Draft Submittal
(Pink Paper)

Senior Reactor Operator Written Exam

CRYSTAL RIVER OCTOBER 2007

EXAM NO. 50-302/2007-301

EXAMINER COPY

NRC EXAM – 2007

References Provided

Both Exams: *Steam Tables*

NRC EXAM	REQUIRED REFERENCES		**RO**
Question #	K/A	Reference	Title
21	024AG2.4.47	AP-490	Reactor Coolant System Boration
43	062A2.12	AP-770	Emergency Diesel Generator Actuation *(Enclosure 1 only)*
53	076AG2.2.22	ITS 3.4.15	RCS Specific Activity
64	BW/E05EK1.2		RM-A12 Conversion Table (Simulator Version)

NRC EXAM	REQUIRED REFERENCES		**SRO**
Question #	K/A	Reference	Title
5	025AG2.1.25	OP-103H	RCS and SFP Decay Heat Tables & Figures
8	029G2.1.23	EM-225A	Post Accident RB Hydrogen Control
14	064G2.1.22	OP-103F ITS 3.8.3	Tank Volumes *(Figure 8 only)* Electrical Power Systems
15	067AG2.4.41	EM-202	Duties of the Emergency Coordinator *(Enclosure 1 only)*
19	G2.1.34	CP-142	Primary Water Chemistry Program *(Remove Page 9 & Enclosure 6 & Enclosure 11)*
22	G2.2.26	AI-504	Guidelines for Cold Shutdown and Refueling *(Enclosure 1 through 6 only)*

1. 003A2.02 001/2/1/SRO#1/C/A 3.7/3.9/NEW/S/CR03701/
The following plant conditions exist:

- The plant is at 100% power.
- Seal injection flow has been lost.
- RCP-1B vibrations are in the "action" range on one monitor and in the "alert" range on another monitor for the same parameter.
- The following RCP information is noted:

Parameter	RCP-1A	RCP-1B	RCP-1C	RCP-1D
SW Out	165° F	170° F	160° F	165° F
Thrust Brg	170° F	200° F	170° F	260° F

Which of the following actions are required to be taken for the above conditions?

A. Reduce plant power to 90% using AP-510, stop RCP-1B, perform AP-545.

B. Reduce plant power to 90% using AP-510, stop RCP-1D, perform AP-545.

C.✔ Trip the reactor and perform EOP-2. Stop RCP-1B and RCP-1D.

D. Trip the reactor and perform EOP-2. Stop all RCPs.

Reasons:

A. This would be the correct path if only RCP-1B vibrations were of concern.

B. This would be the correct path if only RCP-1D thrust bearing temperature were of concern.

C. Correct. With both RCP-1B and RCP-1D meeting trip criteria, the reactor must be tripped and both pumps secured.

D. With only 2 pumps jeopardized, securing all RCPs would be incorrect. With only a loss of seal injection, RCP-1A and RCP-1C can continue to operate.

OPS 4-60, Obj. 9; OP-302 Step 3.2.13 & 3.2.16 & Note prior to 4.10.1; 10 CFR 55.43.b.5

SRO - New

Reference(s) provided: None

4.10 Abnormal RCP Thrust Bearing Temperature (Reference)

ACTION	DETAILS

NOTE

The "RCP MTR THRUST BRG TEMP HIGH" alarm will actuate if thrust bearing temperature is > 185 °F.

NOTE

To prevent seal failure, RCPs MUST be tripped if thrust bearing exceeds 250 °F.

4.10.1	VERIFY Thrust Bearing Temperatures to determine affected RCP	• RC-133-TI • Plant Computer 1A ____/____/____ 1B ____/____/____ 1C ____/____/____ 1D ____/____/____ Initials / Date / Time
4.10.2	Start Available Lift Oil Pumps and contact Nuclear Plant Technical Support (NPTS)	• START available Lift Oil Pumps 2A ____/____/____ 2B/3B ____/____/____ 2C/3C ____/____/____ 2D/3D ____/____/____ Initials / Date / Time
4.10.3	IF temperature continues to increase, THEN REDUCE power to ≤ 72% per AP-510, Rapid power Reduction, to allow for RCP Shutdown	1. __ MONITOR RCP operating parameters: • RC-133-TI • Computer Groups 78 through 81 • RCP seal stage pressures • REFER TO OP-204, Power Operations 2. __ REDUCE power to ≤ 72% per AP-510, Rapid Power Reduction ____/____/____ Initials / Date / Time

3.2 Limits and Precautions

	LIMIT	BASIS
3.2.1	Maintain Seal Injection Flow of 8-10 gpm/pump to all RCPs during normal operations when RCS is > 150 psig and > 150 °F	Ensures proper cooling and flushing flows to seals to prevent seal damage
3.2.2	RCPs may be operated without Seal Injection Flow provided SW Cooling is in operation	Cooling is maintained by internally circulating RCS
3.2.3	Idle RCPs are required to have their Controlled Bleed-Off valves closed if seal injection has been lost	Restricts the heatup rate of the seal cartridge to minimize the possibility of seal damage
3.2.4	RCPs must be stopped within two minutes upon loss of both Seal Injection Flow and SW flow	Cooling water supplies to seals has been lost and seal damage may occur
3.2.5	Maximum allowable time to operate RCP Motor without SW flow is 5 minutes	To prevent motor damage
3.2.6	Maximum allowable time to operate RCPs with Controlled Bleed-off line secured is 5 minutes (30 minutes when in an EOP)provided that Seal Injection and/or SW cooling is maintained	Ensures proper seal staging RCP operation with CBO isolated should be limited to one time during life of seal
3.2.7	RCP may be started three successive times from ambient conditions or two successive times from operating conditions (Additional starts may be allowed if RCP has been running for 25 minutes or shutdown for 2.5 Hrs)	Prevent motor damage due to overheating windings

3.2 Limits and Precautions

	LIMIT	BASIS
3.2.8	Seal Injection Flow Control Valve MUV-16 should be closed if seal injection is lost and then gradually restored to 8 - 10 gpm/pump per Section 4.9	Prevent seal damage on restoration of Seal Injection Flow
3.2.9	Operation of one RCP at system pressures < 400 psig should be minimized	Prevent reduction of RCPs Seal Lifetimes and increased RCP vibration
3.2.10	Single RCP operation shall be minimized to brief periods during transition onto and off of Decay Heat Removal	To prevent run times in an area that causes increased radial load on pump shafts and reduced performance of hydrostatic bearing
3.2.11	IF RCP cooling water temperature leaving pump is ≥ 180 °F, THEN TRIP affected pump	Indicative of RCP problem and prevents further damage
3.2.12	IF for any reason the pump motor winding heaters are to be turned off while pump is shutdown, THEN SW to the air cooler and upper and lower bearings must also be isolated	Prevents damage from collected condensation
3.2.13	IF thrust bearing temperature exceeds 250 °F AND the temperature is increasing THEN trip the affected RCP	Thrust bearing temp > 250 °F indicates bearing failure is occurring which places excessive axial loading on the RCP mechanical seal. Rapid and significant seal failure is likely if the pump is not secured
3.2.14	IF thrust bearing temp ≥ 185 °F, THEN START lift oil pump(s) AND CONTACT Nuclear Plant Technical Support (NPTS) for evaluation	Indication of loss of lubrication. Lift oil pump(s) will provide additional lubrication to bearing shoes

3.2 Limits and Precautions

	LIMIT	BASIS
3.2.15	IF any RCP vibration is in the "Alert" range on any parameter, THEN IMMEDIATELY NOTIFY NPTS and continue to monitor affected RCP	Evaluate RCP performance and initiate additional monitoring
3.2.16	IF any RCP vibration is in the "Action" range on a parameter, AND that same parameter is in the "Alert" range on another monitor for that RCP, THEN REDUCE power to <90% per AP-510, Rapid Power reduction AND SECURE affected RCP	Significant pump damage may occur with continued RCP operation
3.2.17	IF any RCP has less than two operable vibration monitors, THEN SECURE the affected RCP within 30 days, OR ESTABLISH temporary monitoring capabilities	FPC commitment
3.2.18	WHEN RCS Temp is ≥ 150 °F, AND SW cooling is lost to the RCPs for ≥ 5 minutes THEN GRADUALLY RESTORE SW and NOTIFY NPTS	Ref. SER 19-85 to prevent cyclic stresses on the pump cover
3.2.19	Seal Injection Flow to RCP must be initiated prior to opening CBO Valves	Prevent unfiltered water from RCS entering seal area
3.2.20	DO NOT start a RCP after its static seal temperature has exceeded 250 °F without NPTS approval	Possible heat damage to polymer based materials in the seal package
3.2.21	DO NOT Operate a RCP below 100 °F during cooldown and 95 °F during heatup.	PTLR limits

2. 004G2.2.25 001/2/1/SRO#2/MEM 2.5/3.7/NEW/S/CR03701/
The following plant conditions exist:

- The plant is at 100% power.
- Engineering reports that, due to a recently completed modification, HPI Recirc to RB Sump valves MUV-543, MUV-544, MUV-545, and MUV-546 could not be opened if needed.

Which LCOs (if any) should be entered for the given condition?

A.✓ All LCOs are met.

B. Enter LCO 3.5.2 Condition A; do not enter LCO 3.0.3.

C. Enter LCO 3.5.2 Condition A; enter LCO 3.0.3.

D. Do not enter LCO 3.5.2 Condition A; enter LCO 3.0.3.

Reasons:

Per ITS 3.5.2 Basis: *"...removing a train of the recirculation line to the RB sump or the entire bank of valves for maintenance does not render the HPI System inoperable, given the diverse ability to recirculate to the Makeup Tank. HPI satisfies Criterion 3 of the NRC Policy Statement which addresses SSCs that are part of the primary success path, and which function or actuate to mitigate a design basis accident or transient challenging a fission product barrier. Since this recirculation line supports piggyback operation in long-term cooling, and piggyback operation is not a primary success path, LCO 3.5.2 need not be entered when this recirculation path is not available."*

A. Correct. See discussion above.

B. Plausible if student believes that an HPI train is inoperable.

C and D. Plausible if student believes that 100% HPI flow does not exist.

OPS 5-01, Obj. 8; ITS 3.5.2 Basis; 10 CFR 55.43.b.2

SRO - New

Reference(s) provided: None

BASES

ACTIONS

A.1

With one or more ECCS trains inoperable and at least 100% of the flow equivalent to a single OPERABLE ECCS train available, the inoperable components must be returned to OPERABLE status within 72* hours. The 72 hour Completion Time is based on NRC recommendations (Ref. 3) that are based on a risk evaluation and is a reasonable time for many repairs.

*On a one-time basis, an Emergency Core Cooling System train may be inoperable as specified by Required Action A.1 for up to 10 days to allow performance of Decay Heat Seawater System Pump RWP-3B repairs online. Upon completion of the refurbishment and system restoration this footnote is no longer applicable.

An ECCS train is inoperable if it is not capable of delivering the design flow to the RCS.

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. The intent of this Condition is to maintain a combination of equipment such that the safety injection (SI) flow equivalent to 100% of a single train remains available. This allows increased flexibility in plant operations under circumstances when components in opposite trains are inoperable.

For example, removing a train of the recirculation line to the RB sump or the entire bank of valves for maintenance does not render the HPI System inoperable, given the diverse ability to recirculate to the Makeup Tank. HPI satisfies Criterion 3 of the NRC Policy Statement which addresses SSCs that are part of the primary success path, and which function or actuate to mitigate a design basis accident or transient challenging a fission product barrier. Since this recirculation line supports piggyback operation in long-term cooling, and piggyback operation is not a primary success path, LCO 3.5.2 need not be entered when this recirculation path is not available.

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 3) has shown the risk of having one full ECCS train inoperable to be sufficiently low to justify continued operation for 72 hours.

With one or more components inoperable such that the flow equivalent to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be immediately entered.

(continued)

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS-Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more trains inoperable. AND At least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.	A.1 Restore train(s) to OPERABLE status.	72* hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

*On a one-time basis, an Emergency Core Cooling System train may be inoperable as specified by Required Action A.1 for up to 10 days to allow performance of Decay Heat Seawater System Pump RWP-3B repairs online. Upon completion of the refurbishment and system restoration this footnote is no longer applicable.

3. 005AA2.03 001/1/2/SRO#3/C/A 3.5/4.4/BANK/S/CR03701/5-001-019
The following plant conditions exist:

- The plant is at 75% power.
- 3 RCPs are in operation.
- SP-333, Control Rod Exercises, is in progress.
- A malfunction occurs during the evolution and control rods 5-3 and 5-4 drop to 70% withdrawn and remain there. (assume control rods are still trippable).
- No automatic actions occur.

Which of the following action(s) is required?

A. Trip the reactor and enter EOP-2, Vital System Status Verification.

B. Reduce plant power to 60% using AP-545, Plant Runback.

C. ✓ Enter TS 3.1.4 "Control Rod Group Alignment Limits."

D. Enter TS 3.1.5 "Safety Rod Insertion Limits."

Reasons:

A. This was the old requirement. AI-505 does not require tripping the reactor on 2 misaligned rods anymore.

B. If AP-545 is entered then power must be reduced to 45% due to only 3 RCPs running.

C. Correct. TS 3.1.4 should be entered.

D. TS 3.1.5 is only required if a Safety Rod is not fully withdrawn.

OPS 5-01, Obj. 12; ITS 3.1.4 and Basis; 10 CFR 55.43.b.2

SRO - Bank

Reference(s) provided: None

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 CONTROL ROD Group Alignment Limits

LCO 3.1.4 Each CONTROL ROD shall be OPERABLE and aligned to within 6.5% of its group average height.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One trippable CONTROL ROD inoperable, or not aligned to within 6.5% of its group average height, or both.	A.1 Align all CONTROL RODS in the group to within 6.5% of the group average height, while maintaining the rod insertion, group sequence, and group overlap limits in accordance with LCO 3.2.1, "Regulating Rod Insertion Limits."	1 hour
	OR	
	A.2.1.1 Verify SDM is ≥ 1% Δk/k.	1 hour AND Once per 12 hours thereafter
	OR	
	A.2.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2 Reduce THERMAL POWER to ≤ 60% of the ALLOWABLE THERMAL POWER.	2 hours
	AND	
	A.2.3 Reduce the nuclear overpower trip setpoint to ≤ 70% of the ALLOWABLE THERMAL POWER.	10 hours
	AND	
	A.2.4 Verify the potential ejected rod worth is within the assumptions of the rod ejection analysis.	72 hours
	AND	
	A.2.5 Perform SR 3.2.5.1.	72 hours
B. Required Action and associated Completion Time for Condition A not met.	B.1 Be in MODE 3.	6 hours
C. More than one trippable CONTROL ROD inoperable, or not aligned within 6.5% of its group average height, or both.	C.1.1 Verify SDM is ≥ 1% Δk/k.	1 hour
	OR	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	C.2 Be in MODE 3.	6 hours
D. One or more CONTROL RODS untrippable.	D.1.1 Verify SDM is ≥ 1% Δk/k.	1 hour
	OR	
	D.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	D.2 Be in MODE 3.	6 hours

4. 009G2.4.4 001/1/1/SRO#4/C/A 4.0/4.3/NEW/S/CR03701/
The following plant conditions exist:

- The plant is operating at 100% power with Tave stable at 579° F.
- Pressurizer level is 221" and MUT level is 92".
- An unisolable RCS leak occurs.
- After 3 minutes, pressurizer level is 182" and MUT level is 88".

Which of the following procedures will be used for this event?

Procedure names are as follows:

AP-520	Loss of RCS Coolant or Pressure
EOP-2	Vital System Status Verification
EOP-3	Inadequate Subcooling Margin
EOP-8A	LOCA Cooldown
EOP-10	Post Trip Stabilization
OP-208	Power Shutdown
OP-209	Plant Cooldown
OP-211	Reactor Shutdown

A. AP-520, OP-208, OP-209, and OP-211

B.✔ AP-520, EOP-2, and EOP-8A

C. EOP-2, EOP-3, and EOP-8A

D. AP-520, EOP-2, and EOP-10

Reasons:

A. Plausible if the leak were small enough to be maintained using normal makeup via MUV-31.

B. Correct. Leak rate would be calculated to be ~ 200 gpm, which would exceed the capacity of normal makeup via MUV-31. AP-520 would be entered and then the reactor would be tripped. AP-520 and EOP-2 would be performed concurrently. With leakage > 100 gpm, EOP-2 will ultimately lead to EOP-8A.

C. Plausible if the leak were large enough to cause a loss of subcooling margin. With a leak size of 200 gpm, normal operator actions would be expected to maintain adequate subcooling margin throughout the event.

D. This path would be chosen if RCS leakage were excessive, but less than 100 gpm.

OPS 5-114, Obj. 1; OPS 5-96, Obj. 6; AP-520 Steps 1.0 & 3.14; EOP-2 Step 3.29; 10 CFR 55.43.b.5

SRO - New

Reference(s) provided: None

LOSS OF RCS COOLANT OR PRESSURE

1.0 ENTRY CONDITIONS

IF any of the following occur:

- A significant rise in RCS leakage
- An uncontrolled RCS PRESS reduction

THEN use this procedure.

2.0 IMMEDIATE ACTIONS

NOTE
There are no immediate actions for this procedure.

3.0 FOLLOW-UP ACTIONS (CONT'D)

ACTIONS	DETAILS
3.14 ___ IF at any time, all the following exist: ___ PZR level can NOT be maintained using MUV-31 ___ Rx NOT tripped THEN trip the Rx and **CONCURRENTLY PERFORM** EOP-02, Vital System Status Verification, beginning with Step 2.1	
3.15 ___ Maintain MUT level ≥ 55 in.	• ___ IF feed from BWST is desired, THEN cycle appropriate BWST outlet valve: ___ MUV-73 ___ MUV-58

3.0 FOLLOW-UP ACTIONS (CONT'D)

ACTIONS	DETAILS
3.29 ___ Verify RCS leakage is acceptable.	• Observe the following: ___ RM-A6 ___ MUT level ___ MU flow ___ PZR level ___ RB sump level
___ IF RCS leakage is not acceptable, THEN **CONCURRENTLY PERFORM** AP-520, Loss of RCS Coolant or Pressure, beginning with Step 3.1	
___ IF RCS leakage is > 100 gpm, THEN **GO TO** EOP-08A, LOCA Cooldown, beginning with Step 3.1	
3.30 ___ **GO TO** EOP-10, Post-Trip Stabilization, beginning with Step 3.1	

5. 025AG2.1.25 001/1/1/SRO#5/C/A 2.8/3.1/BANK/S/CR03701/5-078-003
The following plant conditions exist:

- The Reactor has been shutdown for 10 days.
- "A" Decay Heat Removal train is in service.
- RCS temperature is 90°F.
- RCS level is 136 feet.
- Cold Leg Nozzle Dams are installed.
- The RCS is vented.
- The core has not been unloaded (RV head on).
- "B" ES 4160V Bus (including EDG-1B) is inoperable due to bus bar work.

A complete loss of the "A" ES 4160V bus occurs and EDG-1A will not start. If no Operator action is performed, approximately how long will it take to core uncovery?

A. 199 minutes

B. 295 minutes

C.✔ 453 minutes

D. 503 minutes

Reasons:

Correct use of OP-103H will yield "C" as the correct answer. Other choices will be found by using the incorrect curve or using the incorrect Decay Heat Correction Factor. Step 5.8: 2.2732 (Step 3.3) x 199.32 (Step 5.4) = 453

OPS 5-78, Obj. 5; OP-103H; 10 CFR 55.43.b.5

SRO - Bank

Reference(s) provided: OP-103H

DETERMINATION OF RCS TIME TO BOIL, TIME TO 200°F, TIME TO SATURATION, AND TIME TO CORE UNCOVERY [NOCS 095314]

1.0 RECORD the following plant conditions:

1.1 RCS Temperature (°F) _____

1.2 RCS Level (ft) _____

1.3 Number of burned fuel assemblies in core _____

1.4 Shutdown Date/Time _____

1.5 Current Date/Time _____

2.0 DETERMINE the number of days shutdown _____

3.0 DETERMINE the correction factors to apply:

> **NOTE**
>
> **Tables 1 – 15 provide time to boil, time to 200°F, time to saturation, and time to core uncovery values for one point in time, which was chosen to be 1 day following reactor shutdown. A correction factor has been provided for all other times after shutdown.**

3.1 OBTAIN the Decay Heat Correction Factor for number of days shutdown (Step 2.0) from Table 16.

Decay Heat Correction Factor (Table 16) = 2.2732

> **NOTE**
>
> **Tables 1 – 15 provide time to boil, time to 200°F, time to saturation, and time to core uncovery values for a full core of 177 burned fuel assemblies. A correction factor can be calculated for periods when the reactor has less than 177 fuel assemblies or has been reloaded with one-third unburned (new) assemblies.**

3.2 DETERMINE the Number of Fuel Assemblies Correction Factor:

Fuel Assembly Correction Factor = 177 ÷ (Step 1.3)
= 177 ÷ 177 = 1

3.3 CALCULATE Overall Correction Factor:

Overall Correction Factor = (Step 3.1) × (Step 3.2)
= 2.2732 × 1 = 2.2732

DETERMINATION OF RCS TIME TO BOIL, TIME TO 200°F, TIME TO SATURATION, AND TIME TO CORE UNCOVERY [NOCS 095314]

4.6 Apply the Overall Correction Factor to determine the current Time to 200°F:

Time to 200°F = (Step 3.3) x (Step 4.2)
= _____ x _____ = _____ min

4.7 Apply the Overall Correction Factor to determine the current Time to Saturation:

Time to Saturation = (Step 3.3) x (Step 4.3)
= _____ x _____ = _____ min

4.8 Apply the Overall Correction Factor to determine the current Time to Core Uncovery:

Time to Core Uncovery = (Step 3.3) × (Step 4.4)
= _____ × _____ = _____ min

4.9 Perform either section 5.0 or 6.0 depending on SG Nozzle Dam status. Use the longer time of 4.0 and 5.0 or 4.0 and 6.0 for AI-504 purposes.

5.0 IF ALL of the following conditions are met:

- RCS Vented (at least one vent open)
- SG Nozzle Dams Installed

THEN GO TO Step 5.1.

IF NOT,
THEN GO T0 Step 6.0

5.1 OBTAIN Time to Boil from Table 2 for the RCS temperature and level recorded in Steps 1.1 and 1.2.

Time to Boil (Table 2) = _____ min

5.2 OBTAIN Time to 200°F from Table 5 for the RCS temperature and level recorded in Steps 1.1 and 1.2.

Time to 200°F (Table 5) = N/A min

5.3 OBTAIN Time to Saturation from Table 9 for the RCS temperature and level recorded in Steps 1.1 and 1.2.

Time to Saturation (Table 9) = _____ min

5.4 OBTAIN Time to Core Uncovery from Table 13 for the RCS temperature and level recorded in Steps 1.1 and 1.2.

Time to Core Uncovery (Table 13) = 199.32 min

DETERMINATION OF RCS TIME TO BOIL, TIME TO 200°F, TIME TO SATURATION, AND TIME TO CORE UNCOVERY [NOCS 095314]

5.5 Apply the Overall Correction Factor to determine the current Time to Boil:

Time to Boil = (Step 3.3) × (Step 5.1)
= _____ × _____ = _____ min

5.6 Apply the Overall Correction Factor to determine the current Time to 200°F:

Time to 200°F = (Step 3.3) x (Step 5.2)
= _____ x _____ = _____ min N A

5.7 Apply the Overall Correction Factor to determine the current Time to Saturation:

Time to Saturation = (Step 3.3) x (Step 5.3)
= _____ x _____ = _____ min

5.8 Apply the Overall Correction Factor to determine the current Time to Core Uncovery:

Time to Core Uncovery = (Step 3.3) × (Step 5.4)
= 2.2732 × 199.32 = 453 min

6.0 IF ALL of the following conditions are met:

- RCS Vented (at least one vent open)
- SG Nozzle Dams NOT Installed

THEN GO TO Step 6.1.

IF NOT,
THEN GO T0 Step 7.0

6.1 OBTAIN Time to Boil from Table 2 for the RCS temperature and level recorded in Steps 1.1 and 1.2. N A

Time to Boil (Table 2) = _____ min

6.2 OBTAIN Time to 200°F from Table 6 for the RCS temperature and level recorded in Steps 1.1 and 1.2.

Time to 200°F (Table 6) = _____ min

6.3 OBTAIN Time to Saturation from Table 10 for the RCS temperature and level recorded in Steps 1.1 and 1.2.

Time to Saturation (Table 10) = _____ min

Decay Heat Correction Factor

	Decay Heat
Time	Days
1	1.0000
2	1.2341
3	1.4076
4	1.5646
5	1.7071
6	1.8428
7	1.9664
8	2.0770
9	2.1769
10	2.2732
11	2.3669
12	2.4584
13	2.5482
14	2.6345
15	2.7168
16	2.7948
17	2.8684
18	2.9376
19	3.0026
20	3.0637
21	3.1216
22	3.1771
23	3.2310
24	3.2865
25	3.3423
26	3.3979
27	3.4532
28	3.5083
29	3.5630
30	3.6174

	Decay Heat
Time	Days
31	3.6713
32	3.7249
33	3.7779
34	3.8305
35	3.8825
36	3.9340
37	3.9850
38	4.0354
39	4.0853
40	4.1347
41	4.1836
42	4.2319
43	4.2798
44	4.3273
45	4.3744
46	4.4212
47	4.4683
48	4.5153
49	4.5622
50	4.6087
51	4.6550
52	4.7009
53	4.7465
54	4.7917
55	4.8366
56	4.8811
57	4.9252
58	4.9689
59	5.0122
60	5.0550

LINEAR INTERPOLATION OF DH CORRECTION FACTORS BETWEEN DAYS IS PERMISSIBLE (INTERPOLATE BETWEEN 0.0000 AND 1.0000 IF LESS THAN 1 DAY)

Time to Core Uncovery (minutes) with RCS Vented, Nozzle Dams Installed

Initial Temperature (°F)	60.00	70.00	80.00	90.00	100.00	110.00	120.00	130.00	140.00	150.00	160.00	170.00	180.00	190.00	200.00
Level(ft)	**T=60°**	**T=70°**	**T=80°**	**T=90°**	**T=100°**	**T=110°**	**T=120°**	**T=130°**	**T=140°**	**T=150°**	**T=160°**	**T=170°**	**T=180°**	**T=190°**	**T=200°**
173.00	314.31	335.58	331.76	327.94	324.12	320.30	316.49	312.69	308.89	305.11	301.33	297.56	293.81	290.06	286.32
172.00	314.31	335.58	331.76	327.94	324.12	320.30	316.49	312.69	308.89	305.11	301.33	297.56	293.81	290.06	286.32
171.00	314.31	335.58	331.76	327.94	324.12	320.30	316.49	312.69	308.89	305.11	301.33	297.56	293.81	290.06	286.32
170.00	314.31	335.58	331.76	327.94	324.12	320.30	316.49	312.69	308.89	305.11	301.33	297.56	293.81	290.06	286.32
169.00	314.31	335.58	331.76	327.94	324.12	320.30	316.49	312.69	308.89	305.11	301.33	297.56	293.81	290.06	286.32
168.00	314.31	335.58	331.76	327.94	324.12	320.30	316.49	312.69	308.89	305.11	301.33	297.56	293.81	290.06	286.32
167.00	314.31	335.58	331.76	327.94	324.12	320.30	316.49	312.69	308.89	305.11	301.33	297.56	293.81	290.06	286.32
166.00	314.31	335.58	331.76	327.94	324.12	320.30	316.49	312.69	308.89	305.11	301.33	297.56	293.81	290.06	286.32
165.00	314.31	335.58	331.76	327.94	324.12	320.30	316.49	312.69	308.89	305.11	301.33	297.56	293.81	290.06	286.32
164.00	314.31	335.58	331.76	327.94	324.12	320.30	316.49	312.69	308.89	305.11	301.33	297.56	293.81	290.06	286.32
163.00	314.31	335.58	331.76	327.94	324.12	320.30	316.49	312.69	308.89	305.11	301.33	297.56	293.81	290.06	286.32
162.00	314.31	335.58	331.76	327.94	324.12	320.30	316.49	312.69	308.89	305.11	301.33	297.56	293.81	290.06	286.32
161.00	314.31	335.58	331.76	327.94	324.12	320.30	316.49	312.69	308.89	305.11	301.33	297.56	293.81	290.06	286.32
160.00	311.23	332.03	328.25	324.46	320.67	316.89	313.11	309.34	305.57	301.82	298.08	294.34	290.62	286.90	283.20
159.00	307.68	327.95	324.21	320.46	316.71	312.96	309.22	305.49	301.76	298.05	294.34	290.64	286.96	283.28	279.61
158.00	304.14	323.88	320.17	316.46	312.74	309.04	305.33	301.63	297.95	294.27	290.60	286.94	283.29	279.65	276.02
157.00	300.59	319.80	316.13	312.45	308.78	305.11	301.44	297.78	294.13	290.49	286.86	283.24	279.63	276.02	272.42
156.00	297.04	315.72	312.09	308.45	304.82	301.18	297.55	293.93	290.32	286.72	283.12	279.54	275.96	272.39	268.83
155.00	293.50	311.64	308.05	304.45	300.85	297.26	293.67	290.08	286.51	282.94	279.38	275.84	272.30	268.77	265.24
154.00	289.95	307.57	304.01	300.45	296.89	293.33	289.78	286.23	282.69	279.17	275.65	272.14	268.63	265.14	261.65
153.00	286.41	303.49	299.97	296.45	292.92	289.40	285.89	282.38	278.88	275.39	271.91	268.43	264.97	261.51	258.06
152.00	282.86	299.41	295.93	292.44	288.96	285.48	282.00	278.53	275.07	271.61	268.17	264.73	261.30	257.89	254.47
151.00	279.31	295.33	291.89	288.44	285.00	281.55	278.11	274.68	271.25	267.84	264.43	261.03	257.64	254.26	250.88
150.00	275.77	291.26	287.85	284.44	281.03	277.62	274.22	270.83	267.44	264.06	260.69	257.33	253.98	250.63	247.29
149.00	272.22	287.18	283.81	280.44	277.07	273.70	270.33	266.98	263.63	260.28	256.95	253.63	250.31	247.00	243.70
148.00	268.68	283.10	279.77	276.44	273.10	269.77	266.44	263.12	259.81	256.51	253.21	249.93	246.65	243.38	240.11
147.00	265.13	279.02	275.73	272.43	269.14	265.84	262.56	259.27	256.00	252.73	249.47	246.22	242.98	239.75	236.52
146.00	261.58	274.95	271.69	268.43	265.17	261.92	258.67	255.42	252.19	248.96	245.74	242.52	239.32	236.12	232.93
145.00	258.04	270.87	267.65	264.43	261.21	257.99	254.78	251.57	248.37	245.18	242.00	238.82	235.65	232.49	229.34
144.00	254.49	266.79	263.61	260.43	257.25	254.07	250.89	247.72	244.56	241.40	238.26	235.12	231.99	228.87	225.75
143.00	250.95	262.71	259.57	256.43	253.28	250.14	247.00	243.87	240.74	237.63	234.52	231.42	228.33	225.24	222.16
142.00	247.40	258.64	255.53	252.42	249.32	246.21	243.11	240.02	236.93	233.85	230.78	227.72	224.66	221.61	218.57
141.00	243.85	254.56	251.49	248.42	245.35	242.29	239.22	236.17	233.12	230.08	227.04	224.02	221.00	217.98	214.98
140.00	237.65	247.69	244.69	241.69	238.68	235.68	232.68	229.68	226.70	223.72	220.75	217.79	214.83	211.88	208.94
139.00	229.35	238.91	235.99	233.07	230.14	227.22	224.30	221.39	218.48	215.59	212.70	209.81	206.94	204.07	201.20
138.00	218.55	227.64	224.82	222.00	219.18	216.36	213.55	210.74	207.94	205.15	202.36	199.58	196.80	194.04	191.28
137.00	207.75	216.46	213.75	211.03	208.31	205.60	202.89	200.18	197.49	194.79	192.11	189.43	186.76	184.09	181.43
136.00	196.21	204.53	201.92	199.32	196.71	194.11	191.51	188.91	186.33	183.74	181.17	178.60	176.04	173.48	170.93
135.00	168.61	191.21	188.73	186.25	183.77	181.29	178.81	176.34	173.87	171.41	168.96	166.51	164.07	161.63	159.20
134.00	156.44	177.23	174.88	172.53	170.17	167.82	165.47	163.13	160.79	158.46	156.14	153.82	151.50	149.19	146.89
133.00	146.87	166.22	163.97	161.72	159.47	157.22	154.97	152.73	150.50	148.27	146.04	143.82	141.61	139.40	137.19
132.00	130.55	147.45	145.38	143.30	141.23	139.15	137.08	135.01	132.95	130.89	128.84	126.79	124.75	122.71	120.67
131.00	105.95	119.17	117.35	115.54	113.73	111.91	110.10	108.30	106.49	104.69	102.90	101.11	99.32	97.54	95.76
130.00	88.81	99.46	97.83	96.20	94.57	92.94	91.31	89.69	88.06	86.45	84.83	83.22	81.62	80.01	78.41
129.00	76.03	84.76	83.27	81.78	80.28	78.79	77.29	75.80	74.32	72.83	71.36	69.88	68.41	66.94	65.47
128.00	56.41	62.21	60.92	59.63	58.35	57.06	55.78	54.50	53.22	51.94	50.67	49.40	48.13	46.87	45.61
127.00	36.78	39.63	38.56	37.48	36.40	35.33	34.25	33.18	32.11	31.04	29.97	28.91	27.85	26.79	25.73
126.00	26.54	27.85	26.88	25.92	24.95	23.98	23.01	22.05	21.09	20.13	19.17	18.21	17.26	16.31	15.36
125.00	17.92	17.94	17.07	16.19	15.31	14.44	13.57	12.69	11.82	10.95	10.08	9.22	8.36	7.49	6.63

6. 026A2.08 001/2/1/SRO#6/C/A 3.2/3.7/NEW/S/CR03701/
The following plant conditions exist:

- A large break LOCA is in progress.
- Building Spray was actuated 3 hours ago.
- RB pressure is 8 psig and stable.
- RB atmosphere I-131 is 10 μCi/cc.
- TSC has approved securing building spray if all other requirements are met.

Which ONE of the following choices represents the correct action to take?

A. Building spray should be secured. EOP-8A will provide the required guidance.

B. Building spray should be secured. EOP-3 will provide the required guidance.

C.✓ Building spray cannot be secured. EOP-8A will provide the required guidance.

D. Building spray cannot be secured. EOP-3 will provide the required guidance.

Reasons:

A. With only 3 hours elapsed, building spray cannot be secured.

B. With only 3 hours elapsed, building spray cannot be secured. Plausible, since EOP-3 is used to address LOCAs. However, it contains no guidance for securing building spray.

C. Correct. Securing RB Spray requires the following:
- More than 5 hours running
- RB Press < 10 psig and stable or lowering
- I-131 < 13 μCi/cc

With spray only in service for 3 hours, the requirements are not met. Guidance for securing RB spray can be provided by EOP-8A.

D. Plausible, since EOP-3 is used to address LOCAs. However, it contains no guidance for securing building spray.

OPS 5-95, Obj. 5; OPS 4-62, Obj. 9; EOP-8A Step 3.38; 10 CFR 55.43.b.5

SRO - New

Reference(s) provided: None

3.0 FOLLOW-UP ACTIONS (CONT'D)

ACTIONS	DETAILS
3.37 ___ Verify ≥ 1% SDM for 73°F RCS TEMP exists.	• ___ See OP-103C, Reactivity Worth Curves, Curve 18 or 19
___ IF ≥ 1% SDM for 73°F RCS TEMP does NOT exist, THEN ensure boration is in progress.	1 ___ Ensure at least 1 post-filter is in service. 2 ___ Open CAV-60 3 ___ Start CAP-1A or CAP-1B 4 ___ **CONCURRENTLY PERFORM** AP-490, Reactor Coolant System Boration, beginning with Step 3.1 (to terminate boration).

3.38 ___ IF at any time, all of the following exist: ___ BSPs running ≥ 5 hrs ___ RB PRESS < 10 psig ___ RB PRESS stable or lowering ___ RB atmosphere I^{131} < 13 μCi/cc ___ TSC has approved BS termination THEN stop BSPs.	1 Ensure BSPs are stopped: ___ BSP-1A ___ BSP-1B 2 Select BSVs to "MAN" and closed: ___ BSV-3 ___ BSV-4 3 Select BSVs to "AUTO": ___ BSV-3 ___ BSV-4

7. 027AA2.11 001/1/1/SRO#7/C/A 4.0/4.1/NEW/S/CR03701/
The following plant conditions exist:

- The plant is operating at 70% power.
- RCP-1C was secured due to abnormal vibrations and RCS pressure control has been selected to the 'B' Loop.
- The spray valve has failed open and has been manually isolated.
- RCS pressure is currently stable at:
  A Loop: 2050 psig
  B Loop: 2075 psig

The departure from nucleate boiling ratio (DNBR) is ___(1)___ now than before the spray valve malfunction and the LCO for DNB is ___(2)___.

A.✓ (1) lower
(2) not met because 'A' Loop pressure is less than the limit

B. (1) lower
(2) met because 'B' Loop pressure is greater than the limit

C. (1) higher
(2) not met because 'A' Loop pressure is less than the limit

D. (1) higher
(2) met because 'B' Loop pressure is greater than the limit

Reasons:

DNBR= heat flux necessary to cause DNB / actual heat flux

A. Correct. With a lower RCS pressure, the margin to DNB has been reduced. This leads to a lower DNBR. Per LCO 3.4.1 and the COLR, the DNB pressure limit is 2064 psig. SR 3.4.1.1 has a note that requires the loop with two RCPs in operation be used to determine if the limit is met. Since B Loop has only 1 RCP in operation, A Loop pressure would be used.

B. This answer would be correct if the A Loop only had 1 RCP in service and the B Loop had 2.

C. Plausible if the candidate misunderstands the definition of DNBR.

D. Plausible if the candidate misunderstands the definition of DNBR.

OPS 5-01, Obj. 12; ITS LCO 3.4.1 and LCO 3.4.1 Basis; COLR; 10 CFR 55.43.b.2

SRO - New

Reference(s) provided: None

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for loop pressure, hot leg temperature, and RCS total flow rate shall be within limits for the number of reactor coolant pumps (RCPs) in operation.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	----------NOTE---------- With three RCPs operating, the limit is applied to the loop with two RCPs in operation. ---------- Verify RCS loop pressure meets the RCS loop pressure limits specified in the COLR.	12 hours
SR 3.4.1.2	----------NOTE---------- With three RCPs operating, the limit is applied to the loop with two RCPs in operation. ---------- Verify RCS hot leg temperature meets the RCS hot leg temperature limit specified in the COLR, AND is ≤ 605.8°F.	12 hours
SR 3.4.1.3	Verify RCS total flow rate meets the RCS total flow rate limits specified in the COLR, AND is ≥ 133.5 E6 lb/hr with four RCPs operating or ≥ 99.7 E6 lb/hr with three RCPs operating.	12 hours
SR 3.4.1.4	----------NOTE---------- Only required to be performed when stable thermal conditions are established > 90% of ALLOWABLE THERMAL POWER. ---------- Verify RCS total flow rate is within limit by measurement.	24 months

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within the limits assumed in the safety analyses. The analyses of normal operating conditions, including anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on DNB related parameters ensure that these parameters will not be less conservative than were assumed in the analyses and thereby provide assurance that the minimum departure from nucleate boiling ratio (DNBR) will meet the required criteria for each of the transients analyzed.

The RCS pressure limit is consistent with operation within the nominal operating envelope and is above the value used as the initial pressure in the analyses. A pressure greater than the minimum specified will produce a higher DNBR. A pressure lower than the minimum specified will cause the plant to approach the DNB limit.

The RCS coolant hot leg temperature limit is consistent with full power operation within the nominal operating envelope and is lower than the initial hot leg temperature in the analyses. A hot leg temperature lower than that specified will produce a higher DNBR. A temperature higher than that specified will cause the plant to approach the DNB limit.

The RCS flow rate is a function of the number of RCPs in operation and is not expected to vary during operation. The minimum RCS flow rate limit corresponds to that assumed for the DNBR analyses. A higher RCS flow rate will produce a higher DNBR. A lower RCS flow will cause the plant to approach the DNB limit.

APPLICABLE SAFETY ANALYSES

The requirements of LCO 3.4.1 represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown

(continued)

Reactor Coolant System DNB Pressure Limits

RCS loop pressure ≥ 2064 psig

(Assumes 20% tube plugging and bounds either four or three RCPs operating).

These limits are referred to by SR 3.4.1.1

8. 029G2.1.23 001/2/2/SRO#8/C/A 3.9/4.0/BANK/S/CR03701/007-003
Which ONE of the following is the required post-LOCA hydrogen purge flow rate?

A.✓ Continuous purge is to be started 1350 hours after a LOCA. The required purge flow rate is 14.5 scfm.

B. Continuous purge is to be started 1350 hours after a LOCA. The required purge flow rate is 17.0 scfm.

C. Intermittent purge is in progress. Initial purge flow rate was 20 scfm, the second start of the purge flow rate is 14.5 scfm.

D. Intermittent purge is in progress. Initial purge flow rate was 20 scfm, the second start of the purge flow rate is 17.0 scfm.

Reasons:

A. Correct. This is the value of the required purge flowrate.

B. This is the error corrected value for the purge. This would represent the "actual" flow rate with the required flow rate in progress.

C. and D. If the purge was previously performed then the purge flow from the first purge should be used (20 scfm).

OPS 5-121, Obj. 2; EM-225A; 10 CFR 55.43.b.5

SRO - Bank

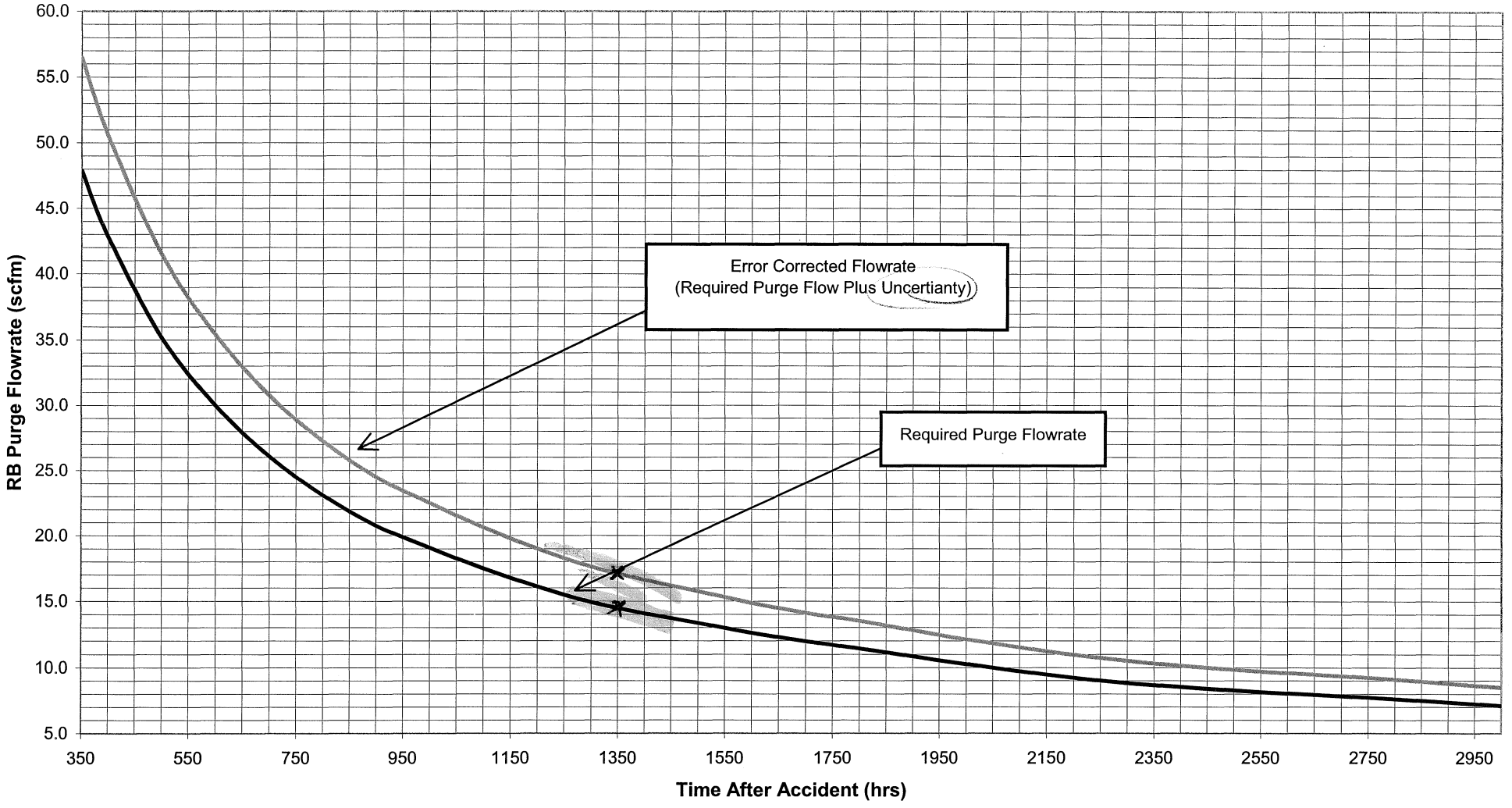
Reference(s) provided: EM-225A

PURGING RB

STATUS
• RB Purge Is Required

		ACTIONS	DETAILS
1.1	____	Ensure Enclosure 2, 3, 4, and 5 of this procedure have been completed (AAT).	• ___ Enclosure 2 complete • ___ Enclosure 3 complete • ___ Enclosure 4 complete • ___ Enclosure 5 complete
1.2	✓	Determine required purge flow rate (AAT/DAT).	• N/A IF H_2 purge has been previously performed, THEN use flows from previous purge. • ✓ IF H_2 purge has NOT been previously performed, THEN refer to Enclosure 10 to determine flows: ✓ Required Purge Flow 14.5 scfm ✓ Error Corrected Flow 17.0 scfm • ✓ Record Error Corrected Flow on Enclosure 6.
1.3	____	Consult Radiation Monitoring Team to determine routes and precautions to be used while performing RB Pressurization (ERT).	• Refer to Enclosure 9 for locations of required actions/components and suggested routes.
1.4	____	WHEN Enclosure 6, Purge Release Authorization Form is complete and approved by the EC, THEN continue with this enclosure.	

CONTINUOUS PURGE FLOW RATES AFTER A LOCA



9. 034A4.02 001/2/2/SRO#9/MEM 3.5/3.9/BANK/S/CR03701/ROT-5-001-074
The plant is in Mode 6 with fuel handling in progress when the Nuclear Operator notices that the audible Source Range Monitor in the Control Room is inoperable. Investigation reveals audible indication is operable in containment. Which of the following describes the required action(s) for this condition?

A. Core alteration may continue as long as both visual indications are still available.

B.✔ Immediately suspend all core alterations and positive reactivity changes.

C. Core alteration may continue as long as audible indication is still available in the containment.

D. Immediately suspend fuel movement but rod shuffle activities may continue.

Reasons:

A. TS 3.9.2 basis states that the audible indication is required in both the control room and the containment.

B. Correct. This condition represents non-compliance with TS 3.9.2. Condition A must be entered. The required actions are to immediately suspend all core alterations or positive reactivity changes.

C. TS 3.9.2 basis states that the audible indication is required in both the control room and the containment.

D. Rod shuffle is a positive reactivity change.

OPS 5-01, Obj. 15; ITS 3.9.2; ITS 3.9.2 Basis; FP-203 Step 3.2.1.1; 10 CFR 55.43.b.2 & 7

SRO - Bank

Reference(s) provided: None

3.9 REFUELING OPERATIONS

3.9.2 Nuclear Instrumentation

LCO 3.9.2 Two source range neutron flux monitors shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required source range neutron flux monitor inoperable.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend positive reactivity additions.	Immediately
B. Two required source range neutron flux monitors inoperable.	B.1 Initiate action to restore one source range neutron flux monitor to OPERABLE status.	Immediately
	AND	
	B.2 Perform SR 3.9.1.1.	4 hours AND Once per 12 hours thereafter

BASES

LCO

This LCO requires two source range neutron flux monitors OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. This LCO is met by use of either the BF3 detectors, or the post-accident monitoring wide range neutron flux instrumentation, since they have been shown to be functionally equivalent. The use of portable detectors is also permitted for purposes of complying with this LCO. If used, portable detectors should be functionally equivalent to the installed source range monitors and satisfy the applicable Surveillance Requirements.

Audible indication from one of the neutron monitors is required in both the control room and containment in order to alert personnel to changes in count rate.

APPLICABILITY

In MODE 6, the source range neutron flux monitors are required to be OPERABLE to determine changes in core reactivity. There is no other direct means available to check core reactivity levels. In portions of MODE 2, MODES 3, 4 and 5, the monitors are required to be OPERABLE by LCO 3.3.9, "Source Range Neutron Flux."

In MODE 1, the neutron flux level is above the indicated range of the monitors, the BF3 monitors are de-energized and no longer relied upon for reactivity and power level monitoring. Thus, there are no requirements on source range neutron flux monitors in MODE 1.

ACTIONS

A.1 and A.2

With only one source range neutron flux monitor OPERABLE, redundant neutron count rate indication has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position, but it does preclude choosing a position that would result in a positive reactivity addition to the core.

(continued)

3.2 LIMITS AND PRECAUTIONS

3.2.1 No operation which may increase core reactivity shall be made unless the conditions of this Section are met.

IF any conditions of this section are not met and fuel handling is underway, THEN fuel handling shall cease immediately and action shall be initiated to satisfy the conditions.

NOTE

Audible counts are not required at the tagboard.

NOTE

The audible count rate amplifier must be located in a manner so as to prevent inadvertent control knob adjustment during refueling operations. [NOCS 000796]

3.2.1.1 During core alterations, at least two (2) source range monitors shall be operable, each with visual indication in the Control Room and one with audible indication in both the Control Room and Containment (Refueling Floor Area). (ITS LCO 3.9.2)

3.2.1.1.1 IF the neutron count rate from two (2) neutron monitors unexpectedly increases by more C_H (1.5 times the base countrate (C_O)) during any single fuel assembly/component step,
THEN

1. Withdraw the fuel assembly in question and
2. Immediately cease all other core alterations and
3. Perform Step 4.2.4 of Section 4.2, Subcritical Multiplication, and
4. Obtain a boron analysis of the reactor coolant and record on Enclosure 2.

Core alterations shall not resume until the Reactor Engineer has evaluated the occurrence and the Refueling Supervisor has given permission to resume core alterations.

3.2.1.1.2 IF the count rate on two (2) or more neutron monitors increases to C_H (1.5 times the base countrate (C_O)) while no geometry changes are being made, THEN immediately begin boration of the RCS until the count rate stops increasing. The Reactor Engineer and Refueling Supervisor will investigate the cause of the count rate increase. Core alterations shall not resume until the cause is determined and rectified.

3.2.1.1.3 IF the count rate on two (2) or more neutron monitors decreases to C_L (0.5 times the base countrate (C_O)) or more while no core geometry changes are being made, THEN cease core alterations until the cause of the count rate decrease has been determined and rectified.

10. 036AA2.02 001/1/2/SRO#10/MEM 3.4/4.1/MOD/S/CR03701/
Fuel movement activities in both the RB and SFP are in progress. Determine which ONE of the following conditions meet the criteria for a "Significant Fuel Handling Event" IAW FP-203, Defueling and Refueling Operations?

A. RM-A1 rises to its "Warning" setpoint. Investigation reveals that work on the reactor vessel head caused the rise.

B. The operating Decay Heat Removal train secured to facilitate core mapping operations.

C. Fuel movement is stopped due to reduced pool clarity.

D.✔ Refueling canal water level is discovered to be 155 ft.

Reasons:

Significant fuel handling event – are fuel handling events that have resulted in fission product gas release, loss of required shutdown margin, personnel injury, damage to fuel, control components or fuel handling equipment, violation of Tech Specs associated with fuel movement, or loss of SNM.

A. This would not meet the definition of a "significant" fuel handling event since it did not result in any of the conditions listed above.

B. See above definition.

C. See above definition.

D. Correct. LCO 3.9.6 requires canal level > 156' during fuel handling inside the RB. This would represent a violation of TS associated with fuel handling and meets the definition of a significant fuel handling event.

OPS 4-26, Obj. 6; FP-203, Section 4.4; 10 CFR 55.43.b.6 & 7

History: NRC-LOI0401

SRO - Modified

Reference(s) provided: None

3.9 REFUELING OPERATIONS

3.9.6 Refueling Canal Water Level

LCO 3.9.6 Refueling canal water level shall be maintained ≥ 156 ft Plant Datum.

APPLICABILITY: During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Refueling canal water level not within limit.	A.1	Suspend movement of recently irradiated fuel assemblies within containment.	Immediately
	AND		
	A.2	Initiate action to restore refueling canal water level to within limit.	Immediately

3.1.3.5 Top hat (upper fuel assembly) shoehorn use considerations:

- Ensure the rope to the top hat shoehorns are clear of the location into which fuel is being inserted so that the rope will not be snagged or trapped. This concern also holds for lower shoehorn ropes.
- Two top hat shoehorns are generally used. One on the fuel assembly leaning into the insertion location (hole) and the other on the opposite side, creating a "funnel" effect on those two sides. For severe cases four shoehorns could be used on a fully enclosed hole but in such a case consideration should also be given to removing one or more leaning fuel assemblies.
- After partial insertion of the fuel assembly (prior to use of JOG switch after second slow zone indication), the top hat shoehorns shall be removed. The inner fuel mast is wider than a fuel assembly and can become wedged between the top hat shoehorns. AR 108101 documents such an occurrence.

3.1.4 Depressions Resembling Debris

3.1.4.1 There are two depressions on the upper core support assembly flange that resemble real debris.

3.1.4.2 On the North, near the X axis, is a U shaped depression towards the outer wall of the flange.

3.1.4.3 On the South, near the Z axis, is an impression that resembles a large bolt, complete with threads. This depression is towards the interior vessel side of the flange.

3.1.5 Definitions

3.1.5.1 **Fuel Handling Accident** – for purposes of notifying the control room to enter CP-341, a fuel handling accident is an event that damages fuel and results in release of fuel pin gas gap activity. An event resulting in actual or possible fuel damage but with no indication of the release of gas gap activity (such as rad monitor alarms, bubbles, etc) is a significant fuel handling event, but not a fuel handling accident.

3.1.5.2 **Recently irradiated fuel** – fuel that has occupied part of a critical reactor core within the previous 72 hours.

3.1.5.3 **Significant fuel handling event** – are fuel handling events that have resulted in fission product gas release, loss of required shutdown margin, personnel injury, damage to fuel, control components or fuel handling equipment, violation of Technical Specifications associated with fuel movement, or loss of SNM. See Step 4.4.1 for further detail.

4.4 RESPONSE TO FUEL HANDLING EVENTS

4.4.1 Significant Fuel Handling Event are when a refueling activity has resulted in the following:

- Fission product gas release
- Loss of required shutdown margin
- Personnel injury
- Damage to fuel, control components or fuel handling equipment
- Violation of Tech. Specs. associated with fuel movement
- Loss of SNM (Special Nuclear Material)

The following are examples of Fuel Handling Events to provide guidance on what type of refueling events require compliance with this section.

Examples of Fuel Handling Event include, but are not limited to:

- Setting one fuel assembly on another
- Attempting to place a rod in an assembly that already contains a rod
- Attempting to withdraw a rod or a fuel assembly from a location and discovering there is not one
- Movesheet errors that could have resulted in misplaced fuel or rods, or in unplanned uncoupling of fuel assemblies from the rest of the core
- Unexpected count rate doubling in the core
- The sounding of a radiation monitor alarm associated with fuel handling
- Fire on or near the fuel handling equipment.
- Improper operation of fuel handling equipment components provided for fuel handling equipment or fuel/component protection for which no redundant backup component is available by design (failure to stop, loss of slow zones, etc.)
- Loss of required water level or boron concentration
- Loss of communications
- Misplacement of a fuel assembly or rod such that its location is not known.

4.4.2 Upon the report of a Fuel Handling Event in either the SF pool or the RB, the Refueling Supervisor shall order all fuel moves to cease.

4.4.2.1 <u>IF</u> the event is a Fuel Handling Accident,
<u>THEN</u> IMMEDIATELY NOTIFY the CRS/SSO.

4.4.3 The Refueling Supervisor shall perform an investigation. This may be as short as a few minutes for a simplistic, obvious problem to hours or even days for a more complex event.

QUESTION PRIOR TO MODIFICATION

036AA2.02 001/1/2/SR0 #9/MEM 3.4/4.1/NEW/S/CR03501/

Fuel movement activities are in progress. Determine which ONE of the following conditions meet the criteria for a "Significant Fuel Handling Event" IAW FP-203, Defueling and Refueling Operations?

A.✔ Movesheet error discovered that would have resulted in setting one fuel assembly on top of another.

B. Securing the operating Decay Heat Removal train to facilitate core mapping operations.

C. Fuel movement is stopped due to decreasing pool clarity.

D. RM-A1 increase to its "Warning" setpoint. Investigation reveals that work on the reactor vessel head caused the increase.

Reasons:

Significant fuel handling event – fuel handling events are events that have, or could have, resulted in fission product gas release, loss of required shutdown margin, personnel injury, damage to fuel, control components or fuel handling equipment, violation of Tech Specs associated with fuel movement, or loss of SNM.

A. Correct. Example provided in FP-203, Step 4.4.1.

B. See above definition.

C. See above definition.

D. This would be correct if fuel handling activities caused this alarm.

OPS 4-26, Obj. 6; FP-203, Section 4.4; 10 CFR 55.43.b.6 & 7

SRO - New

Reference(s) provided: None

11. 054AA2.04 001/1/1/SRO#11/C/A 4.2/4.3/NEW/S/CR03701/
The following plant conditions exist:

- A normal plant shutdown is in progress.
- All 4 of the "<725 PSI GEN PRESS EFIC ACT BYPASS" pushbuttons on the PSA panel were depressed and are back lighted.
- RCS temperature is 500°F.

The only operating Main Feed Pump trips concurrent with a steam leak on the 'B' OTSG. Several minutes into the transient, the following parameters are noted (assume **NO** manual actions have been taken):

	'A' OTSG	'B' OTSG
Level	24" and stable	5" and lowering
Pressure	550 psig	100 psig
EFW Flow from EFP-3	100 gpm	0 gpm
EFW Flow from EFP-2	100 gpm	0 gpm

Given the above conditions, what equipment operability concerns exist?

A. EFIC should not have actuated for the given conditions. The AMSAC system should be declared inoperable.

B. 'A' OTSG level is too low for the given conditions. The EFIC level control modules should be declared inoperable.

C.✓ 'B' OTSG should have EFW flow. The < 725 EFIC Bypass circuitry should be declared inoperable.

D. 'A' OTSG should not have EFW flow. The EFIC FOGG circuitry should be declared inoperable.

Reasons:

A. EFW would have actuated on the loss of both MFW Pumps (if the RPS is not in shutdown bypass) or on OTSG low level.

B. With the given initial plant conditions, RCPs would still be running. As a result, the EFIC system would still be controlling on low level limits. This corresponds to a level of 24".

C. Correct. With the 725 PSI bypasses actuated, the 'B' OTSG should still have EFW flow. While the correct action would be to secure feeding the generator, no manual actions have been taken yet.

D. The FOGG circuitry should not isolate EFW to both OTSGs. In this case, if the 725 PSI bypasses were not actuated the 'B' OTSG would isolate, but the 'A' OTSG would still have EFW flow, even with pressure < 600 psig.

OPS 4-31, Obj. 2; OPS 4-31 Section 1-4.0.G; ITS 3.3.11 Basis; 10 CFR 55.43.b.2

SRO - New

Reference(s) provided: None

BASES

LCO

OTSG Pressure-Low (continued)

transient conditions not requiring secondary system isolation, and has been shown to be an appropriate indicator of secondary side breaks for ensuring automatic EFW actuation. The OTSG Pressure-Low Function includes a bypass enable and removal function. The bypass removal Allowable Value is chosen to allow sufficient operating margin (time) for the operator to bypass the actuation during plant cooldown prior to reaching the actuation setpoint. The 750 psig setpoint allows at least a 10 minute window to perform the bypass assuming the maximum allowed cooldown rate and instrument error.

OTSG Differential Pressure-High

Four EFIC channels for OTSG differential pressure shall be OPERABLE with an allowable value of ≤ 125 psid. The actual plant setpoint is set lower than 125 psid to account for instrument loop uncertainties and setability. The setpoint ensures that automatic EFW isolation to a depressurized OTSG occurs for the range of sizes of SLBs or feedwater line breaks that require rapid actuation early in the event. The setpoint has also been chosen to avoid spurious isolation of EFW during conditions due to relatively small deviations in OTSG pressures that can be caused by primary system conditions. The OTSG Differential Pressure-High Function is bypassed when the OTSG Pressure-Low Function is bypassed.

RCP Status

Four EFIC channels for RCP status are required to be OPERABLE to ensure that upon the loss of all four RCPs, EFW will be automatically initiated. Additionally, EFW will automatically raise and control level to natural circulation value providing a higher driving head for establishing and maintaining natural circulation conditions when forced RCS flow is lost. No setpoint is specified since the status indication used by EFIC is binary in nature. The RCP Status Function includes a bypass enable and removal function from the RPS. The Allowable Value for the bypass removal is set high enough to avoid spurious actuations during low power operation.

RCP status is monitored by two RCPPMs for each RCP. This safety function is lost if both RCPPMs for a single RCP are inoperable and neither is tripped.

(continued)

BASES

APPLICABILITY

EFIC instrumentation OPERABILITY requirements are applicable during the MODES and specified conditions listed in Table 3.3.11-1. Each Function has its own requirements based on the specific accidents and conditions for which it is designed to provide protection.

The initiation of EFW on the Loss of MFW Pumps is applicable in MODE 1 and in MODES 2 and 3 when not in shutdown bypass. Below these plant conditions, EFW initiation on low OTSG level occurs fast enough to prevent primary system overheating.

EFW Initiation on low OTSG level shall be OPERABLE at all times the OTSG is required for heat removal. These conditions include MODES 1, 2, and 3. To avoid automatic actuation of the EFW pumps during heatup and cooldown, the low OTSG pressure Function can be bypassed at or below a secondary pressure of 750 psig. This secondary-side pressure occurs during MODE 3 operation.

EFW initiation on loss of all RCPs is required to be OPERABLE at ≥ 10% RTP. This power level coincides with the bypass permissive signal provided by RPS.

The MFW, Main Steam Line Isolation, and EFW Vector Valve Control Functions shall be OPERABLE in MODES 1, 2, and 3 with OTSG pressure ≥ 750 psig because OTSG inventory can be high enough to contribute significantly to the peak pressure following a secondary side break. Both the normal feedwater and the EFW must be isolatable on each OTSG to limit overcooling of the primary and mass and energy releases to the RB. Once either OTSG pressure decreases below 750 psig, the Main Steam Line and MFW Isolation Functions can be bypassed to prevent actuation during cooldown. The EFW Vector Valve Control logic will not perform any function when both OTSG pressures are low; thus, the logic is also bypassed at the same time the OTSG pressure low Functions is bypassed. In MODES 4, 5, and 6, primary and secondary side energy levels are reduced and the feedwater flow rate is low or nonexistent. Because of this, EFIC instrumentation is not required to be OPERABLE in these MODES.

(continued)

BASES

BACKGROUND (continued)

The valve open/close commands are determined by the relative values of OTSG pressures as follows:

PRESSURE STATUS	VECTOR VALVES "A"	VECTOR VALVES "B"
If OTSG "A" & OTSG "B" > 600 psig	Open	Open
If OTSG "A" > 600 psig & OTSG "B" < 600 psig	Open	Close
If OTSG "A" < 600 psig & OTSG "B" > 600 psig	Close	Open
If OTSG "A" & OTSG "B" < 600 psig		
AND		
OTSG "A" & OTSG "B" within 125 psid	Open	Open
OTSG "A" 125 psid > OTSG "B"	Open	Close
OTSG "B" 125 psid > OTSG "A"	Close	Open

APPLICABLE SAFETY ANALYSES

The SLB analysis was re-performed crediting all "as built" systems and their associated response times. This is documented in FTI Summary Analysis Report 86-266223-00, "CR-3 MSLB with MFP Trip Failure." In this analysis, EFIC isolation of Main Feedwater and Main Steam were credited. The Vector Valve Logic, Feed Only Good Generator (FOGG) was credited for termination of an overfeed condition for either OTSG during postulated DC power failures. No operator action was credited or required.

EFW vector valve logic response time is included in the response time for each EFW instrumentation Function and is not specified separately.

(continued)

G. FEED ONLY GOOD GENERATOR (FOGG) LOGIC

1. When the vector logic is enabled and no EFW overfill signal is present, the valve open/close commands are determined by the relative values of OTSG pressures as follows:

Pressure Status*	OTSG A Valves Command	OTSG B Valves Command
OTSG A and B > 600 psig	OPEN	OPEN
OTSG A > 600 psig and OTSG B < 600 psig	OPEN	CLOSE
OTSG A < 600 psig and OTSG B > 600 psig	CLOSE	OPEN
OTSG A and B < 600 psig AND OTSG A and B within125 psig	OPEN	OPEN
OTSG A and B < 600 psig AND OTSG A 125 psig > OTSG B	OPEN	CLOSE
OTSG A and B < 600 psig AND OTSG B 125 psig > OTSG A	CLOSE	OPEN

* Nominal values. Actual isolation setpoint is 624.5 psig. Actual ΔP setpoint is 94.5 psig.

2. The alarm status panels located in the top of each EFIC cabinet include lights that indicate the Open/Close status of the respective channel vector valves (Figure 25).

3. FOGG logic is bypassed when the low OTSG pressure bypass is selected (see the Shutdown Bypass discussion below).

H. MANUAL OPERATION (Figure 10)

1. After an EFW initiation, the operator may regain control of the pumps and valves in one of two ways, depending on the status of the initiating signal.

   a) If initiating condition still exists, the operator must take MANUAL control of the EFW system.

12. 055EA2.02 001/1/1/SRO#12/C/A 4.4/4.6/NEW/S/CR03701/
Given the following conditions:

- Reactor power was at 100%.
- A station blackout has occurred.
- Reactor Coolant Pressure is 1500 psig.
- Reactor Coolant Temperature is 600°F.

Which ONE of the following choices represents the EOP-13 Rules in effect and the appropriate actions to be taken?

A. Rule 1. Maintain RCS temperature as stable as possible to minimize further RCS shrinkage.

B. Rule 1. Initiate maximum possible cooldown to allow CFT makeup to the RCS.

C. Rule 3. Maintain RCS temperature as stable as possible to minimize further RCS shrinkage.

D.✓ Rule 3. Initiate maximum possible cooldown to allow CFT makeup to the RCS.

Reasons:

A. Plausible since Rule 1 would normally be performed for a loss of SCM. Also plausible since reducing RCS temperature will cause the RCS to 'shrink' which would cause further pressure loss.

B. Plausible since Rule 1 would normally be performed for a loss of SCM.

C. Plausible since reducing RCS temperature will cause the RCS to 'shrink' which would cause further pressure loss.

D. Correct. While in EOP-12 no other rules or EOPs should be performed unless directed by EOP-12. EOP-12 does direct Rule 3, but does not direct Rule 1. With adequate SCM lost, EOP-12 directs a maximum possible cooldown. Per the EOP-12 basis, this is consistent with the SBLOCA guidance with HPI not available to allow for CFT injection.

OPS 5-100, Obj. 4; EOP-12 Step 3.20; EOP-12 Note prior to Step 3.1; EOP-TBD cross-step document for EOP-12 Step 3.20; 10 CFR 55.43.b.5

SRO - New

Reference(s) provided: Steam Tables

EOP-TBD Cross-Step Document
EOP-12, STATION BLACKOUT

EOP Step: 3.20
TBD Step: None **Section:** None **Deviation Type:** Add
Generic Deviation: None
Justification:

This guidance is directed by the TBD Volume 3, Part III.A.3.3, Part II.D.2.3.2 and Part IV Chapter H Section 3.0. The SBO analysis assumed a 111 gpm leakage rate (ITS + 25 gpm/RCP seal) and showed that without a cooldown the core remains covered until makeup can be restored at the end of the SBO coping period. The analysis assumptions are very conservative since actual seal testing, which simulated SBO conditions, resulted in no increase in leakage.

The guidance to commence a cooldown , if SCM is lost, is based on providing mitigation guidance for any size SBLOCA with HPI not available. That is, this guidance addresses any size SBLOCA, i.e., not just the 87-00 assumptions, and is intended to reduce RCS pressure such that the core flood system provides makeup to the RCS. FTI has performed analysis indicating that adequate core cooling is maintained during the cooldown.

This step provides actions for loss of adequate SCM during an SBO. The instructions in this step differ from the strategy used in the NUMARC 87-00 analysis. That analysis assumed a specific leak size (111 gpm). While the SBO analysis shows that the core will not be uncovered before HPI is restored for the specific size leak, it does not address other leak sizes.

Step Basis:

To ensure that the core remains covered for the widest possible range of leak sizes, the TBD guidance for loss of adequate SCM with no HPI must be used. This strategy is to cooldown as rapidly as possible using the OTSGs. This results in an RCS depressurization to the point that CFTs inject water into the RCS to replace the inventory being lost out the leak

Rule 3 has the requirement to ensure OTSG levels are rising to the ISCM setpoint. The details of this step depress the inadequate subcooling margin pushbutton for EFIC channel A and B to ensure OTSG level rise to support boiler cooling condenser cooling.

Setpoint Code: None **Setpoint Value**: None
References:

1. B&W letter to FPC, FPC-92-075, 51-1212183-00
2. NUMARC 87-00, Guidelines and Technical Bases for NUMARC initiatives Addressing Station Blackout at Light Water Reactors, section 4.2.1 guideline #4.
3. EDBD 6.1, Reactor Coolant System
4. NOCS 001611
5. NOCS 002550

3.0 FOLLOW-UP ACTIONS

ACTIONS	DETAILS
NOTE **Do not perform other EOPs, EOP Rules or APs except for AP-880, unless directed by this procedure or TSC.**	
3.1 ___ Verify Rx is shut down.	• ___ Verify Rx is shutdown using edgewise meters.
___ IF Rx is NOT shut down, THEN trip Rx.	1 ___ Depress Rx trip push button. 2 ___ Verify CRD groups 1 through 7 are fully inserted. 3 ___ Verify Rx is shutdown using edgewise meters. 4 ___ Depress Main Turbine trip push button. 5 ___ Remain in this procedure.

3.0 FOLLOW-UP ACTIONS (CONT'D)

ACTIONS	DETAILS
3.20 ___ IF at any time, adequate SCM is lost, THEN establish and maintain maximum possible cooldown rate.	• ___ Depress "ISCM" push buttons for EFIC channels A and B. • Open ADVs: ___ MSV-25 ___ MSV-26 • ___ Establish maximum EFW flow (2 EFPs if available), OR maximum AFW flow. • ___ Lowering OTSG level is acceptable. [Rule 3, EFW/AFW Control] • ___ Remain in this procedure.
3.21 ___ IF at any time, adequate SCM is lost, AND RCS PRESS is > 2400 psig, THEN reduce RCS PRESS.	• Cycle PORV to reduce RCS PRESS to 1600 psig.

13. 062A2.16 001/2/1/SRO#13/C/A 2.5/2.9/NEW/S/CR03701/
The plant is operating at 100% power when Q-04-02 "4KV ES Bus Degraded Volt Trip" alarm is received.

Assuming 2 of the A ES Bus degraded voltage (SLUR) relays have actuated, the 'A' EDG:

A. should start. Only AP-770 "Emergency Diesel Generator Actuation" should be entered.

B. should start. Both AP-770 "Emergency Diesel Generator Actuation" and AP-730 "Grid Instability" should be entered.

C. should not start. Both AP-770 "Emergency Diesel Generator Actuation" and AP-730 "Grid Instability" should be entered.

D.✔ should not start. Only AP-730 "Grid Instability" should be entered.

Reasons:

A. The SLUR relays require 3 out of 3 to cause a diesel start. Plausible since FLUR relays require 2 out of 3 logic. AP-730 would be applicable but AP-770 would not.

B. The SLUR relays require 3 out of 3 to cause a diesel start. Plausible since FLUR relays require 2 out of 3 logic.

C. An actual undervoltage (FLUR) has not occurred. AP-770 should not be entered.

D. Correct. The degraded voltage relays (SLUR) require 3 out of 3 logic to start a diesel. AP-730 would be entered since bus voltage is < 4150V (SLUR setpoint is 3952V. An actual undervoltage condition has not occurred (FLUR) so AP-770 should not be entered.

OPS 4-90, Obj. 4; OPS 5-30, Obj. 1; OPS 5-126, Obj. 1; AP-730 Step 1.0; AP-770 Step 1.0; AR-702 EP 1747; 10 CFR 55.43.b.5

SRO - New

Reference(s) provided: None

EMERGENCY DIESEL GENERATOR ACTUATION

1.0 ENTRY CONDITIONS

IF any ES 4160V bus UV occurs,

THEN use this procedure.

2.0 IMMEDIATE ACTIONS

NOTE
There are no immediate actions for this procedure.

GRID INSTABILITY

1.0 ENTRY CONDITIONS

IF any ES 4160V bus is aligned to offsite power,
AND any of the following exists:

- ECC notification of 230KV voltage ≤ 234.0KV
- ECC notification of existing or imminent grid instability
- ES 4160V bus voltage < 4.15KV OR > 4.40KV
- Unstable grid frequency

THEN use this procedure.

2.0 IMMEDIATE ACTIONS

NOTE
There are no immediate actions for this procedure.

to annunciate (27-Z relay, "Potential Transformer Trouble").

3) At normal bus voltage the LH contacts will be closed. On a loss of voltage the relays will rotate and close the RH contacts.

4) Opening the AY knife switch during the performance of AP-770 will de-energize the control circuit above.

b) The 27 undervoltage relays (Figure 14) physically take ≈7.8 seconds to actuate, in either direction, assuming a total loss or full recovery of voltage. The time required to actuate may be longer if the voltage does not immediately drop all the way to zero or recover to nominal voltage.

1) When the EDG reaches rated speed (frequency) and voltage, and three seconds after the bus supply breakers are stripped, the EDG output breaker will close.

3. Degraded Voltage Protection (SLUR) (Figures 15 and 16)

a) The 4160V ES buses are also provided with Second Level Undervoltage Relaying (SLUR) protection, or "degraded undervoltage relaying", designed to protect the 4160V ES buses from sustained periods of low voltage.

b) Degraded voltage relaying will actuate if bus voltage drops to approximately 3952 volts or less for greater than 5 seconds, as sensed by 3 out of 3 degraded voltage relays.

1) Once actuated, a start command is sent to the affected emergency diesel generator. If the degraded voltage condition persists for an additional 13 seconds (total of 18 seconds), the bus is stripped (27BY), and a three second time delay is started to allow time to strip the bus before closing the diesel breaker (27BZ).

2) Total time to breaker closure is 21 seconds from the time the degraded condition was sensed.

SSF ANNUNCIATOR RESPONSE	SF-A2-04-02	Q-04-02

4KV ES BUS DEGRADED VOLT TRIP

EVENT POINT 1747

INDICATED CONDITION:

- "A" ES 4160V BUS VOLTAGE **<3952 VOLTS** FOR **>5 SECONDS**.

REDUNDANT INDICATION WHICH WILL VERIFY ALARM:

- "A" ES 4160V BUS VOLTAGE METER INDICATING < 3952 VAC.
- "A" ES 4160V BUS VOLTAGE ON MT-190-EIR INDICATING < 3952 VAC.

OPERATOR ACTIONS FOR A VALID ALARM:

- REFER TO AP-770.

DISCUSSION:

- THIS ALARM WILL OCCUR WHEN ANY OF THREE VOLTAGE SENSE RELAYS DETECT LESS THAN 3952 VOLTS FOR GREATER THAN 5 SECONDS. A DIESEL GENERATOR START SIGNAL IS INITIATED IF ALL THREE VOLTAGE RELAYS SENSE THIS CONDITION.

REFERENCES: DRAWING 208-040 MT-131

SENSING ELEMENT: 27B-A/32EA, 27B-B/32EA, 27B-C/32EA

14. 064G2.1.22 001/2/1/SRO#14/C/A 2.8/3.3/BANK/S/CR03701/4-006-013
The following plant conditions exist:

- 'A' Decay Heat train is in service.
- RCS pressure is 300 psig.
- RCS temperature is 190° F.
- The diesel fuel storage tank readings are as follows:

EDG "A" - 9' 8"
EDG "B" - 7' 1"

Which of the following describes required Technical Specification action(s), if any, for this situation?

A.✓ No actions required.

B. Restore fuel oil to within limits in 48 hours.

C. Immediately declare the "B" EDG inoperable and restore to operable status within 72 hours.

D. Immediately declare the "A" & "B" EDG inoperable and verify combined stored fuel oil level > 45,834 gallons within 1 hour.

Reasons:

A. Correct. With the given conditions the plant is in Mode 5. In Mode 5 only 1 EDG is required to be operable. While the "B" EDG would be inoperable for the given tank level, no actions are required with "A" still operable.

B, C. & D. These choices could be selected if the student determined the incorrect plant Mode and/or misapplied LCO 3.8.3.

OPS 4-06, Obj. 1-10; ITS 3.8.2 & 3.8.3; OP-103F Figure 8; 10 CFR 55.43.b.2

SRO - Bank

Reference(s) provided: ITS 3.8.3, OP-103F Figure 8

DIESEL GENERATOR FUEL OIL STORAGE TANKS (DFT-1A/1B)

INDICATED LEVELS ARE NOT ERROR CORRECTED UNLESS OTHERWISE NOTED

Tank Level Dip Stick (ft)	(in)	Tank Volume (gal)	Usable Volume (gal)	Remarks
6'	6 1/2"	19,678	17,660	
6'	6 3/4"	19,752	17,733	
6'	7 "	19,825	17,807	
6'	7 1/4"	19,899	17,880	
6'	7 1/2"	19,972	17,954	
6'	7 3/4"	20,045	18,027	
6'	8 "	20,119	18,100	
6'	8 1/4"	20,192	18,173	
6'	8 1/2"	20,265	18,246	
6'	8 3/4"	20,338	18,319	
6'	9 "	20,410	18,392	
6'	9 1/4"	20,483	18,465	
6'	9 1/2"	20,556	18,537	
6'	9 3/4"	20,628	18,610	
6'	10 "	20,701	18,682	
6'	10 1/4"	20,773	18,755	
6'	10 1/2"	20,846	18,827	
6'	10 3/4"	20,918	18,899	
6'	11 "	20,990	18,971	
6'	11 1/4"	21,062	19,043	
6'	11 1/2"	21,134	19,115	
6'	11 3/4"	21,205	19,187	
7'	0 "	21,277	19,258	
7'	0 1/4"	21,349	19,330	
7'	0 1/2"	21,420	19,401	
7'	0 3/4"	21,491	19,473	
7'	1 "	21,563	19,544	
7'	1 1/4"	21,634	19,615	
7'	1 1/2"	21,705	19,686	
7'	1 3/4"	21,776	19,757	
7'	2 "	21,846	19,828	
7'	2 1/4"	21,917	19,898	
7'	2 1/2"	21,987	19,969	
7'	2 3/4"	22,058	20,039	
7'	3 "	22,128	20,110	
7'	3 1/4"	22,198	20,180	
7'	3 1/2"	22,268	20,250	
7'	3 3/4"	22,338	20,320	
7'	4 "	22,408	20,389	
7'	4 1/4"	22,478	20,459	

DIESEL GENERATOR FUEL OIL STORAGE TANKS (DFT-1A/1B)

INDICATED LEVELS ARE NOT ERROR CORRECTED UNLESS OTHERWISE NOTED

Tank Level Dip Stick (ft)	(in)	Tank Volume (gal)	Usable Volume (gal)	Remarks
9'	0 1/2"	27,520	25,501	
9'	0 3/4"	27,572	25,554	
9'	1 "	27,624	25,606	
9'	1 1/4"	27,676	25,658	
9'	1 1/2"	27,727	25,709	
9'	1 3/4"	27,779	25,760	
9'	2 "	27,829	25,811	
9'	2 1/4"	27,880	25,861	
9'	2 1/2"	27,930	25,911	
9'	2 3/4"	27,979	25,961	
9'	3 "	28,029	26,010	
9'	3 1/4"	28,078	26,059	
9'	3 1/2"	28,126	26,108	
9'	3 3/4"	28,175	26,156	
9'	4 "	28,223	26,204	
9'	4 1/4"	28,270	26,252	
9'	4 1/2"	28,317	26,299	
9'	4 3/4"	28,364	26,346	
9'	5 "	28,410	26,392	
9'	5 1/4"	28,456	26,438	
9'	5 1/2"	28,502	26,483	
9'	5 3/4"	28,547	26,529	
9'	6 "	28,592	26,573	
9'	6 1/4"	28,636	26,618	
9'	6 1/2"	28,680	26,662	
9'	6 3/4"	28,724	26,705	
9'	7 "	28,767	26,748	
9'	7 1/4"	28,810	26,791	
9'	7 1/2"	28,852	26,833	
9'	7 3/4"	28,893	26,875	
9'	8 "	28,935	26,916	
9'	8 1/4"	28,976	26,957	
9'	8 1/2"	29,016	26,997	
9'	8 3/4"	29,056	27,037	
9'	9 "	29,095	27,077	
9'	9 1/4"	29,134	27,115	
9'	9 1/2"	29,172	27,154	
9'	9 3/4"	29,210	27,192	
9'	10 "	29,247	27,229	
9'	10 1/4"	29,284	27,266	

Table 1.1-1 (page 1 of 1)
MODES [a]

MODES	TITLE	REACTIVITY CONDITION (K_{eff})	% RATED THERMAL POWER [b]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 280
4	Hot Shutdown[c]	< 0.99	NA	280 > T_{avg} > 200
5	Cold Shutdown[c]	< 0.99	NA	≤ 200
6	Refueling[d]	NA	NA	NA

(a) With fuel in the reactor vessel.

(b) Excluding decay heat.

(c) All reactor vessel head closure bolts fully tensioned.

(d) One or more reactor vessel head closure bolts less than fully tensioned.

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air

LCO 3.8.3 The stored diesel fuel oil lube oil, and starting air subsystem shall be within limits for each required emergency diesel generator (EDG).

APPLICABILITY: When associated EDG is required to be OPERABLE.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each EDG.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One EDG with stored fuel oil level < 22,917 gal and > 19,643 gal in storage tank.	A.1 Verify combined stored fuel oil level > 45,834 gal.	1 hour
B. One or more EDGs with stored fuel oil level < 22,917 gal and > 19,643 gal in storage tank. AND Combined stored fuel oil level < 45,834 gal.	B.1 Restore fuel oil level to within limits.	48 hour
C. With stored EDG lube oil inventory < 280 gal and > 240 gal.	C.1 Restore lube oil inventory to within limits.	48 hours OR Declare both EDGs inoperable.

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more EDGs with stored fuel oil total particulates not within limit.	D.1 Restore fuel oil total particulates to within limits.	7 days
E. One or more EDGs with new fuel oil properties not within limits.	E.1 Restore stored fuel oil properties to within limits.	30 days
F. One or more EDGs with starting air receiver pressure < 225 psig and ≥ 160 psig.	F.1 Restore starting air receiver pressure to within limits.	48 hours
G. Required Action and associated Completion Time not met. <u>OR</u> One or more EDGs with diesel fuel oil, lube oil, or starting air subsystem not within limits for reasons other than Condition A, B, C, D, E, or F.	G.1 Declare associated EDG inoperable.	Immediately

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources-Shutdown

LCO 3.8.2 The following AC electrical power sources shall be OPERABLE:

a. One qualified circuit between the offsite transmission network and the onsite Class IE AC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems-Shutdown";-and

b. One emergency diesel generator (EDG) capable of supplying one train of the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10.

APPLICABILITY: MODES 5 and 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required offsite circuit inoperable.	----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.8.10, with one required train de-energized as a result of Condition A. ----------	
	A.1 Declare affected required feature(s) with no offsite power available inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
		(continued)

15. 067AG2.4.41 001/1/2/SRO#15/C/A 2.3/4.1/NEW/S/CR03701/

The plant is operating at 100% power when Alarm Window F-03-04 "Cntrl Complex Fire Alert" is received. Five minutes after receipt of the alarm, Security calls and reports the following:

- A smoldering fire exists in the 'A' Battery Room
- The fire is contained to the 'A' Battery Room.

Which ONE of the following Emergency Classifications, if any, must be entered?

A.✓ No Emergency Classification should be entered

B. Unusual Event

C. Alert

D. Site Area Emergency

Reasons:

A. Correct. EAL 2.14 requires that the fire has not been extinguished within 15 minutes. EAL 2.15 is not met if damage is clearly contained and localized to one train and safe shutdown capability exists. The fire is contained to the 'A' Battery Room.

B. Plausible since this would be the appropriate classification if the event continued for 15 minutes. At this time, only 5 minutes has passed.

C. Plausible since an Alert classification exists for a fire. EAL 2.15 is not met if damage is clearly contained and localized to one train and safe shutdown capability exists. The fire is contained to the 'A' Battery Room.

D. Plausible since a Site Area Emergency classification exists for Loss of Vital DC Power. EAL 4.6 requires a loss of both buses and requires the loss to be sustained for > 15 minutes.

OPS 5-34, Obj. 5; EM-202 Enclosure 1; EAL Bases Manual for EALs 2.14, 2.15, 4.6; 10 CFR 55.43.b.5

SRO - New

Reference(s) provided: EM-202 Enclosure 1

EMERGENCY CLASSIFICATION TABLE
ACCIDENT CONDITION:

NATURAL / MANMADE HAZARDS AND EC JUDGEMENT (Continued)

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
Explosions/ Catastrophic Pressurized Equipment Failure **MODES: ALL**	**2.12 MODES: ALL** Report by Plant personnel of VISIBLE DAMAGE to permanent structures or equipment within the PROTECTED AREA due to an EXPLOSION or catastrophic failure of pressurized equipment *Refer to Security Event*	**2.13 MODES: ALL** **(1 or 2)** 1. EXPLOSION or catastrophic failure of pressurized equipment causes significant VISIBLE DAMAGE to any of the following structures: - Auxiliary Building - BWST - Control Complex - Diesel Generator Building - EFT-2 Building, - Intermediate Building - Reactor Building - EFP-3 Building **OR** 2. Indications show degraded SAFE SHUTDOWN EQUIPMENT performance due to the EXPLOSION or pressurized equipment failure	*Refer to Fission Product Barrier Matrix, System Malfunction, or Emergency Coordinator Judgment*	*Refer to Fission Product Barrier Matrix, System Malfunction, or Emergency Coordinator Judgment*
Fire **MODES: ALL**	**2.14 MODES: ALL** **(1 and 2)** 1. FIRE in or threatening one of the following structures: - Auxiliary Building - BWST - Control Complex, - Diesel Generator Building - EFT-2 Building - Intermediate Building - Reactor Building - EFP-3 Building **AND** 2. FIRE not extinguished within 15 minutes from either Control Room notification or receipt of a VALID fire alarm in the Control Room	**2.15 MODES: ALL** **(1 or 2)** 1. Report by Plant personnel of VISIBLE DAMAGE to SAFE SHUTDOWN EQUIPMENT due to the FIRE **OR** 2. Indications show degraded SAFE SHUTDOWN EQUIPMENT performance due to the FIRE	*Refer to Fission Product Barrier Matrix, Control Room Evacuation, System Malfunctions, or Emergency Coordinator Judgment*	*Refer to Fission Product Barrier Matrix, Control Room Evacuation, System Malfunctions, or Emergency Coordinator Judgment*

Fire

Initiating Condition

FIRE within the PROTECTED AREA that could affect SAFE SHUTDOWN EQUIPMENT

Emergency Action Level:

UNUSUAL EVENT
2.14 **MODES: ALL** (1 and 2) 1. FIRE in or threatening one of the following structures: - Auxiliary Building - BWST - Control Complex, - Diesel Generator Building - EFT-2 Building - Intermediate Building - Reactor Building - EFP-3 Building AND 2. FIRE not extinguished within 15 minutes from either Control Room notification or receipt of a VALID fire alarm in the Control Room

FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or **overheated** electrical equipment do not constitute fires. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

VALID: An indication or report or condition is considered VALID when it is conclusively verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by Plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment (e.g., within 15 minutes).

Basis:

This EAL is to address the magnitude and extent of fires that may be potentially significant precursors to damage to safety systems. This excludes such items as fires within administration buildings, wastebasket fires, and other small fires of NO safety consequence. This Initiating Condition applies to buildings and areas contiguous to Plant vital areas or other significant buildings or areas.

Validation of the alarm in this context means those actions taken in the control room or other location to determine the control room alarm is NOT spurious. Fire in other areas adjacent to vital areas may warrant classification if the fire is of a magnitude that threatens vital areas.

The 15-minute time period begins with the time when a credible notification that a fire is occurring or the time a VALID fire detection system alarm is received. The intent of the 15-minute duration is to discriminate against small fires that are readily extinguished.

OP-880A, Appendix "R" Post-Fire Safe Shutdown Information contains additional information on fire damage assessment

CR3 Matrix Reference Number: 2.14

NEI 97-03 Reference: HU2

Fire

Initiating Condition:

FIRE affecting the operability of SAFE SHUTDOWN EQUIPMENT

Emergency Action Level:

ALERT
2.15 **MODES: ALL** (1 or 2) 1. Report by Plant personnel of VISIBLE DAMAGE to SAFE SHUTDOWN EQUIPMENT due to the FIRE OR 2. Indications show degraded SAFE SHUTDOWN EQUIPMENT performance due to the FIRE

VISIBLE DAMAGE: Damage to equipment or structure that is readily observable without measurements, testing, or analyses. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example system/component damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering due to fire. Example structure damage includes exposed and/or broken rebar, failed supports/pipe hangers, etc. Surface blemishing (e.g., paint chipping, scratches, concrete spalling) should NOT be included.

SAFE SHUTDOWN EQUIPMENT: Equipment necessary to achieve and maintain the reactor sub critical with controlled decay heat removal to bring the Plant to the ITS applicable shutdown condition/mode.

FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or **overheated** electrical equipment do not constitute fires. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

Basis:

The key to classifying fires as an Alert is the damage because of the incident. The fact that the equipment required for safe shutdown of the unit has been affected or damaged because of the fire is the driving force for declaring the Alert.

If damage from the fire is clearly contained and localized to one train, and safe shutdown capability exists, then the EAL is NOT met. If the extent of the damage is uncertain in terms of loss of safe shutdown capability, then entry into this EAL is required.

CR3 Matrix Reference Number: 2.15

NEI 97-03 Reference: HA2

16. 069AG2.1.12 001/1/2/SRO#16/C/A 2.9/4.0/NEW/S/CR03701/
The following plant conditions exist:

- The plant is in Mode 6.
- Engineering reports that overall Containment leakage is excessive (exceeds TS allowable level by 5%) based on the most recent leakage testing.

Which of the following restrictions apply based on these conditions?

A. The plant can ascend to Mode 4, but not Mode 3. Moving irradiated fuel in containment is acceptable.

B.✓ The plant cannot ascend to Mode 4. Moving irradiated fuel in containment is acceptable.

C. The plant can ascend to Mode 4, but not Mode 3. Irradiated fuel cannot be moved in containment.

D. The plant cannot ascend to Mode 4. Irradiated fuel cannot be moved in containment.

Reasons

A. Plausible if the candidate fails to recall the Mode of applicability for LCO 3.6.1.

B. Correct. LCO 3.6.1 is applicable in Modes 1-4, and would not be met with the given leakage. LCO 3.9.3 requires containment closure, not containment operability, so this amount of leakage would be acceptable.

C. Plausible if the candidate fails to recall the Mode of applicability for LCO 3.6.1. Also plausible if the student fails to recognize that LCO 3.9.3 only requires containment closure rather than operability.

D. Plausible if the student fails to recognize that LCO 3.9.3 only requires containment closure rather than operability.

OPS 5-01, Obj. 12, 15; ITS LCO 3.6.1 and Basis; ITS LCO 3.9.3 and Basis; 10 CFR 55.43.b.1, 2, 3

SRO - New

Reference(s) provided: None

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	Perform required visual examinations and leakage rate testing except for containment air lock testing, in accordance with the Containment Leakage Rate Testing Program. The maximum allowable leakage rate, L_a, is 0.25% of containment air weight per day at the calculated peak containment pressure, P_a.	In accordance with the Containment Leakage Rate Testing Program.
SR 3.6.1.2	Verify containment structural integrity in accordance with ITS 5.6.2.8.	In accordance with the Containment Inspection Program

BASES

BACKGROUND (continued)

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

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBAs that result in a challenge to containment from high pressures and temperatures are a loss of coolant accident (LOCA), a steam line break, and a rod ejection accident (REA) (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or REA. In the analyses of DBAs involving release of fission product radioactivity, it is assumed that the containment is OPERABLE so that the release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.25% 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 leakage rate at the calculated maximum peak containment pressure (P_a) resulting from the limiting DBA. 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 0.25% of containment air weight per day in the safety analysis at P_a = 54.2 psig (Ref. 3).

The dose acceptance criteria applied to accidental releases of radioactive material to the environment are given in 10 CFR 50.67 (Ref. 5).

The containment satisfies Criterion 3 of the NRC Policy Statement.

(continued)

3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

LCO 3.9.3 The containment penetrations shall be in the following status:

a. The equipment hatch or outage equipment hatch (OEH) installed and held in place by four bolts;

b. A minimum of one door in each installed air lock and the door in the OEH (if installed) closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent; or

2. capable of being closed by an OPERABLE containment purge or mini-purge valve.

APPLICABILITY: During movement of recently irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend movement of recently irradiated fuel assemblies within containment.	Immediately

B 3.9 REFUELING OPERATIONS

B 3.9.3 Containment Penetrations

BASES

BACKGROUND

An accident which occurs during movement of recently irradiated fuel assemblies within containment will have any released radioactivity limited from escaping to the environment. In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, the requirement to isolate the containment from the outside atmosphere is less stringent than those established for MODES 1 through 4. In order to make this distinction, the penetration requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths for radioactivity are closed or capable of being closed by an OPERABLE containment purge or mini-purge valve.

The containment equipment hatch or outage equipment hatch (OEH) provides a means for moving large equipment and components into and out of containment. During movement of recently irradiated fuel assemblies within containment, the equipment hatch or OEH must be held in place by at least four bolts. The required number of bolts is based on dead weight and is acceptable due to the low likelihood of a pressurization event. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced. During movement of recently irradiated fuel assemblies within containment, containment closure is required; therefore, the door in the OEH (if installed) must always remain closed.

The containment air locks provide a means for personnel access during MODES 1, 2, 3, and 4 in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. However, during periods of unit shutdown when containment OPERABILITY is not required, the door interlock mechanism may be disabled, allowing both doors of an installed air lock to remain open for extended periods when frequent containment ingress and egress is necessary. During movement of recently irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed.

(continued)

17. 071A2.02 001/2/2/SRO#17/C/A 3.3/3.6/BANK/S/CR03701/4-061-001
The primary plant operator stops a WDT-1B ("B" Waste Gas Decay Tank) release when WD-19-FQI (release flow monitor) fails low. Which ONE of the following actions must be taken, if any, to re-initiate the release if the flow recorder cannot be fixed?

A. The release may not be re-initiated until the flow monitor is repaired.

B.✓ The release flow rate must be estimated at least once per four hours.

C. Grab samples must be collected and analyzed at least once per 4 hours.

D. Two independent samples with an independent valve alignment must be performed.

Reasons:

A., C, and D. The release is allowed to continue as long as the flow rate is estimated at least once per 4 hours. No other allowance are made for continuing the release.

B. Correct. The release can continue as long as the flow rate is estimated at least once per 4 hours.

OPS 4-61, Obj. 9; ODCM Table 2-3 Action 26; 10 CFR 55.43.b.1

SRO - Bank

Reference(s) provided: None

TABLE 2-3

RADIOACTIVE GASEOUS EFFLUENT AND PROCESS MONITORING INSTRUMENTATION

	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Waste Gas Decay Tank Monitor (RM-A11)			
a. Noble Gas Activity Monitor*	1	ALL MODES	24
b. Effluent System Flow Rate Monitor	1	ALL MODES	26
2. Reactor Building Purge Exhaust Duct Monitor (RM-A1)			
a. Noble Gas Activity Monitor			
i. Operating Range*	1	**	27
ii. Mid Range#	1	**	29
iii. High Range#	1	**	29
b. Iodine Sampler	1	**	25
c. Particulate Sampler	1	**	25
d. Effluent System Flow Rate Monitor	1	**	26
e. Sampler Flow Rate Monitor	1	**	26
3. Auxiliary Building and Fuel Handling Area Exhaust Duct Monitor (RM-A2)			
a. Noble Gas Activity Monitor			
i. Operating Range *	1	ALL MODES	28
ii. Mid Range #	1	1, 2, 3 & 4	29
iii. High Range #	1	1, 2, 3 & 4	29
b. Iodine Sampler	1	ALL MODES	25
c. Particulate Sampler	1	ALL MODES	25
d. Effluent System Flow Rate Monitor	1	ALL MODES	26
e. Sampler Flow Rate Monitor	1	ALL MODES	26
4. Condenser Vacuum Pump Exhaust - Gaseous Activity Monitor (RM-A12)	1	1, 2, 3, 4	30

* Provides control room alarm and automatic termination of release.
** During periods of reactor building purge, except during fuel movement. During fuel movement surveillance requirements for RM-A1 (operating range) are specified by ITS 3.3.15. Fuel movement includes preparation for and demobilization from AI-504, Shutdown Condition 3.
# There is no isolation setpoint or release termination function for this monitor. Alarm setpoints are determined by the appropriate system procedures.

TABLE 2-3 (Continued)
TABLE NOTATION

ACTION 24 With less than the required number of OPERABLE channels, the contents of the Waste Gas Decay Tank may be released to the environment, provided that prior to initiating a release:

1. The Auxiliary Building & Fuel Handling Area Exhaust Duct Monitor (RM-A2) is OPERABLE with its setpoints set to ensure that the limits of Specification 2.7 are not exceeded. The setpoint shall be determined in accordance with the OFFSITE DOSE CALCULATION MANUAL, or

2. a. At least two independent samples of the tank's contents are analyzed in accordance with Table 2-6 and

   b. Two qualified persons independently verify the release rate calculations, and

   c. Two qualified persons independently verify the discharge valve lineup.

Otherwise, suspend releases of radioactive effluents via this pathway.

ACTION 25 RM-A1

With the affected sampler inoperable, operation of the RB purge may continue for 1 hour with no auxiliary sampling, provided that RB airborne levels are steady state or declining. If indicators of RB atmospheric activity, such as RM-A6, RCS leakage, or general area air samples, show an increase in RB activity while the sampler is inoperable, then immediately restore the affected sampler, or implement auxiliary sampling, or shut down the purge.

With the affected sampler inoperable, operation of the RB purge may continue for more than 1 hour provided that samples (reference Tables 2-6) are continuously taken (except for filter changes) with auxiliary sampling equipment.

Auxiliary sampling equipment includes general area RB air samples or RMA-15. Other sampling regimes are acceptable provided results are representative of plant effluents.

Note: Coordination of sampling during core alterations or fuel movement is required in order to comply with Technical Specifications.

TABLE 2-3 (Continued)
TABLE NOTATION

ACTION 25 (Continued) RM-A2

With the affected channel inoperable, effluent releases may continue for 1 hour with no auxiliary sampling, provided that AB airborne levels are steady state or declining. If indicators of AB atmospheric activity, such as RM-A3, RM-A4, and RM-A8 show an increase in activity then restore the affected sampler, or implement auxiliary sampling, or shut down the release.

With the affected sampler inoperable, effluent releases may continue for more than 1 hour provided that samples (reference Table 2-6) are continuously taken (except for filter changes) with auxiliary sampling equipment.

Auxiliary sampling equipment includes 1) RM-A4 and RM-A8 used together 2) general area AB air samples, or 3) RMA-15. Other sampling regimes are acceptable provided results are representative of plant effluents.

ACTION 26 With the number of OPERABLE channels less than required, effluent releases via this pathway may continue, provided flow rate is estimated at least once per 4 hours.

ACTION 27 With the noble gas monitor (operating range) inoperable, operation of the RB purge may continue for 1 hour, provided that RB airborne levels are steady state or declining. If indicators of RB atmospheric activity such as RM-A6, RCS leakage, or general area air samples show an increase in RB activity while the monitor is inoperable, then immediately restore the noble gas monitor or shut down the purge.

Note: Coordination of sampling during core alterations or fuel movement is required in order to comply with Technical Specifications.

ACTION 28 With the number of OPERABLE channels less than required, releases via this pathway may continue, provided grab samples are collected at least once per 12 hours and analyzed within 24 hours, and either the requirements of ACTION 24 Part 2 are met or Radiation Monitor RM-A11 is OPERABLE prior to releasing the contents of the Waste Gas Decay Tanks.

* Gas grabs may be taken from RM-A4 and RM-A8.

18. BW/E05EG2.1.7 001/1/1/SRO#18/C/A 3.7/4.4/BANK/S/CR03701/5-014-009
The following plant conditions exist:
(assume all Immediate Actions are complete for EOP-2, Vital System Status Verification)

- A reactor trip has occurred.
- The "A" steam generator level is 87% and rising.
- The "B" steam generator level is 43% and rising.
- All Reactor Coolant Pumps (RCP) are operating.
- Both Emergency Feedwater Pumps (EFW) are operating.
- Reactor coolant temperature has lowered from 535° F to 533° F in 30 seconds.
- Reactor coolant pressure is 1850 psig.

Based on the above conditions which of the following describes the appropriate Emergency Operating Procedure and associated rule for this situation?

A. EOP-2, Vital System Status Verification; EOP-13 Rule 4, Pressurized Thermal Shock.

B. EOP-2, Vital System Status Verification; EOP-13 Rule 3, EFW Control.

C. EOP-5, Excessive Heat Transfer; EOP-13 Rule 4, Pressurized Thermal Shock.

D.✔ EOP-5, Excessive Heat Transfer; EOP-13 Rule 3, EFW Control.

Reasons:

A. & B. Due to RCS temperature the conditions are met for entering EOP-5.

C. RCS temperature is not low enough for entry into Rule 4 (< 400° F.)

D. Correct. For the given conditions an overcooling event would be in progress and EOP-5 would be in use. With OTSG levels excessive and rising, Rule 3 would be provide the guidance to manually control levels.

OPS 5-14, Obj. 11; OPS 3-22 Section 1-3.0; EOP-13 Rules 3 & 4; 10 CFR 55.43.b.5

SRO - Bank

Reference(s) provided: None

1-3.0 EXCESSIVE HEAT TRANSFER (OVERCOOLING)

A. Excessive heat transfer (overcooling) is undesired heat removal by one or both OTSGs. Indications of excessive heat transfer include:

1. RCS pressure and temperature decreasing below the desired range.

2. OTSG pressure is low or decreasing in one or both OTSGs.

3. Secondary Tsat is equal to or less than Tcold for the affected OTSG(s).

B. Excessive heat transfer may result from overfeeding or loss of OTSG pressure control. Regardless of the cause, all three indications will be present.

1-4.0 CONCERNS

A. Excessive primary to secondary heat transfer is caused by a failure in the control of secondary side parameters. This failure manifests itself as a loss of steam pressure, excessive steam or feedwater flow, or perhaps a combination of both. While a momentary overcooling is only troublesome, an extended overcooling is a severe shock to the plant and requires quick and effective action by the operator to mitigate the transient. There are several concerns associated with an excessive heat transfer transient:

1. Loss of Pressurizer Level

a) An extended overcooling can result in a loss of pressurizer level. This in turn causes a loss of RCS pressure control. An extended overcooling can empty the surge line, which will result in the RCS becoming saturated at the hottest point in the system.

1) The highest temperature in the RCS following initiation of an overcooling transient with RCPs on will be in the upper head region.

RULE 3, EFW/AFW CONTROL

- ___ Ensure available OTSGs are at or trending toward required level.

"LLL"	> 20 in
"NAT CIRC"	> 70%
"ISCM"	> 90%

- ___ IF manual control of EFW flow is desired,
  THEN establish manual EFIC control.

1 ___ Obtain SRO concurrence to place EFIC in manual.

2 ___ Control EFW to maintain required EFW flow and OTSG level.

3 ___ IF EFW flow is NOT controlled,
THEN depress EFIC channels A and B "MANUAL PERMISSIVE" push buttons and close affected EFW block valve.

- ___ IF adequate SCM does NOT exist,
  AND level in available OTSGs is NOT at or trending toward "ISCM" level,
  THEN establish manual required flow.

EFW	2 OTSGs	> 280 gpm in 1 line to each OTSG
	1 OTSG	> 470 gpm in 1 line to 1 OTSG
AFW	2 OTSGs	> 250 gpm to < 300 gpm/OTSG
	1 OTSG	> 450 gpm to < 600 gpm

- ___ IF adequate SCM exists,
  THEN throttle flow to prevent OTSG PRESS from lowering > 100 psig below desired PRESS.

- Do not allow OTSG level to lower.

RULE 4, PTS

- ___ IF any of the following exist:

  ___ $T_{incore} < 400°F$,
  AND cooldown rate exceeds ITS limit

  ___ RCPs off,
  AND HPI flow exists

  THEN perform required PTS actions.

- ___ Throttle HPI flow to minimize adequate SCM.

- ___ Throttle LPI flow to minimize adequate SCM.

- ___ PTS is applicable until an Engineering evaluation has been completed.

19. G2.1.34 001/GENERIC/1/SRO#19/C/A 2.3/2.9/NEW/S/CR03701/
The following plant conditions exist:

- The plant is at 100% power.
- RCS pH is 7.3
- RCS Hydrogen concentration is 30 cc/Kg.
- RCS Dissolved Oxygen is 25 ppb.
- RCS Fluoride concentration is 40 ppb.
- RCS Chloride concentration is 40 ppb.

Which ONE of the following actions should be taken?

A. Restore the out of spec parameter to within limits as soon as possible.

B.✓ Restore the out of spec parameter to within limits within 7 days.

C. Restore the out of spec parameter to within limits within 24 hours or commence an orderly plant shutdown to cold shutdown conditions.

D. Immediately commence an orderly plant shutdown to cold shutdown conditions.

Reasons:

A. This would be the correct action if pH were out of spec.

B. Correct. Dissolved Oxygen (DO) is greater than Action Level 1 limits. Action Level 1 for DO is >5 ppb. No Action Level 2 is specified for DO. Action Level 3 is >100 ppb.

C. This would be the correct action for any parameter Action Level 2 exceeded.

D. This would be the correct action for any parameter Action Level 3 exceeded.

OPS 5-115, Obj. 4; CP-142 Section 3.1.4 and Enclosure 1 Table A; 10 CFR 55.43.b.5

SRO - New

Reference(s) provided: CP-142

3.1.4 **Action Level 1** – The Action Level 1 value of a parameter represents the range, outside of which data or engineering judgment indicates that long-term system reliability may be affected, thereby warranting an improvement of operating practices, or indicates increasing impurity concentrations which warrant investigation prior to further degradation of water quality. Actions that shall be taken if a parameter exceeds the Action Level 1 value:

- Efforts shall be made to bring the parameter within the appropriate limit within seven days.
- If the parameter has not been restored to the appropriate range within seven days, a technical review shall be performed and a program for implementing corrective measures instituted. Such a program may require equipment additions or modifications over the long term.

3.1.5 **Action Level 2** – The Action Level 2 value of a parameter represents the value, outside of which data or engineering judgment indicates significant damage could be done to the system in the short term, whereby warranting a prompt correction of the abnormal condition. Actions that shall be taken if a parameter exceeds the Action Level 2 value:

- Efforts shall be made to bring the parameter within the appropriate Action Level 2 value within 24 hours.
- If the parameter has not been restored to within the Action Level 2 limit condition within 24 hours, an orderly unit shutdown shall be initiated and the plant shall be brought to a cold shutdown condition as quickly as permitted by other plant constraints. If chemistry is improved to within the requirements of Action Level 2 prior to plant shutdown, full power operation may be resumed.
- Following a unit shutdown caused by exceeding the time limit on an Action Level 2 value, a technical review of the incident shall be performed and appropriate corrective measures taken before the unit is restarted.

3.1.6 **Action Level 3** – The Action Level 3 value of a parameter represents the limit, beyond which data or engineering judgment indicates that it is not advisable to continue operating the plant. Actions that shall be taken if a parameter exceeds the Action Level 3 value:

- An orderly unit shutdown shall be initiated immediately, with reduction of coolant temperature to < 250 degrees F as rapidly as other plant constraints permit.
- If chemistry is improved to within the requirements of Action Level 3 prior to plant shutdown, power operation may be resumed, subject to the requirements of other Action Levels and plant technical specifications.
- Following a unit shutdown caused by entering an Action Level 3 condition, a technical review of the incident shall be performed and appropriate corrective measures taken before the unit is restarted.

ENCLOSURE 1 - POWER OPERATION, REACTOR CRITICAL
TABLE A (Page 1 of 3)

REACTOR COOLANT SYSTEM

CONTROL PARAMETER	NORMAL VALUE	INITIATE ACTION	ACTION RESPONSE	CORRECTIVE ACTION
Boron	≤ 2270 ppm	> 2270 ppm	Inform Operations value is above limit.	Operations corrective actions.
Lithium **(Note 1)**	≤ 3.5 (≤ 4.5) ppm and as per B-Li Control Band (Enclosures 9 and 10)	High or Low limits of B-Li Control Band exceeded (Enclosures 9 and 10) > 3.5 (> 4.5 ppm)	Action Level 2 Action Level 3	1. Check to ensure reactor coolant dilution or boration is not in progress. 2. Ensure that a lithiated ion exchange bed is in service. 3. Verify flow through cation or other purification ion exchangers. 4. Adjust lithium as required to bring within station control program.
pH_t @ 300°C **(Note 2)**	> 6.4 or < 7.8	≤ 6.9 or ≥ 7.4	Adjust lithium as necessary.	1. Adjust lithium as required to bring within station control program.
Hydrogen **(Note 3)**	≥ 25 cc/Kg and ≤ 50 cc/Kg	< 25 or >50 cc/Kg ≤ 15 cc/Kg ≤ 5 cc/Kg	Action Level 1 Action Level 2 Action Level 3	1. Ensure appropriate hydrogen overpressure in the MUT. 2. Ensure adequate letdown flow. 3. Check for indications of leaking valves or air ingress in the Make-up system.
Dissolved Oxygen	≤ 5 ppb	> 5 ppb > 100 ppb	Action Level 1 Action Level 3	1. Verify hydrogen concentration is within limits. 2. Check for abnormal concentration of oxygen in makeup water. 3. Check for air leaks into the makeup system.

REACTOR COOLANT SYSTEM

CONTROL PARAMETER	NORMAL VALUE	INITIATE ACTION	ACTION RESPONSE	CORRECTIVE ACTION
Fluoride	≤ 50 ppb	> 50 ppb >150 ppb >1500 ppb	Action Level 1 Action Level 2 Action Level 3	1. Check ion exchange beds in purification system for flow and removal efficiencies. 2. Check for high RCS ammonia or conductivity that may cause fluoride release from resin. 3. Check makeup water purity. 4. Isolate makeup water source or change to alternate source if required. 5. Seek other potential sources of fluoride.
Chloride	≤ 50 ppb	> 50 ppb >150 ppb >1500 ppb	Action Level 1 Action Level 2 Action Level 3	1. Check ion exchange beds in purification system for flow and removal efficiencies. 2. Check for high RCS ammonia or conductivity that may cause chloride release from resin. 3. Check makeup water purity. 4. Isolate makeup water source or change to alternate source if required. 5. Seek other potential sources of chloride.
Sulfate	≤50 ppb	>50 ppb >150 ppb >1500 ppb	Action Level 1 Action Level 2 Action Level 3	1. Check for indications of resin released from the purification system. 2. Check ion exchanger removal efficiency and isolate if necessary. 3. Check makeup water purity. 4. Isolate makeup water source or change to alternate source if required. 5. Seek other potential sources of sulfate.

REACTOR COOLANT SYSTEM

CONTROL PARAMETER	NORMAL VALUE	INITIATE ACTION	ACTION RESPONSE	CORRECTIVE ACTION
Silica **(Note 4)**	≤ 1000 ppb	>1000 ppb >3000 ppb	Reduce concentration to < 1000 ppb. Reduce concentration to <3000 ppb prior to 100% reactor power.	1. Investigate and determine the source of the silica increase in the RCS. 2. Perform system feed and bleed to reduce silica in the coolant. 3. Monitor RCS and all sources of makeup water for Si, Al, Ca, and Mg weekly until RCS Si concentration < 1000 ppb. WHEN RCS Silica is ≥ 1000 ppb, THEN perform Silica, Aluminum, Calcium, and Magnesium analysis weekly on RCS and RCS feed sources including Demin Water, BAST's, and the in service RCBT. The following limits apply for the RCS feed sources: 1. Silica 1000 ppb 2. Aluminum 80 ppb 3. Calcium 40 ppb 4. Magnesium 40 ppb 5. Ca + Mg 80 ppb

Note 1: Lithium limits apply only to steady state operations and do not apply during transient conditions. The 4.5 ppm upper lithium limit applies only to the first 4 EFPD of the cycle. The 3.5 ppm lithium limit applies after the first 4 EFPD of cycle.

Note 2: pH limits (Normal Values) are stated FSAR limits. However, a constant pH control regime is in effect at 7.0 pH units, maintained per B-Li Control Band (Enclosures 9 and 10). A pH of 6.9 shall be achieved as soon as possible following reactor criticality from shutdown.

Note 3: On start-up, new fuel may require a lower initial hydrogen concentration. Follow fuel vendor's recommendations for this specific case.

Note 4: Silica is an EPRI and Framatome (B&W) diagnostic parameter.

20. G2.1.4 001/GENERIC/1/SRO#20/MEM 2.3/3.4/BANK/S/CR03701/5-001-007
The following plant conditions exist:

- The plant is in Mode 3.
- One of the two available PPOs slips and severely sprains his ankle while performing a walkdown of the Reactor Building.
- The PPO is contaminated and is escorted to the hospital by both available Health Physics technicians.

Which ONE of the following describes the appropriate response, relating to shift staffing, for this situation?

A. No action is required. Minimum staffing levels are still met.

B. If it is two hours or less until shift turnover is scheduled to occur no action is required.

C. Another PPO should be called in immediately and should arrive within two hours.

D.✓ Another HP technician should be called in immediately and should arrive within two hours.

Reasons:

A. A minimum of one HP technician is required when fuel is in the reactor.

B. Efforts must be made immediately to replace the HP technician within two hours.

C. Only one PPO is required to meet staffing levels.

D. Correct. An individual qualified in Radiation Protection procedures shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

OPS 5-01, Obj. 14; TS 5.2.2; AI-500 Section 4.4; 10 CFR 55.43.b.1 & 2

SRO - Bank

Reference(s) provided: None

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions responsible for activities affecting safety of the nuclear power plant.

a. Lines of authority, responsibility, and communications shall be established and defined from the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of department responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These shall be documented in the FSAR;

b. The Vice President - Crystal River Nuclear Plant shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety. The Vice President - Crystal River Nuclear Plant shall be responsible for the overall safe operation of the plant and shall have control over those onsite activities necessary for the safe operation and maintenance of the plant; and

c. The individuals who train the operating staff, carry out health physics or perform quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

5.2.2 Unit Staff

The unit staff organization shall include the following:

a. One auxiliary nuclear operator shall be assigned to the operating shift any time there is fuel in the reactor and

(continued)

5.2 Organization

5.2.2 Unit Staff (continued)

an additional auxiliary nuclear operator shall be assigned in MODES 1, 2, 3 and 4.

b. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

c. At least one licensed Reactor Operator (RO) shall be present in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Reactor Operator (SRO) shall be present in the control room.

d. An individual qualified in Radiation Protection procedures shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

e. The amount of overtime worked by unit staff members performing safety related functions shall be limited and controlled in accordance with approved administrative procedures.

4.3 Monitoring And Assessment Of Performance

4.3.1 Plant operation is monitored on a regular basis using the SP-300 series procedures. In addition, the shift turnover process provides for a peer review of plant status and activities.

4.3.2 Personnel performance is informally assessed continually through the peer evaluation process. It is formally assessed in AI-501 by management and supervision.

4.4 Shift Staffing Requirements

4.4.1 A Reactor Operator or Senior Reactor Operator shall remain present at the controls at all times.

4.4.2 During all Operating Modes, the normal operating shift crew will include:

- One Superintendent Shift Operations (SRO)
- One Control Room Supervisor (SRO)
- Two Chief Nuclear Operators (RO)
- One Nuclear Operator (RO)
- Three Non-Licensed Operators (one must be PPO qualified)
- One Shift Technical Advisor (STA)* (may be assigned other functions such as OWC)
- An Operations Work Coordinator (OWC) [NOCS 006010]
- Additional personnel may be assigned to the shift as Operations staffing level permits
- Shift fire brigade coverage – see Appendix 11 for specific details

* Coverage will be 24 hours/day and will be maintained available to the control room within 10 minutes, when summoned, during Modes 1, 2, 3, and 4. [NOCS 009824, 001094]

NOTE

Shift crew composition may be below the minimum requirements stated below for a period of not more than two hours in order to accommodate unexpected absence of on-duty shift crew members, provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

NOTE

The minimum shift crew composition requirements must exclude personnel required for fire brigade manning.

4.4.3 During Operating Modes 1, 2, 3, and 4, the minimum shift crew will include:

- Two Senior Reactor Operators
- Two Reactor Operators
- Two Non-licensed Operators (one must be PPO qualified)
- One Shift Technical Advisor

4.4.4 During Operating Modes 5 and 6, the minimum shift crew will include: [NOCS 006010]

- One Senior Reactor Operator
- One Reactor Operator
- One Non-Licensed Operator

21. G2.2.17 001/GENERIC/2/SRO#21/MEM 2.3/3.5/NEW/S/CR03701/
Which of the following is **NOT** a duty of the Operations' Work Coordinator (OWC)?

A. Performs work week reviews using Form Appendix 20-01, Work Week Checklist.

B. Representing Operations at the 0615 Daily Plant Status Meeting.

C.✔ Approving manpower per Appendix 11, Operations Schedules.

D. Representing Operations at the T-5 Meeting.

Reasons:

A, B, D. Per AI-500, Appendix 20, these are duties of the OWC.

C. Correct. AI-500, Appendix 20, Step 2.4 specifically lists this as a SSO duty.

OPS 5-38, Obj. 1, 2, 3; AI-500 Appendix 20; 10 CFR 55.43.b.7

SRO - New

Reference(s) provided: None

OPERATIONS WORK CONTROL CENTER

2.0 Responsibilities

2.1 Operations Management is responsible for:

- Ensuring that Operations is represented at all required schedule meetings.
- Performing an aggregate, systematic monthly review to assess the prioritized backlogged plant deficiency lists. Ensure the proper schedule integration of backlogged work requests is occurring.

2.2 The OWC is responsible for:

- Performing Work Release for scheduled activities.
- Reviewing and authorizing equipment clearances.
- Performing work week reviews using FORM APPENDIX 20-01, Work Week Checklist.
- Performing the Operations Impact Review for new work requests.
- Performing planning and / or reviews of work order tasks routed to Operations.
- Creating required post maintenance testing tasks for the scheduled work prior to T-4.
- Performing the Operations Plant Condition Review for routed work order tasks.
- Performing Operations department schedule updates.
- Providing shift schedule comments to the Work Week Manager for inclusion into the Progress Report for the work week.
- Performing Operations work order task completion in PassPort.
- Transmitting completed Operations work order tasks to Records Management.
- Reviewing and coordinate post maintenance testing and return to service requirement tasks for plant equipment.
- Representing Operations at the 0615 Daily Plant Status Meeting.
- Interfacing with the Work Week Manager for execution of the current schedule.
- Ensuring that post maintenance testing and associated tasks are created for the scheduled work.
- Accepting Engineering Change (EC) turnovers to Operations.
- Ensuring operations personnel receive work assignments, attend or receive required pre-jobs for assigned work activities, support daily clearance activity.
- Authorizing Fire Protection Work Permits.
- Authorizing and closing Fire Barrier Breach Reports.
- Authorizing, tracking, and closing Control Complex Breach Reports.
- Reviewing and resolving Operations issues in the schedule for their respective weeks.
- Ensuring the work schedule has been risk assessed for Maintenance Rule activities and an EOOS screening.
- Reviewing the work week schedule to ensure that scheduled work activities do not have carry over post maintenance or return to service testing.
- Remaining cognizant of the Degraded Equipment List and Degraded Annunciator List items in their respective work weeks. Maintain contact with key personnel to support resolution.
- Represents Operations at the T-5 meeting.

OPERATIONS WORK CONTROL CENTER

2.3 The CNO is responsible for:

- Writing and verify equipment clearances and checklists.
- Coordinating hanging and pulling of clearances.
- Hanging or pulling clearances to support the schedule, if building operators are not available.
- Coordinating with clearance holders or work order task holders to modify clearances, add work order tasks, etc., as described in OPS-NGGC-1301 and CP-115B.
- Supporting FIN Team activities.
- Reviewing upcoming work activities, surveillances, and procedures to ensure that preparations are adequate.
- Represents Operations at the T-5 meeting when available.
- Performing CNO duties as described in AI-500 and AI-505.

2.4 The SSO is responsible for:

- Maintaining overall responsibility for the daily work schedule during the implementation week.
- Ensuring their respective implementation work week activities are reviewed prior to the T-2 schedule meeting.
- Approving manpower per APPENDIX 11, Operations Schedules.
- Approving the implementation week Maintenance Rule risk assessment and EOOS screening.
- Ensuring appropriate schedule review and risk assessment occurs based upon Emergency Energy Management notifications.
- Preparing work scripts and providing to interfacing crews, for complicated evolutions such as ECCS outages. Work scripts are informal documents which provide detailed sequencing of an evolution beyond the scope of the daily work schedule.
- Approving work scope changes inside T-2 for respective implementation week.
- Ensuring that Equipment Clearance Orders are generated for his/her onshift dayshift week.
- Ensuring that all clearances are written and reviewed prior to the end of the T-2 week.
- Ensuring that manpower scheduled for Operations is identified by T-4, and assigned as required prior to the T-2 week to ensure that there is adequate coverage for the scheduled evolutions.
- Performing the EOOS analysis whenever emergent equipment problems occur that potentially impact the performance of risk significant systems/components.
- Performing the EOOS analysis whenever a status change occurs for any of the twelve (12) different equipment items which are capable of being manually input into the System Status in EOOS. This is only applicable for alignment changes other than those made to support implementation of the approved work week schedule.
- Chairs the Daily Plan of the Day meeting representing Operations focus and alignment to the rest of the site management team.

22. G2.2.26 001/GENERIC/2/SRO#22/C/A 2.5/3.7/BANK/S/CR03701/5-078-004
The following plant conditions exist:

- The refueling team is placing fuel assembly #20 in the core.
- Spent Fuel Pool level is 157 feet.

Which of the following are the minimum requirements for this condition?

A. Two operable qualified off-site power sources; one RCS boration flow path.

B.✔ One operable qualified off-site power source; one RCS boration flow path.

C. Two operable qualified off-site power source; two RCS boration flow paths.

D. One operable qualified off-site power sources; two RCS boration flow paths.

Reasons:

B. Correct. In Shutdown Condition 3, one qualified operable off-site power source and one RCS boration flow path.

A., C., D. These are all plausible combinations of required equipment, but not for the given plant conditions

OPS 5-78, Obj. 14; AI-504 Enclosure 3; 10 CFR 55.43.b.7

SRO - Bank

Reference(s) provided: AI-504 Enclosures 1-6

SHUTDOWN CONDITION 3 FTC FLOODED ≥ 156 ft
REFUELING OPERATIONS IN PROGRESS
FUEL IN REACTOR VESSEL [NOCS 100333]

[√] all that are AVAILABLE/OPERABLE

A. DECAY HEAT REMOVAL EQUIPMENT REQUIREMENTS

1. One DECAY Heat Removal Train OPERABLE and in OPERATION:
   [ ] a. A DHR train
   [ ] b. B DHR train

2. One RB sump suction flow path associated with an OPERABLE DHR train shall be AVAILABLE: (Active)
   [ ] a. A LPI sump suction flow path
   [ ] b. B LPI sump suction flow path

3. Other AVAILABLE Decay Heat Removal methods (Not required):
   [ ] a. Alternate DHR train (Active)
   [ ] b. CFT Injection (Passive)
   [ ] c. Not AVAILABLE

4. Spent Fuel Pool Cooling AVAILABLE:
   [ ] a. One SFP Cooling Train with ≤1/3 core recently discharged Fuel assemblies or
   [ ] b. Two SFP Cooling Trains with >1/3 core recently discharged Fuel assemblies

B. INVENTORY CONTROL REQUIREMENTS

1. AVAILABLE normal RCS inventory makeup flow paths (None are required for this shutdown condition):
   [ ] a. RCBT via DH Purification flow path
   [ ] b. DW via DH Purification flow path
   [ ] c. None AVAILABLE

2. At least one Back-up inventory flow path shall be maintained AVAILABLE:
   [ ] a. LPI sump suction flow path (Active)
   [ ] b. Core Flood Injection (Passive)

3. One AVAILABLE SF Pool inventory makeup flow path:
   [ ] a. DW via SFP-2 flow path
   [ ] b. BWST via SFP-2
   [ ] c. RCBT via installed temporary hose
   [ ] d. Emergency makeup water hose, DW hose bib to the Spent Fuel Pools
   [ ] e. Fire Service via temporary hose to SF pool

C. ELECTRICAL POWER DISTRIBUTION REQUIREMENTS

1. One qualified off-site power source is required to be OPERABLE:
   [ ] a. 500Kv Backfeed
   [ ] b. Offsite Power Transformer
   [ ] c. BEST Transformer

2. One Emergency Diesel Generator associated with the Protected Decay Heat Removal Train is required to be OPERABLE:
   [ ] a. A Emergency Diesel Generator
   [ ] b. B Emergency Diesel Generator

3. One additional power source is required to be maintained AVAILABLE:
   [ ] a. 500Kv Backfeed
   [ ] b. Offsite Power Transformer
   [ ] c. BEST Transformer
   [ ] d. Alternate Emergency Diesel Generator

4. The necessary portions of AC, DC, and AC Vital bus electrical distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.
   [ ] a. A Train Electrical Distribution Subsystems
   [ ] b. B Train Electrical Distribution Subsystems

D. REACTIVITY CONTROL REQUIREMENTS

1. Reactor Coolant System reactivity control requirements:
   [ ] a. RCS boron concentration maintained greater than that required by the COLR.

   [ ] b. Reactor Coolant System dilution flow paths tagged closed (except those used for performing inventory additions per approved procedures.)

List below dilution flow paths not closed:

_____
_____
_____
_____
_____

2. One RCS Boration flow path required:
   [ ] a. A BAST
   [ ] b. B BAST
   [ ] c. HPI Injection from BWST
   [ ] d. LPI Injection from BWST

3. Spent Fuel Pool boron concentration greater than that required by:
   [ ] a. COLR, and
   [ ] b. SP-318, Spent Fuel Pool Boron Concentration Verification.

4. An AVAILABLE Spent Fuel Pool boration flow path:
   [ ] a. BWST via SFP-2
   [ ] b. BAST

23. G2.3.4 001/GENERIC/3/SRO#23/MEM 2.5/3.1/NEW/S/CR03701/
The following plant conditions exist:

- A General Emergency has been declared based on the fission product matrix.
- An operator is to be sent into the auxiliary building to perform a discretionary damage assessment.
- The operator has received 1000 mrem TEDE this year, but has received NO dose during this event.

Per EM-104 "Operation of the Operational Support Center" what is the maximum dose the operator can receive for this entry?

A. 4 rem

B.✔ 5 rem

C. 9 rem

D. 10 rem

Reasons:

A. Plausible since the operator would have a total of 5 rem for the year with this activity.

B. Correct. During declared emergencies, emergency workers are allowed to receive up to 5 rem TEDE for the duration of the emergency regardless of normal exposure to date for the year.

C. Plausible since this would be the normal 10 rem emergency limit (described in D below) with the operator's 1 rem YTD dose subtracted.

D. Plausible since 10 rem would be the limit for preventing serious injury, protecting valuable property, or preventing a catastrophic incident. Performing a damage assessment meets none of these criteria.

OPS 5-34, Obj. 2; EM-104 Section 3.3.1; EM-202 Enclosure 7; 10 CFR 55.43.b.4

SRO - New

Reference(s) provided: None

3.2.15 TSC Radiation Controls Coordinator. Updates and makes recommendations to the Emergency Coordinator on radiological and chemistry-related activities. Reports to the Emergency Coordinator.

3.2.16 TSC Emergency Coordinator. Provides for overall coordination and direction of emergency response and authorizes re-entries and exposure limits in excess of 5 rem TEDE. Reports to the EOF Director when EOF becomes operational.

3.2.17 TSC Repairs Coordinator. Updates the Emergency Coordinator on Emergency Repair Team activities. Reports to the Emergency Coordinator.

3.2.18 Radiation Protection Sub-Unit maintains inventory of emergency kits as identified in Section 3.4.1.

3.2.19 Document Services / Plant Support maintain manuals, procedures, and drawings in the TSC/OSC.

3.2.20 HP Responders assist the Radiation Monitoring Teams in providing access control, in-plant / out-of-plant surveys, and personnel monitoring activities and report to / take direction from an RMT member.

3.3 **Limits & Precautions**

3.3.1 OSC personnel follow the guidelines for exposure of emergency workers during re-entry activities as identified below or as stated on Emergency Team Authorization Form: [NOCS 10546, 15020, 17040]

Dose Limit Rem TEDE	Activity	Condition
5	All	
10	Prevent serious injury, protect valuable property, prevent catastrophic incident	
25	Life saving, protect large populations	
>25	Life saving, protect large populations	Voluntary > 45 years old, trained, and understand radiological health risks involved

a. Health Physics personnel provide recommended courses of action to minimize exposure.
b. Any exposure in excess of 25 rem TEDE should be on a voluntary basis. To perform activities NOT addressed in the above Table, personnel exposure in excess of 5 rem TEDE may be authorized by the Emergency Coordinator with guidance from the Radiation Controls Coordinator.
c. During declared emergencies, emergency workers are allowed to receive up to 5 rem TEDE for duration of emergency regardless of normal exposure to date for the year.

3.3.2 There will be NO re-entry into any emergency evacuated areas unless authorized by the EC or designee.

Guidelines for PE Emergency Worker Exposure

CONDITION	DOSE LIMIT (REM TEDE)	GUIDANCE
1. Emergency conditions NOT requiring actions to prevent serious injury or protect valuable property.	5	Emergency worker exposure should NOT exceed 5 REM TEDE.
2. Emergency conditions requiring actions to prevent serious injury or protect valuable property.	10	Exposure greater than 5 REM TEDE should receive approval of the Emergency Coordinator. Appropriate controls for emergency workers include time limitations and respirators.
3. Emergency conditions requiring lifesaving actions or actions to protect large populations.	25	Exposure greater than 5 REM TEDE should receive approval of the Emergency Coordinator. Appropriate controls for emergency workers include time limitations, respirators, and thyroid blocking.
4. Emergency conditions requiring lifesaving actions or actions to protect large populations.	> 25	Exposure greater than 5 REM TEDE receive approval of the Emergency Coordinator. Exposure at this level should be to volunteers who are healthy, above the age of 45, have an understanding of the health risks involved, and, preferably, be those whose normal duties have trained them for such missions. Appropriate controls for emergency workers include time limitations, respirators, and thyroid blocking.

NOTE: Reference for this table is Table 2.2 in the Manual of Protective Action Guides and Protective Actions for Nuclear Incidents (EPA 400-R/92-001).

24. G2.4.25 001/GENERIC/4/SRO#24/MEM 2.9/3.4/BANK/S/CR03701/5-040-001
The following plant conditions exist:

- Plant is in Mode 1.

Which ONE of the following areas would require the Fire Protection Engineer to evaluate prior to authorizing fire permits in accordance with FIR-NGGC-0003 "Hot Work Permit"?

A. Makeup Pump Room.

B. Unit 4160 V Switchgear Room.

C. Nuclear Services Seawater Room.

D.✔ Vital Bus Inverter Room.

Reasons:

A, B, & C: Plausible since these are all rooms that are important to safety or plant operation, but they are not listed in FIR-NGGC-0003.

D. Correct. In Mode 1-3, the Fire Protection Engineer must evaluate hot work in any "Risk Significant Plant Area". FIR-NGGC-0003 specifically lists the Vital Bus Inverter Rooms as such an area.

OPS 5-40, Obj. 2; FIR-NGGC-0003 Step 3.14 & Step 4.6.1; 10 CFR 55.43.b.3

SRO - Bank

Reference(s) provided: None

3.0 DEFINITIONS

3.14 Risk Significant Plant Area [CR3 only]

Plant areas in which it has been determined the significance of a fire is high and includes the following areas.

- Cable Spreading Room
- Engineered Safeguards (ES) Switchgear Rooms
- Remote Shutdown Room and associated Hallway
- Station Battery Rooms
- Control Rod Drive (CRD) Equipment Room
- Control Room
- Vital Bus Inverter Rooms
- Battery Charger Rooms
- EFIC Rooms
- HVAC Rooms

3.15 Roving Hot Work Fire Watch

A fire watch may be designated to rove through an area to cover several hot work jobs in relatively close proximity to each other. [HNP and RNP - During Modes 5 or 6] [BNP - During Modes 4 or 5] [CR3 - A fire watch may be designated to rove through an area to cover several hot work jobs that are within close proximity and line of sight of each other.] In all cases, all provisions of the Fire Watch Responsibilities must be controlled in their entirety at all times for all areas covered.

3.16 Spark Producing Work

Work that produces a trail of visible sparks which have sufficient volume and mass to comprise an ignition source for a class A or B fire.

4.0 RESPONSIBILITIES

4.5.2 Ensure a Fire Protection Engineer evaluates cutting, welding, spark-producing work, and heat treating processes during Modes 1 through 3 in Risk Significant Plant Areas before authorizing work to begin.

4.5.3 Document that the appropriate systems have been inhibited or isolated.

4.5.4 Authorize / Approve / Concur with Hot Work Permits.

4.5.5 Maintain a Hot Work Log with information similar to that contained in Attachment 3.

4.5.6 Assigns sequential Hot Work Permit numbers on a Logsheet form similar to Attachment 3.

4.5.7 Lists the start and completed time on the Hot Work Permit.

4.5.8 Ensure that any fire protection feature disabled during the course of the hot work activity has been returned to operable status before the hot work permit is cancelled.

4.5.9 Ensure the permit has been cancelled and removed from the Hot Work Log.

4.6 Fire Protection Engineer [CR3 only] [CR3-NOCS 4565]

4.6.1 Evaluates / Approves cutting, welding, spark-producing work, and heat treating processes during Modes 1 through 3 in Risk Significant Plant Areas as defined in step 3.14 to ensure adequate compensatory measures exist.

4.6.2 Evaluates heat treating processes throughout the plant on a case-by-case basis for their fire potential to nearby exposures.

4.6.3 Directs compensatory measures taken when ignition sources are used in areas with impaired automatic fire suppression systems.

25. G2.4.40 001/GENERIC/4/SRO#25/MEM 2.3/4.0/BANK/S/CR03701/ROT-5-034-027
The emergency coordinator (EC) has declared a general emergency. The emergency operations facility, EOF, is not staffed. Besides classification what other duty can NOT be delegated to another emergency team member by the EC?

A.✓ Determine protective action recommendations

B. Direct site evacuation

C. Make notifications to the state

D. Direct the shutdown of the plant

Reasons:

A. Correct. Per EM-202 this item cannot be delegated.

B., C., and D. are things that the EC has the authority to do but can also be delegated.

OPS 5-34, Obj. 2; EM-202 Step 3.2.3; 10 CFR 55.43.b.5

SRO - Bank

Reference(s) provided: None

3.1.33 **Significant Transient** - An UNPLANNED event involving one or more of the following:

(1) Automatic turbine trip at >25% reactor thermal power
(2) Electrical load rejection >25% full electrical load
(3) Plant runback
(4) Reactor trip
(5) Safety injection system actuation
(6) >10% thermal power oscillations
(7) Loss of decay heat removal in Mode 4 ("Significant Transient" is NOT used in any Mode 5 or 6 EAL)

3.1.34 **Site Boundary -** That area, including the PROTECTED AREA that extends 4400 feet or 0.83 miles in a circle around the Reactor Building. Also referred to as the Owner Controlled Area.

3.1.35 **Strike Action -** Is a work stoppage within the PROTECTED AREA by a body of workers to enforce compliance with demands made. The strike actions must threaten to interrupt normal Plant operations.

3.1.36 **Thyroid CDE Dose -** Dose to the thyroid due to intake of radioactive iodine.

3.1.37 **Total Effective Dose Equivalent (TEDE -** The sum of external dose (DDE) and the equivalent amount of whole body dose due to individual organ uptakes.

3.1.38 **Unplanned -** An event or action is UNPLANNED if it is NOT the expected result of normal operations, testing, or maintenance. Events that result in corrective or mitigative actions being taken in accordance with abnormal or emergency procedures are UNPLANNED.

3.1.39 **Valid -** An indication or report or condition is considered VALID when it is conclusively verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by Plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment (e.g., within 15 minutes).

3.1.40 **Visible Damage -** Damage to equipment or structure that is readily observable without measurements, testing, or analyses. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example system/component damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering due to fire. Example structure damage includes exposed and/or broken rebar, failed supports/pipe hangers, etc. Surface blemishing (e.g., paint chipping, scratches, concrete spalling) should NOT be included.

3.2 Responsibilities [NOCS 6120, 13010, 14050]

3.2.1 The Emergency Coordinator controls all activities at CR3 during activation of the Radiological Emergency Response Plan.

3.2.2 The Emergency Coordinator shall NOT delegate the decisions related to classification of the emergency condition. [NOCS 9010]

3.2.3 The Emergency Coordinator shall NOT delegate the decisions related to notification and protective action recommendations to State and Local authorities who implement off-site emergency measures, until the EOF Director communicates to the Emergency Coordinator the EOF accepts the State notification and Protective Action Recommendations (PARs) responsibilities. At this time, the EOF completes the Florida Nuclear Plant Emergency Notification Form. [NOCS 9010]

3.2.4 Upon arrival on-site, the Plant General Manager (PGM) or designated alternate contacts the Control Room Emergency Coordinator or goes to the Control Room and receives a briefing about the status of the emergency condition and the implementation of the Radiological