing Pzr PORVs may cause low temperature over pressure concern.**

__ 1) Stop or reduce makeup to NC System.

2) IF PORV (s) not required open as vent path, AND makeup to NC System stopped, THEN:

__ a) Close PZR PORVs.

__ b) Place closed Pzr PORVs in "AUTO".

__ 3) Control ND flow to maintain NC System subcooled.

__ 43. **Ensure ND flow greater than 1500 GPM.**

__ 44. **Dispatch operator to reclose breaker 1EMXA - F12B (1A ND Pump & Hx Miniflow Isol Motor (1ND-68A)).**

__ 45. **Ensure 1ND-68A (A ND Pump & A Hx Miniflow) remains closed.**

UNIT 1

OP/1/A/6100/22
Enclosure 4.3
Curve 1.6 b

Heatup and Cooldown Limits for LTOP

Valid Thru Cycle 16 and with a Maximum of One NI or One NV Pump Capable of Injection

Heatup and cooldown instantaneous rate (P1246 - P1249) and rate of change over the last hour of operation should be reviewed for compliance **prior to changing temperature ranges**, and at least every 30 minutes during heatup and cooldown.

Table 1: With All NCPs OFF (Note 1)

Temperature Range (°F)	Indicating Temperature	Cooldown Rate (°F/hr)	Heatup Rate (°F/hr)
less than 89	Lowest of: • ND Hx Outlet Temp • Lowest WR T-cold	> 2.75 sq. inch vent	
89 - 119		40	50
119 - 149		60	50
> 149		75	50

Note 1: Minimum temperature to operate pressurized is 89°F. If indicating temperatures fall below 89°F, restore to > 89°F within 15 minutes or immediately depressurize by adjustment of charging and letdown and open 1 PORV.

Table 2: With One or Two NCPs Running (Note 2)

Temperature Range (°F)	Indicating Temperature	Cooldown Rate (°F/hr)	Heatup Rate (°F/hr)
89 - 119	Lowest of: • ND Hx Outlet Temp • Lowest WR T-cold	40	50
119 - 149		60	50
> 149		75	50

Note 2: Minimum temperature required to operate RCPs is 89°F. If indicating temperatures fall below 89°F, restore to > 89°F within 15 minutes or immediately depressurize by adjustment of charging and letdown and open 1 PORV.

Table 3: With Three or Four NCPs Running (Note 3)

Temperature Range (°F)	Indicating Temperature	Cooldown Rate (°F/hr)	Heatup Rate (°F/hr)
91 - 114	• Lowest WR T-cold	20	50
114 - 139		40	50
139 - 164		60	50
> 164		75	50

Note 3: Minimum temperature required to operate 3rd RCP is 91°F. Minimum temperature required to operate 4th RCP is 140°F. If lowest WR Tcold falls below these temperature limits, stop at least one pump and comply with next lower tier of requirements.

UNIT 1

UNIT 1

OP/1/A/6100/22
Enclosure 4.3
Curve 1.6 b

Heatup and Cooldown Limits for LTOP

Valid Thru Cycle 16 and with a Maximum of One NI or One NV Pump Capable of Injection

Heatup and cooldown instantaneous rate (P1246 - P1249) and rate of change over the last hour of operation should be reviewed for compliance **prior to changing temperature ranges,** and at least every 30 minutes during heatup and cooldown.

Table 1: With All NCPs OFF (Note 1)

Temperature Range (°F)	Indicating Temperature	Cooldown Rate (°F/hr)	Heatup Rate (°F/hr)
less than 89	Lowest of:	> 2.75 sq. inch vent	
89 - 119	• ND Hx Outlet Temp	40	50
119 - 149	• Lowest WR T-cold	60	50
> 149		75	50

Note 1: Minimum temperature to operate pressurized is 89°F. If indicating temperatures fall below 89°F, restore to > 89°F within 15 minutes or immediately depressurize by adjustment of charging and letdown and open 1 PORV.

Table 2: With One or Two NCPs Running (Note 2)

Temperature Range (°F)	Indicating Temperature	Cooldown Rate (°F/hr)	Heatup Rate (°F/hr)
89 - 119	Lowest of:	40	50
119 - 149	• ND Hx Outlet Temp	60	50
> 149	• Lowest WR T-cold	75	50

Note 2: Minimum temperature required to operate RCPs is 89°F. If indicating temperatures fall below 89°F, restore to > 89°F within 15 minutes or immediately depressurize by adjustment of charging and letdown and open 1 PORV.

Table 3: With Three or Four NCPs Running (Note 3)

Temperature Range (°F)	Indicating Temperature	Cooldown Rate (°F/hr)	Heatup Rate (°F/hr)
91 - 114		20	50
114 - 139	• Lowest WR T-cold	40	50
139 - 164		60	50
> 164		75	50

Note 3: Minimum temperature required to operate 3rd RCP is 91°F. Minimum temperature required to operate 4th RCP is 140°F. If lowest WR Tcold falls below these temperature limits, stop at least one pump and comply with next lower tier of requirements.

UNIT 1

Bank Question: 1057 ***Answer: C***

1 Pt. As a result of thunderstorms Unit 2 has experienced a Loss of Offsite Power and Reactor trip. E-0 (*Reactor Trip or Safety Injection*) was implemented and the crew has transitioned to ES-0.1 (*Reactor Trip Response*).

The SRO asks the RO to check NC temperatures.

Which one (1) of the following would the RO use to describe the response of the NC system?

A. NC Tave STABLE or trending to 557 degrees

B. NC T hots STABLE or trending to 553 degrees

C. NC T colds STABLE or trending to 557 degrees

D. NC Tave STABLE or trending to 553 degrees.

Distracter Analysis:. The reactor coolant pumps have tripped in this scenario. Tave is only checked if the NC pumps are on. T hot should be increasing initially on the establishment of natural circulation. T colds will go to 557. 553 degrees is a commonly used number for steam dump P-12.

A. Incorrect:
Plausible:
B. Incorrect:
Plausible:
C. Correct:
Plausible:
D. Incorrect
Plausible:

LEVEL: SRO

KA: 00056 AA2.32 (4.3/4.3)

SOURCE: NEW

LEVEL OF KNOWLEDGE: Comprehension

AUTHOR: CWS

LESSON: AP/09 Background Document

OBJECTIVES:

REFERENCES: AP/09 Background Document pages 3 & 4
EP/1/A/5000/ES-0.1 page 3

1 Pt. As a result of thunderstorms Unit 2 has experienced a Loss of Offsite Power and Reactor trip. E-0 (*Reactor Trip or Safety Injection*) was implemented and the crew has transitioned to ES-0.1 (*Reactor Trip Response*).

The SRO asks the RO to check NC temperatures.

Which one (1) of the following would the RO use to describe the response of the NC system?

A. **NC Tave STABLE or trending to 557 degrees**

B. **NC T hots STABLE or trending to 553 degrees**

C. **NC T colds STABLE or trending to 557 degrees**

D. **NC Tave STABLE or trending to 553 degrees.**

APE: **056 Loss of Offsite Power**

AA2.30	Switch gear room cooling unit run indicator	2.0	2.2
AA2.31	Ventilation supply fan and run indicators for the service water building, control room and battery room	2.1	2.2
[redacted]			
AA2.33	ESF channels, A and B breaker-trip alarms, indicators and bus voltage indicators	3.6?	3.7?
AA2.34	Rod bottom lights	4.1	4.2
AA2.35	Reactor trip alarm	4.1	4.1
AA2.36	Turbine stop valve indicator	3.9	4.1
AA2.37	ED/G indicators for the following: voltage, frequency, load, load-status, and closure of bus tie breakers	3.7*	3.8
AA2.38	Load sequencer status lights	3.7*	3.8
AA2.39	Safety injection pump ammeter and flowmeter	3.5*	3.6
AA2.40	Service water pump ammeter and flowmeter	3.3	3.4
AA2.41	HVAC chill water pump run and alarm indicators	2.3*	2.3*
AA2.42	Occurrence of a reactor trip	4.1	4.1
AA2.43	Occurrence of a turbine trip	3.9	4.1
AA2.44	Indications of loss of offsite power	4.3	4.5
AA2.45	Indicators to assess status of ESF breakers (tripped/not-tripped) and validity of alarms (false/not-false)	3.6*	3.9
AA2.46	That the ED/Gs have started automatically and that the bus tie breakers are closed	4.2	4.4
AA2.47	Proper operation of the ED/G load sequencer	3.8	3.9
AA2.48	Reactor coolant temperature, pressure, and PZR level following a power outage transient	4.3	4.4
AA2.49	Nonessential equipment to be secured to avoid overload of ED/Gs	3.0	3.4
AA2.50	That load and VAR limits, alarm setpoints, frequency and voltage limits for ED/Gs are not being exceeded	2.8*	3.1
AA2.51	_T, (core, heat exchanger, etc.)	3.3*	3.4*
AA2.52	PZR level required for a given power level	2.6*	2.8*
AA2.53	Status of emergency bus under voltage relays	2.9	3.2
AA2.54	Breaker position (remote and local)	2.9	3.0
AA2.55	Subcooled margin monitors	3.8	3.9
AA2.56	RCS T-ave	3.6*	3.7
AA2.57	RCS hot-leg and cold-leg temperatures	3.9	4.1
AA2.58	Air compressors (indicating lights)	2.3	2.6*
AA2.59	Gland seal pressure gauge	1.5	1.6
AA2.60	MSIV open	2.7*	2.9*
AA2.61	Condensate pump	1.6	1.7
AA2.62	Breaker for feedwater pumps	1.7	1.9*
AA2.63	Feedwater heater drain pump breaker trip	1.5	1.5
AA2.64	Circulating water pump switch	1.6	1.7
AA2.65	Screen wash pump	1.5	1.7
AA2.66	CVCS charging flow	3.2	3.4
AA2.67	Seal injection flow (for the RCPs)	2.9	3.1
AA2.68	CVCS letdown flow	2.7	2.9
AA2.69	Valve position	2.3*	2.5*
AA2.70	Reactor building CCW temperature	2.1	2.2
AA2.71	Turbine service water heat exchange	1.7	1.7
AA2.72	Auxiliary feed flow	4.1	4.3
AA2.73	PZR heater on/off	3.5	3.6
AA2.74	PORV position	3.6	3.7
AA2.75	CVCS makeup	3.0	3.2
AA2.76	Reactor makeup water pump (running)	2.6	2.6
AA2.77	Auxiliary feed pump (running)	4.1	4.4
AA2.78	Bus voltmeters	2.7	3.0

bOW

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
	·	0.5	0.5	0.5

OBJECTIVES

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	Concerning AP/1(2)/5500/09 (Natural Circulation): • State the purpose of the AP • Recognize the symptoms that would require implementation of the AP. AP09001			X	X	X
2	Given scenarios describing accident events and plant conditions, evaluate the basis for any caution, note, or step. AP09002			X	X	X

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

5. **Check NC temperatures:**

__ • IF any NC pump on, THEN check NC T-Ave - STABLE OR TRENDING TO 557° F.

OR

__ • IF all NC pumps off, THEN check NC T-Colds - STABLE OR TRENDING TO 557° F.

Perform the following based on plant conditions:

a. IF temperature less than 557° F AND going down, THEN:

__ 1) Ensure all steam dump valves closed.

2) IF MSR "RESET" light is dark, THEN:

__ a) Depress "SYSTEM MANUAL".

__ b) Depress "RESET".

__ 3) Ensure all SM PORVs closed.

4) IF any SM PORV can not be closed, THEN:

__ a) Close its isolation valve.

__ b) IF SM PORV isolation valve can not be closed, THEN dispatch operator to close SM PORV isolation valve.

__ 5) Ensure S/G blowdown is isolated.

6) IF cooldown continues, THEN control feed flow as follows:

a) IF S/G N/R level is less than 11% in all S/Gs, THEN throttle feed flow to achieve the following:

__ • Minimize cooldown
__ • Maintain total feed flow greater than 450 GPM.

b) WHEN N/R level is greater than 11% in at least one S/G, THEN throttle feed flow further to:

__ • Minimize cooldown
__ • Maintain at least one S/G N/R level greater than 11%.

(RNO continued on next page)

STEP DESCRIPTION FOR AP/09

STEP 1:

PURPOSE:

Ensure adequate secondary heat sink.

DISCUSSION:

One of the design requirements for natural circulation flow in the NC System is to have a heat sink.

To ensure the S/Gs are maintained as a heat sink, feed flow to ALL the S/Gs is ensured. This can be from the CM/CF system or the CA System. If feed flow is not present, direction is given to use CA, since this system can typically be established much quicker than CM/CF.

S/G NR level greater than 11% is used to indicate the water level just in the narrow range (MCC-1552.08-00-0208, EP setpoint file). Water level in the NR ensures the tube bundles are covered. If NR indication is not met, direction is given to maintain feed flow greater than 450 GPM until greater than 11%. The setpoint of 450 GPM ensures decay heat removal via steam release and a net inventory gain in the S/Gs. Note: 450 GPM is conservative since this amount of flow covers the maximum amount needed decay heat removal and NC pump heat, and of course there is no NC pump heat during natural circulation.

REFERENCES:
MCC-1552.08-00-0208, EP setpoint file

STEP 2:

PURPOSE:

Provide the parameter values necessary to indicate natural circulation is occurring. If it's not, direction is given to increase dumping steam in an attempt to establish it.

DISCUSSION:

The parameters used to verify natural circulation are:

NC Subcooling > 0-F
Although two-phase and reflux boiling are also forms of natural circulation, the preferred form is subcooled natural circulation. It is the most efficient and results in the lowest core temperatures given everything else equal.

S/G press – stable or going down
Natural circulation cooling is established when the heat generation rate in the core equals the heat transfer rate from the core to the NC System, to the S/Gs, and out of the S/Gs. If S/G pressures are INCREASING, then the heat transfer rate out of the S/Gs is not sufficient, and core temperatures will be increasing.

NC T-Hots – Stable or Going Down
When forced NC flow is lost, it takes a few minutes (5 – 10 minutes) for natural circulation flow to set up. During this time, T-Hots are increasing, as expected. T-Colds are relatively stable since they are tied to S/G pressure, and the assumption is S/G pressure is stable unless a cooldown is taking place. Once the delta-T develops (T-Hot increase) sufficiently for driving head for natural circulation flow, T-Hot should no longer be increasing. From this point forward, T-Hots should be going down as decay heat drops off, or be stable as the decay heat curve levels off.

After the initial time period for natural circulation to develop, if T-Hots are going up this means the heat removal from natural circulation is less than the decay heat generation rate, or in another words, inadequate nat. circ for whatever reason.

Core exit T/Cs – Stable or Going Down
When forced NC flow is lost, it takes a few minutes (5 – 10 minutes) for natural circulation flow to set up. During this time, Core Exit T/Cs are increasing, as expected. T-Colds are relatively stable since they are tied to S/G pressure, and the assumption is S/G pressure is stable unless a cooldown is taking place. Once the delta-T develops (Core Exit T/Cs increase) sufficiently for driving head for natural circulation flow, Core Exit T/Cs should no longer be increasing. From this point forward, Core Exit T/Cs should be going down as decay heat drops off, or be stable as the decay heat curve levels off.

After the initial time period for natural circulation to develop, if Core Exit T/Cs are going up this means the heat removal from natural circulation is less than the decay heat generation rate, or in another words, inadequate nat. circ for whatever reason.

NC T-Colds – At Saturation Temperature For S/G Pressure
The following is an excerpt from the ERGs Generic Issues section concerning T-Colds: "The cold leg temperature readings can be used as additional verification that heat removal through the steam generators is occurring. The loop T-Cold readings in active loops are quite sensitive to changes in heat transfer rates from the reactor to the secondary sides of the steam generators. Actual test have shown that loop T-Colds follow almost exactly the steam generator pressure with minimal time lag".

In another words, if T-Colds are not following steam generator pressure, it is likely heat removal from the S/Gs is not occurring. If this is the case, the core is not being adequately cooled. To facilitate determining whether NC T-Colds are at saturation temperature for S/G pressures, a graph is provided to correlate the two, with a band provided to allow for instrument inaccuracies.

1 Pt. Radwaste is in the process of releasing WGDT 'A'. 1EMF -36 L is inoperable due to PM. Trip 2 is received on 0EMF-50 (*Waste Gas Discharge*). The gaseous waste release is secured as a result of 1WG-160 closing. Radwaste calls the control room SRO and reports 0EMF-50 has been purged and is ready to reinitiate the release.

Which one (1) of the following describes the actions of the control room SRO?

A. **The SRO can authorize up to two (2) restarts without re-sampling.**

B. **The SRO has Radwaste terminate existing GWR paperwork, and generate new paperwork.**

C. **The SRO can authorize one (1) restart without re-sampling.**

D. **The SRO can authorize Radwaste to jumper control actions of 0EMF-50, restart release and take grab samples once per four (4) hours during release.**

Bank Question: 1058 **Answer: B**

1 Pt. Radwaste is in the process of releasing WGDT 'A'. 1EMF -36 L is inoperable due to PM. Trip 2 is received on 0EMF-50 (*Waste Gas Discharge*). The gaseous waste release is secured as a result of 1WG-160 closing. Radwaste calls the control room SRO and reports 0EMF-50 has been purged and is ready to reinitiate the release.

Which one (1) of the following describes the actions of the control room SRO?

A. **The SRO can authorize up to two (2) restarts without re-sampling.**

B. **The SRO has Radwaste terminate existing GWR paperwork, and generate new paperwork.**

C. **The SRO can authorize one (1) restart without re-sampling.**

D. **The SRO can authorize Radwaste to jumper control actions of 0EMF-50, restart release and take grab samples once per four (4) hours during release.**

Distracter Analysis:.

A. **Incorrect**: Would be correct if both EMF 36 and 50 were operable
Plausible:

B. **Correct**:
Plausible:

C. **Incorrect**:
Plausible:

D. **Incorrect**
Plausible:

LEVEL: SRO

KA: 00060 G 2.3.8 (2.3/3.2)

SOURCE: NEW

LEVEL OF KNOWLEDGE: Comprehension

AUTHOR: CWS

LESSON: OP-MC-WE-RGR

OBJECTIVES: OP-MC-WE-RGR Obj 5

REFERENCES: OP-MC-WE-RGR page 13

2.3 Radiation Control

2.3.1 **Knowledge of 10 CFR: 20 and related facility radiation control requirements.**
(CFR: 41.12 / 43.4. 45.9 / 45.10)
IMPORTANCE RO 2.6 SRO 3.0

2.3.2 **Knowledge of facility ALARA program.**
(CFR: 41.12 / 43.4 / 45.9 / 45.10)
IMPORTANCE RO 2.5 SRO 2.9

2.3.3 **Knowledge of SRO responsibilities for auxiliary systems that are outside the control room (e.g., waste disposal and handling systems).**
(CFR: 43.4 / 45.10)
IMPORTANCE RO 1.8 SRO 2.9

2.3.4 **Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.**
(CFR: 43.4 / 45.10)
IMPORTANCE RO 2.5 SRO 3.1

2.3.5 **Knowledge of use and function of personnel monitoring equipment.**
(CFR: 41.11 / 45.9)
IMPORTANCE RO 2.3 SRO 2.5

2.3.6 **Knowledge of the requirements for reviewing and approving release permits.**
(CFR: 43.4 / 45.10)
IMPORTANCE RO 2.1 SRO 3.1

2.3.7 **Knowledge of the process for preparing a radiation work permit.**
(CFR: 41.10 / 45.12)
IMPORTANCE RO 2.0 SRO 3.3

2.3.8 **Knowledge of the process for performing a planned gaseous radioactive release.**
(CFR: 43.4 / 45.10)
IMPORTANCE RO 2.3 SRO 3.2

2.3.9 **Knowledge of the process for performing a containment purge.**
(CFR: 43.4 / 45.10)
IMPORTANCE RO 2.5 SRO 3.4

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
N/A	N/A	2.0	2.0	2.0

OBJECTIVES

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	State the purpose of the Radiological Gaseous Releases.			X	X	X
2	Given a completed GWR, state the recommended release rate.			X	X	X
3	State what responsibility the Control Room SRO is accepting when he signs to authorize a release.			X	X	X
4	Given a completed GWR, state the proper EMF to be used for the release.			X	X	X
5	Evaluate plant parameters to determine any abnormal system conditions that may exist			X	X	X
6	Concerning the Selected Licensee Commitments (SLC) related to Gaseous Waste Releases;					
	• Given the SLC Manual, discuss any commitments and their applicability.			X	X	X
	• For any commitments that have action required within one hour, state the action.			X	X	X
	• Given a set of parameter values or system conditions, determine if any commitment is (are) not met and any action(s) required within one hour.			X	X	X
	• Given the SLC Manual, discuss the basis for a given commitment.				X	*
	* **SRO only**					

3.0 GASEOUS RELEASES

The three types of releases discussed in the section are:

- Waste Gas Decay Tank (WG)
- Containment Air Release (VQ)
- Containment Purge (VP)

3.1. Waste Gas Decay Tank Release

3.1.1. Limits and Precautions

If a WGDT will be released, then ensure that bank is not in-service. If an in-service tank must be removed from service in order for a release to be made, then the change must be made per OP/0/A/6200/18, Waste Gas Operation.

Neither Shutdown (S/D) Tank can be in service when making a release, due to the release flowpath which must be used.

For the release of a S/D Tank, it must first be transferred to a WGDT (A, B, C, D, E, F) per OP/0/A/6200/18 (Waste Gas Operation). The release can then be made following this procedure.

The Unit 1 Auxiliary Building Ventilation System or Unit 1 Fuel Building Ventilation System should be in service during a release to ensure that the gas leaves the unit vent completely.

No release will be made without proper verification of flow rate. 0WGLP6140 (WG Disch Flow Loop) is the normal instrument for verifying flow. If inoperable, flow can be monitored using the decay tank pressure (for the tank being released) vs. release time.

Bulk hydrogen or nitrogen **cannot** be added to the waste gas system while releasing a tank.

Based on mutual agreement between MNS Radiation Protection Manager, General Office Protection Manager, and MNS Radwaste, if EMF-36 and EMF-50 are operable, then releases interrupted by EMF Hi Rad Discharge Trips may be reinitiated up to a maximum of two times (total of 3 attempts) without resampling before terminating release procedure. If EMF-36 or EMF-50 are inoperable only one release attempt shall be made.

If EMF-36 and EMF-50 are inoperable, RP management approval is required to make a WG release.

Immediately after each release purge EMF-50. Immediately subsequent to purge, reposition and lock appropriate valves per applicable enclosure.

1 Pt. Given the following conditions on Unit 1:

- SGTR in the '1A' S/G
- E-0 (*Reactor Trip or Safety Injection*) complete
- E-3 (*Steam Generator Tube Rupture*) implemented.
- Cooldown is secured due to operator exceeding Main Steam Isolation set point.

Which one (1) of the following describes how the operator continues to cooldown?

A. **Go to Bypass Interlock on steam dumps and continue cooldown with steam dumps.**

B. **Reset Main Steam Isolation, open MSIVs and continue cooldown with steam dumps.**

C. **Reset Main Steam Isolation, and PORVs and continue cooldown using PORVs in manual.**

D. **Reset Main Steam Isolation, and PORVs and continue cooldown using PORVs in automatic.**

Bank Question: 1060 ***Answer: C***

1 Pt. Given the following conditions on Unit 1:

- SGTR in the '1A' S/G
- E-0 (*Reactor Trip or Safety Injection*) complete
- E-3 (*Steam Generator Tube Rupture*) implemented.
- Cooldown is secured due to operator exceeding Main Steam Isolation set point.

Which one (1) of the following describes how the operator continues to cooldown?

A. Go to Bypass Interlock on steam dumps and continue cooldown with steam dumps.

B. Reset Main Steam Isolation, open MSIVs and continue cooldown with steam dumps.

C. Reset Main Steam Isolation, and PORVs and continue cooldown using PORVs in manual.

D. Reset Main Steam Isolation, and PORVs and continue cooldown using PORVs in automatic.

Distracter Analysis:.

A. Incorrect:
Plausible:
B. Incorrect:
Plausible:
C. Correct:
Plausible:
D. Incorrect
Plausible:

LEVEL: SRO

KA: 0041 G2.4.20 (3.3/4.0)

SOURCE: NEW

LEVEL OF KNOWLEDGE: Memory

AUTHOR: *CWS*

LESSON: OP-MC-EP-E3

OBJECTIVES: OP-MC-EP-E3 Obj 4

REFERENCES: OP-MC-EP-E3 pages 75,77,79
EP/1A/5500/E-3 page 19-21

2.4 Emergency Procedures /Plan (Continued)

2.4.18 Knowledge of the specific bases for EOPs.
(CFR: 41.10 / 45.13)
IMPORTANCE RO 2.7 SRO 3.6

2.4.19 Knowledge of EOP layout, symbols, and icons.
(CFR: 41.10 / 45.13)
IMPORTANCE RO 2.7 SRO 3.7

2.4.20 Knowledge of operational implications of EOP warnings, cautions, and notes.
(CFR: 41.10 / 45.13)
IMPORTANCE RO 3.3 SRO 4.0

2.4.21 Knowledge of the parameters and logic used to assess the status of safety functions including:
1. Reactivity control
2. Core cooling and heat removal
3. Reactor coolant system integrity
4. Containment conditions
5. Radioactivity release control.

(CFR: 43.5 / 45.12)
IMPORTANCE RO 3.7 SRO 4.3

2.4.22 Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.
(CFR: 43.5 / 45.12)
IMPORTANCE RO 3.0 SRO 4.0

2.4.23 Knowledge of the bases for prioritizing emergency procedure implementation during emergency operations.
(CFR: 41.10 / 45.13)
IMPORTANCE RO 2.8 SRO 3.8

2.4.24 Knowledge of loss of cooling water procedures.
(CFR: 41.10 / 45.13)
IMPORTANCE RO 3.3 SRO 3.7

2.4.25 Knowledge of fire protection procedures.
(CFR: 41.10 / 45.13)
IMPORTANCE RO 2.9 SRO 3.4

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
		4.0	4.0	3.0

OBJECTIVES

SEQ	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	Explain the purpose for each procedure in the E-3 series. EPE3001			X	X	
2	Discuss the entry and exit guidance for each procedure in the E-3 series. EPE3002			X	X	
3	Discuss the mitigating strategy (major actions) of each procedure in the E-3 series. EPE3003			X	X	X
4	Discuss the basis for any note, caution or step for each procedure in the E-3 series. EPE3004			X	X	X
5	Given the Foldout page, discuss the actions included and the basis for these actions. EPE3005			X	X	X
6	Given the appropriate procedure, evaluate a given scenario describing accident events and plant conditions to determine any required action and its basis. EPE3006			X	X	X
7	Discuss the time critical task(s) associated with the E-3 series procedures including the time requirements and the basis for these requirements. EPE3007			X	X	X

STEP 16 WHEN "P-11 PRESSURIZER S/I BLOCK PERMISSIVE" status light (1SI-18) lit, THEN perform the following:

PURPOSE: To prevent MSIV closure on low steamline pressure during controlled NC cooldown.

BASIS: An automatic protection feature is provided to close the MSIVs when steam pressure approaches the Low Pressure Steamline Isolation setpoint. In Step 17, the operator is instructed to dump steam from the intact S/Gs, which is expected to reduce their pressure below this setpoint. If automatic isolation occurred, steam flow to the condenser would be terminated requiring the operator to continue the cooldown by dumping steam to the atmosphere. In addition to delaying recovery, this would raise the radiological releases and reduce feedwater supply.

NOTE 1 NC pump trip criteria on subcooling does not apply after starting a controlled cooldown.

NOTE 2 After the Lo Pressure Steamline Isolation signal is blocked, maintaining steam pressure negative rate less than 2 psig per second will prevent Main Steam Isolation.

PURPOSE: To alert the operator of the following.

1. This particular NC pump trip criteria does not apply after starting a controlled cooldown. All other trip criteria remain in effect.
2. The potential for inadvertent steamline isolation during the subsequent S/G depressurization.

BASIS: The NC pump trip criteria is based on subcooled conditions not applicable during the controlled cooldown.

An automatic protection feature is provided to close the MSIVs when the steam pressure rate signal is exceeded. In the following step, the operator is instructed to dump steam from the intact S/Gs, which may result in exceeding the rate setpoint.

STEP 17 Initiate NC System cooldown:

PURPOSE: To establish sufficient subcooling in the NC so the primary system will remain subcooled after pressure is decreased to stop primary-to-secondary leakage.

BASIS: The principal goal of E-3 is to stop primary-to-secondary leakage and to establish and maintain sufficient indications of adequate coolant inventory. These indications include the following:

1. A Pzr level indication used to trend coolant inventory.
2. NC subcooling used to ensure the indicated Pzr level is reliable.

This step is designed to establish sufficient subcooling in the NC so the primary system will remain subcooled after NC pressure is lowered in subsequent steps to stop primary-to-secondary leakage.

The pressure of the intact S/Gs must be maintained less than the pressure of the ruptured S/Gs in order to maintain NC subcooling. Since flow from the ruptured S/G should be isolated, this pressure differential is established by dumping steam only from the intact S/Gs. Steam dump to the condenser is preferred to minimize radiological releases and conserve feedwater supply. However, the PORVs on the intact S/Gs provide an alternative steam release path.

If no intact S/G is available, then an exit transition is provided to ECA-3.1, SGTR With Loss Of Reactor Coolant - Subcooled Recovery Desired.

NC cooldown should proceed as quickly as possible and should not be limited by the 100°F/hr Technical Specification limit. Integrity limits should not be exceeded since the final temperature will remain above 350°F.

The table provides for 20°F subcooling for each pressure range. Core exit T/Cs are used because they provide input for S/I termination and reinitiation. The 20°F subcooling is provided as operating margin to accommodate fluctuation in NC temperature, perturbations in ruptured S/G pressure, interpolation between listed ruptured S/G pressures, and overshoot during NC depressurization.

STEP 17 (CONTINUED)

The preferred cooldown method is to dump steam to the condenser at max rate while attempting to avoid main steam isolation. If the condenser is not available or steam dump to the condenser is not possible, the SG PORVs should be used per the RNO for step 17e. Any delay in initiating cooldown can lead to ruptured S/G overfill. If main steam isolation occurs, while using S/G PORVs, it will have little impact on dumping of steam. The PORVs can be quickly reset and cooldown reinitiated. Although depressurizing at rate specified (2 psig/sec) is very close to rate that would give isolation, it is justified to push rate as hard as possible (when using S/G PORVs). Ensuring steamline Isolation is blocked using NC PORV will prevent any delay in dumping steam with S/G PORVs.

If any intact SG PORV cannot be opened from the Control Room, local operation at the valve is directed. The PORVs should be opened fully using the valve handwheel not the manual loader in the doghouse.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE
- NC pump trip criteria based on subcooling does not apply after starting a controlled cooldown.
- After the Low Pressure Steamline Isolation signal is blocked, maintaining steam pressure negative rate less than 2 PSIG per second will prevent a Main Steam Isolation.

17. **Initiate NC System cooldown as follows:**

__ a. Determine required core exit temperature based on lowest ruptured S/G pressure:

LOWEST RUPTURED S/G PRESSURE (PSIG)	CORE EXIT T/Cs (°F)
GREATER THAN 1099	520 (519 ACC)
1000 - 1099	508 (507 ACC)
900 - 999	494 (493 ACC)
800 - 899	480 (479 ACC)
700 - 799	463 (462 ACC)
600 - 699	444 (444 ACC)
500 - 599	423 (422 ACC)
400 - 499	396 (395 ACC)
300 - 399	362 (361 ACC)
280 - 299	353 (353 ACC)

b. Check condenser available:

__ • "C-9 COND AVAILABLE FOR STEAM DUMP" status light (1SI-18) - LIT

__ • MSIV on intact S/G(s) - OPEN.

__ b. **GO TO** RNO for Step 17.e.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

17. (Continued)

c. Perform the following to place steam dumps in steam pressure mode:

__ 1) Place "STM PRESS CONTROLLER" in manual.

__ 2) Adjust "STM PRESS CONTROLLER" output to equal "STEAM DUMP DEMAND" signal.

__ 3) Place "STEAM DUMP SELECT" in steam pressure mode.

__ d. **WHEN** "P-12 LO-LO TAVG" status light (1SI-18) lit, **THEN** place steam dumps in bypass interlock.

__ e. Dump steam from intact S/G(s) to condenser at maximum rate while attempting to avoid a Main Steam Isolation.

RNO:

e. Perform the following:

__ 1) **IF** Pzr pressure is greater than 1955 PSIG, **THEN** depressurize to 1900 PSIG using Pzr PORV.

__ 2) Depress "BLOCK" on Low Pressure Steamline Isolation block switches.

__ 3) Maintain NC pressure less than 1955 PSIG.

__ 4) Ensure Main Steam Isolation reset.

__ 5) Ensure S/G PORVs reset.

6) **IF** any intact SG PORV isolation valve is closed, **AND** associated PORV is operable, **THEN** perform the following:

__ a) Open S/G PORV isolation valve(s).

__ b) **IF** isolation valve will not open, **THEN** dispatch operator to open isolation valve.

(RNO continued on next page)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

17. (Continued)

7) Dump steam using all intact S/G(s) PORVs at maximum rate as follows:

__ a) Close S/G PORV manual loader on ruptured S/G(s).

__ b) Place intact S/G PORV manual loaders at 50%.

__ c) Select "MANUAL" on "SM PORV MODE SELECT".

__ d) Adjust manual loaders on intact S/G PORVs as required to control intact S/G depressurization rate at approximately 2 PSIG per second.

8) IF any intact S/G PORV closed, THEN dump steam as follows, at maximum rate:

a) Dispatch operators to:

- __ Immediately fully open intact S/G(s) PORVs (at valves).
- __ Establish communication with control room.

__ b) Monitor pressures in all S/G(s) to ensure the correct S/G PORVs are locally operated.

c) IF any intact S/G PORV is unavailable, THEN evaluate using the following to dump steam:

- __ Run TD CA pump.
- __ Use steam drains PER EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 19 (S/G Depressurization Using Steam Drains).

(RNO continued on next page)

1 Pt. Given the following:

- Both Units operating at 100% power.
- 'A' train RN is operating on both units.
- Operations Test Group is performing 'B' train RN valve stroke timing.
- SRO is instructed to evaluate the consequences of stroking 0RN-284B (*Train 1B and 2B Discharge to RC*)

Which one of the following describes the consequences of allowing the technician to test this valve?

A. No consequences due to 'A' Train RN running on both units.

B. Closing 0RN-284B will isolate the RN non-essential header return from Unit 2.

C. Closing 0RN-284B will isolate the RN non-essential header return from Unit 1.

D. Closing 0RN-284B will isolate RV pump discharge.

Bank Question: 1065 ***Answer: B***

1 Pt. Given the following:

- Both Units operating at 100% power.
- 'A' train RN is operating on both units.
- Operations Test Group is performing 'B' train RN valve stroke timing.
- SRO is instructed to evaluate the consequences of stroking 0RN-284B (*Train 1B and 2B Discharge to RC*)

Which one of the following describes the consequences of allowing the technician to test this valve?

A. **No consequences due to 'A' Train RN running on both units.**

B. ***Closing 0RN-284B will isolate the RN non-essential header* return from Unit 2.**

C. **Closing 0RN-284B will isolate the RN non-essential header return from Unit 1.**

D. **Closing 0RN-284B will isolate RV pump discharge.**

Distracter Analysis:.

A. **Incorrect:**
Plausible:
B. **Correct:**
Plausible:
C. **Incorrect:**
Plausible:
D. **Incorrect**
Plausible:

LEVEL: SRO

KA: G 2.2.3 (3.1/3.3)

SOURCE: NEW

LEVEL OF KNOWLEDGE: Comprehension

AUTHOR: CWS

LESSON: OP-MC- PSS-RN

OBJECTIVES: OP-MC-PSS.RN Obj. 8

REFERENCES: OP-MC-PSS-RN page 67

2.2 Equipment Control

2.2.1 **Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.**
(CFR: 45.1)
IMPORTANCE RO 3.7 SRO 3.6

2.2.2 **Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.**
(CFR: 45.2)
IMPORTANCE RO 4.0 SRO 3.5

2.2.3 (multi-unit) **Knowledge of the design, procedural, and operational differences between units.**
(CFR: 41 / 43 / 45)
IMPORTANCE RO 3.1 SRO 3.3

2.2.4 (multi-unit) **Ability to explain the variations in control board layouts, systems, instrumentation and procedural actions between units at a facility.**
(CFR: 45.1 / 45.13)
IMPORTANCE RO 2.8 SRO 3.0*

2.2.5 **Knowledge of the process for making changes in the facility as described in the safety analysis report.**
(CFR: 43.3 / 45.13)
IMPORTANCE RO 1.6 SRO 2.7

2.2.6 **Knowledge of the process for making changes in procedures as described in the safety analysis report.**
(CFR: 43.3 / 45.13)
IMPORTANCE RO 2.3 SRO 3.3

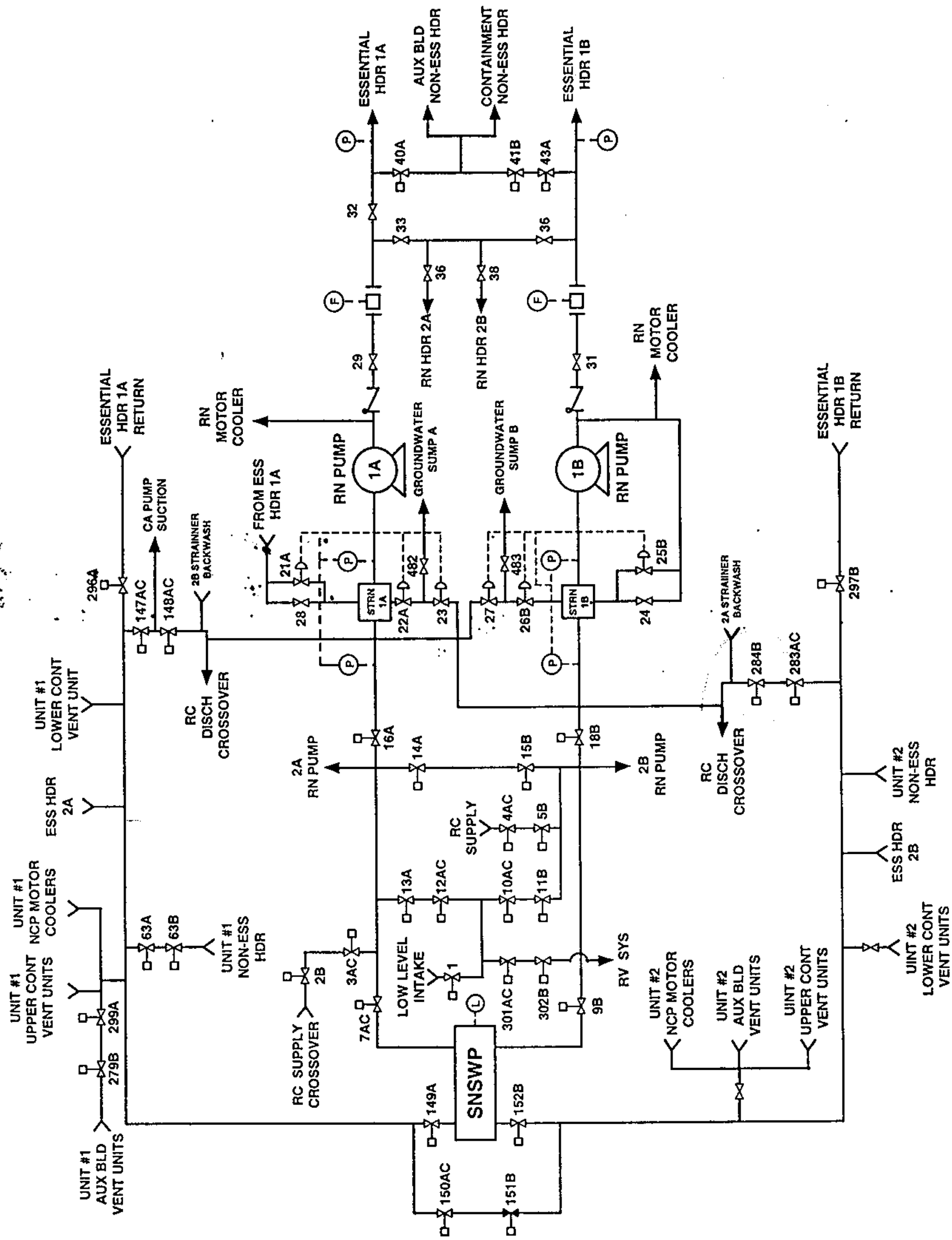
2.2.7 **Knowledge of the process for conducting tests or experiments not described in the safety analysis report.**
(CFR: 43.3 / 45.13)
IMPORTANCE RO 2.0 SRO 3.2

2.2.8 **Knowledge of the process for determining if the proposed change, test, or experiment involves an unreviewed safety question.**
(CFR: 43.3 / 45.13)
IMPORTANCE RO 1.8 SRO 3.3

OBJECTIVES

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
8	Describe the RN System Flow path (suction source, essential and non-essential header alignment and discharge point) for the following:					
	• Normal operation	X	X	X	X	
	• Operation following a Blackout	X	X	X	X	X
	• Operation following a Safety Injection	X	X	X	X	X
9	Explain the reason for taking a suction on the low level intake.	X	X	X	X	
10	Concerning the RN essential and non-essential headers:					
	• List the loads supplied by each header	X	X	X	X	
	• Identify which loads are automatically supplied on a Blackout, Safety injection and/or Phase B.	X	X	X	X	X
11	Explain the reason for **NOT** isolating the auxiliary building non-essential header on a Blackout signal.	X	X	X	X	X
12	Describe the operation including any interlocks for the following valves: • RN42A (AB Non Ess Supply Isol) • RN-70A (171B) (A(B) D/G Supply Isol) • 1RN1 (Low Level Intake Isolation) • Engineering Safeguards Modulating Control Valves and Reset Circuitry		X	X	X	X
13	Given a parameter associated with the RN system, describe the indications for that parameter.	X	X	X	X	
14	Given a Limit and Precaution associated with the RN System, discuss its basis and when it applies.		X	X	X	X

7.2 RN System Basic Layout (5/01/01)



1 Pt Which one of the following describes the bases for prioritizing Critical Safety Functions (CSF)?

A. **The CSFs are prioritized to address challenges to the boundaries that protect the general public from exposure to radiation.**

B. **The CSFs are prioritized to address design bases accidents that are described in the USFAR.**

C. **The CSFs are prioritized to ensure the proper optimal response procedure is implemented.**

D. **The CSFs are prioritized to address challenges to parameters that would affect operation of Engineered Safeguard Features equipment.**

Bank Question: 1067	***Answer: A***

1 Pt Which one of the following describes the bases for prioritizing Critical Safety Functions (CSF)?

A. **The CSFs are prioritized to address challenges to the boundaries that protect the general public from exposure to radiation.**

B. **The CSFs are prioritized to address design bases accidents that are described in the USFAR.**

C. **The CSFs are prioritized to ensure the proper optimal response procedure is implemented.**

D. **The CSFs are prioritized to address challenges to parameters that would affect operation of Engineered Safeguard Features equipment.**

Distracter Analysis:.

A. **Correct:**
Plausible:
B. **Incorrect:**
Plausible:
C. **Incorrect:**
Plausible:
D. **Incorrect**
Plausible:

LEVEL: SRO

KA: G 2.4.22 (3/0/4.0)

SOURCE: NEW

LEVEL OF KNOWLEDGE: Memory

AUTHOR: CWS

LESSON: OP-MC- EP-INTRO

OBJECTIVES: OP-MC-EP-E-1 Obj. 1 & 3

REFERENCES: OP-MC-EP-INTRO pages 21, 27, 29

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2.4 **Emergency Procedures /Plan (Continued)**

2.4.18 **Knowledge of the specific bases for EOPs.**
(CFR: 41.10 / 45.13)
IMPORTANCE RO 2.7 SRO 3.6

2.4.19 **Knowledge of EOP layout, symbols, and icons.**
(CFR: 41.10 / 45.13)
IMPORTANCE RO 2.7 SRO 3.7

2.4.20 **Knowledge of operational implications of EOP warnings, cautions, and notes.**
(CFR: 41.10 / 45.13)
IMPORTANCE RO 3.3 SRO 4.0

2.4.21 **Knowledge of the parameters and logic used to assess the status of safety functions including:**
1. Reactivity control
2. Core cooling and heat removal
3. Reactor coolant system integrity
4. Containment conditions
5. Radioactivity release control.
(CFR: 43.5 / 45.12)
IMPORTANCE RO 3.7 SRO 4.3

2.4.22 **Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.**
(CFR: 43.5 / 45.12)
IMPORTANCE RO 3.0 SRO 4.0

2.4.23 **Knowledge of the bases for prioritizing emergency procedure implementation during emergency operations.**
(CFR: 41.10 / 45.13)
IMPORTANCE RO 2.8 SRO 3.8

2.4.24 **Knowledge of loss of cooling water procedures.**
(CFR: 41.10 / 45.13)
IMPORTANCE RO 3.3 SRO 3.7

2.4.25 **Knowledge of fire protection procedures.**
(CFR: 41.10 / 45.13)
IMPORTANCE RO 2.9 SRO 3.4

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
N/A	N/A	3.0	3.0	3.0

OBJECTIVES

SEQ	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	List the six Critical Safety Functions in order of importance. EPINTRO001			X	X	
2	List the two EP's which provide the entry points into the EP set. EPINTRO002			X	X	
3	Explain when and how the CSF Status Trees are evaluated. EPINTRO003			X	X	
4	Apply the EP Rules of Usage to determine required actions for a step in an EP that is not satisfied when no contingency action (no RNO column) is provided. EPINTRO004			X	X	X
5	Apply the EP Rules of Usage to determine required actions while performing an EP contingency action when the action cannot be performed or is not successful. EPINTRO005			X	X	X
6	State when Foldout Page actions or transitions are applicable. EPINTRO006			X	X	
7	Describe how to determine if sequence is important when performing subtasks within a step of an EP. EPINTRO007			X	X	
8	Discuss the purpose and applicability of Notes and Cautions. EPINTRO008			X	X	
9	Define the "Constrained Language" terms listed in OMP 4-3, Use of Abnormal and Emergency Procedures. EPINTRO009			X	X	X

1.3. Critical Safety Functions

The concept of *Critical Safety Functions* (CSF's) came about after the TMI accident, and has been implemented in the WOG ERG's. The CSF's define parameters that if maintained within specific limits will assure that radioactive materials will not be released from the plant.

Objective # 1

The six CSF's, in the order of their priority are:

1. **Achievement of Subcriticality (S)**
2. **Maintenance of Core Cooling (C)**
3. **Maintenance of the Heat Sink (H)**
4. **Maintenance of the Reactor Coolant System Integrity (P)**
5. **Protection of the Containment Boundary Integrity (Z)**
6. **Maintenance of the Reactor Coolant System Inventory (I)**

The ERG's address these concerns first, and only after challenges to the CSF's are handled is it appropriate to turn the attention of the operating crew to the event cause.

The CSF's address the secureness of the four boundaries that protect the general public from exposure to radioactive materials that could be released during an accident. These boundaries are the

- Fuel matrix/cladding,
- NC system pressure boundary,
- Containment barriers, and
- Site boundary.

The ERG's implemented as the Emergency Procedures (EP's), address only the first three of these. The Site Emergency Plan addresses the fourth.

2.3.3. Function Restoration Procedures

Objective # 3

Challenges to the Critical Safety Functions (CSF's) are addressed by the Function Restoration Procedures. Each CSF is monitored by *Status Tree* in order of priority. Monitoring of Status Trees begins either when directed by E-0, or upon any transition from E-0.

The six Status Trees, one for each CSF, are found in procedure F-0.

1.	F-0.1, SUBCRITICALITY	(Procedure series "S")
2.	F-0.2, CORE COOLING	(Procedure series "C")
3.	F-0.3, HEAT SINK	(Procedure series "H")
4.	F-0.4, INTEGRITY	(Procedure series "P")
5.	F-0.5, CONTAINMENT	(Procedure series "Z")
6.	F-0.6, INVENTORY	(Procedure series "I")

Objective # 3

Each Status Tree includes four color-coded challenges to the CSF being monitored. The color coding represents the severity of the challenge, and thus the priority of the required response. The table lists the colors in their order of priority, defines the challenge, and shows the symbols that are used in the CSF Status Trees.

Color prioritization is important. A red path is addressed before any orange path. Any orange path is addressed before any yellow path. If more than one Status Tree indicates the same color, then priority is addressed by the monitoring order of S - C - H - P - Z - I.

STATUS TREE PRIORITY IDENTIFICATION			
Color	**Severity of Challenge**	**Line Code**	**Symbol**
Red	The CSF is under extreme challenge. Immediate operator action is required.	▬▬▬	●
Orange	The CSF is under severe challenge. Prompt operator action is required.	v v v v v v	◕
Yellow	The CSF condition is off-normal or not satisfied. Operator action may be taken.	λλλλλλλ	◔
Green	The CSF is NOT challenged. No operator action is needed.	═══	○

There is only one entry point to each Status Tree in F-0. However, there are multiple exit points, but only one exit point is possible. The path depends on the plant parameters which are symptomatic of the particular CSF. The exit is always to one of the several procedures associated with the specific CSF Status Tree, or, if the path is green, to remain in the E series procedure being executed. If the exit is to an FR procedure, then the E series procedure is suspended. **The FRP's are designed to restore the condition of the CSF and not the plant.** When the FRP is exited, the operator is directed back to the appropriate ORP to continue plant recovery.