Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	003 K3.01	
	Importance Rating	3.7	

K/A Statement

Knowledge of the effect that a loss or malfunction of the RCPS will have on the following: RCS

Proposed Question: RO 1 Rev: 0

The following conditions exist:

- The plant is in Mode 3 with a cooldown in progress with 3 RCPs running.
- RCP 2A was secured 2 minutes ago.

Which ONE of the following conditions would result in a TOTAL loss of RCS flow?

A. RCP 1B Oil Pressure is LOW.

B. RCP 1A Oil Reservoir level is LOW.

C. RCP 2B Bearing Temperature is HIGH.

D. RCP 2A has Reverse Rotation Detected.

Proposed Answer: D

Explanation (Optional):

A. INCORRECT: Low oil pressure condition requires stopping the affected pump ONLY.
B. INCORRECT: Low oil level condition may require stopping the affected pump ONLY.
C. INCORRECT: High temperature condition requires stopping the affected pump ONLY.
D. CORRECT: Anti-Reverse Rotation Device is defective and the RCP 2A is running in reverse which requires securing ALL running RCPs – Loss of RCS flow.

Technical Reference(s) SD-RCS Table 2
OP-901-130 E_5 (Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-RCP00 Obj 7 (As available)

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 3

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	003 K5.05	
	Importance Rating	2.8	

K/A Statement

Knowledge of the operational implications of the following concepts as they apply to the RCPS: The dependency of RCS flow rates upon the number of operating RCPs

Proposed Question: RO 2 Rev: 0

The following plant conditions exist:

- Cooldown to Mode 5 is in progress
- Boration to Refuel Boron requirements is in progress
- All RCPs are running

Which ONE of the following RCP pump combinations describes the minimum required RCS flow alignment for RCS flow to ensure proper boron concentration is reached in both loops?

A. 1B and 2B are running

B. 1A and 1B are running

C. 2A and 2B are running

D. All 4 RCPs are required

Proposed Answer: A

Explanation (Optional):

A. CORRECT: This configuration has a RCP in each loop running. Procedure prefers one RCP in each loop remain running for equalization of flow through each SG to ensure proper cooldown and boron mixing.

B. INCORRECT: Procedure prefers one RCP in each loop remain running for equalization of flow through each SG to ensure proper cooldown and boron mixing.

C. INCORRECT: Procedure prefers one RCP in each loop remain running for equalization of flow through each SG to ensure proper cooldown and boron mixing.

D. INCORRECT: This would be optimum but is not the minimum requirement. Procedure prefers one RCP in each loop remain running during cooldown for equalization of flow through each SG to ensure proper cooldown and boron mixing.

Technical Reference(s) OP-010-005 pg 35 & Att 9.15 (Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-RCP00 Obj 10 (As available)

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	004 A3.14	
	Importance Rating	3.4	

K/A Statement

Ability to monitor automatic operation of the CVCS, including: Letdown and charging flows

Proposed Question: RO 3 Rev: 0

The following plant conditions exist:

- Reactor power is 100%
- Charging Pump B is running
- Charging Pump A and AB control switches are in AUTO
- Standby Pump Selector switch is in the AB – A position

Which ONE of the following describes the Letdown flow response following a trip of the running Charging Pump?

A. Lowers to 28 GPM and stabilizes

B. Isolates upstream of the Regenerative Heat Exchanger

C. Lowers to a new value and bypasses the in-service CVCS ion exchanger

D. Lowers to a new value until the standby pump starts, then returns to normal flow

Proposed Answer: B

Explanation (Optional):

A. INCORRECT: Letdown will no longer be cooled and will isolate on a high temperature a short time later producing NO flow condition.
B. CORRECT: Letdown will no longer be cooled and will isolate on a high temperature a short time later producing NO flow condition.
C. INCORRECT: Letdown will no longer be cooled but will bypass Ion Exchanger flow for the in-service ion exchanger a short time later.
D. INCORRECT: Letdown will no longer be cooled and will isolate on a high temperature a short time later producing NO flow condition.

Technical Reference(s) SD-CVC pg 10 (Attach if not previously provided)
OP-500-007 A-1 (including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-CVC00 Obj 2 (As available)

Question Source: Bank # X 08390
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam 2007 RO/SRO Exam

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

10 CFR Part 55 Content: 55.41 7
_____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	005 K1.06	
	Importance Rating	3.5	

K/A Statement

Knowledge of the physical connections and/or cause effect relationships between the RHRS and the following systems: ECCS

Proposed Question: RO 4 Rev: 0

The following plant conditions exist:

- Shutdown Cooling Train A is in service at 4000 gpm
- RCS pressure is 125 PSIA with depressurization in progress for Crud Burst cleanup

Which ONE of the following valves is required to be CLOSED and CAUTION tagged to prevent inadvertent makeup to the RCS from the RWSP as RCS pressure is lowered?

A. SI-109A, LPSI Pump A Suction Isolation

B. SI-116A, LPSI Pump A Minimum Flow Recirculation Stop Check

C. CVC-1661, Purification Ion Exchanger Outlet Header Isolation

D. SI-138A, LPSI Header to RXC Loop 2A Control

Proposed Answer: A

Explanation (Optional):

A. CORRECT: “NOTE” prior to Step 5.3.14 requires SI-109A to be CLOSED and CAUTION tagged during the lineup to prevent makeup to RCS from RWSP.
B. INCORRECT: Step 5.3.11 requires SI-116A to be CLOSED to prevent loss of inventory from the RCS in the SDC lineup..
C. INCORRECT: CVC-1661 is required to be closed to prevent a loss of RCS inventory during SDC purification operations.
D. INCORRECT: SI-138A is required to be throttled closed during RCS mid-loop operation to prevent LPSI Pump vortexing.

Technical Reference(s) OP-009-005 Sec 5.3 (Attach if not previously provided)
SD-SDC Fig 1 (including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-SDC00 Obj 3 (As available)

Question Source: Bank # 07906
Modified Bank # X (Note changes or attach parent)
New

Question History: Last NRC Exam 2006 RO Exam

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

10 CFR Part 55 Content: 55.41 10

Comments:

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

K/A Statement

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

Proposed Question: RO 5 Rev: 0

What is the power supply to SI-138 A, LPSI Header to RC Loop 2A Control Isolation?

A. Bus 213 A

B. Bus 311 A

C. Bus 314 A

D. Bus 315 A

Proposed Answer: B

Explanation (Optional):

A. INCORRECT: SI-138 A is not powered from Bus 213A
B. CORRECT: SI-138 A is powered from Bus 311A
C. INCORRECT: SI-138 A is not powered from Bus 314A
D. INCORRECT: SI-138 A is not powered from Bus 315A

Technical Reference(s) OP-901-313 Att 2 (pg 42) (Attach if not previously provided)
SD-SDC Table 4 (including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: WLP-OPS-SDC00 Obj: 2 (As available)

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

Question History: Last NRC Exam 2009 RO/SRO Q# 31

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

10 CFR Part 55 Content: 55.41 5
_____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	007 A2.01	
	Importance Rating	3.9	

K/A Statement

Ability to (a) predict the impacts of the following malfunctions or operations on the PRTS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Stuck-open PORV or code safety

Proposed Question: RO 6 Rev: 0

The following annunciator alarms are LIT on CP-2:

- Pressurizer Relief Line Temp HIGH
- Quench Tank Level HI/LO
- Quench Tank Temperature Hi

Which ONE of the following conditions <u>and</u> corrective actions would address these alarms?

A. Leaking Pressurizer Safety Valve; Fill and Drain the Quench Tank to reduce temperature.

B. Leaking Quench Tank Drain Valve; Fill and Drain the Quench Tank to reduce temperature.

C. Leaking Reactor Vessel Vent To Quench Tank, enter Containment and isolate the leaking solenoid valve.

D. Stuck Closed Quench Tank Drain Valve; Perform a Reactor Coolant Drain Tank Leakage Diagnostic to determine the source of in-leakage.

Proposed Answer: A

Explanation (Optional):

A. CORRECT: All alarms can be related to a leaking Safety Valve which requires action to maintain Quench Tank level and temperature by fill and drain method.
B. INCORRECT: Quench Tank LOW level would indicate a problem with the Quench Tank Drain Valve, but does NOT relate to Pressurizer Relief Line or Quench Tank HIGH Temp.
C. INCORRECT: A leaking Head vent would not cause a PZR Relief Line Temperature High alarm but would give the remaining alarms.
D. INCORRECT: Quench Tank HIGH level would NOT indicate a problem with the Quench Tank Drain Valve since it does NOT function in AUTO.

Technical Reference(s) OP-901-111
OP-500-008 A-2, D-2, F-1
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: WLP-OPS-RCS00 Obj 2 (As available)

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	008 K4.09	
	Importance Rating	2.7	

K/A Statement

Knowledge of CCWS design feature(s) and/or interlock(s) which provide for the following:
The "standby" feature for the CCW pumps

Proposed Question: RO 7 Rev: 0

Which ONE of the following conditions will send an automatic start signal to ACCW Pump A?

A. Low ACCW system pressure

B. High ACCW system temperature

C. Dry Cooling Tower A bypass opens

D. Low Component Cooling Water flow

Proposed Answer: A

Explanation (Optional):

A. CORRECT: ACCW pumps start on system low pressure.
B. INCORRECT: ACCW pumps start on high CCW temperature.
C. INCORRECT: ACCW pumps are NOT started on DCT in bypass.
D. INCORRECT: ACCW pumps are NOT started on Low CCW flow.

Technical Reference(s) SD-CC pg 45-46 and Table 1.28 (Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: WLP-OPS-CC00 Obj 3 (As available)

Question Source: Bank # 2136-A

Modified Bank # X (Note changes or attach parent)

New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 8

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	008 A3.04	
	Importance Rating	2.9	

K/A Statement

Ability to monitor automatic operation of the CCWS, including: Requirements on and for the CCWS for different conditions of the power plant

Proposed Question: RO 8 Rev: 0

The following plant conditions exist:

- Dry Cooling Tower Fans 1 – 5 A are running in SLOW.
- Component Cooling Water Heat Exchanger A Outlet temperature rose to 93.
- 15 minutes later, CCW Heat Exchanger A Outlet Temperature drops from 92 °F to 91° F and stabilizes.

Which ONE of the following conditions identifies the number of Train A Dry Cooling Tower Fans that will be running and their respective speed(s) at the 15 minute point? (Assume ALL DCT Fans are operable and will function as designed)

	FANS	**SPEED**
A.	1 thru 15	SLOW
B.	1 thru 15	FAST
C.	1 thru 5; 6 thru 15	FAST; SLOW
D.	1 thru 5; 6 thru 15	SLOW; FAST

Proposed Answer: C

Explanation (Optional):

A. INCORRECT: DCT fans cycle on 1 thru 15 every 60 seconds in SLOW until ALL fans are running, BUT continue to cycle on 1 thru 15 every 60 seconds in FAST until temperature is less than 92° F.

B. INCORRECT: DCT fans cycle on 1 thru 15 every 60 seconds in SLOW until ALL fans are running, THEN cycle on 1 thru 15 every 60 seconds in FAST until temperature is less than 92° F.

C. CORRECT: DCT fans cycle on 1 thru 15 every 60 seconds in SLOW until ALL fans are running, THEN cycle on 1 thru 15 every 60 seconds in FAST until temperature is less than 92° F.

D. INCORRECT: DCT fans cycle on 1 thru 15 every 60 seconds in SLOW until ALL fans are running, BUT cycle on 1 thru 15 every 60 seconds in FAST until temperature is less than 92° F.

Technical Reference(s) SD-CC pg 19-20 and 73 (Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: WLP-OPS-CC00 Obj 3 (As available)

Question Source:

Bank #		6065A
Modified Bank #	X	(Note changes or attach parent)
New		

Question History: Last NRC Exam N/A

Question Cognitive Level:

Memory or Fundamental Knowledge	
Comprehension or Analysis	X

10 CFR Part 55 Content: 55.41 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	010 A1.06	
	Importance Rating	3.1	

K/A Statement

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PZR PCS controls including: RCS heatup and cooldown effect on pressure

Proposed Question: RO 9 Rev: 0

Plant conditions are as follows:

- RCS pressure is 160 psia
- Pressurizer level is 98% and lowering
- The crew is drawing a Pressurizer bubble in accordance with OP-001-001, Reactor Coolant System Fill and Vent

Which of the following describes how the operator knows that a bubble has been formed in the Pressurizer during this evolution?

A. Pressurizer Backup heaters cycle off.

B. Pressurizer water temperature reaches 212 °F.

C. Pressurizer water temperature reaches saturation temperature for RCS pressure.

D. Pressurizer Pressure no longer drops while Pressurizer level is being lowered.

Proposed Answer: D

Explanation (Optional):

A. Incorrect: Pressurizer pressure will not be rising during this evolution.

B. Incorrect: The Pressurizer bubble does not form at 212 °F for the conditions given.

C. Incorrect: OP-001-001 directs raising Pressurizer temperature to this point, and then directs lowering Pressurizer level while monitoring Pressurizer pressure.

D. Correct: OP-001-001 has a note with this information. While raising Letdown flow and lowering Pressurizer level, pressure will stop dropping when the Pressurizer bubble is formed.

Technical Reference(s) OP-001-001 Step 6.6 (Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: _____ (As available)

Question Source: Bank # _____ 1452-A

Modified Bank # X (Note changes or attach parent)

New _____

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	010 G2.1.25	
	Importance Rating	3.9	

K/A Statement

Pressurizer Pressure Control System: Ability to interpret reference materials, such as graphs, curves, tables, etc.

Proposed Question: RO 10 Rev: 0

The plant is performing a Cooldown to Cold Shutdown in accordance with OP-010-005 using Att 9.4, Cooldown to Cold Shutdown (Mode 4 to Mode 5). The CRS has directed you to monitor the Pressurizer Cooldown evolution using Att. 9.5, Pressurizer Saturation & P_{sat} + 100 PSIA curve.

Which ONE of the following data points indicates an excessive pressure condition requiring operator action?

	Press (PSIA)	Temp (°F)
A.	167	308
B.	216	347
C.	319	386
D.	348	415

Proposed Answer: C

Explanation (Optional):

A. INCORRECT: :Pressure/Temperature conditions are met on the P_{sat} + 100 PSIA curve with NO entry into the NOT allowed region on Att 9.5.
B. INCORRECT: :Pressure/Temperature conditions are met on the P_{sat} + 100 PSIA curve with NO entry into the NOT allowed region on Att 9.5.
C. CORRECT: Pressure/Temperature condition exceeds the P_{sat} + 100 PSIA curve with entry into the NOT allowed region on Att 9.5.
D. INCORRECT: :Pressure/Temperature conditions are met on the P_{sat} + 100 PSIA curve with NO entry into the NOT allowed region on Att 9.5.l

Technical Reference(s) OP-010-005 Step 3.2.29 and Att 9.5 (Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: OP-010-005 Att 9.5

Learning Objective: WLP-OPS-PLC00 Obj 3 (As available)

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	012 A1.01	
	Importance Rating	2.9	

K/A Statement

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RPS controls including: Trip setpoint adjustment

Proposed Question: RO 11 Rev: 0

The following plant conditions exist:

- The plant is operating at 100% with NOT and NOP
- Performance of OP-903-107, Plant Protection System Channel A Functional Test, is in progress

The NPO reports that he inadvertently depressed the LOW PZR PRESS Setpoint Reset pushbutton.

Which of the following pressures (PSIA) indicates the approximate value at which Channel A would generate a low Pressurizer pressure trip?

A. 1850

B. 1684

C. 1484

D. 1284

Proposed Answer: B

Explanation (Optional):

A. INCORRECT: The manual RESET pushbutton will only lower the existing setpoint of a maximum of 400 PSIA below existing RCS pressure. The setpoint has a ceiling of 1684 PSIA. This number would be plausible if the candidate does not realize that there is an upper ceiling on the setpoint.

B. CORRECT: The manual RESET pushbutton will only lower the existing setpoint of a maximum of 400 PSIA below existing RCS pressure. The setpoint has a ceiling of 1684 PSIA.

C. INCORRECT: The manual RESET pushbutton will only lower the existing setpoint of a maximum of 400 PSIA below existing RCS pressure. The setpoint has a ceiling of 1684 PSIA. This number is 200 psia below existing setpoint and would be plausible if the candidate believes that it works like the Steam Pressure Lo setpoint which is 200 psia and uses the setpoint instead of process pressure to determine the new setpoint.

D. INCORRECT: The manual RESET pushbutton will only lower the existing setpoint of a maximum of 400 PSIA below existing RCS pressure. The setpoint has a ceiling of 1684 PSIA. This number is 400 psia below given setpoint and is plausible if the candidate uses the setpoint vs. process pressure to determine the new setpoint.

Technical Reference(s) SD-PPS pg 33 & Fig 30 (Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: WLP-OPS-PPS00 Obj 3 (As available)

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	013 K1.12	
	Importance Rating	4.1	

K/A Statement

Knowledge of the physical connections and/or cause effect relationships between the ESFAS and the following systems: ED/G

Proposed Question: RO 12 Rev: 0

The following plant conditions exist:

- EDG 'A' is running paralleled in TEST mode for surveillance OP-903-068
- A SIAS actuation occurred with **NO** Loss of Offsite Power condition

The EDG output breaker will ___(1)___ and the ESFAS loads will be started ___(2)___ .

	(1)	(2)
A.	OPEN	by the sequencer
B.	OPEN	Immediately
C.	stay CLOSED	by the sequencer
D.	stay CLOSED	Immediately

Proposed Answer: A

Explanation (Optional):

A. CORRECT: EDG output breaker OPENs and after 2 seconds the sequencer loading begins since NO undervoltage conditions existed on the safety bus.

B. INCORRECT: EDG output breaker OPENs and after 2 seconds the sequencer loading begins since NO undervoltage conditions existed on the safety bus to ensure NO excess loading if IMMEDIATELY started.

C. INCORRECT: EDG output breaker OPENs and after 2 seconds the sequencer loading begins since NO undervoltage conditions existed on the safety bus.

D. INCORRECT: EDG output breaker OPENs and after 2 seconds the sequencer loading begins since NO undervoltage conditions existed on the safety bus to ensure NO excess loading if IMMEDIATELY started..

Technical Reference(s) SD-EDG pg 44 - 45 (Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: WLP-OPS-EDG00 Obj 3 (As available)

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	013 K6.01	
	Importance Rating	2.7	

K/A Statement

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

Proposed Question: RO 13 Rev: 0

Regarding a single channel of Plant Protection System, which ONE of the following PPS sensor failures would cause a Reactor Trip signal but NO ESFAS signal from the channel if it failed to ZERO?

A. RCP 1A Speed

B. S/G 1 Level

C. S/G 1 Pressure

D. Wide Range Pressurizer Pressure

Proposed Answer: A

Explanation (Optional):

RPS and ESFAS channels share the same sensors

A. CORRECT: Loss of the 1A RCP speed output (zero speed) signal will cause a reactor trip signal but is not an input to ESFAS.

B. INCORRECT: Loss of the S/G 1 Level output signal will cause a reactor trip and ESFAS (EFAS-1) signal

C. INCORRECT: Loss of the S/G 1 Pressure output signal will cause a reactor trip and ESFAS (MSIS) signal

D. INCORRECT: Loss of the WR Pressurizer pressure output signal will cause a reactor trip and ESFAS (SIAS/CIAS) signal.

Technical Reference(s) SD-PPS Fig 15 & 32 (Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: WLP-OPS-PPS00 Obj 3 (As available)

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	022 A4.03	
	Importance Rating	3.2	

K/A Statement

Ability to manually operate and/or monitor in the control room: Dampers in the CCS

Proposed Question: RO 14 Rev: 0

Which ONE of the following methods will OPEN the CCS-102 A(B), Emergency Discharge Damper(s)?

A. Place CCS-102 A(B) control switch to OPEN at CP-18

B. Start Containment Fan Cooler in FAST at CP-18

C. Start Containment Fan Cooler in SLOW at CP-18

D. SIAS

Proposed Answer: D

Explanation (Optional):

A. INCORRECT: NO control switch available
B. INCORRECT: NO function tied to fan operation
C. INCORRECT: NO function tied to fan operation
D. CORRECT: Designed to OPEN on SIAS to direct cool air to top of containment following a LOCA or MSLB event.

Technical Reference(s) SD-CCS pg 10 and Fig 2 (Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: WLP-OPS-CCS00 Obj 5 (As available)

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 9

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	026 K3.02	
	Importance Rating	4.2	

K/A Statement

Knowledge of the effect that a loss or malfunction of the CSS will have on the following:
Recirculation spray system

Proposed Question: RO 15 Rev: 0

The following plant conditions exist:

- A large break LOCA occurred
- Containment Spray Line A pipe failure occurred in the -35 Wing Area

Which ONE of the following conditions describes the operational concern for the Safety Injection system following RAS initiation?

The ___(1)___ will not have adequate water inventory for automatic operation of the ___(2)___ pumps.

	(1)	(2)
A.	RWSP	LPSI
B.	RWSP	HPSI
C.	SIS sump	LPSI
D.	SIS sump	HPSI

Proposed Answer: D

Explanation (Optional):

A. INCORRECT: RWSP is NOT available once RAS is initiated
B. INCORRECT: RWSP is NOT available once RAS is initiated
C. INCORRECT: SIS sump water will spray into -35 wing area which will cause a loss of inventory until the break can be isolated and loss of inventory for pump suction.
D. CORRECT: SIS sump water will spray into -35 wing area which will cause a loss of inventory until the break can be isolated and loss of inventory for pump suction.

Technical Reference(s) SD-CS pg 7
OP-902-008 pg 30 (IC-2: SI)
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: WLP-OPS-CS00 Obj.5 (As available)

Question Source: Bank # 6504A
Modified Bank # X (Note changes or attach parent)
New

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 1

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	039 K1.08	
	Importance Rating	2.7	

K/A Statement

Knowledge of the physical connections and/or cause-effect relationships between the MRSS and the following systems: MFW

Proposed Question: RO 16 Rev: 0

Steam supply to both Steam Generator Feedwater pump turbines comes from _____ and _____ .

A. Main Steam Header A; MSR A header.

B. Main Steam Header A; MSR B header.

C. Main Steam Header B; MSR A header.

D. Main Steam Header B; MSR B header.

Proposed Answer: D

Explanation (Optional):

A. INCORRECT: The SGFP does **NOT** receive steam from the 'A' Main Steam or the 'A' MSR headers.
B. INCORRECT: The SGFP does **NOT** receive steam from the 'A' Main Steam but does receive steam from the 'B' MSR headers.
C. INCORRECT: The SGFP does receive steam from the 'B' Main Steam but does **NOT** receive steam from the 'A' MSR headers.
D. CORRECT: The SGFP does receive steam from both the 'B' Main Steam and the 'B' MSR headers based upon plant power.

Technical Reference(s) SD-FWP Fig 4A & pg 33 (Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: WLP-OPS-FWP Obj 1 (As available)

Question Source: Bank # X 1871A
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 4

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	039 A2.04	
	Importance Rating	3.4	

K/A Statement

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

Proposed Question: RO 17 Rev: 0

The following plant conditions exist:

- Plant is operating at 92% following Main Turbine testing
- MS-319A, Main Steam Bypass 1A valve, fails OPEN

Which ONE of the following conditions states the initial plant response and the required operator action?

A. Turbine Governor Valve position will lower; lower turbine loading to maintain reactor power ≤ 100% and T_{AVG} stable.

B. Reactor power will rise; lower turbine loading to maintain reactor power ≤ 100% and T_{AVG} stable.

C. Turbine Governor Valve position will lower; manually initiate Reactor Power Cutback.

D. Reactor power will rise; manually initiate Reactor Power Cutback.

Proposed Answer: B

Explanation (Optional):

A. INCORRECT: If DEH feedback loops are in service the Governor valves will open to try and compensate for the loss of MW or impulse pressure. If they are not in service the Governor valves will remain as is. The additional steam flow will cause a rise in reactor power which requires operator action to reduce main steam flow to the main turbine.
B. CORRECT: The additional steam flow will cause a rise in reactor power which requires operator action to reduce main steam flow to the main turbine.
C. INCORRECT: If DEH feedback loops are in service the Governor valves will open to try and compensate for the loss of MW or impulse pressure. If they are not in service the Governor valves will remain as is. The additional steam flow will cause a rise in reactor power which requires operator action to reduce main steam flow to the main turbine.
D. INCORRECT: The additional steam flow will cause a rise in reactor power which requires operator action to reduce main steam flow to the main turbine.

Technical Reference(s) SD-SBC pg 27-28 (Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: WLP-OPS-SBC00 Obj 8 (As available)

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

Question History: Last NRC Exam New

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

10 CFR Part 55 Content: 55.41 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	059 K4.05	
	Importance Rating	2.5	

K/A Statement

Knowledge of MFW design feature(s) and/or interlock(s) which provide for the following:
Control of speed of MFW pump turbine

Proposed Question: RO 18 Rev: 0

Which ONE of the following signals or conditions will result in a SG Feedwater Pump A speed reduction to approximately 3900 RPM?

A. Reactor Trip Override

B. SG1 High Level Override

C. One of two Feedwater Flow Demand output signals fails LOW

D. One of two Speed Pickup units to GE Microprocessor fails

Proposed Answer: A

Explanation (Optional):

A. CORRECT: A Rector Trip Override (RTO) signal is initiated on a Reactor Trip to ensure minimum SGFP output to the Steam Generators.
B. INCORRECT: High Level Override (HLO) generates a zero flow demand to close both FWRVs and a zero flow demand to SGFP control but HIGH select circuit is used and the higher of the two signal controls FWCS to non effected SG level.
C. INCORRECT: Feedwater Flow Demand controls operate on a HIGH select circuit that uses the higher of the two signal to control FWCS.
D. INCORRECT: GE Microprocessor uses two shaft speed pickup units and uses the high select for speed input.

Technical Reference(s) SD-FWC Figs 22 - 24 (Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: WLP-OPS-FW00 Obj. 4 (As available)

Question Source: Bank # 3163A

Modified Bank # X (Note changes or attach parent)

New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	061 K6.02	
	Importance Rating	2.6	

K/A Statement

Knowledge of the effect of a loss or malfunction of the following will have on the AFW components: Pumps

Proposed Question: RO 19 Rev: 0

The AB EFW pump has tripped on OVERSPEED. The NPO was directed to take the Turbine Stop Valve (MS-416) control switch to the OPEN position.

Which ONE of the following reasons would explain why the valve position did NOT change position following this action?

A. The control switch must be taken to the CLOSE position prior to going to the OPEN position.

B. The mechanical overspeed tappet nut and linkage must be locally reset.

C. The emergency steam line drain valve (MS-407) must be closed first

D. The steam supply valves (MS-401 A & B) must be closed first.

Proposed Answer: B

Explanation (Optional):

A. INCORRECT: A mechanical overspeed trip locks out MS-416 from opening in the control room until the mechanical overspeed tappet nut and linkage have been reset.
B. CORRECT: A mechanical overspeed trip locks out MS-416 from opening in the control room until the mechanical overspeed tappet nut and linkage have been reset.
C. INCORRECT: The valve must be reclosed after a high level condition has cleared but is NOT linked to the overspeed function.
D. INCORRECT: Not closing MS-401 A & B will not prevent MS-416 from operating. However, the valves must be closed to reset the ramp generator to prevent another overspeed condition when restoring the pump.

Technical Reference(s) SD-EFW pg 16 (Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: WLP-OPS-EFW00 Obj 5 (As available)

Question Source: Bank # X 139A
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	061 G2.1.28	
	Importance Rating	4.1	

K/A Statement

Auxiliary / Emergency Feedwater System: Knowledge of the purpose and function of major system components and controls.

Proposed Question: RO 20 Rev: 0

Which ONE of the following states the purpose of the bypass lines that are provided from Main Feedwater to the piping downstream of the Emergency Feedwater Pumps?

A. Provide a cleanup path for the emergency feedwater system.

B. Provide a means to test the EFW system check valves in Modes 4 – 6.

C. Keep the EFW lines filled and pressurized to prevent water hammer on EFAS initiation.

D. Keep the EFW discharge lines warm to prevent thermal shock to the SG Feedwater ring.

Proposed Answer: C

Explanation (Optional):

A. INCORRECT: This is a keep filled line with NO cleanup feature provided.
B. INCORRECT: While this line contains a check valve for prevention of backflow, no test feature is provided.
C. CORRECT: Bypass line is provided to ensure the EFW piping is full of water at all times. This feature prevents a water hammer even upon actuation. In addition, the line has a check valve to prevent backflow into the Main Feedwater system
D. INCORRECT: This is a keep filled line with NO warming feature provided

Technical Reference(s): OP-003-033 3.2.2 (Attach if not previously provided)
SD-EFW pg 29
(including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: WLP-OPS-EFW00 Obj 7 (As available)

Question Source: Bank # X 108A
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	062 G2.4.46	
	Importance Rating	4.2	

K/A Statement

A.C. Electrical Distribution: Ability to verify that the alarms are consistent with the plant conditions.

Proposed Question: RO 21 Rev: 0

The ATC reports that indication has been lost on the following CP-4 components:

- CVC-101, Letdown to Regen HX from RCS Loop 2B
- CVC-109, Letdown HX Inlet Header Isolation
- CVC-510, Borated Water to VCT Header Isolation

All 3 components indicate closed on the PMC.

Which ONE of the following SUPS Trouble alarms would be expected for the CVC system indications?

A. SA

B. SB

C. SAB

D. SMD

Proposed Answer: A

Explanation (Optional):

A. CORRECT: A loss of SUPS SA power supply will cause the CVC valves to reposition and CC-636 should AUTO close when CVC-109 goes close but that does NOT occur on loss of AC power.
B. INCORRECT: A loss of SUPS SB does not affect these valves.
C. INCORRECT: A loss of SUPS SAB does not affect these valves.
D. INCORRECT: A loss of SUPS SMD does not affect these valves.

Technical Reference(s): OP-500-004 Att 4.114 (Attach if not previously provided)

OP-901-312 B_2.8 (including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: WLP-OPS-PPO30 Obj 4 (As available)

Question Source: Bank # _____

Modified Bank # _____ (Note changes or attach parent)

New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	063 K2.01	
	Importance Rating	2.9	

K/A Statement

Knowledge of bus power supplies to the following: Major DC loads

Proposed Question: RO 22 Rev: 0

Loss of which DC bus will cause a failure of MS-401A, EFW Pump AB Turbine Steam Supply valve, to perform its AUTO function?

A. A-DC

B. B-DC

C. AB-DC

D. TGB-DC

Proposed Answer: C

Explanation (Optional):

A. INCORRECT: MS-401A(B) motor operated valves receive power from the AB-DC bus.
B. INCORRECT: MS-401A(B) motor operated valves receive power from the AB-DC bus.
C. CORRECT: MS-401A(B) motor operated valves receive power from the AB-DC bus.
D. INCORRECT: MS-401A(B) motor operated valves receive power from the AB-DC bus.

Technical Reference(s) SD-DC pg 28 (Attach if not previously provided)

OP-901-313 Att 5 (pg 50) (including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: WLP-OPS-DC Obj. 8 (As available)

WLP-OPS-GEN Obj. 10

Question Source: Bank # 3297-A

Modified Bank # X (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	064 A3.07	
	Importance Rating	3.6	

K/A Statement

Ability to monitor automatic operation of the ED/G system, including: Load sequencing

Proposed Question: RO 23 Rev: 0

The following plant conditions exist:

- EDG A & B have started due to a Loss of Offsite Power event
- Transformer input to MCC-315A developed a direct short to ground upon being re-energized by the Sequencer 1 second Load Block (S1 relay)
- An undervoltage condition exists on 4160 Volt Bus 3A

Which ONE of the following conditions describes the Sequencer response?

A. Immediately stops the automatic loading process due to Under Voltage Override (UVO) condition.

B. Immediately stops the automatic loading process due to Sequencer Lockout (SLO) condition.

C. Stops the automatic loading process when the 7 second Load Block (S3 relay) is reached.

D. Stops the automatic loading process when the 17 second Load Block (S4 relay) is reached.

Proposed Answer: D

Explanation (Optional):

A. INCORRECT: : A SLO condition will occur at Load Block S4 due to the undervoltage condition on Bus 3A

B. INCORRECT: : A SLO condition will occur at Load Block S4 due to the undervoltage condition on Bus 3A

C. INCORRECT: : A SLO condition will occur at Load Block S4 due to the undervoltage condition on Bus 3A

D. CORRECT: A SLO condition will occur at Load Block S4 due to the undervoltage condition on Bus 3A

Technical Reference(s) SD-EDG pg 46 (Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: WLP-OPS-EDG Obj 3 (As available)

Question Source: Bank # X 2226-A
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	073 K5.03	
	Importance Rating	2.9	

K/A Statement

Knowledge of the operational implications as they apply to concepts as they apply to the PRM system: Relationship between radiation intensity and exposure limits

Proposed Question: RO 24 Rev: 0

The CROAI Radiation monitors isolate the Control Room Ventilation (HVC) on a high radiation level to prevent Control Room staff from receiving a maximum dose of ___(1)___ for the ___(2)___ of the event.

	(1)	(2)
A.	5 rem	duration
B.	5 rem	first 2 hours
C.	2 rem	duration
D.	2 rem	first 2 hours

Proposed Answer: A

Explanation (Optional):

A. CORRECT: The isolation of the Control Room for the duration of the event ensures no one individual will exceed the Federal Limit of 5 rem.
B. INCORRECT: Wrong duration. The isolation of the Control Room for the duration of the event ensures no one individual will exceed the Federal Limit of 5 rem.
C. INCORRECT: Wrong dose. The isolation of the Control Room for the duration of the event ensures no one individual will exceed the Federal Limit of 5 rem.
D. INCORRECT: Wrong dose, wrong duration. The isolation of the Control Room for the duration of the event ensures no one individual will exceed the Federal Limit of 5 rem.

Technical Reference(s) FSAR 6.4.1 (Attach if not previously provided)

10 CFR 50.67(b)(iii) (including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: WLP-OPS-RAD02 Obj 12 (As available)

Question Source: Bank # _____

Modified Bank # _____ (Note changes or attach parent)

New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 12

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	076 A1.02	
	Importance Rating	2.6	

K/A Statement

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SWS controls including: Reactor and turbine building closed cooling water temperatures

Proposed Question: RO 25 Rev: 0

The plant is currently producing 500 MWe output with all systems operating per design.

Which ONE of the following MANUAL operator actions is required as the plant ramps to 1100 MWe output in order to maintain Turbine Cooling Water system temperature and pressure constant?

Throttle the TCW Temperature Control Valve (TC-147) in the _____ direction and throttle the TCW Pressure Control Valve (TC-135) in the _____ direction.

A. OPEN ; OPEN

B. OPEN ; CLOSED

C. CLOSED ; OPEN

D. CLOSED ; CLOSED

Proposed Answer: B

Explanation (Optional):

A. INCORRECT: Need more cooling as heat input rises during power escalation and less pressure control to maintain system in a balanced condition.
B. CORRECT: Need more cooling as heat input rises during power escalation and less pressure control to maintain system operation in a balanced condition.
C. INCORRECT: Need more cooling as heat input rises during power escalation and less pressure control to maintain system in a balanced condition.
D. INCORRECT: Need more cooling as heat input rises during power escalation and less pressure control to maintain system in a balanced condition.

Technical Reference(s) OP-901-512 E_3 SD-TCW pg 11 (Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: WLP-OPS-TC00 Obj 4 (As available)

Question Source: Bank # X 7045A
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 4

Comments:

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

K/A Statement

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

Proposed Question: RO 26 Rev: 0

Plant conditions are as follows:

- The plant is in day 18 of a refueling outage
- Shutdown Cooling Train B is in service and Train A is in standby
- SI-129 B, LPSI Pump B Flow Control, is closed
- SI-415 B, Shutdown Cooling HX B Temperature Control, is 50% open
- CC-963B, SDC HX B TCV is full open
- SI Train B flow is 3000 gpm

An Instrument Air line ruptures in the RCA, causing Instrument Air pressure to drop to 20 psig. (assume pressure drop duration depletes any accumulator capacity)

Based on these conditions, Safety Injection Train B flow will _____(1)_____ and RCS temperature will _____(2)_____

	(1)	(2)
A.	drop	drop
B.	drop	rise
C.	rise	drop
D.	rise	rise

Proposed Answer: D

Explanation (Optional):

A. INCORRECT: SI-129 B fails OPEN on a loss of air and SI-415 will not respond to air leak causing SDC flow to rise but the flow is bypassing the Heat Exchanger.
B. INCORRECT: SI-129 B fails OPEN on a loss of air and SI-415 will not respond to air leak causing SDC flow to rise but the flow is bypassing the Heat Exchanger.
C. INCORRECT SI-129 B fails OPEN on a loss of air and SI-415 will not respond to air leak causing SDC flow to rise but the flow is bypassing the Heat Exchanger.
D. CORRECT: SI-129 B fails OPEN on a loss of air and SI-415 will not respond to air leak causing SDC flow to rise but the flow is bypassing the Heat Exchanger.

Technical Reference(s) SD-SDC Pages 9 & 13 (Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: WLP-OPS-AIR Obj. 5 (As available)

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	064 A4.01	
	Importance Rating	4.0	

K/A Statement

Ability to manually operate and/or monitor in the control room: Local and remote operation of the EDG

Proposed Question: RO 27 Rev: 0

The following plant conditions exist:

- Plant was operating at 100% when a LOOP event occurred
- EDG B started as designed

Which of the following conditions will cause an EDG B trip?

A. BOP presses the Diesel Generator B Trip pushbutton on CP-1

B. EDG B Oil Pressure drops to less than 20 psig

C. EDG B Jacket Water Temperature is 210° F

D. EDG B Speed rises to greater than 680 rpm

Proposed Answer: D

Explanation (Optional):

A. INCORRECT: Diesel Generator B Trip pushbutton on CP-1 is NOT functional during LOOP event
B. INCORRECT: Engine Lube Oil pressure trip limit is 30 PSIG but NOT functional during LOOP event
C. INCORRECT: Jacket Water Temperature trip limit is 205 °F but NOT functional during LOOP event
D. CORRECT: Overspeed condition exist above 660 rpm to TRIP the EDG

Technical Reference(s) OP-009-002 Section 8.8 (Attach if not previously provided)

SD-EDG Table 2 & Fig 17 (including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: WLP-OPS-EDG00 Obj 2 (As available)

Question Source: Bank # _____

Modified Bank # _____ (Note changes or attach parent)

New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 8

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	103 K4.01	
	Importance Rating	3.0	

K/A Statement

Knowledge of containment system design feature(s) and/or interlock(s) which provide for the following: Vacuum breaker protection

Proposed Question: RO 28 Rev: 0

Which ONE of the following conditions describes the setpoint and design basis for the Containment Vacuum Relief valve operation?

At ___(1)___ INWD Containment to Annulus differential pressure, CVR-101(201) open to ensure Containment internal pressure does not become more ____(2)____ than the Containment design limit.

	(1)	(2)
A.	5.5	negative
B.	5.5	positive
C.	8.5	negative
D.	8.5	positive

Proposed Answer: C

Explanation (Optional):

A. INCORRECT: Containment Vacuum Relief valve operation does not occur at 5.5 INWD to ensure the design Containment negative pressure is not exceeded.
B. INCORRECT: Containment Vacuum Relief valve operation does not occur at 5.5 INWD to ensure the design Containment negative pressure is not exceeded.
C. CORRECT: Containment Vacuum Relief valve auto operation occurs at 8.5 INWD to ensure the design Containment negative pressure is not exceeded.
D. INCORRECT: Containment Vacuum Relief valve auto operation occurs at 8.5 INWD to ensure the design Containment negative pressure is not exceeded.

Technical Reference(s) OP-008-005 Section 8.1 & 9.0 (Attach if not previously provided)

FSAR 6.2.3 (including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: WLP-OPS-CB Obj. 2 (As available)

Question Source: Bank # _____

Modified Bank # _____ (Note changes or attach parent)

New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	001 K1.05	
	Importance Rating	4.5	

K/A Statement

Knowledge of the physical connections and/or cause effect relationships between the CRDS and the following systems: NIS and RPS

Proposed Question: RO 29 Rev: 0

Which ONE of the following output signals is used as an input (direct or calculated) signal to prohibit ALL automatic CEA movement below 15% reactor power?

A. RCS Average Temperature

B. Pressurizer Pressure

C. RCS Cold Leg Temperature

D. Reactor Power (control channel)

Proposed Answer: D

Explanation (Optional):

A. INCORRECT: Average Reactor Coolant Temperature output to SBCS for quick opening permissive.
B. INCORRECT: Pressurizer Pressure output to the Summed Error calculation.
C. INCORRECT: RCS Cold Leg Temperature output to Tave Loop calculation.
D. CORRECT: Power Range Control Channel output is used to produce the reactor power error calculation which prohibits auto CEA movement below 15% reactor power.

Technical Reference(s) SD-RR Fig 2 and pg 22 (Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: WLP-OPS-RR00 Obj 3 (As available)

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 6

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	016 K4.03	
	Importance Rating	2.8	

K/A Statement

Knowledge of NNIS design feature(s) and/or interlock(s) which provide for the following:
Input to control systems

Proposed Question: RO 30 Rev: 0

RCS Reference Temperature (Tref) signal is generated in the ____(1)_____ system as a function of the ___(2)____ input.

	(1)	(2)
A.	Reactor Regulating	Main Steam Crossover Header pressure
B.	Steam Bypass Control	Turbine First Stage pressure
C.	Reactor Regulating	Turbine First Stage pressure
D.	Steam Bypass Control	Main Steam Crossover Header pressure

Proposed Answer: C

Explanation (Optional):

A. INCORRECT: : Reactor Regulating System takes input from Turbine First Stage Pressure to calculate the current Tref but does not use MS Crossover Header Pressure.

B. INCORRECT: Steam Bypass Control uses Turbine First Stage Pressure as an input to the AMI circuitry but it does not calculate Tref in Steam Bypass Control.

C. CORRECT: Reactor Regulating System take input from Turbine First Stage Pressure to calculate the current Tref.

D. INCORRECT: Steam Bypass Control input from Main Steam Crossover Header pressure for Master Controller permissive input is NOT part of RR.

Technical Reference(s) SD-RR Fig 02
SD-SBC Fig 02 (Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: WLP-OPS-RR00 Obj 2 (As available)

Question Source: Bank # X 1267B
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	017 A3.01	
	Importance Rating	3.6	

K/A Statement

Ability to monitor automatic operation of the ITM system including: Indications of normal, natural, and interrupted circulation of RCS

Proposed Question: RO 31 Rev: 0

The plant has been operating for 100 days when a shutdown was performed to repair a Reactor Coolant Pump seal. The shutdown was completed on January 4th at 1800 hours. All RCPs are secured and the drain down to replace the seal has been completed.

At 2000 hours on January 8th a total loss of Shutdown Cooling occurs.

Representative Core Exit Thermocouple temperature is 123 degrees °F at the start of the event.

Due to the loss of Shutdown Cooling flow, the RCS temperature will rise to 212 °F by _____ hours on January 8th. (ASSUME: NO operator action.)

A. 2010

B. 2015

C. 2020

D. 2025

Proposed Answer: B

Explanation (Optional):

A. INCORRECT: IF the plant has been shutdown 3 days which equals a value of 6.5 deg F per minute. (10 mins X 6.5 deg F/min) + 123 deg F = 188 deg F final temp.
B. CORRECT: The plant has been shutdown 4 days which equals a value of 6 deg F per minute. **(15 mins X 6 deg F/min) + 123 deg F = 212** deg F final temp.
C. INCORRECT: IF the plant has been shutdown 5 days which equals a value of 5.5 deg F per minute. (20 mins X 5.5 deg F/min) + 123 deg F = 233 deg F final temp.
D. INCORRECT: IF the plant has been shutdown 6 days which equals a value of 5 deg F per minute. (25 mins X 5 deg F/min) + 123 deg F = 248 deg F final temp.

Technical Reference(s) OP-901-131 Att 2 (Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: OP-901-131 Att 2

Learning Objective: WLP-OPS-INI00 Obj 8 (As available)
WLP-OPS-REQ21 Obj 1

Question Source: Bank # 3500A
Modified Bank # X (Note changes or attach parent)
New

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	028 A2.03	
	Importance Rating	3.4	

K/A Statement

Malfunctions or operations on the HRPS; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: The hydrogen air concentration in excess of limit flame propagation or detonation with resulting equipment damage in containment

Proposed Question: RO 32 Rev: 0

The following plant conditions exist:

- Plant tripped due to a Large Break LOCA event and TSC directed that both Hydrogen Recombiners were to be placed in service
- Hydrogen Recombiner A failed and was removed from service
- Hydrogen Recombiner B is operating with a setting of 25 KW output
- Hydrogen Analyzers indicate Containment H_2 concentration of 4.2% and rising slowly.

The CRS directs you to raise Hydrogen Recombiner B output in accordance with OP-008-006 using Attachment 11.2 to reduce Hydrogen concentration.

Post-LOCA Containment pressure is 17.7 PSIA and Pre-LOCA Containment Temperature was 105° F.

The final setting on the B potentiometer will be ____ KW to establish the required value.

A. 53

B. 55

C. 57

D. 59

Proposed Answer: B

Explanation (Optional):

A. INCORRECT: Using Att 11.2 and 11.4, multiply 48 KW by pressure factor (C_p) to get final value: 48 X 1.10 = 53 for (16.7 and 120)
B. CORRECT: Using Att 11.2 and 11.4, multiply 48 KW by pressure factor (C_p) to get final value: **48 X 1.15 = 55** for (17.7 and 105)
C. INCORRECT: Using Att 11.2 and 11.4, multiply 48 KW by pressure factor (C_p) to get final value: 48 X 1.19 = 57 for (18.0 and 90)
D. INCORRECT: Using Att 11.2 and 11.4, multiply 48 KW by pressure factor (C_p) to get final value: 48 X 1.23 = 59 for (18.0 and 60)

Technical Reference(s): OP-902-002 Step 13; OP-008-006 Step 6.1.9; OP-008-006 Att 11.2 & 11.4 (Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: OP-008-006 Att 11.2 & 11.4

Learning Objective: WLP-OPS-HRA00 Obj 3 (As available)

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 8

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	029 G 2.4.30	
	Importance Rating	2.7	

K/A Statement

Containment Purge System: Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator**.**

Proposed Question: RO 33 Rev: 0

With the plant in Mode 1, which of the following conditions would require a 60 day LER for failure to comply with Tech Specs if the condition was not corrected?

A. Pressurizer level stable rose and stabilized at 61%.

B. Containment pressure dropped and stabilized at 14.57 psia.

C. Containment Purge Isolations remained open for 95 hours in a 365 day period.

D. RWSP boron concentration sample results are reported as 2150 ppm.

Proposed Answer: C

Explanation (Optional):

A. INCORRECT: The Tech Spec 3.4.3.1 limit for Pressurizer level is < 62.5 %.
B. INCORRECT: The Tech Spec 3.6.1.4 limit for containment pressure is 14.275 psia.
C. CORRECT: The Tech Spec 3.6.1.7 limit for a Containment Purge Isolation Valve is 90 hours in the previous 365 days.
D. INCORRECT: The Tech Spec 3.5.4 limit for RWSP boron concentration is 2050 ppm.

Technical Reference(s) Tech Spec 3.4.3.1, 3.5.4, 3.6.1.4, 3.6.1.7. (Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: WLP-OPS-TS04 Obj 1 (As available)

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	035 K3.01	
	Importance Rating	4.4	

K/A Statement

Knowledge of the effect that a loss or malfunction of the S/GS will have on the following: RCS

Proposed Question: RO 34 Rev: 0

The following plant conditions exist:

- Mode 3 following a Reactor Trip due to an Excess Steam Demand event in the TGB
- Both ADVs were maintaining SG pressure at 991 psig in AUTO
- Subsequently, both ADVs failed to the CLOSE position

Which ONE of the following temperature (°F) **RISES** corresponds to the MINIMUM RCS temperature rise to reach saturation pressure for the LOWEST lift setpoint for the SG Safety Valves?

A. 12

B. 9

C. 6

D. 3

Proposed Answer: B

Explanation (Optional):

A. INCORRECT: The 2nd safety lifts at 1085 psig (1100 PSIA) which corresponds to 556 °F; therefore a change of 11 °F is required.
B. CORRECT: Current pressure is 991 psig (1006 PSIA) which corresponds to 545 °F and the First safety lifts at 1070 psig (1085 PSIA) which corresponds to 554 °F; therefore a change of **9 °F** is required.
C. INCORRECT: The First safety lifts at 1070 psig (1085 PSIA) which corresponds to 554 °F; therefore a 6 °F change is insufficient
D. INCORRECT: The First safety lifts at 1070 psig (1085 PSIA) which corresponds to 554 °F; therefore a 3 °F change is insufficient

Technical Reference(s) SD-MS Table 1 pg 34 (Attach if not previously provided) (including version/revision number)

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

Learning Objective: WLP-OPS-MS00 Obj 4 (As available)

Question Source: Bank # X 07717
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 1

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	041 G2.4.2	
	Importance Rating	4.5	

K/A Statement

Steam Dump System (SDS) and Turbine Bypass Control: Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.

Proposed Question: RO 35 Rev: 0

The following plant conditions exist:

- Plant is operating at 100%
- RCS Tavg is 573° F
- RCS pressure is 2250 PSIA
- Reactor Cutback is Out of Service
- Reactor Trip on Turbine Trip is ENABLED

Which ONE of the following describes the response of the Steam Bypass Control System immediately following a Turbine Trip?

Valve(s) _____ will quick open.

A. 1 through 3

B. 1 through 5

C. 1 through 6

D. None

Proposed Answer: B

Explanation (Optional):

A. INCORRECT: With RXC out of service and Reactor Power at 100% the Reactor Trip on Turbine Trip is enabled. When the turbine trips the reactor then trips. The reactor trip automatically blocks Quick Open on steam bypass valve 6 With Tavg > 561°F Valves 1,2,3,4,5 will receive quick open signals. Valves 1-3 is plausible because the valves open together on a QO1 signal.

B. CORRECT: With RXC out of service and Reactor Power at 100% the Reactor Trip on Turbine Trip is enabled. When the turbine trips the reactor then trips. The reactor trip automatically blocks Quick Open on steam bypass valve 6 With Tavg > 561°F Valves 1,2,3,4,5 will receive quick open signals. INCORRECT: Valves 1,2,3,4,5 ONLY receive quick open signal with RXC signal OOS and a Reactor Trip is generated.

C. INCORRECT: Valves 1-6 is plausible if the candidate does not know that valve 6 is blocked on a reactor trip.

D. None would be plausible if the candidate misapplies the QO blocks generated by RXC when .a feed pump trips. All 6 valves are blocked from QO in that case. However, this is generated by the RXC system which is out of service. And the event is wrong.

Technical Reference(s) SD-SBC Fig 11 & 21 (Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: WLP-OPS-SBC00 Obj: 5 (As available)

Question Source: Bank # X 07938
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam 2006 RO Exam

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

10 CFR Part 55 Content: 55.41 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	055 K1.06	
	Importance Rating	2.6	

K/A Statement

Knowledge of the physical connections and/or cause effect relationships between the CARS and the following systems: PRM system

Proposed Question: RO 36 Rev: 0

OP-003-001, Condenser Air Evacuation System, requires Condenser Vacuum Pump (1) be operating to provide a suction path for the (2)

	(1)	(2)
A.	A or B	Condenser Air Evacuation WRGM
B.	A or B	Condenser Air Evacuation PIG Rad Monitor
C.	B or C	Condenser Air Evacuation WRGM
D.	B or C	Condenser Air Evacuation PIG Rad Monitor

Proposed Answer: D

Explanation (Optional):

Condenser Air Evacuation Wide Range Was Monitor takes a suction from the common discharge of all 3 Air Evacuation Pumps. The Condenser Air Evacuation PIG Radiation Monitor takes a suction from B and C Air Evacuation Pumps. OP-003-001 specifies that either B or C Pumps must be running to ensure the discharge is monitored by the Condenser Air Evacuation PIG.

Technical Reference(s) SD-AE Fig 1
SD-RMS Table 4 & 5 (Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: WLP-OPS-AE00 Obj 1 (As available)

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	011 A2.04	
	Importance Rating	3.5	

K/A Statement

Ability to (a) predict the impacts of the following malfunctions or operations on the PZR LCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of one, two or three charging pumps

Proposed Question: RO 37 Rev: 0

Plant conditions are as follows:

- Power is 100%
- Charging Pump A is running
- CVC-ILT-0227 fails to 0%

The crew secured Charging Pump A and closed CVC-101 in accordance with OP-901-113, Volume Control Tank Makeup Control Malfunction

Based on these conditions, the first backup Charging Pump will start when Pressurizer level drops ___(1)___ below Pressurizer level setpoint and the crew will be required to ___(2)___ after the Charging Pump start.

	(1)	(2)
A.	3.9%	makeup to the VCT
B.	3.9%	reduce Main Turbine load
C.	2.5%	makeup to the VCT
D.	2.5%	reduce Main Turbine load

Proposed Answer: D

Explanation (Optional):

A. INCORRECT: The first Pump will start at -2.5% below program but the second pump will NOT start until -3.9% below program; Initial operator action is to reduce Main Turbine load due to the boration that occurs.

B. INCORRECT: The first Pump will start at -2.5% below program but the second pump will NOT start until -3.9% below program.

C. INCORRECT: The Initial operator action is to reduce Main Turbine load due to the boration that occurs.

D. CORRECT: The first Pump will start at -2.5% below program; Initial operator action is to reduce Main Turbine load due to the boration that occurs.

Technical Reference(s) SD-CVC pg 23 (Attach if not previously provided)

OP-901-112 C & E_0 (including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: WLP-OPS-CVC Obj 6 (As available)

Question Source: Bank # _____

Modified Bank # _____ (Note changes or attach parent)

New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	086 A4.05	
	Importance Rating	3.0	

K/A Statement

Ability to manually operate and/or monitor in the control room: Deluge valves

Proposed Question: RO 38 Rev: 0

Which of the following is the <u>only</u> air filtration unit that has a Fire Protection Deluge system which can be operated from the Control Room?

A. RAB Normal Exhaust Unit

B. CVAS Emergency Filtration Unit

C. Shield Building Ventilation Filter Unit

D. Airborne Radioactivity Removal Unit

Proposed Answer: D

Explanation (Optional):

A. INCORRECT: RAB Normal Unit has a manual deluge actuation operated locally in the RAB
B. INCORRECT: CVAS Emergency Unit has a manual deluge actuation operated locally in the RAB
C. INCORRECT: SBV Unit has a manual deluge actuation operated locally in the RAB
D. CORRECT: The ARR Unit has a manual deluge actuation operated from the Control Room.

Technical Reference(s) OP-009-004 Step 8.10
SD-FP Table 5 (Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: WLP-OPS-FP00 Obj 2 (As available)

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	E02 EA2.2	
	Importance Rating	3.0	

K/A Statement

Ability to determine and interpret the following as they apply to the (Reactor Trip Recovery): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

Proposed Question: RO 39 Rev: 0

A reactor trip has just occurred.

Which of the following support entry into OP-902-001, Reactor Trip Recovery?
(Assume components and parameters not described are as designed)

A. Two (2) CEAs fully withdrawn, with emergency boration in progress.

B. Pressurizer level is 25% and lowering with three (3) charging pumps running.

C. S/G 1 and 2 levels are 45% NR and lowering after an inadvertent MSIS.

D. Containment pressure is 16.5 psia and stable with the Containment PIG in alert.

Proposed Answer: A

Explanation (Optional):

A. **CORRECT**. With all other parameters normal as stated in the stem this condition supports entry into OP-902-001.
B. Incorrect. This indicates a condition where an RCS leak is greater than charging capacity and would not support entry into OP-902-001
C. Incorrect. The conditions given would not support entry into OP-902-001 because MSIS closes the MFIVs and therefore a loss of main feedwater has occurred.
D. Incorrect. Containment pressure and rising activity in containment would drive you to OP-902-002 or OP-902-008.

Technical Reference(s) OP-902-001, Rev 011

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE01 Obj: 12

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41(b) 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	008 AK1.01	
	Importance Rating	3.2	

K/A Statement

Knowledge of the operational implications of the following concepts as they apply to a Pressurizer Vapor Space Accident: Thermodynamics and flow characteristics of open or leaking valves

Proposed Question: RO 40 Rev: 0

The plant was at 100% power when Pressurizer Safety Valve, RC-317A, failed open. The crew tripped the reactor and initiated SIAS and CIAS.

The following conditions exist:

- RCS pressure - 1250 PSIA
- RCS T_{hot} temperature - 552 °F
- Quench Tank pressure - 41 PSIG

What is the expected temperature indication of the Pressurizer Relief Line?

A. 267 °F

B. 288 °F

C. 552 °F

D. 572 °F

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Value is equal to the given pressure if the candidate does not convert to psia from psig which is the normal units for Quench Tank Pressure indication on CP-2.

B. **CORRECT**. This value corresponds to the value that can be obtained from a Molliere diagram for 56 psia, assuming that the throttling process across the open relief valve is isenthalpic

C. Incorrect. This value is equal to the Hot Leg temperature given, which could be selected if the candidate does not realize that the throttling process is isenthalpic and assumes that an in-surge from the Hot Leg has lowered PZR Temperature to Hot Leg Temperature.

D. Incorrect. This value is equal to the saturation temperature for the RCS pressure given and could be selected if the candidate realizes the RCS is still subcooled, and therefore no significant in-surge has occurred to the PZR, but does not realize that the throttling process is isenthalpic.

Technical Reference(s) Steam Tables – Properties of Saturated and Superheated Steam

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

Learning Objective: WLP-OPS-RCS00 Obj: 2
WLP-OPS-RCS00 Obj: 5

Question Source: Bank # WF3-OPS-7177-A
Modified Bank #
New

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41(b) 14

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	009 EK3.12	
	Importance Rating	3.4	

K/A Statement

Knowledge of the interrelations between the small break LOCA and the following: Letdown Isolation

Proposed Question: RO 41 Rev: 0

The plant was operating at 100% power when a double ended shear of the Letdown line occurs at the weld on the outlet of the Regenerative Heat Exchanger. Given the following timeline:

- **T1** – PZR Pressure 1900 psia, Containment Pressure = 16.5 psia
- **T2** – PZR Pressure 1800 psia, Containment Pressure = 17.0 psia
- **T3** – PZR Pressure 1700 psia, Containment Pressure = 17.5 psia
- **T4** – PZR Pressure 1600 psia, Containment Pressure = 18.0 psia

Letdown initially isolated due to _____ Pressure. At the end of the timeline the break is isolated from the RCS by a total of _____ automatic isolation valves.

A. Containment; two

B. PZR; two

C. Containment; three

D. PZR; three

Proposed Answer: A

Explanation (Optional):

A. **CORRECT**. The timeline indicates that the setpoint is met for the High Containment Pressure (SIAS) (17.1 psia) prior to exceeding any ESFAS setpoint (PZR Press (SIAS) (1684 psia). The break is located inside containment on the outlet of the Regenerative HX upstream of the containment isolation that is outside containment. Therefore, the break is isolated from the RCS by two valves (CVC-101, CVC-103).
B. Incorrect. Wrong isolation parameter, correct number of valves. See explanation A.
C. Incorrect. Correct isolation parameter, wrong number of valves. See explanation A.
D. Incorrect. Wrong isolation parameter, wrong number of valves. See explanation A.

Technical Reference(s) OP-002-005, Chemical and Volume Control, Rev. 32
OP-902-009, Standard Appendices, Rev. 301
OP-902-000, Standard Post Trip Actions, Rev. 10
G-168, Chemical and Volume Control Sheet 1, Rev. 43

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-CVS00 Obj. 11
WLP-OPS-CVC00 Obj.3

Question Source: Bank # _____
Modified Bank # _____
New X

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41(b) 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	015 AA2.10	
	Importance Rating	3.7	

K/A Statement

Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): When to secure RCPs on loss of cooling or seal injection

Proposed Question: RO 42 Rev: 0

Given the Following:

- The Reactor is operating at 100% RTP
- At 1445 the following alarms and indications are noted in the Control Room
  - Panel M, CCW Surge Tank Level Lo-Lo A
  - Panel N, CCW Surge Tank Level Lo-Lo B
  - Panel N, CCW Surge Tank Level Hi-Lo
  - All Automatic Actions have occurred

What is the latest time that CCW flow can be restored to the Reactor Coolant Pumps without requiring they be tripped per OP-901-510, Component Cooling Water System Malfunction?

A. 1447

B. 1448

C. 1450

D. 1455

Proposed Answer: B

Explanation (Optional):

A. Incorrect. See B. This was previously a good answer at Waterford 3.
B. **CORRECT**. OP-901-510, Section E1 states that if CCW flow is lost to the AB header and cannot be restored within 3 minutes, trip the reactor and trip all RCPs. The conditions in the stem provide indication that flow is lost to the AB header.
C. Incorrect. See B.
D. Incorrect. See B. 10 minutes, as mentioned in OP-901-510 Section E1, CAUTION 1 is the time that if exceeded could cause damage to RCP Seals if CCW flow is then restored to an affected RCP.

Technical Reference(s) OP-901-510, Component Cooling Water System Malfunction Rev 300

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPO05 Obj.06

Question Source: Bank # _____
Modified Bank # _____
New X

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41(b) 10

Comments

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	022 AA1.01	
	Importance Rating	3.4	

K/A Statement

Ability to operate and/or monitor the following as they apply to the Loss of Reactor Coolant Makeup: CVCS letdown and charging

Proposed Question: RO 43 Rev: 0

Given the following:

- The plant is performing a heatup to NOP/NOT
- RCS Pressure is 1300 psia
- RCS temperature is 440ºF
- Charging Pump B is the Lead Pump
- Charging pump AB is OOS
- Charging pump A is the first backup pump and in standby

Charging Pump B trips on overcurrent. The direction provided in OP-901-112, Charging or Letdown Malfunction, would be to first:

A. Verify a suction path aligned and manually start Charging Pump A.

B. Verify a suction path aligned and verify Charging Pump A cycles on PZR level.

C. Manually close CVC-101, Letdown Stop Valve.

D. Verify auto closure of CVC-101, Letdown Stop Valve.

Proposed Answer: A

Explanation (Optional):

A. **CORRECT**. OP-901-112 has the Operator verify either the VCT or RWST suction valve open and attempt to restart a charging pump if a charging pump trips and letdown has not isolated.
B. Incorrect. OP-901-112 manually restarts a charging pump vice allowing the auto start to occur. Plausible if candidate realizes that the given conditions will not result in an auto isolation of letdown.
C. Incorrect. This action is performed if a charging pump can not be restarted and letdown is still in service.
D. Incorrect. The auto isolation will not occur since the temperature of the RCS is currently below the auto isolation setpoint of 470ºF.

Technical Reference(s) OP-901-112, Charging and Letdown Malfunction, Rev. 3

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPO10 Obj. 3
WLP-OPS-PPO10 Obj. 4

Question Source: Bank # _____
Modified Bank # WF3-OPS-4273-B
New _____

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41(b) 7 / 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	025 G2.2.38	
	Importance Rating	3.6	

K/A Statement

Knowledge of conditions and limitations in the facility license.

Proposed Question: RO 44 Rev: 0

Given the following:

- The plant is in MODE 6 with core shuffle in progress
- Cavity level is 44' MSL
- LPSI Pump A is OOS
- LPSI Pump B is OPERABLE and in operation

LPSI Pump B trips. Which of the following is prohibited per Technical Specifications?

A. Shuffling CEAs within the core using the CEA Mast.

B. Loading a new fuel assembly into the core from the Spent Fuel Pool.

C. Adding 200 gallons of Boric Acid to the RCS from an OPERABLE BAM Tank.

D. Loading a twice burned fuel assembly into the core from the Spent Fuel Pool.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Shuffling CEAs within the vessel does not increase decay heat load or constitute an operation that would lower boron concentration below that required by TS 3.9.1.
B. Incorrect. Loading new fuel assemblies does not increase decay heat load or constitute an operation that would lower boron concentration below that required by TS 3.9.1.
C. Incorrect. Adding boric acid to the RCS does not increase decay heat load or constitute an operation that would lower boron concentration below that required by TS 3.9.1.
D. **CORRECT**. Loading a previously burned fuel assembly into the core increases decay heat load and is prohibited by TS 3.9.8.1 when the LCO is not met

Technical Reference(s) TS 3/4.9.8.1, Amendment 185

B3/4.9.8, Change 19

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-REQ04 Obj. 2

Question Source: Bank # _____

Modified Bank # _____

New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41(b) 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	026 AA2.01	
	Importance Rating	2.9	

K/A Statement

Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: Location of a leak in the CCW system

Proposed Question: RO 45 Rev: 0

Given the following

- A leak is occurring in the Component Cooling Water System
- Initially CCW Surge Tank Level A and B lowered to 0%
- CCW Surge Tank level A remains at 0%
- CCW Surge Tank level B recovered to ~ 55% and stabilized

Which of the following is potentially the location of the leak?

A. Supply pipe to Waste Gas Compressor A

B. Supply pipe to Shutdown Cooling HX A

C. Return pipe from Spent Fuel Pool HX A

D. Return pipe from CEDM Fan Cooler A

Proposed Answer: B

Explanation (Optional):

A. Incorrect. The Waste Gas compressor is an NNS Loop load. This pipe would have been automatically isolated by the AB loop isolation valves and level in both sides of the CCW surge tank would have recovered.
B. **CORRECT**. Shutdown HX A is a Safety Loop A load. Safety Loops A and B would be split out such that the leak no longer affects Loop B until CCW Surge Tank Level B exceeds 55% which is the top of the baffle separating A and B sides of the CCW surge tank.
C. Incorrect. SFP HX A is an AB Loop load. This pipe would have been automatically isolated by the AB Loop isolation valves and level in both sides of the CCW surge tank would have recovered.
D. Incorrect. CEDM Fan Cooler A is an AB Loop load. This pipe would have been automatically isolated by the AB Loop isolation valves and level in both sides of the CCW surge tank would have recovered.

Technical Reference(s) OP-901-510, Component Cooling Water Malfunction, Rev. 4

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-CC00 Obj. 7

Question Source: Bank # _____
Modified Bank # _____
New X

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41(b) 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	027 AK3.03	
	Importance Rating	3.7	

K/A Statement

Knowledge of the reasons for the following responses as they apply to the Pressurizer Pressure Control Malfunctions: Actions contained in EOP for PZR PCS malfunction

Proposed Question: RO 46 Rev: 0

During a loss of all pressurizer heaters, OP-901-120, Pressurizer Pressure Control Malfunction, provides steps to _____(1)____ during power reduction to __(2)____.

A. (1) maintain PZR level constant
   (2) maintain PZR pressure constant

B. (1) raise PZR level
   (2) maintain PZR pressure constant

C. (1) maintain PZR level constant
   (2) conserve PZR inventory and enthalpy

D. (1) raise PZR level
   (2) conserve PZR inventory and enthalpy

Proposed Answer: C

Explanation (Optional):

A. Incorrect. The methodology used is correct per OP-901-120; however, even if you maintain PZR level constant RCS pressure will lower due to charging water into the PZR via the surge line that is at Hot Leg temperature, the effects of PZR Spray bypass flow into the PZR steam space, and ambient heat losses from the PZR. If the candidate does not have a good grasp on fundamentals this is an attractive answer.
B. Incorrect. Both the methodology and procedure bases are wrong. OP-901-120 has steps to take local control of PZR level controller setpoint and then set the setpoint to current PZR level before starting a down power. The basis is also wrong (see explanation in A).
C. **CORRECT**. Per OP-901-120 this is the correct approved methodology and the correct bases (See note prior to step for setting up PZR level controller).
D. Incorrect. Per OP-901-120 the methodology for PZR level control is wrong,; however, this could be a viable way to control pressure during the down power. The Bases is correct per OP-901-120.

Technical Reference(s) OP-901-120, Pressurizer Pressure Control Malfunction Rev. 301

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPO10 Obj. 4

Question Source: Bank # _____
Modified Bank # WF3-OPS-5995-A
New _____

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41(b) 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	029 G2.4.11	
	Importance Rating	4.0	

K/A Statement

Knowledge of abnormal condition procedures.

Proposed Question: RO 47 Rev: 0

The basis for re-closing the A32 and B32 feeder breakers 5 seconds after opening during an ATWS is to restore power to the:

A. CEDMCS in order to verify that all CEAs have fully inserted.

B. A32 and B32 busses before UV relays strip loads from the busses.

C. CEDM cooling fans to protect CEDM coils from overheating.

D. pressurizer heaters to maintain RCS pressure control.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Power for rod position indication is independent of power from A32 and B32 to CEDM MG Sets.
B. Incorrect. The associated UV relays would have already operated by this point and all loads on the Bus that strip on UV would have already have been stripped.
C. Incorrect. CEDM cooling fans are powered from A31 and B31.
D. **CORRECT**. Re-closing the A32 and B32 feeder breakers allows for maintaining RCS pressure control via the pressurizer heaters. This is in accordance with the basis for Step 1 of OP-902-000, Standard Post Trip Actions as listed in TGOP-902-000, Technical Guide for Standard Post Trip Actions.

Technical Reference(s) TGOP-902-000, Technical Guide for Standard Post Trip Actions, Rev. 9

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE01 Obj 11 (As available)

Question Source: Bank # WF3-OPS-1704-A

Modified Bank #

New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b) 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	038 EK1.02	
	Importance Rating	3.2	

K/A Statement
Knowledge of the operational implications of the following concepts as they apply to the SGTR: Leak rate vs. pressure drop

Proposed Question: RO 48 Rev: 0

With a Steam Generator Tube Rupture in progress it is ultimately desired to depressurize the RCS to within ___(1)___ psid of the ruptured Steam Generator to ___(2)___ and minimize the potential for Steam Generator overfill.

	(1)	(2)
A.	50	minimize the potential release to the environment
B.	50	prevent loss of subcooled margin
C.	100	minimize the potential release to the environment
D.	100	prevent loss of subcooled margin

Proposed Answer: A

Explanation (Optional):

A. **CORRECT**. Per step 12 of OP-902-007 and TG-OP-902-007 this is the correct pressure differential. Per TG-OP-902-007 this minimizes pri-to-sec leakage which minimizes release magnitude and helps to prevent S/G overfill.
B. Incorrect. Right D/P, wrong bases. The substep within step 12 that provides instructions for maintaining RCS pressure within the P/T curve limits prevents loss of subcooled margin and allows continued operation of RCPs.
C. Incorrect. Wrong D/P; right bases.
D. Incorrect. Wrong D/P, wrong bases.

Technical Reference(s) OPO-902-007, Steam Generator Tube Rupture Recovery, Rev. 012

TG-OP-902-007, Rev. 302

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE07 Obj 3 (As available)

Question Source: Bank # _____

Modified Bank # _____

New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41(b) 14

Comments:

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

K/A Statement

Knowledge of the interrelations between the (Excess Steam Demand) and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

Proposed Question: RO 49 Rev: 0

An Excess Steam Demand is in progress and OP-902-004, Excess Steam Demand Recovery is being implemented.

- RCS Subcool Margin 80ºF and stable
- SG 1 pressure is 540 psia and lowering
- SG 2 pressure is 670 psia and stable
- SG 1 level 45% WR and lowering
- SG 2 level 60% WR and lowering slowly
- Containment Pressure 25 psia and rising
- Containment Spray Pump A is tagged out with CS-125A gagged shut
- Containment Spray Header B flow is 1000 gpm
- CVC-103, Letdown Inside Containment Isolation Valve stuck open
- EFW-228A, Emergency Feedwater Isolation SG 1 Primary is tagged closed
- All other systems and components are operating as designed

Which Safety Function is not being met as a result of these failures?

A. RCS Pressure Control

B. RCS Heat Removal

C. Containment Isolation

D. Containment Temperature and Pressure Control

Proposed Answer: D

Explanation (Optional):

A. Incorrect. RCS pressure Control can be met by LPSI and HPSI pumps providing flow > OP-902-009 Appendix 2 curves. This is implied by the last bullet in the conditions given.
B. RCS Heat Removal. Although S/G 2 level is lowering the final bullet implies that that EFAS-2 actuated per design and EFW is operating per design. The level given would not be low enough yet for EFW to feed S/G 2. Operators are trained to evaluate this safety function as SAT if EFW is automatic and working as designed. Tc can also be implied to be lowering during the blowdown phase of an excess steam demand. The conditions given imply that information.
C. Incorrect. Both CVC containment isolation valves would have to fail to jeopardize the Containment Isolation safety function. The last bullet implies that this is not the case.
D. **CORRECT**. At least one Containment Spray pump must be providing > 1750 gpm of flow to meet the CT&PC safety function.

Technical Reference(s) OP-902-004, Excess Steam Demand Recovery, Rev. 11

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE04 Obj. 6 (As available)

Question Source: Bank # _____
Modified Bank # _____
New X

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41(b) 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	E06 G2.1.7	
	Importance Rating	4.4	

K/A Statement

Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

Proposed Question: RO 50 Rev: 0

The plant has tripped due to a loss of both Main Feedwater Pumps. The crew has diagnosed to OP-902-006 and taken all required actions. The plant will be cooled down to SDC entry conditions. The plant conditions are as follows:

- RCPs 1B and 2B are running
- CSP Level = 50%
- DWST Level = 70%
- RCS Thot = 550 degrees F
- Plant Shutdown = 4 hours

The approximate time remaining to place SDC in service is:

A. 29 hours

B. 22 hours

C. 19 hours

D. 14 hours

Proposed Answer: C

Explanation (Optional):

A. Incorrect. This is the value that would be obtained if the no RCPs operating curve were used (OP-902-009 Att. 2-J).
B. Incorrect. This is the value that would be obtained if the correct attachment were used but the candidate did not take into account the water needed to remove the sensible heat between Thot of 550ºF and SDC entry conditions of 350ºF.
C. **CORRECT**. This is the value that should be obtained if the candidate calculates available feedwater and selects and correctly reads the 4 hour curve on Att. 2-I of OP-902-009.
D. Incorrect. This is the value obtained using the 4 RCP operating curve of OP-902-009 (Att. 2-H).

Technical Reference(s) OP-902-009, Standard Appendices, Rev. 301

Proposed references to be provided to applicants during examination: OP-902-009 Att. 2-G thru 2-J

Learning Objective: WLP-OPS-PPE06 Obj. 4
WLP-OPS-PPE06 Obj. 9

Question Source: Bank # _____
Modified Bank # WF3-OPS-6908-A
New _____

Question History: Last NRC Exam _____

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

10 CFR Part 55 Content: 55.41(b) 5 / 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	055 EA1.06	
	Importance Rating	4.1	

K/A Statement

Ability to operate and monitor the following as they apply to a Station Blackout:
Restoration of power with one ED/G

Proposed Question: RO 51 Rev: 0

Given the following:

- The plant is implementing OP-902-005, Station Blackout Recovery.
- Attachments 7-A(B)(C), Switchgear Room A(B)(AB) Removable Loads have been completed as required by OP-902-005.
- Emergency Diesel Generator A has become available to restore power.

OP-902-005, previously directed the Operators to place both Containment Spray Pump control switches to OFF if Containment pressure is < 17.7 psia to prevent _____.

A. an inadvertent spray of containment

B. actuating a Sequencer UV Lockout

C. operating CS Pumps at shutoff head

D. overloading an Emergency Diesel Generator

Proposed Answer: A

Explanation (Optional):

A. **CORRECT**. Due to de-energizing ESFAS relays as part of the Attachment given in the stem a CSAS signal is generated. Restoration of EDG A will result in a start of CS Pump A and initiation of flow through the Train A Spray Header nozzles.
B. Incorrect. Although the restoration of EDG A will start various ESFAS A loads, the Sequencer should still function to start the loads in the prescribed order without initiating a sequencer lockout.
C. Incorrect. Although CS Pump A will start on the restoration of EDG A, the CS A header isolation valve should be open due to the load stripping that occurs in the Attachments given in the stem as complete.
D. Incorrect. The operation of the sequencer on the start of EDG A should prevent an overload condition on EDG A.

Technical Reference(s) OP-902-005, Station Blackout Recovery, Rev. 13

OP-902-009, Standard Appendices, Rev. 301, Attachments 7-A, 7-B, and 7-C

TG-OP-902-005, Technical Guide for Station Blackout, Rev. 302

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE05 Obj. 7

Question Source: Bank # WF3-OPS-2469-A

Modified Bank # _____

New _____

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41(b) 10

_____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	056 AA1.37	
	Importance Rating	3.4	

K/A Statement

Ability to operate and / or monitor the following as they apply to the Loss of Offsite Power:
Instrument air

Proposed Question: RO 52 Rev: 0

Given the following initial conditions:

- EDG A is OOS
- IA compressor A is running; IA compressor B is in standby
- Combined Instrument Air and Station Air usage is within the capacity of one air compressor
- The plant is operating at 100% with all systems aligned normally, except as noted, when a reactor trip and Loss of Offsite Power occurs
- All Automatic Actions occur as designed

Which of the following is correct?

A. IA pressure drops until IA compressor B starts automatically, then IA pressure recovers.

B. IA pressure drops until action is taken to manually cross-connect Instrument Air and Station Air, then IA pressure recovers.

C. IA pressure continues to drop until alternate cooling is aligned to IA compressor B and the compressor is started manually, then IA pressure recovers.

D. IA pressure drops until Station Air and Instrument air are automatically cross-connected, then IA pressure varies around the setpoint of the automatic valve.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. IA Compressor B will not automatically start (see explanation in B).
B. Incorrect. The conditions given in the stem indicate a Loss of Offsite Power has occurred. Train A is lost due to the 87STA actuation which isolates SUT A. Train B is lost due to the Generator Output Breaker B failing to open which then opens the feeder breakers from the grid feeding Offsite power to Train B. This results in loss of all Station Air Compressors and Instrument Air Compressor A. IA compressor B is interlocked such that it will not auto start when the normal power supply is unavailable to Bus B3. Bus B3 which is currently energized by EDG B. Even if the IA and SA systems are manually cross-connected pressure would not recover because no SA air compressors are available.
C. **CORRECT**. IA compressor B can be manually started. This would require alternate cooling be aligned due to the loss of normal cooling (TCW, CW). OP-902-003 contains steps to align potable water to the air compressors and start all available air compressors. In this case, the only available air compressor is IA compressor B (see explanation in A).
D. Incorrect. See explanation A. Even if the IA to SA cross-connect automatically opens without air compressors running pressure is going to continue to lower.

Technical Reference(s): OP-902-003, Loss of Offsite Power/Loss of Forced Circulation, Rev.006
OP-902-009, Standard Appendices, Rev 301

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-AIR00 Obj. 1

Question Source: Bank # _____
Modified Bank # WF3-OPS-6737-A
New _____

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41(b) 4

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	057 AA2.19	
	Importance Rating	4.0	

K/A Statement

Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: The plant automatic actions that will occur on the loss of a vital ac electrical instrument bus.

Proposed Question: RO 53 Rev: 0

Given the following

- The plant is at 100%

On the loss of SUPS MA, Reactor Trip breakers _____ open and the reactor _____.

A. 1, 2, 5, and 6; remains at 100%

B. 1, 2 ,5, and 6; trips

C. 3, 4, 7, and 8; remains at 100%

D. 3, 4, 7, and 8; trips

Proposed Answer: A

Explanation (Optional):

A. **CORRECT**. This is the correct combination of trip breakers that will open on a loss of SUPS MA. These breakers opening will only eliminate one of two paths for power to the CEAs and the reactor does not trip.
B. Incorrect. Correct breakers, wrong effect. See explanation for A.
C. Incorrect. Wrong breaker combination. See explanation A.
D. Incorrect. Wrong breaker combination, wrong effect. See explanation A

Technical Reference(s) OP-901-312, Rev. 2
SD-PPS, Fig. 1, Rev. 0

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPO30 Obj. 1

Question Source: Bank # _____
Modified Bank # WF3-OPS-3316-A
New _____

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41(b) 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	058 AK3.02	
	Importance Rating	4.0	

K/A Statement

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

Proposed Question: RO 54 Rev: 0

On a Loss of TGB-DC Bus, the reactor will be manually tripped when Instrument Air Header Pressure lowers to _____ due to loss of power to _____.

A. 80 psig; IA and SA compressor unloader valves

B. 80 psig; Instrument Air Dryers

C. 65 psig; IA and SA compressor unloader valves

D. 65 psig; Instrument Air Dryers

Proposed Answer: C

Explanation (Optional):

A. Incorrect. The reactor will not be tripped until 65 psig per OP-901-313. the basis is correct per a note in OP-901-313.
B. Incorrect. Wrong pressure. Wrong bases, however, IA dryer malfunctions can significantly affect IA pressure making it plausible.
C. **CORRECT**. Per the step and associated note in OP-901-313 this is the correct value and bases.
D. Incorrect. Correct pressure. Wrong Bases.

Technical Reference(s) OP-901-313, Loss of a 125 DC Bus, Rev. 301

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPO03 Obj. 4

Question Source: Bank # _____
Modified Bank # _____
New X

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41(b) 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	062 G2.4.21	
	Importance Rating	4.0	

K/A Statement

Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.

Proposed Question: RO 55 Rev: 0

Given the following

- A loss of main feedwater event has occurred
- No Wet Cooling Tower Basin is available for alignment to EFW due to loss of both ACCW Pumps.

Which of the following is the level in the Condensate Storage Pool at which the RCS and Core Heat Removal safety function is impacted by the potential loss of the EFW pumps due to cavitation, per TG-OP-902-009, Technical Guide for Standard Appendices?

A. 5%

B. 11%

C. 13%

D. 25%

Proposed Answer: B

Explanation (Optional):

A. Incorrect. This is an old value from a previous revision that does not take level uncertainties into account.
B. **CORRECT**. This is the value stated in TGOP-902-009.
C. Incorrect. This is the value that Operators should wait for to complete the lineup to the Wet Tower Basin to ensure that unnecessary depletion of the Wet Tower Basin does not occur.
D. Incorrect. This is the level at which the CRS should notify personnel outside of the Control Room to start the lineup to the Wet Tower Basin. This allows 20 minutes to complete the lineup.

Technical Reference(s): TGOP-902-009, Technical Guide for Standard Appendices, Rev. 301

OP-902-009, Standard Appendices, Rev. 301

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE06 Obj. 9

Question Source: Bank # _____

Modified Bank # _____

New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41(b) 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	065 AK3.04	
	Importance Rating	3.0	

K/A Statement

Ability to determine and interpret the following as they apply to the Loss of Instrument Air:
Cross-over to backup air supplies

Proposed Question: RO 56 Rev: 0

The following alarm is received in the Control Room on Annunciator Panel L:

- Valve Operators Nitrogen Backup Actuated/Trouble

Which of the following is a potential cause of this alarm?

A. A Nitrogen Accumulator Outlet Valve is open.

B. An Essential Air Nitrogen Station has been aligned.

C. An Essential Air Nitrogen bottle pressure is less than 715 psig.

D. A Nitrogen Tube Trailer has been aligned.

Proposed Answer: A

Explanation (Optional):

A. **CORRECT**. Per Annunciator Response Procedure OP-500-010, any one of eight Nitrogen Accumulator Outlet Valve limit switches indicating that the valve is not fully closed actuates the alarm.
B. Incorrect. None of the 3 inputs to the alarm is associated with the Essential Air Accumulators. However, the Essential Air system is a valid backup source of motive air that is aligned on a loss of IA and valve position is a parameter used as an input to the alarm.
C. Incorrect. None of the 3 inputs to the alarm is associated with the Essential Air Accumulators. However, the Essential Air system is a valid backup source of motive air that is aligned on a loss of IA. 715 psig is the setpoint for the nitrogen accumulator pressure that is an input to the alarm
D. Incorrect. The tube trailer is a backup source of nitrogen but is not an input to the alarm. However, valve position is used as an input to the alarm, but the source is from the nitrogen accumulator outlet valves.

Technical Reference(s) OP-500-010, Control Room Cabinet L, Rev 020

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-AIR00 Obj. 4

Question Source: Bank # _____
Modified Bank # _____
New X

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41(b) 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	001 AK2.08	
	Importance Rating	3.1	

K/A Statement

Knowledge of the interrelations between the Continuous Rod Withdrawal and the following: Individual rod display lights and indications

Proposed Question: RO 57 Rev: 0

Given the following:

- Reactor Power is 12% with power ascension in progress
- Reg Group 6 is being withdrawn manually in MS mode to raise power to 15%
- The ATC releases the IN-HOLD-OUT switch at 135” and Reg Group 6 continues out due to a switch contact failure

Assuming a 0.0” deviation between individual CEAs in Reg Group 6 and 0.0” deviation between the Pulse Counters and RSPTs for Reg Group 6 answer the following:

(1) What will stop CEA motion for Reg Group 6?
(2) What will be the approximate reading of the Pulse Counters on CP-2?
(3) What will be the status of the Red indicators for Reg Group 6 CEAs on the CEDMCS Control Panel on CP-2?

A. (1) Upper Electrical Limit
(2) 150”
(3) Illuminated

B. (1) Upper Electrical Limit
(2) 145.5”
(3) Extinguished

C. (1) Upper Group Stop
(2) 150”
(3) Illuminated

D. (1) Upper Group Stop
(2) 145.5”
(3) Extinguished

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Upper Group Stop would occur, not the Upper Electrical Limit. This would occur at 145.5“ on the pulse counters, not 150”. The red lights would be extinguished because they are actuated by the Upper Electrical Limit for the individual rod, not the Upper Group Stop.
B. Incorrect. Upper Group Stop would occur, not the Upper Electrical Limit. This would occur at 145.5” on the pulse counters as stated. The red lights would be extinguished as stated because they are actuated by the Upper Electrical Limit for the individual rod, not the Upper Group Stop.
C. Incorrect. Upper Group Stop would occur as stated; however, this would occur at 145.5“ on the pulse counters, not 150”. The red lights would be extinguished because they are actuated by the Upper Electrical Limit for the individual rod, not the Upper Group Stop.
D. **CORRECT**. Upper Group Stop would occur as stated. This would occur at 145.5“ on the pulse counters as stated. The red lights would be extinguished as stated because they are actuated by the Upper Electrical Limit for the individual rod, not the Upper Group Stop.

Technical Reference(s) OP-004-004, Rev. 16

Proposed references to be provided to applicants during examination: _____

Learning Objective: WLP-OPS-CED00 Obj. 1

Question Source: Bank # _____
Modified Bank # _____
New X

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41(b) 6

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	036 AA1.02	
	Importance Rating	3.1	

K/A Statement

Ability to operate and / or monitor the following as they apply to the Fuel Handling Incidents: ARM system

Proposed Question: RO 58 Rev: 0

Given the following:

- The plant is in MODE 6 with fuel shuffle in progress in both the Fuel Handling Building and Containment.
- Containment Purge is in the Refueling Mode.
- Containment equipment hatch and airlock doors are closed and no containment penetrations are impaired.
- A fuel bundle is dropped from the Refueling Machine Fuel Hoist.

Which of the following radiation monitors can detect the event **AND** terminate the radioactive gas release?

A. Containment Atmosphere Hi Range Area Radiation Monitor, ARM-IRE-5400AS.

B. Containment Purge Area Radiation Monitor, ARM-IRE-5024.

C. Refueling Machine Area Radiation Monitor, ARM-IRE-5013.

D. Containment PIG Process RM, PRM-IRE-0100S.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. No automatic functions are associated with this Rad Monitor. However, the RM detectors are in containment and could see the radiation from the fuel handling incident.
B. **CORRECT**. ARM 5024 provides isolation of Containment Purge on Hi Rad and is in the general area of fuel handling on the +46' elevation of Containment.
C. Incorrect. This rad monitor is in the general area of the Refueling Cavity; however, it has indication and alarm functions only.
D. Incorrect. The containment PIG monitors containment atmosphere but does not have Containment Purge isolation functions.

Technical Reference(s) OP-901-405, Fuel Handling Incident, Rev. 2

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-RMS00 Obj. 2

Question Source: Bank # WF3-OPS-07922

Modified Bank #

New

Question History: Last NRC Exam 2006 W3 NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b) 11

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	037 AK1.01	
	Importance Rating	2.9	

K/A Statement

Knowledge of the operational implications of the following concepts as they apply to Steam Generator Tube Leak: Use of steam tables

Proposed Question: RO 59 Rev: 0

Given the following:

- A cooldown is being performed to MODE 5 per OP-901-202, Steam Generator Tube Leakage or High Activity due to a leak on SG 2.
- All RCPs are secured
- CET Temp is 382ºF
- T_h Loop1 is 375ºF
- T_h Loop1 is 378ºF
- T_c Loop 1 is 350ºF
- T_c Loop 2 is 382ºF
- The CRS has placed a lower limit of 30ºF on Subcool Margin

Determine the minimum value of RCS pressure that supports the requested RCS Subcool Margin.

A. 196 psia

B. 262 psia

C. 270 psia

D. 283 psia

Proposed Answer: D

Explanation (Optional):

A. Incorrect. This is the value associated with Cold leg 1 temperature. The hottest RCS temperature should be used to determine subcool margin. This would also lower RCS pressure below the isolated S/G pressure.
B. Incorrect. This is the value associated with Hot Leg Loop 1 temperature. The hottest RCS temperature should be used to determine subcool margin.
C. Incorrect. This is the value associated with Hot Leg Loop 2 temperature. The hottest RCS temperature should be used to determine subcool margin.
D. **CORRECT**. This is the value associated with CET and Cold Leg 2 temperatures. The hottest RCS temperature should be used to determine subcool margin. OI-038-000 has guidance to use CET temperatures on natural circulation

Technical Reference(s) Steam Tables – Properties of Saturated and Superheated Steam

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

Learning Objective: WLP-OPS-PPO20 Obj. 3

Question Source: Bank # _____
Modified Bank # _____
New X

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41(b) 10 / 14

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	051 AA2.02	
	Importance Rating	3.9	

K/A Statement

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

Proposed Question: RO 60 Rev: 0

In accordance with OP-901-220, Loss of Condenser Vacuum, a reactor trip is first required when condenser vacuum is approaching _____ inches Hg and not stabilized.

A. 25

B. 23

C. 20

D. 14

Proposed Answer: C

Explanation (Optional):

A. Incorrect. This is the point at which a power reduction is required.
B. Incorrect. This is the point at which the Condenser Vacuum Pumps go into the Hogging mode of operation.
C. **CORRECT**. This is the requirement stated in OP-901-220.
D. Incorrect. This is the point at which the main feedwater pumps would trip and OP-901-220 requires action to be performed to protect the Main Condenser from high pressure.

Technical Reference(s) OP-901-220, Loss of Condenser Vacuum, Rev. 2

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPO02 Obj. 3 (As available)

Question Source: Bank # WF3-OPS-5789-A

Modified Bank #

New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b) 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	CE/E09 EK2.1	
	Importance Rating	3.6	

K/A Statement

Knowledge of the interrelations between the Functional Recovery and the following:
Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Proposed Question: RO 61 Rev: 0

Given the following:

- A large break LOCA concurrent with a failed open secondary safety on S/G 2 has occurred
- RCS Pressure is 100 psia and stable
- RWSP Level is 60% and lowering
- HPSI Pump A/B is OOS for a pump coupling replacement
- HPSI Pump A tripped on overcurrent upon its initial start signal

15 minutes later, MCC 311B feeder breaker trips on overcurrent.

With no Operator action taken to mitigate this condition, which of the following would interrupt Safety Injection flow **FIRST**?

A. Injecting from the RWSP prior to RAS.

B. Injecting from the SIS Sump after RAS.

C. Isolating the SI Pump recircs after RAS.

D. Isolating the RWSP from the SI Pump suction after RAS.

Proposed Answer: B.

Explanation (Optional):

A. Incorrect. Per conditions given, (RWSP level has dropped at least 23% assuming minimum required volume was available, RCS pressure well below SIAS setpoint) the SI system should be in its SIAS alignment. The flow path should be unaffected by the loss of 311B which would cause loss of power to LPSI and HPSI Loop FCV MOVs and SI-120B which is also an MOV. This would fail them in their SIAS position.

B. **CORRECT**. The loss of MCC 311B causes loss of power to SI-602B which is the MOV suction isolation to the Train B ECCS pumps from the SIS Sump. This valve would normally automatically open at 10% level in the RWSP to provide a suction source to the pumps from the alternate source prior to emptying the RWSP. Since only HPSI Pump B is available to inject this would result in SI flow being interrupted after RAS when the RWSP empties.

C. Incorrect. Only one of two pump recirc isolation valves in series has lost power due to the malfunction given (SI-120B). SI-121B is powered from MCC 312B and would still be available to isolate the recirc flow path.

D. Incorrect. While SI-106B (Train B RWSP isolation) can not be isolated until SI-602B is open, this would not be the reason flow would be lost. Flow will be lost as a result of not aligning the alternate source of suction (SIS Sump) prior to primary source (RWSP) depletion. The act of aligning the alternate source would in itself allow isolation of the primary source after RAS.

Technical Reference(s) OP-009-008, Safety Injection, Rev. 29

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-SI00 Obj. 3
WLP-OPS-SI00 Obj. 8

Question Source: Bank # _____
Modified Bank # _____
New X

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41(b) 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	074 EK3.04	
	Importance Rating	3.9	

K/A Statement

Knowledge of the reasons for the following responses as they apply to the Inadequate Core Cooling: Tripping RCPs

Proposed Question: RO 62 Rev: 0

Under what circumstances does OP-902-008, Safety Function Recovery:

(1) require securing all RCPs for a potential inadequate core cooling condition?

AND

(2) What is the basis for this action.

A. (1) Containment Spray Actuation Signal (CSAS)
   (2) Prevent RCP Damage

B. (1) Loss of NPSH requirements for RCPs
   (2) Prevent RCP Damage

C. (1) Loss of Main Feedwater > 30 minutes, only EFW AB operable
   (2) Eliminate RCP heat input to RCS

D. (1) Loss of all Feedwater
   (2) Eliminate RCP heat input to RCS

Proposed Answer: D

Explanation (Optional):

A. Incorrect. CSAS by itself does not indicate a condition that could lead to inadequate core cooling; however, it is a valid reason to secure the RCP and is addressed in Section 4.0 of OP-902-008 prior to entering any Safety Function Success Path for the reason stated.
B. Incorrect. NPSH requirements for the RCP by itself does not indicate an inadequate core cooling condition; however, it is a valid reason to secure the RCP and is addressed in Section 4.0 of OP-902-008 prior to entering any Safety Function Success Path for the reason stated.
C. Incorrect. Inadequate feedwater can lead to an inadequate core cooling condition. However, all RCPs would only be secured in this circumstance if only one motor driven EFW Pump is available. EFW Pump AB has 100% capacity
D. **CORRECT**. Inadequate feedwater can lead to an inadequate core cooling condition. The RCPs are required to be secured for this reason. With no feedwater the RCPs are secured to eliminate the heat input into the RCS from the RCPs that must be removed by steaming the S/Gs.

Technical Reference(s)	OP-902-008, Safety Function Recovery, Rev. 15	(Attach if not previously provided)
	TG-OP-902-008, Technical Guide for Safety Function Recovery, Rev.302	(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE08 Obj.9
WLP-OPS-PPE08 Obj. 4

Question Source:
Bank # _____
Modified Bank # _____
New X

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41(b) 10
_____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	069 AA1.01	
	Importance Rating	3.5	

K/A Statement

Ability to operate and/or monitor the following as they apply to the Loss of Containment Integrity: Isolation valves, dampers, and electro-pneumatic devices

Proposed Question: RO 63 Rev: 0

Given the following:

- A Loss of Coolant Accident has occurred
- Containment pressure is 17.2 psia and slowly rising
- All systems operate as designed and equipment lineups were normal at the start of the event

Which of the following pipe breaks would constitute a loss of containment integrity at the current conditions?

A. on the downstream weld of CVC-101, Letdown Stop Valve.

B. just upstream of CS-128B, Containment Spray Riser B Check Valve.

C. on the supply line of Containment Fan Cooler C at the cooling coils.

D. just downstream of CVC-216B, Pressurizer Auxiliary Spray Valve B

Proposed Answer: C

A. Incorrect. A leak at this location would be upstream of the Containment Isolation valves CVC-103 and CVC-109. Both of these valves closed on a CIAS at 17.1 psia in the containment.
B. Incorrect. CS-125B, Containment Spray Header B Isolation Valve would still be closed until pressure reaches 17.7 psia.
C. **CORRECT**. On an SIAS all flow control valves for CC flow through CFC C would be open due to the containment pressure providing a path outside containment. A Keyswitch allows override of the valves to isolate the leak.
D. Incorrect. Although CVC-216B does not get any automatic isolation signals it is normally closed and conditions indicate that the valve is in its normal position.

Technical Reference(s) G-160 Sht 4, Rev. 16
SD-CC00, Rev. 12

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-CC00 Obj. 7

Question Source: Bank # _____
Modified Bank # _____
New X

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41(b) 9

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	076 G2.1.25	
	Importance Rating	3.9	

K/A Statement

Ability to interpret reference materials, such as graphs, curves, tables, etc.

Proposed Question: RO 64 Rev: 0

Given the following:

- Reactor power is 100%
- OP-901-410, High Activity in Reactor Coolant System, was entered today at 0230 due to the Isotopic Analysis for DEQ I-131 sample reading 3.0 µci/gm.
- The last sample taken for Isotopic Analysis for Iodine activity was taken today at 0400.

A reactor trip occurs at 0430. Using TS Table 4.4-4, determine the **LATEST** time possible to perform the next sample for Iodine isotopic analysis.

A. 0630 today

B. 0800 today

C. 1030 today

D. 0400 in 14 days

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Even though the reactor tripped, the sample is not required to be taken due to the power change until 1030. A sample at two hours is not required.
B. **CORRECT**. The table requires a 4 hour sample be taken as long as Iodine activity is exceeding 1.0 µci/gm.
C. Incorrect. If the plant was not already in a 4 hr sampling requirement this would be the required time to take the sample post trip due to the power change. Waiting until this point would however, violate the 4 hr requirement.
D. Incorrect. The increased Iodine sampling requirements must continue on a 4 hour basis until Iodine activity is less than 1.0 µci/gm.

Technical Reference(s)	TS Table 4.4-4, Amendment 184
	OP-901-410, High Activity in Reactor Coolant System, Rev. 3

Proposed references to be provided to applicants during examination: TS 3.4.7 and Table 4.4-4

Learning Objective: WLP-OPS-PPO40 Obj. 3
WLP-OPS-PPO40 Obj. 5

Question Source:

Bank #	
Modified Bank #	
New	X

Question History: Last NRC Exam N/A

Question Cognitive Level:

Memory or Fundamental Knowledge	
Comprehension or Analysis	X

10 CFR Part 55 Content: 55.41(b) 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	CE/A16 AA2.1	
	Importance Rating	2.7	

K/A Statement

Ability to determine and interpret the following as they apply to the (Excess RCS Leakage) Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Proposed Question: RO 65 Rev: 0

Given the following conditions:

- Letdown is secured to determine the location of an RCS leak
- RCS Tavg is 575 degrees and steady
- Current RCS leakage is 86 gpm
- AB Charging pump is out of service
- Charging pumps A and B are running

Which of the following is the appropriate action to be performed?

A. Remain in OP-901-111, Reactor Coolant System Leak, and attempt to locate the leak.

B. Commence a normal shutdown in accordance with OP-010-005, Plant Shutdown.

C. Commence a rapid plant shutdown in accordance with OP-901-212, Rapid Plant Power Reduction.

D. Initiate a manual reactor trip, SIAS/CIAS, and go to OP-902-000, Standard Post Trip Actions.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. The RCS leakage and the 4-6 gpm RCP controlled bleedoff that normally exists exceeds the capacity of two charging pumps. This is an option if pressurizer level can be maintained.
B. Incorrect. The RCS leakage and the 4-6 gpm RCP controlled bleedoff that normally exists exceeds the capacity of two charging pumps. This is an option if pressurizer level can be maintained and a shutdown is required.
C. Incorrect. The RCS leakage with the 4-6 gpm RCP controlled bleedoff that normally exists exceeds the capacity of two charging pumps. This is a valid shutdown option; however, OP-901-111 specifically ignores this option based on the possible inability to control pressurizer level on the shutdown.
D. **CORRECT**. Based on the total inventory loss from the RCS (RCS leakage + RCP controlled bleedoff) exceeding Charging pump capacity, Pressurizer level can not be maintained. OP-901-111 requires that answer D be performed per step 2 when this is the case.

Technical Reference(s) OP-901-111, Reactor Coolant System Leak, Rev. 301

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPO10 Obj. 3

Question Source: Bank # WF3-OPS-5199-A
Modified Bank # _____
New _____

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41(b) 8 / 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	G 2.1.40	
	Importance Rating	2.8	

K/A Statement
Conduct of Operations: Knowledge of refueling administrative requirements.

Proposed Question: RO 66 Rev: 0

The reactor was operating at 100% power at EOC. The following sequence of events occurs:

- At 1330 on 10/4 the reactor trips
- At 1245 on 10/9 the cause of the trip has been corrected and the crew has commenced withdrawing the Shutdown Banks.
- At 1315 on 10/9 with Shutdown Bank B at 50", the reactor trip breakers open due to low S/G levels

A decision is made to enter the Refueling outage early. At what minimum point in time does the plant meet the time requirements to remove irradiated fuel from the reactor per TS 3.9.3, Decay Time?

A. 1330 on 10/7

B. 1730 on 10/8

C. 1315 on 10/12

D. 1715 on 10/13

Proposed Answer: A

Explanation (Optional):

A. **CORRECT**. The conditions given for the startup indicate that the reactor did not obtain criticality prior to the trip. Therefore, the 72 hours starts from the first reactor trip time. The time limit is based on allowing the decay of short lived fission products.
B. Incorrect. The base time used is correct but a minimum subcriticality time of 100 hours is used. This time is common in Tech Specs in the industry.
C. Incorrect. This is 72 hours from the second trip. Based on the conditions given in the stem the reactor never achieved criticality prior to second trip.
D. Incorrect. This is 100 hrs from the second trip. This is the wrong duration after shutdown and also would not apply since we did not reach criticality prior to the second trip.

Technical Reference(s) TS 3/4.9.3, Decay Time, Amendment 0
B3/4.9.3, Decay Time, Change 21

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-FHS00 Obj. 5

Question Source: Bank # _____
Modified Bank # _____
New X

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41(b) 1 / 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	G 2.1.19	
	Importance Rating	3.9	

K/A Statement
Conduct of Operations: Ability to use plant computers to evaluate system or component status.

Proposed Question: RO 67 Rev: 0

Which of the following control board annunciator would be if the Plant Monitoring Computer fails?

A. CEA DISABLED

B. CEDMCS TIMER FAILURE

C. CEDMCS MAINTENANCE ERROR

D. POWER DEPENDENT INSERTION LIMIT

Proposed Answer: D

Explanation (Optional):

A. Incorrect. This alarm is generated by contacts for CEA Circuit Breakers in OFF.
B. Incorrect. This alarm is generated by ACTM Micro-processer Card.
C. Incorrect. This alarm is generated by CEDMCS when trying to place more than one subgroup on hold bus.
D. **CORRECT**. This alarm is generated by the PMC.

Technical Reference(s) SD-PMC, Rev. 7
OP-500-00, Control Room Cabinet H, Rev. 26

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PMC00 Obj. 3

Question Source: Bank # _____
Modified Bank # _____
New X

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41(b) 4

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	G 2.1.37	
	Importance Rating	4.3	

K/A Statement

Equipment Control: Knowledge of procedures, guidelines, or limitations associated with reactivity management.

Proposed Question: RO 68 Rev: 0

Which of the following accurately describes the procedure requirement regarding reactivity management?

A. Planned Primary Makeup additions less than 100 gallons are exempt from peer checks in Modes 3, 4, and 5.

B. Positive reactivity additions may be made by two methods simultaneously if in accordance with an approved reactivity plan.

C. An approved reactivity plan is only needed for planned reactor power changes of greater than 5%.

D. During approach to criticality the designated Reactor Engineer gives permission to continue rod withdrawal after verifying 1/M plots do not predict criticality.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. EN-OP-115, Conduct of Operations requires that all planned reactivity manipulations are verified/peer checked.
B. **CORRECT**. Per EN-OP-103 and EN-OP-115, Conduct of Operations it is permissible to add positive reactivity to the reactor by more than one means with an approved reactivity plan.
C. Incorrect. EN-OP-115 guidance is to develop a written reactivity plan for power changes of greater than 2%.
D. Incorrect. EN-OP-115 specifically requires direction that affects reactivity to come through the individual holding Command and Control to the Control Room Operators.

Technical Reference(s) EN-OP-103, Reactivity Management Program, Rev. 4

EN-OP-115, Conduct of Operations, Rev. 9

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-EXP00 Obj. 15
WLP-OPS-PPA00 Obj. 3

Question Source: Bank # _____
Modified Bank # _____
New X

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41(b) 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	G 2.2.38	
	Importance Rating	3.6	

K/A Statement

Equipment Control: Knowledge of conditions and limitations in the facility license.

Proposed Question: RO 69 Rev: 0

Per the Waterford 3 Operating License, Entergy Operations Inc. is authorized to operate the Waterford 3 reactor core at power levels not to exceed:

A. 3441 MW_t

B. 3461 MW_t

C. 3716 MW_t

D. 3739 MW_t

Proposed Answer: C

Explanation (Optional):

A. Incorrect. This value is equal to the intermediate power uprate value.
B. Incorrect. This value is equal to the intermediate power uprate value + the old estimated RCP heat contribution.
C. **CORRECT**. This is the value listed in the current Operating License post power uprate.
D. Incorrect. This is the value listed in the current Operating License post power uprate + the most recent evaluation of RCP heat contribution.

Technical Reference(s) Waterford 3 Operating License, Amendment 225

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-TS00 Obj. 2

Question Source: Bank # _____
Modified Bank # WF3-OPS-06762
New _____

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41(b) 2

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	G 2.2.41	
	Importance Rating	3.5	

K/A Statement

Equipment Control: Ability to obtain and interpret station electrical and mechanical drawings.

Proposed Question: RO 70 Rev: 0

Given the following:

- Main Steam Drip Pot MS-ILS-0313A has detected a high level
- MS Drain Valve, MS-127A has automatically opened

MS-127A will close:

A. when the high level condition clears and the control switch is taken to close.

B. if valve limit switch contact (bo) indicates the valve did not stroke fully open.

C. when the white Water Detected lights on the control switch extinguishes.

D. when the high level condition clears and relay LSX de-energizes.

Proposed Answer: A

Explanation:

A. **CORRECT**. The high level condition must be cleared and then the operator can manually close the valve.
B. Incorrect. This valve does not have an auto close feature if the valve does not fully stroke.
C. Incorrect. The white lights will extinguish when the high level condition clears but it will not cause closure of the valve.
D. Incorrect. The LSX relay disables closure of the valve with a high level condition but does not provide auto closure when the high condition clears.

Technical Reference(s) CWD B424-1648, Rev. 8

Proposed references to be provided to applicants during examination: CWDs B424-1648

Learning Objective: WLP-OPS-CD00 Obj. 3

Question Source: Bank # WF3-OPS-2357-A
Modified Bank #
New

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41(b) 4

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	G 2.3.15	
	Importance Rating	2.9	

K/A Statement

Radiation Control: Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question: RO 71 Rev: 0

The _____(1)_____ Radiation Monitor(s) is(are) susceptible to Thermally Induced Currents (TIC) during a LOCA/Steam Line Break and will initially read erroneously _____(2)_____ with rising temperatures in Containment.

A. (1) Containment High Range
   (2) low

B. (1) Containment High Range
   (2) high

C. (1) Containment PIG
   (2) low

D. (1) Containment PIG
   (2) high

Proposed Answer: B

Explanation:

A. Incorrect. Correct rad monitors, wrong indication effect.
B. **CORRECT**. The Containment High Range Radiation Monitors have been determined to be susceptible to TIC post accident which will cause erroneously high readings for at least 15 minutes from the time containment temperature stabilizes.
C. Incorrect. Wrong rad monitor, wrong effect.
D. Incorrect. Wrong rad monitor, correct effect.

Technical Reference(s) WLP-OPS-MCD06, Radiation Monitoring, Rev. 4

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-MCD06 Obj.3

Question Source: Bank # _____
Modified Bank # WF3-OPS-5638-A
New _____

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41(b) 11

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	G 2.3.12	
	Importance Rating	3.2	

K/A Statement

Radiation Control: Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Proposed Question: RO 72 Rev: 0

Given the following:

- The plant is at 10% Power

During a containment entry, which of the following areas inside containment is forbidden to enter?

A. Pressurizer Cubicle below +21' elevation

B. +46' elevation at the Quench Tank

C. Main Steam Line Crossovers on the +46' elevation

D. 1A Cold Leg penetration through the 'D' Ring Wall

Proposed Answer: D

A. Incorrect. This is an area known to have high radiation levels, but is not listed in HP-001-213 as being forbidden in MODE 1. Not an area listed as needing RP Manager approval to enter.
B. Incorrect. This area is in close proximity to the Reactor Cavity but is a sufficient distance away that it is not forbidden, or need RP Manager approval to enter.
C. Incorrect. This is an exception to the requirement for obtaining RP Manager approval for going above the actual +46' elevation in Containment. It is not listed as forbidden.
D. **CORRECT**. Per HP-001-213, Step 5.2.2 and Attachment 7.1, this is a forbidden area in MODE 1. (> 5% RTP).

Technical Reference(s) HP-001-213, Control of Containment Building Power Entries, Rev. 301

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPA00 Obj. 3

Question Source: Bank # _____
Modified Bank # WF3-NRC-5770-A
New _____

Question History: Last NRC Exam 2000 W3 NRC Exam

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

10 CFR Part 55 Content: 55.41(b) 12

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	G 2.4.50	
	Importance Rating	4.2	

K/A Statement

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

Proposed Question: RO 73 Rev: 0

Which of the following alarms/status is used to verify a SIAS has occurred during Standard Post Trip Action verification, per OI-038-000, Emergency Operations Procedures Operations Expectations/ Guidance?

A. RPS Channel Trip PZR Pressure Lo - illuminated

B. Train A(B) ESF Valves Overload Override - extinguished

C. SIAS Train A(B) Logic initiated – illuminated

D. LPSI/HPSI Pump A(B) Pump unavailable – extinguished

Proposed Answer: C

Explanation (Optional):

A. Incorrect. This alarm does come in when SIAS occurs but is not used by OI-038-000 and would be in if only one RPS channel has tripped.
B. Incorrect. This alarm actually comes in when SIAS occurs.
C. **CORRECT**. These are the alarms called out in OI-038-000 to support verifying a SIAS.
D. Incorrect. These alarms, if extinguished > 47 seconds after an SIAS, occurs would tell the Operator that the breakers for the associated pumps closed. However, OI-038-000 does not use them for verification of an SIAS.

Technical Reference(s) OI-038-000, Rev. 11

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE01 Obj. 4

Question Source: Bank # _____
Modified Bank # _____
New X

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41(b) 5 / 7 / 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	G 2.4.30	
	Importance Rating	2.7	

K/A Statement

Emergency Procedures/Plan: Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator

Proposed Question: RO 74 Rev: 0

Which of the following is the lowest classification that requires notification of the Coast Guard and Union Pacific Railroad to establish Exclusion Area Boundary Controls?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Per EP-002-010, Notifications and Communications the USCG and Union Pacific Railroad will be notified at a Site Area Emergency or General Emergency.
B. Incorrect. Per EP-002-010, Notifications and Communications the USCG and Union Pacific Railroad will be notified at a Site Area Emergency or General Emergency.
C. **CORRECT**. Per EP-002-010, Notifications and Communications the USCG and Union Pacific Railroad will be first be notified at a Site Area Emergency.
D. Incorrect. Per EP-002-010, Notifications and Communications the USCG and Union Pacific Railroad will first be notified at a Site Area Emergency.

Technical Reference(s) EP-002-010, Notifications and Communications, Rev. 304

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-EP00 Obj. 12

Question Source: Bank # ____
Modified Bank # ____
New X

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41(b) 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	G 2.4.21	
	Importance Rating	4.0	

K/A Statement

Emergency Procedures / Plan: Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.

Proposed Question: RO 75 Rev: 0

Which of the following would indicate failure to meet a Safety Function acceptance criteria for Containment Isolation per OP-902-008, Safety Function Recovery?

A. S/G 1 activity rising without explanation, Control Room is commencing rapid cooldown to T_{hot} < 520ºF per HR-2 of OP-902-008.

B. Containment pressure is 16.5 psia and Containment Area Radiation Monitors are rising without explanation, and no containment isolation.

C. Containment Pressure is 17.5 psia and only RC-606, RCP Controlled Bleedoff Inside Isolation Valve closes on the associated penetration.

D. MSL 2 Radiation Monitor is in high alarm, S/G 2 isolation has been completed, S/G 2 pressure is 945 psia, no steam is issuing from ADV 2.

Proposed Answer: A

Explanation:

A. **CORRECT**. Conditions given in the answer indicate that the S/G has not been isolated yet per procedure.
B. Incorrect. Containment isolation is not required until Containment Pressure is > 17.1 psia. Therefore, at this time no containment isolation valve is required to be closed.
C. Incorrect. OP-902-008 only requires one valve per penetration to be closed to meet the acceptance criteria.
D. Incorrect. All acceptance criteria for OP-902-008 are met as long as the S/G is isolated and the ADV is not steaming.

Technical Reference(s) OP-902-008, Safety Function Recovery, Rev.15

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE08 Obj. 8

Question Source: Bank # _____
Modified Bank # _____
New X

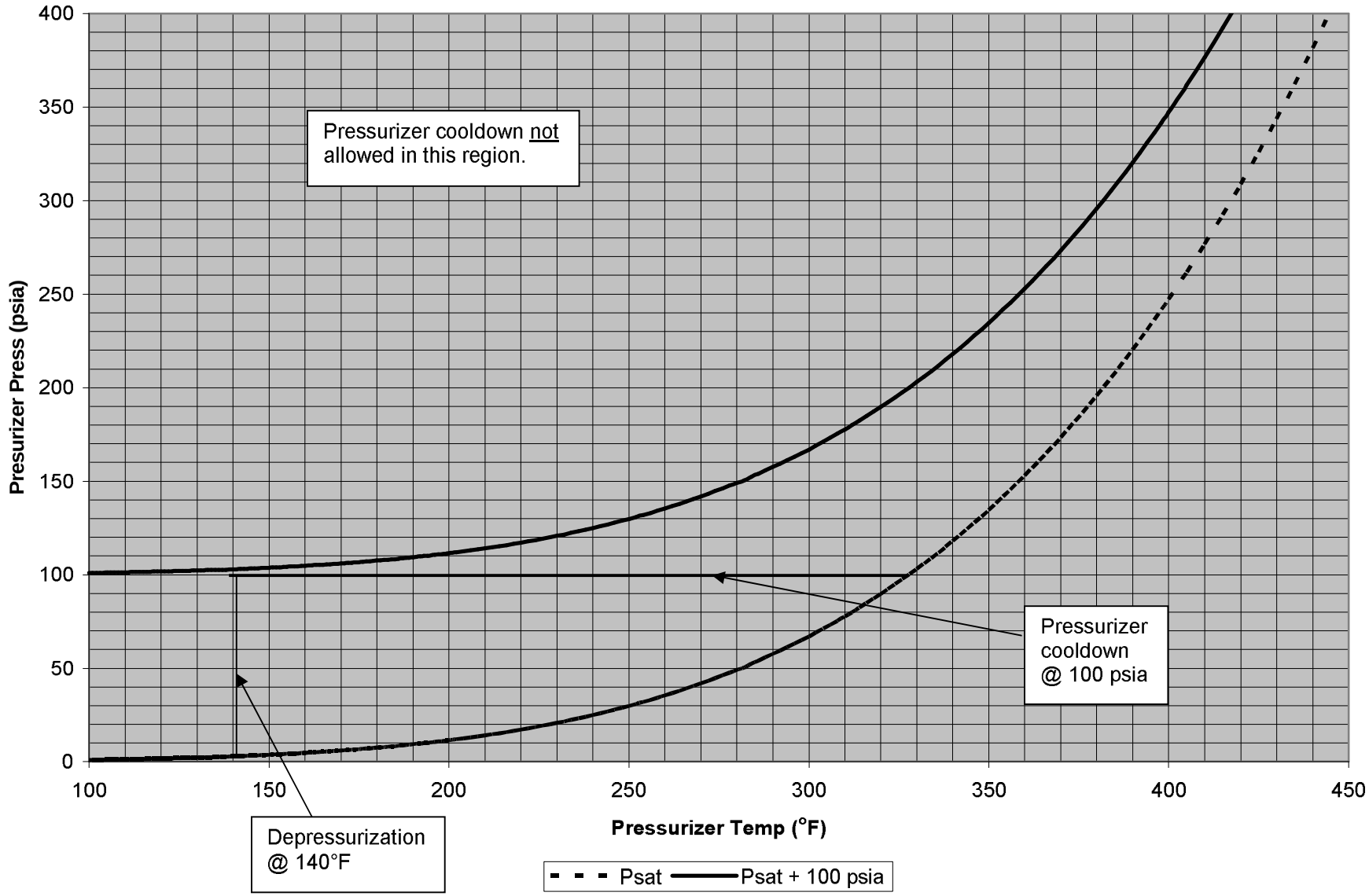
Question History: Last NRC Exam N/A

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

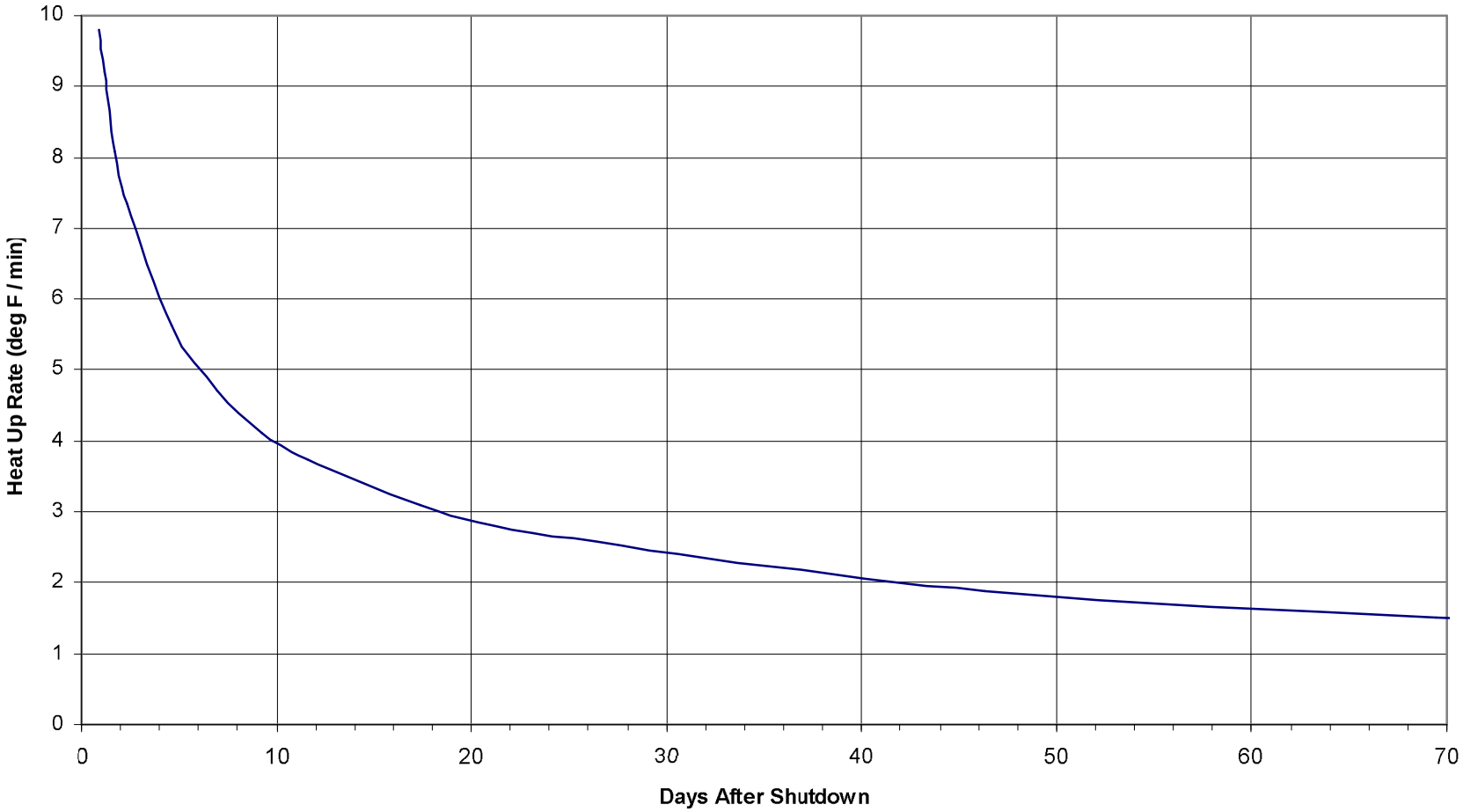
10 CFR Part 55 Content: 55.41(b) 10

Comments:

9.5 PRESSURIZER SATURATION & P_{SAT} + 100 PSIA



ATTACHMENT 2: HEATUP RATE VS TIME AFTER SHUTDOWN



Note: This graph assumes RCS level at midloop.

11.2 HYDROGEN RECOMBINER POWER CONTROL SETTING DATA SHEET

SECTION 6.1 (6.2)

STEP # _____

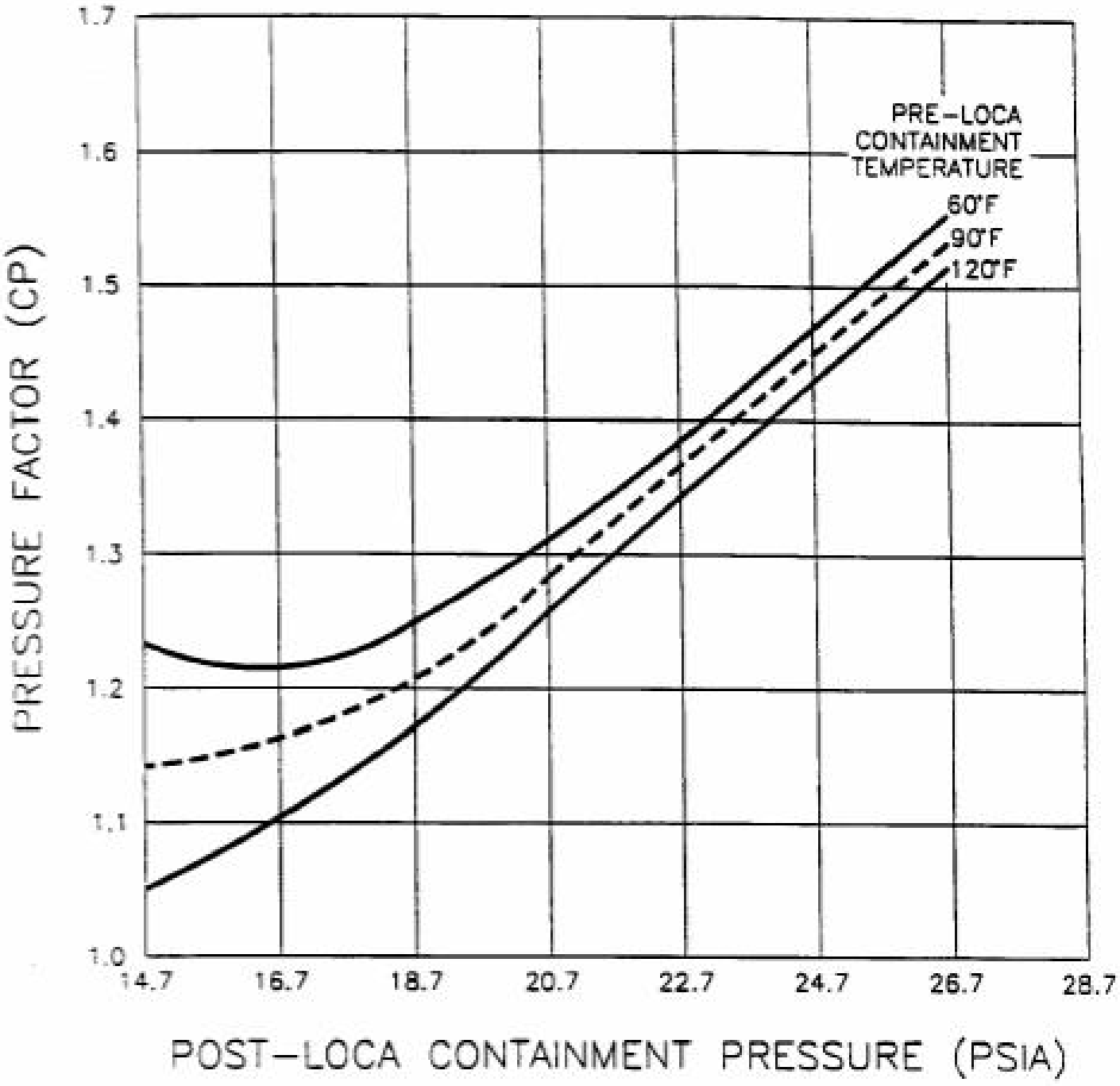
6.1.1.1 (6.2.4.1)	Post-LOCA Containment Pressure, ESF-IPI-6750A (CP-8)	_____
6.1.1.2 (6.2.4.2)	Pre-LOCA Containment Temperature (from OP-903-001)	_____
6.1.1.3.1 (6.2.4.3.1)	_____ Cp (from Attachment 11.4)	
6.1.1.4.1 (6.2.4.4.1)	48 KW X Cp = Recombiner Power Control Setting	
	48 KW X _____ = _____	

1

Performed by: _____ _____
(Signature) (Date)

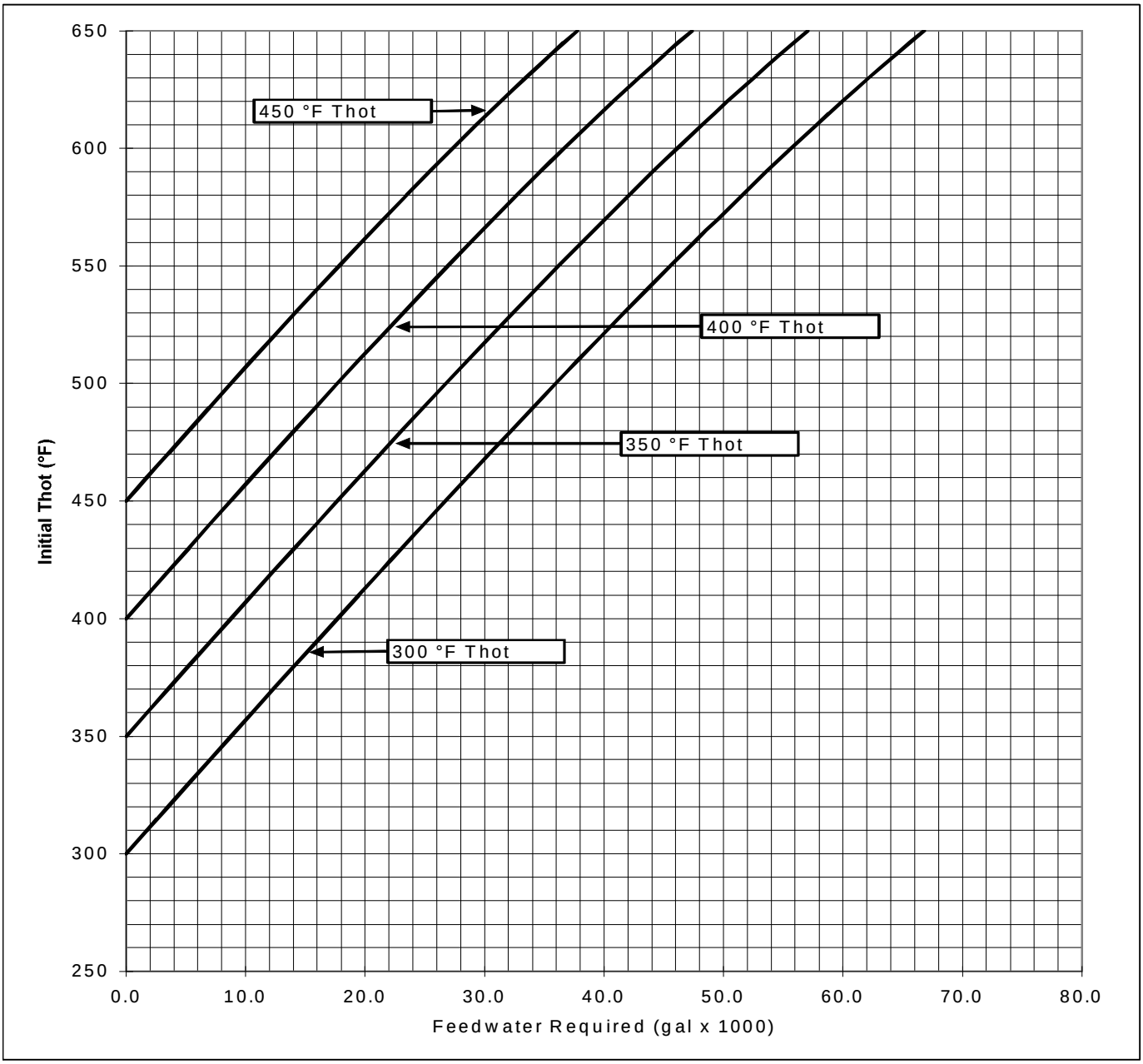
Verified by: _____ _____
(Signature) (Date)

11.4 DRY CONTAINMENT RECOMBINER POWER CORRECTION FACTOR GRAPH



Figures

Attachment 2-G: Feedwater for Sensible Heat Removal

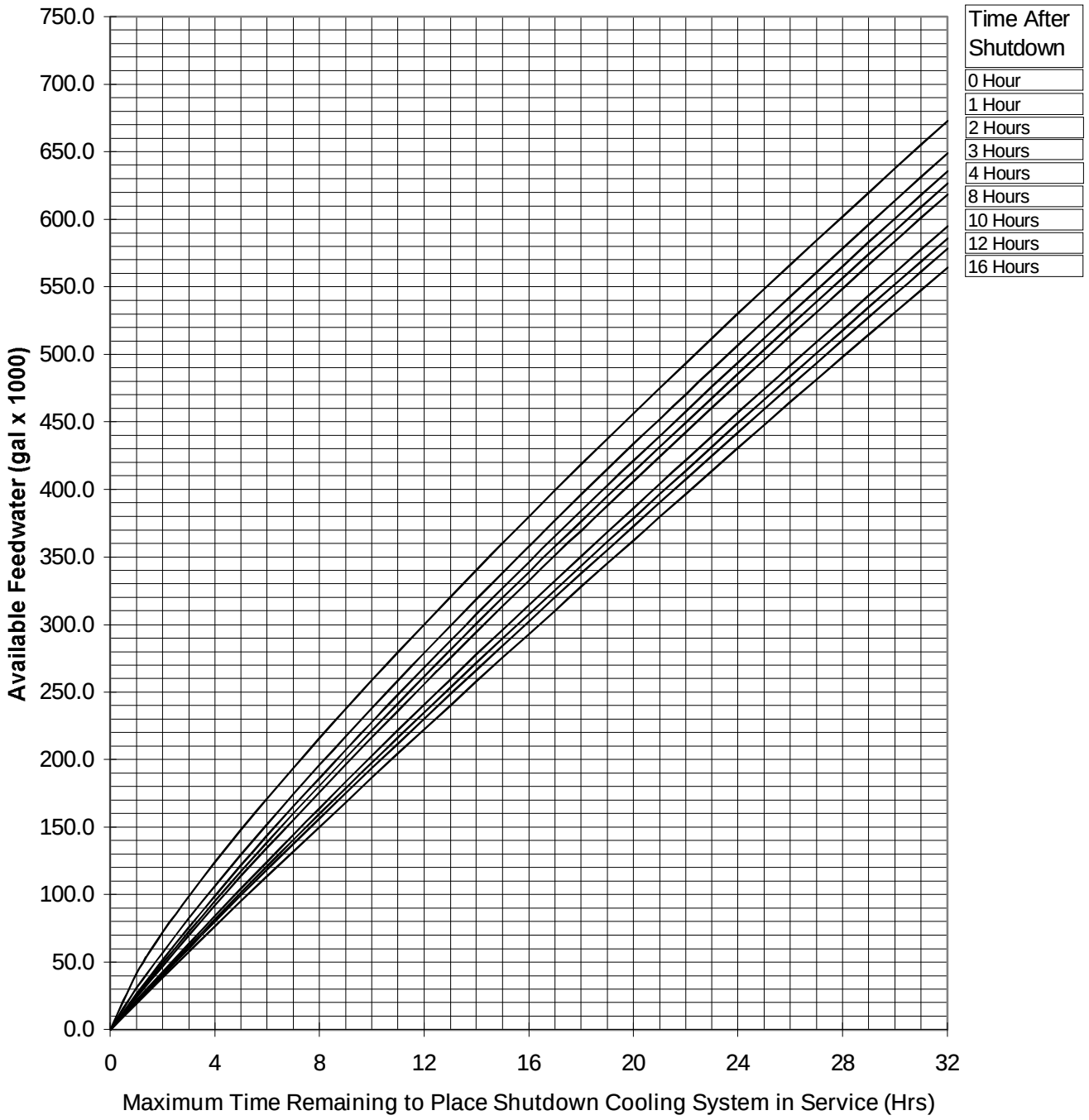


DWST Volume = 5000 gal / % x _____ % - 58,000 gal = _____ gal
CSP Volume = 2106 gal / % x _____ % -53,000 gal = _____ gal
Total Feedwater = DWST Volume + CSP Volume = _____ gal
Available Feedwater = Total Feedwater – Feedwater Required from Att. 2-G
= _____ Total - _____ Required = _____gal

End of Attachment 2-G

Figures

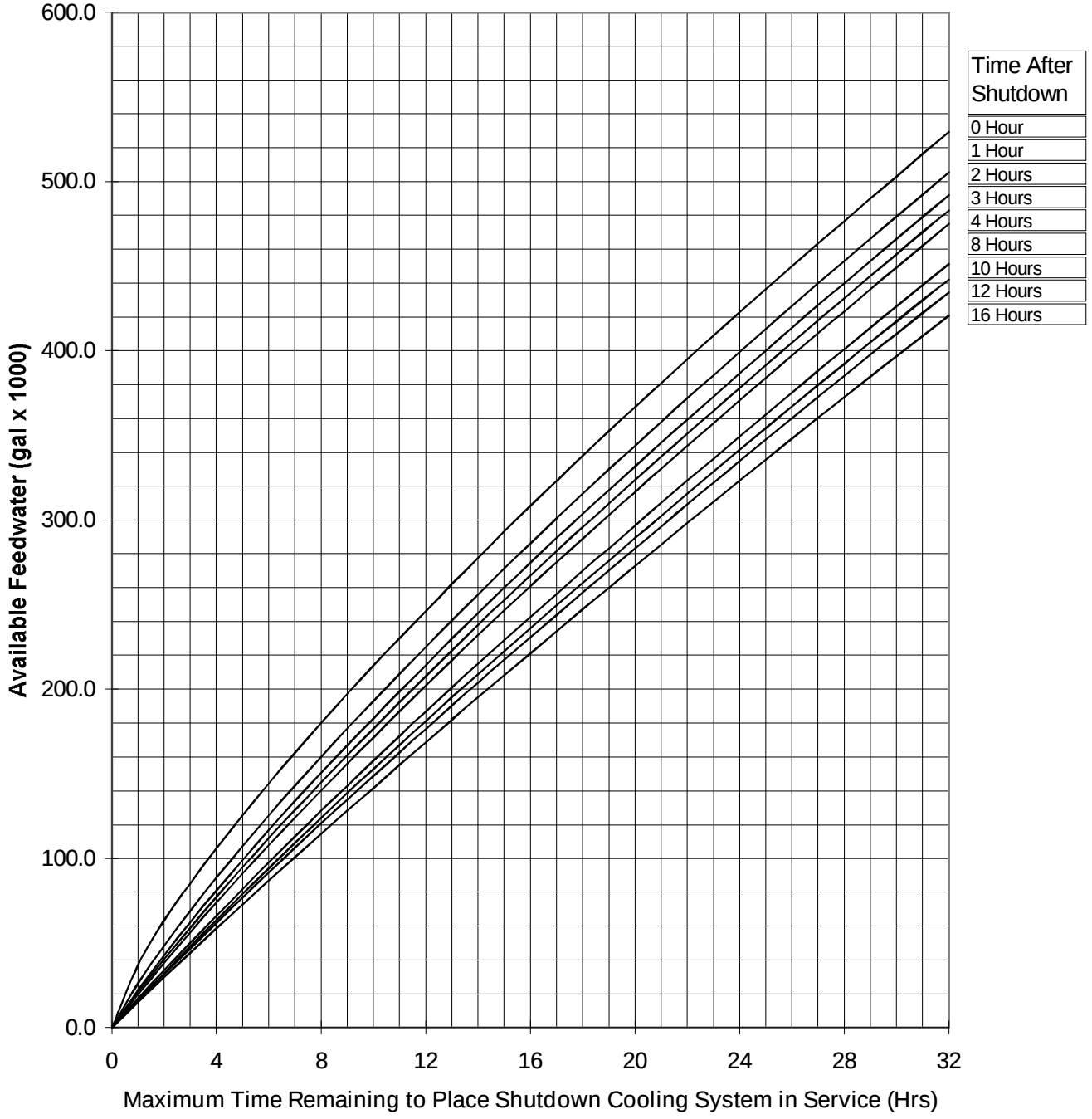
End of Attachment 2-H: Condensate Inventory Curve Four RCPs Operating



End of Attachment 2-H

Figures

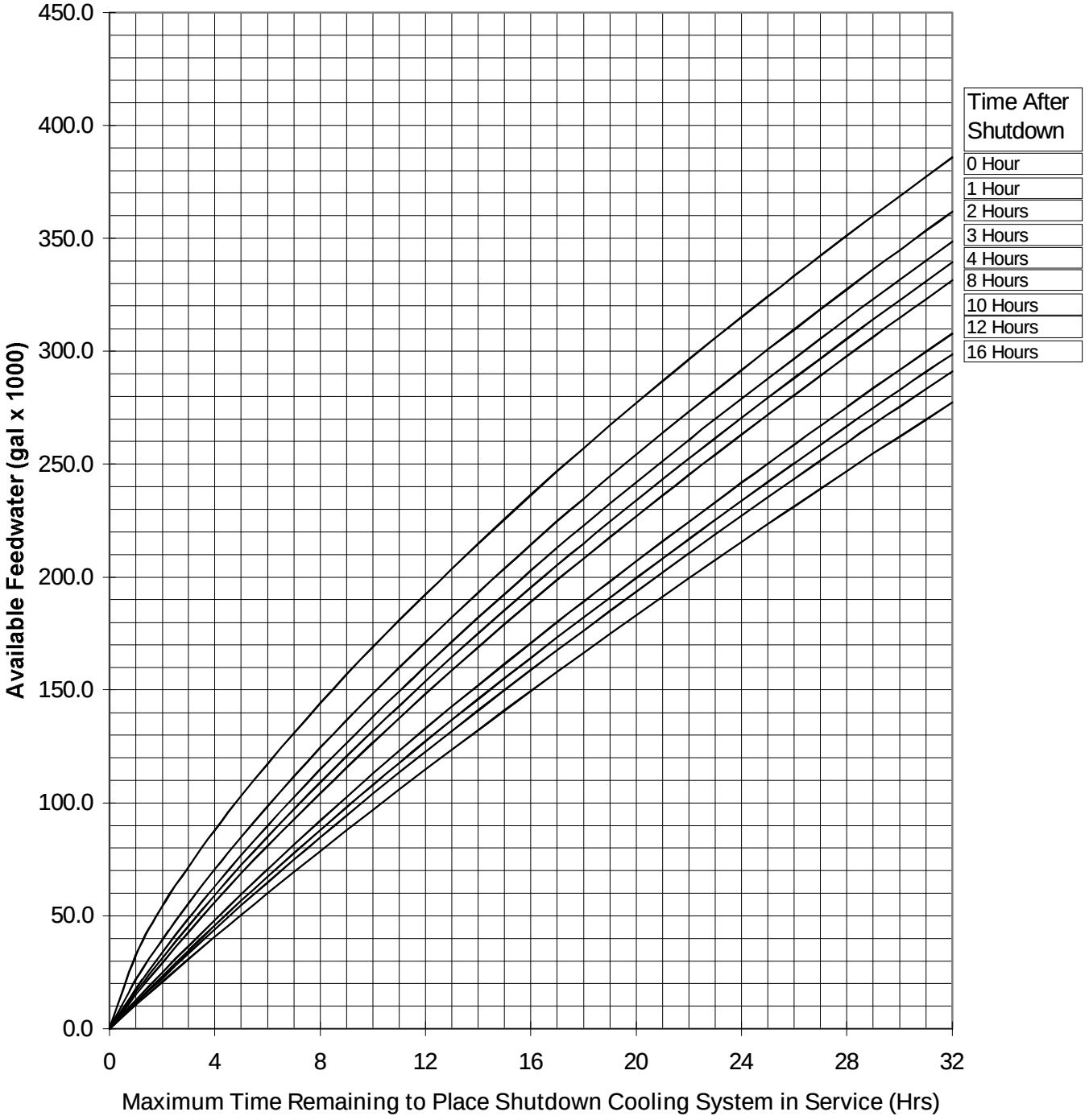
Attachment 2-I: Condensate Inventory Curve Two RCPs Operating



End of Attachment 2-I

Figures

Attachment 2-J: Condensate Inventory Curve NO RCPs Operating



End of Attachment 2-J

3/4.4.7 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.7 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 1.0 microcurie/gram DOSE EQUIVALENT I-131, and

b. Less than or equal to 100/Î microcuries/gram.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1, 2, and 3*:

a. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding 60 microcuries/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY with Tavg less than 500°F within 6 hours.

b. With the specific activity of the primary coolant greater than 100/Î microcuries/gram, be in at least HOT STANDBY with Tavg less than 500°F within 6 hours.

MODES 1, 2, 3, 4, and 5:

c. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 or greater than 100/Î microcuries/gram, perform the sampling and analysis requirements of item 4 a) of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

4.4.7 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

---

* With T_{avg} greater than or equal to 500°F.

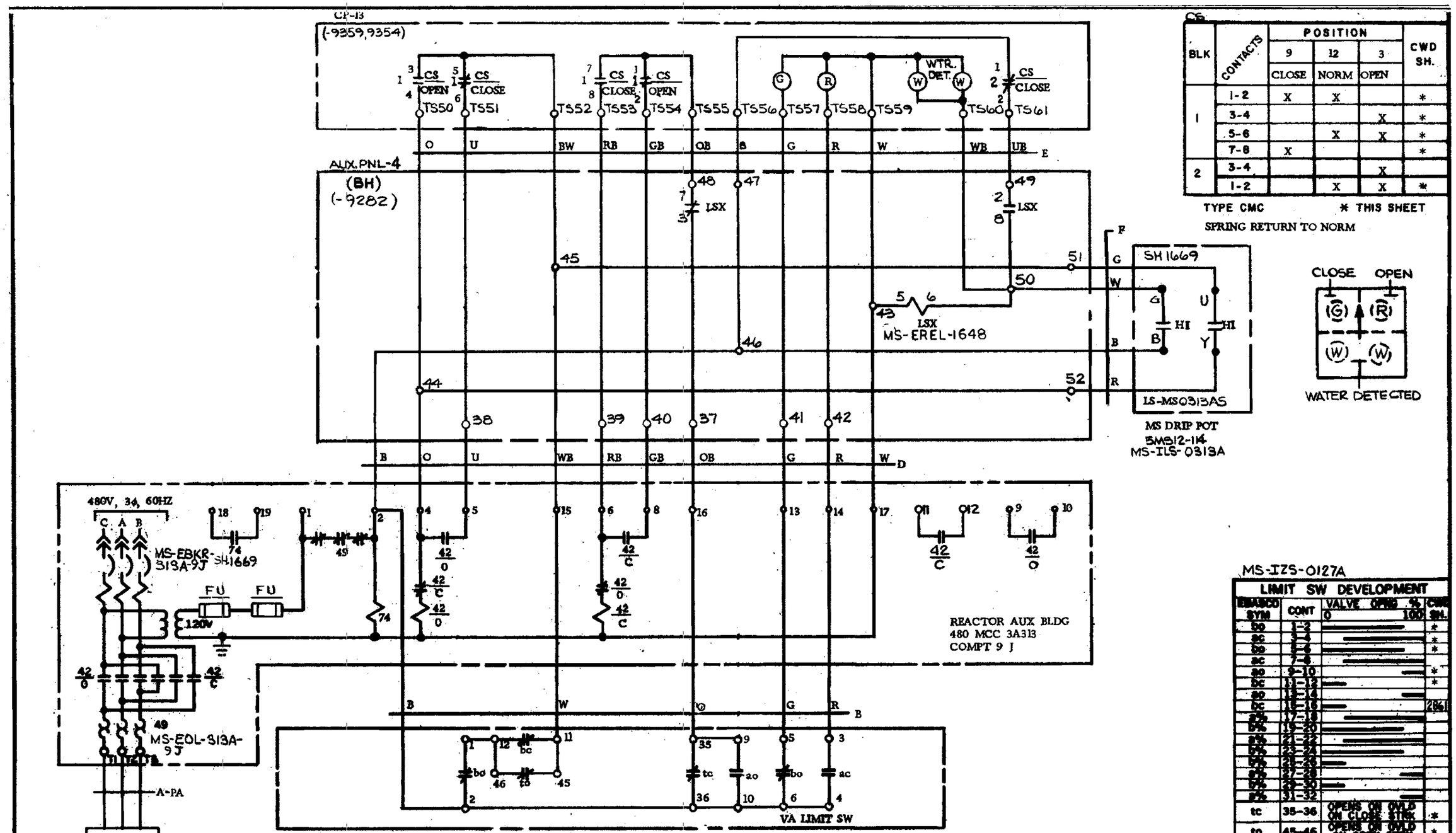
TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Activity Determination	At least once per 72 hours	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1
3. Radiochemical for $\bar{E}$ Determination	1 per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1.0 µCi/gram, DOSE EQUIVALENT I-131 or 100/$\bar{E}$ µCi/gram, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 % of the RATED THERMAL POWER within a 1-hour period.	1, 2, 3

* Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

Until the specific activity of the primary coolant system is restored within its limits.



CS

BLK	CONTACTS	POSITION 9 CLOSE	12 NORM	3 OPEN	CWD SH.
1	1-2	X	X		*
	3-4			X	*
	5-6		X	X	*
	7-8	X			*
2	3-4			X	
	1-2		X	X	*

TYPE CMC * THIS SHEET

SPRING RETURN TO NORM

MS-IZS-0127A

LIMIT SW DEVELOPMENT

EBASCO SYM	CONT	VALVE OPENS % 0 100	CWD SH.
bo	1-2		*
ac	3-4		*
bo	5-6		*
ac	7-8		
ao	9-10		*
bc	11-12		*
ao	13-14		
bc	15-16		2861
ao	17-18		
bo	19-20		
ao	21-22		
bo	23-24		
bo	25-26		
ao	27-28		
bo	29-30		
ao	31-32		
tc	35-36	OPENS ON OVLD ON CLOSE STRK	*
to	45-46	OPENS ON OVLD ON OPEN STRK	*

CONT CLOSED * = THIS SHEET

This drawing was used in its entirety to scan and rasterize to produce, this revision, REV 7

REV	DATE	BY	APPROVED	REV	DATE	BY	APPROVED
8	9-19-95	LLB	MC	4	2-13-84		
7	9-18-90	EJB	VI	3	10-25-82		
6	4/25/86			2	1-22-79	AC	
5	7-31-84						

EBASCO SERVICES INCORPORATED
NEW YORK
DIV. I&C DR. S.P.
SCALE NONE CH. MAZ
DATE JUN 16, 1976.
APPROVED

LOUISIANA POWER & LIGHT CO.
WATERFORD S. E. S. UNIT No.3
CONTROL WIRING DIAGRAM
STM.LINE I DOWNSTREAM DRAIN
VA 5MS-V683

LOU-1564
B-424
SHEET 1648
2

(G-165-2)

INCORP. DRN E-9500913

ORIGINAL

Received 950927-008

LP&L Microfilmed

FILENAME: 00006949.bwa

2010 NRC Exam
SRO Written Exam Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	004 A2.01	
	Importance Rating		4.2

K/A Statement

Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: RCS pressure allowed to exceed limits

Proposed Question: SRO 1 Rev: 0

The in service Pressurizer level channel has failed low. All 3 Charging Pumps are running. The crew has entered OP-901-110, Pressurizer Level Control Malfunction.

With these conditions, Pressurizer pressure will _____(1)_____ until _____(2)_____

	(1)	(2)
A.	rise	Letdown flow is raised to ~ 126 gpm.
B.	rise	the Pressurizer Spray valves are manually opened.
C.	drop	Letdown flow is raised to ~ 126 gpm.
D.	drop	the Pressurizer Spray valves are manually opened.

Proposed Answer: A

Explanation (Optional):

A. CORRECT: Due to the failure, Pressurizer level will rise which will in turn make Pressurizer pressure rise. OP-901-110 directs raising Letdown flow to match Charging flow to restore Pressurizer level. With Charging and Letdown matched, the Pressurizer level rise will stop and pressure will stabilize.
B. INCORRECT: OP-901-110 does not direct taking the Pressurizer Spray valves to manual. This malfunction does not impair the Spray valves and taking manual control would not be appropriate.
C. INCORRECT: Pressure will not drop with these conditions.
D. INCORRECT: Pressure will not drop with these conditions.

2010 NRC Exam
SRO Written Exam Worksheet

Technical Reference(s) OP-901-110 (Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PLC00 Obj. 7 (As available)
WLP-OPS-PPO10 Obj. 3

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 _____
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	005 G2.2.37	
	Importance Rating		4.6

K/A Statement
Residual Heat Removal System (RHRS): Ability to determine operability and/or availability of safety related equipment.

Proposed Question: SRO 2 Rev: 0

The following plant conditions exist:

- Plant is in Mode 6, core off load has **not** commenced
- Reactor has been shutdown for 6.5 days
- RCS temperature is 130 °F and stable
- Reactor Cavity water level is 24 feet above the fuel
- Shutdown Cooling (SDC) flow is 4050 gpm

The Refueling Team supervisor has contacted you to request a reduction in SDC flow to limit turbulence during submarine operation in the Refuel Cavity.

Which ONE of the following actions is allowed for the current plant conditions?

Shutdown Cooling flow can be lowered to ____(1)____ gpm ____(2)____

	(1)	(2)
A.	3100	if RCS temperature is verified < 135 °F every hour.
B.	3100	since the plant has been shutdown for > 150 hours.
C.	2100	if RCS temperature is verified < 135 °F every hour.
D.	2100	since the plant has been shutdown for > 150 hours.

Proposed Answer: A

Explanation (Optional):

A. CORRECT: RCS temperature less than 135° F allows reduction in flow to a minimum of 3000 gpm by verifying hourly that RCS temperature remains below the limit.
B. INCORRECT: The plant must be shutdown for > 175 hours before flow can be lowered to < 4000 gpm with no other restrictions.
C. INCORRECT: RCS temperature less than 135° F allows reduction in flow to a minimum of 3000 gpm. The plant must be shutdown for > 375 hours for flow to be lowered to ≥ 2000 gpm, regardless of RCS temperature.
D. INCORRECT: The plant must be shutdown for > 375 hours for flow to be lowered to ≥ 2000 gpm, regardless of RCS temperature.

2010 NRC Exam
SRO Written Exam Worksheet

Technical Reference(s) T.S. 3.9.8.1 (Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-REQ04 Obj 2 (As available)

Question Source:
- Bank # _____ 5529A
- Modified Bank # X (Note changes or attach parent)
- New _____

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content:
- 55.41 _____
- 55.43 6

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	008 G2.2.37	
	Importance Rating		3.6

K/A Statement

Component Cooling Water System: Ability to determine operability and/or availability of safety related equipment.

Proposed Question: SRO 3 Rev: 0

Component Cooling Water Surge Tank level switch CC-ILS-7011 A has failed low. CC-538 A, Component Cooling Water Makeup Valve A, has opened as a result of the failure, and the CCW Surge Tank level is rising.

As a result of this failure, the CRS directed isolating CC-538 A. Maintenance is planning their repair of CC-ILS-7011 A. The Work Management SRO should verify that the maintenance will not prevent alignment of CCW Makeup for greater than ____(1)____ hours or ____(2)____ will require entry.

	(1)	(2)
A.	2	Tech Spec 3.8.1.1 and cascading Tech Specs
B.	2	Tech Spec 3.7.3 and cascading Tech Specs
C.	4	Tech Spec 3.8.1.1 and cascading Tech Specs
D.	4	Tech Spec 3.7.3 and cascading Tech Specs

Proposed Answer: B

2010 NRC Exam
SRO Written Exam Worksheet

Explanation (Optional):

OP-100-014, Technical Specification and Technical Requirements Compliance, describes that CCW Makeup can be isolated for 2 hours without any Tech Spec consequences, not 4 hours. If CCW Makeup is not available for over 2 hours, then entry into Tech Spec 3.7.3 and cascading Tech Specs is required. Cascading Tech Specs does include Tech Spec 3.8.1.1, but Waterford does not cascade on entry into Tech Spec 3.8.1.1. Both answers are incorrect that state enter Tech Spec 3.8.1.1 and cascading Tech Specs.

Technical Reference(s) OP-100-014, Att. 6.6 (8 of 26) (Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-CC00 Obj. 9 (As available)

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 _____
55.43 2

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	064 A2.11	
	Importance Rating		2.9

K/A Statement

Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Conditions (minimum load) required for unloading an ED/G

Proposed Question: SRO 4 Rev: 0

Plant conditions are as follows:

- Emergency Diesel Generator B has been started for OP-903-068, Emergency Diesel Generator and Subgroup Relay Operability Verification.
- EDG B load is 1.9 MW
- EDG B Speed Adjust switch has failed and the BOP operator cannot raise or lower load.
- Following PME troubleshooting and fuse replacement, EDG B speed control has been restored.
- EDG B operated at 1.9 MW for a total of 7 hours.

The impact of operating in the described condition is that there is a potential for ____(1)____ on EDG B; the CRS should direct the BOP to use OP-009-002, Emergency Diesel Generator, and ____(2)____ to address this issue.

	(1)	(2)
A.	the fuel injection pumps to develop leaks due to low flow	Establish reactive load (MVAR) ≤ 0.1 MW real load
B.	the fuel injection pumps to develop leaks due to low flow	raise load to > 4.0 MW over a 90 minute period
C.	the buildup of unburned exhaust products	Establish reactive load (MVAR) ≤ 0.1 MW real load
D.	the buildup of unburned exhaust products	raise load to > 4.0 MW over a 90 minute period

Proposed Answer: D

Explanation (Optional):

Selections A & B describe the impact as the fuel injection pumps developing leaks due to low flow. There is a precaution that describes the effects of operating a low loads and the effects on the fuel injection pumps. However, there is no limit to load or duration based on the injection pumps,. Additionally, the injection pump concern at load loads is long term reliability.

Selection C has the correct impact, but the action described is the action the BOP must take to prevent Reverse Power trips and has no applicability to long term operation at low loads.

Technical Reference(s): OP-009-002 Section 3.0 (Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-EDG00 Obj 8 (As available)

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 _____
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	061 A2.04	
	Importance Rating		3.8

K/A Statement
Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: pump failure or improper operation

Proposed Question: SRO 5 Rev: 0

The following plant conditions exist:

- EFW pump A is tagged OOS due to pump maintenance
- A loss of both Main Feedwater pumps required a manual reactor trip
- EFAS-1 and EFAS-2 actuated 60 seconds after the reactor trip
- EFW pump B is running
- EFW pump AB started but tripped on overspeed due to a failure of the pump governor
- S/G 1 level 10% NR and slowly rising
- S/G 2 level 60% WR and stable

Which ONE of the following states the procedure to be entered and the <u>required</u> operator action to re-establish proper feedwater flow?

A. OP-902-006, Loss of Main Feedwater Recovery; immediately stop all running RCPs then attempt to restart the EFW AB pump

B. OP-902-006, Loss of Main Feedwater Recovery; Attempt to restart the EFW AB pump, if unsuccessful in 30 minutes stop all running RCPs.

C. OP-902-008, Safety Function Recovery; immediately stop all running RCPs then attempt to restart the EFW AB pump

D. OP-902-008, Safety Function Recovery; Attempt to restart the EFW AB pump, if unsuccessful in 30 minutes stop all running RCPs.

Proposed Answer: B

Explanation (Optional):

A. INCORRECT: OP-902-006 supports staying in the procedure with only 1 MD EFW pump running provided at least one S/G is being maintained or restored 50-70% NR, but only 2 RCPs are initially required to be secured, the other 2 RCPs are required to be secured if MFW is lost > 30 minutes and only one MD EFW pump is operating

B. CORRECT: OP-902-006, Loss of Main Feedwater Recovery, directs a restart of the EFW AB pump but after 30 minutes with only one motor driven pump operating then stop all running RCPs.

C. INCORRECT: OP-902-008 does not need to be entered if at least one S/G is being restored to 50-70% NR; only 2 RCPs are initially required to be secured, the other 2 RCPs are required to be secured if MFW is lost > 30 minutes and only one MD EFW pump is operating.

D. INCORRECT: OP-902-008 does not need to be entered if at least one S/G is being restored to 50-70% NR; actions are correct.

Technical Reference(s): OP-902-006 Section 4.7 and 4.11 (Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-EFW00 Obj 4 (As available)
WLP-OPS-PPE06 Obj. 8 & 9

Question Source: Bank # _____ 08417
Modified Bank # X (Note changes or attach parent)
New _____

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 _____
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	015 A2.04	
	Importance Rating		3.8

K/A Statement

Ability to (a) predict the impacts of the following malfunctions or operations on the NIS; and (b based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Effects on axial flux density of control rod alignment and sequencing, xenon production and decay, and boron vs. control rod reactivity change

Proposed Question: SRO 6 Rev: 0

CEA 51, a CEA located on the outer perimeter of the core, has dropped into the core with the plant at 100% power.

Because of this event, core ____(1)____ will exceed Tech Spec and Core Protection Calculator limits. OP-901-102, CEA or CEDMCS Malfunction, requires starting a power reduction within 15 minutes of the dropped CEA and that ____(2)____ indications be used to monitor reactor power during the power reduction.

	(1)	(2)
A.	Axial Shape Index	Calibrated Excore Nuclear Power
B.	Axial Shape Index	Secondary Cal Power
C.	Azimuthal Power Tilt	Calibrated Excore Nuclear Power
D.	Azimuthal Power Tilt	Secondary Cal Power

Proposed Answer: D

Explanation (Optional):

Selections A and B both incorrectly list ASI as the parameter that is impacted by the dropped CEA. OP-901-102 does direct the operator to verify ASI in this procedure because it is written for any CEA deviation of > 7 inches. If a CEA was misaligned but not dropped, ASI would be affected. For a dropped outer perimeter CEA, Azimuthal Power Tilt will be affected. Selection A and C list the excore power indications as the procedure directed power indication to use during the power reduction. The excore powers will be significant affected by the dropped CEA, since an outer perimeter CEA was specified. OP-901-102 directs using several power indications, including Secondary Calorimetric Power.

Technical Reference(s) OP-901-102, E1 Step 8 (Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPO10 Obj. 3 (As available)

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 _____
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	033 G2.4.4	
	Importance Rating		4.7

K/A Statement
Spent Fuel Pool Cooling System (SFPCS): Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Proposed Question: SRO 7 Rev: 0

The following plant conditions exist:

- Plant is in Mode 6
- A full core off load to the Spent Fuel Pool is in progress.
- The following alarms have been received at CP-2:
  - Fuel Pool Temperature Hi
  - Fuel Pool Pumps Disch Press Lo
  - Fuel Pool Level Lo

Based on these indications, the CRS should enter
_____(1)_____ and suspend _____(2)_____

	(1)	(2)
A.	OP-901-510, Component Cooling Water System Malfunction	spent fuel movement and fill the Spent Fuel Pool.
B.	OP-901-510, Component Cooling Water System Malfunction	core alterations and open the CCW isolations to the AB Header.
C.	OP-901-513, Spent Fuel Pool Cooling Malfunction	spent fuel movement and fill the Spent Fuel Pool.
D.	OP-901-513, Spent Fuel Pool Cooling Malfunction	core alterations and open the CCW isolations to the AB Header.

Proposed Answer: C

Explanation (Optional):
Indications given include both low level and high temperature condition in the Spent Fuel Pool. OP-901-513 directs the operator to leave OP-901-513 and enter OP-901-510 if the event in progress is a loss of CCW to the Spent Fuel Pool. With the alarms for low level and low discharge pressure, the CRS should stay in OP-901-513. Suspending spent fuel movement and filling the Spent Fuel Pool is consistent with the guidance in OP-901-513. Aligning flow to CCW Header AB would be appropriate if the event was related to a loss of CCW.

Technical Reference(s) OP-901-513
OP-901-510
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPO50 Obj 3 (As available)

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content:
55.41 _____
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	071 A2.08	
	Importance Rating		2.8

K/A Statement
Ability to (a) predict the impacts of the following malfunctions or operations on the Waste Gas Disposal System ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Meteorological changes

Proposed Question: SRO 8 Rev: 0

The following plant conditions exist:

- Gaseous Decay Tank C release is in progress per OP-007-003
- National Weather Service issued a Severe Thunderstorm Warning for St. Charles Parish area

Current Meteorological Conditions are:

- Primary 10 Meter Wind Speed is 2.58 m/s
- Primary 10 Meter Wind Direction is 70.6 Deg
- Primary Delta T is -0.34 DegC

Which ONE of the following meteorological conditions would require a termination of the release?

	10m Wind Speed	Delta T
A.	0.75	-0.45
B.	1.85	-0.36
C.	3.37	-0.27
D.	2.95	-0.18

Proposed Answer: D

Explanation (Optional):

A. INCORRECT: Per Att 11.5, Stability Class D has been established and release should continue since Delta-T is NOT entered Class E (-0.25 to +0.75) values.
B. INCORRECT: Per Att 11.5, Stability Class D has been established and release should continue since Delta-T is NOT entered Class E (-0.25 to +0.75) values.
C. INCORRECT: Per Att 11.5, Stability Class D has been established and release should continue since Delta-T is NOT entered Class E (-0.25 to +0.75) values.
D. CORRECT: Per Att 11.5, Stability Class E has been entered and release should be terminated since Delta-T is NOT trending toward Class D (-0.75 to -0.25) values.

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Technical Reference(s):	OP-007-003 pg 14 and Attachment 11.5	(Attach if not previously provided)
		(including version/revision number)

Proposed references to be provided to applicants during examination: OP-007-003 Att 11.5

Learning Objective: WLP-OPS-GWM00 Obj 9 (As available)

Question Source:	Bank #		6134-B
	Modified Bank #	X	(Note changes or attach parent)
	New		

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content:	55.41	
	55.43	5

COMMENTS

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	022 AA2.03	
	Importance Rating		3.6

K/A Statement

Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup: Failures of flow control valve or controller

Proposed Question: SRO 9 Rev: 0

The following plant conditions exist:

- Plant is operating at 100%
- Pressurizer Level is 53% and slowly lowering
- Annunciator Letdown Flow Hi/Lo at CP-4 is Lit
- The ATC reports letdown flow is 135 gpm
- The Letdown Flow Control Valves controller, RC-IHIC-0110, is in AUTO with output at 100%
- The Pressurizer Level Controller, RC-ILIC-0110, is in AUTO with output at 15%.

Based on these indications, the CRS should enter ____(1)____ and direct the ATC to take manual control of the ____(2)____ controller.

	(1)	(2)
A.	OP-901-112, Charging or Letdown Malfunction	Letdown Flow Control Valves
B.	OP-901-112, Charging or Letdown Malfunction	Pressurizer Level
C.	OP-901-110, Pressurizer Level Control Malfunction	Letdown Flow Control Valves
D.	OP-901-110, Pressurizer Level Control Malfunction	Pressurizer Level

Proposed Answer: A

Explanation (Optional):

A. CORRECT: Requires action in OP-901-112 to regain control of failed letdown flow control valve to restore normal flow.
B. INCORRECT: Correct procedure, but the Pressurizer Level Controller is not failed.
C. INCORRECT: Wrong procedure to enter for this failure even though the distractor references the correct controller.
D. INCORRECT: Incorrect procedure to enter and the wrong controller to use to lower Letdown flow..

Technical Reference(s) OP-901-112 E_2 (Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-CVC00 Obj 10 (As available)

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 _____
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	026 G 2.4.45	
	Importance Rating		4.3

K/A Statement
Loss of Component Cooling Water: Ability to prioritize and interpret the significance of each annunciator or alarm.

Proposed Question: SRO 10 Rev: 0

The following plant conditions exist:

- Plant is operating at 100% with the following Annunciators in alarm
  - CCW Surge Tank Level LO-LO on Panel M & N
  - CCW FLOW LO red alarms for all 4 Reactor Coolant Pumps
  - CCW Header A Flow LO on Panel M
- CCW Surge Tank A is reading 0%
- CCW Surge Tank B is reading 50%
- CCW pumps A and B are running

The CRS has entered OP-901-510, CCW System Malfunction. Based on these annunciators, the CRS should use sub-section ____(1)____ and direct the reactor operator to ____(2)____ to restore CCW flow to the Reactor Coolant Pumps

	(1)	(2)
A.	E_2 Loss of CCW Pump(s)	open CC-200B / 563, CCW Header B TO AB Supply & Return Isolations
B.	E_2 Loss of CCW Pump(s)	align and start CCW Pump AB to replace CCW Pump A
C.	E_1 System Leakage	open CC-200B / 563, CCW Header B TO AB Supply & Return Isolations
D.	E_1 System Leakage	align and start CCW Pump AB to replace CCW Pump A

Proposed Answer: C

Explanation (Optional):

A. INCORRECT: Incorrect procedure sub-section with the correct mitigating action. CCW Surge Tank level went low enough to split the A & B headers. The Train B side restored, the Train A side did not, indicating a leak on Train A. Flow can be restored to the RCPs from Train B.
B. INCORRECT: Incorrect procedure sub-section and the incorrect action. Aligning CCW Pump AB to the A Header will not restore flow, since there are indications of a leak on Train A.
C. CORRECT: That sub-section will direct restoring CCW flow to the RCPs.
D. INCORRECT: Correct procedure sub-section, but incorrect mitigating actions..

Technical Reference(s)	OP-901-510 OP-500-011 C-2, F-2, G-2	(Attach if not previously provided)
	SD-CC Table 1.27 and Fig 15	(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective:	WLP-OPS-CC00 Obj 7 WLP-OPS-PPO50 Obj 1	(As available)

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content:	55.41	
	55.43	5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	G2.4.41	
	Importance Rating		4.6

K/A Statement

Anticipated Transient Without Scram (ATWS): Knowledge of the emergency action level thresholds and classifications

Proposed Question: SRO 11 Rev: 0

Given the following:

- Plant is at 24% power with a power ascension in progress
- Feedwater Control #1 fails and S/G 1 levels drop
- At 23%NR in S/G 1 OP-902-000 contingency steps are performed in order and all CEAs insert when the 32A and 32B supply breakers are opened on CP-1
- After performing Standard Post Trip Actions the CRS diagnoses and enters OP-902-001 Reactor Trip Recovery
- The STA verifies that all SFSC criteria are met in OP-902-001

Classify the event per EP-001-001, Recognition and Classification of Emergency Conditions and determine the correct implementing procedure to enter.

A. EP-001-010, Unusual Event

B. EP-001-020, Alert

C. EP-001-030, Site Area Emergency

D. EP-001-040, General Emergency

Proposed Answer: C

Explanation (Optional):

A. Incorrect. The lowest level of classification for the conditions given is an Alert.
B. Incorrect. This classification would be correct if the trip had been successful using manual trip pushbuttons or DRTS pushbuttons.
C. **CORRECT**. EP-001-001 basis states that tripping the reactor using the 32 Supply breakers after exceeding a RPS setpoint would be escalation criteria from an alert to SAE.

Incorrect. The stem of the question does not support classifying a General Emergency.

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Technical Reference(s):	EP-001-001, Recognition and Classification of Emergency Conditions, Rev. 025	(Attach if not previously provided)
		(including version/revision number)

Proposed references to be provided to applicants during examination: EP-001-001

Learning Objective:	WLP-OPS-EP02 Obj. 17		(As available)
Question Source:	Bank #		
	Modified Bank #	WF3-OPS-6876-A	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	N/A	
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		
	55.43	5	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	E06 AA2.2	
	Importance Rating		4.2

K/A Statement
Ability to determine and interpret the following as they apply to the (Loss of Feedwater): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

Proposed Question: SRO 12 Rev: 0

A steam line break and subsequent MSIS has created a loss of all Main Feedwater condition. Which of the following describe a condition where 100% Emergency Feedwater flow is **NOT** available to both Steam Generators A and B as described by Tech Spec 3.7.1.2?

A. EFW-102 A, Condensate Storage Pool Outlet to Emergency Feedwater Header A Isol, is discovered fully closed with EFW-102 B, Condensate Storage Pool Outlet to Emergency Feedwater Header B Isol, fully open during the event.

B. Nitrogen Accumulator #6 ruptures, leaving EFW-224 A, Emergency Feedwater Hdr A to SG1 Primary Flow Control, and EFW-229 A, Emergency Feedwater to SG1 Backup Isolation, with no Instrument Air.

C. EFW-228 A, Emergency Feedwater to SG1 Primary Isolation, and EFW-228 B, Emergency Feedwater to SG2 Primary Isolation, failed closed during the event.

D. Emergency Feedwater Pump AB and Emergency Feedwater Pump A trip during the event, Emergency Feedwater Pump B remains running during the event.

Proposed Answer: D

Explanation (Optional):

A: Incorrect Each suction valve EFW-102 A and EFW-102 B are 100% flow capacity valves

B: Incorrect With Nitrogen accumulator 6 failed, EFW-224 A will fail to be able to control flow, but flow isolation EFW-229 A will fail open, allowing flow control with EFW-223 A.

C: Incorrect With EFW-228 A & B shut, EFW-229 A and B can pass 100% flow capacity.

D: Correct: Selection A describes a condition where the only remaining Emergency Feedwater Pump is EFW Pump B, which is not a 100% capacity pump. The Emergency Feedwater Pump AB and its steam flow valves are 100% capacity.

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Technical Reference(s)	Tech Spec 3.7.1.2 bases OP-100-014 SD – EFW	(Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-EFW00 Obj 8 (As available)

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content:	55.41	
	55.43	2

Comments

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	055 G2.1.20	
	Importance Rating		4.6

K/A Statement
Loss of Offsite and Onsite Power (Station Blackout): Ability to interpret and execute procedure steps.

Proposed Question: SRO 13 Rev: 0

The following initial plant conditions exist:

- The reactor tripped, standard post trip actions were completed, and OP-902-001, Reactor Trip Recovery is being implemented
- EDG A is OOS to replace a turbocharger bearing
- All other equipment is OPERABLE

Subsequently:

- A Loss of Offsite Power occurs
- EDG B starts and trips on Overspeed
- Battery A Disconnect shorts and fails open

Which of the following procedures should be implemented and what action should be performed to protect the main condenser?

A. OP-902-005, Station Blackout Recovery; place all Steam Bypass Valve Permissive Switches to OFF and trip both Main Feedwater Pumps.

B. OP-902-005, Station Blackout Recovery; close both Main Steam Isolation Valves and both S/G Blowdown and Sample isolations.

C. OP-902-008, Safety Function Recovery; place all Steam Bypass Valve Permissive Switches to OFF and trip both Main Feedwater Pumps.

D. OP-902-008, Safety Function Recovery; close both Main Steam Isolation Valves and both S/G Blowdown and Sample isolations.

Proposed Answer: B

Explanation (Optional):

A. INCORRECT: OP-902-005 would be implemented as long as one DC bus is energized. Per the conditions in the stem the B DC bus is still energized. The isolation methodology eliminates most steam to the Main Condenser but not all of it. Loads such as gland steam and some steam to the MSRs could potentially overpressurize the main condenser.

B. CORRECT: OP-902-005 would be implemented as long as one DC bus is energized. Per the conditions in the stem the B DC bus is still energized. Per step 6 of OP-902-005 this is the correct methodology for protecting the main condenser.

C. INCORRECT: OP-902-008 does not need to be implemented with B DC Bus energized. The isolation methodology eliminates most steam to the Main Condenser but not all of it. Loads such as gland steam and some steam to the MSRs could potentially overpressurize the main condenser.

D. INCORRECT: OP-902-008 does not need to be implemented with B DC Bus energized. Per step 6 of OP-902-005 this is the correct methodology for protecting the main condenser.

Technical Reference(s) OP-902-005 Step 6 (Attach if not previously provided)
OP-902-009 Diagnostic Flow Chart (including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE05 Obj 4 (As available)

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 _____
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	057 AA2.04	
	Importance Rating		4.0

K/A Statement
Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: ESF system panel alarm annunciators and channel status indicators

Proposed Question: SRO 14 Rev: 0

The plant is operating at 100% when the following conditions are reported:

- ALL ESFAS actuation annunciators for Train A & B are in alarm
- ESFAS lights on CP-7B & 7D are extinguished
- Reactor Trip breakers 1, 2, 5, and 6 are OPEN
- CP-7 ROM lights are extinguished on Channel B
- CPC B is de-energized

Which ONE of the following off-normal procedural actions is required for these conditions?

A. Enter OP-901-311 section E_0 to address a loss of MCC 312B

B. Enter OP-901-311 section E_0 to address a loss of MCC 313B

C. Enter OP-901-312 section E_2 to address a loss of SUPS MB

D. Enter OP-901-312 section E_4 to address a loss of SUPS MD

Proposed Answer: C

Explanation (Optional):
The indications given are consistent with a loss of SUPS MB. Selections A and B select the wrong procedure. Selection D implements the wrong procedure section. The correct procedure and section is stated in selection C.

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Technical Reference(s) OP-901-312 E_2 1 and 2 (Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPO30 Obj 4 (As available)

Question Source: Bank # 04054a
Modified Bank # X (Note changes or attach parent)
New

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	003 AA2.03	
	Importance Rating		3.8

K/A Statement
Ability to determine and interpret the following as they apply to the Dropped Control Rod: Dropped rod, using in-core/ex-core instrumentation, in-core or loop temperature measurements

Proposed Question: SRO 15 Rev: 0

CEA 22 has dropped into the core with the following resultant parameters on all 4 Core Protection Calculators:

ASI = – 0.1
DNBR = 2.05

The required power reduction has commenced and power is currently 99%. All other equipment is functioning as designed.

Based on these conditions, the crew will meet the minimum requirements of Tech Spec 3.2.4 when DNBR is ___(1)___ on ___(2)___ Core Protection Calculator

	(1)	(2)
A.	2.06	any operable
B.	2.06	each operable
C.	2.62	any operable
D.	2.62	each operable

Proposed Answer: A

Explanation (Optional):
This question requires the use of the correct graph for COLR 3.2.4. There are graphs for any CEAC operable and graphs for no CEAC operable. They must use the graph for any CEAC operable. For the second part, they must choose the option for any operable CPC. When using the COLSS power operating limits to meet DNBR, every channel must meet the required limit. For this example, a power reduction has commenced, which invalidates the COLSS power operating limits, requiring comparing the CPC DNBR values to the curves in COLR 3.2.4.

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Technical Reference(s)	OP-901-102 COLR for Tech Spec 3.4.2.	(Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: COLR for Tech Spec 3.2.4.

Learning Objective: WLP-OPS-PPO10 Obj. 5 (As available)

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content:	55.41	
	55.43	2

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	028 AA2.06	
	Importance Rating		2.8

K/A Statement
Ability to determine and interpret the following as they apply to the Pressurizer Level Control Malfunctions: Letdown flow indicator

Proposed Question: SRO 16 Rev: 0

Given the following:

- The plant is at 100% power
- All systems are in Automatic
- The ATC reports that letdown flow is 0 gpm
- The PZR level controller on CP-2 output is zero percent

Based on these indications, the ____(1)____ and the CRS should implement ____(2)____

	(1)	(2)
A.	in service Letdown flow control valve has failed closed	OP-901-112 Section E2, Letdown Malfunction
B.	in service Letdown flow control valve has failed closed	OP-901-110, Section E3, Pressurizer Level Controller Malfunction
C.	Pressurizer level controller has failed low	OP-901-112 Section E2, Letdown Malfunction
D.	Pressurizer level controller has failed low	OP-901-110, Section E3, Pressurizer Level Controller Malfunction

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Wrong failure, wrong procedure.
B. Incorrect. Wrong failure , correct procedure.
C. Incorrect. Correct failure, wrong procedure.
D. **CORRECT**. If the Pressurizer level controller output failed to 0 the letdown valve would completely close and give the stated indications. The listed procedure and section is correct for this malfunction.

Technical Reference(s) OP-901-110, Rev. 005 (Attach if not previously provided)
OP-901-112, Rev 3
SD-PLC, Rev. 8 (including version/revision number)
WLP-OPS-PLC00, Rev. 10

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPO10 Obj. 1 (As available)

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 _____
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	067 AA2.12	
	Importance Rating		3.9

K/A Statement

Ability to determine and interpret the following as they apply to the Plant Fire on Site: Location of vital equipment within fire zone

Proposed Question: SRO 17 Rev: 0

RAB-7C Fire Area covers the Isolation Panels located in the ____(1)____
If a fire were to develop in this area, OP-901-503, Isolation Panel Fire, would direct ____(2)____

	(1)	(2)
A.	RAB +35' Relay Room	tripping the reactor and initiating a Main Steam Isolation Signal (MSIS).
B.	RAB +35' Relay Room	Evacuating the Control Room and entering OP-901-502, Evacuation of Control Room and Subsequent Plant Shutdown
C.	RAB +35' Cable Vault	tripping the reactor and initiating a Main Steam Isolation Signal (MSIS).
D.	RAB +35' Cable Vault	Evacuating the Control Room and entering OP-901-502, Evacuation of Control Room and Subsequent Plant Shutdown

Proposed Answer: A

Explanation (Optional):

A. **CORRECT**. The Isolation Panels are located in the Relay Room and OP-901-503 mitigating actions include tripping the reactor and initiating a MSIS
B. Incorrect. Location is correct but evacuating the Control Room is not appropriate for an isolation panel fire.
C. Incorrect. Wrong location. Correct action.
D. Incorrect. Evacuation of the Control Room would be appropriate for a fire in the +35 Cable Vault Area, but the isolation panels are located in the +35 Relay Room.

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Technical Reference(s) OP-901-503, Rev. 303 (Attach if not previously provided)
OP-901-502, Rev. 018
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPO50 Obj. 3 (As available)

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 _____
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	G2.1.36	
	Importance Rating		4.1

K/A Statement

Conduct of Operations: Knowledge of procedures and limitations involved in core alterations.

Proposed Question: SRO 18 Rev: 0

Given the following:

- Fuel movement is in progress
- An irradiated fuel assembly has just been ungrappled in the Containment Upender and the Refueling Machine is in transit to the Reactor Vessel
- Reactor Cavity level begins lowering uncontrolled at 2 ft/min
- Personnel performing S/G tube inspections report a large amount of water gushing from S/G 2 hot leg manway

The Control Room Supervisor and Fuel Handling Supervisor should implement OP-010-006, Outage Operations, Attachment 9.23, Loss of Refuel Cavity Water Level Guidelines, and direct the refueling equipment operators to close the Transfer Tube Gate Valve and _____.

A. lower the Containment Upender to the horizontal position

B. transfer the fuel assembly from the Containment Upender to the Reactor Vessel

C. transfer the fuel assembly from the Containment Upender to one of the Temporary Fuel Storage Racks

D. transfer the fuel assembly from the Containment Upender to the Refueling Machine and lower the fuel assembly in the Refuel Cavity deep end

Proposed Answer: A

Explanation (Optional):

A. CORRECT. Attachment 9.23 of OP-010-006 The procedure states that the reactor vessel is the preferred location but states final decision is up to the Fuel Handling Supervisor. In this case the fuel assembly would have more shielding in its present location with the Upender lowered and it poses an unnecessary risk to the Refueling Machine operators to take the time to move back to the upender, lower the mast, grapple, raise the mast, move the assembly to a core location and then lower the assembly into the reactor while level is lowering at 2 ft/min.

B. Incorrect. See explanation A for why it is wrong to go the reactor vessel.

C. Incorrect. This location is listed in the suggested places to place the fuel assembly but it would not provide as much shielding for the assembly

D. Incorrect. This location is listed in the suggested places to place the fuel assembly but it would not provide as much shielding for the assembly

Technical Reference(s) OP-010-006, Att. 9.23, Rev. 310 (Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-REQ04 Obj. 4 & 5 (As available)

Question Source: Bank # _____
Modified Bank # WF3-NRC-5911-B (Note changes or attach parent)
New _____

Question History: Last NRC Exam 2004 W3 SRO Exam

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

10 CFR Part 55 Content: 55.41 _____
55.43 6

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	G2.1.37	
	Importance Rating		4.6

K/A Statement

Conduct of Operations: Knowledge of procedures, guidelines, or limitations associated with reactivity management.

Proposed Question: SRO 19 Rev: 0

Initial Conditions:

- The plant had been at 80% power for the past 5 days
- ASI control is in progress with Group P and Reg Group 6 partially inserted

Current Conditions:

- Cabinet H, Annunciator H-9, Pre-power Dependent Insertion Limit has actuated and been verified valid
- Linear Power by CP-7 indications have risen by 1% over the last 30 minutes without operator action
- Tavg and Tavg-Tref Deviation have risen 1 degree
- Group P and Reg Group 6 have not moved

Based on indications which of the following describes the status of Shutdown Margin (SDM) and states the procedure to implement?

A. SDM has lowered but still meets TS requirements; OP-901-103, Emergency Boration.

B. SDM has lowered but still meets TS requirements; OP-901-104, Inadvertent Positive Reactivity Addition.

C. SDM is below TS requirements; OP-901-103, Emergency Boration.

D. SDM is below TS requirements; OP-901-104, Inadvertent Positive Reactivity Addition.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. The correct procedure to implement is OP-901-104. OP-901-104 has a step that verifies SDM and then if it is not met it is required to implement OP-901-103. Based on information in the stem it can be determined that SDM is not being violated because the Transient Insertion Limits of the COLR are based on meeting SDM. The pre-power dependent alarm comes in prior to going below those limits.
B. **CORRECT**. Based on information in the stem it can be determined that SDM is not being violated because the Transient Insertion Limits of the COLR are based on meeting SDM. The pre-power dependent alarm comes in prior to going below those limits. The correct procedure to implement is OP-901-104.
C. Incorrect. Based on information in the stem it can be determined that SDM is not being violated because the Transient Insertion Limits of the COLR are based on meeting SDM. Wrong procedure per ARP for the given alarm.
D. Incorrect. Based on information in the stem it can be determined that SDM is not being violated because the Transient Insertion Limits of the COLR are based on meeting SDM. Correct procedure per ARP for the given alarm.

Technical Reference(s): OP-500-008, Control Room Cabinet H, Rev.026 (Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPO10 Obj. 1 (As available)

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 _____
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		2
	K/A #	G2.2.14	
	Importance Rating		4.3

K/A Statement

Equipment Control: Knowledge of the process for controlling equipment configuration or status.

Proposed Question: SRO 20 Rev: 0

The plant is at 100%. The rubber seal has been reported torn on Door 153, a CVAS boundary door, and the door has been declared inoperable. Door 161, the other door in that airlock, is operable. Which of the following is true concerning requirements to establish CVAS Boundary operability?

A. Perform OP-903-124, CVAS Pressure Boundary Testing, with Door 153 held open and Door 161 closed.

B. Perform OP-903-124, CVAS Pressure Boundary Testing, both Door 153 and Door 161 closed.

C. Perform OP-903-124, CVAS Pressure Boundary Testing, with Door 153 closed and Door 161 held open.

D. No testing is required; only one door is required to be operable in each CVAS air lock.

Proposed Answer: A

Explanation (Optional):

A. **CORRECT**. Per OP-100-014 this is required guidance for an inoperable air lock door.
B. Incorrect. This selection would interfere with ensuring the operable door provides sufficient isolation to ensure the CVAS fans can perform their function.
C. Incorrect. This configuration tests the inoperable door.
D. Incorrect. This plausible if the candidate assumes that no requirement exists to verify the acceptability of the remaining door.

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SRO Written Exam Worksheet

Technical Reference(s):	OP-100-014, Rev. 306, Pg 48	(Attach if not provided)
	TS 3.7.4, Amendment 170	(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPA00 Obj. 3 (As available)

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content:	55.41	
	55.43	2

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		2
	K/A #	G2.2.39	
	Importance Rating		4.5

K/A Statement

Equipment Control: Knowledge of less than or equal to one hour Technical Specification action statements for systems.

Proposed Question: SRO 21 Rev: 0

Given the following:

- On 10/13 the plant was operating at 100% power
- At 1600 Engineering determined that both CVAS trains are inoperable and the Control Room entered TS 3.0.3
- The Control Room staff started a shutdown at 1700
- The reactor trip breakers were opened at 2000

Which of the following describe the time that the crew must complete all of the required actions of Tech Spec 3.0.3?

A. 0200 on 10/14

B. 0500 on 10/14

C. 0200 on 10/15

D. 0500 on 10/15

Proposed Answer: D

Explanation (Optional):

A. Incorrect. No penalty is incurred for reaching HOT STANDBY in less than 6 hours and the plant must be brought to Cold Shutdown.
B. Incorrect. The plant must be brought to Cold Shutdown. The time would be correct if Hot shutdown was correct.
C. Incorrect. No penalty is incurred for reaching HOT STANDBY in less than 6 hours and the plant must be brought to Cold Shutdown.
D. **CORRECT**. The plant must be brought to Cold Shutdown. 3.0.3 allows 37 hours to get to Cold Shutdown from 100% power. No penalty is incurred for reaching MODE 3 early

Technical Reference(s): TS 3.0.3, Amendment 219 and Bases Amendment 99 (Attach if not previously provided)
TS 3.7.7, Amendment 170 (including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-TS02 Obj. 6 (As available)

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 _____
55.43 2

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		3
	K/A #	G2.3.6	
	Importance Rating		3.8

K/A Statement

Radiation Control: Ability to approve release permits.

Proposed Question: SRO 22 Rev: 0

The following plant conditions exist:

- Waste Condensate Tank 'B' is being discharged to the Circ Water system
- EFFLUENT RAD MONT SYS ACT HI-HI annunciator is alarm on CP-36
- WASTE LIQUID RAD MONITOR TROUBLE annunciator is in alarm on CP-4
- LWM-IFRR-0647, Liquid Waste Flow Recorder, indicates rising activity level
- The automatic isolations did not automatically close as designed
- The ATC operator has closed LWM-441 and LWM-442.

Based on these conditions, for Waste Condensate Tank B to be approved for discharge before the Liquid Waste Radiation Monitor is repaired, which of the following is a required action per TRM 3.3.3.10, Radioactive Liquid Effluent?

A. Complete a valve lineup to ensure that Waste Condensate Tank A is isolated from the discharge.

B. Initiate a Department Action Statement Notice requiring samples every 4 hours during discharge.

C. Initiate a Technical Specification Addendum Log to calculate release flow rate every 4 hours during the discharge.

D. Ensure release rate calculations have been verified by two technically qualified personnel prior to the discharge.

Proposed Answer: D

Explanation (Optional):

A. Incorrect: Independent valve lineups of the discharge path of the tank to be released must be performed.
B. Incorrect: Independent analyzed samples are required prior to the release of the tank.
C. Incorrect: This is a requirement for an inoperable flow instrument.
D. **CORRECT**: This is a requirement of TRM 3.3.3.10 for an inoperable LWM Rad Monitor.

Technical Reference(s) TRM 3.3.3.10 (Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-LWM00 Obj: 8 (As available)

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

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 55.41 ____
55.43 2

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		3
	K/A #	G2.4.18	
	Importance Rating		4.0

K/A Statement
Emergency Procedures / Plan: Knowledge of the specific bases for EOPs.

Proposed Question: SRO 23 Rev: 0

The following plant conditions exist:

- A Main Steam Line Break has occurred on SG #2
- The reactor tripped on Low Steam Generator Pressure
- SG #2 has just blown dry and the CRS has ordered performance of the necessary steps of OP-902-004 to stabilize RCS temperature and pressure

Which ONE of the following states the reason for feeding the LEAST affected SG with EFW manually?

A. To restore subcooled margin to with the P-T Curve limits

B. To reduce the positive reactive addition from the rapid cooldown

C. To provide additional cooling since the ADV alone will not provide sufficient cooling

D. To override the Main Steam Isolation Signal that blocks the automatic Emergency Feedwater signal.

Proposed Answer: C

Explanation (Optional):.

A. Incorrect: By maintaining adequate heat removal will ensure proper cooling
B. Incorrect: By maintaining adequate heat removal will not prevent a rapid cooldown.
C. **CORRECT**: Maintain level to provide adequate heat removal capability under forced or natural circulation conditions.
D. Incorrect: By maintaining adequate heat removal will not override a MSIS signal or block automatic operation of EFW signal

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Technical Reference(s) OP-902-004 Step 4.16, Rev. 011 (Attach if not previously provided)
TGOP-902-004 Step 16, Rev. 302
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: WLP-OPS-PPE04 Obj: 4 (As available)

Question Source: Bank # WF3-OPS-6794-A
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam 2009 W3 SRO Exam

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

10 CFR Part 55 Content: 55.41 _____
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	068 G2.4.30	
	Importance Rating		4.1

K/A Statement

Control Room Evacuation: Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.

Proposed Question: SRO 24 Rev: 0

Implementation of which of the following would require notification to the NRC within one hour? (assume all other equipment operable).

A. OP-902-001, Reactor Trip Recovery due to a Turbine trip with Reactor Power Cutback OOS.

B. OP-902-006, Loss of Main Feedwater Recovery due to Main Feedwater Pumps tripping.

C. OP-901-502, Evacuation of Control Room and Subsequent Plant Shutdown, due to a fire in CP-18.

D. OP-901-504, Inadvertent ESFAS Actuation due to CSAS initiated during OP-903-107, PPS A(B)(C)(D) Functional Test.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. 4 hour report.
B. Incorrect. 4 hour report.
C. **CORRECT**. Evacuation of Control Room by itself meets threshold criteria for E-Plan entry and classification.
D. Incorrect. 8 hour report if considered valid or defect in system

2010 NRC Exam
SRO Written Exam Worksheet

Technical Reference(s)	EP-001-001, Recognition and Classification of Emergency Conditions, Rev. 025	(Attach if not previously provided)
	UNT-006-010, Event Notification and Reporting, Rev. 302	(including version/revision number)

Proposed references to be provided to applicants during examination: UNT-006-010, EP-001-001

Learning Objective: WLP-PPO51 obj.3 (As available)

Question Source:

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

Question History: Last NRC Exam N/A

Question Cognitive Level:

Memory or Fundamental Knowledge	
Comprehension or Analysis	X

10 CFR Part 55 Content:

55.41	
55.43	5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #	G2.4.44	
	Importance Rating		4.4

K/A Statement
Emergency Procedures / Plan: Knowledge of emergency plan protective action recommendations

Proposed Question: SRO 25 Rev: 0

The following plant conditions exist:

- A radiological release is in progress following a LOCA outside of Containment
- The Emergency Coordinator is evaluating E Plan
- Data is now available for a PARs determination
- The EC directs you to perform this determination per EP-002-052, Protective Action Guidelines.

The following data is available:

- Duration of release is unknown
- Wind Directions is 345 º

EAB TEDE Mr/hr	EAB CDE Thyroid Mr/hr	2 mile TEDE Mr/hr	2 mile CDE Thyroid Mr/hr	5 mile TEDE Mr/hr	5 mile CDE Thyroid Mr/hr
1250	3200	750	1900	50	730

Which ONE of the following PAR actions meets the required recommendations of EP-002-052, Protective Action Guidelines?

<u>EVACUATE</u>

A. A1, B1, C1, D1

B. A1, B1, C1, D1, D2

C. A1, B1, C1, D1, B2, D2

D. A1, B1, C1, D1, D2, D3, D4

Proposed Answer: B

Explanation (Optional):

A. INCORRECT: Student would arrive at this answer if he doesn't multiply the dose by 2 hours as required, since duration length is NOT known.
B. CORRECT: Double rate and use response areas from sector R
C. INCORRECT: Correct dose but uses sector Q vice R.
D. INCORRECT: Uses correct sector, but evacuates all 3 response areas.

Technical Reference(s)	EP-002-052 pgs 13 - 14 (Att 7.2 & 7.3) EP-002-050, page 7	(Attach if not previously provided) (including version/revision number)

Proposed references to be provided to applicants during examination: EP-002-052 Att 7.2 & 7.3

Learning Objective: WLP-OPS-EP02 Obj: 24 (As available)

Question Source:	Bank #	X	SRO #23
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam 2009 SRO Exam

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

10 CFR Part 55 Content:	55.41	
	55.43	4

Comments:

REQUEST/APPROVAL PAGE

SAFETY RELATED
PROCEDURE

Normal Review Class (check one):
☒ **OSRC**
☐ **QUALIFIED REVIEW**

PROCEDURE NUMBER: EP-001-001		REVISION: 025

TITLE: Recognition and Classification of Emergency Conditions

PROCEDURE OWNER (Position Title): Emergency Planning Manager

TERM (check one): ☒ **PERMANENT** ☐ **TEMPORARY**

Effective Date / Milestone (if applicable): 03/31/10

Expiration Date / Milestone (if applicable): N/A

PROCEDURE ACTION (check one):
☒ Revision ☐ Deletion ☐ New Procedure

DESCRIPTION AND JUSTIFICATION:

(1) Revised security event related EALs to incorporate NEI 99-01 Revision 5 security event related EALs in their entirety to support Security Department implementation of NEI 03-12 Revision 6 in accordance with NRC generated EAL FAQ 2009-48.
(2) Changed the title of "Duty Plant Manager" to "TSC duty Emergency Coordinator" in sections 3.3 and 3.4 to correspond to a title change in the Emergency Plan.

☐ Request/Approval Page Continuation Sheet(s) attached.

REVIEW PROCESS (check one): ☒ Normal ☐ Editorial Correction (Revisions only) ☐ Technical Verification (Revisions only)

REVIEW AND APPROVAL ACTIVITIES				PRINT NAME OR SIGNATURE	DATE
PREPARER				J.J. Lewis	03/08/10
EC SUPERVISOR	Administrative Review and Approval			(sign) N/A	
CROSS-DISCIPLINE and INTERNAL REVIEWS (List Groups, Functions, Positions, etc.)				N/A	
				N/A	
				N/A	
				N/A	
				N/A	
PROCESS APPLICABILITY DETERMINATION	Performed ☐	PA Exclusion ☒		OSRC meeting 03-003	
TECHNICAL	Review ☒	Verification ☐		R.J. Perry	3/17/10
QUALIFIED REVIEWER	Review ☐			N/A	
GROUP/DEPT. HEAD	Review ☒	Approval ☐		(sign) G.L. Fey/	3/18/10
GM, PLANT OPERATIONS	Review ☐	Approval ☒		(sign)	3/30/10
VICE PRESIDENT, OPERATIONS		Approval ☐		(sign) N/A	

TABLE OF CONTENTS

LIST OF EFFECTIVE PAGES

1-144 Revision 025

Reference Use

1.0 PURPOSE

1.1 This procedure describes the immediate actions to be taken to recognize and classify the four emergency classifications: Unusual Event, Alert, Site Area Emergency, and General Emergency.

2.0 REFERENCES

2.1 Waterford 3 SES Emergency Plan

2.2 Title 10, Code of Federal Regulations Part 50, Appendix E

2.3 NEI 99-01 Methodology for Development of Emergency Action Levels

2.4 Waterford 3 SES Final Safety Analysis Report

2.5 EP-001-010, Unusual Event

2.6 EP-001-020, Alert

2.7 EP-001-030, Site Area Emergency

2.8 EP-001-040, General Emergency

2.9 EP-002-015, Emergency Responder Activation

2.10 EP-002-052, Protective Action Guidelines

2.11 EP-002-150, Emergency Plan Implementing Records

2.12 EP-004-010, Toxic Chemical Contingency Procedure

2.13 NUREG-1022, Event Reporting Guidelines: 10CFR50.72 and 50.73

2.14 UNT-006-010, Event Notification and Reporting

2.15 HP-CALC-2001-001, PASS System Elimination (Dose Rates Calculation)

2.16 HP-CALC-2005-002, Emergency Action Levels (EALs) (Abnormal Rad Levels and Radiological Effluent) Based on Power Uprate Source Terms

2.17 NRC Bulletin 2005-02, Emergency Preparedness and Response Actions for Security-Based Events

3.0 RESPONSIBILITIES

3.1 The Emergency Coordinator (EC) is responsible for implementation of this procedure

3.2 The Emergency Coordinator (EC) is responsible for declaration of the appropriate emergency classification whenever, in his judgment, the station status warrants such a declaration.

3.3 The Shift Manager shall assume the responsibility and authority of the Emergency Coordinator (EC) until such time that he is properly relieved of this duty by the TSC Duty Emergency Coordinator.

3.4 If the Shift Manager cannot immediately assume the duty of Emergency Coordinator, then the Control Room Supervisor (CRS) shall assume the duty of Emergency Coordinator until properly relieved by the TSC Duty Emergency Coordinator or Shift Manager.

3.5 When the EOF is activated and responsibilities are transferred, then the EOF Director is responsible for implementation of this procedure and declaration of the appropriate emergency classification whenever, in his judgment, the station status warrants such a declaration.

4.0 INITIATING CONDITIONS

4.1 An off-normal event has occurred or is in progress.

NOTE

This instruction does not replace any plant operating procedure. Ensure that any immediate actions (for example, use of Emergency Procedures) are taken for the proper operation of the plant. During an emergency condition, continue to use the appropriate plant procedures in parallel with this instruction.

4.2 An action step in a plant procedure refers to this instruction for classification of the indicated plant conditions.

5.0 PROCEDURE

5.1 Definitions

5.1.1 Emergency Class - One of a minimum set of names or titles, established by the Nuclear Regulatory Commission (NRC), for grouping off-normal nuclear power plant conditions according to (1) their relative radiological seriousness, and (2) the time-sensitive onsite and off-site radiological emergency preparedness actions necessary to respond to such conditions. The existing radiological emergency classes, in ascending order of seriousness, are called: Notification of Unusual Event (Unsual Event), Alert, Site Area Emergency, and General Emergency.

5.1.2 Initiating Condition (IC) - One of a predetermined subset of nuclear power plant conditions where either the potential exists for a radiological emergency, or such an emergency has occurred.

5.1.3 Emergency Action Level (EAL) - A pre-determined, site-specific, observable threshold for a plant Initiating Condition that places the plant in a given emergency class. An EAL can be: an instrument reading; an equipment status indicator; a measurable parameter (onsite or offsite); a discrete, observable event; results of analyses; entry into specific emergency operating procedures; or another phenomenon which, if it occurs, indicates entry into a particular emergency class.

5.2 Classification

5.2.1 Verify the off-normal event to ensure that the event is real.

5.2.2 Match the off-normal event with one of the following five emergency categories:

5.2.2.1	Abnormal Radiation Levels/Radiological Effluents	TAB A
5.2.2.2	Cold Shutdown/Refueling System Malfunction	TAB C
5.2.2.3	Fission Product Barrier Degradation	TAB F
5.2.2.4	Hazards and Other Conditions Affecting Plant Safety	TAB H
5.2.2.5	System Malfunction	TAB S

5.2.3 Refer to Attachment 7.1, Emergency Categories, under the category TAB selected in step 5.2.2 above, match the off-normal condition with the appropriate IC to determine the emergency classification.

5.2.4 If an event or condition existed which met or exceeded an IC but no emergency was declared and the basis for the emergency classification no longer exists at the time of the discovery (rapidly concluded event, missed classification or misclassified event), then do not classify the emergency or make offsite notifications.

5.2.4.1 Notify the NRC within one hour of the discovery of the undeclared or misclassified event in accordance with UNT-006-010.

NOTE

The effects of combinations of initiating conditions that individually constitute a lower classification may be considered as a possibly higher emergency classification.

5.2.5 Declare the highest emergency classification for which an IC has been met or exceeded.

5.2.6 Perform the emergency actions in accordance with the appropriate Emergency Plan Implementing Instruction, one of which is provided for each classification, as follows:

5.2.6.1 Unusual Event - EP-001-010

5.2.6.2 Alert - EP-001-020

5.2.6.3 Site Area Emergency - EP-001-030

5.2.6.4 General Emergency - EP-001-040

5.2.7 Assessment actions shall be continued, and if necessary, the emergency classification escalated (or downgraded) as more definitive information becomes available or if the plant conditions change.

6.0 FINAL CONDITIONS

6.1 The plant conditions which activated this instruction have been declassified to non-emergency status.

7.0 ATTACHMENTS

7.1 Emergency Categories

Index of Initiating Conditions

TAB A Abnormal Radiation Levels/Radiological Effluents

TAB C Cold Shutdown/Refueling System Malfunction

TAB F Fission Product Barrier Degradation

TAB H Hazards and Other Conditions Affecting Plant Safety

TAB S System Malfunction

7.2 Waterford 3 EAL Basis Document

8.0 RECORDS

None

INDEX OF INITIATING CONDITIONS

TAB A ABNORMAL RADIATION LEVELS/RADIOLOGICAL EFFLUENTS

1. Unplanned releases of gaseous or liquid radioactivity to the environment
2. Unexpected rise in plant radiation/damage to irradiated fuel/loss of water level
3. Release of radioactive material or rise in radiation levels within the facility that impedes operation of systems required to maintain safe operation

TAB C COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

1. RCS Leakage/loss of reactor vessel inventory
2. Loss of RCS inventory/loss of reactor vessel inventory
3. Loss of decay heat removal capability with irradiated fuel in the reactor vessel
4. Loss of offsite/onsite AC power
5. Unplanned loss of required DC power
6. Inadvertent criticality
7. Loss of onsite or offsite communications

TAB F FISSION PRODUCT BARRIER DEGRADATION

1. Loss of Containment, RCS, or Fuel Clad barrier(s)

TAB H HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

1. Security
2. Judgment
3. Control Room evacuation
4. Fire or explosion
5. Toxic or flammable gases
6. Natural and destructive phenomena

TAB S SYSTEM MALFUNCTION

1. Loss of offsite/onsite AC power (No IC number 2)
3. Failure of Reactor Protection System
4. Loss of Vital DC power
5. Complete loss of heat removal capability
6. Loss of safety system annunciators/indicators
7. RCS Leakage
8. Loss of onsite or offsite communications
9. Fuel clad degradation
10. Inadvertent criticality
11. Inability to reach required shutdown within Tech Spec limits

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

ABNORMAL RADIATION LEVELS/RADIOLOGICAL EFFLUENTS

Radiological Effluents

AG1 Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 1000 mR TEDE or 5000 mR CDE Thyroid for the actual or projected duration of the release using actual meteorology.

1 2 3 4 5 6 D

Emergency Action Level(s): (1 or 2 or 3)

NOTE: *If dose assessment results are available at the time of declaration, then the classification should be based on EAL #2 instead of EAL #1. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to more accurately characterize the nature of the release.*

1. VALID reading on one or more of the following radiation monitors that exceeds or is expected to exceed the reading shown for ≥ 15 minutes:
   - CONDENSER EXHAUST WRGM (PRM-IRE-0002, RE0002-4) indicates release rate > 2.69E+09 uCi/sec
   - FUEL HANDLING BUILDING EXHAUST WRGM (PRM-IRE-3032, RE3032-4) indicates release rate > 1.75E+09 uCi/sec
   - PLANT STACK WRGM (PRM-IRE-0110, RE0110-4) indicates release rate > 2.55E+09 uCi/sec

OR

2. Dose assessment using actual meteorology indicates doses > 1000 mR TEDE or > 5000 mR CDE Thyroid at or beyond the EAB.

OR

3. Field survey results indicate closed window dose rates >1000 mR/hr expected to continue for > one hour; or analyses of field survey samples indicate CDE Thyroid ≥ 5000 mR for one hour of inhalation, at or beyond the EAB.

AS1 Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mR TEDE or 500 mR CDE Thyroid for the actual or projected duration of the release.

1 2 3 4 5 6 D

Emergency Action Level(s): (1 or 2 or 3)

NOTE: *If dose assessment results are available at the time of declaration, then the classification should be based on EAL #2 instead of EAL #1. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated.*

1. VALID reading on one or more of the following radiation monitors that exceeds or is expected to exceed the reading shown for ≥ 15 minutes:
   - CONDENSER EXHAUST WRGM (PRM-IRE-0002, RE0002-4) indicates release rate > 2.69E+08 uCi/sec
   - FUEL HANDLING BUILDING EXHAUST WRGM (PRM-IRE-3032, RE3032-4) indicates release rate > 1.75E+08 uCi/sec
   - PLANT STACK WRGM (PRM-IRE-0110, RE0110-4) indicates release rate > 2.55E+08 uCi/sec

OR

2. Dose assessment using actual meteorology indicates doses > 100 mR TEDE or > 500 mR CDE Thyroid at or beyond the EAB.

OR

3. Field survey results indicate closed window dose rates >100 mR/hr expected to continue for > one hour; or analyses of field survey samples indicate CDE Thyroid ≥ 500 mR for one hour of inhalation, at or beyond the EAB.

AA1 Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds 200 times the radiological effluent ODCM limits for ≥ 15 minutes.

1 2 3 4 5 6 D

Emergency Action Level(s): (1 or 2 or 3)

1. VALID reading on any effluent monitor that exceeds 200 times the alarm setpoint established by a current radioactivity discharge permit for ≥ 15 minutes.

OR

2. VALID reading on one or more of the following radiation monitors that exceeds the reading shown for ≥ 15 minutes:

MONITOR	CONC.	EFFLUENT RATE
CONDENSER EXHAUST WRGM PRM-IRE-0002, RE0002-4		1.51E+07 uCi/sec
FUEL HANDLING BUILDING EXHAUST WRGM, PRM-IRE-3032, RE3032-4		2.25E+07 uCi/sec
PLANT STACK WRGM PRM-IRE-0110, RE0110-4		1.51E+07 uCi/sec

OR

3. Confirmed grab sample analyses for gaseous or liquid releases indicates concentrations or release rates, with a release duration of ≥ 15 minutes, in excess of 200 times ODCM based limits from the Technical Requirements Manual (TRM) (Table A1).

Table A1
TRM Limits

	ALERT	UE
Gaseous Release		
Noble Gases: ≤ 500 mrem/yr whole body	1.00E+05	1000
Noble Gases: ≤ 3000 mrem/yr skin	6.00E+05	6000
I-131, I-133, H-3 and particulates with half-lives > 8 days: ≤ 1500 mrem/year to any organ	3.00E+05	3000
Liquid Release		
Whole body: < 1.50 mrem/quarter	300	3
< 3 mrem/yr	600	6
Any Organ: < 5 mrem/quarter	1000	10
< 10 mrem/yr	2000	20

AU1 Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds 2 times the radiological effluent ODCM limits for ≥ 60 minutes.

1 2 3 4 5 6 D

Emergency Action Level(s): (1 or 2 or 3)

1. VALID reading on any effluent monitor that exceeds 2 times the alarm setpoint established by a current radioactivity discharge permit for ≥ 60 minutes.

OR

2. VALID reading on one or more of the following radiation monitors that exceeds the reading shown for ≥ 60 minutes:

MONITOR	CONC.	EFFLUENT RATE
CONDENSER EXHAUST WRGM PRM-IRE-0002, RE0002-4		1.51E+05 uCi/sec
FUEL HANDLING BUILDING EXHAUST PIG, GAS CHANNEL , PRM-IRE-5107A or B, RE5107A-1 or RE5107B-1	1.61E-02 uCi/cc	
FUEL HANDLING BUILDING EXHAUST WRGM, PRM-IRE-3032, RE3032-4		2.25E+05 uCi/sec
PLANT STACK PIG GAS CHANNEL PRM-IRE-0100.1S or 2S, RE0100.1-1 or RE0100.2-1	3.45E-03 uCi/cc	
PLANT STACK WRGM PRM-IRE-0110, RE0110-4		1.51E+05 uCi/sec
[1]DRY COOLING TOWER SUMPS MONITOR, PRM-IRE-6775 or PRM-IRE-6776, RE6775-1 or RE6776-1	8.49E-04 uCi/ml	
[1]TURBINE BUILDING INDUSTRIAL WASTE SUMP MONITOR, PRM-IRE-6778, RE6778-1	8.49E-04 uCi/ml	

[1]Monitor reading not applicable if sump discharge is aligned to circulating water discharge.

OR

3. Confirmed grab sample analyses for gaseous or liquid releases indicates concentrations or release rates, with a release duration of ≥ 60 minutes, in excess of 2 times ODCM based limits from the Technical Requirements Manual (TRM) (Table A1).

Plant Modes (white boxes indicate applicable modes) 1 **Power Operations** 2 **Startup** 3 **Hot Standby** 4 **Hot Shutdown** 5 **Cold Shutdown** 6 **Refueling** D **Defueled**

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

ABNORMAL RADIATION LEVELS/RADIOLOGICAL EFFLUENTS

Abnormal Radiation Levels

AA2 Damage to irradiated fuel or loss of water level that has or will result in uncovering of irradiated fuel outside the reactor vessel. 1 2 3 4 5 6 D

Emergency Action Level(s): (1 or 2)

1. VALID alarm or reading ≥ HIGH alarm limits on one or more of the following radiation monitors:
   - CONTAINMENT AREA RADIATION MONITORS (PURGE ISOLATION), (ARM-IRE-5024S, 5025S, 5026S OR 5027S, RE5024-1, RE5025-1, RE5026-1 OR RE5027-1) ≥ HIGH alarm
   - CONTAINMENT +46 STAIRS MONITORS, (ARM-IRE-5014 OR 5015, RE5014-1 OR RE5015-1) ≥ HIGH alarm
   - REFUELING BRIDGE AREA RADIATION MONITOR (ARM-IRE-5013, RE5013-1) ≥ HIGH alarm
   - FHB AREA RADIATION MONITORS (ISOLATION), (ARM-IRE-0300.1S, .2S, .3S OR .4S, RE0300.1-1, RE0300.2-1, RE0300.3-1, OR RE0300.4-1) ≥ 1000 mR/hr
   - FUEL HANDLING BUILDING EXHAUST PIG, GAS CHANNEL , PRM-IRE-5107A OR B, RE5107A-1 OR RE5107B-1 ≥ HIGH alarm

OR

2. Valid indication of uncontrolled water level drop in the reactor refueling cavity, spent fuel pool or fuel transfer canal that will result in irradiated fuel uncovering.

AU2 Unexpected rise in plant radiation. 1 2 3 4 5 6 D

Emergency Action Level(s): (1 or 2)

1. a. VALID indication of uncontrolled water level drop in the reactor refueling cavity, spent fuel pool, or fuel transfer canal with all irradiated fuel assemblies remaining covered by water.
   - Level drop may be indicated by personnel observation, spent fuel pool level below level plate, refueling crew report, indication on area security camera, RWSP level drop due to makeup demands.

AND

   b. Unplanned VALID Area Radiation Monitor rise on any of the following:
   - CONTAINMENT AREA RADIATION MONITORS (PURGE ISOLATION), (ARM-IRE-5024S, 5025S, 5026S OR 5027S, RE5024-1, RE5025-1, RE5026-1 OR RE5027-1)
   - CONTAINMENT +46 STAIRS MONITORS, (ARM-IRE-5014 OR 5015, RE5014-1 OR RE5015-1)
   - REFUELING BRIDGE AREA RADIATION MONITOR (ARM-IRE-5013, RE5013-1)
   - FHB AREA RADIATION MONITORS (ISOLATION), (ARM-IRE-0300.1S, .2S, .3S OR .4S, RE0300.1-1, RE0300.2-1, RE0300.3-1, OR RE0300.4-1)

OR

2. Unplanned VALID Area Radiation Monitor readings indicate a rise in plant radiation levels by a factor of 1000 over normal levels (highest reading in the past 24 hours excluding the current peak value).

Table A2
Radiation Levels in Areas Requiring Infrequent Access

VCT Room – 10 R/hr	Safeguards Rooms – 10 R/hr
VALUE FOR ALL AREAS BELOW IS 2.5 R/hr:	
+46 Chiller Area	BAM Tank Rooms
MSIV Areas	Relay Room
Electrical Penetration Area	Remote Shutdown Room
EDG Rooms	Battery Rooms
Valve Bay	Wing Areas
CVC-507 Valve Area	CCW Heat Exchanger Rooms
CCW Pump Rooms	

AA3 Release of radioactive material or rise in radiation levels within the facility that impedes operation of systems required to maintain safe operations or to establish or maintain cold shutdown. 1 2 3 4 5 6 D

Emergency Action Level(s): (1 or 2)

1. VALID radiation level > 15 mR/hr in areas requiring continuous occupancy to maintain plant safety functions.
   - Main Control Room Area Radiation Monitor (ARM-IRE- 5001, RE5001-1) > 15 mR/hr
   - Radiation level in CAS >15 mR/hr

OR

2. VALID radiation level > Table A2 value in plant vital areas requiring infrequent access to maintain plant safety functions (Table A2).

Plant Modes (white boxes indicate applicable modes) 1 **Power Operations** 2 **Startup** 3 **Hot Standby** 4 **Hot Shutdown** 5 **Cold Shutdown** 6 **Refueling** D **Defueled**

COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Cold Shutdown Loss of RCS Inventory	**CG1** Loss of reactor vessel inventory affecting fuel clad integrity with containment challenged with irradiated fuel in the reactor vessel. (5 6) **Emergency Action Level(s):** (1 and 2 and 3) 1. Loss of reactor vessel inventory as indicated by unexplained containment sump level or reactor drain tank level rise **AND** 2. Reactor vessel level cannot be monitored with indication of core uncovery > 30 minutes as evidenced by one or more of the following: • Containment High Range Radiation Monitor (ARM-IRE-5400AS or ARM-IRE-5400BS) ≥ 10R/hr • Erratic Source Range Monitor indication • Core Exit Thermocouples indicate superheat **AND** 3. Indication of CONTAINMENT challenged as indicated by one or more of the following: • Explosive mixture inside containment • Containment pressure > 50 PSIA • CONTAINMENT CLOSURE not established.	**CS1** Loss of reactor vessel inventory affecting core decay heat removal capability. (5) **Emergency Action Level(s):** (1 or 2) 1. With CONTAINMENT CLOSURE not established: a. Reactor vessel inventory as indicated by RVLMS upper plenum level 0% **OR** b. Reactor vessel level cannot be monitored > 30 minutes with a loss of reactor vessel inventory as indicated by unexplained containment sump level or reactor drain tank level rise **OR** 2. With CONTAINMENT CLOSURE established: Reactor vessel level cannot be monitored > 30 minutes with a loss of reactor vessel inventory as indicated by either: • Unexplained containment sump or reactor drain tank level rise. • Erratic Source Range Monitor indication.	**CA1** Loss of RCS inventory. (5) **Emergency Action Level(s):** (1 or 2) 1. Loss of RCS inventory as indicated by RVLMS upper plenum level ≤ 20% **OR** 2. a. Loss of RCS inventory as indicated by unexplained containment sump level or reactor drain tank level rise **AND** b. RCS level cannot be monitored > 15 minutes.	**CU1** RCS leakage. (5) **Emergency Action Level(s):** (1 or 2) 1. Unidentified or pressure boundary leakage > 10 gpm. **OR** 2. Identified leakage > 25 gpm.
Refueling Loss of RCS Inventory		**CS2** Loss of reactor vessel inventory affecting core decay heat removal capability with irradiated fuel in the reactor vessel. (6) **Emergency Action Level(s):** 1. Reactor vessel level cannot be monitored WITH indication of core uncovery as evidenced by one or more of the following: • Containment High Range Radiation Monitor (ARM-IRE-5400AS or ARM-IRE-5400BS) ≥ 10R/hr • Erratic Source Range Monitor indication • Core Exit Thermocouples indicate superheat.	**CA2** Loss of reactor vessel inventory with irradiated fuel in the reactor vessel. (6) **Emergency Action Level(s):** (1 or 2) 1. Loss of reactor vessel inventory as indicated by reactor vessel level at 12 ft. **OR** 2. a. Loss of reactor vessel inventory as indicated by unexplained containment sump level or reactor drain tank level rise **AND** b. Reactor vessel level cannot be monitored > 15 minutes.	**CU2** UNPLANNED loss of RCS inventory with irradiated fuel in the reactor vessel. (6) **Emergency Action Level(s):** (1 or 2) 1. UNPLANNED RCS level drop below the vessel flange for ≥ 15 minutes **OR** 2. a. Loss of reactor vessel inventory as indicated by unexplained containment sump level or reactor drain tank level rise **AND** b. Reactor vessel level cannot be monitored.

Plant Modes (white boxes indicate applicable modes) **1 Power Operations** **2 Startup** **3 Hot Standby** **4 Hot Shutdown** **5 Cold Shutdown** **6 Refueling** **D Defueled**

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION			
Loss of Decay Heat Removal		**CA3** Inability to maintain plant in Cold Shutdown with irradiated fuel in the reactor vessel. 5 6 **Emergency Action Level(s):** (1 or 2 or 3) 1. With CONTAINMENT CLOSURE and RCS integrity not established, an UNPLANNED event results in RCS temperature exceeding the Technical Specification cold shutdown temperature limit **OR** 2. With CONTAINMENT CLOSURE established and RCS integrity not established, or RCS inventory reduced, an UNPLANNED event results in RCS temperature exceeding the Technical Specification cold shutdown temperature limit for > 20 minutes[1] **OR** 3. An UNPLANNED event results in RCS temperature exceeding the Technical Specification cold shutdown temperature limit for > 60 minutes[1] or results in an RCS pressure rise of > 10 psig. *[1]Note: **If** shutdown cooling is in operation within this time frame **and** RCS temperature is being reduced **then** this EAL is **not** applicable.*	**CU3** UNPLANNED loss of decay heat removal capability with irradiated fuel in the reactor vessel. 5 6 **Emergency Action Level(s):** (1 or 2) 1. An UNPLANNED event results in RCS temperature exceeding the Technical Specification cold shutdown temperature limit **OR** 2. Loss of all RCS temperature and reactor vessel level indication for > 15 minutes.

Plant Modes (white boxes indicate applicable modes) 1 **Power Operations** 2 **Startup** 3 **Hot Standby** 4 **Hot Shutdown** 5 **Cold Shutdown** 6 **Refueling** D **Defueled**

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION (cont.)				
Loss of Power			**CA5** Loss of all offsite power and loss of all onsite AC power to essential busses. (5 6 D) **Emergency Action Level(s):** 1. a. Loss of power to all unit auxiliary and startup transformers **AND** b. Failure of the 'A' and 'B' emergency diesel generators to supply power to emergency busses **AND** c. Failure to restore power to at least one emergency bus within 15 minutes from the time of loss of both offsite and onsite AC power.	**CU5** Loss of all offsite power to essential busses > 15 minutes. (5 6) **Emergency Action Level(s):** 1. a. Loss of power to all unit auxiliary and startup transformers > 15 minutes **AND** b. At least emergency diesel generator 'A" or 'B' is supplying power to emergency busses.
Loss of DC Power				**CU6** UNPLANNED loss of required DC power > 15 minutes. (5 6) **Emergency Action Level(s):** 1. a. UNPLANNED loss of vital DC power to required DC busses based on bus voltage indication < 108 volts **AND** b. Failure to restore power to at least one required DC bus within 15 minutes from the time of loss.
Inadvertent Criticality				**CU7** Inadvertent criticality. (5 6) **Emergency Action Level(s):** 1. An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

Plant Modes (white boxes indicate applicable modes) 1 **Power Operations** 2 **Startup** 3 **Hot Standby** 4 **Hot Shutdown** 5 **Cold Shutdown** 6 **Refueling** D **Defueled**

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION (cont.)

Loss of Communications

Table C1 Onsite Communications Equipment
Plant radio system
Plant paging system
In-plant telephones
Sound powered phones

Table C2 Offsite Communications Equipment
All telephone lines (commercial and microwave)
Industrial Hot Line
ENS
Civil Defense Radios
Operational Hotline

5 6

CU8 UNPLANNED loss of all onsite or offsite communications capabilities.

Emergency Action Level(s): (1 or 2)

1. Loss of all Table C1 onsite communications systems affecting the ability to perform routine operations

OR

2. Loss of all Table C2 offsite communications systems.

Plant Modes (white boxes indicate applicable modes) 1 **Power Operations** 2 **Startup** 3 **Hot Standby** 4 **Hot Shutdown** 5 **Cold Shutdown** 6 **Refueling** D **Defueled**

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
FISSION PRODUCT BARRIER DEGRADATION			
FG1 Loss of ANY two Barriers AND Loss or Potential Loss of Third barrier. 1 2 3 4	**FS1** Loss or Potential Loss of ANY two Barriers. 1 2 3 4	**FA1** ANY loss or ANY Potential Loss of EITHER Fuel Clad or RCS. 1 2 3 4	**FU1** ANY loss or ANY Potential Loss of Containment. 1 2 3 4

Plant Modes (white boxes indicate applicable modes) **1 Power Operations** **2 Startup** **3 Hot Standby** **4 Hot Shutdown** **5 Cold Shutdown** **6 Refueling** **D Defueled**

Note: Determine which combination of the three barriers are lost or have a potential loss and use the above key to classify the event. Also an event or multiple events could occur which result in the conclusion that exceeding the loss or potential loss thresholds is imminent (i.e., within 1 to 2 hours). In this imminent loss situation use judgment and classify as if the thresholds are exceeded.

SEE FOLLOWING PAGE FOR EALS FOR BARRIER LOSS AND POTENTIAL LOSS

Fuel Clad Barrier EALs

LOSS	POTENTIAL LOSS
1. Primary Coolant Activity Level (FCB1)	
RCS Dose Equivalent Iodine > 300 µCi/gm as indicated by: a. Dose Rate at one foot from Primary Sample Panel > 950 mR/hr **OR** b. -4 RAB RADIOCHEMISTRY LAB area radiation monitor (ARM-IRE-5020) > 125 mR/hr **OR** c. Chemistry sample results	Not Applicable
2. Core Exit Thermocouple Readings (FCB2)	
≥ 1200 degrees F	≥ 700 degrees F
3. Reactor Vessel Water Level (FCB3)	
Not applicable.	RVLMS upper plenum level 0%.
4. Containment Radiation Monitoring (FCB4)	
Containment High Range Radiation Monitor (ARM-IRE-5400AS or ARM-IRE-5400BS) > 1000 R/hr.	Not Applicable
5. Emergency Coordinator/EOF Director Judgment (FCB5)	
Any condition in the opinion of the Emergency Coordinator/EOF Director that indicates Loss or Potential Loss of the Fuel Clad barrier.	

RCS Barrier EALs

LOSS	POTENTIAL LOSS
1. RCS Leak Rate (RCB1)	
GREATER THAN available makeup capacity as indicated by RCS subcooling < 28º F.	Unisolable RCS leak > 44 gpm.
2. SG Tube Rupture (RCB2)	
SGTR that results in an ECCS (SI) actuation.	Not Applicable
3. Containment Radiation Monitoring (RCB3)	
Containment High Range Radiation Monitor (ARM-IRE-5400AS or ARM-IRE-5400BS) > 100 R/hr.	Not Applicable
4. Other Indications (RCB4)	
Not Applicable	RCS pressure dropping due to primary relief not reseating
5. Emergency Coordinator/EOF Director Judgment (RCB5)	
Any condition in the opinion of the Emergency Coordinator/EOF Director that indicates Loss or Potential Loss of the RCS barrier.	

Containment Barrier EALs

LOSS	POTENTIAL LOSS
1. Containment Pressure (CNB1)	
a. Rapid unexplained drop following initial rise **OR** b. Containment parameters not consistent with LOCA conditions	a. Containment pressure 50 PSIA and rising **OR** b. Explosive mixture exists **OR** c. Containment pressure > 17.7 PSIA with LESS THAN one full train of Containment Spray operating (1750 gpm)
2. Core Exit Thermocouple Readings (CNB2)	
Not Applicable	Core exit thermocouples >1200 degrees F and restoration procedures not effective within 15 minutes **OR** Core exit thermocouples > 700 degrees F with RVLMS upper plenum level equal to 0% or LOWER and restoration procedures not effective within 15 minutes
3. SG Secondary Side Release With Primary-to-Secondary Leakage (CNB3)	
Ruptured S/G is also faulted outside of containment **OR** Primary-to-Secondary leakrate >10 gpm with nonisolable steam release from affected S/G to the environment	Not Applicable
4. Containment Isolation Valve Status After Containment Isolation (CNB4)	
Unisolable breach of containment with a direct release path to the environment following containment isolation actuation.	Not Applicable
5. Significant Radioactive Inventory in Containment (CNB5)	
Not Applicable	Containment High Range Radiation Monitor (ARM-IRE-5400AS or ARM-IRE-5400BS) > 4000 R/hr.
6. Emergency Coordinator/EOF Director Judgment (CNB6)	
Any condition in the opinion of the Emergency Coordinator/EOF Director that indicates Loss or Potential Loss of the Containment barrier.	

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY				
Security	**HG1** HOSTILE ACTION resulting in loss of physical control of the facility. 1 2 3 4 5 6 D **Emergency Action Level(s):** 1. A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions. **OR** 2. A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and IMMINENT fuel damage is likely for a freshly off-loaded reactor core in pool.	**HS1** HOSTILE ACTION within the PROTECTED AREA. 1 2 3 4 5 6 D **Emergency Action Level(s):** 1. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Waterford 3 Security Shift Supervision.	**HA1** HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat. 1 2 3 4 5 6 D **Emergency Action Level(s):** 1. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Waterford 3 Security Shift Supervision **OR** 2. A validated notification from NRC of an airliner attack threat within 30 minutes of the site.	**HU1** Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant. 1 2 3 4 5 6 D **Emergency Action Level(s):** (1 or 2 or 3) 1. A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by the Waterford 3 Security Shift Supervision **OR** 2. A credible site specific security threat notification **OR** 3. A validated notification from NRC providing information of an aircraft threat.
Discretionary	**HG2** Other conditions existing which in the judgment of the Emergency Coordinator/EOF Director warrant declaration of General Emergency. 1 2 3 4 5 6 D **Emergency Action Level(s):** 1. Other conditions exist which in the judgment of the Emergency Coordinator/EOF Director indicate that events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	**HS2** Other conditions existing which in the judgment of the Emergency Coordinator/EOF Director warrant declaration of Site Area Emergency. 1 2 3 4 5 6 D **Emergency Action Level(s):** 1. Other conditions exist which in the judgment of the Emergency Coordinator/EOF Director indicate that events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the Exclusion Area Boundary.	**HA2** Other conditions existing which in the judgment of the Emergency Coordinator/EOF Director warrant declaration of an Alert. 1 2 3 4 5 6 D **Emergency Action Level(s):** 1. Other conditions exist which in the judgment of the Emergency Coordinator/EOF Director indicate that events are in process or have occurred which involve actual or likely potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	**HU2** Other conditions existing which in the judgment of the Emergency Coordinator warrant declaration of an Unusual Event. 1 2 3 4 5 6 D **Emergency Action Level(s):** 1. Other conditions exist which, in the judgment of the Emergency Coordinator, indicate that events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

Plant Modes (white boxes indicate applicable modes) 1 **Power Operations** 2 **Startup** 3 **Hot Standby** 4 **Hot Shutdown** 5 **Cold Shutdown** 6 **Refueling** D **Defueled**

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY				
Main Control Room Evacuation		**HS3** Control Room evacuation has been initiated and plant control cannot be established. 1 2 3 4 5 6 D **Emergency Action Level(s):** 1. Control Room evacuation has been initiated **AND** Control of the plant cannot be established in accordance with OP-901-502, Evacuation of Control Room & Subsequent Plant Shutdown within 15 minutes.	**HA3** Control Room evacuation has been initiated. 1 2 3 4 5 6 D **Emergency Action Level(s):** 1. Entry into OP-901-502, Evacuation of Control Room & Subsequent Plant Shutdown.	
Fire			**HA4** FIRE or EXPLOSION affecting the operability of plant safety systems required to establish or maintain safe shutdown. 1 2 3 4 5 6 D **Emergency Action Level(s):** 1. FIRE or EXPLOSION in the Reactor Auxiliary Building, Containment or Cooling Tower Areas **AND** Affected system parameter indications show degraded performance or plant personnel report VISIBLE DAMAGE to permanent structures or equipment within the specified area.	**HU4** FIRE within PROTECTED AREA boundary not extinguished within 15 minutes of detection. 1 2 3 4 5 6 D **Emergency Action Level(s):** 1. FIRE in or contiguous to Condensate Polisher Building, Containment, Fuel Handling Building, Reactor Auxiliary Building, Cooling Tower Areas or Turbine Building not extinguished within 15 minutes of Control Room notification or verification of a Control Room alarm.

Plant Modes (white boxes indicate applicable modes) 1 **Power Operations** 2 **Startup** 3 **Hot Standby** 4 **Hot Shutdown** 5 **Cold Shutdown** 6 **Refueling** D **Defueled**

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY			
Toxic Gas		**HA5** Release of toxic or flammable gases within or contiguous to VITAL AREA which jeopardizes operation of systems required to maintain safe operations or establish or maintain safe shutdown. 1 2 3 4 5 6 D **Emergency Action Levels:** (1 or 2) 1. Report or detection of toxic gases within or contiguous to VITAL AREA in concentrations that may result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH) **OR** 2. Report or detection of gases in concentration > LOWER FLAMMABILITY LIMIT within or contiguous to VITAL AREA.	**HU5** Release of toxic or flammable gases deemed detrimental to normal operation of the plant. 1 2 3 4 5 6 D **Emergency Action Level(s):** (1 or 2) 1. Report or detection of toxic or flammable gases that has or could enter the Exclusion Area Boundary in amounts that can affect NORMAL PLANT OPERATIONS **OR** 2. Report by St. Charles Parish for evacuation or sheltering of site personnel based on an offsite event.

Plant Modes (white boxes indicate applicable modes) 1 **Power Operations** 2 **Startup** 3 **Hot Standby** 4 **Hot Shutdown** 5 **Cold Shutdown** 6 **Refueling** D **Defueled**

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY			
Natural Events		**HA6** Natural and destructive phenomena affecting the plant VITAL AREA. 1 2 3 4 5 6 D **Emergency Action Level(s):** (1 or 2 or 3 or 4 or 5) 1 RED LIGHT on the seismic monitor panel indicates a VALID Seismic Event > Operating Basis Earthquake (OBE) **OR** 2. Tornado or high winds > 100 mph within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the following plant structures/equipment or Control Room indication of degraded performance of those systems • Containment • Reactor Auxiliary Building • Turbine Building • Cooling Tower Areas **OR** 3. Vehicle crash within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the following plant structures or equipment therein or Control Room indication of degraded performance of those systems • Containment • Reactor Auxiliary Building • Turbine Building • Cooling Tower Areas **OR** 4. Turbine failure-generated missiles result in any VISIBLE DAMAGE to or penetration of any of the following plant areas • Containment • Reactor Auxiliary Building • Cooling Tower Areas **OR** 5. Uncontrolled flooding in the Reactor Auxiliary Building or Cooling Tower Areas that results in degraded safety system performance as indicated in the Control Room or that creates industrial safety hazards (e.g., electric shock) that preclude access necessary to operate or monitor safety equipment.	**HU6** Natural and destructive phenomena affecting the PROTECTED AREA. 1 2 3 4 5 6 D **Emergency Action Level(s):** (1 or 2 or 3 or 4 or 5 or 6 or 7 or 8) 1. Earthquake felt in plant and detected on station seismic instrumentation **OR** 2. Report by plant personnel of tornado or high winds > 100 mph striking within PROTECTED AREA boundary **OR** 3. Vehicle crash into plant structures or systems within PROTECTED AREA boundary **OR** 4. Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment **OR** 5. Report of turbine failure resulting in casing penetration or damage to turbine or generator seals **OR** 6. Uncontrolled flooding in Reactor Auxiliary Building or Cooling Tower Areas that has the potential to affect safety related equipment needed for the current operating mode **OR** 7. Hurricane force winds (≥ 74 mph) expected to arrive on site in ≤ 12 hours as projected by the National Weather Service for a hurricane event **OR** 8. River water level at the intake structure > +27 FT MSL.

Plant Modes (white boxes indicate applicable modes) | 1 **Power Operations** | 2 **Startup** | 3 **Hot Standby** | 4 **Hot Shutdown** | 5 **Cold Shutdown** | 6 **Refueling** | D **Defueled**

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
SYSTEM MALFUNCTION				
Loss of AC Power	**SG1** Prolonged loss of all offsite power and prolonged loss of all onsite AC power to essential busses. (1 2 3 4) **Emergency Action Level(s):** 1. Loss of power to all unit auxiliary and startup transformers **AND** Failure of both 'A' and 'B' emergency diesel generators to supply power to emergency busses **AND** Either of the following: (a or b) a. Restoration of at least one emergency bus within 4 hours is not likely **OR** b. FA1 entry conditions met.	**SS1** Loss of all offsite power and loss of all onsite AC power to essential busses. (1 2 3 4) **Emergency Action Level(s):** 1. Loss of power to all unit auxiliary and startup transformers **AND** Failure of the 'A' and 'B' emergency diesel generators to supply power to emergency busses **AND** Failure to restore power to at least one emergency bus within 15 minutes from the time of loss of both offsite and onsite AC power.	**SA1** AC power capability to essential busses reduced to a single power source > 15 minutes such that any additional single failure would result in station blackout. (1 2 3 4) **Emergency Action Level(s):** 1. AC power capability to essential busses reduced to a single power source > 15 minutes **AND** Any additional single failure will result in station blackout.	**SU1** Loss of all offsite power to essential busses > 15 minutes. (1 2 3 4) **Emergency Action Level(s):** 1. Loss of power to all unit auxiliary and startup transformers > 15 minutes **AND** At least 'A' and 'B' emergency diesel generators supplying power to emergency busses.
Failure of Reactor Protection System	**SG3** Failure of the Reactor Protection System to complete an automatic trip and manual trip was NOT successful and there is indication of an extreme challenge to the ability to cool the core. (1 2) **Emergency Action Level(s):** 1. Indications exist that automatic and manual trip were not successful **AND** Either of the following: (a or b) a. Indication(s) exists that core cooling is extremely challenged as indicated by CET temperatures at or approaching 1200° F **OR** b. Indication(s) exists that heat removal is extremely challenged as indicated by inability to maintain at least one steam generator level > 50% wide range.	**SS3** Failure of Reactor Protection System instrumentation to complete or initiate an automatic reactor trip once a Reactor Protection System setpoint has been exceeded and manual trip was NOT successful. (1 2) **Emergency Action Level(s):** 1. Indication(s) exist that automatic and manual trip were not successful.	**SA3** Failure of Reactor Protection System instrumentation to complete or initiate an automatic reactor trip once a Reactor Protection System setpoint has been exceeded and manual trip was successful. (1 2 3) **Emergency Action Level(s):** 1. Indication(s) exist that indicate that the Reactor Protection System setpoint was exceeded and automatic trip did not occur and a successful manual trip occurred.	

Plant Modes (white boxes indicate applicable modes) 1 **Power Operations** 2 **Startup** 3 **Hot Standby** 4 **Hot Shutdown** 5 **Cold Shutdown** 6 **Refueling** D **Defueled**

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
SYSTEM MALFUNCTION				
Loss of DC		**SS4** Loss of all vital DC power. 1 2 3 4 **Emergency Action Level(s):** 1. Loss of all Vital DC power based on bus voltage indications < 108 volts for > 15 minutes.		
Heat Sink		**SS5** Complete loss of heat removal capability. 1 2 3 4 **Emergency Action Level(s):** 1. Loss of core cooling and heat sink.		
Loss of Annunciators		**SS6** Inability to monitor a SIGNIFICANT TRANSIENT in progress. 1 2 3 4 **Emergency Action Level(s):** 1. a. Loss of most or all annunciator cabinets C, D, H, K, M, N, SA, SB annunciators associated with safety systems **AND** b. Compensatory non-alarming indications are unavailable **AND** c. Indications needed to monitor safety functions (reactivity control, core cooling, maintaining reactor coolant system integrity or maintaining containment integrity) are unavailable **AND** d. SIGNIFICANT TRANSIENT in progress. .	**SA6** UNPLANNED loss of most or all safety system annunciation or indication in the Control Room with either (1) a SIGNIFICANT TRANSIENT in progress, or (2) compensatory non-alarming indicators are unavailable. 1 2 3 4 **Emergency Action Level(s):** 1. UNPLANNED loss of most or all annunciator cabinets C, D, H, K, M, N, SA, SB annunciators or indicators associated with safety systems > 15 minutes **AND** Either of the following (a or b): a. SIGNIFICANT TRANSIENT is in progress **OR** b. Compensatory non-alarming indications are unavailable.	**SU6** UNPLANNED loss of most or all safety system annunciation or indication in the Control Room > 15 minutes. 1 2 3 4 **Emergency Action Level(s):** 1. UNPLANNED loss of most or all annunciator cabinets C, D, H, K, M, N, SA, SB annunciators or indicators associated with safety systems > 15 minutes.

Plant Modes (white boxes indicate applicable modes) 1 **Power Operations** 2 **Startup** 3 **Hot Standby** 4 **Hot Shutdown** 5 **Cold Shutdown** 6 **Refueling** D **Defueled**

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

SYSTEM MALFUNCTION

RCS Leakage

SU7 RCS leakage. 1 2 3 4

Emergency Action Levels: (1 or 2)

1. Unidentified or pressure boundary leakage > 10 gpm

OR

2. Identified leakage > 25 gpm.

Loss of Communications

SU8 Unplanned loss of all onsite or offsite communications capabilities. 1 2 3 4

Emergency Action Levels: (1 or 2)

1. Loss of all Table M1 onsite communications equipment affecting the ability to perform routine operations

OR

2. Loss of all Table M2 offsite communications capability.

Table M1 Onsite Communications Equipment
Plant radio system
Plant paging system
In-plant telephones
Sound powered phones

Table M2 Offsite Communications Equipment
All telephone lines (commercial and microwave)
Industrial Hot Line
ENS
Civil Defense Radios
Operational Hotline

Fuel Clad Degredation

SU9 Fuel clad degradation. 1 2 3 4

Emergency Action Level(s):

1. Reactor coolant sample activity value indicating fuel clad degradation > Technical Specification allowable limits
   - > 60 µCi/gm DEI

OR

   - >1.0 µCi/gm DEI for more than 48 hours during one continuous time interval

OR

   - >100/Ē µCi/gm.

Inadvertent Criticality

SU10 Inadvertent criticality. 3 4

Emergency Action Level(s):

1. An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

Tech. Spec. Shutdown

SU11 Inability to reach required shutdown within Technical Specification time limits. 1 2 3 4

Emergency Action Level(s):

1. Plant is not brought to required operating mode within Technical Specifications LCO Action Statement time.

Plant Modes (white boxes indicate applicable modes) 1 **Power Operations** 2 **Startup** 3 **Hot Standby** 4 **Hot Shutdown** 5 **Cold Shutdown** 6 **Refueling** D **Defueled**

WATERFORD 3 EAL BASIS DOCUMENT

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WATERFORD 3 EAL BASIS DOCUMENT

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General Notes on Basis Document Use

Plant Operating Mode Usage for Waterford 3 EALs:

Mode 1 = Power Operations – Reactor Power > 5%, Keff ≥ 0.99

Mode 2 = Startup – Reactor Power ≤ 5%, Keff ≥ .99

Mode 3 = Hot Standby – RCS ≥ 350º F, Keff < .99

Mode 4 = Hot Shutdown – 200º F< RCS <350º F, Keff < .99

Mode 5 = Cold Shutdown – RCS < 200º F, Keff < .99

Mode 6 = Refueling – RCS < 140º F, Keff < .95, Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed

Defueled (D) – All reactor fuel removed from reactor pressure vessel (full core offload during refueling or extended outage). This is not an operating mode designation by Technical Specifications.

This basis document serves two basic functions:

- It provides background and explanatory information based on NEI 99-01 to present a basis for the origination of the Waterford 3 EALs for reviewers and users.
- The second function this basis document may provide is an aid to decision makers when making a determination to classify an emergency event. It is intended that decision makers have all the information in Attachment 7.1 of this procedure that they need to make a sound classification decision. Information that may be useful to a decision maker in classifying emergency events is indicated in red font in the Basis section for each IC in the Basis Document.

The expectation is that emergency classifications are to be made as soon as conditions are present and recognizable for the classification, but within 15 minutes or less in all cases of conditions present. A decision maker's use of this Basis Document for assistance is not intended to delay the classification.

DEFINITIONS

The following definitions are taken from NEI 99-01, the NEI White Paper on NRC Bulletin 2005-02 and the Waterford 3 Emergency Plan and applicable to the Waterford 3 emergency classification system:

AFFECTING SAFE SHUTDOWN: Event in progress has adversely affected functions that are necessary to bring the plant to and maintain it in the applicable HOT or COLD SHUTDOWN condition. Plant condition applicability is determined by Technical Specification LCOs in effect.

Example 1: Event causes damage that results in entry into an LCO that requires the plant to be placed in HOT SHUTDOWN. HOT SHUTDOWN is achievable, but COLD SHUTDOWN is not. This event **is not** "AFFECTING SAFE SHUTDOWN."

Example 2: Event causes damage that results in entry into an LCO that requires the plant to be placed in COLD SHUTDOWN. HOT SHUTDOWN is achievable, but COLD SHUTDOWN is not. This event **is** "AFFECTING SAFE SHUTDOWN."

BOMB: refers to an explosive device suspected of having sufficient force to damage plant systems or structures.

CIVIL DISTURBANCE: is a group of persons violently protesting station operations at the site.

CONFINEMENT BOUNDARY: is the barrier(s) between areas containing radioactive substances and the environment.

CONTAINMENT CLOSURE: Those actions taken by procedure within acceptable times as specified by procedure to close containment when in modes 5 or 6. Reference OP-901-131, Shutdown Cooling Malfunction, Attachment 7.1.

CONTAINMENT INTEGRITY: refers to that condition of the containment described in Technical Specifications definition 1.7.

EMERGENCY ACTION LEVEL (EAL): A pre-determined, site-specific, observable threshold for a plant Initiating Condition that places the plant in a given emergency class. An EAL can be: an instrument reading; an equipment status indicator; a measurable parameter (onsite or offsite); a discrete, observable event; results of analyses; entry into specific emergency operating procedures; or another phenomenon which, if it occurs, indicates entry into a particular emergency class.

DEFINITIONS

EMERGENCY CLASS: One of a minimum set of names or titles, established by the Nuclear Regulatory Commission (NRC), for grouping off-normal nuclear power plant conditions according to (1) their relative radiological seriousness, and (2) the time-sensitive onsite and off-site radiological emergency preparedness actions necessary to respond to such conditions. The existing radiological emergency classes, in ascending order of seriousness, are called:

- Notification of Unusual Event (Unusual Event)
- Alert
- Site Area Emergency
- General Emergency

EXCLUSION AREA BOUNDARY (EAB): For Waterford 3 EALs, the Emergency Plan Exclusion Area Boundary is the site boundary. The term “Exclusion Area Boundary” or “EAB" is used throughout the Waterford 3 EALs as the site boundary. The Emergency Plan defines the Exclusion Area Boundary (EAB) as “The border of the EXCLUSION AREA or an area corresponding to a distance of 914 meters from the Waterford 3 reactor.”

EXPLOSION: is a rapid, violent, unconfined combustion, or catastrophic failure of pressurized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components.

EXTORTION: is an attempt to cause an action at the station by threat of force.

FAULTED: in a steam generator, the existence of secondary side leakage that results in an uncontrolled decrease in steam generator pressure or the steam generator being completely depressurized.

FIRE: is 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.

HOSTAGE: is a person(s) held as leverage against the station to ensure that demands will be met by the station.

DEFINITIONS

HOSTILE ACTION: an act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidates the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the NPP. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the OWNER CONTROLLED AREA.)

HOSTILE FORCE: One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH): A condition that either poses an immediate threat to life and health or an immediate threat of severe exposure to contaminants which are likely to have adverse delayed effects on health.

IMMINENT: Mitigation actions have been ineffective, additional actions are not expected to be successful, and trended information indicates that the event or condition will occur. Where IMMINENT timeframes are specified, they shall apply.

INITIATING CONDITION (IC): One of a predetermined subset of nuclear power plant conditions where either the potential exists for a radiological emergency, or such an emergency has occurred.

INTRUSION / INTRUDER: is a person(s) present in a specified area without authorization. Discovery of a BOMB in a specified area is indication of INTRUSION into that area by a HOSTILE FORCE.

LOWER FLAMMABILITY LIMIT (LFL): The minimum concentration of a combustible substance that is capable of propagating a flame through a homogenous mixture of the combustible and a gaseous oxidizer.

NORMAL PLANT OPERATIONS: activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures. Entry into offnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from NORMAL PLANT OPERATIONS.

OWNER CONTROLLED AREA (OCA) The external area contiguous to the PROTECTED AREA extending outward to Entergy Louisiana, Inc. property lines.

DEFINITIONS

PROTECTED AREA: The area encompassed by physical barriers (the security fence) and to which access is controlled into the VITAL AREAS of the plant.

RUPTURED: in a steam generator, existence of primary-to-secondary leakage of a magnitude sufficient to require or cause a reactor trip and safety injection.

SABOTAGE: is deliberate damage, mis-alignment, or mis-operation of plant equipment with the intent to render the equipment inoperable. Equipment found tampered with or damaged due to malicious mischief may NOT meet the definition of SABOTAGE until this determination is made by security supervision.

SECURITY CONDITION: Any Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.

SIGNIFICANT TRANSIENT: is an UNPLANNED event involving one or more of the following: (1) automatic turbine runback >25% thermal reactor power, (2) electrical load rejection >25% full electrical load, (3) Reactor Trip, (4) Safety Injection Activation, or (5) thermal power oscillations >10%.

STRIKE ACTION: is a work stoppage within the PROTECTED AREA by a body of workers to enforce compliance with demands made on Entergy or its affiliates. The STRIKE ACTION must threaten to interrupt NORMAL PLANT OPERATIONS.

UNPLANNED: a parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

VALID: an indication, report, or condition, is considered to be VALID when it is 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.

VISIBLE DAMAGE: is damage to equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

VITAL AREA: is any area, normally within the PROTECTED AREA, which contains equipment, systems, components, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.

ABNORMAL RADIATION LEVELS/RADIOLOGICAL EFFLUENTS

ABNORMAL RADIATION LEVELS/RADIOLOGICAL EFFLUENTS

AU1

Initiating Condition – NOTIFICATION OF UNUSUAL EVENT

Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds 2 times the radiological effluent ODCM limits for ≥ 60 minutes.

Operating Mode Applicability: All

Emergency Action Level(s): (1 or 2 or 3)

1. VALID reading on any effluent monitor that exceeds 2 times the alarm setpoint established by a current radioactivity discharge permit for ≥ 60 minutes.

OR

2. VALID reading on one or more of the following radiation monitors that exceeds the reading shown for ≥ 60 minutes:

MONITOR	CONC.	EFFLUENT RATE
CONDENSER EXHAUST WRGM PRM-IRE-0002, RE0002-4		1.51E+05 uCi/sec
FUEL HANDLING BUILDING EXHAUST PIG, GAS CHANNEL , PRM-IRE-5107A or B, RE5107A-1 or RE5107B-1	1.61E-02 uCi/cc	
FUEL HANDLING BUILDING EXHAUST WRGM, PRM-IRE-3032, RE3032-4		2.25E+05 uCi/sec
PLANT STACK PIG GAS CHANNEL PRM-IRE-0100.1S or 2S, RE0100.1-1 or RE0100.2-1	3.45E-03 uCi/cc	
PLANT STACK WRGM PRM-IRE-0110, RE0110-4		1.51E+05 uCi/sec
[1]DRY COOLING TOWER SUMPS MONITOR, PRM-IRE-6775 or PRM-IRE-6776, RE6775-1 or RE6776-1	8.49E-04 uCi/ml	
[1]TURBINE BUILDING INDUSTRIAL WASTE SUMP MONITOR, PRM-IRE-6778, RE6778-1	8.49E-04 uCi/ml	

[1]Monitor reading not applicable if sump discharge is aligned to circulating water discharge.

OR

3. Confirmed grab sample analyses for gaseous or liquid releases indicates concentrations or release rates, with a release duration of ≥ 60 minutes, in excess of 2 times ODCM based limits from the Technical Requirements Manual (TRM) (Table A1).

Table A1 TRM Limits		
	ALERT	UE
Gaseous Release		
Noble Gases: ≤ 500 mrem/yr whole body	1.00E+05	1000
Noble Gases: ≤ 3000 mrem/yr skin	6.00E+05	6000
I-131, I-133, H-3 and particulates with half-lives > 8 days: ≤ 1500 mrem/year to any organ	3.00E+05	3000
Liquid Release		
Whole body: < 1.50 mrem/quarter	300	3
< 3 mrem/yr	600	6
Any Organ: < 5 mrem/quarter	1000	10
< 10 mrem/yr	2000	20

Basis:

This IC addresses a potential or actual decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time. Waterford 3 SES incorporates features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. These controls are located in the Offsite Dose Calculation Manual (ODCM). The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

The ODCM multiples are specified in AU1 (and AA1) only to distinguish between non-emergency conditions, and from each other. While these multiples obviously correspond to an offsite dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, NOT the magnitude of the associated dose or dose rate.

ABNORMAL RADIATION LEVELS/RADIOLOGICAL EFFLUENTS

AU1

The ODCM contains the specific release limits and appropriate surveillance requirements which normally monitor these limits. Releases should not be prorated or averaged over 60 minutes. For example, a release exceeding 4 times ODCM limits for 30 minutes does not meet the threshold for this event classification. The one-hour time period allows sufficient time to isolate any release after exceeding ODCM limits. Releases continuing for more than one hour represent inability to isolate or control the release. The Emergency Coordinator should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, then the Emergency Coordinator should, in the absence of data to the contrary, assume that the release has exceeded 60 minutes and make the emergency declaration.

UNPLANNED, as used in this context, includes any release for which a liquid waste release or a gaseous waste release discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm set points, etc.) on the applicable package permit. Unplanned releases in excess of two times of the technical specification limit that continue for 60 minutes or longer represent an uncontrolled situation and a potential degradation in the level of safety. It is not intended that the release be averaged over 60 minutes. The event should be declared as soon as it is determined that the release duration has or will likely exceed one hour.

EAL #1 addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed two times the Technical Specification limit and releases are not terminated within 60 minutes. In all cases, the applicable monitor is expected to be in **high alarm**, but AU1 and AA1 EAL #1 are based on the reading on the monitor and not its alarm status. The emergency classification is not made simply on the basis that the monitor has been in high alarm for 60 minutes. This alarm setpoint may be associated with a planned batch release, or a continuous release path. In either case, the setpoint is established by the ODCM to warn of a release that is not in compliance. Indexing the EAL threshold to the ODCM setpoints in this manner insures that the EAL threshold will never be less than the setpoint established by a specific discharge permit.

ABNORMAL RADIATION LEVELS/RADIOLOGICAL EFFLUENTS

AU1

EAL #2 is similar to EAL #1, but is intended to address effluent or accident radiation monitors on release pathways for which a discharge permit would not be prepared for a non-routine release. The ODCM establishes a methodology for determining effluent radiation monitor setpoints. The ODCM specifies default source terms from the UFSAR and, for gaseous releases, prescribes the use of pre-determined annual average meteorology in the most limiting downwind sector for showing compliance with the regulatory commitments. These monitor reading EALs have been determined using this methodology. The values used on the Dry Cooling Tower and Turbine Building sump discharge are based on the release pathway being aligned to the Storm Water System or Discharge Canal vice the circulating water system and are not applicable if the pathway is aligned to the circulating water system. Grab sample analysis of the circulation water discharge, IAW EAL #3, would be necessary to determine the appropriate action.

EAL #3 addresses uncontrolled releases that are detected by sample analyses, particularly on unmonitored pathways, e.g., spills of radioactive liquids into storm drains, leakage into Mississippi river water system, etc.

Calculation HP-CALC-2005-002, “Emergency Action Levels (EALs) Abnormal Rad Levels and Radiological Effluent Based on Power Uprate Source Terms” and HP-CALC-2005-012, “ Emergency Action Levels (EALs) (Fuel Handling Building Accident) Based on Power Uprate Source Terms” provide the basis for the radiation monitor readings selected for AU1, AA1, AS1 and AG1. The guidance from NEI 99-01 (Basis for Radiological Effluent Initiating Conditions) and Appendix A were used for these calculations. The calculations assume the same meteorology (annual average meteorology) and source term (Offsite Dose Calculation Manual – ODCM default source term) for all four emergency classifications. The back calculation methodology for the Site Area and General Emergency values utilizes the dose assessment method used by responders in emergency facilities to determine offsite doses and its corresponding dose factors and iodine to noble gas ratios. The NEI 99-01 Appendix A caution regarding overly conservative iodine to noble gas ratios was also considered in the calculation with an appropriate ratio correction factor selected.

ABNORMAL RADIATION LEVELS/RADIOLOGICAL EFFLUENTS

AU2

Initiating Condition – NOTIFICATION OF UNUSUAL EVENT

Unexpected rise in plant radiation.

Operating Mode Applicability: All

Emergency Action Level(s): (1 or 2)

1. a. VALID indication of uncontrolled water level drop in the reactor refueling cavity, spent fuel pool, or fuel transfer canal with all irradiated fuel assemblies remaining covered by water.

- Level drop may be indicated by personnel observation, spent fuel pool level below level plate, refueling crew report, indication on area security camera, RWSP level drop due to makeup demands.

AND

b. Unplanned VALID Area Radiation Monitor rise on any of the following:

- CONTAINMENT AREA RADIATION MONITORS (PURGE ISOLATION), (ARM-IRE-5024S, 5025S, 5026S OR 5027S, RE5024-1, RE5025-1, RE5026-1 OR RE5027-1)
- CONTAINMENT +46 STAIRS MONITORS, (ARM-IRE-5014 OR 5015, RE5014-1 OR RE5015-1)
- REFUELING BRIDGE AREA RADIATION MONITOR (ARM-IRE-5013, RE5013-1)
- FHB AREA RADIATION MONITORS (ISOLATION), (ARM-IRE-0300.1S, .2S, .3S OR .4S, RE0300.1-1, RE0300.2-1, RE0300.3-1, OR RE0300.4-1)

OR

2. Unplanned VALID Area Radiation Monitor readings indicate a rise in plant radiation levels by a factor of 1000 over normal levels (highest reading in the past 24 hours excluding the current peak value).

Basis:

This IC addresses increased radiation levels as a result of water level decreases above the reactor vessel flange or events that have resulted, or may result, in unexpected increases in radiation dose rates within plant buildings. These radiation increases represent a loss of control over radioactive material and may represent a potential degradation in the level of safety of the plant.

ABNORMAL RADIATION LEVELS/RADIOLOGICAL EFFLUENTS

AU2

In light of Reactor Cavity Seal failure incidents at two different PWRs and loss of water in the Spent Fuel Pit/Fuel Transfer Canal at a BWR, explicit coverage of these types of events via EAL #1 is appropriate given their potential for increased doses to plant staff. Classification as an Unusual Event is warranted as a precursor to a more serious event.

Specific indications may include instrumentation such as water level and local area radiation monitors, and personnel (e.g., refueling crew) reports. Depending on available level instrumentation, the declaration may be based on indications of water makeup rate or decrease in Refueling Water Storage Pool level. Video cameras (Security or outage-related) may allow remote observation of level. Credit should not be taken for inventory additions to maintain level above the threshold.

While a radiation monitor could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered. For example, the reading on an area radiation monitor located on the refueling bridge may increase due to planned evolutions such as head lift, or even a fuel assembly being raised in the refuel mast. Generally, increased radiation monitor indications will need to be combined with another indicator (or personnel report) of water loss. For refueling events where the water level drops below the reactor vessel flange, classification would be via CU2. This event escalates to an Alert per AA2 if irradiated fuel outside the reactor vessel is uncovered. For events involving irradiated fuel in the reactor vessel, escalation would be via the Fission Product Barrier Matrix for events in operating modes 1-4.

EAL #2 addresses UNPLANNED increases in in-plant radiation levels that represent a degradation in the control of radioactive material, and represent a potential degradation in the level of safety of the plant. Normal levels can be considered as the HIGHEST reading in the past twenty-four hours excluding the current peak value.

This event escalates to an Alert in accordance with AA3 if the increase in dose rates impedes personnel access necessary for safe operations.

ABNORMAL RADIATION LEVELS/RADIOLOGICAL EFFLUENTS

AA1

Initiating Condition – ALERT

Any UNPLANNED release of gaseous or liquid radioactivity to the environment that exceeds 200 times the radiological effluent ODCM limits for ≥ 15 minutes.

Operating Mode Applicability: All

Emergency Action Level(s): (1 or 2 or 3)

1 VALID reading on any effluent monitor that exceeds 200 times the alarm setpoint established by a current radioactivity discharge permit for ≥ 15 minutes.

OR

2. VALID reading on one or more of the following radiation monitors that exceeds the reading shown for ≥ 15 minutes:

- CONDENSER EXHAUST WRGM (PRM-IRE-0002, RE0002-4) indicates release rate > 1.51E+07 uCi/sec.
- FUEL HANDLING BUILDING EXHAUST WRGM (PRM-IRE-3032, RE3032-4) indicates > 2.25E+07 uCi/sec.
- PLANT STACK WRGM (PRM-IRE-0110, RE0110-4) indicates > 1.51E+07 uCi/sec.

OR

AA1

3. Confirmed grab sample analyses for gaseous or liquid releases indicates concentrations or release rates, with a release duration of ≥ 15 minutes, in excess of 200 times ODCM based limits from the Technical Requirements Manual (TRM) (Table A1).

Table A1 TRM Limits		
	ALERT	UE
Gaseous Release		
Noble Gases: ≤ 500 mrem/yr whole body	1.00E+05	1000
Noble Gases: ≤ 3000 mrem/yr	6.00E+5	6000
I-131, I-133, H-3 and particulates with half-lives > 8 days: ≤ 1500 mrem/year to any organ	3.00E+05	3000
Liquid Release		
Whole body: < 1.50 mrem/quarter	300	3
< 3 mrem/yr	600	6
Any Organ: < 5 mrem/quarter	1000	10
< 10 mrem/yr	2000	20

Basis:

This IC addresses a potential or actual decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time. Waterford 3 SES incorporates features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. These controls are located in the ODCM. The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of degradation in these features and/or controls.

The ODCM multiples are specified in AA1 (and AU1) only to distinguish between non-emergency conditions, and from each other. While these multiples obviously correspond to an offsite dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, NOT the magnitude of the associated dose or dose rate.

Releases should not be prorated or averaged. For example, a release exceeding 100 times ODCM limits for 30 minutes does not meet the threshold for this event classification.

ABNORMAL RADIATION LEVELS/RADIOLOGICAL EFFLUENTS

AA1

UNPLANNED, as used in this context, includes any release for which a liquid waste release or a gaseous waste release discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm set points, etc.) on the applicable package permit. The Emergency Coordinator/EOF Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, then the Emergency Coordinator/EOF Director should, in the absence of data to the contrary, assume that the release has exceeded 15 minutes and make the emergency declaration.

EAL #1 addresses radioactivity releases that for whatever reason cause effluent radiation monitor readings that exceed 200 times the alarm setpoint established by the radioactivity discharge permit. In all cases, the applicable monitor is expected to be in **alarm**, but AU1 and AA1 EAL #1 are based on the reading on the monitor and not its alarm status. The emergency classification is not made simply on the basis that the monitor has been in high alarm for 15 minutes. This alarm setpoint may be associated with a planned batch release, or a continuous release path. In either case, the setpoint is established by the ODCM to warn of a release that is not in compliance. Indexing the EAL threshold to the ODCM setpoints in this manner insures that the EAL threshold will never be less than the setpoint established by a specific discharge permit.

EAL #2 is similar to EAL #1, but is intended to address effluent or accident radiation monitors on release pathways for which a discharge permit would not be prepared for a non-routine release. The ODCM establishes a methodology for determining effluent radiation monitor setpoints. The ODCM specifies default source terms from the UFSAR and, for gaseous releases, prescribes the use of predetermined annual average meteorology in the most limiting downwind sector for showing compliance with the regulatory commitments. These monitor reading EALs have been determined using this methodology.

EAL #3 addresses uncontrolled releases that are detected by sample analyses, particularly on unmonitored pathways, e.g., spills of radioactive liquids into storm drains, leakage into Mississippi river water system, etc.

ABNORMAL RADIATION LEVELS/RADIOLOGICAL EFFLUENTS

AA1

Calculation HP-CALC-2005-002, “Emergency Action Levels (EALs) Abnormal Rad Levels and Radiological Effluent Based on Power Uprate Source Terms” and HP-CALC-2005-012, “ Emergency Action Levels (EALs) (Fuel Handling Building Accident) Based on Power Uprate Source Terms” provide the basis for the radiation monitor readings selected for AU1, AA1, AS1 and AG1. The guidance from NEI 99-01 (Basis for Radiological Effluent Initiating Conditions) and Appendix A were used for these calculations. The calculations assume the same meteorology (annual average meteorology) and source term (Offsite Dose Calculation Manual – ODCM default source term) for all four emergency classifications. The back calculation methodology for the Site Area and General Emergency values utilizes the dose assessment method used by responders in emergency facilities to determine offsite doses and its corresponding dose factors and iodine to noble gas ratios. The NEI 99-01 Appendix A caution regarding overly conservative iodine to noble gas ratios was also considered in the calculation with an appropriate ratio correction factor selected.

ABNORMAL RADIATION LEVELS/RADIOLOGICAL EFFLUENTS

AA2

Initiating Condition – ALERT

Damage to irradiated fuel or loss of water level that has or will result in uncovering of irradiated fuel outside the reactor vessel.

Operating Mode Applicability: All

Emergency Action Level(s): (1 or 2)

1. VALID alarm or reading ≥ HIGH alarm limits on one or more of the following radiation monitors:

   - CONTAINMENT AREA RADIATION MONITORS (PURGE ISOLATION), (ARM-IRE-5024S, 5025S, 5026S OR 5027S, RE5024-1, RE5025-1, RE5026-1 OR RE5027-1) ≥ HIGH alarm

   - CONTAINMENT +46 STAIRS MONITORS, (ARM-IRE-5014 OR 5015, RE5014-1 OR RE5015-1) ≥ HIGH alarm

   - REFUELING BRIDGE AREA RADIATION MONITOR (ARM-IRE-5013, RE5013-1) ≥ HIGH alarm

   - FHB AREA RADIATION MONITORS (ISOLATION), (ARM-IRE-0300.1S, .2S, 3S OR .4S, RE0300.1-1, RE0300.2-1, RE0300.3-1, OR RE0300.4-1) ≥ 1000 mR/hr

   - FUEL HANDLING BUILDING EXHAUST PIG, GAS CHANNEL , PRM-IRE-5107A OR B, RE5107A-1 OR RE5107B-1 ≥ HIGH alarm

OR

2. Valid indication of uncontrolled water level drop in the reactor refueling cavity, spent fuel pool or fuel transfer canal that will result in irradiated fuel uncovering.

Basis:

This IC addresses specific events that have resulted, or may result, in unexpected increases in radiation dose rates within plant buildings, and may be a precursor to a radioactivity release to the environment. These events represent a loss of control over radioactive material and represent degradation in the level of safety of the plant. These events escalate from AU2 in that fuel activity has been released, or is anticipated due to fuel heatup.

Uncontrolled water level decrease may be detected by visual observation, increased radiation levels or various other symptoms that are considered valid indicators of the event. Fuel uncovery may be expected based on abnormal radiation levels, visual observation, or best judgment of the Emergency Coordinator/EOF Director based on present and past trends.

EAL #1 addresses radiation monitor indications of fuel uncovery and/or fuel damage. Increased readings on ventilation monitors may be indication of a radioactivity release from the fuel, confirming that damage has occurred. Increased background at the monitor due to water level decrease may mask increased ventilation exhaust airborne activity and needs to be considered. While a radiation monitor could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered. For example, the monitor could in fact be properly responding to a known event involving transfer or relocation of a source, stored in or near the fuel pool or responding to a planned evolution such as removal of the reactor head. Application of these Initiating Conditions requires understanding of the actual radiological conditions present in the vicinity of the monitor.

In EAL #2, indications may include instrumentation such as water level and local area radiation monitors, and personnel (e.g., refueling crew) reports. Depending on available level indication, the declaration may be based on indications of water makeup rate or decrease in Refueling Water Storage Pool level. Video cameras (Security or outage-related) may allow remote observation of level.

Escalation, if appropriate, would occur via AS1 or AG1 or Emergency Coordinator/EOF Director Judgment.

ABNORMAL RADIATION LEVELS/RADIOLOGICAL EFFLUENTS

AA3

Initiating Condition – ALERT

Release of radioactive material or rise in radiation levels within the facility that impedes operation of systems required to maintain safe operations or to establish or maintain cold shutdown.

Operating Mode Applicability: All

Emergency Action Level(s): (1 or 2)

1. VALID radiation level > 15 mR/hr in areas requiring continuous occupancy to maintain plant safety functions.

   - Main Control Room Area Radiation Monitor (ARM-IRE- 5001, RE5001-1) > 15 mR/hr
   - Radiation level in CAS >15 mR/hr

OR

2. VALID radiation level > Table A2 value in plant vital areas requiring infrequent access to maintain plant safety functions (Table A2).

Table A2 Radiation Levels in Areas Requiring Infrequent Access	
VCT Room – 10 R/hr	Safeguards Rooms – 10 R/hr
VALUE FOR ALL AREAS BELOW IS 2.5 R/hr:	
+46 Chiller Area	BAM Tank Rooms
MSIV Areas	Relay Room
Electrical Penetration Area	Remote Shutdown Room
EDG Rooms	Battery Rooms
Valve Bay	Wing Areas
CVC-507 Valve Area	CCW Heat Exchanger Rooms
CCW Pump Rooms	

Basis:
The radiation levels in the EALs for this IC may be identified by a radiation monitor value or direct survey.

This IC addresses increased radiation levels that impede necessary access to operating stations, or other areas containing equipment that must be operated manually or that requires local monitoring, in order to maintain safe operation or perform a safe shutdown. It is this impaired ability to operate the plant that results in the actual or potential substantial degradation of the level of safety of the plant. The cause and/or magnitude of the increase in radiation levels is not a concern of this

ABNORMAL RADIATION LEVELS/RADIOLOGICAL EFFLUENTS

AA3

IC. The Emergency Coordinator/EOF Director must consider the source or cause of the increased radiation levels and determine if any other IC may be involved. For example, a 15 mR/hr dose rate in the control room or a high radiation monitor reading may be a problem in itself. However, the increase may also be indicative of high dose rates in the containment due to a LOCA. In this latter case, an SAE or GE may be indicated by the fission product barrier matrix EALs.

This IC is not meant to apply to increases in the containment dome radiation monitors as these are events which are addressed in the fission product barrier matrix EALs. Nor is it intended to apply to anticipated temporary increases due to planned events (e.g., incore detector movement, radwaste container movement, depleted resin transfers, etc.).

The value of 15mR/hr is derived from the GDC 19 value of 5 Rem in 30 days with adjustment for expected occupancy times. Although Section III.D.3 of NUREG-0737, *"Clarification of TMI Action Plan Requirements"*, provides that the 15 mR/hr value can be averaged over the 30 days, the value is used here without averaging, as a 30 day duration implies an event potentially more significant than an Alert.

For areas requiring infrequent access, the value of 2.5 R/hr was selected for those areas that are not already high radiation areas because it is a value with a specific action for Radiation Protection Superintendent approval addressed in RP-105, Radiation Work Permits that would result in exposure control measures intended to maintain doses within normal occupational guidelines and limits (i.e., 10CFR20), and in doing so, will impede necessary access. The 10 R/hr value is selected for those areas that are already high radiation areas because some greater amount of radiological control is already in place as a baseline condition for these areas (such as the requirement to notify Radiation Protection and get a briefing) prior to entry. In selecting both the 2.5 R/hr value and the 10 R/hr value, consideration was given to preclude unnecessary EAL entry for radiological conditions that may fluctuate during normal plant operations (e.g., incore detector movement, radwaste container movement, depleted resin transfers, etc.). As used here, *impede,* includes hindering or interfering provided that the interference or delay is sufficient to significantly threaten the safe operation of the plant. The list of plant areas was selected from a review of the 10 CFR 50 Appendix R analysis.

ABNORMAL RADIATION LEVELS/RADIOLOGICAL EFFLUENTS

AS1

Initiating Condition – SITE AREA EMERGENCY

Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mR TEDE or 500 mR CDE Thyroid for the actual or projected duration of the release.

Operating Mode Applicability: All

Emergency Action Level(s): (1 or 2 or 3)

Note: *If dose assessment results are available at the time of declaration, then the classification should be based on EAL #2 instead of EAL #1. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated.*

1. VALID reading on one or more of the following radiation monitors that exceeds or is expected to exceed the reading shown for ≥ 15 minutes:

   - CONDENSER EXHAUST WRGM (PRM-IRE-0002, RE0002-4) indicates release rate > 2.69E+08 µCi/sec

   - FUEL HANDLING BUILDING EXHAUST WRGM (PRM-IRE-3032, RE3032-4) indicates release rate > 1.75E+08 µCi/sec

   - PLANT STACK WRGM (PRM-IRE-0110, RE0110-4) indicates release rate > 2.55E+08 µCi/sec

OR

2. Dose assessment using actual meteorology indicates doses > 100 mR TEDE or > 500 mR CDE Thyroid at or beyond the EAB.

OR

3. Field survey results indicate closed window dose rates >100 mR/hr expected to continue for > one hour; or analyses of field survey samples indicate CDE Thyroid ≥ 500 mR for one hour of inhalation, at or beyond the EAB.

ABNORMAL RADIATION LEVELS/RADIOLOGICAL EFFLUENTS

AS1

Basis:
This IC addresses radioactivity releases that result in doses at or beyond the EAB that exceed a small fraction of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public. While these failures may be addressed by other ICs, this IC provides appropriate diversity and addresses events which may not be able to be classified on the basis of plant status alone, e.g., fuel handling accident in spent fuel building.

The actual or projected dose of 100 mR TEDE is set at 10% of the EPA Protective Action Guide (PAG) values given in EPA-400-R-92-001, while the 500 mR CDE thyroid was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE. The TEDE integrated dose value also provides a desirable gradient between the Alert, Site Area Emergency and General Emergency classes.

The Emergency Coordinator/EOF Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

The monitor list in EAL #1 includes monitors on the primary potential release pathways (Plant stack, Primary/Secondary leak, Fuel Handling Accident) for Waterford 3. The EPA PAGs are expressed in terms of the sum of the effective *dose equivalent (EDE)* and the *committed effective dose equivalent (CEDE)*, or as the thyroid *committed dose equivalent (CDE).* For the purpose of these EALs, the dose quantity *total effective dose equivalent (TEDE),* as defined in 10 CFR 20, is used in lieu of *"…sum of EDE and CEDE…."* The EPA PAG guidance in EPA-400R-92-001 provides for the use of adult thyroid dose conversion factors.

The monitor reading EALs were determined using a dose assessment method that back calculates from the dose values specified in the IC. Calculation HP-CALC-2005-002, "Emergency Action Levels (EALs) Abnormal Rad Levels and Radiological Effluent Based on Power Uprate Source Terms" and HP-CALC-2005-012, "Emergency Action Levels (EALs) (Fuel Handling Building Accident) Based on Power Uprate Source Terms" provide the basis for the radiation monitor readings selected for AU1, AA1, AS1 and AG1. The guidance from NEI 99-01 (Basis for Radiological Effluent Initiating Conditions) and Appendix A were used for these calculations. The calculations assume the same meteorology (annual average meteorology) and source term (Offsite Dose Calculation Manual – ODCM default source term) for all four emergency classifications. The back calculation methodology for the Site Area and General Emergency values utilizes the dose assessment method used by responders in emergency facilities to determine offsite doses and its corresponding dose factors and iodine to noble gas ratios. The NEI 99-01 Appendix A caution regarding overly conservative iodine to noble gas ratios was also considered in the calculation with an appropriate ratio correction factor selected.

Since doses are generally not monitored in real-time, a release duration of one hour was assumed, and the EALs are based on a EAB (or beyond) dose of 100 mR/hour whole body or 500 mR/hour thyroid, whichever is more limiting (as was done for EALs #2 and #3). If analyses indicate a longer or shorter duration for the period in which the substantial portion of the activity is released, the longer duration should be used.

Since dose assessment in EALs #2 and #3 is based on actual meteorology, whereas the monitor readings in EAL #1 are not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, decision makers should ensure performance of dose assessments using actual meteorology and release information are performed in a timely manner when release conditions are detected. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), then the dose assessment results override the monitor reading EALs. However, classification should not be delayed pending the results of these dose assessments. If dose assessment team calculations can not be completed in 15 minutes, then valid monitor readings should be used for emergency classification.

Field team surveys in EAL #3 are performed at or beyond the EAB and at the most accurate indicator of the condition. Field data are independent of release elevation and meteorology. The assumed release duration is one hour for the basis of the EAL. Expected post accident source terms would be dominated by noble gases providing the dose rate value. Sampling of radioiodine by adsorption on a charcoal cartridge should determine the iodine value.

ABNORMAL RADIATION LEVELS/RADIOLOGICAL EFFLUENTS

AG1

Initiating Condition – GENERAL EMERGENCY

Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 1000 mR TEDE or 5000 mR CDE Thyroid for the actual or projected duration of the release using actual meteorology.

Operating Mode Applicability: All

Emergency Action Level(s): (1 or 2 or 3)

Note: *If dose assessment results are available at the time of declaration, then the classification should be based on EAL #2 instead of EAL #1. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to more accurately characterize the nature of the release.*

1. VALID reading on one or more of the following radiation monitors that exceeds or is expected to exceed the reading shown for ≥ 15 minutes:

   - CONDENSER EXHAUST WRGM (PRM-IRE-0002, RE0002-4) indicates release rate > 2.69E+09 uCi/sec

   - FUEL HANDLING BUILDING EXHAUST WRGM (PRM-IRE-3032, RE3032-4) indicates release rate > 1.75E+09 uCi/sec

   - PLANT STACK WRGM (PRM-IRE-0110, RE0110-4) indicates release rate > 2.55E+09 uCi/sec

OR

2. Dose assessment using actual meteorology indicates doses > 1000 mR TEDE or > 5000 mR CDE Thyroid at or beyond the EAB.

OR

3. Field survey results indicate closed window dose rates >1000 mR/hr expected to continue for > one hour; or analyses of field survey samples indicate CDE Thyroid ≥ 5000 mR for one hour of inhalation, at or beyond the EAB.

ABNORMAL RADIATION LEVELS/RADIOLOGICAL EFFLUENTS

AG1

Basis:

This IC addresses radioactivity releases that result in doses at or beyond the EAB that exceed the EPA Protective Action Guides (PAGs). Public protective actions are required. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage. While these failures are addressed by other ICs, this IC provides appropriate diversity and addresses events which may not be able to be classified on the basis of plant status alone. It is important to note that, for the more severe accidents, the release may be unmonitored or there may be large uncertainties associated with the source term and/or meteorology.

The Emergency Coordinator/EOF Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

The actual or projected dose of 1000 mR TEDE and 5000 mR CDE thyroid integrated doses are based on the EPA Protective Action Guide (PAG) values given in EPA-400-R-92-001, which indicates that public protective actions are indicated if doses exceed these values. This is consistent with the emergency class description of a General Emergency.

The monitor list in EAL #1 includes monitors on potential gaseous effluent release pathways (Plant stack, Primary/Secondary Leak, Fuel Handling Accident). The EPA PAGs are expressed in terms of the sum of the effective *dose equivalent (EDE)* and the *committed effective dose equivalent (CEDE)*, or as the thyroid *committed dose equivalent (CDE).* For the purpose of these EALs, the dose quantity *total effective dose equivalent (TEDE),* as defined in 10 CFR 20, is used in lieu of *"…sum of EDE and CEDE…."* The EPA PAG guidance in EPA-400R-92-001 provides for the use of adult thyroid dose conversion factors.

The monitor reading EALs were determined using a dose assessment method that back calculates from the dose values specified in the IC.

Calculation HP-CALC-2005-002, "Emergency Action Levels (EALs) Abnormal Rad Levels and Radiological Effluent Based on Power Uprate Source Terms" and HP-CALC-2005-012, "Emergency Action Levels (EALs) (Fuel Handling Building Accident) Based on Power Uprate Source Terms" provide the basis for the radiation monitor readings selected for AU1, AA1, AS1 and AG1. The guidance from NEI 99-01 (Basis for Radiological Effluent Initiating Conditions) and Appendix A were used for these calculations. The calculations assume the same meteorology (annual average meteorology) and source term (Offsite Dose Calculation Manual – ODCM default source term) for all four emergency classifications. The back calculation methodology for the Site Area and General Emergency values utilizes the dose assessment method used by responders in emergency facilities to determine offsite doses and its corresponding dose factors and iodine to noble gas ratios. The NEI 99-01 Appendix A caution regarding overly conservative iodine to noble gas ratios was also considered in the calculation with an appropriate ratio correction factor selected.

Since doses are generally not monitored in real-time, a release duration of one hour was assumed, and the EALs are based on a EAB (or beyond) dose of 1000 mR/hour whole body or 5000 mR/hour thyroid, whichever is more limiting (as was done for EALs #2 and #3). If analyses indicate a longer or shorter duration for the period in which the substantial portion of the activity is released, the longer duration should be used.

Since dose assessment in EALs #2 and #3 is based on actual meteorology, whereas the monitor readings in EAL #1 are not, the results from these assessments may indicate that the classification is not warranted. For this reason, decision makers should ensure performance of dose assessments using actual meteorology and release information are performed in a timely manner when release conditions are detected. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), then the dose assessment results override the monitor reading EALs. However, classification should not be delayed pending the results of these dose assessments. If dose assessment team calculations can not be completed in 15 minutes, then valid monitor readings should be used for emergency classification.

Field team surveys in EAL #3 are performed at or beyond the EAB and at the most accurate indicator of the condition. Field data are independent of release elevation and meteorology. The assumed release duration is one hour for the basis of the EAL. Expected post accident source terms would be dominated by noble gases providing the dose rate value. Sampling of radioiodine by adsorption on a charcoal cartridge should determine the iodine value.

COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

CU1

Initiating Condition – NOTIFICATION OF UNUSUAL EVENT

RCS leakage.

Operating Mode Applicability: **Cold Shutdown (Mode 5)**

Emergency Action Level(s): (1 or 2)

1. Unidentified or pressure boundary leakage > 10 gpm.

OR

2. Identified leakage > 25 gpm .

Basis:

This IC is included as a NOUE because it is considered to be a potential degradation of the level of safety of the plant. The 10 gpm value for the unidentified and pressure boundary leakage was selected as it is sufficiently large to be observable via normally installed instrumentation (e.g., Pressurizer level, RCS loop level instrumentation, etc.) or reduced inventory instrumentation such as level hose indication. Lesser values must generally be determined through time-consuming surveillance tests (e.g., mass balances). The EAL for identified leakage is set at a higher value due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage. Steam generator tube leakage is identified leakage. Prolonged loss of RCS inventory may result in escalation to the Alert level via either CA1 or CA3.

The difference between CU1 and CU2 deals with the RCS conditions that exist between cold shutdown and refueling mode applicability. In cold shutdown, the RCS will normally be intact and RCS inventory and level monitoring means such as Pressurizer level indication and makeup volume control tank levels are normally available. In the refueling mode, the RCS is not intact and reactor vessel level and inventory are monitored by different means.

COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

CU2

Initiating Condition – NOTIFICATION OF UNUSUAL EVENT

UNPLANNED loss of RCS inventory with irradiated fuel in the reactor vessel.

Operating Mode Applicability: **Refueling (Mode 6)**

Emergency Action Level(s): (1 or 2)

1. UNPLANNED RCS level drop below the vessel flange for ≥ 15 minutes

OR

2. a. Loss of reactor vessel inventory as indicated by unexplained containment sump level or reactor drain tank level rise

AND

   b. Reactor vessel level cannot be monitored

Basis:

This IC is included as a NOUE because it may be a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant. Refueling evolutions that decrease RCS water level below the reactor vessel flange are carefully planned and procedurally controlled. An UNPLANNED event that results in water level decreasing below the reactor vessel flange warrants declaration of a NOUE due to the reduced inventory that is available to keep the core covered. The allowance of 15 minutes was chosen because it is reasonable to assume that level can be restored within this time frame using one or more of the redundant means of refill that should be available. If level cannot be restored in this time frame then it may indicate a more serious condition exists. Continued loss of Inventory will result in escalation to the Alert level via either CA2 or CA3.

The difference between CU1 and CU2 deals with the RCS conditions that exist between cold shutdown and refueling modes. In cold shutdown, the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the refueling mode, the RCS is not intact and reactor vessel level and inventory are monitored by different means.

In the refueling mode, normal means of core temperature indication and RCS level indication may not be available. Redundant means of reactor vessel level indication

will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of inventory event, the operators would need to determine that reactor vessel inventory loss was occurring by observing containment sump and reactor drain tank level changes. Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage. Escalation to Alert would be via either CA2 or RCS heatup via CA3.

EAL 1 involves a decrease in RCS level below the top of the reactor vessel flange that continues for 15 minutes due to an **UNPLANNED** event. This EAL is **not** applicable to decreases in flooded reactor cavity level (covered by AU2 EAL1) **until** such time as the level decreases to the level of the vessel flange. If reactor vessel level continues to decrease and reaches the Bottom ID of the RCS Loop (12 ft. MSL for these ICs), then escalation to CA2 would be appropriate.

COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

CU3

Initiating Condition – NOTIFICATION OF UNUSUAL EVENT

UNPLANNED loss of decay heat removal capability with irradiated fuel in the reactor vessel.

Operating Mode Applicability: **Cold Shutdown (Mode 5)**
Refueling (Mode 6)

Emergency Action Level(s): (1 or 2)

1. An UNPLANNED event results in RCS temperature exceeding the Technical Specification cold shutdown temperature limit

OR

2. Loss of all RCS temperature and reactor vessel level indication for > 15 minutes.

Basis:

This IC is included as a NOUE because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In cold shutdown, the ability to remove decay heat relies primarily on forced cooling flow. Operation of the systems that provide this forced cooling may be jeopardized due to the unlikely loss of electrical power or RCS inventory. Temporary instrumentation and jumpers are maintained in service such that the operators are able to monitor RCS temperature and reactor vessel level so that escalation to the alert level via CA3 or CA1 will occur if required.

Loss of forced decay heat removal at reduced inventory may result in more rapid increases in reactor coolant temperatures depending on the time since shutdown. Escalation to the Alert level via CA3 is provided dependent upon containment closure and RCS integrity conditions.

Redundant means of reactor vessel level indication are procedurally installed in accordance with OP-001-003, Reactor Coolant System Drain Down, to assure that the ability to monitor level will not be interrupted. However, if all level and temperature indication were to be lost in either the cold shutdown or refueling modes, EAL 2 would result in declaration of a NOUE if either temperature or level indication cannot be restored within 15 minutes from the loss of both means of indication. Escalation to Alert would be via CA1 based on an inventory loss or CA3 based on exceeding RCS temperature criteria. The Emergency Coordinator must remain attentive to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the Emergency Coordinator, an imminent situation is at hand, the classification should be made as if the threshold has been exceeded.

COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

CU5

Initiating Condition – NOTIFICATION OF UNUSUAL EVENT

Loss of all offsite power to essential busses > 15 minutes.

Operating Mode Applicability: **Cold Shutdown (Mode 5)**
Refueling (Mode 6)

Emergency Action Level(s):

1. a. Loss of power to all unit auxiliary and startup transformers > 15 minutes.

AND

b. At least emergency diesel generator 'A' or 'B' is supplying power to emergency busses.

Basis:

Prolonged loss of AC power reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete loss of AC power (e.g. station blackout). Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Credit for Temporary Emergency Diesel Generators (TEDs) may **NOT** be taken because they are not a credited power source in the Technical Specifications for modes 5 and 6.

COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

CU6

Initiating Condition – NOTIFICATION OF UNUSUAL EVENT

UNPLANNED loss of required DC power > 15 minutes.

Operating Mode Applicability: **Cold Shutdown (Mode 5)**
Refueling (Mode 6)

Emergency Action Level:

1. a. UNPLANNED loss of vital DC power to required DC busses based on bus voltage indication < 108 volts.

AND

b. Failure to restore power to at least one required DC bus within 15 minutes from the time of loss.

Basis:

This IC refers to loss of vital DC power from the 3A-DC, 3B-DC or 3AB-DC busses dependant upon designated protected train operability status. The purpose of this IC and its associated EALs is to recognize a loss of DC power compromises the ability to monitor and control the removal of decay heat during Cold Shutdown or Refueling operations. This EAL is intended to be anticipatory in as much as the operating crew may not have necessary indication and control of equipment needed to respond to the loss.

UNPLANNED is included in this EAL to preclude the declaration of an emergency as a result of planned maintenance activities. Routinely, Waterford 3 performs maintenance on a Train related basis during shutdown periods. It is intended that the loss of the operating (operable) train is to be considered. If this loss results in the inability to maintain Cold Shutdown, then the escalation to an Alert will be in accordance with CA3.

The minimum voltage necessary, based on plant design, is 105 volts; however, the lowest battery voltage attained on a loss of off site power at the end of the 4 hour period is 107.4 volts on the 'B' battery bank. 108 volts is used for the EAL indication because the Control Room instrumentation reads in 2 volt increments. Reference calculations ECE91-058, "Battery 3A-S "A Train" Calculation for Station Blackout" and ECE91-059, "Battery 3B-S "B Train" Calculation for Station Blackout."

COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

CU7

Initiating Condition – NOTIFICATION OF UNUSUAL EVENT

Inadvertent criticality.

Operating Mode Applicability: **Cold Shutdown (Mode 5)**
Refueling (Mode 6)

Emergency Action Level(s):

1. An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

Basis:

This IC addresses criticality events that occur in Cold Shutdown or Refueling modes (NUREG 1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States) such as fuel misloading events and inadvertent dilution events. This IC indicates a potential degradation of the level of safety of the plant, warranting a NOUE classification. This IC excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups (e.g., criticality earlier than estimated) which are addressed in the companion SU10.

This condition can be identified using the startup rate meter. If the startup rate meter is not in service in Mode 6, then the neutron count rate is used for this EAL. The term "sustained" is used in order to allow exclusion of expected short term positive startup rates from planned fuel bundle or control rod movements during core alteration. These short term positive startup rates are the result of the increase in neutron population due to subcritical multiplication.

Escalation would be by Emergency Coordinator Judgment.

COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

CU8

Initiating Condition – NOTIFICATION OF UNUSUAL EVENT

UNPLANNED loss of all onsite or offsite communications capabilities.

Operating Mode Applicability: **Cold Shutdown (Mode 5)**
Refueling (Mode 6)

Emergency Action Level(s): (1 or 2)

1. Loss of all Table C1 onsite communications systems affecting the ability to perform routine operations.

OR

2. Loss of all Table C2 offsite communications systems.

Basis:

The purpose of this IC and its associated EALs is to recognize a loss of communications capability that either defeats the plant operations staff ability to perform routine tasks necessary for plant operations or the ability to communicate problems to offsite authorities. The loss of offsite communications ability is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72. The availability of one method of ordinary offsite communications is sufficient to inform state and local authorities of plant problems. This EAL is intended to be used only when extraordinary means (e.g., relaying of information from radio transmissions, individuals being sent to offsite locations, etc.) are being utilized to make communications possible.

Table C1 Onsite Communications Equipment
Plant radio system Plant paging system In-plant telephones Sound powered phones

Table C2 Offsite Communications Equipment
All telephone lines (commercial and microwave) Industrial Hot Line ENS Civil Defense Radios Operational Hotline

COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

CA1

Initiating Condition – ALERT

Loss of RCS inventory.

Operating Mode Applicability: **Cold Shutdown (Mode 5)**

Emergency Action Level(s): (1 or 2)

1. Loss of RCS inventory as indicated by RVLMS upper plenum level ≤ 20%.

OR

2. a. Loss of RCS inventory as indicated by unexplained containment sump level or reactor drain tank level rise

AND

b. RCS level cannot be monitored > 15 minutes

Basis:

These EALs serve as precursors to a loss of ability to adequately cool the fuel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further reactor vessel level decrease and potential core uncovery. This condition will result in a minimum classification of Alert. The Reactor Vessel Level Monitoring System (RVLMS) provides a reading in percentage level remaining in the upper plenum. Procedure OP-001-003, Reactor Coolant System Drain Down, Attachment 11.4 lists the RVLMS sensing element elevations. The area corresponding to 20 % level is at 11.80 ft. MSL (bottom ID of RCS loop determined to be 11.625 ft. MSL from basis for CA2). Therefore a level equal to or below 20% indicates that level has dropped to an area at (or below) the low point of the RCS loop. The inability to restore and maintain level after reaching this setpoint would therefore be indicative of a failure of the RCS barrier.

In cold shutdown the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode. Entry into cold shutdown conditions may be attained within hours of operating at power or hours after refueling is completed. Entry into the refueling mode procedurally may not occur for typically 100 hours or longer after the reactor has been shutdown. Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the reactor vessel (note that the heatup threat could be lower for cold shutdown conditions if the entry into cold shutdown was following a refueling). The above forms the basis for needing both a cold shutdown specific IC (CA1) and a refueling specific IC (CA2).

COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

CA1

In the cold shutdown mode, normal RCS level and reactor vessel level instrumentation systems will normally be available. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that reactor vessel inventory loss was occurring by observing sump and tank level changes. Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage. The 15-minute duration for the loss of level indication was chosen because it is half of the CS1 Site Area Emergency EAL duration. The 15-minute duration allows CA1 to be an effective precursor to CS1. Significant fuel damage is not expected to occur until the core has been uncovered for greater than 1 hour in accordance with the analysis referenced in the CS1 basis. Therefore this EAL meets the definition for an Alert emergency.

The difference between CA1 and CA2 deals with the RCS conditions that exist between cold shutdown and refueling mode applicability. In cold shutdown, the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the refueling mode, the RCS is not intact and reactor vessel level and inventory are monitored by different means.

If reactor vessel level continues to decrease, then escalation to Site Area Emergency will be via CS1.

COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

CA2

Initiating Condition – ALERT

Loss of reactor vessel inventory with irradiated fuel in the reactor vessel.

Operating Mode Applicability: **Refueling (Mode 6)**

Emergency Action Level(s): (1 or 2)

1. Loss of reactor vessel inventory as indicated by reactor vessel level at 12 ft.

OR

2. a. Loss of reactor vessel inventory as indicated by unexplained containment sump level or reactor drain tank level rise

AND

b. Reactor vessel level cannot be monitored > 15 minutes

Basis:

These EALs serve as precursors to a loss of heat removal. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further reactor vessel level decrease and potential core uncovery. The bottom ID of the RCS Loop is chosen for this IC because at this level remote RCS level indication is lost and loss of normal suction for the shutdown cooling system will occur below this point. RVLMS is not used as an indicator for this EAL because it is not expected to be in service in Mode 6. The bottom ID of the RCS loop is determined to be at 11.625 ft. MSL by the following: The centerline elevation of the RCS hot leg is 13.375' MSL (from drawing 1564-G146), the hot leg piping inside diameter is 42" (from UFSAR section 5.4.3.2), therefore 13.375' – 21" = bottom ID of RCS loop = elevation 11.625' MSL. Other reactor vessel level monitoring systems for mode 6 provide lowest indication in the Control Room at 12.0 ft. MSL (from RCS System Description SD-RCS). Thus the level corresponding to a loss of suction to decay heat removal systems (bottom ID of the RCS loop) for upgrade to an Alert is taken to be 12 ft. MSL for this IC. The inability to restore and maintain level after reaching this setpoint would therefore be indicative of a failure of the RCS barrier.

In cold shutdown the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode. Entry into cold shutdown conditions may be attained within hours of operating at power or hours after refueling is completed. Entry into the refueling mode procedurally may not occur for typically 100 hours or longer after the reactor has been shutdown.

Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the reactor vessel (note that the heatup threat could be lower for cold shutdown conditions if the entry into cold shutdown was following a refueling). The above forms the basis for needing both a cold shutdown specific IC (CA1) and a refueling specific IC (CA2).

In the refueling mode, normal means of reactor vessel level indication may not be available. Redundant means of reactor vessel level indication will be normally installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that reactor vessel inventory loss was occurring by observing sump and tank level changes. Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage. The 15-minute duration for the loss of level indication was chosen because it is half of the CS2 Site Area Emergency EAL duration. The 15-minute duration allows CA2 to be an effective precursor to CS2. Significant fuel damage is not expected to occur until the core has been uncovered for greater than 1 hour in accordance with the analysis referenced in the CS2 basis. Therefore this EAL meets the definition for an Alert.

The difference between CA1 and CA2 deals with the RCS conditions that exist between cold shutdown and refueling mode applicability. In cold shutdown, the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the refueling mode, the RCS is not intact and reactor vessel level and inventory are monitored by different means.

If reactor vessel level continues to decrease, then escalation to Site Area Emergency will be via CS2.

COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

CA3

Initiating Condition – ALERT

Inability to maintain plant in Cold Shutdown with irradiated fuel in the reactor vessel.

Operating Mode Applicability: **Cold Shutdown (Mode 5)**
Refueling (Mode 6)

Emergency Action Level(s): (1 or 2 or 3)

1. With CONTAINMENT CLOSURE and RCS integrity not established, an UNPLANNED event results in RCS temperature exceeding the Technical Specification cold shutdown temperature limit.

OR

2. With CONTAINMENT CLOSURE established and RCS integrity not established, or RCS inventory reduced, an UNPLANNED event results in RCS temperature exceeding the Technical Specification cold shutdown temperature limit for > 20 minutes[1].

OR

3. An UNPLANNED event results in RCS temperature exceeding the Technical Specification cold shutdown temperature limit for > 60 minutes[1] or results in an RCS pressure rise of > 10 psig.

*[1]Note: **If** shutdown cooling is in operation within this time frame **and** RCS temperature is being reduced **then** this EAL is **not** applicable.*

Basis:

This IC and its associated EALs are based on concerns raised by Generic Letter 88-17, "Loss of Decay Heat Removal." A number of phenomena such as pressurization, vortexing, steam generator U-tube draining, RCS level differences when operating at a mid-loop condition, decay heat removal system design, and level instrumentation problems can lead to conditions where decay heat removal is lost and core uncovery can occur. NRC analyses show sequences that can cause core uncovery in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost.

A loss of Technical Specification components alone is not intended to constitute an Alert. The same is true of a momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available.

COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

CA3

The Emergency Coordinator/EOF Director must remain alert to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the Emergency Coordinator/EOF Director, an imminent situation is at hand, then the classification should be made as if the threshold has been exceeded.

EAL 1 addresses complete loss of functions required for core cooling during refueling and cold shutdown modes when **neither** CONTAINMENT CLOSURE **nor** RCS integrity are established. RCS integrity is in place when the RCS pressure boundary is in its normal condition to be pressurized (e.g., no freeze seals or nozzle dams). No delay time is allowed for EAL1 because the evaporated reactor coolant that may be released into the Containment during this heatup condition could also be directly released to the environment.

EAL 2 addresses the complete loss of functions required for core cooling for > 20 minutes during refueling and cold shutdown modes when CONTAINMENT CLOSURE is established **but** RCS integrity is **not** established **or** RCS inventory is reduced (e.g., mid loop operation). As in EAL 1, RCS integrity should be assumed to be in place when the RCS pressure boundary is in its normal condition to be pressurized (e.g., no freeze seals or nozzle dams). The allowed 20 minute time frame was included to allow operator action to restore the heat removal function, if possible. The allowed time frame is consistent with the guidance provided by Generic Letter 88-17, "Loss of Decay Heat Removal" and is believed to be conservative given that a low pressure Containment barrier to fission product release is established. Note 1 indicates that EAL 2 is **not** applicable if actions are successful in restoring an RCS heat removal system to operation **and** RCS temperature is being reduced within the 20 minute time frame.

EAL 3 addresses complete loss of functions required for core cooling for > 60 minutes during refueling and cold shutdown modes when RCS integrity is established. As in EAL 1 and 2, RCS integrity should be considered to be in place when the RCS pressure boundary is in its normal condition to be pressurized (e.g., no freeze seals or nozzle dams). The status of CONTAINMENT CLOSURE in this EAL is immaterial given that the RCS is providing a high pressure barrier to fission product release to the environment. The 60 minute time frame should allow sufficient time to restore cooling without there being a substantial degradation in plant safety. The 10 psig pressure increase covers situations where, due to high decay heat loads, the time provided to restore temperature control should be less than 60 minutes. The RCS pressure setpoint can be read on installed control board instrumentation. Note 1 indicates that EAL 3 is **not** applicable if actions are successful in restoring a shutdown cooling system to operation **and** RCS temperature is being reduced within the 60 minute time frame **assuming that the RCS pressure increase has remained LESS THAN 10 psig.**

Escalation to Site Area Emergency would be via CS1 or CS2 should boiling result in significant reactor vessel level loss leading to core uncovery.

COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

CA5

Initiating Condition – ALERT

Loss of all offsite power and loss of all onsite AC power to essential busses.

Operating Mode Applicability: **Cold Shutdown (Mode 5)**
Refueling (Mode 6)
Defueled

Emergency Action Level(s):

1. a. Loss of power to all unit auxiliary and startup transformers

AND

b. Failure of the 'A' and 'B' emergency diesel generators to supply power to emergency busses

AND

c. Failure to restore power to at least one emergency bus within 15 minutes from the time of loss of both offsite and onsite AC power.

Basis:

Loss of all AC power compromises all plant safety systems requiring electric power including shutdown cooling, emergency core cooling, containment cooling, spent fuel pool cooling and the ultimate heat sink. When in cold shutdown, refueling, or defueled mode the event can be classified as an Alert, because of the significantly reduced decay heat and lower temperature and pressure which allow increasing the time to restore one of the emergency busses, relative to that specified for the Site Area Emergency EAL. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. Escalating to Site Area Emergency, if appropriate, is by Abnormal Radiation Levels / Radiological Effluents (AS1), or Emergency Coordinator/EOF Director Judgment EALs.

Consideration should be given to available loads necessary to remove decay heat or provide reactor vessel makeup capability when evaluating loss of AC power to essential busses. Even though an essential bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or reactor vessel makeup capability) are not operable on the energized bus, then the bus should not be considered available.

Credit for Temporary Emergency Diesel Generators (TEDs) may **NOT** be taken because they are not a credited power source in the Technical Specifications for modes 5 and 6.

COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

CS1

Initiating Condition – SITE AREA EMERGENCY

Loss of reactor vessel inventory affecting core decay heat removal capability.

Operating Mode Applicability: **Cold Shutdown (Mode 5)**

Emergency Action Level(s): (1 or 2)

1. With CONTAINMENT CLOSURE not established:

   a. Reactor vessel inventory as indicated by RVLMS upper plenum level 0%.

OR

   b. Reactor vessel level cannot be monitored > 30 minutes with a loss of reactor vessel inventory as indicated by unexplained containment sump level or reactor drain tank level rise.

OR

2. With CONTAINMENT CLOSURE established:

   Reactor vessel level cannot be monitored > 30 minutes with a loss of reactor vessel inventory as indicated by either:
   - Unexplained containment sump or reactor drain tank level rise.
   - Erratic Source Range Monitor indication.

Basis:

Under the conditions specified by this IC, continued decrease in reactor vessel level is indicative of a loss of inventory control. Inventory loss may be due to a reactor vessel breach, pressure boundary leakage, or continued boiling in the reactor vessel.

In cold shutdown the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode . Entry into cold shutdown conditions may be attained within hours of operating at power or hours after refueling is completed. Entry into the refueling mode procedurally may not occur for typically 100 hours or longer after the reactor has been shutdown. Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the reactor vessel (note that the heatup threat could be lower for cold shutdown conditions if the entry into cold shutdown was following a refueling). The above forms the basis for needing both a cold shutdown specific IC (CS1) and a refueling specific IC (CS2).

COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

CS1

In the cold shutdown mode, normal RCS level and reactor vessel level indication systems will normally be available. However, If all reactor vessel level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that reactor vessel inventory loss was occurring by observing containment sump level or reactor drain tank level changes. Containment sump level or reactor drain tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

These EALs are based on concerns raised by Generic Letter 88-17, *Loss of Decay Heat Removal*, SECY 91-283, *Evaluation of Shutdown and Low Power Risk Issues*, NUREG-1449, *Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States*, and, NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management.* A number of variables, (mid-loop, reduced level/flange level, head in place, or cavity flooded, RCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining) can have a significant impact on heat removal capability challenging the fuel clad barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery therefore, conservatively, 30 minutes was chosen.

The Reactor Vessel Level Monitoring System (RVLMS) provides a reading in percentage level remaining in the upper plenum. A 0% level is the first observable point below 6" below the bottom ID of the RCS loop penetration in the reactor vessel (NEI 99-01 guidance) but is at a point higher than the Top of Active Fuel (TOAF) at its location 12.6" above the fuel alignment plate (from RCS System Description SD-RCS). Procedure OP-001-003, Reactor Coolant System Drain Down, Attachment 11.4 lists the RVLMS sensing element elevations. The area corresponding to 0% level is at 10.10 ft. MSL (bottom ID of RCS loop determined to be 11.625 ft. MSL from basis for CA2 with 6" below that point at 11.125 ft. MSL). Therefore a 0% level indicates that level has dropped to an area at (or below) the low point of the RCS loop. The inability to restore and maintain level after reaching this setpoint would therefore be indicative of a failure of the RCS barrier.

The 30-minute duration allowed when CONTAINMENT CLOSURE is established allows sufficient time for actions to be performed to recover needed cooling equipment and is considered to be conservative given that level is being monitored via CS1 and CS2. Effluent release is not expected with closure established.

Escalation to a General Emergency is via CG1 or radiological effluent IC AG1.

COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

CS2

Initiating Condition – SITE AREA EMERGENCY

Loss of reactor vessel inventory affecting core decay heat removal capability with irradiated fuel in the reactor vessel.

Operating Mode Applicability: **Refueling (Mode 6)**

Emergency Action Level(s):

1. Reactor vessel level cannot be monitored WITH indication of core uncovery as evidenced by one or more of the following:

- Containment High Range Radiation Monitor (ARM-IRE-5400AS or ARM-IRE-5400BS) ≥ 10R/hr
- Erratic Source Range Monitor indication
- Core Exit Thermocouples indicate superheat

Basis:

Under the conditions specified by this IC, continued decrease in reactor vessel level is indicative of a loss of inventory control. Inventory loss may be due to a reactor vessel breach or continued boiling in the reactor vessel.

In cold shutdown the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode Entry into cold shutdown conditions may be attained within hours of operating at power or hours after refueling is completed. Entry into the refueling mode procedurally may not occur for typically 100 hours or longer after the reactor has been shutdown. Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the reactor vessel (note that the heatup threat could be lower for cold shutdown conditions if the entry into cold shutdown was following a refueling). The above forms the basis for needing both a cold shutdown specific IC (CS1) and a refueling specific IC (CS2).

These example EALs are based on concerns raised by Generic Letter 88-17, Loss of Decay Heat Removal, SECY 91-283, Evaluation of Shutdown and Low Power Risk Issues, NUREG-1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States, and, NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management. A number of variables, (e.g., mid-loop, reduced level/flange level, head in place, or cavity flooded, RCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining) can have a significant impact on heat removal capability challenging the fuel clad barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery therefore, conservatively, 30 minutes was chosen.

COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

CS2

Normal means of reactor vessel level indication may not be available. Redundant means of reactor vessel level indication will be normally installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted.

RVLMS is not used as an indicator for this EAL because it is not expected to be in service in Mode 6. Other reactor vessel level monitoring systems for mode 6 provide lowest indication at 12.0 ft. MSL which is slightly above the bottom ID of the RCS loop penetration to the reactor vessel. Therefore, an indication that the water level has dropped to any point below the bottom of the RCS loop penetration in the reactor vessel is not available in mode 6 and an EAL is selected that uses inability to monitor reactor vessel level.

As water level in the reactor vessel lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in up-scaled Containment High Range Monitor indication and possible alarm. A reading of greater than or equal to 10 R/hr may be indicative of fuel damage. The basis for 10 R/hr is that it is sufficiently above the normal indication of 0.74 R/hr (nominal shutdown) to avoid an unnecessary entry into the EAL but substantially lower than the calculated values for RCS barrier failure (100 R/hr) and fuel clad barrier failure (1000 R/hr) for barrier losses in Section F(Fission Product Barrier) to give an early indication of vessel level lowering to the point of potential fuel damage. The 10 R/hr is also high enough to be indicative of potential fuel uncovery. Additionally, post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations.

Effluent release is not expected if containment closure is established.

Escalation to a General Emergency is via CG1 or radiological effluent IC AG1.

COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

CG1

Initiating Condition – GENERAL EMERGENCY

Loss of reactor vessel inventory affecting fuel clad integrity with containment challenged with irradiated fuel in the reactor vessel.

Operating Mode Applicability: **Cold Shutdown (Mode 5)**
Refueling (Mode 6)

Emergency Action Level(s): (1 and 2 and 3)

1. Loss of reactor vessel inventory as indicated by unexplained containment sump level or reactor drain tank level rise

AND

2. Reactor vessel level cannot be monitored with indication of core uncovery > 30 minutes as evidenced by one or more of the following:
   - Containment High Range Radiation Monitor (ARM-IRE-5400AS or ARM-IRE-5400BS) ≥ 10R/hr
   - Erratic Source Range Monitor indication
   - Core Exit Thermocouples indicate superheat

AND

3. Indication of CONTAINMENT challenged as indicated by one or more of the following:
   - Explosive mixture inside containment
   - Containment pressure > 50 PSIA
   - CONTAINMENT CLOSURE not established

COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

CG1

Basis:

During an outage, installed RCS level and REACTOR VESSEL level instrumentation systems will normally be available when the RCS is filled and redundant means of REACTOR VESSEL level indication will be normally installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted when the RCS is not filled. EAL #1 assumes, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that REACTOR VESSEL inventory loss was occurring by observing sump and tank level changes. Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

EAL 2 represents the inability to restore and maintain reactor vessel level to above the top of active fuel. Fuel damage is probable if reactor vessel level cannot be restored, as available decay heat will cause boiling, further reducing the reactor vessel level.

These EALs are based on concerns raised by Generic Letter 88-17, *Loss of Decay Heat Removal*, SECY 91-283, *Evaluation of Shutdown and Low Power Risk Issues*, NUREG-1449, *Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States*, and, NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management.* A number of variables, (e.g., mid-loop, reduced level/flange level, head in place, or cavity flooded, RCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining) can have a significant impact on heat removal capability challenging the fuel clad barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery therefore, conservatively, 30 minutes was chosen.

As water level in the reactor vessel lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in up-scaled Containment High Range Monitor indication. The basis for 10 R/hr is that it is sufficiently above the normal indication of 0.74 R/hr (nominal shutdown) to avoid an unnecessary entry into the EAL but substantially lower than the calculated values for RCS barrier failure (100 R/hr) and fuel clad barrier failure (1000 R/hr) for barrier losses in Section F (Fission Product Barrier) to give an early indication of vessel level lowering to the point of potential fuel damage. The 10 R/hr is also high enough to be indicative of potential fuel uncovery. Additionally, post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations.

COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

CG1

The GE is declared on the occurrence of the loss or imminent loss of function of all three barriers. Based on the above discussion, RCS barrier failure resulting in core uncovery for 30 minutes or more may cause fuel clad failure. With the CONTAINMENT breached or challenged then the potential for unmonitored fission product release to the environment is high. This represents a direct path for radioactive inventory to be released to the environment. This is consistent with the definition of a GE.

In the context of EAL 3, CONTAINMENT CLOSURE is the action taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant (shutdown) conditions. Site shutdown contingency plans provide for re-establishing CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory functions. If the closure is re-established prior to exceeding the temperature or level thresholds of the RCS Barrier and Fuel Clad Barrier EALs, then escalation to GE would **not** occur.

The pressure at which CONTAINMENT is considered challenged is based on the condition of the CONTAINMENT. If the CONTAINMENT is fully intact, then the CONTAINMENT will be challenged at the design pressure of 44 psig (~59 psia). Because the EOPs use 50 psia as a safety function parameter following a LOCA, this is the value used in the EAL. This is consistent with the owner's groups Emergency Response Procedures. If CONTAINMENT CLOSURE is established, the EAL setpoint is based on an estimate of the pressure CONTAINMENT CLOSURE would be able to sustain. Waterford estimates this pressure to be the design pressure because of the closure actions taken.

In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive mixture of dissolved gasses in CONTAINMENT. However, CONTAINMENT monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that an explosive mixture exists. Existence of an explosive mixture means a hydrogen and oxygen concentration of at least the lower deflagration limit curve exists (refer to Severe Accident Management Guidelines Calculation Aid #7, Containment Challenge due to Hydrogen Combustion).

FISSION PRODUCT BARRIER DEGRADATION

FISSION PRODUCT BARRIER DEGRADATION

FU1 – Initiating Condition – NOTIFICATION OF UNUSUAL EVENT

ANY loss or ANY Potential Loss of Containment.

Operating Mode Applicability:

Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

FA1 – Initiating Condition – Alert

ANY loss or ANY Potential Loss of EITHER Fuel Clad or RCS

Operating Mode Applicability:

Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

FS1 – Initiating Condition – Site Area Emergency

Loss or Potential Loss of ANY two Barriers

Operating Mode Applicability:

Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

FG1 – Initiating Condition – General Emergency

Loss of ANY two Barriers AND Loss or Potential Loss of Third barrier

Operating Mode Applicability:

Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

FISSION PRODUCT BARRIER DEGRADATION

General Bases:

The logic used for these Initiating Conditions reflects the following considerations:

- The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Containment Barrier. Unusual Event ICs associated with RCS and Fuel Clad Barriers are addressed under System Malfunction (S) ICs.

- At the Site Area Emergency level, there must be some ability to dynamically assess how far present conditions are from the threshold for a General Emergency. For example, if Fuel Clad and RCS Barrier "Loss" EALs existed, that, in addition to offsite dose assessments, would require continual assessments of radioactive inventory and containment integrity. Alternatively, if both Fuel Clad and RCS "Potential Loss" EALs existed, the Emergency Coordinator/EOF Director would have more assurance that there was no immediate need to escalate to a General Emergency.

- The ability to escalate to higher emergency classes as an event deteriorates must be maintained. For example, RCS leakage steadily increasing would represent an increasing risk to public health and safety.

  a. Fission Product Barrier ICs must be capable of addressing event dynamics. Thus, the Note associated with the ICs in the EAL Matrix provides guidance that imminent (i.e., within 1 to 2 hours) Loss or Potential Loss should result in a classification as if the affected threshold(s) are already exceeded, particularly for the higher emergency classes.

FISSION PRODUCT BARRIER DEGRADATION

Fuel Clad Barrier Emergency Action Levels:

The Fuel Clad Barrier is the zircalloy or stainless steel tubes that contain the fuel pellets.

Primary Coolant Activity Level (FCB1)

Loss: RCS Dose Equivalent Iodine > 300 µCi/gm as indicated by:

a. Dose Rate at one foot from Primary Sample Panel > 950 mR/hr

OR

b. -4 RAB RADIOCHEMISTRY LAB area radiation monitor (ARM-IRE-5020) > 125 mR/hr

OR

c. Chemistry sample results

Potential Loss: Not Applicable

Basis:

The radiation monitor values given are assumed valid when the primary sample panel valves are open receiving flow from the RCS.

The radiation monitor values were determined by calculating various coolant radionuclide concentrations postulated to result from a 10% gap inventory release at Waterford 3. This alternate method to PASS sampling of determining fuel degradation was developed in HP-CALC-2001-001, PASS Elimination and accepted by NRC when Waterford 3 eliminated the PASS. This amount of coolant activity is well above that expected for iodine spikes and corresponds to less than 5% fuel clad damage. This amount of radioactivity indicates significant clad damage and thus the Fuel Clad Barrier is considered lost.

There is no equivalent “Potential Loss” EAL for this item.

FISSION PRODUCT BARRIER DEGRADATION

Fuel Clad Barrier Emergency Action Levels:

Core Exit Thermocouple Readings (FCB2)

Loss: Core Exit Thermocouple readings ≥ 1200 degrees F

Potential Loss: Core Exit Thermocouple readings ≥ 700 degrees F

Basis:

The Loss EAL of ≥ 1200º F is consistent with the NEI 99-01. The elevated temperature corresponds to significant superheating of the coolant and is indicative of a loss of the Fuel Clad Barrier. Other references (EC-S98-001, "EOP Action Value Bases" and CE-NPSD-241, "Development of the Comprehensive Procedure Guideline for Core Damage Assessment," Task 467) indicate that clad rupture due to high temperature is not expected for CET temperature readings of less than 1200º F.

The Potential Loss setpoint of CET temperatures ≥ 700º F is consistent with Emergency Operating Procedures (EOPs) and is used as an indication of a loss of subcooling conditions in the RCS. It is consistent with the criteria developed in NEI 99-01. The elevated temperature corresponds to a loss of subcooling and is indicative of a Potential Loss of the Fuel Clad Barrier. This criteria is supplemented by further plant specific criteria for diagnosis of loss of subcooling given in Potential Loss EAL FCB3.

FISSION PRODUCT BARRIER DEGRADATION

Fuel Clad Barrier Emergency Action Levels:

Reactor Vessel Water Level (FCB3)

Loss: Not Applicable

Potential Loss: RVLMS upper plenum level 0%.

Basis:

There is no "Loss" EAL corresponding to this item because it is better covered by the other Fuel Clad Barrier "Loss" EALs.

As part of its Inadequate Core Cooling Instrumentation, Waterford 3 uses a Reactor Vessel Level Monitoring System (RVLMS) that is displayed to the operators and can measure water level from near the top of the active fuel. The lowest point where monitoring is provided in this system is 12.6" above the fuel alignment plate. This monitoring point is equal to 0% upper plenum RVLMS level. This is consistent with the EOPs as follows: The Waterford 3 EOPs, in OP-902-008, Functional Recovery, use an acceptance criteria for RCS and core heat removal of RVLMS upper plenum level $\geq$ 20%. If the level is below 20%, then contingency actions must be taken and the criterion is considered not met. The next discrete measurement point below 20% upper plenum level is 0% level.

FISSION PRODUCT BARRIER DEGRADATION

Fuel Clad Barrier Emergency Action Levels:

Containment Radiation Monitoring (FCB4)

Loss: Containment High Range Radiation Monitor (ARM-IRE-5400AS or ARM-IRE-5400BS) > 1000 R/hr.

Potential Loss: Not Applicable

Basis:

This reading is a value which indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the containment. The reading is calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 µCi/gm dose equivalent I-131 into the containment atmosphere. Reference Waterford 3 Engineering Calculation EC-S03-008. Source documents are HP-CALC-93-005, "Containment Atmosphere Radiation Monitor Setpoint Calculation," NUREG 1228, "Source Term Estimation During Incident Response to Severe Nuclear Power Plant Accidents" and EC-S98-002, "Waterford 3 Chapter 15 Non-LOCA Dose Calculation." It assumes normal (NUREG 1228) gas gap fractions, leak into RCS and then into containment, and containment spray initiation impact. Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within Technical Specifications and are therefore indicative of fuel damage. This radiation monitor value is higher than that specified for RCS barrier Loss EAL **RCB3**. Thus, this EAL indicates a loss of **both** the fuel clad barrier **and** a loss of the RCS barrier.

There is no "Potential Loss" EAL associated with this item.

FISSION PRODUCT BARRIER DEGRADATION

Fuel Clad Barrier Emergency Action Levels:

Emergency Coordinator/EOF Director Judgment (FCB5)

Any condition in the opinion of the Emergency Coordinator/EOF Director that indicates Loss or Potential Loss of the Fuel Clad Barrier.

Basis:

This EAL addresses any other factors that are to be used by the Emergency Coordinator/EOF Director in determining whether the Fuel Clad barrier is lost or potentially lost. An event or multiple events could occur which result in the conclusion that exceeding the loss or potential loss thresholds is imminent (i.e., within 1 to 2 hours). In this imminent loss situation, use judgment and classify as if the thresholds are exceeded. In addition, the inability to monitor the barrier is also incorporated in this EAL as a factor in Emergency Coordinator/EOF Director judgment that the barrier may be considered lost or potentially lost. *(See also SG1, "Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power", for additional information.)*

FISSION PRODUCT BARRIER DEGRADATION

RCS Barrier Emergency Action Levels:

The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.

RCS Leak Rate (RCB1)

Loss: RCS leak rate GREATER THAN available makeup capacity as indicated by RCS subcooling < 28º F.

Potential Loss: Unisolable RCS leak > 44 gpm.

Basis:

The “Loss” EAL addresses conditions where leakage from the RCS is greater than available inventory control capacity such that a loss of subcooling has occurred. The loss of subcooling is the fundamental indication that the inventory control systems are inadequate in maintaining RCS pressure and inventory against the mass loss through the leak.

The “Potential Loss” EAL is based on the inability to maintain normal liquid inventory within the RCS by normal operation of the Chemical and Volume Control System which is considered as one charging pump discharging to the charging header. A second charging pump being required is indicative of a substantial RCS leak.

FISSION PRODUCT BARRIER DEGRADATION

RCS Barrier Emergency Action Levels:

SG Tube Rupture (RCB2)

Loss: SGTR that results in an ECCS (SI) actuation

Potential Loss: Not Applicable

Basis:

This EAL is intended to address the full spectrum of Steam Generator (SG) tube rupture events in conjunction with Containment Barrier "Loss" EAL **CNB3** and Fuel Clad Barrier EALs. The "Loss" EAL addresses RUPTURED SG(s) for which the leakage is large enough to cause actuation (either automatic or manual) of ECCS (SI) This is consistent to the RCS Barrier "Potential Loss" EAL **RCB1**. By itself, this EAL will result in the declaration of an Alert. However, if the SG is also FAULTED (i.e., two barriers failed), the declaration escalates to a Site Area Emergency in accordance with Containment Barrier "Loss" EAL **CNB3**.

There is no "Potential Loss" EAL.

FISSION PRODUCT BARRIER DEGRADATION

RCS Barrier Emergency Action Levels:

Containment Radiation Monitoring (RCB3)

Loss: Containment High Range Radiation Monitor (ARM-IRE-5400AS or ARM-IRE-5400BS) > 100 R/hr.

Potential Loss: Not Applicable

Basis:

The specific radiation monitor reading is a value which indicates the release of reactor coolant to the containment. The reading was calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within Technical Specifications) into the containment atmosphere. Reference Waterford 3 Engineering Calculation EC-S03-008. Source documents used for the determination of this monitor reading are NUREG 1228, "Source Term Estimation During Incident Response to Severe Nuclear Power Plant Accidents" and EC-S98-002, "Waterford 3 Chapter 15 Non-LOCA Dose Calculation." This reading is less than that specified for Fuel Clad Barrier EAL **FCB4**. Thus, this EAL is indicative of a RCS leak only. If the radiation monitor reading increased to that specified by Fuel Clad Barrier EAL **FCB4**, then fuel damage is indicated.

There is no "Potential Loss" EAL associated with this item.

FISSION PRODUCT BARRIER DEGRADATION

RCS Barrier Emergency Action Levels:

Other Indications (RCB4)

Loss: Not Applicable

Potential Loss: RCS pressure dropping due to primary relief not reseating

Basis:

The setpoint for the pressurizer code safety valves is 2500 psia +/- 3%. Their purpose is to provide RCS overpressure protection. The safety valves pass sufficient pressurizer steam to limit the RCS pressure to 2750 psia (110 % of design) following a complete loss of turbine generator load without simultaneous reactor trip. In the event of a primary relief valve lifting and not reseating the loss of mass inventory of the RCS is large enough to uncover the core in a short period of time.
Source document: Technical Specifications sections 3.4.2.1 and 3.4.2.2.

FISSION PRODUCT BARRIER DEGRADATION

RCS Barrier Emergency Action Levels:

Other Indications (RCB)

Emergency Coordinator/EOF Director Judgment (RCB5)

Any condition in the opinion of the Emergency Coordinator/EOF Director that indicates Loss or Potential Loss of the RCS Barrier.

Basis:

This EAL addresses any other factors that are to be used by the Emergency Coordinator/EOF Director in determining whether the RCS barrier is lost or potentially lost. An event or multiple events could occur which result in the conclusion that exceeding the loss or potential loss thresholds is imminent (i.e., within 1 to 2 hours). In this imminent loss situation, use judgment and classify as if the thresholds are exceeded. In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in Emergency Coordinator/EOF Director judgment that the barrier may be considered lost or potentially lost. *(See also SG1, "Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power", for additional information.)*

FISSION PRODUCT BARRIER DEGRADATION

Containment Barrier Emergency Action Levels:

The Containment Barrier includes the containment building, its connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side isolation valve.

Containment Pressure (CNB1)

Loss: Rapid unexplained drop following initial rise

OR

Containment parameters not consistent with LOCA conditions

Potential Loss:

Containment pressure 50 PSIA and rising

OR

Explosive mixture exists

OR

Containment pressure > 17.7 PSIA with LESS THAN one full train of Containment Spray operating (1750 gpm)

Basis:

Rapid unexplained loss of pressure (i.e., not attributable to containment spray or condensation effects) following an initial pressure increase indicates a loss of containment integrity. Containment pressure and sump levels should increase as a result of the mass and energy release into containment from a LOCA. Thus, sump level or pressure not increasing indicates containment bypass and a loss of containment integrity.

The Containment pressure used for potential loss of containment is based on the containment design pressure. Existence of an explosive mixture means a hydrogen and oxygen concentration of at least the lower deflagration limit curve exists (refer to Severe Accident Management Guidelines Calculation Aid #7, Containment Challenge due to Hydrogen Combustion). This EAL is primarily a discriminator between Site Area Emergency and General Emergency representing a potential loss of the third barrier following a LOCA.

FISSION PRODUCT BARRIER DEGRADATION

Containment Barrier Emergency Action Levels:

Containment Pressure (CNB1)

The last potential loss EAL represents a potential loss of containment in that the containment heat removal/depressurization system (Containment Spray, but not including containment venting strategies) are either lost or performing in a degraded manner, as indicated by containment pressure greater than the setpoint at which the equipment was supposed to have actuated or Containment Spray pump providing LESS THAN 1750 gpm flow. Credit is not taken for Containment Fan Coolers in this EAL as mitigating Containment Spray losses.

FISSION PRODUCT BARRIER DEGRADATION

Containment Barrier Emergency Action Levels:

Core Exit Thermocouples (CNB2)

Loss: Not Applicable

Potential Loss:

Core exit thermocouples >1200 degrees F and restoration procedures not effective within 15 minutes

OR

Core exit thermocouples > 700 degrees F with RVLMS upper plenum level equal to 0% or LOWER and restoration procedures not effective within 15 minutes

Basis:

In this EAL, the functional restoration procedures are those emergency operating procedures that address the recovery of the core cooling critical safety functions. The procedure is considered effective if the temperature is decreasing **or** if the vessel water level is increasing.

Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation within the reactor vessel in a significant fraction of the core damage scenarios, and that the likelihood of containment failure is very small in these events. Given this, it is appropriate to provide a reasonable period to allow function restoration procedures to arrest the core melt sequence. Whether or not the procedures will be effective should be apparent within 15 minutes. The Emergency Coordinator/EOF Director should make the declaration as soon as it is determined that the procedures have been, or will be ineffective.

The conditions in this potential loss EAL represent an imminent core melt sequence which, if not corrected, could lead to vessel failure and an increased potential for containment failure. In conjunction with the Core Cooling and Heat Sink criteria in the Fuel and RCS barrier columns, this EAL would result in the declaration of a General Emergency – loss of two barriers and the potential loss of a third. If the function restoration procedures are ineffective, then there is no “success” path.

There is no “Loss” EAL associated with this item.

FISSION PRODUCT BARRIER DEGRADATION

Containment Barrier Emergency Action Levels:

SG Secondary Side Release With Primary to Secondary Leakage (CNB3)

Loss: Ruptured S/G is also faulted outside of containment

OR

Loss: Primary-to-Secondary leakrate >10 gpm with nonisolable steam release from affected S/G to the environment

Potential Loss: Not Applicable

Basis:

This "loss" EAL recognizes that SG tube leakage can represent a bypass of the containment barrier as well as a loss of the RCS barrier. The first "loss" EAL addresses the condition in which a RUPTURED (primary-to-secondary leakage of a magnitude sufficient to require or cause a reactor trip and safety injection) steam generator is also FAULTED (secondary side leakage that results in an uncontrolled decrease in steam generator pressure or the steam generator being completely depressurized). This condition represents a bypass of the RCS and containment barriers. In conjunction with RCS Barrier "loss" EAL **RCB2**, this would always result in the declaration of a Site Area Emergency.

The second "loss" EAL addresses SG tube leaks that exceed 10 gpm in conjunction with a nonisolable release path to the environment from the affected steam generator. The threshold for establishing the nonisolable secondary side release is intended to be a prolonged release of radioactivity from the RUPTURED steam generator directly to the environment. This could be expected to occur when the main condenser is unavailable to accept the contaminated steam (i.e., SGTR with concurrent loss of offsite power and the RUPTURED steam generator is required for plant cooldown or a stuck open relief valve or failed open atmospheric dump valve).

If the main condenser is available, then there may be releases via air ejectors, gland seal exhausters, and other similar controlled, and often monitored, pathways. Also, releases from the Steam Driven Emergency Feedwater Pump Turbine result in a very small minor release that can be isolated with the MS-401A and B valves. These pathways do **not** meet the intent of a nonisolable release path to the environment. These minor releases are assessed using Abnormal Radiation Levels / Radiological Effluents EALs.

A pressure boundary leakage of 10 gpm was used as the threshold in SU7 and is deemed appropriate for this EAL. For smaller breaks, not exceeding the normal

FISSION PRODUCT BARRIER DEGRADATION

Containment Barrier Emergency Action Levels:

SG Secondary Side Release With Primary to Secondary Leakage (CNB3)

charging capacity threshold in RCS Barrier "Potential Loss" EAL **RCB1** (RCS Leak Rate) or not resulting in ECCS actuation in EAL **RCB2** (SG Tube Rupture), this EAL results in a NOUE. For larger breaks, RCS barrier EALs **RCB1** and **RCB2** would result in an Alert. For SG tube ruptures which may involve multiple steam generators or unisolable secondary line breaks, this EAL would exist in conjunction with RCS barrier "Loss" EAL **RCB2** and would result in a Site Area Emergency. Escalation to General Emergency would be based on "Potential Loss" of the Fuel Clad Barrier.

FISSION PRODUCT BARRIER DEGRADATION

Containment Barrier Emergency Action Levels:

Containment Isolation Valve Status After Containment Isolation (CNB4)

Loss: Unisolable breach of containment with a direct release path to the environment following containment isolation actuation.

Potential Loss: Not Applicable

Basis:

This EAL is intended to address incomplete containment isolation that allows direct release to the environment. It represents a loss of the containment barrier.

The use of the modifier "direct" in defining the release path discriminates against release paths through interfacing liquid systems. The existence of an in–line charcoal filter does **not** make a release path indirect since the filter is not effective at removing fission noble gases. Typical filters have an efficiency of 95-99% removal of iodine. Given the magnitude of the core inventory of iodine, significant releases could still occur. In addition, since the fission product release would be driven by boiling in the reactor vessel, the high humidity in the release stream can be expected to render the filters ineffective in a short period. Therefore, a failure of a containment penetration in the annulus with frequent cycling of the shield building ventilation system (a filtered release path) meets the loss criteria of this EAL and constitutes a loss of the containment barrier.

There is no "Potential Loss" EAL associated with this item.

FISSION PRODUCT BARRIER DEGRADATION

Containment Barrier Emergency Action Levels:

Significant Radioactive Inventory in Containment (CNB5)

Loss: Not Applicable

Potential Loss: Containment High Range Radiation Monitor (ARM-IRE-5400AS or ARM-IRE-5400BS) > 4000 R/hr.

Basis:

The containment high range radiation monitor reading is a value which indicates significant fuel damage well in excess of the EALs associated with both loss of Fuel Clad and loss of RCS Barriers. A major release of radioactivity requiring offsite protective actions from core damage is not possible unless a major failure of fuel cladding allows radioactive material to be released from the core into the reactor coolant.

Regardless of whether containment is challenged, this amount of activity in containment, if released, could have such severe consequences that it is prudent to treat this as a potential loss of containment, such that a General Emergency declaration is warranted. Because the monitor reading exceeds the readings for Fuel Clad Barrier loss in **FCB4** and RCS Barrier loss in **RCB3**, the Emergency Coordinator/EOF Director should declare a General Emergency when this value on the Containment High Range Radiation Monitor is exceeded as a loss of two barriers (fuel clad and RCS) and potential loss of the third (containment). NUREG-1228, "Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents," indicates that such conditions do not exist when the amount of clad damage is less than 20%. The radiation monitor reading specified corresponds to approximately 20% fuel clad damage. Reference Waterford 3 Engineering Calculation EC-S03-008.

There is no "Loss" EAL associated with this item.

FISSION PRODUCT BARRIER DEGRADATION

Containment Barrier Emergency Action Levels:

Emergency Coordinator/EOF Director Judgment (CNB6)

Any condition in the opinion of the Emergency Coordinator/EOF Director that indicates Loss or Potential Loss of the Containment barrier.

Basis:

This EAL addresses any other factors that are to be used by the Emergency Coordinator/EOF Director in determining whether the Containment barrier is lost or potentially lost. An event or multiple events could occur which result in the conclusion that exceeding the loss or potential loss thresholds is imminent (i.e., within 1 to 2 hours). In this imminent loss situation, use judgment and classify as if the thresholds are exceeded. In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in Emergency Coordinator/EOF Director judgment that the barrier may be considered lost or potentially lost. *(See also SG1, "Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power", for additional information.)*

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU1

Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant.

Operating Mode Applicability: **All**

Emergency Action Level(s): (1 or 2 or 3)

1. A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by the Waterford 3 Security Shift Supervision

OR

2. A credible site specific security threat notification

OR

3. A validated notification from NRC providing information of an aircraft threat .

Basis:

NOTE: Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs.

Security events which do not represent a potential degradation in the level of safety of the plant are reported under 10 CFR 73.71 or in some cases under 10 CFR 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under HA1, HS1 and HG1 .

A higher initial classification could be made based upon the nature and timing of the security threat and potential consequences. Consideration shall be given to upgrading the emergency response status and emergency classification in accordance with the Safeguards Contingency Plan and Emergency Plan.

EAL #1

The Security Shift Supervisor is the designated individual on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Safeguards Contingency Plan.

This threshold is based on the Safeguards Contingency Plan. The Safeguards Contingency Plan is based on guidance provided in NEI 03-12.

EAL #2

This threshold is included to ensure that appropriate notifications for the security threat are made in a timely manner. This includes information of a credible threat. Only the plant to which the specific threat is made need declare the Notification of Unusual Event.

EAL #3

The intent of this EAL is to ensure that notifications for the aircraft threat are made in a timely manner and that Offsite Response Organizations and plant personnel are at a state of heightened awareness regarding the credible threat. It is not the intent of this EAL to replace existing non-hostile related EALs involving aircraft.

This EAL is met when a plant receives information regarding an aircraft threat from NRC. Validation is performed by calling the NRC or by other approved methods of authentication. Only the plant to which the specific threat is made need declare the Notification of Unusual Event.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an airliner (airliner is meant to be a large aircraft with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.

Escalation to Alert via HA1 would be appropriate if the threat involves an airliner within 30 minutes of the plant.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU2

Other conditions existing which in the judgment of the Emergency Coordinator warrant declaration of an Unusual Event.

Operating Mode Applicability: **All**

Emergency Action Level(s):

1. Other conditions exist which, in the judgment of the Emergency Coordinator, indicate that events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

Basis:

This EAL is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Coordinator to fall under the Unusual Event emergency class.

From a broad perspective, one area that may warrant Emergency Coordinator judgment is related to likely or actual breakdown of site-specific event mitigating actions. Examples to consider include inadequate emergency response procedures, transient response either unexpected or not understood, failure or unavailability of emergency systems during an accident in excess of that assumed in accident analysis, or insufficient availability of equipment and/or support personnel.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU4

FIRE within PROTECTED AREA boundary not extinguished within 15 minutes of detection.

Operating Mode Applicability: **All**

Emergency Action Level(s):

1. FIRE in or contiguous to Condensate Polisher Building, Containment, Fuel Handling Building, Reactor Auxiliary Building, Cooling Tower Areas or Turbine Building not extinguished within 15 minutes of Control Room notification or verification of a Control Room alarm.

Basis:

The purpose of this IC is to address the magnitude and extent of FIREs that may be potentially significant precursors to damage to safety systems. As used here, *Detection is* visual observation and report by plant personnel or sensor alarm indication. The 15-minute time period begins with a credible notification that a FIRE is occurring, or indication of a VALID fire detection system alarm. Verification of a fire detection system alarm includes actions that can be taken within the Control Room to ensure that the alarm is not spurious. A verified alarm is assumed to be an indication of a FIRE unless it is disproved within the 15-minute period by personnel dispatched to the scene. In other words, a personnel report from the scene may be used to disprove a sensor alarm if received within 15 minutes of the alarm, but shall not be required to verify the alarm.

The intent of this 15-minute duration is to size the FIRE and to discriminate against small FIREs that are readily extinguished (e.g., smoldering waste paper basket). The buildings listed are limited and ONLY include buildings and areas contiguous (in actual contact with or immediately adjacent) to plant VITAL AREAs or other significant buildings or areas. The intent of this EAL is not to include buildings (i.e., MSB, Service Building, Construction Support Building, Chiller Building, etc.) or areas that are not contiguous to plant VITAL AREAs. This IC excludes FIREs within administration buildings, waste-basket FIREs, and other small FIREs of no safety consequence.

Escalation to a higher emergency class is by IC HA4, “FIRE or EXPLOSION affecting the operability of plant safety systems required to establish or maintain safe shutdown.”

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU5

Release of toxic or flammable gases deemed detrimental to normal operation of the plant.

Operating Mode Applicability: **All**

Emergency Action Level(s): (1 or 2)

1. Report or detection of toxic or flammable gases that has or could enter the Exclusion Area Boundary in amounts that can affect NORMAL PLANT OPERATIONS.

OR

2. Report by St. Charles Parish for evacuation or sheltering of site personnel based on an offsite event.

Basis:

This IC is based on the existence of uncontrolled releases of toxic or flammable gas that may enter the EAB and affect normal plant operations. It is intended that releases of toxic or flammable gases are of sufficient quantity, and the release point of such gases is such that normal plant operations would be affected. This would preclude small or incidental releases, or releases that do not impact structures needed for plant operation. The EALs are intended to not require significant assessment or quantification. The EALs assume an uncontrolled process that has the potential to affect plant operations or personnel safety. Information from a neighboring plant provided over the Taft Industrial Complex Communication (TICC) radio in the Control Room meets the intent of the term "report" as used in EAL #1 and is considered to be information from a credible source.

Escalation is via HA5, which involves a quantified release of toxic or flammable gas affecting VITAL AREAS.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU6

Natural and destructive phenomena affecting the PROTECTED AREA

Operating Mode Applicability: **All**

Emergency Action Level(s): (1 or 2 or 3 or 4 or 5 or 6 or 7 or 8)

1. Earthquake felt in plant and detected on station seismic instrumentation.

OR

2. Report by plant personnel of tornado or high winds > 100 mph striking within PROTECTED AREA boundary.

OR

3. Vehicle crash into plant structures or systems within PROTECTED AREA boundary.

OR

4. Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment.

OR

5. Report of turbine failure resulting in casing penetration or damage to turbine or generator seals.

OR

6. Uncontrolled flooding in Reactor Auxiliary Building or Cooling Tower Areas that has the potential to affect safety related equipment needed for the current operating mode.

OR

7. Hurricane force winds (≥ 74 mph) expected to arrive on site in ≤12 hours as projected by the National Weather Service for a hurricane event

OR

8. River water level at the intake structure > +27 FT MSL.

Basis:

An Unusual Event in this IC would be declared on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators. Areas identified in the EALs define the location of the event based on the potential for damage to equipment contained therein. Escalation of the event to an Alert occurs when the magnitude of the event is sufficient to result in damage to equipment contained in the specified location.

EAL #1 is based on damage that may be caused to some portions of the site, but should not affect ability of safety functions to operate. The method of detection is based on instrumentation, validated by a reliable source, or operator assessment.

As defined in the EPRI sponsored "Guidelines for Nuclear Plant Response to an Earthquake", dated October 1989, a "felt earthquake" is: An earthquake of sufficient intensity such that: (a) the vibratory ground motion is felt at the nuclear plant site and recognized as an earthquake based on a consensus of control room operators on duty at the time, and (b) for plants with operable seismic instrumentation, the seismic switches of the plant are activated.

EAL #2 is based on the assumption that a tornado striking (touching down) or high winds within the PROTECTED AREA may have potentially damaged plant structures containing functions or systems required for safe shutdown of the plant. The high wind value in EAL#2 is based on the FSAR design basis 100 year recurrence interval projected wind velocity of 100 miles per hour. The actual site design basis for Seismic Class one structures is 200 mph. If damage is confirmed visually or by other plant indications, then the event may be escalated to Alert.

EAL #3 is intended to address crashes of vehicle types large enough to cause significant damage to plant structures containing functions and systems required for safe shutdown of the plant. Minor accidents involving smaller vehicles or golf carts where the potential for significant damage to site structures is not a concern or "fender bender" type accidents do **not** warrant declaration under this EAL. If the crash is confirmed to affect a plant VITAL AREA, the event may be escalated to Alert.

For EAL #4 only those EXPLOSIONs of sufficient force to damage permanent structures or equipment within the PROTECTED AREA should be considered. No attempt is made in this EAL to assess the actual magnitude of the damage. The occurrence of the EXPLOSION with reports of evidence of damage is sufficient for declaration. The Emergency Coordinator also needs to consider any security aspects of the EXPLOSION, if applicable.

EAL #5 is based on main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Of major concern is the potential for leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. Actual FIREs and flammable gas build up are appropriately classified via HU4 and HU5. Generator seal damage observed after generator purge does not meet the intent of this EAL because it did not impact normal operation of the plant. This EAL is consistent with the definition of a NOUE while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment. Escalation of the emergency classification is based on potential damage done by missiles generated by the failure or in conjunction with a steam generator tube rupture. The latter event would be classified by the radiological EALs or Fission Product Barrier EALs.

EAL #6 is based on the effect of flooding caused by internal events such as component failures, equipment misalignment, or outage activity mishaps. The areas noted include those areas that contain systems required for safe shutdown of the plant, and that are not designed to be wetted or submerged. Site specific areas containing functions and systems required for safe shutdown of the plant are taken from the Waterford 3 Post-Fire Safe Shutdown Analysis, EC-F00-026 for this EAL. These areas are reflected in FP-001-022, Design Change Fire Protection/Safe Shutdown Review. The Containment Building is not included in the EAL because of the guidance in the NEI 99-01 basis that this EAL applies to areas not designed to be wetted or submerged. Escalation of the emergency classification is based on the damage caused or by access restrictions that prevent necessary plant operations or systems monitoring.

EAL #7 addresses the potential for the site to experience high level (hurricane force) winds and associated flooding and storm surge over an extended period of time (usually several hours). This EAL is selected because it will generally be associated with significant levels of site severe weather response such as a potential precautionary shutdown, diesel testing, staff call-outs, etc. The site experiencing a hurricane can also be a precursor of more serious events. It is not necessary to declare this event based on issuance of a Hurricane Warning for St. Charles Parish alone.

EAL #8 addresses Mississippi River flooding. The levee system is designed to protect people and property from the most severe effects of river flooding. The Waterford 3 UFSAR section 2.4 indicates that a flood less severe than the Probable Maximum Flood (PMF) but more severe than the Project Design Flood (PDF) may pose the greatest threat to the site in the event of a nearby levee failure. The UFSAR refers to Mississippi River water level of +27 ft. MSL as that corresponding level for such an event that includes appropriate conservatism. Therefore, this level of flooding can also be a precursor of more serious events and is used as an EAL here.

Initiating Condition -- ALERT

HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat

Operating Mode Applicability: **All**

Emergency Action Level(s):

1. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Waterford 3 Security Shift Supervision.

OR

2. A validated notification from NRC of an airliner attack threat within 30 minutes of the site.

Basis:

NOTE: Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs.

These EALs address the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. They are not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land or water attack elements.

The fact that the site is under serious attack or is an identified attack target with minimal time available for further preparation or additional assistance to arrive requires a heightened state of readiness and implementation of protective measures that can be effective (such as on-site evacuation, dispersal or sheltering).

EAL #1

This EAL addresses the potential for a very rapid progression of events due to a HOSTILE ACTION. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the OWNER CONTROLLED AREA. Those events are adequately addressed by other EALs.

Note that this EAL is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA. This includes Independent Spent Fuel Storage Installations that may be outside the PROTECTED AREA but still in the OWNER CONTROLLED AREA.

EAL #2

This EAL addresses the immediacy of an expected threat arrival or impact on the site within a relatively short time.

The intent of this EAL is to ensure that notifications for the airliner attack threat are made in a timely manner and that Offsite Response Organzations and plant personnel are at a state of heightened awareness regarding the credible threat. Airliner is meant to be a large aircraft with the potential for causing significant damage to the plant.

This EAL is met when a plant receives information regarding an airliner attack threat from NRC and the airliner is within 30 minutes of the plant. Only the plant to which the specific threat is made need declare the Alert.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an airliner (airliner is meant to be a large aircraft with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA2

Initiating Condition -- ALERT

Other conditions existing which in the judgment of the Emergency Coordinator/EOF Director warrant declaration of an Alert

Operating Mode Applicability: All

Emergency Action Level(s):

1. Other conditions exist which in the judgment of the Emergency Coordinator/EOF Director indicate that events are in process or have occurred which involve actual or likely potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Basis:

This EAL is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Coordinator/EOF Director to fall under the Alert emergency class.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA3

Initiating Condition -- ALERT

Control Room evacuation has been initiated

Operating Mode Applicability: **All**

Emergency Action Level(s):

1. Entry into OP-901-502, Evacuation of Control Room & Subsequent Plant Shutdown.

Basis:

With the Control Room evacuated, additional support, monitoring and direction through the Technical Support Center and/or other emergency response facility is necessary. Inability to establish plant control from outside the Control Room will escalate this event to a Site Area Emergency.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA4

Initiating Condition -- ALERT

FIRE or EXPLOSION affecting the operability of plant safety systems required to establish or maintain safe shutdown.

Operating Mode Applicability: **All**

Emergency Action Level(s):

1. FIRE or EXPLOSION in the Reactor Auxiliary Building, Containment or Cooling Tower Areas

AND

Affected system parameter indications show degraded performance or plant personnel report VISIBLE DAMAGE to permanent structures or equipment within the specified area.

Basis:

Site specific areas containing functions and systems required for safe shutdown of the plant are taken from the Waterford 3 Post-Fire Safe Shutdown Analysis, EC-F00-026 for this IC. These areas are reflected in FP-001-022, Design Change Fire Protection/Safe Shutdown Review.

This EAL addresses a FIRE / EXPLOSION and not the degradation in performance of affected systems. System degradation is addressed in the System Malfunction (S) EALs. The reference to damage of systems is used to identify the magnitude of the FIRE / EXPLOSION and to discriminate against minor FIREs / EXPLOSIONs. The reference to safety systems is included to discriminate against FIREs/EXPLOSIONs in areas having a low probability of affecting safe operation. The significance here is not that a safety system was degraded but the fact that the FIRE / EXPLOSION was large enough to cause damage to these systems. Thus, the designation of a single train was intentional and is appropriate when the FIRE / EXPLOSION is large enough to affect more than one component.

This situation is not the same as removing equipment for maintenance that is covered by Technical Specifications. Removal of equipment for maintenance is a planned activity controlled in accordance with procedures and, as such, does not constitute a substantial degradation in the level of safety of the plant. A FIRE / EXPLOSION is an UNPLANNED activity and, as such, does constitute a substantial degradation in the level of safety of the plant. In this situation, an Alert classification is warranted.

HA4

The inclusion of a "report of VISIBLE DAMAGE" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The occurrence of the EXPLOSION with reports of evidence of damage is sufficient for declaration. The declaration of an Alert and the activation of the Technical Support Center will provide the Emergency Coordinator/EOF Director with the resources needed to perform these damage assessments. The Emergency Coordinator/EOF Director also needs to consider any security aspects of the EXPLOSIONs, if applicable.

Escalation to a higher emergency class, if appropriate, will be based on System Malfunction (S), Fission Product Barrier Degradation (F), Abnormal Radiation Levels / Radiological Effluents (A), or Emergency Coordinator/EOF Director Judgment EALs (H…2).

HA5

Initiating Condition -- ALERT

Release of toxic or flammable gases within or contiguous to VITAL AREA which jeopardizes operation of systems required to maintain safe operations or establish or maintain safe shutdown.

Operating Mode Applicability: **All**

Emergency Action Level(s): (1 or 2)

1. Report or detection of toxic gases within or contiguous to VITAL AREA in concentrations that may result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH).

OR

2. Report or detection of gases in concentration > LOWER FLAMMABILITY LIMIT within or contiguous to VITAL AREA.

Basis:

This IC is based on gases that affect the safe operation of the plant. These EALs apply to buildings and areas contiguous to plant VITAL AREAs or other significant buildings or areas. The intent of these EALs is not to include buildings (e.g., warehouses, MSB, Construction Support Building, etc.) or other areas that are not contiguous or immediately adjacent to plant VITAL AREAs. It is appropriate that increased monitoring be done to ascertain whether consequential damage has occurred.

EAL #1 is met if measurement of toxic gas concentration results in an atmosphere that is IDLH within a VITAL AREA or any area or building contiguous to a VITAL AREA. Exposure to an IDLH atmosphere will result in immediate harm to unprotected personnel, and would preclude access to any such affected areas.

EAL #2 is met when the flammable gas concentration in a VITAL AREA or any building or area contiguous to a VITAL AREA exceeds the LOWER FLAMMABILITY LIMIT. Flammable gasses, such as hydrogen and acetylene, are routinely used to maintain plant systems (hydrogen) or to repair equipment/components (acetylene - used in welding). This EAL addresses concentrations at which gases can ignite/support combustion. An uncontrolled release of flammable gasses within a facility structure has the potential to affect safe operation of the plant by limiting either operator or equipment operations due to the potential for ignition and resulting equipment damage/personnel injury. Once it has been determined that an uncontrolled release is occurring, then sampling must be done to determine if the concentration of the released gas is within this range.

HA5

Escalation to a higher emergency class, if appropriate, will be based on System Malfunction (S), Fission Product Barrier Degradation (F), Abnormal Rad Levels / Radioactive Effluent (A), or Emergency Coordinator/EOF Director Judgment EALs (H…2).

HA6

Initiating Condition -- ALERT

Natural and destructive phenomena affecting the plant VITAL AREA.

Operating Mode Applicability: All

Emergency Action Level(s): (1 or 2 or 3 or 4 or 5)

1 RED LIGHT on the seismic monitor panel indicates a VALID Seismic Event > Operating Basis Earthquake (OBE).

OR

2. Tornado or high winds > 100 mph within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the following plant structures/equipment or Control Room indication of degraded performance of those systems.
   - Containment
   - Reactor Auxiliary Building
   - Turbine Building
   - Cooling Tower Areas

OR

3. Vehicle crash within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the following plant structures or equipment therein or Control Room indication of degraded performance of those systems.
   - Containment
   - Reactor Auxiliary Building
   - Turbine Building
   - Cooling Tower Areas

OR

4. Turbine failure-generated missiles result in any VISIBLE DAMAGE to or penetration of any of the following plant areas.
   - Containment
   - Reactor Auxiliary Building
   - Cooling Tower Areas

OR

5. Uncontrolled flooding in the Reactor Auxiliary Building or Cooling Tower Areas that results in degraded safety system performance as indicated in the Control Room or that creates industrial safety hazards (e.g., electric shock) that preclude access necessary to operate or monitor safety equipment.

Basis:

These EALs escalate from the NOUE EALs in HU6 in that the occurrence of the event has resulted in VISIBLE DAMAGE to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by control indications of degraded system response or performance. The occurrence of VISIBLE DAMAGE and/or degraded system response is intended to discriminate against lesser events. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation. Escalation to higher classifications occurs on the basis of other EALs (e.g., System Malfunction (S)).

EAL #1 is based on seismic events of a magnitude that can result in a plant VITAL AREA being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to plant safety systems. See EPRI-sponsored "Guidelines for Nuclear Plant Response to an Earthquake", dated October 1989, for information on seismic event categories.

EAL #2 is based on the FSAR design basis 100 year recurrence interval projected wind velocity of 100 miles per hour. Wind loads of this magnitude can cause significant damage to site structures, and is well below the actual site design basis for Seismic Class One structures of 200 mph. Wind speed for this EAL is based on a 15-minute average from the 10 meter (low level) meteorological monitoring point.

EAL #3 is intended to address crashes of **vehicle types large enough** to cause **significant damage** to plant structures containing functions and systems required for safe shutdown of the plant. Minor accidents involving smaller vehicles or golf carts where significant damage to site structures is not a concern or “fender bender” type accidents do **not** warrant declaration under this EAL.

EAL #4 is intended to address the threat to safety related equipment imposed by missiles generated by main turbine rotating component failures. The list of areas includes areas containing safety-related equipment, their controls, and their power supplies that a turbine missile could penetrate. This EAL is, therefore, consistent with the definition of an ALERT in that if missiles have damaged or penetrated areas containing safety-related equipment the potential exists for substantial degradation of the level of safety of the plant.

HA6

EAL #5 is intended to address the effect of internal flooding (OR external flooding that is of such magnitude that it affects the Reactor Auxiliary Building or Cooling Tower Areas) that has resulted in degraded performance of systems affected by the flooding, or has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to operate or monitor safety equipment represents a potential for substantial degradation of the level of safety of the plant. This flooding may have been caused by internal events such as component failures, equipment misalignment, or outage activity mishaps. The areas include those areas that contain systems required for safe shutdown of the plant that are not designed to be wetted or submerged. Site specific areas containing functions and systems required for safe shutdown of the plant are taken from the Waterford 3 Post-Fire Safe Shutdown Analysis, EC-F00-026 for this EAL. These areas are reflected in FP-001-022, Design Change Fire Protection/Safe Shutdown Review. The Containment Building is not included in the EAL because of the guidance in the NEI 99-01 basis that this EAL applies to areas not designed to be wetted or submerged.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS1

Initiating Condition – SITE AREA EMERGENCY

HOSTILE ACTION within the PROTECTED AREA.

Operating Mode Applicability: **All**

Emergency Action Level(s):

1. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Waterford 3 Security Shift Supervision.

Basis:

This condition represents an escalated threat to plant safety above that contained in the Alert in that a HOSTILE FORCE has progressed from the OWNER CONTROLLED AREA to the PROTECTED AREA.

This Initiating Condition addresses the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. It is not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land or water attack elements.

The fact that the site is under serious attack or is an identified attack target with minimal time available for further preparation or additional assistance to arrive requires Offsite Response Organization readiness and preparation for the implementation of protective measures.

This Initiating Condition addresses the potential for a very rapid progression of events due to a HOSTILE ACTION. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the PROTECTED AREA. Those events are adequately addressed by other EALs.

Escalation of this emergency classification level, if appropriate, would be based on actual plant status after impact or progression of attack.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HS2

Initiating Condition – SITE AREA EMERGENCY

Other conditions existing which in the judgment of the
Emergency Coordinator/EOF Director warrant declaration of Site Area Emergency.

Operating Mode Applicability: **All**

Emergency Action Level(s):

1. Other conditions exist which in the judgment of the Emergency Coordinator/EOF Director indicate that events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the Exclusion Area Boundary.

Basis:

This EAL is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Coordinator/EOF Director to fall under the emergency class description for Site Area Emergency.

HS3

Initiating Condition – SITE AREA EMERGENCY

Control Room evacuation has been initiated and plant control cannot be established.

Operating Mode Applicability: **All**

Emergency Action Level(s):

1. Control Room evacuation has been initiated

AND

Control of the plant cannot be established in accordance with OP-901-502, Evacuation of Control Room & Subsequent Plant Shutdown within 15 minutes

Basis:

The Waterford 3 Post-Fire Safe Shutdown Analysis, EC-F00-026 provides the basis for these EALs.

Expeditious transfer of safety systems has not occurred but fission product barrier damage may not yet be indicated. The intent of this IC is to capture those events where control of the plant cannot be reestablished in a timely manner. The determination of whether or not control is established at the remote shutdown panel is based on Emergency Coordinator/EOF Director judgment. The Emergency Coordinator/EOF Director is expected to make a reasonable, informed judgment within 15 minutes that control of the plant from the remote shutdown panel has been established.

The intent of the EAL is to establish control of important plant equipment and knowledge of important plant parameters in a timely manner. Primary emphasis should be placed on those components and instruments that supply protection for and information about safety functions such as reactivity control (ability to shutdown the reactor and maintain it shutdown), RCS inventory (ability to cool the core), and decay heat removal (ability to maintain a heat sink).

Escalation of this event, if appropriate, would be by Fission Product Barrier Degradation (F), Abnormal Radiation Levels/Radiological Effluents (A), or Emergency Coordinator/EOF Director Judgment (H…2) EALs.

Initiating Condition – GENERAL EMERGENCY

HOSTILE ACTION resulting in loss of physical control of the facility

Operating Mode Applicability: **All**

Emergency Action Level(s):

1. A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions.

OR

2. A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and IMMINENT fuel damage is likely for a freshly off-loaded reactor core in pool.

Basis:

EAL #1:

This EAL encompasses conditions under which a HOSTILE ACTION has resulted in a loss of physical control of VITAL AREAs (containing vital equipment or controls of vital equipment) required to maintain safety functions **and** control of that equipment can **not** be transferred to and operated from another location. These safety functions are reactivity control (ability to shut down the reactor and keep it shutdown) RCS inventory (ability to cool the core), and decay heat removal (ability to maintain a heat sink).

Loss of physical control of the Control Room or LCP-43 (remote shutdown panel) capability alone may not prevent the ability to maintain safety functions per se. Design of the remote shutdown capability, the location of the transfer switches and areas of the plant where physical control has been lost should be taken into account. Primary emphasis should be placed on those components and instruments that supply protection for and information about safety functions.

If control of the plant equipment necessary to maintain safety functions can be transferred to another location, then the above initiating condition is not met.

EAL #2:

This EAL addresses failure of spent fuel cooling systems as a result of HOSTILE ACTION if IMMINENT fuel damage is likely, such as when a freshly off-loaded reactor core is in the spent fuel pool. At Waterford 3, the term “freshly off-loaded reactor core” refers to fuel that has been discharged from the core and stored in the spent fuel pool for a period of LESS THAN one year.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HG2

Initiating Condition – GENERAL EMERGENCY

Other conditions existing which in the judgment of the Emergency Coordinator/EOF Director warrant declaration of General Emergency.

Operating Mode Applicability: **All**

Emergency Action Level(s):

1. Other conditions exist which in the judgment of the Emergency Coordinator/EOF Director indicate that events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Basis:

This EAL is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Coordinator/EOF Director to fall under the General Emergency class.

SYSTEM MALFUNCTION

SYSTEM MALFUNCTION

SU1

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

Loss of all offsite power to essential busses > 15 minutes.

Operating Mode Applicability:

Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Emergency Action Level(s):

1. Loss of power to all unit auxiliary and startup transformers > 15 minutes.

AND

At least 'A' and 'B' emergency diesel generators supplying power to emergency busses.

Basis:

Prolonged loss of AC power reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete Loss of AC power (e.g., Station Blackout). Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Loss of all offsite power varies depending on the plant mode and source transformers. If the unit is back feeding via the Unit Auxiliary Transformers and offsite power is lost, declaration of an Unusual Event is warranted.

SYSTEM MALFUNCTION

SU6

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

UNPLANNED loss of most or all safety system annunciation or indication in the Control Room > 15 minutes.

Operating Mode Applicability:

Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Emergency Action Level(s):

UNPLANNED loss of most or all annunciator cabinets C, D, H, K, M, N, SA, SB annunciators or indicators associated with safety systems > 15 minutes.

Basis:

This IC and its associated EAL are intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment.

Recognition of the availability of computer based indication equipment is considered (e.g., SPDS, plant computer, etc.).

Indicators associated with safety systems are those indicators for reactivity control, core cooling, maintaining reactor coolant system integrity or maintaining containment integrity.

Quantification of "Most" is arbitrary, however, it is estimated that if approximately **75%** of the safety system annunicators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected.

It is not intended that Operations personnel perform a detailed count of the instrumentation lost, but use the value as a judgment threshold for determining the severity of plant conditions.

These EALs also recognize that redundant safety system indication powered from separate uninterruptible power supplies is provided. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions.

SYSTEM MALFUNCTION

SU6

The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, then the UNUSUAL EVENT is based on SU11 "Inability to reach required shutdown within Technical Specification time limits."

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. Due to the limited number of safety systems in operation during cold shutdown, refueling, and defueled modes, no EAL is indicated during these modes of operation.

This UNUSUAL EVENT will be escalated to an Alert if a transient is in progress during the loss of annunciation or indication.

SYSTEM MALFUNCTION

SU7

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

RCS Leakage.

Operating Mode Applicability:

Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Emergency Action Level(s): (1 or 2)

1. Unidentified or pressure boundary leakage > 10 gpm.

OR

2. Identified leakage > 25 gpm.

Basis:

This IC is included as an Unusual Event because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. The 10 gpm value for the unidentified and pressure boundary leakage was selected as it is observable with normal Control Room indications. Lesser values must generally be determined through time-consuming surveillance tests (e.g., mass balances). The EAL for identified leakage is set at a higher value due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage. Steam generator tube leakage is identified leakage.

Escalation to the Alert level is via Fission Product Barrier Degradation (F) EALs.

SYSTEM MALFUNCTION

SU8

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

UNPLANNED loss of all onsite or offsite communications capabilities.

Operating Mode Applicability:

Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Emergency Action Level(s): (1 or 2)

1. Loss of all Table M1 onsite communications equipment affecting the ability to perform routine operations.

OR

2. Loss of all Table M2 offsite communications capability

Table M1 Onsite Communications Equipment
Plant radio system Plant paging system In-plant telephones Sound powered phones

Table M2 Offsite Communications Equipment
All telephone lines (commercial and microwave) Industrial Hot Line ENS Civil Defense Radios Operational Hotline

Basis:

The purpose of this IC and its associated EALs is to recognize a loss of communications capability that either defeats the plant operations staff ability to perform routine tasks necessary for plant operations or the ability to communicate problems with offsite authorities. The loss of offsite communications ability is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

The availability of one method of ordinary offsite communications is sufficient to inform State and local authorities of plant problems. This EAL is intended to be used only when extraordinary means (e.g., relaying of information from radio transmissions, individuals being sent to offsite locations, etc.) are being utilized to make communications possible.

SYSTEM MALFUNCTION

SU9

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

Fuel clad degradation.

Operating Mode Applicability:

Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Emergency Action Level(s):

1. Reactor coolant sample activity value indicating fuel clad degradation > Technical Specification allowable limits.

   - >60 µCi/gm DEI

OR

   - >1.0 µCi/gm DEI for more than 48 hours during one continuous time interval

OR

   - >100/Ē µCi/gm

Basis:

This IC is included because it is a precursor of more serious conditions and, as a result, is considered to be a potential degradation in the level of safety of the plant.

The EAL addresses coolant samples exceeding coolant Technical Specifications for transient iodine spiking limits and coolant samples exceeding coolant Technical Specifications for nominal operating iodine limits for the time period specified in the Technical Specifications.

Escalation of this IC to the Alert level is via the Fission Product Barrier Degradation Monitoring (F) ICs.

SYSTEM MALFUNCTION

SU10

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

Inadvertent criticality.

Operating Mode Applicability: **Hot Standby (Mode 3)**
Hot Shutdown (Mode 4)

Emergency Action Level(s):

1. An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

Basis:

This IC addresses inadvertent criticality events. While the primary concern is criticality events that occur in Cold Shutdown or Refueling modes (NUREG 1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States), the IC is applicable in other modes in which inadvertent criticalities are possible. This IC indicates a potential degradation of the level of safety of the plant, warranting an Unusual Event classification. This IC excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups (e.g., criticality earlier than estimated). This IC corresponds to Cold Shutdown/Refueling CU7.

This condition can be identified using the startup rate meter. The term “sustained“ is used in order to allow exclusion of expected short term positive startup rates from planned control rod movements (such as shutdown bank withdrawal). These short term positive startup rates are the result of the increase in neutron population due to subcritical multiplication.

Escalation would be by the Fission Product Barrier Matrix (F), as appropriate to the operating mode at the time of the event, or by Emergency Coordinator Judgment.

SYSTEM MALFUNCTION

SU11

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

Inability to reach required shutdown within Technical Specification time limits.

Operating Mode Applicability:

Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Emergency Action Level(s):

1. Plant is not brought to required operating mode within Technical Specifications LCO Action Statement time

Basis:

Limiting Conditions of Operation (LCOs) require the plant to be brought to a required shutdown mode when the Technical Specification required configuration cannot be restored. Depending on the circumstances, this may or may not be an emergency or precursor to a more severe condition. In any case, the initiation of plant shutdown required by the site Technical Specifications requires a one-hour report under 10 CFR 50.72 (b) Non-emergency events. The plant is within its safety envelope when being shut down within the allowable action statement time in the Technical Specifications. An **immediate** UNUSUAL EVENT is required when the plant is not brought to the required operating mode within the allowable action statement time in the Technical Specifications. Declaration of an Unusual Event is based on the time at which the LCO-specified action statement time period elapses under Technical Specifications and is not related to how long a condition may have existed**.** Other required Technical Specification shutdowns that involve precursors to more serious events are addressed by other System Malfunction (S), Hazards (H), or Fission Product Barrier Degradation (F) EALs.

SYSTEM MALFUNCTION

SA1

Initiating Condition -- ALERT

AC power capability to essential busses reduced to a single power source > 15 minutes such that any additional single failure would result in station blackout.

Operating Mode Applicability:

Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Emergency Action Level(s):

1. AC power capability to essential busses reduced to a single power source > 15 minutes.

AND

Any additional single failure will result in station blackout.

Basis:

This IC and its associated EAL is intended to provide an escalation from IC SU1. The condition indicated by this IC is the degradation of the offsite and onsite power systems such that any additional single failure would result in a station blackout. This condition could occur due to a loss of all offsite power with a concurrent failure of one emergency diesel generator to supply power to its emergency busses. Another related condition could be the loss of all offsite power **and** loss of the onsite emergency diesel generators with **only** one train of emergency busses being backfed from the unit main generator, **or** the loss of onsite emergency diesels with only one train of emergency busses being backfed from offsite power. The subsequent loss of this single power source would escalate the event to a Site Area Emergency in accordance with SS1, "Loss of All Offsite and Loss of All Onsite AC Power to Essential Busses."

When temporary emergency diesels (TEDs) are used to supplement onsite AC power for essential busses in the event diesels are lost, they are credited in this EAL. The EAL condition does not apply unless the TED also failed.

SYSTEM MALFUNCTION

SA3

Initiating Condition -- ALERT

Failure of Reactor Protection System instrumentation to complete or initiate an automatic reactor trip once a Reactor Protection System setpoint has been exceeded and manual trip was successful.

Operating Mode Applicability:

Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)

Emergency Action Level(s):

Indication(s) exist that indicate that the Reactor Protection System setpoint was exceeded and automatic trip did not occur and a successful manual trip occurred.

Basis:

This condition indicates failure of the automatic protection system to trip the reactor. This condition is more than a potential degradation of a safety system in that a front line automatic protection system did not function in response to a plant transient and thus the plant safety has been compromised, and design limits of the fuel may have been exceeded. An Alert is indicated because conditions exist that lead to potential loss of fuel clad or RCS. Reactor protection system setpoint being exceeded, rather than limiting safety system setpoint being exceeded, is specified here because failure of the automatic protection system is the issue. A manual trip is any set of actions by the reactor operator(s) at the **reactor control panel** which causes control rods to be rapidly inserted into the core and brings the reactor subcritical (e.g., reactor trip button, DRTS buttons). Failure of manual trip would escalate the event to a Site Area Emergency. Opening the A32 and B32 Bus Feeders to facilitate insertion of all CEAs requires declaration of a Site Area Emergency under SS3. If the RPS, Automatic Reactor trip, fails and a manual reactor trip is initiated, the EAL is satisfied and an Alert must be declared.

SYSTEM MALFUNCTION

SA6

Initiating Condition -- ALERT

UNPLANNED loss of most or all safety system annunciation or indication in the Control Room with either (1) a SIGNIFICANT TRANSIENT in progress, or (2) compensatory non-alarming indicators are unavailable.

Operating Mode Applicability:

Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Emergency Action Level(s):

1. UNPLANNED loss of most or all annunciator cabinets C, D, H, K, M, N, SA, SB annunciators or indicators associated with safety systems > 15 minutes.

AND

Either of the following (a or b):

a. SIGNIFICANT TRANSIENT is in progress.

OR

b. Compensatory non-alarming indications are unavailable.

Basis:

This IC and its associated EAL are intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment during a transient. Recognition of the availability of computer based indication equipment is considered (e.g., SPDS, plant computer, etc.) in this IC.

"Planned" loss of annunciators or indicators includes scheduled maintenance and testing activities.

Quantification of “Most” is arbitrary, however, it is estimated that if approximately **75%** of the safety system annunicators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected.

Indicators associated with safety systems are those indicators for reactivity control, core cooling, maintaining reactor coolant system integrity or maintaining containment integrity.

It is **not** intended that Operators perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift Manager be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation.

It is further recognized that redundant safety system indication powered from separate uninterruptible power supplies is provided. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This is addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, then the UNUSUAL EVENT is based on SU11 "Inability to Reach Required Shutdown Within Technical Specification Limits."

"Compensatory non-alarming indications" in this context includes computer-based information such as SPDS, QSPDS, COLSS, etc. This includes all computer systems available for this use. If **both** a major portion of the annunciation system **and** all computer monitoring are unavailable, then the Alert is required.

This Alert will be escalated to a Site Area Emergency if the operating crew can not monitor a transient in progress.

Due to the limited number of safety systems in operation during cold shutdown, refueling, and defueled modes, no EAL is indicated during these modes of operation.

SYSTEM MALFUNCTION

SS1

Initiating Condition -- SITE AREA EMERGENCY

Loss of all offsite power and loss of all onsite AC power to essential busses.

Operating Mode Applicability:

Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Emergency Action Level(s):

1. Loss of power to all unit auxiliary and startup transformers

AND

Failure of the 'A' and 'B' emergency diesel generators to supply power to emergency busses

AND

Failure to restore power to at least one emergency bus within 15 minutes from the time of loss of both offsite and onsite AC power.

Basis:

Loss of all AC power compromises all plant safety systems requiring electric power including Shutdown Cooling, ECCS, Containment Heat Removal and the Ultimate Heat Sink. Prolonged loss of all AC power will cause core uncovering and loss of containment integrity, thus this event can escalate to a General Emergency.

Escalation to General Emergency is via Fission Product Barrier Degradation (F) or SG1, "Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power."

Consideration should be given to operable loads necessary to remove decay heat or provide Reactor Vessel makeup capability when evaluating loss of AC power to essential busses. Even though an essential bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or Reactor Vessel makeup capability) are not operable on the energized bus, then the bus should not be considered operable for this IC. If this bus was the only energized bus, then a Site Area Emergency in accordance with SS1 should be declared.

SYSTEM MALFUNCTION

SS1

Loss of all offsite power varies depending on the plant mode and source transformers. If the unit is back feeding via the unit Auxiliary Transformers and offsite power is lost in conjunction with loss of onsite AC power from the emergency diesel generators, then declaration of a Site Area Emergency is warranted.

When temporary emergency diesels (TEDs) are used to supplement onsite AC power for essential busses in the event diesels are lost, they are credited in this EAL. The EAL condition does not apply unless the TED also failed, provided the TED powers necessary loads as described above.

SYSTEM MALFUNCTION

SS3

Initiating Condition -- SITE AREA EMERGENCY

Failure of Reactor Protection System instrumentation to complete or initiate an automatic reactor trip once a Reactor Protection System setpoint has been exceeded and manual trip was NOT successful.

Operating Mode Applicability: **Power Operations (Mode 1)**
Startup (Mode 2)

Emergency Action Level(s):

Indication(s) exist that automatic and manual trip were not successful.

Basis:

Automatic and manual trip are not considered successful if action away from the **reactor control console** was required to trip the reactor. For example, opening the A32 and B32 Bus Feeders to facilitate insertion of all CEAs requires declaration of a Site Area Emergency.

Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed. A Site Area Emergency is indicated because conditions exist that lead to imminent loss or potential loss of both fuel clad and RCS. Although this may be viewed as redundant to the Fission Product Barrier Degradation (F) EALs, its inclusion is necessary to better assure timely recognition and emergency response. Escalation of this event to a General Emergency would be via Fission Product Barrier Degradation (FG1) or Emergency Coordinator/EOF Director Judgment EALs (HG2).

SYSTEM MALFUNCTION

SS4

Initiating Condition -- SITE AREA EMERGENCY

Loss of all vital DC power.

Operating Mode Applicability:

Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Emergency Action Level(s):

1. Loss of all Vital DC power based on bus voltage indications < 108 volts for > 15 minutes.

Basis:

Loss of all DC power compromises ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system. The minimum voltage necessary, based on plant design, is 105 volts; however, the lowest battery voltage attained on a loss of off site power at the end of the 4 hour period is 107.4 volts on the 'B' battery bank. 108 volts is used for the EAL indication because the Control Room instrumentation reads in 2 volt increments. Reference calculations ECE91-058, "Battery 3A-S "A Train" Calculation for Station Blackout "and ECE91-059, "Battery 3B-S "B Train" Calculation for Station Blackout."

Escalation to a General Emergency would occur by Abnormal Radiation Levels/Radiological Effluents (AG1), Fission Product Barrier Degradation (FG1), or Emergency Coordinator/EOF Director Judgment (HG2) EALs. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

SYSTEM MALFUNCTION

SS5

Initiating Condition -- SITE AREA EMERGENCY

Complete loss of heat removal capability.

Operating Mode Applicability:

Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Emergency Action Level(s):

1. Loss of core cooling and heat sink.

Basis:

This EAL addresses complete loss of functions, including ultimate heat sink, required to attain and maintain Hot Shutdown (Mode 4) with the reactor at pressure and temperature. Reactivity control is addressed in other EALs.

Under these conditions, there is an actual major failure of systems intended for protection of the public. Thus, declaration of a Site Area Emergency is warranted. Escalation to General Emergency would be via Abnormal Radiation Levels / Radiological Effluents (AG1), Emergency Coordinator/EOF Director Judgment (HG2), or Fission Product Barrier Degradation (FG1) EALs.

Steam Generator levels and natural circulation may be used as indicators because RCS temperatures in Mode 4 will be high enough to use Steam Generators as a heat sink. The inability to makeup to the RCS will prevent establishing or maintaining Hot Shutdown due to the inability to maintain adequate RCS inventory.

SYSTEM MALFUNCTION

SS6

Initiating Condition -- SITE AREA EMERGENCY

Inability to monitor a SIGNIFICANT TRANSIENT in progress.

Operating Mode Applicability:

Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Emergency Action Level(s):

1. a. Loss of most or all annunciator cabinets C, D, H, K, M, N, SA, SB annunciators associated with safety systems.

AND

b. Compensatory non-alarming indications are unavailable

AND

c. Indications needed to monitor safety functions (reactivity control, core cooling, maintaining reactor coolant system integrity or maintaining containment integrity) are unavailable

AND

d. SIGNIFICANT TRANSIENT in progress

Basis:

This IC and its associated EAL are intended to recognize the inability of the Control Room staff to monitor the plant response to a transient. A Site Area Emergency is considered to exist if the Control Room staff can **not** monitor safety functions needed for protection of the public.

Waterford 3 has defined "most" for the first indicator in the EAL to be a loss of **75%** or more of annunciator cabinets C, D, H, K, M, N, SA, SB annunciators. Loss of these annunciator cabinet annunciators or instrumentation has been identified as having the greatest impact on normal operations and safe shutdown of the plant. It is **not** intended that Operations personnel perform a detailed count of the instrumentation lost, but use the value as a judgment threshold for determining the severity of plant conditions. It is also **not** intended that the Shift Manager be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation.

"Compensatory non-alarming indications" in this context includes computer based information such as SPDS, QSPDS, COLSS, etc. This includes all computer systems available for this use.

Indicators associated with safety systems are those indicators for reactivity control, core cooling, maintaining reactor coolant system integrity or maintaining containment integrity. Indications needed to monitor safety functions necessary for protection of the public must include Control Room indications, computer generated indications and dedicated annunciation capability.

"Planned" and **"UNPLANNED"** actions are **not** differentiated since the loss of instrumentation of this magnitude is of such significance during a transient that the cause of the loss is not an ameliorating factor.

This event is required to be declared **regardless** of the length of time equipment is out of service and whether or not equipment is unavailable due to failure or planned maintenance or testing.

SYSTEM MALFUNCTION

SG1

Initiating Condition -- GENERAL EMERGENCY

Prolonged loss of all offsite power and prolonged loss of all onsite AC power to essential busses.

Operating Mode Applicability:

Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Emergency Action Level(s):

1. Loss of power to all unit auxiliary and startup transformers.

AND

Failure of both 'A' and 'B' emergency diesel generators to supply power to emergency busses.

AND

Either of the following: (a or b)

a. Restoration of at least one emergency bus within 4 hours is not likely

OR

b. FA1 entry conditions met.

Basis:

Loss of all AC power compromises all plant safety systems requiring electric power including Shutdown Cooling, ECCS, Containment Heat Removal and the Ultimate Heat Sink. Prolonged loss of all AC power will lead to loss of fuel clad, RCS, and containment.

This IC is specified to assure that in the unlikely event of a prolonged station blackout, timely recognition of the seriousness of the event occurs and that declaration of a General Emergency occurs as early as is appropriate, based on a reasonable assessment of the event trajectory. The likelihood of restoring **at least one** emergency bus should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions. The 4 hours to restore AC power is based on the site blackout coping analysis performed in conformance with 10 CFR 50.63 and Regulatory Guide 1.155, "Station Blackout."

SYSTEM MALFUNCTION

SG1

Appropriate allowance for offsite emergency response, including evacuation of surrounding areas has been considered. Although this EAL may be viewed as redundant to the Fission Product Barrier Degradation (FG1) EALs, its inclusion is necessary to better assure timely recognition and emergency response.

When temporary emergency diesels (TEDs) are used to supplement onsite AC power for essential busses in the event diesels are lost, they are credited in this EAL.

In addition, under these conditions, fission product barrier monitoring capability may be degraded. Although it may be difficult to predict when power can be restored, it is necessary to give the Emergency Coordinator/EOF Director a reasonable idea of how quickly (s)he may need to declare a General Emergency based on two major considerations:

1. Are there any present indications that core cooling is already degraded to the point that Loss or Potential Loss of Fission Product Barriers is imminent?

2. If there are no present indications of such core cooling degradation, then how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented?

Thus, indication of continuing core cooling degradation must be based on Fission Product Barrier monitoring with particular emphasis on Emergency Coordinator/EOF Director judgment as it relates to imminent Loss or Potential Loss of fission product barriers and degraded ability to monitor fission product barriers using the barrier indicators in section F of the EALs.

SYSTEM MALFUNCTION

SG3

Initiating Condition -- GENERAL EMERGENCY

Failure of the Reactor Protection System to complete an automatic trip and manual trip was NOT successful and there is indication of an extreme challenge to the ability to cool the core.

Operating Mode Applicability: **Power Operations (Mode 1)**
Startup (Mode 2)

Emergency Action Level(s):

1. Indications exist that automatic and manual trip were not successful.

AND

Either of the following: (a or b)

a. Indication(s) exists that core cooling is extremely challenged as indicated by CET temperatures at or approaching 1200° F

OR

b. Indication(s) exists that heat removal is extremely challenged as indicated by inability to maintain at least one steam generator level > 50% wide range.

Basis:

Automatic and manual trip are **not** considered successful if action away from the **reactor control console** was required to trip the reactor. For example, opening the A32 and B32 Bus Feeders to facilitate insertion of all CEAs is **NOT** considered as a successful manual trip under this IC.

Under the conditions of this IC and its associated EALs, the efforts to bring the reactor subcritical have been unsuccessful and, as a result, the reactor is producing more heat than the maximum decay heat load for which the safety systems were designed. Although there are capabilities away from the reactor control console, such as emergency boration, the continuing temperature rise indicates that these capabilities are not effective. This situation could be a precursor for a core melt sequence.

SYSTEM MALFUNCTION

SG3

For Waterford 3, the extreme challenge to the ability to cool the core means that core exit thermocouple temperatures are at or approaching 1200 degrees F. Another consideration is the inability to initially remove heat during the early stages of this sequence. If feedwater flow is insufficient to remove the amount of heat required by design (SG level less than 50% Wide Range) from at least one steam generator, then an extreme challenge should be considered to exist. This level is taken from OP-902-002, Loss of Coolant Accident Recovery Procedure.

In the event either of these challenges exist at a time that the reactor has not been brought below the power associated with the safety system design (typically 3 to 5% power) a core melt sequence exists. In this situation, core degradation can occur rapidly. For this reason, the General Emergency declaration is intended to be anticipatory of the fission product barrier matrix declaration to permit maximum offsite intervention time.

DNBR MARGIN

3.2.4 The DNBR margin shall be maintained by one of the following methods:

a) When COLSS is in service and neither CEAC is operable: maintain COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR decreased by 13% RATED THERMAL POWER.

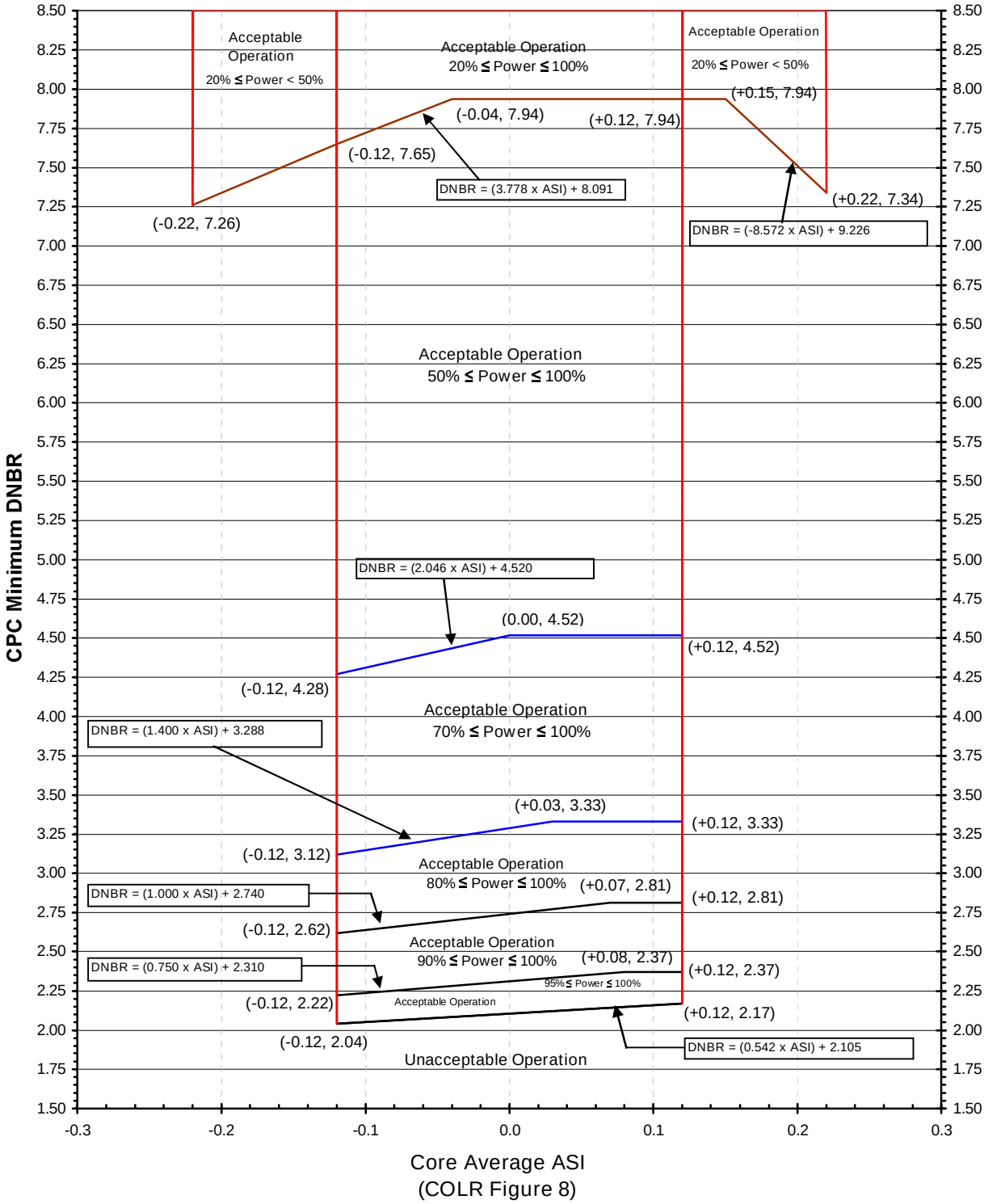
b) When COLSS is out of service and at least one CEAC is operable: operate within the region of acceptable operation shown on COLR Figure 8 (or 8A as appropriate), using any operable CPC channel.

c) When COLSS is out of service and neither CEAC is operable: operate within the region of acceptable operation shown on COLR Figure 9 (or 9A as appropriate), using any operable CPC channel.

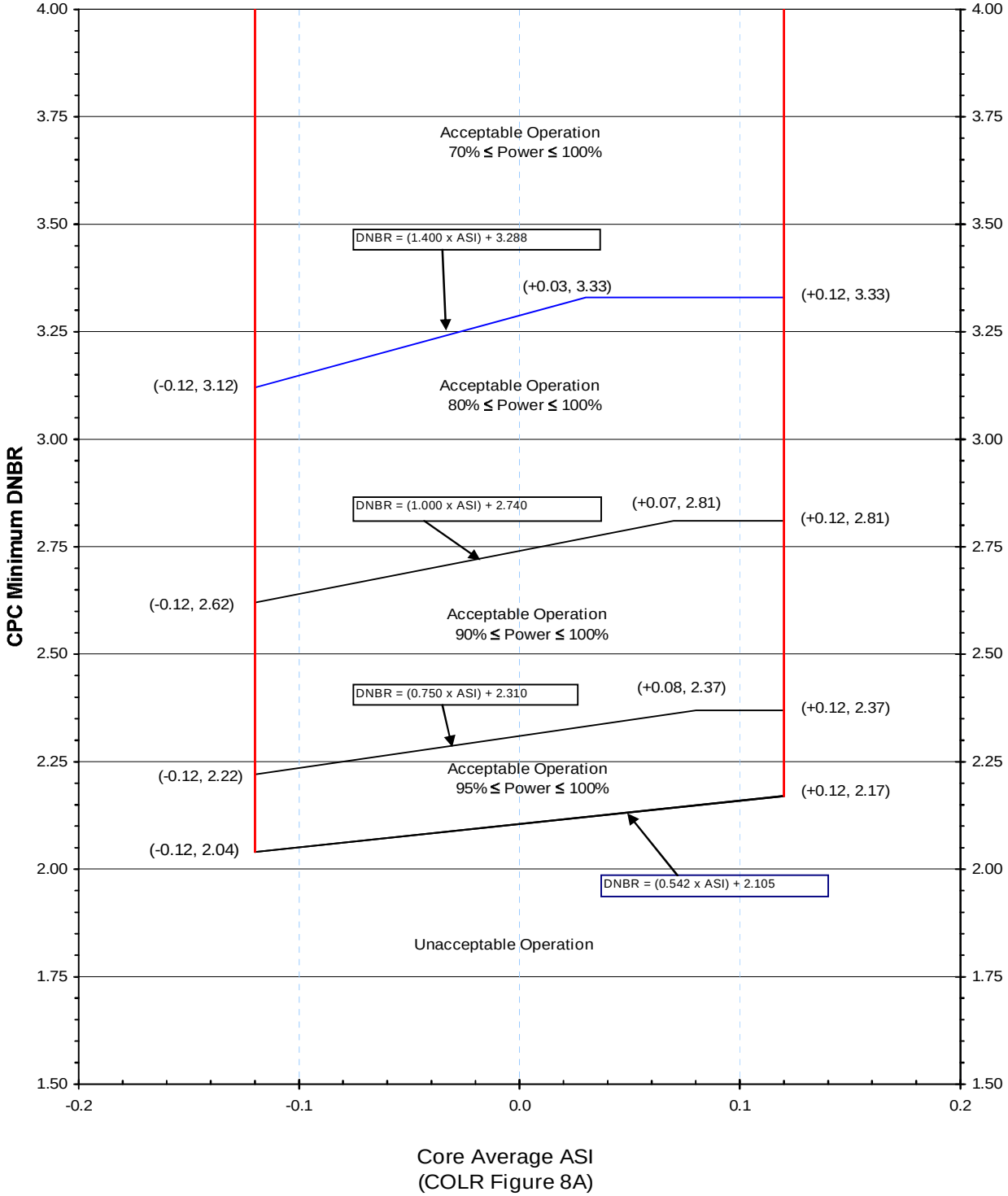
NOTES

1. The various DNBR limit lines shown between the vertical ASI limit lines at ±0.12 and ±0.22 on Figures 8, 8A, 9, and 9A represent the minimum CPC-calculated DNBR value required for operation in the power range displayed in the area above each line. Operation at lower power levels requires that a larger DNBR value be maintained. For example, with ASI equal to -0.12 and a core power of 85%, CPC calculated DNBR must be a minimum of 2.62 with any CEAC Operable. At 79% power and the same ASI value with any CEAC Operable, the calculated DNBR must be at least 3.12. At 65% power and the same ASI value, DNBR must be a minimum of 4.28. At 90% power and an ASI value of +0.08, DNBR must be no less than 2.37.
2. The vertical ASI limit lines shown at ±0.12 and ±0.22 on Figures 8, 8A, 9, and 9A may be considered as extending beyond the maximum DNBR value on the Y-axis of the charts. Therefore, when monitoring DNBR with these figures, compliance is achieved at all power levels shown on a given figure when DNBR is greater than the largest DNBR value on the vertical scale.
3. Figure 8A is provided to offer better resolution for the four power ranges in the lower portion of Figure 8. Figure 9A is provided to offer better resolution for the four power ranges in the lower portion of Figure 9.
4. Equations are provided on Figures 8, 8A, 9, and 9A to assist in determining DNBR limits in the sloped portions of the plots.

Allowable DNBR with Any CEAC Operable
(COLSS Out of Service)

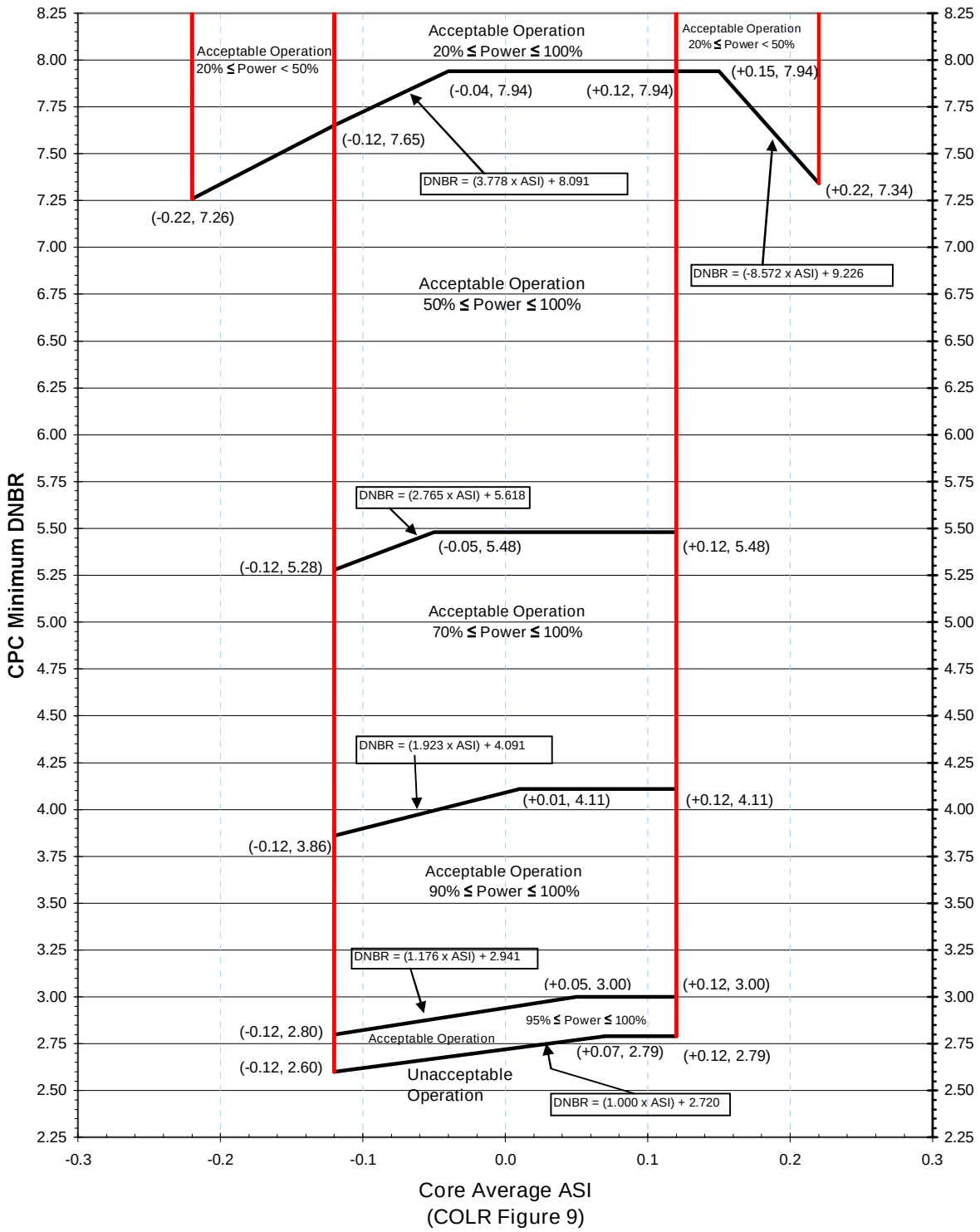


Subset of Allowable DNBR with Any CEAC Operable
(COLSS Out of Service)

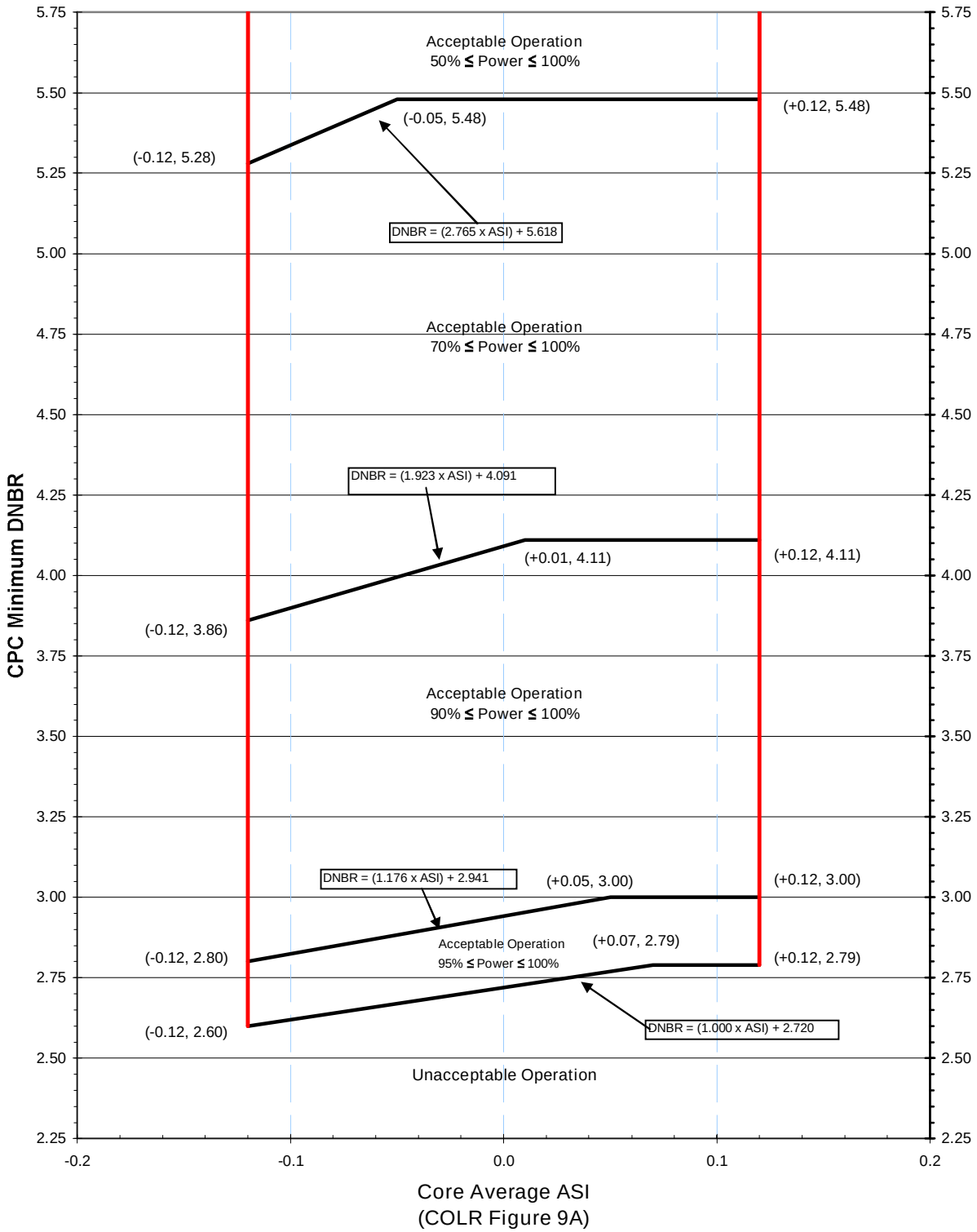


Core Average ASI
(COLR Figure 8A)

Allowable DNBR with No CEAC(s) Operable
(COLSS Out of Service)



Subset of Allowable DNBR with No CEAC(s) Operable
(COLSS Out of Service)



REQUEST/APPROVAL PAGE

SAFETY RELATED
PROCEDURE

Normal Review Class (check one):
☒ **OSRC**
☐ **QUALIFIED REVIEW**

PROCEDURE NUMBER: UNT-006-010		REVISION: 302

TITLE: Event Notification and Reporting

PROCEDURE OWNER (Position Title): Bob Murillo / Licensing

TERM (check one): ☒ **PERMANENT** ☐ **TEMPORARY**

Effective Date / Milestone (if applicable): 04/01/2010

Expiration Date / Milestone (if applicable):

PROCEDURE ACTION (check one):

☒ Revision ☐ Deletion ☐ New Procedure

DESCRIPTION AND JUSTIFICATION:

Page 1 – The level of use for this procedure is changed from INFORMATION USE to REFERENCE USE, per Licensing Manager. Updated LIST OF EFFECTIVE PAGES.

Throughout the procedure editorial changes were made to references clarifying fleet procedure designation (EN-...).

References to deleted site specific procedures were changed to that of the respective corporate procedure. Added associated Sub-Reference 2.3.1, 10CFR26.719, Significant FFD policy violations or programmatic failures.

(See Continuation sheet)

☒ Request/Approval Page Continuation Sheet(s) attached.

REVIEW PROCESS (check one): ☐ Normal ☒ Editorial Correction (Revisions only) ☐ Technical Verification (Revisions only)

REVIEW AND APPROVAL ACTIVITIES				PRINT NAME OR SIGNATURE	DATE
PREPARER				Joe Williams	02/20/2010
EC SUPERVISOR	Administrative Review and Approval			(sign)	3/23/10
CROSS-DISCIPLINE and INTERNAL REVIEWS (List Groups or Functions)	Operations				3/23/10
	Training			Ronald Bailey	2/20/2010
	Licensing			Jim Pollock	2/22/2010
				N/A	
PROCESS APPLICABILITY DETERMINATION	Performed ☐	PA Exclusion ☐		N/A	
TECHNICAL	Review ☐	Verification ☐		N/A	
QUALIFIED REVIEWER	Review ☐			N/A	
GROUP/DEPT. HEAD	Review ☐	Approval ☒		(sign)	3/23/10
GM, PLANT OPERATIONS	Review ☐	Approval ☐		(sign) N/A	
VICE PRESIDENT, OPERATIONS		Approval ☐		(sign) N/A	

CONTINUATION SHEET

PROCEDURE/INSTRUCTION NUMBER: UNT-006-010		REVISION: 302

DESCRIPTION AND JUSTIFICATION (continued):

Att. 7.2 (Reactor Operation Related Notification Requirements):

- Modified 11. to clarify the reportability of losing capacity of TSC as an Emergency Response Facility per (Ref. CR-WF3-2009-3108).
- Deleted the word "planned" from the sentence in 11. "Performing *planned* maintenance on an offsite emergency response facility is not reportable if the facility can be returned to service promptly in event of an accident" on page 25. This is to match the wording in note #15 on page 76 of NUREG 1022 Rev. 2, to more closely align with regulatory guidance.
- Added clarifying statement to 9. "If during a normal reactor shutdown, conditions develop during the process of shutting down that require an unplanned reactor scram, the RPS actuation (manual or automatic) is reportable." (Ref. NUREG 1022).
- Added editorial clarification to 10. that the point of discovery for Containment leakage is not mode dependent, but is based on the overall total containment minimum pathway leak-rate specified in TS. An example was added with emphasis that "In evaluating LLRT failures for Containment, an investigation should be completed to ensure that containment leakage does not exceed 630,000 sccm."

Attachment 7.4 (Security Related Notification Requirements):

- Deleted the Event numbers referencing to NS-200 (old) Att. 9.3 since these numbers were incorrect in EN-NS-200 (new) Att. 9.3. Combined and revised Att. 7.4 items associated with sale/use of illegal drugs and Fitness for Duty to align with new rule 10 CFR 26.719(b) notification requirements [no longer 10 CFR 26.73(a)(1)].
- Renumbered Att. 7.4 item 10.(electrical system threat) as 9., as a result of combining 7. & 8. and deleted the word "faxed" since the DOE form is preferred to be emailed, per EN-NS-200 Att. 9.4.

ADDITIONAL CROSS-DISCIPLINE REVIEWS		PRINT NAME OR SIGNATURE	DATE
CROSS-DISCIPLINE and INTERNAL REVIEWS (List Groups, Functions, Positions, etc.)		N/A	
		N/A	
		N/A	
		N/A	
		N/A	

TABLE OF CONTENTS

LIST OF EFFECTIVE PAGES	
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REFERENCE USE

1.0 PURPOSE

1.1 To provide a comprehensive source of information to ensure that reporting requirements arising from the existence of an adverse condition are quickly, accurately, and completely identified. Reportable events have been categorized and sorted into event-driven functional areas (Attachments 7.2-7.7).

2.0 REFERENCES

2.1 10CFR20, “Standards for Protection Against Radiation.”

2.2 10CFR21, “Reporting of Defects and Noncompliance.”

2.3 10CFR26, “Fitness for Duty Programs.”

2.3.1 10CFR26.719, Significant FFD policy violations or programmatic failures.

2.4 10CFR50, “Domestic Licensing of Production and Utilization Facilities.”

2.4.1 10CFR50.72, “Immediate Notification Requirements for Operating Nuclear Power Reactors.”

2.4.2 10CFR50.73, “Licensee Event Report System.”

2.5 10CFR55, “Operators' Licenses.”

2.6 10CFR70, “Domestic Licensing of Special Nuclear Material.”

2.7 10CFR71, “Packaging and Transportation of Radioactive Material.”

2.8 10CFR73, “Physical Protection of Plants and Materials.”

2.9 10CFR95, “Facility Security Clearance and Safeguarding of National Security Information and Restricted Data.”

2.10 33CFR64, “Marking of Structures, Sunken Vessels and other Obstructions.”

2.11 33CFR66, “Private Aids to Navigation.”

2.12 33CFR153, “Control of Pollution by Oil and Hazardous Substances, Discharge Removal.”

2.13 40CFR110, “Discharge of Oil.”

2.14 40CFR117, “Determination of Reportable Quantities for Hazardous Substances.”

2.15 40CFR190, “Environmental Radiation Protection Standards for Nuclear Power Operations.”

2.16 40CFR302, “Designations, Reportable Quantities, and Notification.”

2.17 US NRC Regulatory Guide 1.133, “Loose Part Detection Program for the Primary System of Light-Water-Cooled Reactors,” Revision 1, May 1981.

2.18 NUREG-0787, “Safety Evaluation Report related to the operation of Waterford Steam Electric Station Unit No. 3,” July 1981, 22.2 (II.K.3.3).

2.19 NUREG-0737, “Three Mile Island-2 Action Plan Requirements for Applicants for an Operating License,” Item II.K.3.17, “Report on Emergency Core-Cooling Systems Licensee Report and Proposed Technical Specification Changes.”

2.20 NUREG-1022, , “Event Reporting Guidelines,” Revision 2, October 2000.

2.21 Waterford 3 Technical Specifications

2.22 Waterford 3 Technical Requirements Manual Sections

2.23 Waterford 3 SES, Final Safety Analysis Report, Section 1.9.34.

2.24 EN-LI-101, “10CFR50.59 Review Program.”

2.25 EN-LI-102, “Corrective Action Process.”

2.26 W2.301, “Identification, Evaluation, and Reporting Process for 10CFR21 Compliance.”

2.27 EN-NS-204, “Handling of Safeguards Information.”

2.28 EN-EV-106, “Waste Management Program”.

2.29 EN-EV-113, “Drum Control Program”.

2.30 UNT-005-013, “Fire Protection Program.”

2.31 UNT-005-014, “Offsite Dose Calculation Manual.”

2.32 OP-100-014, “Technical Specification and Technical Requirements Manual Compliance.”

2.33 EN-NS-200, “Security Reporting Requirements.”

2.34 Environmental Protection Plan

2.35 State of Louisiana, Department of Environmental Quality, “Notification Regulations and Procedures for Unauthorized Discharges” (as amended August, 1993).

2.36 Private Aids to Navigation Application.

2.37 NRC Letter dated March 26, 1993, “Reporting Reactor Power Cutback System Actuation,” (ILN 93-0058).

2.38 LPDES Permit No. LA0007374.

2.39 Waterford 3 letter W3A89-0155 dated August 2, 1989, from J.R. McGaha to Distribution, “Operating Waterford 3 at the License Limit.”

2.40 NRC letter dated August 22, 1980 from E.L. Jordan, “Licensed Power Level.”

2.41 EP-001-001, “Recognition and Classification of Emergency Conditions.”

2.42 EP-002-010, “Notifications and Communications.”

2.43 UNT-007-064, “Hazardous Materials Emergency Response Plan and SPCC.”

2.44 Waterford 3 Calculation No. EC-I01-002, “COLSS Secondary Calorimetric Measurement Uncertainty”

2.45 NRC SER to Waterford 3, dated April 3, 2003 entitled “Waterford Steam Electric Station , Unit 3 – Issuance of Amendment Re: Corrections and Clarifications to Various Technical Specification Pages (TAC No. MB7150), Amendment 188.

2.46 Engineering Request ER-W3-2004-0377, Roll up ER to address drawing discrepancies

2.47 Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No.165 to Facility Operating License No. NPF-29 Grand Gulf Nuclear Station , Unit 1, Docket No. 50-416, Related to Amendment No. 138 to Facility Operating License No. NPF-47 River Bend Station, Unit 1 Docket No. 50-458, And Related to Amendment No. 193 to Facility Operating License No. NPF-38 Waterford Steam Electric Station, Unit 3, Docket No. 50-382 Entergy Operations, INC..ET AL.

2.48 Permit modification, Waterford III, Entergy Operations Inc., Killona, St. Charles Parish, Louisiana. Air Emission Permit No. 2520-00091-00, dated April 21, 2004.

2.49 North American Electric Reliability Corporation (NERC) Standard EOP-004, “Disturbance Reporting”

2.50 EN-LI-108, “Event Notification and Reporting”

3.0 DEFINITIONS

3.1 **Discharge** – The placing, releasing, spilling, percolating, draining, pumping, leaking, seeping, emitting, disposing, or other escaping of pollutants into the air, waters, subsurface or the ground.

3.2 **Hazardous Material** – Any substance deemed a hazardous material and included on the most recent list developed as a result of the Comprehensive Environmental Response Compensation Liability Act or certain substances included on the most recent U.S. Dept. of Transportation Hazardous Materials List. Hazardous material also means any substance deemed a physical or health hazard in the Occupational Safety and Health Act (OSHA) as found in 29CFRPart 1910.1200 et seq.

3.3 **Immediate Notification** – Appropriate verbal report to the NRC or other outside agencies as required by regulations, and/or the operating license.

3.4 **Invalid Actuations** – Those actuations which do not meet the criteria for being valid (refer to the definition of 'valid actuation'). Therefore, invalid actuations include those actuations that are not the result of valid signals and are not intentional manual actuations. Invalid actuations may include actuations resulting from instrument drift, spurious signals, human error, or signals which are the result of other signals. Examples of invalid actuations include actuations that result from: jarring a cabinet, an error in use of jumpers, an error in actuation of switches or controls, equipment failure, or radio frequency interference. (See Attachment 7.2 Item 9 for the systems affected)

3.5 **Licensed Material** – Source material, special nuclear material, or byproduct material received, possessed, used, transferred, or disposed of under a general or specific license issued by the NRC (i.e. Any material that contains detectable radioactivity).

3.6 **Reportable Event** – Any operation or condition occurring at Waterford 3 that is reportable to the NRC or other outside agencies.

3.7 **Safeguards Systems** – The equipment, personnel, and processes that make up the physical protection program necessary to meet 10CFR73 requirements.

3.8 **Source Material** – Uranium or thorium or any combination thereof in any physical or chemical form not including special nuclear material.

3.9 **Special Nuclear Material** – Plutonium, uranium-233, uranium enriched in the isotope 233 or in the isotope 235 or any material enriched by any of the foregoing, not including source material.

3.10 **Valid Actuations** – Those actuations that are initiated in response to actual plant conditions or parameters satisfying the requirements for initiation of the safety function of the system. A valid actuation must result from a valid signal or an intentional manual actuation, unless it is part of a preplanned test. (See Attachment 7.2 Item 9 for the systems affected.)

4.0 RESPONSIBILITIES

4.1 The Shift Manager (SM) is responsible for ensuring the completion of immediate actions and reviews as required by corporate procedure EN-LI-102 “Corrective Action Process”.

4.2 The Duty Plant Manager (DPM) is responsible for:

4.2.1 Assisting the SM in making immediate notification determinations when requested.

4.3 The Licensing Manager is responsible for:

4.3.1 Assisting the SM in performing the immediate reportability determination, if requested.

4.3.2 Review of and concurrence with the SM’s immediate reportability determination.

4.3.3 Approving final reportability determinations performed to identify notification requirements other than immediate reporting requirements.

4.3.4 Approving final determinations for reportability under 10CFR21.

4.3.5 Coordinating the development of information necessary to retract notifications made to agencies if Licensing’s reportability determination concludes that the initial notification was not required.

4.4 The Environmental Specialist is responsible for:

4.4.1 Assisting the SM in performing the immediate reportability determination for conditions involving hazardous material or environmental considerations.

4.4.2 Review of and concurrence with the SM’s immediate reportability determination for conditions involving hazardous material or environmental considerations.

4.4.3 Preparation and submittal of follow-up reports for conditions involving hazardous material or environmental considerations.

4.5 The General Manager Plant Operations (GMPO) ultimately responsible for making required immediate reports and all environmental and hazardous material-related reports.

4.6 The Director Nuclear Safety Assurance is responsible for:

4.6.1 Maintenance of this procedure.

4.6.2 Identification of reporting requirements in areas that are not the responsibility of the GMPO.

4.7 Security Shift Supervisor is responsible for assisting the Shift Manager in determining the reportability of security or safeguards events, as required.

5.0 PROCEDURE

5.1 Immediate Reportability Review
Upon receiving a Condition Report, the SM should:

> **NOTE**
> If the Emergency Plan is implemented, follow the instructions in the plan for making appropriate notifications. Note that EN-LI-102 requires that a Condition Report be generated for all events that are reportable to the NRC.

5.1.1 Perform an immediate reportability determination by comparing the circumstances of the event to the criteria in Attachments 7.2 through 7.7 as appropriate.

> **NOTE**
> For security or safeguards information-related events, this procedure includes only the immediate and follow-up reporting requirements (i.e., those events for which an ENS notification and follow-up Security Incident Report are required). Certain less significant events are required to be logged by Security in accordance with Appendix G to 10CFR73. The Security Shift Supervisor should be informed of all security or safeguards information-related events so that the event or condition may be logged if necessary.

5.1.2 For those events involving Safeguards Information or the Security System, notify the Security Shift Supervisor of the condition so that necessary compensatory measures may be implemented and the event documented in the Security log if necessary.

5.1.3 If the SM's reportability review indicates that:

a. The event is REPORTABLE; go to Section 5.2.

b. Event reportability is INDETERMINATE; go to Section 5.3.

c. The event is NOT REPORTABLE; go to Section 5.4.

5.2 Event Notification: Report Required

The SM's reportability determination indicates that the NRC or other outside agency should be notified. The SM should:

5.2.1 Notify the Duty Plant Manager of the circumstances of the event and the basis for the reportability determination.

5.2.2 Complete an Event Notification Worksheet if the notification will be made to the NRC (Attachment 7.1).

To assist the NRC in determining their proper response to an event, notifications should include enough information so that an independent assessment of the event can be made. If possible, the notification should include:

- **current conditions**
- **cause of the event**
- **safety significance of the event**
- **possible near-term effects of the event**

CAUTION

Events involving Safeguards Information may be described in general terms when making telephone notifications. Actual Safeguards Information, however, should not be discussed over non-secure telephone lines. Additionally, Safeguards Information should not ordinarily be transmitted with a FAX machine.

5.2.3 Notify the NRC and/or appropriate outside agencies within the specified reporting time limit.

- **In general, the reporting time clock starts at the time of the event or the discovery of the condition.**
- **"Discovery of the condition" generally refers to the time that management responsible for the event is informed that a condition exists or that there is reasonable expectation that the condition exists.**

- **Whether or not there is "reasonable expectation" that a condition exists is a judgment call which should be made by the SM based on the best available information at the time. One conservative interpretation provided by the NRC is that there is "reasonable expectation" that a condition exists if the condition is significant enough to warrant entry into a Technical Specification LCO or to take other compensatory measures.**

- **The Emergency Notification System (ENS) is the primary method of notification to the NRC. This system should be used if it is available.**

5.2.4 During the course of the event, immediately report any further degradation in the level of safety of the plant or other worsening plant conditions. Also, report results of evaluations or assessments of plant conditions, the effectiveness of response or protective measures taken and information related to plant behavior that is not understood.

NOTE
The NRC Resident Inspector should be notified as soon as practical of all notifications made to the NRC Operations Center, regardless of the event time or significance. The Resident Inspector should be contacted at home if the NRC notification occurs after normal business hours.

5.2.5 Complete the 'Immediate Reportability Determination' section of the Condition Report. Attach the Event Notification Worksheet, the Post Trip Review, or any other pertinent information to the Condition Report.

5.2.6 Process the Condition Report in accordance with Corporate Procedure EN-LI-102, "Corrective Action Process."

5.3 Event Notification: Reportability is Indeterminate

The information available to the SM is not sufficient to clearly establish whether a particular event is or is not reportable. The SM should:

> **NOTE**
>
> The reportability of an event or condition may be considered to be 'indeterminate' so long as a reasonable expectation exists that further evaluation will confirm that the event or condition is not reportable. If this reasonable expectation does not exist, the event should be reported.

5.3.1 Request assistance in clarifying specific reporting requirements if necessary.

5.3.2 Attempt to obtain additional information about the event if it is available.

> **NOTE**
>
> The time clock for reporting starts when management responsible for reporting is informed of the existence of the condition or event. The SM should always err on the conservative side when making reportability determinations. If subsequent review indicates that the initial reportability determination was overly conservative, the notification may be retracted without adverse consequences.

5.3.3 Complete the 'Immediate Reportability Determination' section of the Condition Report to indicate 'indeterminate' if this additional information does not resolve the reportability question.

5.3.4 Process the Condition Report in accordance with Corporate Procedure EN-LI-102, "Corrective Action Process."

5.4 Event Notification: Report Not Required

The SM's reportability determination indicates that the event is not reportable. The SM should:

5.4.1 Complete the 'Immediate Reportability Determination' section of the Condition Report to indicate that the event or condition is not reportable.

5.4.2 Process the Condition Report in accordance with Corporate Procedure EN-LI-102, "Corrective Action Process."

5.5 Licensing Reportability Review

Upon receiving a Condition Report, Licensing should:

NOTE
If Licensing's final reportability review indicates that the initial notification was overly conservative, the event notification should be retracted. Consideration should also be given to correcting factual errors (i.e., incorrect reporting requirement specified), particularly when a follow-up written report (LER) will not be provided.

5.5.1 Review the SM's immediate reportability determination to ensure that notification requirements have been properly identified and that appropriate reports have been made.

5.5.2 Perform a reportability review for Condition Reports as specified by Corporate Procedure EN-LI-102, "Corrective Action Process," except those dealing with hazardous material or environmental matters.

5.5.3 Document the basis of the reportability determination in the Condition Report.

5.5.4 In cases where the Condition Report identifies a potentially reportable condition (for example, past operability concern) but does not contain sufficient information to conclusively determine reportability, the reportability determination shall be classified as "Indeterminate" until the additional information required to determine reportability (for example, engineering evaluation, past operability) is received and reviewed.

5.5.4.1 To obtain the information necessary to determine reportability, the Manager, Licensing may issue a corrective action to the appropriate department in accordance with EN-LI-102.

5.5.4.2 The due date for the action item should be commensurate with the safety significance of the condition. It should be limited to 30 days from the date of initiation of the Condition Report unless there is reasonable assurance that the condition will ultimately be determined non-reportable.

5.5.5 Process Condition Reports in accordance with Corporate Procedure EN-LI-102, "Corrective Action Process."

5.6 Environmental Reportability Review

Upon receiving a Condition Report that deals with hazardous materials or an environmental condition, Environmental should:

5.6.1 Review the SM's immediate reportability determination to ensure that notification requirements have been properly identified and that appropriate reports have been made.

5.6.2 Perform a reportability review for all Condition Reports dealing with hazardous material or environmental matters.

5.6.3 Document the basis of the reportability determination in the Condition Report.

5.6.4 Process Condition Reports in accordance with Corporate Procedure EN-LI-102, "Corrective Action Process."

5.7 Event Notification Retraction

If an event notification is to be retracted or corrected:

5.7.1. Licensing should coordinate the development of information to retract or revise the original notification.

5.7.2. Licensing should prepare an Event Notification Worksheet (Attachment 7.1). Deliver the completed worksheet to the SM.

5.7.2.1 The Event Notification Worksheet should provide the technical basis for any changes that are made in the retraction or revised notification. This is particularly important for event retractions.

NOTE
The NRC Resident Inspector should be notified as soon as practical of all notifications made to the NRC Operations Center, regardless of the event time or significance. The Resident Inspector should be contacted at home if the NRC notification occurs after normal business hours.

5.7.3. The SM should process the Event Notification Worksheet in accordance with the guidance of Section 5.2, modifying that guidance to fit the circumstances of the situation.

5.7.3.1 As a general rule, individuals or groups that would normally be apprised of an event should be notified that a retraction or correction is being made.

6.0 RECORDS

NONE

7.0 ATTACHMENTS

7.1 Sample Event Notification Worksheet (NRC Form 361).

7.2 Reactor Operation-Related Notification Requirements.

7.3 Radiation-Related Notification Requirements.

7.4 Security-Related Notification Requirements.

7.5 Miscellaneous Notification Requirements.

7.6 Environmental Notification Requirements.

7.7 Hazardous Material Notification Requirements.

7.1 Sample Event Notification Worksheet (NRC Form 361)

PAGE 1 OF 2

NRC FORM 361
(12-2000)

U.S. NUCLEAR REGULATORY COMMISSION
OPERATIONS CENTER

REACTOR PLANT
EVENT NOTIFICATION WORKSHEET

EN #

NRC OPERATION TELEPHONE NUMBER: PRIMARY -- 301-816-5100 or 800-532-3469*, BACKUPS – [1st] 301-951-0550 or 800-449-3694*, [2nd] 301-415-0550 and [3rd] 301-415-0553
*Licensees who maintain their own ETS are provided these telephone numbers.

NOTIFICATION TIME	FACILITY OR ORGANIZATION	UNIT	NAME OF CALLER	CALL BACK #

EVENT TIME & ZONE	EVENT DATE	POWER/MODE BEFORE	POWER/MODE AFTER

EVENT CLASSIFICATIONS		1-Hr. Non-Emergency 10 CFR 50.72(b)(1)					
	GENERAL EMERGENCY GEN/AAEC		TS Deviation ADEV		(v)(A)	Safe S/D Capability	AINA
	SITE AREA EMERGENCY SIT/AAEC	4-Hr. Non-Emergency 10 CFR 50.72(b)(2)			(v)(B)	RHR Capability	AINB
	ALERT ALE/AAEC	(i)	TS Required S/D ASHU		(v)(C)	Control of Rad Release	AINC
	UNUSUAL EVENT UNU/AAEC	(iv)(A)	ECCS Discharge to RCS ACCS		(v)(D)	Accident Mitigation	AIND
	50.72 NON-EMERGENCY (see next columns)	(iv)(B)	RPS Actuation (scram) ARPS		(xii)	Offsite Medical	AMED
	PHYSICAL SECURITY (73.71) DDDD	(xi)	Offsite Notification APRE		(xiii)	Loss Comm/Asmt/Resp	ACOM
	MATERIAL/EXPOSURE B???	8-Hr. Non-Emergency 10 CFR 50.72(b)(3)		60-Day Optional 10 CFR 50.73(a)(1)			
	FITNESS FOR DUTY HFIT	(ii)(A)	Degraded Condition ADEG			Invalid Specified System Actuation	AINV
	OTHER UNSPECIFIED REQMT. (see last column)	(ii)(B)	Unanalyzed Condition AUNA	Other Unspecified Requirement (Identify)			
	INFORMATION ONLY NINF	(iv)(A)	Specified System Actuation AESF				NONR
							NONR

DESCRIPTION

Include: Systems affected, actuations and their initiating signals, causes, effect of event on plant, actions taken or planned, etc. *(Continue on back)*

NOTIFICATIONS	YES	NO	WILL BE
NRC RESIDENT			
STATE(s)			
LOCAL			
OTHER GOV AGENCIES			
MEDIA/PRESS RELEASE			

ANYTHING UNUSUAL OR NOT UNDERSTOOD?	☐ YES (Explain above)	☐ NO
DID ALL SYSTEMS FUNCTION AS REQUIRED?	☐ YES	☐ NO (Explain above)
MODE OF OPERATION UNTIL CORRECTED:	ESTIMATED RESTART DATE:	ADDITIONAL INFO ON BACK ☐ YES ☐ NO

NRC FORM 361 (12-2000)

PRINTED ON RECYCLED PAPER

ADDITIONAL INFORMATION

RADIOLOGICAL RELEASES: CHECK OR FILL IN APPLICABLE ITEMS ***(specific details/explanations should be covered in event description)***

	LIQUID RELEASE		GASEOUS RELEASE		UNPLANNED RELEASE		PLANNED RELEASE		ONGOING		TERMINATED
	MONITORED		UNMONITORED		OFFSITE RELEASE		T. S. EXCEEDED		RM ALARMS		AREAS EVACUATED
	PERSONNEL EXPOSED OR CONTAMINATED				OFFSITE PROTECTIVE ACTIONS RECOMMENDED				*State release path in description		

	Release Rate (Ci/sec)	% T. S. LIMIT	HOO GUIDE	Total Activity (Ci)	% T. S. LIMIT	HOO GUIDE
Noble Gas			**0.1 Ci/sec**			**1000 Ci**
Iodine			**10 uCi/sec**			**0.01 Ci**
Particulate			**1 uCi/sec**			**1 mCi**
Liquid ***(excluding tritium and dissolved noble gases)***			**10 uCi/min**			**0.1 Ci**
Liquid (tritium)			**0.2 Ci/min**			**5 Ci**
Total Activity						

	PLANT STACK	CONDENSER/AIR EJECTOR	MAIN STEAM LINE	SG BLOWDOWN	OTHER
RAD MONITOR READINGS					
ALARM SETPOINTS					
% T. S. LIMIT ***(if applicable)***					

RCS OR SG TUBE LEAKS: CHECK OR FILL IN APPLICABLE ITEMS: ***(specific details/explanations should be covered in event description)***

LOCATION OF THE LEAK *(e.g., SG #, valve, pipe, etc.)*

LEAK RATE	UNITS: gpm/gpd	T. S. LIMITS	SUDDEN OR LONG-TERM DEVELOPMENT
LEAK START DATE	TIME	**COOLANT ACTIVITY AND UNITS:** PRIMARY	SECONDARY

LIST OF SAFETY RELATED EQUIPMENT NOT OPERATIONAL

EVENT DESCRIPTION *(Continued from front)*

7.2 Reactor Operation-Related Notification Requirements

1. Emergency Declaration. **Declaration of any of the emergency classes specified in the approved Emergency Plan.**

Initiation: One hour notification **10CFR50.72(a)(1)(i)**

Discussion (Item 1):
The use of the Notification Message Form in EP-002-010 when an emergency is declared meets this notification requirement.

Occasionally, a licensee may discover that an event or condition had existed which met the emergency plan criteria but no emergency had been declared and the basis for the emergency class no longer exists at the time of this discovery. This may be due to a rapidly concluded event or an oversight in the emergency classification made during the event or it may be determined during a post-event review. Frequently, in cases of this nature, which were discovered after the fact, licensees have declared the emergency class, immediately terminated the emergency class and then made the appropriate notifications. However, the staff does not consider actual declaration of the emergency class to be necessary in these circumstances; an ENS notification (or an ENS update if the event was previously reported but misclassified) within one hour of the discovery of the undeclared (or misclassified) event will provide an acceptable alternative. State and local agencies are not notified for this type of event or condition.

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2. Tech. Spec. Deviation. **Deviation from Technical Specifications pursuant to 10CFR50.54(x).**

One hour notification via ENS **10CFR50.72(b)(1)(i)**

60 Day LER **10CFR50.73(a)(2)(i)(C)**

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3. External conditions that pose an actual threat. **Natural phenomenon or other external condition that poses an actual threat to the safety of the nuclear power plant or significantly hampers site personnel in the performance of duties necessary for the safe operation of the plant.**

For immediate reporting, refer to Item 1, Emergency Declaration.
Previous requirement has been deleted.

60 Day LER **10CFR50.73(a)(2)(iii)**

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4. Internal conditions that pose an actual threat. **Event or condition that poses an actual threat to the safety of the nuclear power plant or significantly hampers site personnel in the performance of duties necessary for the safe operation of the nuclear power plant, including fires, toxic gas releases, or radioactive releases.**

For immediate reporting, refer to Item 1, Emergency Declaration.
Previous requirement has been deleted.

60 Day LER	**10CFR50.73(a)(2)(x)**

5. Safety Limit Exceeded**. Safety limit specified in the plant's Technical Specifications 2.1 exceeded.**

4 hour notification via ENS	**10CFR50.72(b)(2)**
60 Day LER	**10CFR50.73**
If any safety limit is exceeded, the reactor must be shutdown. Operation must not be resumed until authorized by the Commission.	**10CFR50.36(c)(1)**

Discussion (Item 5) Vice President-Nuclear Operations and SRC shall be notified within 24 hours.

6. Safety System Failure. **Automatic safety system (a system needed to maintain safety limits) fails to function as required at any time during operation.**

See Item 12 for the reporting requirements for prevented safety function	**10CFR50.36(c)(1)(ii)**

7. Plant Shutdown. **Plant Shutdown required by Technical Specifications.**

Initiation: 4 hour notification	**10CFR50.72(b)(2)(i)**
Completion: 60 Day LER	**10CFR50.73(a)(2)(i)(A)**

Discussion (Item 7): Initiation of a plant shutdown includes action to start reducing reactor power, i.e., adding negative reactivity to achieve a nuclear plant shutdown required by TS. This includes initiation due to expected inability to restore equipment prior to exceeding the LCO action time.

Completion of a plant shutdown is defined as the point in time during a TS required shutdown when the plant enters the first shutdown condition required by a LCO. An LER is not required under this section if a failure was or could have been corrected before a plant has completed shutdown.

8. ECCS Discharge. **Event or condition that resulted <u>or</u> should have resulted in the Emergency Core Cooling System (ECCS) being discharged into the reactor coolant system as a result of a valid signal, except when the actuation results from and is part of a pre-planned sequence during testing or reactor operations.**

4 hour notification via ENS **10CFR50.72(b)(2)(iv)(A)**

60 Day LER **10CFR50.73(a)(2)(iv)**

Reference: See Item 19b

9. System Actuation.

a. **Actuation of the Reactor Protection System (RPS) when the reactor is critical, except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation. This includes valid <u>or</u> invalid system actuations.**

If an operator were to manually scram the reactor in anticipation of receiving an automatic reactor scram, this would be reportable just as an automatic scram would be reportable. If during a normal (or planned) reactor shutdown, conditions develop during the process of shutting down that require an unplanned reactor scram, the RPS actuation (manual or automatic) is reportable.
(Reference NUREG 1022, Section 3.2.6, "System Actuation")

4 hour notification via ENS **10CFR50.72(b)(2)(iv)(B)**

b. **A Valid Actuation of any of the following systems, except when the the actuation results from and is part of a pre-planned sequence during testing or reactor operation.**

1. **RPS, while the reactor is NOT critical (see Item 9.a for RPS actuation while critical);**
2. **Containment isolation signals affecting CIVs (CIAS) or MSIVs(CIAS or MSIS);**
3. **Emergency core cooling systems (ECCS): HPSI and LPSI;**
4. **Emergency feedwater systems;**
5. **Containment heat removal and depressurization systems: containment spray and fan cooler systems;**
6. **Emergency AC electrical power systems: EDGs.**

8 hour notification via ENS **10CFR50.72(b)(3)(iv)(A)**

NOTE

An invalid actuation reported under 10CFR50.73(a)(2)(iv) may be provided to the NRC via telephone notification to the NRC Operations Center within 60 days after discovery of the event instead of submitting a written LER. Invalid actuations are no longer reportable under 10CFR50.72, with the exception of actuations of the RPS system (see Item 9a).

c. **Any event or condition that resulted in manual or automatic actuation of any system listed in Item 9 section b, except when:**

- **The actuation resulted from and was part of a pre-planned sequence during testing or reactor operation; or**
- **The actuation was invalid and;**
  - **Occurred while the system was properly removed from service; or**
  - **Occurred after the safety function had been already completed.**

60 Day LER **10CFR50.73(a)(2)(iv)(A)**

Discussion (Item 9): Any invalid system actuation that reveals a defect in the system so that the system failed or would fail to perform its intended function is reportable. These invalid system actuations may be provided the NRC via the ENS hotline. The report should provide the following information: (a) specific train(s) and system(s) that were actuated (b) whether each train actuation was complete or partial (c) whether or not the system started and functioned successfully.

Isolation of the control room due to a Toxic Gas Actuation is not reportable unless conditions exist that cause the declaration of an Emergency Class under the criteria of 10CFR50.72(a)(1)(i).

Actuation of multichannel actuation systems is defined as actuation of enough channels to complete the minimum actuation logic. Therefore, single channel actuations, whether caused by failures or otherwise, are not reportable if they do not complete the minimum actuation logic. Note, however, that if only a single logic channel actuates when, in fact, the system should have actuated in response to plant parameters, this would be reportable under this criteria as a valid actuation and potentially other criteria such as loss of safety function.

10. Degraded Condition. **Event or condition that results in either:**

- **the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded;**
- **the nuclear power plant being in an unanalyzed condition that significantly degrades plant safety**

8 hour notification via ENS	**10CFR50.72(b)(3)(ii)**
60 Day LER	**10CFR50.73(a)(2)(ii)**

Discussion (Item 10):

This criteria applies to all modes of plant operation.

Principal safety barriers include: (1) fuel cladding, (2) reactor coolant system pressure boundary, and (3) containment. The specific safety function performed by these principal safety barriers is the protection of public health and safety through limiting the release of radioactive material.

Examples of seriously degraded primary safety barriers include:

- **Fuel cladding failures in the reactor, or in the storage pool**
- **Welding or material defects in the primary coolant system**
- **Serious Steam Generator tube degredation**
- **Loss of containment fuction or integrity, including containment leak rate tests where the total containment as-found, minimum pathway leak rate exceeds the TS value. CR-WF3-2009-7273 documents that LLRT test failures were assigned leakage rates of 630,000 sccm to penetrations in three Condition Reports. The reportability reviews determined the conditions were not reportable even though the As-Found Mini-path Total Leakage for the Containment has a limit of 630,000 sccm. A review determined that an 8 hour report to the NRC should have been made as each CR was written. In evaluating LLRT failures for Containment, an investigation should be completed to ensure that containment leakage does not exceed 630,000 sccm. If the limit is determined to be exceeded, an 8 hour report is required.**

Examples of unanalyzed conditions that significantly affect plant safety include:

- **The discovery that a system required to meet the single failure criterion does not do so;**
- **If fire barriers are found to be missing, such that the required degree of seperation for redundant safe shutdown trains is lacking.**

NOTE

Per the guidance of NUREG-1022, Rev. 2 "Event Reporting Guidelines", if the Operations Center notifies the licensee that an ENS line is inoperable, there is no need for a subsequent licensee notification.

11. Major loss of emergency assessment capability, off-site response capability, or offsite communications capability.

8 hour notification via ENS **10CFR50.72(b)(3)(xiii)**

Discussion (Item 11): Examples of a 'Major loss of emergency assessment capability' include the loss of SPDS for greater than 12 hours or the loss of a significant portion of control room indication. The loss of SPDS concurrent with the loss of other emergency assessment equipment may be immediately reportable. Engineering judgement should be used in these instances to determine if the condition constitutes a "major loss" (Ref. NUREG-1022).

A 10CFR50.72 notification should be made for situations where the Technical Support Center (TSC) is not available for use because of an UNPLANNED event. In the case where the TSC is already staffed and personnel remain in the TSC until CO2 reaches a certain level, this notification should not be reported until it is known the TSC can not be used as a TSC because of reduced staffing. Limiting TSC response capacity and directing personnel to the EOF for use as the backup TSC is not reportable under 10CFR50.72(b)(3)(xiii) when the condition occurs as a result of a PLANNED activity, such as planned maintenance or monitor calibrations (Ref. CR-WF3-2009-3108).

A 'Major loss of offsite response capability' includes those events that would significantly impair the fulfillment of the licensee's approved emergency plan for <u>other than a short time</u>. Loss of offsite response capability may typically include the loss of plant access, emergency offsite response facilities, or public prompt notification systems, including sirens and other alerting systems. Performing maintenance on an offsite emergency response facility is not reportable if the facility can be returned to service promptly in event of an accident.

Although the loss of a single siren for a short time is not a major loss of offsite response capabability, the loss of a large number of sirens, other alerting systems, or more importantly, the lost capability to alert a large segment of the population for 1 hour would warrant immediate notification.

It should be noted that Waterford 3 has determined that in the event that 25% of the sirens being inoperable is considered a major loss of of offsite response capability and would be reportable under this criteria. An example of a 'Major loss of communications capability' is the failure of the Emergency Notification System (ENS) and/or Health Physics Network (HPN). Should an ENS and/or HPN failure occur, the Operations Center in Bethesda, Md. should be so informed by calling (301) 816-5100 or one of the following backup numbers: Backup (301) 951-0550; 2nd Backup (301) 415-0550, and 3rd Backup (301) 415-0553. Emergency Response Data System (ERDS) failures alone are not reportable per 10CFR50.72 or 10CFR50.73. However, if NRC-supplied communications equipment is inoperable, the NRC should be informed so repairs can be ordered.

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12. Prevented Safety Function. **Event that at the time of discovery could have prevented the fulfillment of the safety function of structures or systems that are needed to**

    A. **shutdown the reactor and maintain it in a safe shutdown condition**

    B. **remove residual heat**

    C. **control the release of radioactive material**

    D. **mitigate the consequences of an accident**

8 hour notification via ENS	**10CFR50.72(b)(3)(v)**
60 Day LER	**10CFR50.73(a)(2)(v)**

Discussion (Item 12): Even though the loss of the safety function does not actually have to occur, the condition would be reportable if it reasonably might have occurred. For example, single failure in a component required to perform any of the functions listed in this requirement is reportable when there is reason to expect that the failure mechanism is one that could occur in the redundant component. However, individual component failures need not be reported if redundant equipment in the same system was operable and available to perform the required safety function.

The systems included in the scope of these criteria includes systems required by TS to be operable to perform one of the four functions stated in A – D above.

It is not necessary to assume an additional single failure in a system, however it is necessary to consider other existing plant conditions.

The events covered in this section may include one or more procedural personnel errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies.

13. Loose part in reactor (confirmed).

24 hour notification via ENS	**Reg. Guide 1.133**
30 Day LER	**Reg. Guide 1.133**

14. Failure of spring-loaded pressurizer safety valve to close.

24 hour notification via ENS	**SER 22.2 (II.K.3.3)**
30 Day LER	**SER 22.2 (II.K.3.3)**

15. Power Limit Exceeded. **Plant operation that results in either:**

- **Reactor core power greater than 102%, or**
- **The average power level over any 8-hour period greater than 100.00% reactor core power (3716 MWt full steady-state licensed power level)**

24 hour notification via ENS	**NPF-38**
30 Day LER	**NPF-38**

Discussion (Item 15): In accordance with NRC Inspection Procedure 61706, it is permissible to briefly exceed the “full, steady-state licensed power level” by as much as 2% for as long as 15 minutes. In no case should 102% power be exceeded, but lesser power “excursions” for longer periods should be allowed (e.g. 1% excess for 30 minutes and ½% for 1 hour should be allowed). There are no limits on the number of times these excursions may occur or the time interval that must separate such excursions. (ref. 2.38, 2.39, 2.43)

16. Prohibited Condition/Operation. **Operation or condition prohibited by Technical Specifications, except when:.**

- **The Technical Specification is administrative in nature,**
- **The event consisted solely of a case of a late surveillance test where the oversight was corrected, the test was performed and the equipment was found to be capable of perfoming its specified safety functions; or**
- **The Technical Specificaiton was revised prior to discovery of the event such that the operation or condition was no longer prohibited at the time of discovery of the event.**

60 Day LER	**10CFR50.73(a)(2)(i)(B)**
	10CFR50.36(c)(2)

Discussion (Item 16): Entry into TS 3.0.3 is not necessarily reportable under this criterion. However, it should be considered reportable if this condition is not corrected within an hour.

17. Common Cause Failure. **Did a single cause or condition cause at least one independent train or channel to become inoperable in multiple systems or two independent trains or channels to become inoperable in a single system designed to:**

- **shutdown the reactor and maintain it in a safe shutdown condition**
- **remove residual heat**
- **control the release of radioactive material**
- **mitigate the consequences of an accident**

60 Day LER **10CFR50.73(a)(2)(vii)**

Discussion (Item 17): To be reportable under this criterion, the event or failure must result in or involve the failure of independent portions of more than one train or channel in the same or different systems. For example, if a cause or condition caused components in Train "A" and "B" of a single system to become inoperable, even if additional trains (e.g., Train "C") were still available, the event must be reported. In addition, if the cause or condition caused components in Train "A" of one system and in Train "B" of another system (i.e., a train that is assumed in the safety analysis to be independent) to become inoperable, the event must be reported. However, if a cause or condition caused components in Train "A" of one system and Train "A" of another system to fail (i.e., trains that are not assumed in the safety analysis to be independent), the event need not be reported unless it meets one or more of the other criteria in this section.

Common causes may include such factors as high ambient temperatures, heat up from energization, inadequeate preventative maintenance, oil contaimination or air systems, incorrect lubrication, use of non-qualified components or manufacturing design flaws.

18. Single Cause that Could Have Prevented Fulfillment of the Safety Functions of Trains or Channels in Different Systems. **Could a single cause or condition have prevented the fullfillment of a safety function for** two or more trains **or channels in** different systems **that are needed to:**

- **shutdown the reactor and maintain it in a safe shutdown condition**
- **remove residual heat**
- **control the release of radioactive material**
- **mitigate the consequences of an accident**

Except when, the event results from:

- **a shared dependency among trains or channels that is a natural or expected consequence of the approved plant design; or**
- **normal and expected wear or degradation.**

60 Day LER **10CFR50.73(a)(2)(ix)**

Discussion (Item 18): This may included cases of procedural error, equipment failure, and/or discovery of a design, analysis, fabrication, construction, and/or procedural inadequacy.

19. Technical Specification Required Reports

a. RCS pressure transient mitigated by SDC suction relief valves or RCS vent(s)

30 day Special Report (SR) **TS 3.4.8.3**

b. ECCS actuated:

i. Modes 1, 2, and 3 (when pressurizer pressure ≥ 1750 psia and RCS average temperature is ≥ 500F)

90 day SR **TS 3.5.2**

ii. Mode 3 (with pressurizer pressure < 1750 psia and the RCS average temperature is < 500F) and Mode 4

90 day SR **TS 3.5.3**

Reference: See item 8

c. Inoperable Instrumentation:

i. Seismic Monitoring

10 day SR if inoperable ≥ 30 days TRM 3.3.3.3

ii. Meteorological Monitoring

10 day SR if inoperable > 7 days TRM 3.3.3.4

iii. Loose Part Detection

10 day SR if inoperable > 30 days TRM 3.3.3.9

iv. Explosive Gas Monitoring Instrumentation (Table 3.3-13)

30 day SR if inoperable > 30 days TS 3.3.3.11

d. Seismic Event

10 day SR TRM 4.3.3.3.2

e. Radiation Monitoring Instrumentation

i. Effluent Accident Monitors (ref. Table 3.3-6)

14 day SR if inoperable > 7 days TS 3.3.3.1

ii. Steam Generator Blowdown Monitor

14 day SR if inoperable > 30 days TS 3.3.3.1

iii. CCW Radiation Monitors (A, B, A/B)

14 day SR if inoperable > 30 days TS 3.3.3.1

f. Accident Monitoring Instrumentation (See Table 3.3-10 and Bases for 3.3.3.6)

i. Required Channels not operable

14 day SR if inoperable > 30 days TS 3.3.3.6

ii. Reactor Vessel Level Monitoring System Minimum Channels not operable

14 day SR if inoperable > 7 days TS 3.3.3.6

7.3 Radiation-Related Notification Requirements

NOTE

The Radiation-Related Notification Requirements in Attachment 7.3 can be grouped into four major categories:

Personnel Exposure/Contamination – Items 1, 2, 3, 10, and 18.

Radioactive Releases – Items 4, 5, 11, 12, 13, 14, 16 and 19.

Loss/Theft of Radioactive Material – Items 6 and 15.

Material Shipment/Packaging Discrepancies – Items 7, 8, 9, and 17.

NOTE

If a radiological surveillance or operable radiation monitor required by the Offsite Dose Calculation Manual (ODCM), UNT-005-014, was not met, refer to the ODCM for reporting information (routine and special).

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1. Personnel Exposure/Contamination. **Events involving licensed material that may have caused or threatens to cause an individual to receive any of the following:**

- **a total effective dose equivalent of ≥ 25 rems**
- **an eye dose equivalent of ≥ 75 rem**
- **a shallow dose equivalent to the skin or extremities of ≥ 250 rads**

One hour notification via ENS	**10CFR20.2202(a)(1)**
One hour notification to NRC Region IV	**RG 1.16**
30 day LER	**10CFR20.2203(a)(1)**

Discussion (Item 1): This provision does not include planned special exposures that are reported under 10CFR20.2204. Inoperable radiation instrumentation is addressed in Attachment 7.2 Item 19.

Per 10CFR20.2203(c), occurrences reported in accordance with 10CFR50.73 do not need to be reported by a separate report under 10CFR20.2203(a)(1).

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2. Personnel Exposure/Contamination. **Events involving the loss of control of licensed material that may have caused, or threatens to cause, an individual to receive, in a period of 24 hours, any of the following:**

- **a total effective dose equivalent exceeding 5 rem**
- **an eye dose equivalent exceeding 15 rem**
- **a shallow dose equivalent to the skin or extremities exceeding 50 rem.**

One hour notification to NRC Region IV	**RG 1.16**
24 hour notification via ENS	**10CFR20.2202(b)(1)**
30 day LER	**10CFR20.2203(a)(1)**

Discussion (Item 3): This provision does not include planned special exposures that are reported under 10CFR20.2204.

Per 10CFR20.2203(c), occurrences reported in accordance with 10CFR50.73 do not need to be reported by a separate report under 10CFR20.2203(a)(1).

3. Personnel Exposure/Contamination. **Radiation dose to any individual in excess of any of the following:**

- **the occupational dose limits for adults in 10CFR20.1201.**
- **the occupational dose limits for a minor in 10CFR20.1207.**
- **the limits for an embryo/fetus of a declared pregnant woman in 10CFR20.1208.**
- **the limits for an individual member of the general public in 10CFR20.1301.**
- **any applicable limit in the license.**
- **the ALARA constraints for air emissions established under 10CFR20.1101(d).**

One hour notification to NRC Region IV	**RG 1.16**
30 day LER	**10CFR20.2203(a)(2)**

4. Radioactive Releases. **Events involving licensed material that may have caused or threatens to cause, the release of radioactive material, inside or outside of a retricted area, so that, had an individual been present for 24 hours, the individual could have received an intake five times the annual limit on intake (this provision does not apply to locations where personnel are not normally stationed during routine operations such as hot-cells or process enclosures).**

One hour notification via ENS	**10CFR20.2202(a)(2)**
24 hour telephone notification to Louisiana DEQ	**(225) 765-0160**
Radiation Protection Division	
7 day written follow-up to Louisiana DEQ	
30 day LER	**10CFR20.2203(a)(1)**

Discussion (Item 5): This provision does not include planned special exposures that are reported under 10CFR20.2204.

Per 10CFR20.2203(c), occurrences reported in accordance with 10CFR50.73 do not need to be reported by a separate report under 10CFR20.2203(a)(1).

If the release is associated with a radioactive spill outside the restricted area, then environmental reporting requirements also may apply. See Attachment 7.6.

5. Radioactive Releases. **Any offsite release of a hazardous substance in a quantity equal to or exceeding the reportable quantity as determined by 40CFR302 in any 24-hour period.**

Immediate notification to National Response Center	**(800) 424-8802**
30 minute notification to Department of Public Safety	**(225) 925-6595**
and Corrections, Office of State Police	
St. Charles Emergency Planning Committee	**(985) 783-5050**

6. Loss/Theft of Radioactive Material. **Loss or theft of licensed material in such quantity and under such circumstances that it appears that an exposure could result to persons in unrestricted areas.**

One hour notification via ENS	**10CFR20.2201(a)(i)**
	10CFR73.71(a)
30 day LER	**10CFR20.2201(b)(1)**
24 hour telephone notification to Louisiana DEQ	**(225) 765-0160**
Radiation Protection Division	
7 day written follow-up to Louisiana DEQ	

Discussion (Item 6): The reportable quantity under 10CFR20.2201(a)(i) is equal to or greater than 1000 times the quantity specified in appendix C to paragraphs 20.1001-20.2401. 10CFR20.2201(d) requires, subsequent to submitting the initial LER that any additional substantive information on the loss or theft be submitted within 30 days after learning of such information. If the licensed material lost or stolen is special nuclear material or spent fuel, <u>then</u> a one hour notification via ENS is required regardless of the quantity. See Security-Related Requirement Item 7.

Per 10CFR20.2201(c), occurrences reported in accordance with 10CFR50.73 (see regulation for additional potential reporting criteria) do not need a duplicate report under by 10CFR20.2201(b).

7. Material Shipment/Packaging Discrepancies. **Removable radioactive contamination in excess of 22000 disintegrations per minute ($1x10^{-3}\mu Ci/100cm^2$) found on the external surfaces of a package at receipt.**

One hour notification to NRC Region IV	**10CFR20.1906(d)(1)**
Immediately notify the final delivering carrier of the shipment.	

Discussion (Item 7): Contaminants reportable under this limit include all β^-/γ emitting radionuclides, all radionuclides with half-lives less than ten days; natural uranium; natural thorium; Uranium-235 and –238; Thorium-228 and –230 when contained in physical concentrates.

8. Material Shipment/Packaging Discrepancies. **Removable radioactive contamination from an alpha-emitting radionuclide in excess of 2200 disintegrations per minute ($1x10^{-4}$ µCi/100cm^2) found on the external surfaces of a package at receipt.**

One hour notification to NRC Region IV	**10CFR20.1906(d)(1)**
Immediately notify the final delivering carrier of the shipment.	

9. Material Shipment/Packaging Discrepancies. **Radiation levels in excess of 200 millirem per hour found on the external surfaces of a package at receipt.**

One hour notification to NRC Region IV	**10CFR20.1906(d)(2)**
Immediately notify the final delivering carrier of the shipment.	

Discussion (Item 9): Certain exceptions to the 200 mrem limit apply. See 10CFR71.47 for additional details.

10. Personnel Exposure/Contamination. **Contaminated individual transported to an off-site medical facility for treatment.**

8 hour notification via ENS	**10CFR50.72(b)(3)(xii)**

11. Radioactive Releases. **Airborne radioactive release that, averaged over 1 hour, resulted in concentrations in an unrestricted area that exceed 20 times the applicable concentration limit specified in appendix B to part 20, table 2, column 1.**

Refer to Attachment 7.2 Item 1, Emergency Declaration and Attachment 7.5 Item 2, News Release. Previous requirement has been deleted.	
24 hour telephone notification to Louisiana DEQ Radiation Protection Division	**(225) 765-0160**
7 day written follow-up to Louisiana DEQ	
60 day LER	**10CFR50.73(a)(2)(viii)**

Discussion (Item 11): Reports submitted under this criterion satisfy the reporting requirements of 10CFR20.2202 and 20.2203(a)(3).

12. Radioactive Releases. **Liquid effluent release that, averaged over 1 hour, exceeds 20 times the applicable concentration specified in appendix B to part 20, table 2, column 2, at the point of entry into the receiving water (i.e., unrestricted area) for all radionuclides except tritium and dissolved noble gasses.**

Refer to Attachment 7.2 Item 1, Emergency Declaration and Attachment 7.5 Item 2, News Release. Previous requirement has been deleted.

24 hour telephone notification to Louisiana DEQ Radiation Protection Division	**(225) 765-0160**
7 day written follow-up to Louisiana DEQ	
60 day LER	**10CFR50.73(a)(2)(viii)**

Discussion (Item 12): Reports submitted under this criterion satisfy the reporting requirements of 10CFR20.2202 and 20.2203(a)(3).

13. Radioactive Releases. **Events involving the loss of control of licensed material that may have caused, or threatens to cause, the release of radioactive material inside or outside of a restricted area, so that, had an individual been present for 24 hours, the individual could have received an intake in excess of one occupational annual limit on intake (this provision does not apply to locations where personnel are not normally stationed during routine operations such as hot-cells or process enclosures).**

24 hour notification via ENS	**10CFR20.2202(b)(2)**
24 hour telephone notification to Louisiana DEQ Radiation Protection Division	**(225) 765-0160**
30 day LER	10CFR20.2203(A)(1)

Discussion (Item 13): This provision does not include planned special exposures that are reported under 10CFR20.2204.

If the release is associated with a radioactive spill outside the restricted area, then environmental reporting requirements also may apply. See Attachment 7.6.

Per 10CFR20.2203(c), occurrences reported in accordance with 10CFR50.73 do not need to be reported by a separate report under 10CFR20.2203(a)(1).

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14. Radioactive Releases. **Levels of radiation or releases of radioactive material in excess of the limits specified in 40CFR190 or in excess of license conditions related to compliance with 40CFR190.**

24 hour telephone notification to Louisiana DEQ Radiation Protection Division	**(225) 765-0160**
7 day written follow-up to Louisiana DEQ	
30 day LER	**10CFR20.2203(a)(4) and TRM 3.11.4**

15. Loss/Theft of Radioactive Material. **Lost, stolen, or missing licensed material that is not recovered within 30 days after it was discovered to be lost, stolen or missing.**

30 day notification via ENS	**10CFR20.2201(a)(1)(ii)**
30 day follow-up LER	**10CFR20.2201(b)(1)**

Discussion (Item 15): Required for quantities greater than 10 times the quantity specified in Appendix C to 10CFR parts 20.1001-20.2401. 10CFR20.2201(d) requires, subsequent to submitting the initial LER that any additional substantive information on the loss or theft be submitted within 30 days after learning of such information. If the licensed material lost or stolen is special nuclear material or spent fuel, <u>then</u> a one hour notification via ENS is required regardless of the quantity. See Security-Related Requirement Item 7.

Per 10CFR20.2201(c), occurrences reported in accordance with 10CFR50.73 (see regulation for additional potential reporting criteria) do not need a duplicate report under by 10CFR20.2201(b).

16. Radioactive Releases. **Levels of radiation or concentrations of radioactive material in**

- **restricted area (protected area) in excess of any applicable limit in the license.**
- **an unrestricted area (outside protected area) in excess of 10 times any applicable limit set forth in 10CFR20 or in the license (whether or not involving exposure of any individual).**

30 day LER	**10CFR20.2203(a)(3)**

17.	Material Shipment/Packaging Discrepancies. **Significant reduction in the effectiveness of any radioactive material packaging during use.**	
	30 day report to the Director, Office of Nuclear Material Safety and Safeguards	**10CFR71.95**
18.	Personnel Exposure/Contamination. **Conduct of a planned special exposure in accordance with 10CFR20.1206.**	
	Written report to NRC Region IV 30 days following the planned special exposure	**10CFR20.2204**
19.	Radioactive Releases. **Special reports are required by the ODCM if:** - **Quarterly and/or Annual Effluent Dose Limits of ODCM / TRM are exceeded.** **Liquid Releases – TRM 3.11.1.2** **Gaseous Releases – TRM 3.11.2.2 and 3.11.2.3** - **31-day Projected Effluent Dose Limits of the ODCM / TRM are exceeded without the use of either Ventilation Exhaust Treatment, Waste Gas Holdup System, or Liquid Radwaste Treatment.** **Liquid Releases – TRM 3.11.1.3** **Gaseous Releases – TRM 3.11.2.4**	
	30 day Special Report	

7.4	Security-Related Notification Requirements

1. Threat to cause damage. **Any event in which there is reason to believe that a person has committed or caused, or attempted to commit or cause, or has made a creditable threat to commit or cause significant physical damage to the nuclear island or to the equipment, or carrier equipment transporting nuclear fuel or spent nuclear fuel; or to the nuclear fuel or spent fuel the facility or carrier possesses.**

Notify the Security Shift Supervisor

One hour notification via ENS	**10CFR73.71(b)(1)**
60 Day LER	**10CFR73.71(d)**

Ref: **10CFR73 Appendix G I.(a)(2)**

EN-NS-200, Attachment 9.3

2. Threat to operations. **Any event in which there is reason to believe that a person has committed or caused, or attempted to commit or cause, or has made a creditable threat to commit or cause interruption of normal operation through the unauthorized use of or tampering with its machinery, components, or controls including the security system.**

Notify the Security Shift Supervisor

One hour notification via ENS	**10CFR73.71(b)(1)**
60 Day LER	**10CFR73.71(d)**

Ref: **10CFR73 Appendix G I.(a)(2)**

EN-NS-200, Attachment 9.3, and

EN-NS-200, 5.3 and Attachment 9.4

3. Unauthorized access. **Actual entry of an unauthorized person into a protected area, material access area, controlled access area, vital area, or transport.**

Notify the Security Shift Supervisor

One hour notification via ENS	**10CFR73.71(b)(1)**
60 Day LER	**10CFR73.71(d)**

Ref: 10CFR73 Appendix G I.(b)

EN-NS-200, Attachment 9.3

4. Safeguard system vulnerability. **Any failure, degradation, or the discovered vulnerability in a safeguard system that could allow unauthorized or undetected access to a protected area, material access area, controlled access area, vital area, or transport for which compensatory measures have not been employed.**

Notify the Security Shift Supervisor

One hour notification via ENS	**10CFR73.71(b)(1)**
60 Day LER	**10CFR73.71(d)**

Ref: 10CFR73 Appendix G I.(c)

EN-NS-200, Attachment 9.3.

5. Introduction of contraband **– actual or attempted – into a protected area, material access, vital area, or transport.**

Notify the Security Shift Supervisor

One hour notification via ENS	**10CFR73.71(b)(1)**

60 Day LER	**10CFR73.71(d)**

Ref: 10CFR73 Appendix G I.(d)

EN-NS-200, Attachment 9.3.

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6. Recovery of, or accounting for a lost shipment of special nuclear material or spent fuel.

Notify the Security Shift Supervisor	
One hour notification via ENS	**10CFR73.71(a)(1)**
60 Day LER	**10CFR73.71(a)(4)**

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7. Loss/Theft of SNM or Spent Fuel. **Any loss (other than normal operating loss) or actual or attempted theft or unlawful diversion of special nuclear material or spent fuel.**

Notify the Security Shift Supervisor	
One hour notification via ENS	**10CFR70.52(a)**
	10CFR70.52(b)
	10CFR73.67(g)(3)(iii)
	10CFR73.71(a)(1)
24 hour telephone notification to Louisiana DEQ Radiation Protection Division	**(225) 765-0160**
7 day written follow-up to Louisiana DEQ	
60 Day LER	**10CFR73.71(a)(4)**
	10CFR73.71(d)

EN-NS-200, Attachment 9.3

10CFR73 App. G I.(a)(I)

8. **Significant FFD policy violations or programmatic failures. The following significant FFD policy violations and programmatic failures must be reported to the NRC Operations Center by telephone within 24 hours after the licensee or other entity discovers the violation: 10CFR Part 26.719(b).**

   1) The use, sale, distribution, possession, or presence of illegal drugs, or the consumption or presence of alcohol within a protected area;

   2) Any acts by any person that is a licensed operator, SSNM transporters, FFD program personnel, or any supervisory personnel who are authorized under this part, if such acts:
      - **i. Involve the use, sale, or possession of a controlled substance;**
      - **ii. Result in a determination that the individual has violated the licensee's or other entity's FFD** *[including Fatigue/Work-Hour]* **policy (including subversion as defined in § 26.5); or**
      - **iii. Involve the consumption of alcohol within a protected area or while performing the duties that require the individual to be subject to the FFD program;**

   3) Any intentional act that casts doubt on the integrity of the FFD program; and

   4) Any programmatic failure, degradation, or discovered vulnerability of the FFD program that may permit undetected drug or alcohol use or abuse by individuals within a protected area, or by individuals who are assigned to perform duties that require them to be subject to the FFD program.

   Notify the Security Shift Supervisor

   24 hour notification via ENS **10CFR26.719(b)**

   EN-NS-200, Attachment 9.3, Event 9

   Discussion (Item 8): If an event with safeguards significance is caused by a Fitness For Duty (FFD) event, the FFD aspects must be submitted to the NRC in accordance with 10CFR26 and the safeguards aspects reported in accordance with 10CFR73.71. When a telephone report is required by both rules, the licensee need only make one call to the NRC Operations Center within one hour.

9. Physical and/or Cyber threat to major electrial systems

a **Actual or suspected physical sabotage, terrorism or vandalism to major electric system shall be reported to the Department of Energy (DOE)**

Notify the Security Shift Supervisor

One hour notification via form OE-417 (Attachment 9.4 of EN-NS-200)

48 hour update via form OE-417 (Attachment 9.4 of EN-NS-200)

b **Actual or suspected cyber sabotage, terrorism or vandalism to major electrial system shall be reported to Department of Energy (DOE)**

Notify the Security Shift Supervisor

One hour notification via form OE-417 (Attachment 9.4 of EN-NS-200)

48 hour update via form OE-417 (Attachment 9.4 of EN-NS-200)

7.5	Miscellaneous Notification Requirements.	
1.	Any incident of an accidental criticality involving special nuclear material.	
	One hour notification via ENS	**10CFR70.52(a)**
2.	News Release **planned or notification to other government agencies regarding an event or situation related to the health and safety of the public or on-site personnel, or protection of the environment.**	
	4 hour notification via ENS	**10CFR50.72(b)(2)(xi)**
3.	Discrepancy in operation or integrity of intake and/or discharge dolphins.	

Notify the 8th Coast Guard District promptly (normally within 24 hours) upon the discovery of a discrepancy in the operation of the dolphin lights, or if any dolphin (with or without lights) is damaged such that an obstruction or hazard to river navigation exists.

Report correction of the discrepancy by the same method.

Mon-Fri (0630-1500):	**8th District Private Aids to Navigation Section (504) 589-6236**
All other times:	**8th District Operations (504) 589-6225**

Discussion (Item 3): A 'Discrepancy in operation' exists if any of the green lights marking the intake and discharge structures (three on the intake structure and one on the discharge structure) is not in the proper physical location, not illuminated during periods of darkness, or not displaying the proper light characteristic (0.5 seconds on, 2 seconds off).

A 10CFR50.72 report to the NRC is not required for dolphin lighting discrepancies.

Ref: Private Aids To Navigation Application. 33CFR66.01-20, 25, 50 Sunken Vessels and Other Obstructions/Marking and Notification Requirements. 33CFR64.10-1.

4. Failure by Entergy Operations, Inc., to comply with the antitrust conditions in Appendix C of the operating license.

   24 hour notification via ENS **NPF-38**

   30 day LER

5. Failure of the fire protection system to meet all of the provisions described in Amendment 36 of the FSAR and SER through Supplement 9.

   24 hour notification via ENS **NPF-38**

   30 day LER

6. Information previously submitted to the Commission is not complete or accurate.

   Two Day Notification to NRC Region IV **10CFR50.9(b)**

7. Failure, defect, malfunction, or noncompliance of a basic component as defined in 10CFR21.3(a).

   Notification and reporting of defects as required by 10CFR21 should be made in accordance with Site Procedure W2.301, "Identification, Evaluation and Reporting Process for 10CFR21 Requirements."

8. Acknowledgment of receipt of a radioactive material shipment not received within 20 days after transfer.

   Investigate and file a written report with the nearest Regional Office within 2 weeks of completion of the investigation. Ref: 10CFR20, Appendix F to sections 20.1001-20.2401, item III(E)(2).

9. In-service Inspection Reports:

a. Steam generator tube plugging

15 day SR **TS 4.4.4.5a**

b. Summary of complete ISI for Steam generator

SR within 12 months **TS 4.4.4.5b**

c. C-3 Steam generator tube inspection

SR within 30 days and prior to resumption of plant operation **TS 4.4.4.5c**

d. Degradation of containment structure

15 day SR **TS 4.6.1.6**

e. Degradation of shield building structure

14 day SR **TS 4.6.6.3**

10. Significant change or error in the emergency core cooling systems (ECCS) evaluation model or in the application of such a model resulting in:

a. peak fuel cladding temperature different by more than 50 °F from the temperature calculated for the limiting transient using the last acceptable model.

b. a culmination of changes and errors such that the sum of the absolute magnitudes of the respective temperature changes is greater than 50 °F.

c. peak cladding temperature in excess of 2200 °F.

d. maximum cladding oxidation in excess of 0.17 times the total cladding thickness before oxidation.

e. total hydrogen generation exceeding 0.01 times the maximum hypothetical amount.

f. core geometry not remaining amenable to cooling.

g. inability to maintain core temperature by decay heat removal for the required extended period.

8 hour notification via ENS **10CFR50.72(b)(3)(ii)**

30 day written report	**10CFR50.46(a)(3)(ii)**
	10CFR50.73(a)(2)(ii)

Discussion (Item 10): Notify the NRC Operations Center within eight hours of discovering any change or error in an acceptable evaluation model or in the application of such a model that results in one or more of the conditions listed in c through g, above. Conditions a and b do not, by themselves, meet the eight hour reporting threshold. For example, if the application of additional HPSI flow uncertainty to an acceptable evaluation model (e.g., a model reviewed and approved by the NRC for use at Waterford 3) results in a calculated peak cladding temperature in excess of 2200 °F, then the condition is reportable within eight hours an unanalyzed condition that significantly degraded plant safety.

Provide NRC with a written report within 30 days of discovering any change or error in an acceptable evaluation model or in the application of such a model that results in one or more of the conditions (a through g) listed above. The report should discuss the nature of the change or error, its estimated effect on the limiting ECCS analysis, and a proposed schedule for providing a reanalysis or taking other action as may be needed to comply with 10CFR50.46. For those conditions reported in accordance with 10CFR50.72, the follow-up written report shall be a LER.

7.6 Environmental Notification Requirements.

I. GENERAL DISCUSSION

- An Environmental Specialist should be made aware of all chemical or oil discharges or LPDES noncompliances to aid in evaluation and reporting.
- All discharges of chemicals and/or oils are potentially reportable to state and/or federal agencies.
- All noncompliances with the LPDES permit are reportable to the Louisiana Department of Envirnomental Quality.
- Additional details regarding reporting requirements for oil or PCB spills or releases are covered in UNT-007-064, “Hazardous Materials Emergency Response Plan and SPCC.”
- The release of any hazardous substance into the environment that is not addressed in the Federal regulations or state permits may be reportable in certain quantities. Contact Chemistry for interpretation or clarification.
- Discharges of radionuclides may be reportable in certain quantities. Contact Chemistry for clarification concerning these discharges. Telephone numbers for chemistry personnel can be obtained on Waterford 3’s chemistry department homepage.
- The following information will be included, as appropriate, in verbal reports of environmental conditions:

  a. name and call-back phone number

  b. name and location of facility where discharge occurred

  c. date and time incident began and ended and expected duration if continuing

  d. extent of any injuries and identification of personnel hazards which response agencies may face

  e. common or scientific chemical name, U.S. Department of Transportation Hazard Classification and best estimate of amounts of discharged pollutants brief description of incident sufficient to allow response agencies to formulate level and extent of response activity

f. for radioactive material or other unauthorized emission of toxic air pollutants, include the following supplemental information:

   i. location of the source facility or stack.

   ii. time at the onset of the emission.

   iii. prevailing local wind direction and estimated velocity at time of onset.

   iv. duration of emission if stopped at time of notification.

- Radioactive Spills

  a. Significant radioactive spills inside or outside of a restricted area that threatens or could cause radioactive effluent limits to be exceeded should be reported as stated in item 4 or 13 of Attachment 7.3. As a guideline, spills that require corrective action to prevent exceeding any effluent limits should be considered reportable. Radioactive spills are reportable under environmental reporting requirements if they involve noncompliance's with the LPDES permit or the release of hazardous substances into the environment that are not addressed in federal or state permits. Contact the Environmental Engineer for assistance.

  b. Minor radioactive spills are those for which effluent limits are not exceeded or challenged. These spills will be reported in the Annual Effluent Release Report and no notifications are necessary.

- Phone Numbers

  Louisiana Department of Public Safety (State Police) – (225-925-6595)

  Louisiana Department of Environmental Quality - (225-342-1234)

  National Response Center - (800-424-8802)

  Environmental Protection Agency, Region VI - (214-665-2222)

  Lower Mississippi River Waterworks Warning Network (504-599-0100)

  St. Charles Parish Emergency Operations Center (985-783-5050)

  St. Charles Sheriff's Office (985-783-6807)

REPORTING REQUIREMENTS

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1. Environmental Emergency Condition. **An environmental "Emergency Condition" exists at the plant site (including the Energy Education Center) due to an unauthorized discharge.**

"Emergency Condition" **means any condition which could reasonably be expected to endanger the health and safety of the public, cause significant adverse impact to the land, water or air environment, or cause severe damage to property.**

One hour notification to LA DPS (225) 925-6595	**LAC33:I.3915**
7 day written notification to LA DEQ	**LAC33:I.3925(A)**

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2. Visible emission from the site (other than steam) may require a report to LADEQ within 7 days. Contact Chemistry for clarification concerning these emissions**. Telephone numbers for chemistry personnel can be obtained on Waterford 3's chemistry department homepage.**

7 day written notification to LA DEQ **Louisiana Air Emission Permit, General Condition XI**

3. Discharge of Oil to the environment.

DETERMINATION OF REPORTABILITY (Non-PCB Oil Spills)

Spill exceeded 42 gallons and caused Emergency Condition	**Yes è** **No** **ê** Proceed to next block below	**Notify within one-hour:** 1. Department of Public Safety (State Police) 24-Hour Louisiana Emergency Hazardous Materials Hotline (State Police notifies LADEQ) 2. National Response Center if spill causes sheen on navigable waterway 3. Local Drainage Authority (St. Charles Parish Emergency Operations Center) if spill enters subsurface drains **Submit written report to LADEQ within seven calendar days after notification (see UNT-007-064).**
Spill exceeded 42 gallons and did not cause Emergency Condition	**Yes è** **No** **ê** Proceed to next block below	**Notify within 24-hours:** 1. Louisiana Department of Environmental Quality 2. National Response Center if spill causes sheen on navigable waterway 3. Local Drainage Authority (St. Charles Parish Emergency Operations Center) if spill enters subsurface drains **Submit written report to LADEQ within seven calendar days after notification (see UNT-007-064).**
Spill less than 42 gallons	**Yes è**	**Notify within 24-hours:** 1. National Response Center if spill causes sheen on navigable waterway 2. Local Drainage Authority (St. Charles Parish Emergency Operations Center) if spill enters subsurface drains **No notification necessary if spill did not cause a sheen on navigable waterway or enter subsurface drains.**

7 day written follow-up report to LA DEQ

4 hour notification via ENS **10CFR50.72(b)(2)(xi)**

DETERMINATION OF REPORTABILITY (PCB Oil Spills)

Spill outside facility boundary, contains ≥1.0 pounds of PCBs and caused Emergency Condition	**Yes è** **No ê** Proceed to next block below	**Notify within one-hour:** 1. Department of Public Safety (State Police) 24-Hour Louisiana Emergency Hazardous Materials Hotline (State Police notifies LADEQ) 2. St. Charles Parish EOC 3. National Response Center 4. Local Drainage Authority (St. Charles Parish Emergency Operations Center) if spills enters subsurface drains 5. EPA if spill directly contaminates surface water, sewers, drinking water, grazing lands or vegetable gardens, or if spill contains >10 pounds of PCB. 6. Lower Mississippi River Waterworks Warning Network if spill could impact downstream potable or industrial usage water. **Submit written report to LADEQ within seven calendar days after notification (see UNT-007-064).**
Oil spilled within facility boundary, contains ≥1.0 pounds of PCBs and did not cause Emergency Condition	**Yes è** **No ê** Proceed to next block below	**Notify within 24-hours:** 1. Louisiana Department of Environmental Quality 2. National Response Center 3. EPA if spill directly contaminates surface water, sewers, drinking water, grazing lands or vegetable gardens, or if spill contains greater than 10 pounds of PCB. **Submit written report to LADEQ within seven calendar days after notification (see UNT-007-064).**
Oil spill contains <1.0 pounds of PCBs	**Yes è**	**Notify within 24-hours:** 1. National Response Center if spill causes sheen on navigable waterway 2. Local Drainage Authority (St. Charles Parish Emergency Operations Center) if spill enters subsurface drains 3. EPA if spill directly contaminates surface water, sewers, drinking water, grazing lands or vegetable gardens. **No notification is necessary if spill did not cause a sheen on navigable waterway, enter subsurface drains, or contaminate water, land or vegetable gardens.**

7 day written follow-up report to LA DEQ

4 hour notification via ENS **10CFR50.72(b)(2)(xi)**

4. Any occurrence of an unusual or important event that indicates or could result in Significant environmental impact causally related to plant operation. **Examples include excessive bird impaction events, on-site plant or animal disease outbreaks, mortality or unusual occurrence of any species protected by the Endangered Species Act of 1973, fish kills, increase in nuisance organisms or conditions, and unanticipated or emergency discharge of waste water or chemical substances.**

   24 hour notification via ENS **NPF-38, Appendix B**

   30 day LER

5. Any unauthorized discharge to the Mississippi River **occurs which could reasonably be expected to interfere with or significantly impact downstream potable or industrial water usage then, in addition to other reporting requirements, notification must also be made to:**

   Lower Mississippi River Waterworks Warning Network

   New Orleans Office: (504) 599-0100

6. Any event related to the health and safety of the public or on-site personnel, or protection of the environment, for which notification to other government agencies has been or will be made.

   4 hour notification via ENS **10CFR50.72(b)(2)(xi)**

7. Non-Emergency Discharge of Pollutants. **Discharge of any of the following pollutants in a quantity exceeding that listed:**

	Pollutant	Pounds
i.	**Ammonium Hydroxide**	**1,000**
ii.	**Calcium Hypochlorite**	**10**
iii.	**Hydrazine**	**1**
iv.	**Sodium Hydroxide (Caustic)**	**1,000**
v.	**Trisodium Phosphate**	**5000**
vi.	**Sulfuric Acid**	**1000**
vii.	**PCBs**	**1**
viii.	**Unlisted Hazardous Wastes***	
	(1) Characteristic of Ignitability	**100**
	(2) Characteristic of Corrosivity	**100**
	(3) Characteristic of Reactivity	**100**
	(4) Characteristic of EP Toxicity	**1**
ix.	**Any unlisted chemical or hazardous waste**	*****

Contact Environmental/Safety Coordinator or specialist for characterization or exceptions.

If discharged into water: Immediately notify the US Coast Guard's National Response Center Duty Officer [(800) 424-8802]

24 hour notification to appropriate division of LA DEQ [(225) 342-1234] [LAC33:I.3917]

7 day written follow-up report to LA DEQ [LAC33:I.3925]

8. Discharges to the Atmosphere. **LPDES noncompliance which may endanger health or the environment.**

   a. Any unanticipated emission control device bypass.

   b. Any "upset" event which exceeds any effluent limitation in the permit; as per the definition of "upset" in LPDES Permit No. LA0007374.

24 hour notification to US EPA	**(214) 665-2222**
24 hour notification to LA DEQ	**LAC33:III.5107**
Air Qual. Div	**(225) 765-0219**
5 day written report to US EPA	
7 day written report to LA DEQ	**LAC33:III.5107**

9. Outfall Limitations**. If any of the following conditions exist, make notifications as follows:**

24 hour telephone notification to LA DEQ	**LAC33:I.3917**
	LPDES Permit# LA0007374, Part II.E.
	(225) 342-1234
5 day written notification to LA DEQ	**LAC33:I.3925**

   a. Daily maximum discharge limitations from any outfall is exceeded for the following:

   Total Chromium – 0.2 mg/l

   Total Copper – 1.0 mg/l

   Total Zinc – 1.0 mg/l

Written notification via next Discharge Monitoring Report (DMR) submittal to LA DEQ	**LPDES Permit# LA0007374, Part III.D.7**

   b. All instances of noncompliance not reported in item a, above.

10. Discharge resulting in groundwater contamination.

7 day written report to LA DEQ	**LAC33:I.3919**

7.7 Hazardous Material Notification Requirements

I. GENERAL DISCUSSION

- Detailed hazardous substance reporting criteria are included in UNT-007-064, “Hazardous Materials Emergency Response Plan and SPCC.”
- All incidents involving chemicals, substances, elements, compounds in the form of liquids, gases, or solids; radionuclides; or a combination thereof are potentially reportable under this section.
- An Environmental Specialist should be notified of all releases of hazardous materials requiring notification of off-site organizations. Telephone numbers for chemistry personnel can be obtained on Waterford 3’s chemistry department web homepage.

- Phone Numbers

  Louisiana Department of Public Safety (State Police) – (225-925-6595)

  Louisiana Department of Environmental Quality - (225-342-1234)

  National Response Center - (800-424-8802)

  Environmental Protection Agency, Region VI - (214-665-2222)

  Lower Mississippi River Waterworks Warning Network (504-599-0100)

  St. Charles Parish Emergency Operations Center (985-783-5050)

  St. Charles Sheriff’s Office (985-783-6807)

II. REPORTING REQUIREMENTS

<u>DETERMINATION OF REPORTABILITY</u> (Chemical or Hazardous Waste Spills) *

Spill is outside the facility boundary and exceeded the 40CFR302 Reportable Quantity	**Yes è** **No** **ê** Proceed to next block below	**Notify within one-hour:** 1. Department of Public Safety (State Police) 24-Hour Louisiana Emergency Hazardous Materials Hotline (State Police notifies LADEQ) 2. Local Law Enforcement 3. Louisiana Department of Environmental Quality 4. National Response Center 5. Local Drainage Authority (St. Charles Parish Emergency Operations Center) if spills enters subsurface drains 6. EPA Regional Office **Submit written report to LADEQ within seven calendar days after notification (see UNT-007-064).** **If applicable, submit written report to EPA within fifteen days for incidents involving hazardous waste fires or explosions (see UNT-007-064).**
Spill is within the facility boundary and exceeded the 40CFR302 Reportable Quantity	**Yes è** **No** **ê** Proceed to next block below	**Notify within 24-hours:** 1. Louisiana Department of Environmental Quality 2. National Response Center 3. EPA Regional Office **Submit written report to LADEQ within seven calendar days after notification (see UNT-007-064).** **If applicable, submit written report to EPA within fifteen days for incidents involving hazardous waste fires or explosions (see UNT-007-064).**
Spill less than the 40CFR302 Reportable Quantity	**Yes è**	**No notification necessary.**

* If the chemical or hazardous waste spill is within the facility boundary, is <u>not</u> volatile to the atmosphere, and is entirely contained within a concrete containment with <u>no</u> chance of exposure to the public, <u>no</u> reporting is necessary.

1. Transportation Accident. **Incident, accident, or the cleanup of an incident or accident during the transportation, loading, unloading, or related storage in any place of a hazardous material involving:**

- **A fatality due to fire, explosion, or exposure to any hazardous material.**
- **The hospitalization of any person due to fire, explosion, or exposure to any hazardous material.**
- **A continuing danger to life, health, or property at the place of the incident or accident.**
- **An estimated property damage of more than ten thousand dollars.**

One hour notification to LA DPS **[(225) 925-6595]**

Written follow-up report to LA DEQ

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2. Any event related to the health and safety of the public or on-site personnel, or protection of the environment, for which notification to other government agencies has been or will be made.

4 hour notification via ENS **10CFR50.72(b)(2)(xi)**

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3. Any occurrence of an unusual or important event that indicates or could result in significant environmental impact causally related to plant operation. **Examples include excessive bird impaction events, on-site plant or animal disease outbreaks, mortality or unusual occurrence of any species protected by the Endangered Species Act of 1973, fish kills, increase in nuisance organisms or conditions, and unanticipated or emergency discharge of waste water or chemical substances.**

24 hour notification via ENS **NPF-38, Appendix B**

30 day LER

[LAST PAGE]

PROTECTIVE ACTION GUIDELINES WORKSHEET, RELEASE OCCURRING

USE ATT. 7.1 FOR INITIAL GENERAL EMERGENCY CLASSIFICATION RECOMMENDATIONS

EVALUATION PERFORMED BY: _____ DATE: _____ TIME: _____

(COMPLETE SECTIONS A, B & C EACH TIME AN EVALUATION IS PERFORMED.)

A. 0-2 MILE RESPONSE AREA

PROJECTED DOSE AT THE EAB

TEDE = _____ mrem

CDE THYROID = _____ mrem

IS THE PROJECTED DOSE ≥ 1000 mrem TEDE? **OR** ≥ 5000 mrem CDE THYROID?

NO → NO IMMEDIATE PROTECTIVE ACTIONS REQUIRED

↓ YES

EVACUATE (or Shelter)

0-2 MILE RECOMMENDATION:

☐ NO PROTECTIVE ACTIONS REQUIRED

☐ EVACUATE AREAS A1, B1, C1, D1

☐ SHELTER AREAS _____

B. 2-5 MILE RESPONSE AREA

PROJECTED DOSE AT 2 MILES

TEDE = _____ mrem

CDE THYROID = _____ mrem

IS THE PROJECTED DOSE ≥ 1000 mrem TEDE? **OR** ≥ 5000 mrem CDE THYROID?

NO → NO IMMEDIATE PROTECTIVE ACTIONS REQUIRED

↓ YES

EVACUATE (or Shelter)

2-5 MILE RECOMMENDATION: (See NOTE 1)

☐ NO PROTECTIVE ACTIONS REQUIRED

☐ EVACUATE AREAS _____

☐ SHELTER AREAS _____

C. 5-10 MILE RESPONSE AREA

PROJECTED DOSE AT 5 MILES

TEDE = _____ mrem

CDE THYROID = _____ mrem

IS THE PROJECTED DOSE ≥ 1000 mrem TEDE? **OR** ≥ 5000 mrem CDE THYROID?

NO → NO IMMEDIATE PROTECTIVE ACTIONS REQUIRED

↓ YES

EVACUATE (or Shelter)

5-10 MILE RECOMMENDATION: (See NOTE 1)

☐ NO PROTECTIVE ACTIONS REQUIRED

☐ EVACUATE AREAS _____

☐ SHELTER AREAS _____

NOTE 1: DETERMINE THE APPROPRIATE PROTECTIVE RESPONSE AREAS BY USING ATTACHMENT 7.3.

AFFECTED COMPASS SECTORS/PROTECTIVE RESPONSE AREAS CHART

Directions for use:

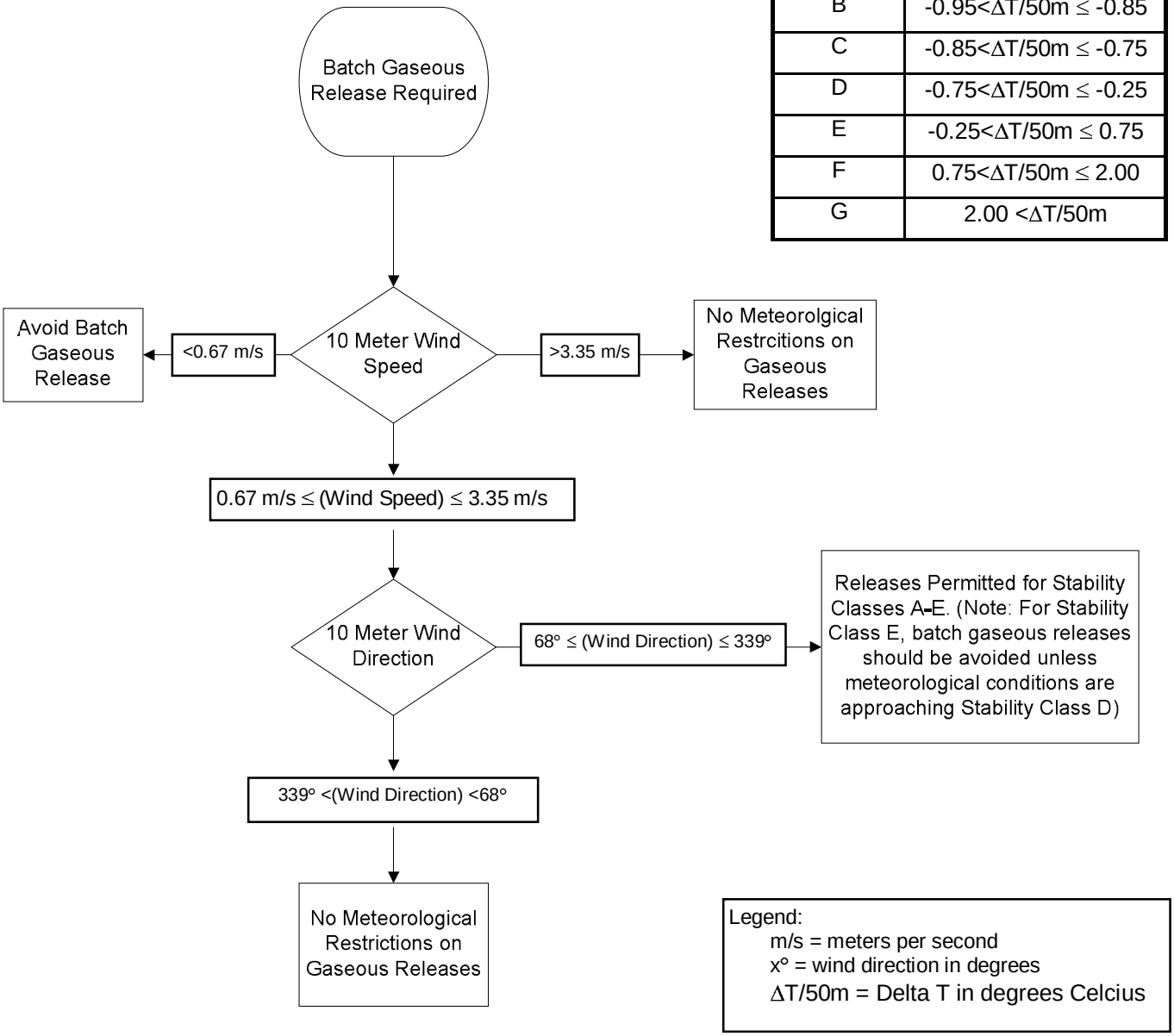
Locate appropriate wind direction and find plume center line sector and two adjacent sectors in the "Affected Compass Sectors" column. The centerline sector is the middle sector of each set. The corresponding "Protective Response Areas" in which protective actions are to be implemented can then be found for downwind distances of interest by reading across the page.

Note that protective actions for ***any*** compass sectors in the two-mile radius (0-2 miles column) should be recommended for all 4 protective response areas in the two-mile radius as shown below.

WIND DIRECTION FROM	AFFECTED COMPASS SECTORS	PROTECTIVE RESPONSE AREAS		
		0-2 MILES	2-5 MILES (DOWNWIND)	5-10 MILES (DOWNWIND)
191.3 TO 213.8 (Sector K)	A, B, C	A1, B1, C1, D1	A2, B2	A3, B4
213.8 TO 236.3 (Sector L)	B, C, D	A1, B1, C1, D1	A2, B2	A3, B4
236.3 TO 258.8 (Sector M)	C, D, E	A1, B1, C1, D1	B2, D2	B3, B4
258.8 TO 281.3 (Sector N)	D, E, F	A1, B1, C1, D1	B2, D2	B3, B4, D3
281.3 TO 303.8 (Sector P)	E, F, G	A1, B1, C1, D1	B2, D2	B3, B4, D3
303.8 TO 326.3 (Sector Q)	F, G, H	A1, B1, C1, D1	B2, D2	B3, D3, D4
326.3 TO 348.8 (Sector R)	G, H, J	A1, B1, C1, D1	D2	D3, D4
348.8 TO 11.3 (Sector A)	H, J, K	A1, B1, C1, D1	D2	D4
11.3 TO 33.8 (Sector B)	J, K, L	A1, B1, C1, D1	C2, D2	C4, D4
33.8 TO 56.3 (Sector C)	K, L, M	A1, B1, C1, D1	C2, D2	C4, D4
56.3 TO 78.8 (Sector D)	L, M, N	A1, B1, C1, D1	C2	C4
78.8 TO 101.3 (Sector E)	M, N, P	A1, B1, C1, D1	C2	A4, C3, C4
101.3 TO 123.8 (Sector F)	N, P, Q	A1, B1, C1, D1	C2	A4, C3, C4
123.8 TO 146.3 (Sector G)	P, Q, R	A1, B1, C1, D1	A2, C2	A3, A4, C3
146.3 TO 168.8 (Sector H)	Q, R, A	A1, B1, C1, D1	A2, C2	A3, A4, C3
168.8 TO 191.3 (Sector J)	R, A, B	A1, B1, C1, D1	A2, C2	A3, A4

11.5 METEOROLOGICAL CONDITIONS REQUIREMENTS
[P11585]

Pasquill Stability Classes	
Class	(ΔT °C/50m)
A	ΔT/50m ≤ -0.95
B	-0.95<ΔT/50m ≤ -0.85
C	-0.85<ΔT/50m ≤ -0.75
D	-0.75<ΔT/50m ≤ -0.25
E	-0.25<ΔT/50m ≤ 0.75
F	0.75<ΔT/50m ≤ 2.00
G	2.00 <ΔT/50m



Notes:
1. All parameters should be obtained from the 15 minute average values displayed on the PMC.
2. 10 meter wind speed and wind direction may be obtained from the primary or back-up tower 33' reading.
3. ΔT/50m may be obtained from the primary or back-up tower 199-33' Delta T reading.