Arkansas Nuclear One

Name:	**KEY**
Login:	N/A
Date:	6/15/2017

ANO-1
Subject: **2017 ILO NRC**

Total Points:	75/25/100
Percent Grade:	

	T	F		
1	⊏A⊐	⊏B⊐	■	⊏D⊐
2	⊏A⊐	⊏B⊐	⊏C⊐	■
3	■	⊏B⊐	⊏C⊐	⊏D⊐
4	⊏A⊐	⊏B⊐	⊏C⊐	■
5	⊏A⊐	■	⊏C⊐	⊏D⊐
6	⊏A⊐	⊏B⊐	⊏C⊐	■
7	⊏A⊐	■	⊏C⊐	⊏D⊐
8	⊏A⊐	⊏B⊐	⊏C⊐	■
9	⊏A⊐	⊏B⊐	⊏C⊐	■
10	⊏A⊐	■	⊏C⊐	⊏D⊐
11	⊏A⊐	⊏B⊐	■	⊏D⊐
12	⊏A⊐	⊏B⊐	⊏C⊐	■
13	⊏A⊐	■	⊏C⊐	⊏D⊐
14	⊏A⊐	⊏B⊐	■	⊏D⊐
15	⊏A⊐	⊏B⊐	⊏C⊐	■
16	■	⊏B⊐	⊏C⊐	⊏D⊐
17	■	⊏B⊐	⊏C⊐	⊏D⊐
18	⊏A⊐	⊏B⊐	■	⊏D⊐
19	⊏A⊐	■	⊏C⊐	⊏D⊐
20	⊏A⊐	■	⊏C⊐	⊏D⊐
21	⊏A⊐	■	⊏C⊐	⊏D⊐
22	■	⊏B⊐	⊏C⊐	⊏D⊐
23	⊏A⊐	■	⊏C⊐	⊏D⊐
24	⊏A⊐	⊏B⊐	⊏C⊐	■
25	⊏A⊐	■	⊏C⊐	⊏D⊐
26	⊏A⊐	⊏B⊐	⊏C⊐	■
27	⊏A⊐	⊏B⊐	⊏C⊐	■
28	■	⊏B⊐	⊏C⊐	⊏D⊐
29	■	⊏B⊐	⊏C⊐	⊏D⊐
30	■	⊏B⊐	⊏C⊐	⊏D⊐
31	⊏A⊐	⊏B⊐	■	⊏D⊐
32	⊏A⊐	⊏B⊐	⊏C⊐	■
33	⊏A⊐	⊏B⊐	■	⊏D⊐
34	⊏A⊐	⊏B⊐	■	⊏D⊐
35	■	⊏B⊐	⊏C⊐	⊏D⊐
36	⊏A⊐	■	⊏C⊐	⊏D⊐
37	⊏A⊐	⊏B⊐	■	⊏D⊐
38	■	⊏B⊐	⊏C⊐	⊏D⊐
39	⊏A⊐	⊏B⊐	■	⊏D⊐
40	■	⊏B⊐	⊏C⊐	⊏D⊐
41	⊏A⊐	■	⊏C⊐	⊏D⊐
42	⊏A⊐	⊏B⊐	■	⊏D⊐
43	⊏A⊐	⊏B⊐	■	⊏D⊐
44	⊏A⊐	⊏B⊐	⊏C⊐	■
45	⊏A⊐	■	⊏C⊐	⊏D⊐
46	⊏A⊐	■	⊏C⊐	⊏D⊐
47	⊏A⊐	■	⊏C⊐	⊏D⊐
48	⊏A⊐	⊏B⊐	■	⊏D⊐
49	⊏A⊐	⊏B⊐	■	⊏D⊐
50	⊏A⊐	■	⊏C⊐	⊏D⊐

	T	F		
51	■	⊏B⊐	⊏C⊐	⊏D⊐
52	■	⊏B⊐	⊏C⊐	⊏D⊐
53	⊏A⊐	⊏B⊐	■	⊏D⊐
54	⊏A⊐	■	⊏C⊐	⊏D⊐
55	⊏A⊐	⊏B⊐	⊏C⊐	■
56	⊏A⊐	■	⊏C⊐	⊏D⊐
57	⊏A⊐	⊏B⊐	■	⊏D⊐
58	⊏A⊐	⊏B⊐	■	⊏D⊐
59	⊏A⊐	■	⊏C⊐	⊏D⊐
60	■	⊏B⊐	⊏C⊐	⊏D⊐
61	⊏A⊐	⊏B⊐	■	⊏D⊐
62	⊏A⊐	■	⊏C⊐	⊏D⊐
63	■	⊏B⊐	⊏C⊐	⊏D⊐
64	⊏A⊐	■	⊏C⊐	⊏D⊐
65	■	⊏B⊐	⊏C⊐	⊏D⊐
66	⊏A⊐	⊏B⊐	■	⊏D⊐
67	⊏A⊐	⊏B⊐	■	⊏D⊐
68	⊏A⊐	■	⊏C⊐	⊏D⊐
69	⊏A⊐	⊏B⊐	■	⊏D⊐
70	⊏A⊐	⊏B⊐	⊏C⊐	■
71	⊏A⊐	■	⊏C⊐	⊏D⊐
72	⊏A⊐	⊏B⊐	■	⊏D⊐
73	⊏A⊐	⊏B⊐	⊏C⊐	■
74	■	⊏B⊐	⊏C⊐	⊏D⊐
75	⊏A⊐	■	⊏C⊐	⊏D⊐
76	■	⊏B⊐	⊏C⊐	⊏D⊐
77	⊏A⊐	⊏B⊐	■	⊏D⊐
78	■	⊏B⊐	⊏C⊐	⊏D⊐
79	■	⊏B⊐	⊏C⊐	⊏D⊐
80	⊏A⊐	■	⊏C⊐	⊏D⊐
81	⊏A⊐	⊏B⊐	⊏C⊐	■
82	⊏A⊐	⊏B⊐	⊏C⊐	■
83	■	⊏B⊐	⊏C⊐	⊏D⊐
84	⊏A⊐	⊏B⊐	■	⊏D⊐
85	⊏A⊐	■	⊏C⊐	⊏D⊐
86	■	⊏B⊐	⊏C⊐	⊏D⊐
87	■	⊏B⊐	⊏C⊐	⊏D⊐
88	⊏A⊐	⊏B⊐	⊏C⊐	■
89	⊏A⊐	⊏B⊐	⊏C⊐	■
90	⊏A⊐	■	⊏C⊐	⊏D⊐
91	⊏A⊐	⊏B⊐	■	⊏D⊐
92	⊏A⊐	■	⊏C⊐	⊏D⊐
93	■	⊏B⊐	⊏C⊐	⊏D⊐
94	■	⊏B⊐	⊏C⊐	⊏D⊐
95	⊏A⊐	⊏B⊐	⊏C⊐	■
96	⊏A⊐	⊏B⊐	■	⊏D⊐
97	⊏A⊐	■	⊏C⊐	⊏D⊐
98	⊏A⊐	■	⊏C⊐	⊏D⊐
99	⊏A⊐	⊏B⊐	■	⊏D⊐
100	■	⊏B⊐	⊏C⊐	⊏D⊐

QID: 1121 **Rev:** 2 **Rev Date:** 6/8/17 **Source:** Modified **Originator:** Cork

TUOI: A1LP-RO-NNI **Objective:** 4 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 008 **System Title:** Pressurizer Vapor Space Accident

Description: Ability to operate and / or monitor the following as they apply to the Pressurizer Vapor Space Accident:
Control of PZR level

K/A Number: AA1.06 **CFR Reference:** 41.7 / 45.5 / 45.6

Tier: 1 **RO Imp:** 3.6 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 3.6 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 1 **SRO:**

Given:
*Unit 1 100% power

A leak occurs on the upper tap of the controlling Pressurizer level transmitter sensing line, causing a PZR steam space leak.

Initially, actual PZR level will ___________ and RCS makeup flow will _________.

A. Decrease, Decrease

B. Decrease, Increase

C. Increase, Decrease

D. Increase, Increase

Answer:

C. Increase, Decrease

Notes:

"C" is correct since a steam space leak will cause actual level to rise. A leak on the upper tap will cause the differential pressure to decrease on the Pressurizer (PZR) level transmitter. The PZR level transmitter is reverse acting so a lowering DP will cause indicated level to rise. Since indicated level is rising RCS makeup flow will drop as indicated level rises above setpoint.

"A" is incorrect but plausible since the RCS makeup flow trend given is correct. However, actual PZR level will rise with a steam space leak, not drop.

"B" is incorrect, a steam space leak will cause actual level to rise. This answer (drop, rise) has trends which work together if applicant believe the loss of inventory (steam space leak) will cause indicated level to drop and makeup flow will rise to counteract.

"D" is incorrect but plausible since actual level trend is correct but makeup flow trend is wrong.

This is a modified version of QID 371. Added unit condition of 100% power and changed "actual PZR level" to "RCS makeup flow". This made "C" the correct answer (vs. D in 371).

This question matches the K/A since it involves a vapor space accident (upper PZR level transmitter tap) and evaluates applicant's knowledge on how the PZR level control system works in response to the steam leak.

References:

1304.022, Unit 1 Pressurizer Level & Temperature Channel Calibration
STM 1-69, Non-Nuclear Instrumentation System

History:

Modified QID 371 for 2017 re-exam.
Rev.1, 5/18/17
Replaced Drop, Rise with Decrease, Increase.
Added "the controlling" in the description of the leak, and "initially" at beginning of stem.
Rev. 2, 6/8/17
Corrected typo.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0368 **Rev:** 4 **Rev Date:** 5/18/17 **Source:** Bank **Originator:** Cork

TUOI: A1LP-RO-EOP02 **Objective:** 8 **Point Value:** 1

Section: 4.1 **Type:** Generic EPEs

System Number: 009 **System Title:** Small Break LOCA

Description: Knowledge of the interrelations between the small break LOCA and the following: S/Gs

K/A Number: EK2.03 **CFR Reference:** 41.7 / 45.7

Tier:	1	**RO Imp:**	3.0	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.3	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 2 **SRO:**

Given:
* Reactor tripped from 100% power due to low RCS pressure

The following conditions have existed for three (3) minutes:

* CETs 590 °F and rising
* RCS pressure 1700 psig and lowering
* RCPs are running

Which of the following operator actions are required to be performed?

A. Trip all running RCPs

B. Go to Overheating (1202.004)

C. Verify EFW flow 320 gpm to each SG

D. Verify Reflux Boiling setpoint selected on both EFIC trains

Answer:

D. Verify Reflux Boiling setpoint selected on both EFIC trains

Notes:

"D" is correct, Core Exit Thermocouple (CET) indication and RCS pressure condition show a loss of subcooling margin. With subcooling margin lost, the Reflux Boiling setpoint is required to be selected per RT-5, Verify Proper EFW Actuation and Control.

"A" is incorrect, plausible since this action would be taken on a loss of subcooling margin but only if less than 2 minutes had expired. Question condition says three minutes have gone by making this choice incorrect.

"B" is incorrect, plausible since this is an entry condition for the Overheating EOP but a loss of subcooling margin exists which has a higher priority than Overheating (1015.043, ANO-1 EOP/AOP User Guide, p. 13).

"C" is incorrect, plausible since (per RT-5) EFW flow is verified to both SGs but the flow value given is less than the required minimum flow rate of greater than or equal to 340 gpm when EFW flow is determined to be less than adequate.

Added "due to low RCS pressure" in first sentence, changed "RCS temperatures" to "CETs", and revised stem to be current with question development philosophy. Shuffled answer choices so they are short to long.

This matches the K/A since condtions are given for a SBLOCA and the applicant is required to know a related action for the SGs (Reflux Boiling Setpoint selected) due to the loss of SCM from the SBLOCA.

References:

1202.012, Repetitive Tasks, RT-5, Verify Proper EFW Actuation and Control

History:

Direct from regular exambank QID 3030.
Selected for use in 2002 SRO exam.
Modified for use in 2005 RO exam, replacement question. EK1.01
Selected for use in 2017 Re-take exam.
Rev. 3
1. Changed Distractor C based on NRC comment
Rev. 4, 5/18/17
Changed Distractor C back to original version since it was also a correct answer.
Editorial changes.

QID: 0491 **Rev:** 3 **Rev Date:** 5/18/17 **Source:** Bank **Originator:** Pullin

TUOI: A1LP-RO-EOP10 **Objective:** 6 **Point Value:** 1

Section: 4.1 **Type:** Generic EPEs

System Number: 011 **System Title:** Large Break LOCA

Description: Ability to determine or interpret the following as they apply to a Large Break LOCA: Conditions for throttling or stopping HPI.

K/A Number: EA2.11 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier:	1	**RO Imp:**	3.9	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.3	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 3 **SRO:**

Given:
* Large break LOCA occurred
* ESAS actuated
* A3 bus locked out
* RCS pressure 50 psig and stable
* BWST level 12 ft. and lowering
* LPI flow 3150 gpm
* HPI total flow 275 gpm

Which of the following actions are required per ESAS (1202.010) for these conditions?

A. Secure P-36B HPI pump

B. Raise HPI flow on P-36B HPI pump

C. Swap to RB sump recirculation using Attachment 1

D. Open Decay Heat supply to Makeup Pump suction CV-1277

Answer:

A. Secure P-36B HPI pump

Notes:

"A" is correct, the HPI pump can be secured since LPI flow is greater than 3050 gpm with only one LPI pump (A3 bus is locked out so "A" LPI pump cannot run) per step 14 of 1202.010.

"B" is incorrect, this action would be taken if LPI flow was inadequate (<3050 gpm), but LPI flow is adequate.

"C" is incorrect but plausible since for a large break LOCA BWST level would be dropping but transfer to RB recirculation is not performed until BWST level is less than 6 ft. and BWST level is 12 ft.

"D" is incorrect, plausible if LPI flow was inadequate (or RCS pressure was higer than LPI discharge pressure) and thus HPI pump operation was to continue via "piggyback" mode following transfer to RB sump suction, but LPI flow is adequate.

This matches the K/A since condtions are given for a large break LOCA and LPI flow is greater than the value required to allow for stopping HPI.

References:

1202.010, ESAS

History:

New Question on 2013 RO Exam EK2.02
Selected for 2017 RO Re-exam
Rev. 1: added condition of RCS pressure at 50 psig and BWST level of 12 ft. Replaced "C" distractor since

validators thought it was not plausible. Added component numbers.
Rev. 2
1. Re-ordered answers from shortest to longest and deleted "B" from the stem in front of the flows for LPI and HPI.
Rev. 3, 5/18/17
Deleted "and cannot be reenergized" from 3rd bullet.
Deleted "LPI/HPI flow rates are as follows:" and moved bullets up.
Changed LOD to 3.
Editorial changes.
Deleted "pump" from last bullet.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0396 **Rev:** 3 **Rev Date:** 5/18/17 **Source:** Bank **Originator:** S.Pullin

TUOI: A1LP-RO-ICS **Objective:** 4 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 015 **System Title:** Reactor Coolant Pump (RCP) Malfunctions

Description: Knowledge of the operational implications of the following concepts as they apply to Reactor Coolant Pump Malfunctions (Loss of RC Flow): Consequences of an RCPS failure

K/A Number: AK1.02 **CFR Reference:** 41.8 / 41.10 / 45.3

Tier:	1	**RO Imp:**	3.7	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.1	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 4 **SRO:**

Given:
* 70% power

Subsequently, following indications observed:

* "A" MFW flow 4.7 x E6 lbm/hr
* "B" MFW flow 2.3 x E6 lbm/hr

What event would cause this MFW flow discrepancy?

A. "B" MFW pump trip

B. "A" Tcold instrument failed high

C. "D" RCP trip

D. "A" RCP trip

Answer:

D. "A" RCP trip

Notes:

"D" is correct, "A" RCP trip (B loop) will cause ICS to re-ratio FW flow so that the highest MFW flow will be in the loop with the highest RCS flow (A loop, two RCPs running). This is to balance heat input with heat removal. With power unchanged (there would be no ICS runback on an RCP trip since power is less than 75%) 2/3 of the heat is now being removed from the A loop since it has 2/3 of the RCS flow and 1/3 of RCS flow will be going through the B loop so it will receive 1/3 of the total FW flow. Total FW flow will still be equivalent to 70% power.

"A" is incorrect, but plausible since B MFW pump trip would lower FW to the B SG (until the cross-tie valve opened) but this would also cause a runback to 60% power and FW flows would not be skewed as they are given.

"B" is incorrect, this is plausible as MFW would be affected by a Tcold failure but this would cause FW to decrease to the "A" SG to attempt to reduce "A" Tcold temperature, FW flow would also increase to the "B" SG making this choice incorrect.

"C" is incorrect, this is plausible as MFW would re-ratio but a "D" RCP trip (A loop) would cause FW to decrease to the "A" SG and increase to the "B" SG.

This question matches the K/A since re-ratioing of MFW is an operational implication, and consequence, of a RCP failure.

References:

STM 1-64, Integrated Control System

History:

New question created for 2001 RO/SRO Exam. AA1.05
Selected for 2013 Exam
Selected for 2017 RO Re-exam.
Rev. 1 Changed order of correct answer so now "D" is correct. Deleted "instrument" from B so they are all about the same length.
Rev. 2 Changed Distractor A based on NRC comment and reinserted "instrument" into B so that two answers are the same length.
Rev. 3, 5/18/17
Changed distractor A back to B MFW pump trip.
Editorial changes.

QID: 0610 **Rev:** 2 **Rev Date:** 5/18/17 **Source:** Bank **Originator:** Cork/Pullin

TUOI: A1LP-RO-MU **Objective:** 10 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 022 **System Title:** Loss of Reactor Coolant Makeup

Description: Knowledge of the reasons for the following responses as they apply to the Loss of Reactor Coolant Makeup:
Actions contained in SOPs and EOPs for RCPs, loss of makeup, loss of charging, and abnormal charging

K/A Number: AK3.02 **CFR Reference:** 41.5, 41.10 / 45.6 /45.13

Tier: 1 **RO Imp:** 3.5 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 3.8 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 5 **SRO:**

Given:
* 100% power
* Makeup pump P-36B tripped

What is the primary reason the AOP directs isolation of Letdown for this condition?

A. Prevent water hammer

B. Maintain RCS inventory

C. Prevent overfilling Makeup Tank

D. Reduce heat load on Nuclear ICW system

Answer:

B. Maintain RCS inventory

Notes:

"B" is correct, RCS inventory is maintained by isolating letdown.

"A" is incorrect, this is plausible since this is the same reason for closing SW valves in sequence on the RB coolers.

"C" is incorrect, this is plausible since Makeup Tank level would rise with the Makeup Pump tripped and letdown still going but the primary reason is to maintain RCS inventory.

"D" is incorrect, this is plausible since isolating letdown will reduce heat load on Nuclear ICW but the reason is to maintain RCS inventory.

This question matches the K/A since it is about a loss of RCS makeup (makeup pumt trip) and it tests knowledge of the reason for an action from the loss of charging AOP.

References:

1203.026, Loss of Reactor Coolant Makeup

History:

New for 2005 RO exam, replacement question. AK3.04
Selected for 2017 RO Re-exam.
Rev.2, 5/18/17
Editorial changes.
Re-ordered choices short to long.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1122 **Rev:** 3 **Rev Date:** 6/8/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-TS **Objective:** 5 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 025 **System Title:** Loss of RHR System

Description: Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

K/A Number: 2.2.42 **CFR Reference:** 41.7 / 41.10 / 43.2 / 43.3 / 45.3

Tier:	1	**RO Imp:**	3.9	**RO Select:**	Yes	**Difficulty:**	4
Group:	1	**SRO Imp:**	4.6	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 6 **SRO:**

Given at 0915:
* Plant Heatup to 200 °F in progress to prepare for starting RCPs
* Both Decay Heat pumps in operation
* "A" DH suction temperature 185 °F
* RCS pressure 190 psig

At 0945:
* Pressurizer level 95 inches
* "A" DH suction temperature 205 °F
* RCS pressure 215 psig

NOW
P-34A Decay Heat pump trips

Which of the following Technical Specifications LCO's is NOT met?

A. 3.4.3, Reactor Coolant Pressure/Temperature Limits

B. 3.4.6, RCS Loops - Mode 4

C. 3.4.9, Pressurizer

D. 3.5.3, ECCS - Shutdown

Answer:

D. 3.5.3, ECCS - Shutdown

Notes:

"D" is correct, with RCS temperature at 205 °F, Mode 4 has been entered (> 200 °F) and 3.4.3 requires two operable ECCS trains in Mode 4. With P-34A DH pump breaker tripped, operability is not satisfied.

"A" is incorrect but plausible, a temperature rise of 20 °F (205 - 185) has taken place over 30 minutes (0915 to 0945) but the 3.4.3 limit of 50 °F/hr has not yet been exceeded.

"B" is incorrect but plausible since 3.4.6 requires two operable loops with one loop in operation but RCS pressure is at the point (>200 psig where an RCP can be started and the other DH pump is running, so this spec is met for the given conditions.

"C" is incorrect but plausible since an 1102.002 procedural requirement of less than 95 inches in the Pressurizer is not met but 3.4.9 does not have a minimum Pressurizer level for operability. LCO 3.4.11, LTOP does have a surveillance requirement to verify the pressurizer does not represent a water solid condtion but that is not part of 3.4.9.

This question matches the K/A since it involves a loss of an RHR system (DH pump) with parameters requiring entry into TS action for 3.5.3.

References:

Technical Specificatlions 3.5.3

History:

New question for 2017 RO Re-take exam
Rev.1 Reordered choices from longest to shortest
Rev.2, 5/18/17
Swapped C and D to keep TS in numerical order.
Changed LOD to 4.
Editiorial changes.
Rev. 3, 6/8/17
Corrected typo.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1123 **Rev:** 1 **Rev Date:** 5/18/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-AOP **Objective:** 4 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 026 **System Title:** Loss of Component Cooling Water

Description: Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water: Control of flow rates to components cooled by the CCWS.

K/A Number: AA1.06 **CFR Reference:** 41.7 / 45.5 / 45.6

Tier: 1 **RO Imp:** 2.9 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 2.9 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 7 **SRO:**

Given:
* 100% power
* P-33B ICW pump tagged out for maintenance

ATC reports:
* ICW FLOW LO (K12-B4) alarms
* P-33C Nuclear ICW pump tripped and will not re-start

Which of the following actions will be in compliance with 1203.052, Loss of Intermediate Cooling Water?

A. Total ICW flow 3250 gpm and ICW isolated to one SFP Cooler.

B. Total ICW flow 3050 gpm and Letdown re-established with one Letdown Cooler.

C. Total ICW flow 3250 gpm and Letdown re-established with one Letdown Cooler.

D. Total ICW flow 3050 gpm and ICW isolated to one SFP Cooler.

Answer:

B. Total ICW flow 3050 gpm and Letdown re-established with one Letdown Cooler.

Notes:

"B" is correct, with the "B" ICW pump tagged out and the Nuclear ICW pump tripping there will only be the "A" ICW pump running so total ICW flow must be maintained less than or equal to 3100 gpm to prevent pump runout. Initiallly Letdown is isolated but then is re-stablished following isolation of ICW to one Letdown cooler.

"A" is incorrect but plausible, the total ICW flow is close to the required value but is too high. Also, ICW is isolated to both SFP Coolers, not just one.

"C" is incorrect but plausible, since the action for Letdown flow is correct. The total ICW flow value is close to the required value but is too high.

"D" is incorrect but plausible, the total ICW flow value is correct but ICW is isolated to both SFP Coolers, not just one.

This question matches the K/A as a loss of CCW has occurred (ICW pump trip) and the letdown flow rate

References:

1203.052, Loss of Intermediate Cooling Water

History:

New question for 2017 RO Re-take exam
Rev.1, 5/18/17
Editorial changes.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0933 **Rev:** 1 **Rev Date:** 5/18/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-NNI **Objective:** 14 **Point Value:** 1

Section: 4.2 **Type:** Generic Abnormal Plant Evolutions

System Number: 027 **System Title:** Pressurizer Pressure Control Malfunction

Description: Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following: Controllers and positioners.

K/A Number: AK2.03 **CFR Reference:** 41.7 / 45.7

Tier:	1	**RO Imp:**	2.6	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	2.8	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 8 **SRO:**

Given:
* 100% power
* "A" MFW pump trips

Current plant conditions are:
* 85% power and lowering
* RCS pressure has lowered to 2110 psig

In accordance with Pressurizer Systems Failure (1203.015) which of the following indicates a malfunction of the Pressurizer Pressure Control System?

A. Pressurizer spray valve OPEN

B. Heater bank 3 ON

C. Heater bank 4 ON

D. Heater bank 5 ON

Answer:

D. Heater bank 5 ON

Notes:

"D" is correct, Bank 5 should not be on until 2105 psig so if Bank 5 were ON at 2110 psig it would be a malfunction of the Pressurizer Pressure controller.

"A" is incorrect but plausible, the spray valve should normally close at 2155 psig. However a MFW Pump trip above 80% power reduces the 2205 open setpoint for the spray valve to 2080 psig and the close setpoint drops to 2030 psig so the spray valve should be open (it is NOT a malfunction).

"B" is incorrect but plausible since this is another heater bank but Bank 3 turns on at 2135 psig so it should be on at 2110 psig and thus is not a malfunction.

"C" is incorrect but plausible since this is another heater bank but Bank 4 turns on at 2120 psig so it should be on at 2110 psig and thus is not a malfunction.

This question matches the K/A since it involves the Pressurizer Pressure Control system and determines if applicant can recognize a malfunction of the Pressure Controller.

References:

1103.005, Pressurizer Operation
1203.015, Pressurizer Systems Failure, Section 3, Inoperative Pressurizer Heaters

History:

New for 2014 Exam - NOT USED due to similarities with another question.
New for 2017 RO Re-exam.
Rev. 1, 5/18/17

Added 1203.015 to stem and references.
Editorial changes.

QID: 1124 **Rev:** 2 **Rev Date:** 6/8/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-ESAS **Objective:** 5 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 040 **System Title:** Steam Line Rupture

Description: Ability to determine and interpret the following as they apply to the Steam Line Rupture: Conditions requiring ESFAS initiation.

K/A Number: AA2.04 **CFR Reference:** 41.7 / 43.5 / 45.13

Tier: 1 **RO Imp:** 4.5 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 4.7 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 9 **SRO:**

Given:
* Steam line break in Reactor Building

When performing RT-10, what RB pressure would require actuation of ESAS Channels 1-10?

A. 15.2 psia

B. 18.7 psia

C. 30.0 psia

D. 44.7 psia

Answer:

D. 44.7 psia

Notes:

"D" is correct, RB pressure of 44.7 psia (30 psig) requires actuation of all ESAS channels.

"A" is incorrect, but plausible since this is an abnormal RB pressure and would require operator action to address, but no actuations would occur.

"B" is incorrect, but plausible since this is the actuation setpoint for channels 1-6, but not for channels 7-10.

"C" is incorrect, but plausible since this is similar to the actuation setpoint for channels 7-10, i.e., 30 psig.

This question matches the K/A since it requires the recognize the correct setpoint for ESFAS actuation following a major steam line rupture.

References:

1202.012, Repetitive Tasks, RT-10, Verify Proper ESAS Actuation

History:

New for 2017 RO Re-exam
Rev. 1, 5/18/17
Editorial changes.
Rev. 2, 6/8/17
Added CFR number.

QID: 0623 **Rev:** 1 **Rev Date:** 5/18/17 **Source:** Bank **Originator:** Cork

TUOI: A1LP-RO-ICS **Objective:** 28 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 054 **System Title:** Loss of Main Feedwater (MFW)

Description: Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW): Occurrence of reactor and/or turbine trip.

K/A Number: AA2.01 **CFR Reference:** 41.5 / 41.10

Tier:	1	**RO Imp:**	4.3	**RO Select:**	Yes	**Difficulty:**	4
Group:	1	**SRO Imp:**	4.4	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 10 **SRO:**

Given:
* Plant startup in progress
* Per procedure, second MFW pump P-1B being placed in service at 350 MW
* Startup of P-1B complete to point of placing H/A station in AUTO
* B MFP TURBINE TRIP (K07-A8) alarms

Subsequently both Startup and Low Load Control valves are observed to go closed.

Which of the following resulted in the closure of the valves?

A. Turbine trip

B. Reactor trip

C. Loss of Instrument Air

D. Loss of power to H1 bus

Answer:

B. Reactor trip

Notes:

"B" is correct, on a reactor trip the ICS Rapid Feedwater Reduction (RFR) circuit will close the Startup and Low Load control valves. This ICS design feature closes these valves to totally stop Feedwater thereby preventing an overcooling. The RFR circuit will release the Startup and Low Load control valves when SG levels drop to the low level limit setpoint of 45" (40" is normal setpoint but a reactor trip will bias this setpoint by 5" to limit "undershoot").

"A" is incorrect, although plausible since the Turbine trip signal is an input into several ICS control schemes, but this would have no direct effect on the FW system. The condition of 350 Mwe is equivalent to 39% power which is below the 43% threshold of generating a reactor trip due to a turbine trip.

"C" is incorrect, although plausible since both valves are actuated by Instrument Air, but the Low Load Control Valves fail "As-Is" on a loss of Inst. Air and so will not be fully closed.

"D" is incorrect, although plausible since a loss of power to the H1 bus means a loss of one RCP in each loop which will generate an ICS runback. A runback would close the block valves but will not close the control valves.

This question matches the K/A since it involves a loss of Main Feedwater (trip of B pump) and requires interpretation of what could have closed the FW valves, i.e., a reactor trip.

References:

STM 1-64, Integrated Control System
1203.012F, Annunciator K07 Corrective Action

History:

New for 2005 RO re-exam.
Selected for 2017 RO Re-exam.
Rev. 1, 5/18/17
Changed answers to short to long.
Deleted P1B trips from 3rd bullet and added 4th bullet for B MFP trip alarm.
Replaced CFR references.
Re-phrased stem.
Editorial changes.

QID: 1125 **Rev:** 1 **Rev Date:** 5/18/17 **Source:** Modified **Originator:** Cork

TUOI: A1LP-RO-EOP08 **Objective:** 14 **Point Value:** 1

Section: 4.1 **Type:** Generic EPEs

System Number: 055 **System Title:** Station Blackout

Description: Knowledge of the operational implications of the following concepts as they apply to the Station Blackout: Natural circulation cooling

K/A Number: EK1.02 **CFR Reference:** 41.8 / 41.10 / 45.3

Tier: 1 **RO Imp:** 4.1 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 4.4 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 11 **SRO:**

Given:
* After 200 days of continuous operation Unit 1 tripped from 100% due to loss of off site power
* Both EDG's tripped and repair actions in progress
* Natural circulation established for one hour with Tave 545 °F and Thot 565 °F

In accordance with RT-5, Verify Proper EFW Actuation and Control, with primary to secondary heat transfer in progress, the operator should observe:

A. Core delta T rising

B. Tcold tracking CET temperature

C. Tcold tracking SG Tsat temperature

D. ADVs opening periodically to maintain SG pressure

Answer:

C. Tcold tracking SG Tsat temperature

Notes:

"C" is correct. If natural circulation is cooling the core, then Tcold should be tracking SG Tsat temperature.

"A" is incorrect since Core delta T should be stable or dropping, but this is plausible since core delta T will rise early in the event before natural circulation is established, but this event has been going on for an hour.

"B" is incorrect but plausible as an RCS temperature should be tracking CETS but it will be Thot, not Tcold.

"D" is incorrect, as long as heat removal is continuing, secondary heat must be removed by steaming continuously. This is plausible since periodic ADV opening could be observed if forced circulation were in progress.

Modified version of QID 124: changed EDGs to "tripped" so this question would apply to Blackout, replaced Natural Circulation AOP with RT-5, replaced "D" distractor since TBVs aren't available, changed "B" to "rising", changed "A" to "Tcold" to make it the correct answer.

This question matches the K/A since it has Blackout condtions and tests the knowledge of an operational implication: how to verify natural circulation cooling is in progress.

References:

1202.008, Blackout
1202.012, Repetitive Tasks, RT-5, Verify Proper EFW Actuation and Control

History:

Modifed QID 124 for 2017 RO Re-exam
Rev. 1, 5/18/17
Re-ordered answers short to long.

Added "after 200 days continuous operation" to first bullet.
Editorial changes.

QID: 1145 **Rev:** 0 **Rev Date:** 3/22/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-EOP07 **Objective:** 11 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 056 **System Title:** Loss of Off-site Power

Description: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

K/A Number: 2.4.4 **CFR Reference:** 41.10 / 43.2 / 45.6

Tier:	1	**RO Imp:**	4.5	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.7	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 12 **SRO:**

Given:
* Unit 1 tripped from 100% power
* A3 & A4 buses energized from EDGs, all other 4160v/6900v buses de-energized
* CETs 615 °F and rising slowly
* RCS pressure 2000 psig and rising slowly
* Both SG pressures 1000 psig and steady

Which EOP is required to be in use for the above conditions?

A. 1202.002, Loss of Subcooling Margin

B. 1202.004, Overheating

C. 1202.005, Inadequate Core Cooling

D. 1202.007, Degraded Power

Answer:

D. 1202.007, Degraded Power

Notes:

"D" is correct, since only EDGs are powering 4160V buses following a reactor trip, the entry condition for Degraded Power (1202.007) has been met. 1202.007 has sections designed to mitigate a loss of Subcooling Margin (SCM), overheating, and overcooling events.

"A" is incorrect but highly plausible since SCM does not exist in the conditions given, however, Degraded Power beginning at step 24 has actions to mitigate a loss of SCM.

"B" is incorrect but highly plausible since an overheating condition exists (CETs >610 °F with RCPs off), however Degraded Power beginning at step 55 has actions to mitigate overheating.

"C" is incorrect but plausible due to elevated RCS core exit conditions but entry conditions are not met for the ICC procedure (CETs are not moving to the right of the saturation line.)

This question matches the K/A since a loss of offsite power condition is given which is the entry condtion for the loss of offsite power EOP (actually entry conditions are met for serval EOPs).

References:

1202.007, Degraded Power
A1LP-RO-EOP07, Degraded Power lesson plan

History:

New for 2017 RO Re-exam

QID: 1163 **Rev:** 1 **Rev Date:** 5/18/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-NNI **Objective:** 21 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 057 **System Title:** Loss of Vital AC Instrument Bus

Description: Ability to operate and / or monitor the following as they apply to the Loss of Vital AC Instrument Bus: Backup instrument indications.

K/A Number: AA1.05 **CFR Reference:** 41.7 / 45.5 / 45.6

Tier: 1 **RO Imp:** 3.2 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 3.4 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 13 **SRO:**

Given:
* Power lost to 120VAC Panel RS-2

Aside from the Plant Computer or SPDS, where can a backup indicator for RCS Flow be found?

A. Control Room panel C18

B. Control Room panel C03

C. Control Room back panel C486-1

D. Computer Room Dasey panel C166

Answer:

B. Control Room panel C03

Notes:

"B" is correct, although all four RPS Channels have an input from RCS Flow for protective trips, only A and B RPS channels provide RCS flow indication. Panel C03 in the Control Room has alternate indications for RCS Flow from "A" RPS channel.

"A" is incorrect, but plausible since panel C18 has indications for RB pressure, HPI flow, BWST level, and LPI flow but it does not have an indicator for RCS flow.

"C" is incorrect but plausible since C486-1 has alternate indications for other important parameters from RPS, such as RB pressure, but it does not have RCS flow.

"D" is incorrect but plausible since the Dasey Panel alternate indications for other important parameters from RPS, such as RCS pressure from A RPS channel, but it does not have RCS flow.

This matches the K/A since the condition is a loss of a vital AC instrument panel and the question requires applicant to recall the location of backup indication.

References:

STM 1-63, Reactor Protection System

History:

New for 2017 RO Re-exam
Rev. 1, 5/18/17
Deleted 2nd condition.
Arranged answers short to long.
Editiorial changes.

QID: 1156 **Rev:** 1 **Rev Date:** 5/18/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-AOP **Objective:** 3 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 058 **System Title:** Loss of DC Power

Description: Ability to determine and interpret the following as they apply to the Loss of DC Power: DC loads lost; impact on ability to operate and monitor plant systems.

K/A Number: AA2.03 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier:	1	**RO Imp:**	3.5	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.9	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 14 **SRO:**

Given:
* Reactor tripped from 100% power due to loss of 125VDC bus D02

In accordance with Loss of 125V DC (1203.036), which of the following is a de-energized DC load and what is the plant impact?

A. Generator Lockout Relays,
all even buses de-energized

B. Vital AC panels RS2 and RS4,
inadvertent ESAS actuation

C. DG2 control power,
DG2 will not start from control room

D. Main Turbine trip block solenoid,
Main Turbine can not be tripped from control room

Answer:

C. DG2 control power,
DG2 will not start from control room

Notes:

A loss of 125VDC bus D02 will also cause a loss of DC panel RA2. A loss of RA2 causes RPS to see a loss of two RCPs due to loss of the RCP contact monitors powered from RA2. A loss of 2 RCPs at 100% power will cause a reactor trip.

"C" is correct, control power for DG2 comes from D02 via distribution panel D21, without control power the EDG can not be started from the control room, it can only be started locally using the "No DC" start procedure.

"A" is incorrect, the Generator Lockout relays are powered from D01 via D11, but the plant impact is correct, all even buses will be de-energized but they will be de-energized due to the loss of control power to the breakers so they can not auto-transfer to a Startup Transformer.

"B" is incorrect, but plausible since RS2 and RS 4 are lost but only one of the ESAS Analog channels will be lost so there will not be an inadvertent ESAS actuation. An inadvertent ESAS actuation does occur on a loss of D01, but not D02.

"D" is incorrect, control power for Main Turbine trip block solenoid comes from D01, but plausible since it is powered from Vital DC.

This question matches the K/A since it involves a loss of DC power (D01) and evaluates knowledge of DC loads and the impact on the plant from the loss of some of those loads.

References:

1203.036, Loss of 125V DC

History:

New question for 2017 RO re-exam
Rev. 1, 5/18/17
Added 1203.036 to stem.
Re-ordered answers short to long.
Editorial changes.

QID: 0947 **Rev:** 1 **Rev Date:** 5/18/17 **Source:** Bank **Originator:** NRC

TUOI: A1LP-RO-MSSS **Objective:** 3 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 062 **System Title:** Loss of Nuclear Service Water

Description: Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water: The automatic actions (alignments) within the nuclear service water resulting from the actuation of the ESFAS.

K/A Number: AK3.02 **CFR Reference:** 41.4, 41.8 / 45.7

Tier: 1 **RO Imp:** 3.6 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 3.9 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 15 **SRO:**

Given:
* 100% power
* All Service Water pumps in service
* All Service Water cross-tie valves open and in AUTO
* P-4B MOD breaker aligned to A4

Following ESAS channels 1 and 2 actuation with offsite power available, how will the Service Water System respond and what is the reason for this response?

A. All cross-tie valves remain open;
maximize system reliability

B. All cross-tie valves close and P-4B trips;
maintain train separation

C. All cross-tie valves close and P-4B continues to run;
provide cooling water to ACW loop

D. Cross-tie valves between P-4B and P-4C remain open;
prevent deadheading pump

Answer:

D. Cross-tie valves between P-4B and P-4C remain open;
prevent deadheading pump

Notes:

"D" is correct, with the system aligned as described in the stem, the cross-tie valves between P-4B and C will remain open since all 3 SW pumps are running and offsite power is available. This prevents deadheading the pump.

"A" is incorrect since the cross-tie valves between P-4A and B will close, but plausible if the candidate can recall that all 3 pumps will remain running and cannot recall that train separation is required after ESAS actuation. All four crosstie valves open is the normal system configuration and is done so to maximize system reliability.

"B" is incorrect yet plausible since all crosstie valves would close if only A and C SW pumps were running. This could be the case if offsite power were lost coincident with the ESAS actuation. SW pump P-4B would not be re-started after the EDG energizes the bus as long as P-4C was available.

"C" is incorrect yet plausible if the candidate recalls that all 3 SW pumps will remain runnning but cannot recall what happens with the ACW isolation. The ACW isolation can be powered from either B55 or B56, and thus will be affected by ESAS channels 1 or 2 depending on its power supply. The ACW isolation will remain open in certain situations but only if Channel 1 or 2 actuated (not both) with all 3 pumps runnning and offsite power available. This unique set of conditions would prevent deadheading the B pump due to inadvertent ESAS actuation of a single channel, but is incorrect for this question since two channels actuated.

Swapped "A" and "B" since this is a bank question. Separated automatic actions from reasons to make question more clear.

This question matches the K/A as the conditions are given for an ESAS actuation and the candidate is required to know how the Service Water system will re-align and the reason for that alignment.

References:

STM 1-42, Service and Auxiliary Cooling Water

History:

New for 2013 Exam
Selected for 2017 RO Re-exam
Rev.1, 5/18/17
Re-ordered answers short to long (swapped A and D).
Editorial changes.

QID: 0108 **Rev:** 3 **Rev Date:** 5/18/17 **Source:** Modified **Originator:** Cork

TUOI: A1LP-RO-AOP **Objective:** 3 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 065 **System Title:** Loss of Instrument Air

Description: Knowledge of the reasons for the following responses as they apply to the Loss of Instrument Air: Knowing effects on plant operation of isolating certain equipment from instrument air.

K/A Number: AK3.03 **CFR Reference:** 41.5 / 41.10 / 45.6 / 45.13

Tier: 1 **RO Imp:** 2.9 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 3.4 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 16 **SRO:**

Given:
* 100% power
* Instrument air leak requires manual isolation of air to Turbine Bypass Valve (CV-6688)

Since the leak is downstream of the isolation point, the leak eventually causes the instrument air accumulators for CV-6688 to depressurize.

What effect will this have on CV-6688 operation and why?

A. OPENS due to system pressure

B. OPENS due to spring pressure

C. CLOSES due to loss of IA pressure

D. CLOSES due to spring pressure

Answer:

A. OPENS due to system pressure

Notes:

"A" is correct. On a loss of instrument air the accumulator for CV-6688 will maintain the valve closed for a period of time, but the given condition of a failure of the instrument air to the valve operator (with a leak) will result in a loss of accumulator pressure. At the given power level, the valve will be forced open by steam pressure.

"B" is incorrect but plausible since some air operated valves do fail open by spring pressure on loss of air but there is no spring.

"C" is incorrect but plausible if the candidate believes the air pressure is required to open the valve vice hold it closed.

"D" is incorrect but plausible if the candidate believes the actuator contains a spring and the valve failed closed on loss of air but there is no spring.

Modified question to be a "2 x 2" due to two implausible distractors in the original question. Two distractors revised and condition was changed from a severed line to Inst. Air lost with a small leak.

This question matches the K/A due to a loss of Instrument Air condition and the applicant must know the effect on the equipment and the reason for that effect.

References:

1203.024, Loss of Instrument Air, Attachment A

History:

Developed for the 1998 RO/SRO Exam.
Used in 98 RO Re-exam
Selected for 2002 RO/SRO exam.
Selected for 2007 RO Exam. Changed KA from 041 K6.03 to 065 AK3.03.
Modified for 2017 RO Re-exam.
Rev. 2 - Changed Distractor C based on feedback from the NRC.
Rev. 3, 5/18/17
Editorial changes.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0514 **Rev:** 2 **Rev Date:** 5/18/17 **Source:** Modified **Originator:** NRC

TUOI: A1LP-RO-EOP04 **Objective:** 14 **Point Value:** 1

Section: 4.3 **Type:** B&W EPE/APE

System Number: E04 **System Title:** Inadequate Heat Transfer

Description: Knowledge of the interrelations between the (EOP Rules) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

K/A Number: EK2.1 **CFR Reference:** 41.7 / 45.7

Tier:	1	**RO Imp:**	3.8	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.0	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 17 **SRO:**

Given:
* Overheating (1202.004) is in use
* HPI Cooling (RT-4) is also in use

What is the preferred method of operation of the ERV in this situation, AND why is this method preferred?

A. Manual; minimizes the cycles and reduces the chance of ERV failure.

B. Manual; ensures ES actuations occur to simplify the subsequent operator actions.

C. Automatic; minimizes the cycles and reduces the chance of ERV failure.

D. Automatic; ensures ES actuations occur to simplify the subsequent operator actions.

Answer:

A. Manual; minimizes the cycles and reduces the chance of ERV failure.

Notes:

"A" is correct, automatic operation would make the ERV open and close more often in reaction to RCS pressure, more cycling of the valve increases the chance of ERV failure, thus manual operation is preferred.

"B" is incorrect but plausible since opening the ERV manually is the preferred method per the EOP bases but the reason is incorrect since another reason for manual operation of the ERV is also to prevent ES actuations, not cause them as stated in this distractor.

"C" is incorrect but plausible since the reason give is the correct one but this choice lists automatic operation instead of manual operation.

"D" is incorrect, this distractor completes the 2x2 choices with both the incorrect mode of operation and the incorrect reason for the preferred mode.

Modified all answers by making them into a 2x2 format, this eliminates implausible distractors present in the original question. Rephrased stem of question to be more specific.

This question matches the K/A due to an Overheating condition is given and it requires the applicant to know the interrelation between the event and the method of operation of a component, i.e., the ERV, per the EOP bases document for Inadequate Heat Transfer and Repetitive Task (Rule) for HPI Cooling

References:

GEOG bases, HPI Cooling
GEOG bases III.C, Inadequate Heat Transfer
1202.004, Overheating
1202.012, Repetitive Tasks
Bases for 1202.012

History:

Developed by NRC.
Taken from Crystal River, Unit 3, Date: 03/22/1996
Selected for 2017 RO Re-exam
Rev. 2, 5/18/17
Editorial changes.

QID: 1126 **Rev:** 2 **Rev Date:** 5/18/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-GEN **Objective:** 7 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 077 **System Title:** Generator Voltage and Electric Grid Disturbances

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

K/A Number: 2.1.25 **CFR Reference:** 41.10 / 43.5 / 45.12

Tier: 1 **RO Imp:** 3.9 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 4.2 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 18 **SRO:**

***************REFERENCE PROVIDED****************************

Given:

* Thunderstorms in the area

* Due to plant issues, Unit 1 is operating at 720 MWe under-excited with a power factor of .98

* Main Generator Hydrogen pressure is 60 psig

* Dispatcher calls Control Room, says lightning has tripped a capacitor bank, and requests Unit 1 reactive load be raised as much as possible

* Dispatcher states Main Generator electrical load and power factor must NOT change

What is the MAXIMUM change in reactive load allowed for the above conditions per Power Operation (1102.004)?

A. 550 MVARs

B. 340 MVARs

C. 280 MVARs

D. 140 MVARs

Answer:

C. 280 MVARs

Notes:

"C" is correct, operating at a PF of .98 under-excited with a load of 720 MW means the Main Generator reactive load must be -140 MVARs. The most reactive load which could be raised is where the 720 MW line intersets the .98 PF line in the over-excited half of Att. N. This would be -140 to +140 for a total of 280 MVARs. This also meets an Ops log restriction of a max of +160 MVARs.

"A" is incorrect yet plausible if the candidate mistakenly starts at the 720 MW line at .98 PF and goes until the 720 MW line intersects the 60 psig limit line, then performs the calculation.

"B" is incorrect yet plausible if the candidate mistakenly starts where the .98 PF line intersects the 60 psig limit line on the under-excited side and draws a line straight up to where the 60 psig limit intersects the .98 PF on the over-excited side and performs the calculation.

"D" is incorrect but is plausible if the candidate mistakenly starts at the 720 MW line at 1.0 PF and draws as line to the .98 PF line.

This question matches the K/A since it involves a grid disturbance and requires the candidate to use the

provided graph and conditions to arrive at the correct answer.

Rev. 1 - Formatted stem with bullets based on feedback from NRC.

References:

1102.004, Power Operation

1102.004, Attachment N, must be in RO handout!!!

History:

New question for 2017 RO Re-exam
Rev. 2, 5/18/17
Editorial changes.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0397 **Rev:** 3 **Rev Date:** 5/18/17 **Source:** Bank **Originator:** J.Cork

TUOI: A1LP-RO-NOP **Objective:** 4 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 001 **System Title:** Continuous Rod Withdrawal

Description: Ability to determine and interpret the following as they apply to the Continuous Rod Withdrawal: Proper actions to be taken if automatic safety functions have not taken place.

K/A Number: AA2.03 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier:	1	**RO Imp:**	4.5	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	4.8	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 19 **SRO:**

Given:
* Approach to criticality in progress
* Reactor power is in Source Range
* CBOR commences sequential withdrawal of regulating rods

The following indications are observed:
* SR (Source Range) count rate rising
* Sustained SUR of 2.5 DPM
* Continued outward rod motion without a command

What action is required to be taken (or verified to occur) per 1102.008, Approach to Criticality?

A. Select "JOG" on rod speed selector switch.

B. Trip reactor and go to Reactor Trip (1202.001).

C. Verify Out Inhibit on high SUR stops rod motion automatically.

D. Commence Emergency Boration per RT-12 until SUR is negative.

Answer:

B. Trip reactor and go to Reactor Trip (1202.001).

Notes:

"B" is correct, the Source Range rod hold on high SUR of 2.5 DPM should have stopped outward rod motion. A system failure has occurred, control of reactivity has been lost, and reactor should be tripped.

"A" is incorrect, plausible since this is one of the steps to stop rod motion but not all of them.

"C" is incorrect, plausible since the SR High SUR inhibit should have stopped rod motion at 2 DPM but it would be imprudent to wait to see if the IR High SUR inhibit at 3 DPM would work.

"D" is incorrect, plausible since emergency boration will add negative reactivity but this method is too slow.

Revised question by adding procedure to stem and changing order of answers.

This matches the K/A since conditions are given for a continuous rod withdrawal event and to answer the question correctly requires knowledge of the proper procedural action.

Rev. 2 - Reformatted stem based on NRC comment.

References:

1102.008, Approach to Criticality

History:

New created for 2001 SRO Exam.

No longer SRO, selected for 2017 RO Re-exam.
Rev. 3, 5/18/17
Editorial changes.

QID: 1127 **Rev:** 2 **Rev Date:** 6/8/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-ACRD **Objective:** 26 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 003 **System Title:** Dropped Control Rod

Description: Knowledge of the reasons for the following responses as they apply to the Dropped Control Rod: Actions contained in EOP for a dropped control rod.

K/A Number: AK3.04 **CFR Reference:** 41.5, 41.10 / 45.6 / 45.13

Tier: 1 **RO Imp:** 3.8 **RO Select:** Yes **Difficulty:** 3

Group: 2 **SRO Imp:** 4.1 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 20 **SRO:**

Given:
* 38% power
* ICS in full automatic
* Rod 9 in Group 5 has dropped
* CONTROL ROD ASYMMETRIC (K08-C2) alarms
* All actions in response to dropped rod are complete

In accordance with CRD System Operating Procedure (1105.009) which of the following actions must be performed FIRST to recover the dropped rod, and why must the rod be withdrawn within limits of Control Rod Drive Malfunction Action (1203.003)?

A. Depress FAULT RESET on the Diamond panel;
ensure power peaking and shutdown margin stay within design limits.

B. Latch the dropped rod using auxiliary power supply;
ensure power peaking and shutdown margin stay within design limits.

C. Latch the dropped rod using auxiliary power supply;
ensure operation remains within accident analysis for an ejected rod.

D. Depress FAULT RESET on the Diamond panel;
ensure operation remains within accident analysis for an ejected rod.

Answer:

B. Latch the dropped rod using auxiliary power supply;
ensure power peaking limits and shutdown margin stay within design limits.

Notes:

"B" is correct, the rod is in Group 5 which is the first group to be withdrawn and thus will be at the out limit, so per 1105.009 the dropped rod is first latched using the aux power supply, and per lesson plan (A1LP-RO-CRD) a misaligned control rod may cause increased power peaking and reduction in total available shutdown margin.

"A" is incorrect and plausible since the reason given is correct but FAULT RESET is not pressed until after the rod is leveled with its group.

"C" is incorrect and plausible since the action given is the first one to be performed per 1105.009 but the reason given is not correct per lesson plan.

"D" is incorrect as it contains both the incorrect action and the incorrect reason. The incorrect action is plausible if the student forgets CRD logic and assumes the fault must be reset before outward rod motion can take place (e.g., out inhibit). The incorrect reason is plausible since there is an accident analysis in chapter 14 of the SAR for an ejected rod accident.

This question matches the K/A since condtions are given for a dropped rod and the candidate must know how a dropped rod is recovered, and why a dropped rod is re-leveled with its group.

This question has ANO-1 specific OE where a fault occurred with a CRD at 100% power. The crew did not

lower power and depressed FAULT RESET. This removed the withdrawal inhibit and the rods immediately began pulling, causing power to go above 100% (no trip, just greater than 100%).

References:

1105.009, CRD System Operating Procedure
TS 3.1.4 Bases
1203.003, Control Rod Drive Malfunction Action

History:

New question for 2017 RO Re-exam.
Rev. 1, 5/18/17
Added 1105.009 to beginning of stem and added "within the rate limits of 1203.003" to end of stem.
Added TS and 1203.003 to references.
Editorial changes.
Rev. 2, 6/8/17
Corrected typo.

QID: 1128 **Rev:** 1 **Rev Date:** 5/19/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-NI **Objective:** 10 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 033 **System Title:** Loss of Intermediate Range NI

Description: Knowledge of the operational implications of the following concepts as they apply to Loss of Intermediate Range Nuclear Instrumentation: Effects of voltage changes on performance.

K/A Number: AK1.01 **CFR Reference:** 41.8 / 41.10 / 45.3

Tier:	1	**RO Imp:**	2.7	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.0	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 21 **SRO:**

Given:
* Plant startup (approach to criticality) in progress
* The following NI readings are observed:
  * Channel 1 Source Range, NI-501 6 E4 cps
  * Channel 2 Source Range, NI-502 5 E4 cps
  * Intermediate Range Channel NI-3 2 E-11 amps
  * Intermediate Range Channel NI-4 6 E-10 amps
  * Power Range Channels NI-5 thru 8 0%

Which of the following conclusions is correct for the above indications?

A. NI-3 is reading low due to low compensating voltage (under-compensated)

B. NI-3 is reading low due to high compensating voltage (over-compensated)

C. NI-4 is reading high due to low compensating voltage (under-compensated)

D. NI-4 is reading high due to high compensating voltage (over-compensated)

Answer:

B. NI-3 is reading low due to high compensating voltage (over-compensated)

Notes:

"B" is correct, Intermediate Range Channel NI-3 is reading very low and does not exhibit one decade overlap from Source Range to Intermediate Range. The compensating voltage set too high would definitely cause a low reading by subtracting too much from the detector's overall output.

"A" is incorrect, while NI-3 is reading low (making this answer plausible), but a low compensating voltage will cause the detector to read high, not low.

"C" is incorrect, yet plausible since a low compensating voltage could cause NI-4 to read too high, NI-4 is exhibiting one decade of overlap with the Source Range channels and is thus reading correctly.

"D" is incorrect, yet plausible if the candidiate determines NI-4 is reading high (vs. NI-3 reading low), and believes compensating voltage and indication go together (high compensating voltage causes high indication).

This question matches the K/A since it concerns the affect of Intermediate Range detector performance as it is affected by a voltage change.

References:

STM 1-67, Nuclear Instrumentation

History:

New question for 2017 RO Re-take exam
Rev. 1, 5/19/17
Editorial changes.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1129 **Rev:** 1 **Rev Date:** 5/19/17 **Source:** Modified **Originator:** Cork

TUOI: A1LP-RO-ICS **Objective:** 13 **Point Value:** 1

Section: 4.3 **Type:** B&W EPE/APE

System Number: A01 **System Title:** Plant Runback

Description: Knowledge of the interrelations between the (Plant Runback) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

K/A Number: AK2.1 **CFR Reference:** 41.7 / 45.7

Tier:	1	**RO Imp:**	3.7	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.5	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 22 **SRO:**

Given:
* Plant at 88% power for Main Turbine TV/GV testing
* P-1A MFWP suddenly trips

Assuming the reactor does NOT trip, total MFW flow would be __________ E6 lbm/hr.

A. 4.4

B. 4.9

C. 6.0

D. 8.2

Answer:

A. 4.4

Notes:

"A" is correct, these are the indications for 40% power. There is a runback to this power for a loss of 1 MFWP or asymmetric rod. 100% power was assumed to be 11 x e6 lbm/hr total, 11 x 0.4 = 4.4

"B" is incorrect, these are the indications for 45% power. There is a runback to this power for a loss of 1 RCP in each loop which makes this plausible.

"C" is incorrect, these are the indications for 55% power. The runback for a loss of 1 RCP in each loop was formally at this power level which makes this plausible. Also, this runback is only applicable when RX power is less than 55% so this value is credible.

"D" is incorrect, ICS will runback the plant to 75% power for a loss of one RCP which makes this plausible.

This is a modified version of QID 635. The power level condition was changed to 88% power and a MFWP trip replaced the RCP trip. The answer changed from 8.2 x e6 to 4.4 x e6 lbm/hr.

This question matches the K/A since it requires the knowledge of the interrelation between MFW flow and the power level for the automatic runback feature in ICS for the trip of a MFWP.

References:

1105.004, Integrated Control System

History:

Modified QID 635 for 2017 RO Re-exam
Rev. 1, 5/19/17
Revised stem.
Re-ordered answers from low to high.

QID: 0020 **Rev:** 2 **Rev Date:** 5/19/17 **Source:** Bank **Originator:** GGiles

TUOI: A1LP-RO-NNI **Objective:** 6 **Point Value:** 1

Section: 4.3 **Type:** B&W EPEs/APEs

System Number: A02 **System Title:** Loss of NNI-X

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

K/A Number: AA1.1 **CFR Reference:** 41.7 / 45.5 / 45.6

Tier:	1	**RO Imp:**	4.0	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	3.8	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 23 **SRO:**

Given following indications/alarms:

* SASS MISMATCH (K07-B4) in alarm (fast flash)

* B OTSG BTU LIMIT (K07-E3) in alarm (slow flash)

* SG "B" FW Temp signal select switch selected to SASS Enable (AUTO) with white indicating light off and blue indicating light on

What operator action is required per ACA for SASS MISMATCH (K07-B4)?

A. Place both MFW pump H/A stations in HAND.

B. Place SG "B" FW Temp signal select switch to "Y" position.

C. No action necessary, SASS has automatically transferred to "X" NNI.

D. Depress AUTO pushbutton for SG "B" FW Temp on SASS panel in C47-2.

Answer:

B. Place SG "B" FW Temp signal select switch to "Y" position.

Notes:

"B" is the correct response per procedure 1203.012F and 1105.006.

"A" is incorrect, but plausible as 1203.001, ICS Abnormal Operation, directs taking the AFFECTED pump to hand in Section 11, but not both. Also, this action is not necessary since the SASS system has transferred to a good signal and no ICS upset should occur.

"C" is incorrect SASS has transferred to "Y" not "X", placing the selector switch to the input signal automatically transferred is a procedurally required action.

"D" is incorrect, but plausible since this action is performed inside the ICS cabinets when resetting a failed signal.

Changed "D" from "both FW loop demands in Manual" to "both MFW pump H/A stations in HAND" since the former was also a correct answer per 1203.001 section 11. Also, swapped correct answer from A to B position.

This question matches the K/A since a loss of an NNI-X instrument has occurred and the candidate must know that he needs to operate a manual feature in response to the failure.

References:

1203.012F, Annunciator K07 Corrective Action

History:

Developed for 1998 RO/SRO Exam.
Modified QID 3127
Used in 2001 RO/SRO Exam.
Selected for use in 2002 RO/SRO exam.
Selected for 2007 RO Exam.
Selected for 2017 RO Re-exam.
Rev. 1 - Changed Distractor C from "X" to "Y" as recommended by the NRC.
Rev. 2, 5/19/17
Changed "Y" to "X" in C distractor.
Swapped A and D to make answers short to long.
Editorial changes.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0026 **Rev:** 1 **Rev Date:** 5/19/17 **Source:** Bank **Originator:** GGiles

TUOI: A1LP-RO-RPS **Objective:** 14 **Point Value:** 1

Section: 4.3 **Type:** B&W EPEs/APEs

System Number: A04 **System Title:** Turbine Trip

Description: Knowledge of system purpose and/or function.

K/A Number: 2.1.27 **CFR Reference:** 41.7

Tier:	1	**RO Imp:**	3.9	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	4.0	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 24 **SRO:**

Which of the following describes the primary purpose of the Anticipatory Trip/Turbine Trip in RPS?

A. Limit plant cooldown following a loss of heat source.

B. Limit reactor power to prevent exceeding linear heat rate limits.

C. Limit reactor power to prevent exceeding DNBR limits.

D. Limit plant heatup following a loss of heat sink.

Answer:

D. Limit plant heatup following a loss of heat sink.

Notes:

"D" is correct, tripping the Reactor when the Main Turbine trips is an Anticipatory Trip whenever Rx power is greater than 43%. This limits plant heatup due to the loss of secondary heat removal from operation of the Main Turbine at a power level which would exceed the capacity of the Turbine Bypass Valves. At power levels above 43% the Reactor would trip anyway from a Main Turbine trip due to the primary response to a loss of secondary heat sink, hence the term "aniticipatory".

"A" is incorrect yet plausible since this is the reason for tripping the Main Turbine on a Reactor Trip.

"B" is incorrect yet plausible since this is the purpose of most of the RPS trips.

"C" is incorrect yet plausible since this is the purpose of most of the RPS trips.

References:

STM-1-63, Reactor Protection System

History:

Developed for 1998 SRO Exam. This question is no longer SRO Only. JWC 3/2/17
Selected for 2017 RO Re-exam.
Rev. 1, 5/19/17
Added "primary" prior to "purpose" in stem.
Changed "sink" to "source in A distractor.
Editorial changes.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0780 **Rev:** 1 **Rev Date:** 5/19/17 **Source:** Bank **Originator:** S.Pullin

TUOI: A1LP-RO-AOP **Objective:** 4 **Point Value:** 1

Section: 4.3 **Type:** B&W EPEs/APEs

System Number: A07 **System Title:** Flooding

Description: Ability to determine and interpret the following as they apply to the (Flooding): adherence to appropriate procedures and operation within the limitations in the facilities license and amendments.

K/A Number: AA.2.2 **CFR Reference:** 41.10 / 43.5 / 4 5.13

Tier:	1	**RO Imp:**	3.3	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.7	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 25 **SRO:**

Given:

* 100% power
* P-34A Decay Heat pump OOS
* Lake Dardanelle level 346 feet and rising due to heavy rains
* Corps of Engineers predicts peak flood levels will reach 355 feet

What action is required per Natural Emergencies (1203.025) Section 6, Flood?

A. Perform Rapid Plant Shutdown (1203.045) and align "B" Decay Heat pump for Decay Heat removal.

B. Perform Rapid Plant Shutdown (1203.045) and make preparations to transfer plant auxiliaries to SU2 transformer.

C. Perform Power Reduction and Plant Shutdown (1102.016) and align "B" Decay Heat pump for Decay Heat removal.

D. Perform Power Reduction and Plant Shutdown (1102.016) and make preparations to transfer plant auxiliaries to SU2 transformer.

Answer:

B. Perform Rapid Plant Shutdown (1203.045) and make preparations to transfer plant auxiliaries to SU2 transformer.

Notes:

"B" is correct due to 1203.025 directing performance of a shutdown per 1203.045 when lake level is greater than 345 feet, and SU2 transformer is designed for flooding and auxiliaries will be transferred to it following some preparatory activities.

"A" is incorrect yet plausible since 1203.025 directs one to perform a shutdown per 1203.045, but 1203.025 states to ensure the operable DH loop is aligned for ES standby (LPI) if only one loop is available.

"C" is incorrect since the procedure does not state to use the normal plant shutdown procedure and 1203.025 states to ensure the operable DH loop is aligned for ES standby (LPI) if only one loop is available. Both procedure and action are incorrect.

"D" is incorrect since the procedure does not state to use the normal plant shutdown procedure but plausible since second half is correct per 1203.025.

This question matches the K/A since the candidate must adhere to the flooding procedure by performing a rapid plant shutdown when lake level exceeds a threshold value (345 ft.) to ensure ANO-1 does not continue operating when flood waters could exceed our design flood level.

References:

1203.025 , Natural Emergencies

History:

New for 2010 RO/SRO exam
Selected for 2017 RO Re-exam
Rev. 1, 5/19/17
Changed C and D to make this question a 2x2. First half is perform 112.016 and 2nd halves are the second parts of A and B.
Editorial changes.

QID: 1147 **Rev:** 1 **Rev Date:** 5/19/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-ASDCD **Objective:** 4 **Point Value:** 1

Section: 4.3 **Type:** B&W EPE/APE

System Number: E08 **System Title:** LOCA Cooldown

Description: Knowledge of the reasons for the following responses as they apply to the (LOCA Cooldown): Normal, abnormal and emergency operating procedures associated with (LOCA Cooldown).

K/A Number: EK3.2 **CFR Reference:** 41.5 / 41.10, 45.6, 45.13

Tier:	1	**RO Imp:**	3.0	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.6	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 26 **SRO:**

Given:
* Small Break LOCA Cooldown (1203.041) in use
* SCM adequate
* "A" SG has leaking MSSV
* All RCPs running

Considering the above conditions which RCP will be selected to trip and why?

A. P-32A; to leave the combination running with the least stringent NPSH limitations

B. P-32D; to leave the combination running with the least stringent NPSH limitations

C. P-32A; due to degraded SG "A"

D. P-32D; due to degraded SG "A"

Answer:

D. P-32D; due to degraded SG "A"

Notes:

"D" is correct. RCP P-32D should be tripped per 1203.041 since this RCP is in the loop with a degraded SG and it does not provide PZR spray.

"A" is incorrect, but plausible since this is in the "B" loop and would normally be tripped so as to leave C & D RCPs running due to the less stringent NPSH requirements when another RCP is tripped at lower RCS pressures. However, A SG is degraded and an RCP should be tripped in that loop.

"B" is incorrect, but plausible since this is the correct RCP to trip but the reason given is incorrect.

"C" is incorrect, but plausible since this is the correct reason but the wrong RCP, the A RCP is in the B loop at ANO-1.

This question matches the K/A since this is a LOCA cooldown situtation and the question requires knowledge of the reason for a step in an abnormal operating procedure.

References:

1203.041, Small Break LOCA Cooldown

History:

New for 2017 RO Re-exam
Rev. 1, 5/19/77
Editorial changes.

QID: 0348 **Rev:** 3 **Rev Date:** 5/19/17 **Source:** Bank **Originator:** Goad

TUOI: A1LP-RO-MU **Objective:** 10 **Point Value:** 1

Section: 4.3 **Type:** B&W EOP/AOP

System Number: E13 **System Title:** EOP Rules and Enclosures

Description: Knowledge of the operational implications of the following concepts as they apply to the (EOP Rules): Annunciators and conditions, indicating signals, and remedial actions associated with the (EOP Rules).

K/A Number: EK1.3 **CFR Reference:** 41.8 / 41.10 / 45.3

Tier:	1	**RO Imp:**	3.0	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	3.2	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 27 **SRO:**

Given:
* LOCA has occurred
* RCS pressure 950 psig
* CETs 525°F
* ESAS actuated on channels 1 through 4
* "ES" HPI pump has failed

"OP" HPI pump flow:

300 gpm to "A" HPI line
65 gpm to "B" HPI line
85 gpm to "C" HPI line
95 gpm to "D" HPI line

All valves are in ESAS actuated position.

Annunciator A HPI FLOW HI/LO (K11-A4) in alarm.

What operator action is required per RT-10, Verify Proper ESAS Actuation?

A. Close isolation valve for "A" HPI line.

B. Throttle all HPI line flows until they are within 20 gpm of each other.

C. Throttle "A" HPI valve until "A" line flow is within 20 gpm of "B" line flow.

D. Throttle "A" HPI valve until "A" line flow is within 20 gpm of "D" line flow.

Answer:

D. Throttle "A" HPI valve until "A" line flow is within 20 gpm of "D" line flow.

Notes:

"D" is correct, conditions give only one HPI pump running, so in accordance with RT-10 the highest flow HPI line should be throttled to within 20 gpm of the next highest line. This is done in the event of an HPI line break. This will also clear alarm K11-A4 which is also required per RT-10. All 4 flows add up to 545 gpm total which is greater than alarm setpoint of 450 gpm. Throttling A line to 115 gpm will put it within 20 gpm of D and bring total flow down to 360 gpm.

"A" is incorrect but plausible if applicant can recall 20 gpm value but does not recall how to apply it. If they are all within 20 gpm of each other, then this will clear the alarm but will not be in compliance with all procedure steps.

"B" is incorrect but plausible as the conditions are indicative of a break and normally breaks are isolated but the diagnosis of a break is not assured in this case, therefore procedural guidance has flow maintained but balanced.

"C" is incorrect but plausible if applicant can recall 20 gpm value but does not recall how to apply it and mistakenly applies it to be within 20 gpm of the lowest line flow. This will clear the alarm but will not be in compliance with all procedure steps.

Revised original question by adding HPI flow hi alarm to match K/A. Replaced distractor A since it said to do nothing. Added CET temperature of 525 °F to ensure conditions indicate a loss of Subcooling Margin (SCM). Made minor editorial changes.

This question matches the K/A since it involves an EOP Rule/Enclosure (RT-10 and Loss of SCM Rule) and an alarm with associated remedial actions.

References:

1202.012, Repetitive Tasks, RT-10, Verify Proper ESAS Actuation
123.012J, Annunciator K11 Corrective Action

History:

Used in 1999 exam.
Direct from ExamBank, QID# 1711 used in class exam
Selected for the 2008 RO Exam
Selected for 2017 RO Re-exam
Rev. 3, 5/19/17
Deleted reasons from B, C, and D since A did not contain a reason and K/A did not require knowledge of a reason.
Swapped A and B so answers are short to long.

QID: 1148 **Rev:** 1 **Rev Date:** 5/19/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-ARCP **Objective:** 2 **Point Value:** 1

Section: 3.4 **Type:** RCS Heat Removal

System Number: 003 **System Title:** Reactor Coolant Pump

Description: Knowledge of the operational implications of the following concepts as they apply to the RCPS: Effects of RCP shutdown on secondary parameters, such as steam pressure, steam flow, and feed flow.

K/A Number: K5.04 **CFR Reference:** 41.5 / 45.7

Tier: 2 **RO Imp:** 3.2 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 3.5 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 28 **SRO:**

Given:
* 75% power
* ICS in full automatic
* RCP P-32A stopped due to vibration

Subsequently "A" MFWP trips.

When the plant stabilizes (assuming no operator action) Main Feedwater flow should be ________ x E6 lbm/hr for "A" RCS loop and __________ x E6 lbm/hr for "B" RCS loop?

A. 3.0
 1.5

B. 1.5
 3.0

C. 3.3
 1.65

D. 1.65
 3.3

Answer:

A. 3.0
 1.5

Notes:

"A" is correct, the ICS will ratio FW flow so the "A" loop has 2/3 flow and the "B" loop has 1/3 flow with a total of 40% flow which is power level for a MFW Pump trip runback.

"B" is incorrect but plausible, this is the correct FW ratio and total flow but to the wrong loops.

"C" is incorrect but plausible, this would be the correct ratio of FW but to the wrong runback power level of 45% which is the runback limit for a loss of 1 RCP in each loop.

"D" is incorrect but plausible since this is the correct ratio and this is a runback limit but this is the limit (45%) for a loss of 1 RCP in each loop. The proportion of FW flow is backwards, the "A" loop should be higher than the "B".

This question matches the K/A since it requires the knowledge of the operational implications of the secondary effects (proportion of FW flow) due to an RCP shutdown.

References:

1105.004, Integrated Control System
1203.022, Reactor Coolant Pump Trip

History:

New for 2017 RO Re-exam
Rev. 1, 5/19/17
Editorial changes.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1146 **Rev:** 1 **Rev Date:** 5/19/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-RCS **Objective:** 23 **Point Value:** 1

Section: 3.4 **Type:** RCS Heat Removal

System Number: 003 **System Title:** Reactor Coolant Pump

Description: Knowledge of RCPS design feature(s) and/or interlock(s) which provide for the following:: Prevention of cold water accidents or transients.

K/A Number: K4.02 **CFR Reference:** 41.7

Tier:	2	**RO Imp:**	2.5	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	2.7	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 29 **SRO:**

What interlock and/or procedure limit prevents a cold water reactor power excursion when starting a Reactor Coolant Pump?

A. Reactor power < 22% to start any RCP

B. Cold Leg > 241 °F prior to stating the third RCP

C. SG Downcomer < 50 °F deltaT from cold leg temperature

D. Reactor power < 45% with no more than three RCPs running

Answer:

A. Reactor power < 22% to start any RCP

Notes:

"A" is correct per SAR 14.1.2.5 analysis. Allowing RCP starts only when power is less than 22% prevents a cooldown which, if the moderator coefficient is negative, will introduce positive reactivity into the core causing power to rise. An interlock of power less than 22% ensures RCS pressure rise will not exceed code pressure limits and the minimum DNBR will be greater than 1.3.

"B" is incorrect but plausible as this is the reason when starting the third RCP is allowed, this is a limit and precaution in 1103.006, Reactor Coolant Pump Operation.

"C" is incorrect but plausible as this is the reason when starting an RCP is allowed based on SG Downcomer and RCS temperature deltaT. This is a limit and precaution in 1103.006, Reactor Coolant Pump Operation.

"D" is incorrect but plausible as this is an ICS runback setpoint for only having one RCP running in each loop.

This question matches the K/A since it directly asks for the purpose of the RCP power interlock which is to prevent a cold water transient.

References:

1103.006, Reactor Coolant Pump Operation
ANO-1 SAR, Chapter 14

History:

New for 2017 RO Re-exam
Rev. 1, 5/19/17
Added "to start any RCP" to end of A.

QID: 0417 **Rev:** 2 **Rev Date:** 5/19/17 **Source:** Bank **Originator:** S.Pullin

TUOI: A1LP-RO-EOP01 **Objective:** 10 **Point Value:** 1

Section: 3.1 **Type:** Reactivity Control

System Number: 004 **System Title:** Chemical and Volume Control System

Description: Knowledge of the effect of a loss or malfunction on the following CVCS components: Flow paths for emergency boration

K/A Number: K6.17 **CFR Reference:** 41.7 / 45.7

Tier: 2 **RO Imp:** 4.4 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 4.6 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 30 **SRO:**

Given:

* Rx tripped from 100% power
* P-36A OP HPI pump is running
* Three CRDMs indicate 100% withdrawn
* Boric Acid Pump P-39A out of service

Emergency Boration initiated per RT-12.

* Boric Acid Pump P-39B discharge pressure indicates 12 psig
* Boric Acid flow indicates 2 gpm

What operator action is required per RT-12 for these conditions?

A. Open BWST Outlet valve CV-1407.

B. Vent Makup Tank to lower pressure to 10 psig.

C. Raise Batch Controller setting to maximum batch size (999999).

D. Verify Batch Controller Flow Control valve, CV-1249, 100% open.

Answer:

A. Open BWST Outlet valve CV-1407.

Notes:

"A" is correct per RT-12, if Boric Acid pumps or controller are not working (as evidenced by low flow rate on batch controller and low discharge pressure of only Boric Acid Pump), then boration from BWST is initiated.

"B" will lower MUT pressure to less than discharge pressure of Boric Acid Pump P-39B but this is not procedurally directed and thus incorrect.

"C" is incorrect, while this would appear to help, the Batch Controller should already be at maximum batch size prior to starting the Boric Acid Pump P-39B.

"D" is incorrect, this step would have been performed right after starting the Boric Acid Pump P-39B.

Revised question: added P-36A OP HPI pump running to conditions, changed correct answer, formerly said both BWST outlets, changed to only one BWST outlet, this agrees with current revision of 1202.012.

This matches the K/A since the situation requires emergency boration, the makeup and purification system is used for this purpose, and the question involves what action to take if the normal emergency boration flow path is not working.

References:

1202.012, Repetitive Tasks, RT-12 and RT-2

History:

New for 2002 RO/SRO exam.
Selected for 2017 RO Re-exam.
Rev. 2, 5/19/17
Added "per RT-12" to stem.
Re-ordered answers short to long.
Editorial changes.

QID: 0700 **Rev:** 2 **Rev Date:** 5/19/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-ADHR **Objective:** 1 **Point Value:** 1

Section: 3.4 **Type:** RCS Heat Removal from Reactor Core

System Number: 005 **System Title:** Residual Heat Removal System

Description: Knowledge of the effect that a loss or malfunction of the RHRS will have on the following: RCS.

K/A Number: K3.01 **CFR Reference:** 41.7 / 45.6

Tier:	2	**RO Imp:**	3.9	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.0	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 31 **SRO:**

Given:
* Plant in mode 5
* "A" Decay Heat System in service
* RCS intact
* Steam Generators NOT available

A local loss of Instrument Air to the "A" Decay Heat Vault occurs.

What affect would this have on the Decay Heat valves and RCS temperature?

A. Cooler Outlet (CV-1428) stays "as-is"
Cooler Bypass (CV-1433) opens fully
RCS temperature rises

B. Cooler Outlet (CV-1428) opens fully
Cooler Bypass (CV-1433) closes fully
RCS temperature lowers

C. Cooler Outlet (CV-1428) stays "as-is"
Cooler Bypass (CV-1433) closes fully
RCS temperature lowers

D. Cooler Outlet (CV-1428) opens fully
Cooler Bypass (CV-1433) opens fully
RCS temperature rises
.

Answer:

C. Cooler Outlet (CV-1428) stays "as-is"
Cooler Bypass (CV-1433) closes fully
RCS temperature lowers

Notes:

"C" is correct. The Cooler Outlet valve (CV-1428) is an MOV and not be affected by the loss of instrument air. The Cooler Bypass valve (CV-1433) is air operated and will fail closed on loss of instrument air. The increased flow through the Decay Heat cooler will reduce DH outlet temperature and thus RCS temperature will lower.

"A" is incorrect. The Cooler Bypass valve (CV-1433) fails closed, not open. This distractor is plausible since the given RCS parameter effects correspond to increased bypass flow around the cooler, RCS temperature would rise.

"B" is incorrect. The Cooler Outlet valve (CV-1428) is an MOV and will not be affected by a loss of instrument air. However, this distractor is plausible since the Cooler Bypass valve (CV-1433) does fail closed on loss of instrument air. The increased flow through the Decay Heat cooler will reduce DH outlet temperature and thus RCS temperature will lower.

"D" is incorrect. The Cooler Outlet valve (CV-1428) is an MOV and will not be affected by a loss of instrument

air. Also, the Cooler Bypass valve (CV-1433) fails closed, not open. This distractor is plausible since the given RCS parameter effects correspond to increased bypass flow around the cooler, RCS temperature would rise.

Modified this question, previous version featured a trip of the DH pump with effects on RCS pressure, temperature, and Pressurizer level. There were only two possible answers that would correspond to the laws of physics and thus two distractors were not plausible. Revised this question to be "2x2" with a loss of instrument air to the vault with the answer choices being the corresponding effects on valve positions and RCS press/temp.

This matches the K/A since the conditions give a malfunction of the Decay Heat system (loss of air) and requires the applicant to determine the resulting effects on the RCS.

References:

1203.024, Loss of Instrument Air
STM 1-05, Decay Heat System

History:

New for the 2008 RO Exam. Original question had two implausible distractors.
Completely revised for 2017 RO Re-exam, qualifies as New question.
Rev. 2, 5/19/17
Changed "RCS parameters" to "RCS temperature in stem.
Deleted effects on RCS pressure from each answer choice.
Editorial changes.

QID: 1130 **Rev:** 1 **Rev Date:** 5/19/17 **Source:** Modified **Originator:** Cork

TUOI: A1LP-RO-TS **Objective:** 4 **Point Value:** 1

Section: 3.2 **Type:** RCS Inventory Control

System Number: 006 **System Title:** Emergency Core Cooling

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

K/A Number: A1.13 **CFR Reference:** 41.5 / 45.5

Tier: 2 **RO Imp:** 3.5 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 3.7 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 32 **SRO:**

Given:
* 100% power

Core Flood system is properly aligned with following parameters:

T-2A level	13.1 feet	T-2B level	12.5 feet
T-2A pressure	618 psig	T-2B pressure	581 psig

One of above Core Flood Tank (CFT) parameters is unacceptable per Tech Specs because during a LOCA:

A. Level too high, N2 volume insufficient to fully inject CFT contents into vessel.

B. N2 pressure too low, could not fully inject CFT contents into vessel.

C. N2 pressure too high, RCS inventory will be lost out of break.

D. Level too low, thus insufficient to reflood vessel.

Answer:

D. Level too low, thus insufficient to reflood vessel.

Notes:

"D" is correct. B level is less than the Tech Spec limit of 12.6 ft and it takes both CFT to accomplish their design of re-flooding the vessel.

"A" is incorrect. This would be the case if either level were out of spec high, however, the levels are within specs, both less than 13.4 ft. This is plausible since T-2A level is close to the high limit

"B" is incorrect. N2 pressures for both are greater than the 572 psig TS limit from 1104.001. This is plausible since B CFT pressure is close to the admin limit in 1104.001 of 580 psig.

"C" is incorrect, A CFT pressure is less than the Tech Spec limit of 628 psig from 1104.001. This is plausible since A CFT pressure is high but within the admin limit in 1104.001 of 620 psig.

Modified QID 197 by changing all of the CFT parameters, most importantly changing T-2B level to 12.5 ft which is less than the T.S. level. This made "C" the correct answer. Answers were revised to remove vague terms like "may" or "could".

This question matches the K/A as it concerns Core Flood Tank parameters (accumulators) and requires the candidate to know the parameter values that define operability for the CFTs to ensure they are within design limits.

References:

1104.001, Core Flood System Operating Procedure
Technical Specifications, LCO 3.5.1 Bases

History:

Modified QID197 for 2017 RO Re-exam
Rev. 1, 5/19/17
Re-ordered choices long to short to break up string of C's.
Editorial changes, rephrased answers so they are all similar.

QID: 0904 **Rev:** 1 **Rev Date:** 5/19/17 **Source:** New **Originator:** Possage/Cork

TUOI: A1LP-RO-RCS **Objective:** 17 **Point Value:** 1

Section: 3.5 **Type:** Containment Integrity

System Number: 007 **System Title:** Pressurizer Relief Tank/Quench Tank System (PRTS)

Description: Ability to monitor automatic operation of the PRTS, including: Components which discharge to the PRT

K/A Number: A3.01 **CFR Reference:** 41.7 / 45.5

Tier: 2 **RO Imp:** 2.7 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 2.9 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 33 **SRO:**

Given:
* Plant tripped due to secondary transient
* RCS pressure dropped to 1010 psig
* Quench Tank pressure 25 psig
* Reactor Building pressure 15.7 psia

Which of the following Quench Tank temperatures would indicate that the ERV is lifting?

A. 215 °F

B. 239 °F

C. 267 °F

D. 547 °F

Answer:

C. 267 °F

Notes:

"C" is correct, this is saturation temperature for Quench Tank pressure of 40 psia (25 psig). The ERV will be relieving to the Quench Tank which is at ambient, but above normal, temperature. The indicated pressure for the Quench Tank is above the normal nitrogen cover gas pressure of ~3 psig. The relief piping within the Quench Tank is under a water level, so the steam from the ERV will pass through this water, cooling the steam. Therefore, one cannot use the Mollier diagram to derive the correct temperature using an isenthalpic process since the steam condition in the Quench Tank will not be superheated but saturated.

"A" is incorrect, this is saturation temperature for the Reactor Building which is at 15.7 psia. This is above normal but is now ambient temperature for the RB which is elevated due to a coolant leak, or a secondary steam leak. This is plausible but only if the ERV had been lifting continuously and this lifting broke the rupture disc of the Quench Tank, causing the Quench Tank to be at RB pressure but the Quench Tank pressure is given at 25 psig.

"B" is incorrect, this is saturation temperature for 25 psia which is a plausible error an applicant could make.

"D" is incorrect, this is saturation temperature for RCS pressure of 1010 psig but the steam conditions in the Quench Tank will not be the same as the RCS.

This question matches the K/A since to monitor automatic operation of the PRTS (Quench Tank), and the components which discharge to it, one has to deduce the expected temperature if a relief valve was lifting.

References:

Steam Tables
STM 1-03, Reactor Coolant System; Figure 3.21 Reactor Coolant Quench Tank

History:

written for 2014 but not used
New question for 2017 RO Re-exam
Rev. 1, 5/19/17
Editorial changes.

QID: 1132 **Rev:** 1 **Rev Date:** 5/19/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-MSSS **Objective:** 9 **Point Value:** 1

Section: 3.8 **Type:** Plant Service Systems

System Number: 008 **System Title:** Component Cooling Water

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: High/low surge tank level.

K/A Number: A2.02 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

Tier:	2	**RO Imp:**	3.2	**RO Select:**	Yes	**Difficulty:**	4
Group:	1	**SRO Imp:**	3.5	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 34 **SRO:**

Given:
* 100% power
* RCS leakage has increased
* Inside AO dispatched to ICW Surge Tanks
* Inside AO reports ICW Surge Tank T-37B level changed 1.4 to 2.1 psid in five minutes

* RCP Seal Injection flows
  A 9 gpm
  B 8 gpm
  C 7 gpm
  D 8 gpm

* RCP seal bleedoff temperatures are stable

* Letdown Cooler E29A Outlet temperature rising

Which of the following actions are required by Excess RCS Leakage (1203.039) for above conditions?

A. Open ICW Surge Tank Crossconnect Isolation (ICW-165)

B. Trip reactor, perform Reactor Trip (1202.001) in conjunction with 1203.039

C. Isolate Letdown Cooler E29A by closing inlet (CV-1213) and outlet (CV-1214)

D. Close Nuclear ICW RB Inlet (CV-2233) and both Nuclear ICW RB Outlets (CV-2214 and CV-2215)

Answer:

C. Isolate Letdown Cooler E29A by closing inlet (CV-1213) and outlet (CV-1214)

Notes:

"C" is correct. RCS is leaking into ICW system as seen by Nuclear ICW Surge Tank T-37B level risiing. Letdown Cooler E29A outlet temperature is rising, indicating a leak into this cooler. 1203.039 isolates E29A letdown cooler to an attempt to stop the leak.

"A" is incorrect but plausible since the purpose of the cross-connect is to allow the Surge Tanks to overflow from one to the other in case of a leak. However, 1203.039 directs closing, not opening, the cross-connect valve.

"B" is incorrect but plausible since 1203.039 does have a step which states to trip the Reactor if HPI is needed to maintain PZR inventory. However, leak rate via ICW surge tank is 46 gpm, this is less than the 50 gpm threshold normally used to make this determination. ICW surge tank contains 1000 gallons and DP indication is 0.5 to 2.7 psid and 1.0 psid is 333 gallons. An increase of 0.7 psid over five minutes means a leak rate of 46.6 gpm. 0.7 x 333 = 233 gallons/5 min = 46.6 gpm

"D" is incorrect but plausible since this action is taken if there is a leak into ICW from an RCP seal cooler.

One indication of this is skewed Seal Injection flows. The Seal Injection flows given do vary but they are not significantly skewed.

This question matches the K/A since it involves high CCW (ICW at ANO-1) surge tank level due to RCS leakage into the ICW system, predicting the impact on operations, and taking procedural action to mitigate the leakage. The applicant is required to calculate RCS leakage based on the rise in surge tank level (thus predicting the impact of this malfunction), determine approximate RCS leak rate to eliminate one course of action (tripping the reactor), diagnose the conditions to determine if it is a leak into the letdown cooler or into the RCP seal cooler, and then arrive at the proper action as stated in the abnormal operating procedure.

References:

1203.039, Excess RCS Leakage

History:

New for 2017 RO Re-exam
Rev.1, 5/19/17
Re-ordered answers short to long.
Editorial changes

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0562 **Rev:** 1 **Rev Date:** 5/19/17 **Source:** Bank **Originator:** J.Cork

TUOI: A1LP-RO-MSSS **Objective:** 9 **Point Value:** 1

Section: 3.8 **Type:** Plant Service Systems

System Number: 008 **System Title:** Component Cooling Water System (CCWS)

Description: Knowledge of the bus power supplies to the following: CCW pump, including emergency

K/A Number: K2.02 **CFR Reference:** 41.7

Tier:	2	**RO Imp:**	3.0	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	3.2	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 35 **SRO:**

Which of the following identifies the correct power supplies to Intermediate Cooling Water Pumps (P-33A, P-33B, P-33C)?

A. P33A from B-12;
P33B and P33C from B-22

B. P33A and P33B from B-12:
P33C from B-22

C. P33A, P33B and P33C from B-11, B-12 and B-13 respectively

D. P33A, P33B and P33C from B-12, B-22 and B-32 respectively.

Answer:

A. P33A from B-12;
P33B and P33C from B-22

Notes:

"A" lists the correct power supplies for the ICW pumps.

"B" is incorrect since P33B is not powered from B-12 but plausible since the other power supplies are correct.

"C" is incorrect since P33A and P33C power supplies are incorrect but the power supply for the B pump is correct and they are presented in a logical order to enhance plausiblity.

"D" is incorrect since P33C is not powered from B-32 but plausible since the other power supplies are correct.

This question matches the K/A since it requires recall of the power supplies for the ICW (CCW) pumps.

Re-ordered distractors since this is a bank question. Swapped the order of A and B and also swapped the order of C and D.

References:

STM 1-43, Intermediate Cooling Water

History:

Direct from regular exam bank QID#4674
Selected for 2005 RO exam
Selected for 2011 RO exam.
Selected for 2017 RO Re-exam
Rev.1, 5/19/17
Editorial changes.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1149 **Rev:** 1 **Rev Date:** 5/19/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-NOP **Objective:** 4 **Point Value:** 1

Section: 3.3 **Type:** Reactor Pressure Control

System Number: 010 **System Title:** Pressurizer Pressure Control

Description: Ability to monitor automatic operation of the PZR PCS, including: PZR pressure.

K/A Number: A3.02 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

Tier:	2	**RO Imp:**	3.6	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	3.5	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 36 **SRO:**

Given:
* 100% power
* ATC observes RCS pressure rising

Assuming no malfunctions, what is the pressure setpoint at which ATC will observe Pressurizer Spray valve automatically OPEN?

A. 2255 psig

B. 2205 psig

C. 2155 psig

D. 2080 psig

Answer:

B. 2205 psig

Notes:

"B" is correct, the Pressurizer Pressure controller sends a signal to open the Spray valve (CV-1008) at 2205 psig.

"A" is incorrect but plausible, this is the setpoint for the high RCS pressure alarm.

"C" is incorrect but plausible since the Pressurizer Spray valve closes at this pressure.

"D" is incorrect. This is plausible since the Pressurizer Spray valve opens at this pressure but only if it opened due to a MFW pump trip when greater than 80% power. .

This question matches the K/A since the applicant must know when the presurizer spray valve normally opens to be able to monitor automatic operation.

References:

1103.005, Pressurizer Operation

History:

New for 2017 RO Re-exam
Rev.1, 5/19/17
Changed A distractor to high RCS pressure alarm setpoint.
Editorial changes.

QID: 0307 **Rev:** 2 **Rev Date:** 5/19/17 **Source:** Bank **Originator:** J. Cork

TUOI: A1LP-RO-RPS **Objective:** 11 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 012 **System Title:** Reactor Protection System

Description: Knowledge of the effect of a loss or malfunction of the following will have on the RPS: Trip logic circuits

K/A Number: K6.03 **CFR Reference:** 41.7 / 45.7

Tier:	2	**RO Imp:**	3.1	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	3.5	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 37 **SRO:**

Given:

* 100% power
* "B" Reactor Protection System channel declared inoperable and cannot be repaired until next month
* Surveillance test on "D" RPS channel in progress

What is the RPS bistable trip logic under these conditions?

A. Two out-of-two

B. Two out-of-four

C. One out-of-two

D. One out-of-three

Answer:

C. One out-of-two

Notes:

"C" is correct. The reactor protection system logic is normally 2 of 4 required to trip the reactor. The "D" RPS channel is placed in bypass to perform the surveillance test, therefore it cannot trip. Since only one channel can be in bypass at a time, the "B" RPS channel will be tripped due to being inoperable to comply with Tech Specs. The trip logic will therefore be 1 out of 2.

"A" is incorrect but plausible if the applicant does not consider that the "B" channel is already tripped.

"B" is incorrect but plausible if the applicant merely recalls the normal trip logic and does not apply any of the conditions listed.

"D" is incorrect but plausible if the applicant does not consider that the "D" channel is in bypass during a surveillance and can not trip.

Re-ordered answer choices. Revised thrid bullet due to validator comment, used to say "becomes inoperable due to NI-6 failing high", now says simply "declared inoperable yesterday, cannot be repaired until next month". Changed order of bullets so inoperable channel is before surveillance test.

This question matches the K/A since it concerns the RPS and the affect a malfunction has upon the trip logic.

References:

STM-63, Reactor Protection System

History:

Developed for 1999 exam.
Selected for use in 2005 RO exam as replacement.

Selected for 2017 RO Re-exam
Rev. 2, 5/19/17
Changed LOD to 2.
Editorial changes.

QID: 1131 **Rev:** 1 **Rev Date:** 5/19/17 **Source:** Modified **Originator:** Cork

TUOI: A1LP-RO-ESAS **Objective:** 6 **Point Value:** 1

Section: 3.2 **Type:** Reactor Coolant System Inventory Control

System Number: 013 **System Title:** Engineered Safety Features Actuation System

Description: Knowledge of less than or equal to one hour Technical Specification action statements for systems.

K/A Number: 2.2.39 **CFR Reference:** 41.7 / 41.10 / 43.2 / 45.13

Tier: 2 **RO Imp:** 3.9 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 4.5 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 38 **SRO:**

Given:

* 100% power

* ESAS Analog channel 2 RCS pressure transmitter fails HIGH

What action below will be in compliance with Technical Specifications for the above conditions?

A. Place analog instrument channel in trip within one hour.

B. Immediately initiate action to prepare and submit a Special Report.

C. Place associated components in their ES positions within one hour.

D. Initiate plant shutdown within one hour to be in Mode 3 within 7 hours.

Answer:

A. Place analog instrument channel in trip within one hour.

Notes:

"A" is correct per Tech Spec 3.3.5. If the RCS pressure transmitter fails high, then the low RCS pressure parameter in Table 3.3.5-1 cannot perform it's function (will not trip) so the associated analog channel must be tripped within one hour per 3.3.5 required action A.1.

"B" is incorrect, yet plausible since this would be correct if the RCS Wide Range pressure function were inoperable (and not repaired within 30 days) per TS 3.3.15, PAM instrumentation, but the spec requires two channels and two are available from Analog channels 1 and 2.

"C" is incorrect, yet plausible as this is one hour required action A.1 from 3.3.7 for an inoperable digital ESAS channel.

"D" is incorrect, yet plausible since 3.0.3 requires this for more than one inoperable analog channel, but only one channel is inoperable.

Modified QID 350 by making it RO level, changed stem to be a one hour action statement. Changed the RCS pressure instrument from failing low to failing high so "A" is no longer correct, modified "B" so that it is more like 3.3.5 required action B.1, changed "C" so that it is required action A.1 for 3.3.5 (now the correct answer), and revised "D" so it is like required action A.1 for LCO 3.3.7 for a digital channel.

This matches the K/A since it is about a failed ESAS (ESFAS) channel requiring entry into a one hour Tech Spec action statement.

References:

Technical Specifications 3.3.5, 3.3.15, 3.0.3

History:

Modified QID 350 for 2017 RO Re-exam.
Rev. 1, 5/19/17
Re-ordered answer choices from short to long.
Replaced distractor B (previous position was A) with action from TS 3.3.15.
Added "within one hour" to D distractor so it falls within the one hour recall credibility for an RO. Changed 6 hours to 7 hours in D distractor so it reads like TS 3.0.3.
Added "within one hour" to A and C so answers would be consistent with RO expectations for TS recall.
In D explanation replaced TS reference with 3.0.3.
Editorial changes.

QID: 1153 **Rev:** 1 **Rev Date:** 5/19/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-EOP02 **Objective:** 14 **Point Value:** 1

Section: 3.2 **Type:** RCS Inventory Control

System Number: 013 **System Title:** Engineered Safety Features Actuation

Description: Ability to manually operate and/or monitor in the control room: ESFAS-initiated equipment which fails to actuate.

K/A Number: A4.01 **CFR Reference:** 41.7 / 45.5 to 45.8

Tier: 2 **RO Imp:** 4.5 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 4.8 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 39 **SRO:**

Given:
* Unit 1 tripped from 100% power
* ESAS channels 1 though 8 have actuated

CBOT performing RT-10, Verify Proper ESAS Actuation, and reports BWST Outlet valve, CV-1408, failed to open.

What action is required per RT-10?

A. Obtain CRS permission, override ES, open BWST Outlet valve (CV-1408).

B. Override ES, open BWST Outlet valve (CV-1408) ONLY.

C. Obtain CRS permission, override ES, stop HPI pump (P36B) and LPI pump (P34B).

D. Override ES, stop HPI pump (P36B) and LPI pump (P34B) ONLY.

Answer:

C. Obtain CRS permission, override ES, stop HPI pump (P36B) and LPI pump (P34B).

Notes:

"C" is correct. RT-10 requires obtaining permission to override ES, then stop HPI and LPI pumps. Permission is required due to taking the pumps to stop which is not their ES position. The pumps are stopped due to CV-1408 has a relatively long stroke time and the pumps should be stopped ASAP to avoid any damage due to loss of suction. A previous step said to verify CV-1408 open and thus the operator should have taken the handswitch to open to attempt to open the valve.

"A" is incorrect, plausible since open is the desired position but there is no requirement to obtain CRS permission to place a component in its' required ES position. Since open is the ES position, there is no reason to override ES. For CV-1407 and CV-1408 one cannot take the valves to close since the ES signal opens series contacts preventing the handswitch from taking the valve to close but the contacts from ES in the open circuit are in parallel to the handswitch so there is nothing to prevent the handswitch from working.

"B" is incorrect, plausible since open is the desired position but again, there is no reason to override ES.

"D" is incorrect, this is plausible since this is the correct action but CRS permission must be obtained prior to placing any component in any position other than it's required ES position.

This question matches the K/A since it involves an ESFAS component which has failed to operate and the actions which must be taken in response to this failure.

References:

1202.012, Repetitive Tasks

History:

New question for 2017 RO Re-exam

Rev. 1, 5/19/17
Added "ONLY" to end of distractors B and D.
Editorial changes.
Did not change answer order due to answer distribution.

QID: 0256 **Rev:** 2 **Rev Date:** 5/19/17 **Source:** Bank **Originator:** D. Slusher

TUOI: A1LP-RO-MSSS **Objective:** 3 **Point Value:** 1

Section: 3.5 **Type:** Containment Integrity

System Number: 022 **System Title:** Containment Cooling System

Description: Knowledge of the physical connections and/or cause-effect relationships between the CCS and the following systems: SWS/cooling system.

K/A Number: K1.01 **CFR Reference:** 41.2 to 41.9 / 45.7 to 45.8

Tier: 2 **RO Imp:** 3.5 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 3.7 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 40 **SRO:**

Why are Decay Heat Cooler Outlet Valves SW-22A and SW-22B throttled during normal operation?

A. Maintain adequate service water flow to Reactor Building Coolers when ES actuates.

B. Raise Service water flow to Auxiliary Cooling Water System during normal operation.

C. Reduce reactor coolant to service water differential temperature when ES actuates.

D. Maintain Decay heat coolers full and reduce chance of water hammer.

Answer:

A. Maintain adequate service water flow to Reactor Building Coolers when ES actuates.

Notes:

"A" is correct, SW flow to the DH coolers are throttled via SW-22A/B because flow to ES components (specifically the RB coolers) may not be adequate. The valves are marked so that they will not be throttled below the minimum required for the DH coolers. The RB coolers are normally cooled by chilled water and transfer to Service Water upon ESAS actuation (this is the cause-effect relationship between the two systems).

"B" is incorrect, while it is true that ACW demand will be higher during normal operation, this is not the reason for throttling the DH cooler outlets.

"C" is incorrect because while throttling the valves will reduce SW flow to the coolers and thus raise DH outlet temperature, it is not reduce the service water to RCS differential temperature.

"D" is incorrect because while water hammer of service water piping is a concern with SW to the RB coolers, that is not the reason for throttling these valves.

Changed order of answers.

This question matches the K/A since the question requires knowledge of the reason for throttling Service Water (SWS) to the DH coolers, i.e., to ensure adequate flow to the RB coolers (CCS) during an ESAS actuation.

References:

STM 1-09, Reactor Building Ventilation
1309.013, Unit 1 Service Water Flow Test

History:

Used in 1999 exam.
Modified from ExamBank, QID# 1519.
Selected for 2005 RO re-exam. KA 022 A1.04

Selected for 2011 RO Exam.
Seleted for 2017 RO Re-exam
Rev. 1, 5/19/17
Editorial changes.

QID: 0705 **Rev:** 1 **Rev Date:** 4/8/17 **Source:** Bank **Originator:** Pullin

TUOI: A1LP-RO-RBS **Objective:** 4 **Point Value:** 1

Section: 3.5 **Type:** Containment Integrity

System Number: 026 **System Title:** Containment Spray System (CSS)

Description: Knowledge of power supplies to the following: MOVs

K/A Number: K2.02 **CFR Reference:** 41.7

Tier:	2	**RO Imp:**	2.7	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	2.9	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 41 **SRO:**

Which of the following supply power to Containment Spray Motor Operated Valves
CV-2400 (B RB Spray Block) and CV-2401 (A RB Spray Block)?

A. B6371 and B5371

B. B6171 and B5171

C. B5271 and B6271

D. B6571 and B5571

Answer:

B. B6171 and B5171

Notes:

B is correct as these are the correct power supplies for the spray block valves.

A is incorrect but plausible since these are two other ES powered MCCs.

C is incorrect but plausible since these are two other ES powered MCCs.

D is incorrect but plausible since these are two other ES powered MCCs.

Revised "A" and "C" distractors since the MCCs were non-ES and thus not plausible.

This matches the K/A because it asks what the power supplies are for Containment Spray MOVs.

References:

1107.002, ES Electrical System Operation

History:

New for the 2008 RO Exam.
Selected for 2017 RO Re-exam

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1155 **Rev:** 1 **Rev Date:** 5/19/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-STEAM **Objective:** 12 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal from Reactor Core

System Number: 039 **System Title:** Main and Reheat Steam

Description: Knowledge of the operational implications of the following concepts as they apply to the MRSS: Effect of steam removal on reactivity

K/A Number: K5.08 **CFR Reference:** 41.5 / 45.7

Tier: 2 **RO Imp:** 3.6 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 3.6 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 42 **SRO:**

Given:
* 100% power
* Main Steam leak occurs

Control rooom operators will initially observe a Reactor power ______ and Main Generator MW __________.

A. rise; stay the same

B. drop; drop

C. rise; drop

D. drop; stay the same

Answer:

C. rise; drop

Notes:

"C" is correct. With reactor power at 100%, the increase in steam flow will cause primary temperature to decrease. With a negative moderator temperature coefficient, the drop in primary temperature will result in increased moderator density, and a reactor power increase. Main Turbine control system receives pulses from the ICS to control header pressure at setpoint, the steam leak will cause header pressure to drop, the control system will then close the Governor Valves to raise header pressure, and thus Main Generator Megawatts will drop.

"A" and "D" distractors are plausible if the applicant incorrectly believes the Main Turbine control system will function like many control systems and will compensate for the steam leak by opening the governor valves to maintain Main Generator output.

"A" distractor is also plausible since reactor power has the correct trend.

"B" distractor is plausible since the main generator output trend is correct.

This question matches the K/A since it tests the applicant's knowledge of the operational effects of steam removal, including reactor power.

References:

STM 1-65, Integrated Control System

History:

New question for 2017 RO re-exam
Rev.1, 5/19/17
Revised stem to be a fill in the blank with two parts, revised answer choices accordingly.
Editorial changes.

QID: 0656 **Rev:** 0 **Rev Date:** 12/12/06 **Source:** Bank **Originator:** Possage

TUOI: A1LP-RO-EFIC **Objective:** 4 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal from Reactor Core

System Number: 039 **System Title:** Main and Reheat Steam System

Description: Knowledge of MRSS design feature(s) and/or interlock(s) which provide for the following: Automatic isolation of steam line.

K/A Number: K4.05 **CFR Reference:** 41.7

Tier: 2 **RO Imp:** 3.7 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 3.7 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 43 **SRO:**

Unless bypassed, the setpoint for Main Steam Line Isolation (MSLI) actuation is at a decreasing SG pressure of:

A. 700 psig

B. 650 psig

C. 600 psig

D. 550 psig

Answer:

C. 600 psig

Notes:

"C" is correct MSLI and EFW actuate at 600 psig decreasing SG pressure.

All of the other choices are simply bracketed around the correct setpoint in 50 psig increments.

This matches the K/A as it requires knowledge of the setpoint for main steam line isolation.

References:

1105.005, Emergency Feedwater Initiation and Control

History:

Selected for 2007 RO Exam. Direct from regular exambank QID# ANO-OPS1-779.
Selected for 2017 RO Re-exam

QID: 0911 **Rev:** 2 **Rev Date:** 5/19/17 **Source:** New **Originator:** Possage/Cork

TUOI: A1LP-RO-EOP01 **Objective:** 10 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal from Reactor Core

System Number: 059 **System Title:** Main Feedwater System

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

K/A Number: A2.03 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

Tier: 2 **RO Imp:** 2.7 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 3.1 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 44 **SRO:**

Given:

* Reactor tripped from 100% due to Turbine Trip

* Immediate Actions are complete

* ATC reports ‘A’ S/G level is greater than 410 inches

In accordance with Reactor Trip (1202.001) what action is required and why?

A. Trip ‘A’ MFWP; mitigate overcooling event

B. Trip Both MFWPs; mitigate overcooling event

C. Trip ‘A’ MFWP; prevent feedwater carryover into Main Steam Line

D. Trip Both MFWPs; prevent feedwater carryover into Main Steam Line

Answer:

D. Trip Both MFWPs; prevent feedwater carryover into Main Steam Line

Notes:

"D" is correct per 1202.001 step 11 and the corresponding EOP bases document.

"A" is incorrect but plausible as this action is taken in the Overcooling EOP for overfeed events but if SG level is approaching the high limit both pumps are tripped to prevent putting water in the steam lines.

"B" is incorrect but plausible as this is the correct action but the wrong reason.

"C" is incorrect but plausible as this is the correct reason but both pumps should be tripped.

This question matches the K/A since the conditions describe an overfeeding event and requires the ability to predict what is happening and the knowledge of the proper procedural actions to mitigate the event.

References:

1202.001, Reactor Trip
1202.001 Bases Document

History:

New for 2014 Exam - NOT used, never used in an open bank.
New for 2017 RO Re-exam, modified A & B distractors. Rev. 1 Cork 3/7/17
Rev. 2, 5/19/17
Revised 3rd bullet so that A SG level is >410".
Re-arranged stem.

Editorial changes.

QID: 0709 **Rev:** 3 **Rev Date:** 5/19/17 **Source:** Bank **Originator:** Thompson

TUOI: A1LP-RO-EFIC **Objective:** 29 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal From Reactor Core

System Number: 061 **System Title:** Auxiliary / Emergency Feedwater System

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the AFW controls including: AFW flow/motor amps.

K/A Number: A1.05 **CFR Reference:** 41.5 / 45.5

Tier: 2 **RO Imp:** 3.6 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 3.7 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 45 **SRO:**

Given:

* Reactor tripped
* RCPs stopped
* SCM adequate

"A" SG pressure 925 psig
"A" SG level 200" and rising

"B" SG pressure 845 psig
"B" SG level 215" and rising

Per Emergency Feedwater Initiation and Control (1105.005) what are SG fill rates for the above conditions?

A. A at 3"/minute,
B at 5"/minute

B. A at 5"/minute,
B at 3"/minute

C. A at 4"/minute,
B at 6"/minute

D. A at 6"/minute,
B at 4"/minute

Answer:

B. A at 5"/minute,
B at 3"/minute

Notes:

"B" has the correct fill rates based on Table 1 in RT-5 and knowledge of the proper operation of the fill rate controller. The controller automatically fills the SGs on a linear scale from 2" to 8" based on SG pressure. As SG pressure rises from 800 psig to 1050 psig the fill rate rises linearly to prevent overfill and overcooling. Therefore with A OTSG pressure at 925 psig the fill rate would be 925 - 800 = 125 x .024"/psig = 3 + 2 = 5"/minute. B OTSG fill rate is calculated the same way: 845 - 800 = 45 x 0.024"/psig = 1 + 2 = 3"/minute. The 0.024"/psig is derived by: 1050 - 800 = 250, 8" - 2" = 6", 6"/250 = 0.024"/psig

"A" is incorrect yet plausible as these are the correct fill rates but for the wrong SGs.

"C" is incorrect yet plausible if the candidate believed the given fill rates are correct.

D are incorrect yet plausible if the candidate believed the given fill rates should be swapped.

Revised question by changing B SG pressure from 750 psig to 845 psig, changed all fill rates in answer choices. Revised question to be in compliance with latest format expectations.

This question matches the K/A since it requires candidate to evaluate given condition and monitor/predict the proper EFW flow rates for those conditions.

References:

1105.005, Emergency Feedwater Initiation and Control
1202.012, Repetitive Tasks, RT-5

History:

New for 2008 RO Exam.
Selected for 2017 RO Re-exam.
Rev. 3, 5/19/17
Revised stem so answer is in accordance with 1105.005 and it asks for fill rates.
Deleted EFW feed rates from given parameters.
Editorial changes.

QID: 1133 **Rev:** 1 **Rev Date:** 5/19/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-ELECD **Objective:** 14g **Point Value:** 1

Section: 3.6 **Type:** Electrical

System Number: 062 **System Title:** AC Electrical Distribution

Description: Knowledge of the effect that a loss or malfunction of the ac distribution system will have on the following: DC system.

K/A Number: K3.03 **CFR Reference:** 41.7 / 45.6

Tier:	2	**RO Imp:**	3.7	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	3.9	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 46 **SRO:**

If Unit 1 experiences a Blackout, what is the minimum time the 125 VDC Vital Batteries (D06 and D07) are designed to supply power to their loads, and what major loads will be secured to extend battery capacity?

A. 1 hour
Main Turbine Emergency Lube Oil Pump (P20)

B. 2 hours
Main Feed Pump Emergency Oil Pumps (P28A & P28B)

C. 1 hour
Main Feed Pump Emergency Oil Pumps (P28A & P28B)

D. 2 hours
Main Turbine Emergency Lube Oil Pump (P20)

Answer:

B. 2 hours
Main Feed Pump Emergency Oil Pumps (P28A & P28B)

Notes:

"B" is the correct answer per the SAR section on the 125 Volt DC System and step 41 of the Blackout EOP. The batteries will last a minimum of two hours and the DC powered oil pumps on the Main FW Pumps are secured to extend battery life.

"A" is incorrect, plausible since the time given is close to the actual time and plausible since the name of the Main Turbine Emergency Lube Oil Pump is similar to the MFW pump emergency lube oil pumps but the P20 pump is powered from B56, not DC. The one hour time is plausible since that is the same as the FLEX implementation time.

"C" is incorrect, yet plausible since it has the correct loads to secure but has the incorrect time.

"D" is incorrect, yet plausible as this is the correct time but the incorrect load.

This question matches the K/A since it requires knowledge of DC design (how long the batteries will last and what are DC loads) with respect to a loss of the AC distribution system.

References:

1202.008, Blackout
SAR, 8.3.2

History:

New question for 2017 RO Re-exam
Rev. 1, 5/19/17
Addition of FLEX as supporting plausibility of one hour in distractors.

QID: 0336 **Rev:** 2 **Rev Date:** 5/19/17 **Source:** Bank **Originator:** Cork

TUOI: A1LP-RO-AOP **Objective:** 3 **Point Value:** 1

Section: 3.6 **Type:** Electrical

System Number: 063 **System Title:** DC Electrical Distribution

Description: Knowledge of DC electrical system design feature(s) and/or interlock(s) which provide for the following: Manual/automatic transfers of control.

K/A Number: K4.01 **CFR Reference:** 41.7

Tier: 2 **RO Imp:** 2.7 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 3.0 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 47 **SRO:**

Given:

Plant operating at 100% power when the following occurs:

* TURBINE LOCKOUT RELAY DC FAILURE (K04-B5) alarms
* D01 UNDERVOLTAGE (K01-A7) alarms
* D01 TROUBLE (K01-D7) alarms
* Loss of breaker position indicator lights on left side of C10

Which action, with the correct reason for the action, is required to be performed?

A. Start #1 Diesel Generator from C-10 due to loss of power to undervoltage relays.

B. Transfer D11 to its Emergency Power Supply to energize generator lockout relays.

C. Trip Generator Output Breakers to prevent Main Generator from motoring.

D. Line up Battery Charger D03A or D03B to D01 Bus to restore DC power.

Answer:

B. Transfer D11 to its Emergency Power Supply to energize generator lockout relays.

Notes:

"B" is correct, this action is the use of a design feature: manual transfer of power for 125VDC panel D11 from it's normal power supply of D01 (which has been lost) to it's Emergency Supply (D02). This is the most expedient method of restoring power to D11 and is the action prescribed in 1203.036 for Loss of D01. This action cross-connects both ES trains of DC and is only taken due to the rather significant consequences of losing D01 combined with a loss of generator lockout relays, control power to EDG and ES bus A3, and inadvertent ESAS actuation. Additionally, this action is only taken when D11 is shown NOT to be faulted as evidenced by the presence of the D01 undervoltage alarm so as not to transfer a fault from one train to the other.

"A" is incorrect, although plausible since control power will be lost to #1 EDG and this could be construed as a loss of power to the DG undervoltage relays.

"C" is incorrect, the output breakers may or may not trip on a loss of D01, but plausible since this action is taken within 1203.036 if transfer of D11 to D02 is Unsuccessful.

"D" is incorrect, this is plausible since it might restore power to D01 but it would take too much time and may not work if all of "A" train power is lost.

Swapped answer choices A and B.

This question matches the K/A since it requires knowledge of a DC design feature: manual transfer of DC control power from one train to the other.

References:

1203.036, Loss of 125V DC

History:

Used in 1999 exam. Direct from ExamBank, QID# 1891
Selected for 2005 RO exam, but not used.
Selected for 2007 RO exam.
Selected for 2017 RO Re-exam
Rev. 2, 5/19/17
Changed LOK to H and LOD to 3.
Editorial changes

QID: 0712 **Rev:** 2 **Rev Date:** 5/19/17 **Source:** Bank **Originator:** Steve Pullin

TUOI: A1LP-RO-TS **Objective:** 5 **Point Value:** 1

Section: 2 **Type:** Generic Knowledge and Abilities

System Number: 063 **System Title:** DC Electrical Distribution

Description: Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

K/A Number: 2.2.42 **CFR Reference:** 41.7 / 41.10 / 43.2 / 43.3 / 45.3

Tier: 2 **RO Imp:** 3.9 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 4.6 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 48 **SRO:**

With Unit 1 at 100% power, which of the following conditions requires entry into Technical Specification 3.8.4, DC Systems - Operating?

A. D03A, Battery Charger inoperable and
D04A, Battery Charger inoperable.

B. D03B, Battery Charger inoperable and
D04B, Battery Charger inoperable.

C. D03A, Battery Charger inoperable and
D03B, Battery Charger inoperable.

D. D03B, Battery Charger inoperable and
D04A, Battery Charger inoperable.

Answer:

C. D03A, Battery Charger inoperable and
D03B, Battery Charger inoperable.

Notes:

"C" is correct. Two chargers inoperable on the same train render the train inoperable and require TS entry.

"A" is incorrect. One battery charger inoperable on each train does not require TS entry. Plausible if candidate believes both chargers are on the same train or if any charger being inop requires TS entry per given train.

"B" is incorrect. One battery charger inoperable on each train does not require TS entry. Plausible if candidate believes both chargers are on the same train or if any charger being inop requires TS entry per given train.

"D" is incorrect. One battery charger inoperable on each train does not require TS entry. Plausible if candidate believes both chargers are on the same train or if any charger being inop requires TS entry per given train.

This question matches the K/A since the candidate must know what parts make up a DC electrical system and what would be an entry condition into Tech Specs.

Re-worded distractors A and D to make them more plausible.

References:

Technical Specification 3.8.4
STM1-32, Electrical Distribution

History:

New for 2008 RO Exam
Selected for 2017 RO Re-exam
Rev.2, 5/19/17
Added STM 1-32 to references.

QID: 1150 **Rev:** 1 **Rev Date:** 5/19/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-EOP07 **Objective:** 10 **Point Value:** 1

Section: 3.6 **Type:** Electrical

System Number: 064 **System Title:** Emergency Diesel Generator

Description: Ability to manually operate and/or monitor in the control room: Synchroscope.

K/A Number: A4.03 **CFR Reference:** 41.7 / 45.5 to 45.8

Tier: 2 **RO Imp:** 3.2 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 3.3 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 49 **SRO:**

Given:
* DG2 running for monthly test per 1104.036, Emergency Diesel Generator Operation
* CBOT ready to parallel DG2 to grid

In accordance with 1104.036, Supplement 2, the voltage regulator (Incoming) should be adjusted to match or be no greater than ________(1)___________volts of A4 (Running) voltage.
AND
the governor control should be adjusted until frequency is 60 Hz with the synchroscope rotating _________(2)_____________ direction.

A. (1) 50
(2) slowly in FAST

B. (1) 50
(2) fast in SLOW

C. (1) 20
(2) slowly in FAST

D. (1) 20
(2) fast in SLOW

Answer:

C. (1) 20
(2) slowly in FAST

Notes:

"C" is correct per steps 2.15.6 and 2.15.7 of 1104.036 Supplement 2 which states that DG2 voltage must match or be no greater than running voltage and the synchroscope should be rotating slowly in the fast direction.

"A" is incorrect, plausible since part (2) is correct but the part (1) voltage direction does not match the direction in 1104.036. 100 volts is the smallest increment on the panel gauges so half of that is the best accuracy obtainable from these gauges hence the use of 50 in this distractor and in B. Operators therefore use the plant computer for greater accuracy to ensure procedure compliance.

"B" is incorrect, plausible since part (1) is similar to the voltage direction in 1104.036 but the limit of 50 is too high. Part (2) is the opposite of the correct synchroscope requirement.

"D" is incorrect, plausible since the part (1) voltage requirement is correct per 1104.036 but part (2) is the opposite of the correct synchroscope requirement.

This question matches the K/A since the question requires the applicant to know how to monitor for proper synchroscope operation of an Emergency Diesel Generator.

References:

1104.036, Emergency Diesel Generator Operation

History:

New for 2017 RO Re-exam
Rev. 1, 5/19/17
Editorial changes.

QID: 0136 **Rev:** 1 **Rev Date:** 5/19/17 **Source:** Bank **Originator:** M. Cooper

TUOI: ANO-1-LP-RO-RMS **Objective:** 2 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 073 **System Title:** Process Radiation Monitoring System

Description: Knowledge of the physical connections and/or cause-effect relationships between the PRM system and the following systems: Those systems served by PRMs.

K/A Number: K1.01 **CFR Reference:** 41.2 to 41.9 / 45.7 to 45.8

Tier:	2	**RO Imp:**	3.6	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.9	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 50 **SRO:**

Given:

* LOCA inside Reactor Building
* RE-3814, Service Water Loop I Radiation Monitor alarms
* SW Loop II indications are normal.

Which of the conditions below, when combined with above conditions, would make it necessary to isolate "A" & "B" RB Emergency Coolers?

A. RE-8020, RB area monitor in alarm

B. RE-3618, Discharge Flume monitor in alarm

C. RE-3815, Loop II Service Water monitor in alarm

D. RE-7471, RB ATMOS Gaseous Detector in alarm

Answer:

B. RE-3618, Discharge Flume monitor in alarm

Notes:

"B" is the correct answer. Since Loop II is OK, then the confirmation of an actual release via the Discharge Flume monitor necessitates the isolation of Loop I.

"C" would not corroborate a problem with Loop I coolers and indicate a need to isolate the Loop I coolers.

"D" would be expected for a LOCA condition, but does not necessarily impact the SW side of the RB coolers.

"A" is incorrect but plausible since this does seem to confirm a problem with the RB coolers in the Reactor Building but the RB area monitors would be in alarm anyway due to the LOCA.

References:

1203.012I, Annunciator K10 Corrective Action

History:

Taken from Exam Bank QID # 2571
Used in 98 RO Re-exam
Used on 2004 RO/SRO Exam.
Selected for 2017 RO exam
Rev. 1, 5/19/17
Re-sequenced choices so they are short to long.
Ediorial changes.

QID: 0978 **Rev:** 1 **Rev Date:** 5/21/17 **Source:** Bank **Originator:** NRC

TUOI: A1LP-RO-RMS **Objective:** 9 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 073 **System Title:** Process Radiation Monitoring

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRM system controls including: Radiation levels.

K/A Number: A1.01 **CFR Reference:** 41.5 / 45.7

Tier: 2 **RO Imp:** 3.2 **RO Select:** Yes **Difficulty:** 4

Group: 1 **SRO Imp:** 3.2 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 51 **SRO:**

Given:
* 100% power
* A OTSG N-16 TROUBLE (K07-A5) alarms, RI-2691 5.5 x 10^3 cpm and stable
* B OTSG N-16 TROUBLE (K07-A6) alarms, RI-2692 4 x 10^3 cpm and slowly rising

Both RI-2691 and RI-2692 taken from GROSS to ANALYZER per ANNUNCIATOR K07 CORRECTIVE ACTION (1203.012F), following readings observed:

* RI-2691 5.5 x 10^3 cpm and stable
* RI-2692 3.5 x 10^3 cpm and slowly rising

Which of the following explains the above observations?

A. RI-2691 has failed to 5.5 x 10^3 cpm as confirmed by no change in count rate.

B. RI-2692 has low voltage condition as confirmed by drop in count rate following completion of 1203.012F.

C. RI-2691 operating properly and steam generator tube leak is confirmed as detected by steady 5.5 x 10^3 cpm reading.

D. Both RI-2691 and RI-2692 operating properly with confirmed steam generator tube leak present in both steam generators with different leak rates.

Answer:

A. RI-2691 has failed to 5.5 x 10^3 cpm as confirmed by no change in count rate.

Notes:

"A" is correct, 1203.012F directs the rate meter mode to be changed from Gross to Analyzer if the reactor is critical and RI-2691/2692 in Alert or High alarm. This will screen out all activity except N-16 thus the reading would be lower after the change.

"B" is incorrect. A drop in count rate is expected after completion of 1203.012F where the rate meter mode is changed from Gross to Analyzer. A low voltage condition would cause the rad monitor to cease functioning.

"C" is incorrect, but plausible since RI-2692 reading upscale indicates a tube leak is occuring but RI-2691 would read lower after completion of 1203.012F.

"D" is incorrect, but plausible as the indications are that a tube leak is occuring, but RI-2691 would read lower after completion of 1203.012F.

This question matches the K/A since the applicant must know how these monitors detect radiation levels, specifically how they detect N-16 gammas, and requires them to know how the controls (selection switch) changes the reading to arrive at the correct answer: one of the process radiation monitors has failed and this monitor should not be used in determining if design limits have been exceeded.

References:

1203.012F, Annunciator K07 Corrective Action
STM 1-62, Radiation Monitoring

History:

New for 2013 Exam
Selected for 2017 RO Re-exam
Rev. 1, 5/21/17
Editorial changes

QID: 0046 **Rev:** 3 **Rev Date:** 5/21/17 **Source:** Bank **Originator:** Cork

TUOI: A1LP-RO-EOP07 **Objective:** 3 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal From Reactor Core

System Number: 076 **System Title:** Service Water System

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

K/A Number: A4.02 **CFR Reference:** 41.7 / 45.5 to 45.8

Tier: 2 **RO Imp:** 2.6 **RO Select:** Yes **Difficulty:** 4

Group: 1 **SRO Imp:** 2.6 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 52 **SRO:**

Given:
* Degraded Power
* Both EDGs operating
* ESAS has NOT actuated
* SW pump P4C failed to start
* SW pump P4B out of service

Which of the following actions is required for the above conditons?

A. Close ACW Loop Isolation (CV-3643).

B. Close SW Loop II Isolation Valve (SW-10C).

C. Open SW Loop I & II Crossconnects (SW-5 and SW-6).

D. Cross-tie SW Loops at Makeup Pump (SW-14 thru SW-17).

Answer:

A. Close ACW Loop isolation (CV-3643).

Notes:

"A" is correct per the Degraded Power EOP, this action is taken to isolate ACW thereby reducing SW system flow and preventing runout of the sole remaining SW pump. There is no ES actuation so ACW isolation CV-3643 will not close automatically.

"C" is incorrect since this will only isolate the loop flow from P4C with the SW loops still cross-tied at the ICW coolers. This is plausible since it will reduce SW flow but is not required by 1202.007.

"B" is incorrect but plausible since this is a manual isolation which would reduce SW flow by isolating Loop II.

"D" is incorrect but plausible since this would ensure SW flow to the Makeup pumps since only one train of SW is operating but this is not directed by 1202.007.

Re-ordered answer choices due to multiple exam use.

This question matches the K/A since it involves manual operation of a Service Water valve in the control room.

References:

1202.007, Degraded Power

History:

Developed for 1998 RO/SRO Exam.
Revised after 9/98 exam analysis review.
Used in 98 RO Re-exam
Selected for use in 2002 RO exam.
Selected for use on 2007 RO Exam

Selected for 2011 RO Exam.
Selected for 2017 RO Re-exam
Rev. 3, 5/21/17
Editorial changes.

QID: 0534 **Rev:** 3 **Rev Date:** 5/21/17 **Source:** Bank **Originator:** NRC

TUOI: A1LP-RO-MSSS **Objective:** 1 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal From Reactor Core

System Number: 076 **System Title:** Service Water System

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

K/A Number: K2.08 **CFR Reference:** 41.7

Tier:	2	**RO Imp:**	3.1	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.3	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 53 **SRO:**

Given:
* Reactor trip
* ES channels 1 through 4 actuated

While performing RT-10 to verify proper ES actuation, CBOT notes following on C16:

* P4A to P4B Crosstie (CV-3644) has no position indication
* P4B to P4C Crosstie (CV-3642) has no position indication

This is due to loss of power to MCC Load Center _____ on the _____ train.

A. B52 / Red

B. B55 / Red

C. B62 / Green

D. B64 / Green

Answer:

C. B62 / Green

Notes:

"C", this is the correct power supply for both CV-1284 and CV-1285, both are powered from the same MCC, B62. One valve is even numbered and one is odd numbered so applicant must be familiar with power supplies in order to choose correct answer and eliminate distractors.

"A" is incorrect but plausible, this is an ES red train MCC which powers other HPI block valves.

"B" is incorrect but plausible as this is an ES red train MCC.

"D" is incorrect but plausible as this is and ES green train MCC.

Modified this question since the original had two implausible distractors (panel identification eliminated two). Changed valves from Service Water since these valves are in another question, replaced SW valves with HPI block valves. Replaced two distractors which could be eliminated simply by knowing which train they were.

References:

1107.002, ES Electrical System Operation
STM 1-32, Electrical Distribution

History:

Developed by NRC.
Used on 2004 RO/SRO Exam.
Modified for 2005 RO re-exam.
Selected for 2017 RO Re-exam

Rev. 3, 5/21/17
Changed valves from HPI block to SW discharge crosstie valves, answers remain the same, correct answer unchanged.
Editorial changes.

QID: 0988 **Rev:** 1 **Rev Date:** 5/21/17 **Source:** Bank **Originator:** NRC

TUOI: A1LP-RO-MSSS **Objective:** 10 **Point Value:** 1

Section: 3.8 **Type:** Plant Service Systems

System Number: 078 **System Title:** Instrument Air System

Description: Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems: MSIV air.

K/A Number: K1.05 **CFR Reference:** 41.2 to 41.9 / 45.7 to 45.8

Tier: 2 **RO Imp:** 3.4 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 3.5 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 54 **SRO:**

The design purpose of the backup air accumulators for the MSIVs is to ____________________ due to loss of Instrument Air.

A. Operate ADV's for 30 minutes

B. Maintain MSIV's open for 30 minutes

C. Operate ADV's for 60 minutes

D. Maintain MSIV's open for 60 minutes

Answer:

B. Maintain MSIV's open for 30 minutes

Notes:

Answer A is incorrect. The purpose is to ensure the MSIV’s have enough air to hold them open for 30 minutes not for ADV control this can be done locally with no Instrument air pressure although the accumulators supply air to the ADV’s .

Answer B is correct. The purpose is to ensure the MSIV’s have enough air to hold them open for 30 minutes. They are spring to close valves and air to open and maintain open.

Answer C is incorrect. Due to reasons above and the time is too long.

Answer D is incorrect Due to reasons above and the time is too long.

References:

STM1-15, Main Steam System

History:

New for 2013 Exam
Selected for 2017 RO Re-exam
Rev. 1, 5/21/17
Revised stem to incorporate repetitive elements of answer choices.
Editorial changes.

QID: 0867 **Rev:** 3 **Rev Date:** 6/8/17 **Source:** Modified **Originator:** Cork

TUOI: A1LP-RO-EOP05 **Objective:** 9 **Point Value:** 1

Section: 3.5 **Type:** Containment Integrity

System Number: 103 **System Title:** Containment

Description: Ability to (a) predict the impacts of the following malfunction or operations on the containment system, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Phase A and B isolation.

K/A Number: A2.03 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

Tier: 2 **RO Imp:** 3.5 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 3.8 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 55 **SRO:**

Given:
* Reactor tripped from 100% power
* RCS pressure 1750 psig and lowering
* CETs 500 °F and lowering
* RB pressure 19.8 psia and rising

Which of the following correctly lists ESAS channels required to be actuated AND an appropriate corrective action based on the above conditions?

A. Channels 1, 2, 3, 4;
Trip all running RCPs per RT-10, Verify Proper ESAS Actuation

B. Channels 1, 2, 3, 4, 5, 6;
Restore Service Water and ACW per 1104.029 Exhibit B,
Restoring SW to ICW Following ES Actuation

C. Channels 1, 2, 3, 4;
Restore Service Water and ACW per 1104.029 Exhibit B,
Restoring SW to ICW Following ES Actuation

D. Channels 1, 2, 3, 4, 5, 6;
Trip all running RCPs per RT-10, Verify Proper ESAS Actuation

Answer:

D. Channels 1, 2, 3, 4, 5, 6;
Trip all running RCPs per RT-10, Verify Proper ESAS Actuation

Notes:

"D" is correct, with Reactor Building pressure greater than 18.7 psia, ESAS channels 1 thru 6 will actuate and ESAS will be verified to be operating correctly per RT-10. RT-10 states that if either Channnel 5 or 6 have actuated to secure all running RCPS due to ICW isolation. B&W ESAS channels 1-4 actuate on low RCS pressure (1550 psia) and isolate some RB penetrations, this is the equivalent of the Westinghouse Phase A isolation. If RB pressure exceeds 18.7 psia, then ESAS Channels 1-4 AND channels 5&6 actuate with channels 5&6 being the equivalent of a Westinghouse Phase B isolation. FYI, the conditions given are indicative of a steam line break in the building with RCS pressure low but with plenty of subcooling margin and RB pressure rising.

"A" is incorrect ESAS channels 5 and 6 also should be actuated, but is plausible since Trip all running RCPs is the correct action to take.

"B" is incorrect but plausible since ESAS Channels 1 thru 6 are the correct channels but (per RT-10) Service Water is restored to ICW only if channels 5 nor 6 have NOT actuated.

"C" is incorrect ESAS channels 5 and 6 also should be actuated, but plausible since SW is restored to ICW if only channels 1-4 have actuated.

Revised, this question did not match the K/A originally. Replaced "A LOCA is in progress" with "the reactor tripped from 100%" and revised condtions to reflect a steam line break; deleted "ESAS has actuated" from conditions; revised A & B as they were implausible since there is no scenario where channels 7 - 10 would actuate alone. Revised all answer choices with actions from RT-10 appropriate to the conditions.

This question matches the K/A since it requires the applicant to predict the impact of rising RB pressure on the Reactor Building (containment) and requires the applicant to know the actions from RT-10 that are taken specifically if ESAS channels 5 or 6 actuate (phase B isolation).

References:

1202.012, Repetitive Tasks, RT-10, Verify Proper ESAS Actuation

History:

New question written by NRC for 2013 exam
Modified for 2017 RO Re-exam
Rev.2, 5/21/17
Added per RT-10 to A and D
Editorial changes.
Rev. 3, 6/8/17
Copied missing piece of correct answer into answer field.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1134 **Rev:** 1 **Rev Date:** 5/21/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-RPS **Objective:** 4 **Point Value:** 1

Section: 3.1 **Type:** Reactivity Control

System Number: 001 **System Title:** Control Rod Drive System

Description: Knowledge of the physical connections and/or cause effect relationships between the CRDS and the following systems: CCWS must be cut in before energizing CRDS.

K/A Number: K1.09 **CFR Reference:** 41.2 to 41.9 / 45.7 to 45.8

Tier:	2	**RO Imp:**	2.8	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	3.1	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 56 **SRO:**

Given:
* Unit 1 recovering from reactor trip
* Preparations underway for re-start

Which of the follow annunciators ALONE, if in alarm, will prevent closing the CRD AC trip breakers?

A. CRD PUMP PREFILTER DP HI (K08-E1)

B. CRD COOLING RETURN FLOW LO (K08-C1)

C. CRD COOLING RETURN TEMP HI (K08-B1)

D. CRD CLNG SUPP TEMP HI / ROOM FLOOD (K08-A1)

Answer:

B. CRD COOLING RETURN FLOW LO (K08-C1)

Notes:

"B" is correct. FS-2222 causes this annunciator when return flow is ≤127.9 gpm and is an interlock to closing the CRD AC trip breakers. If the alarm is not in, then the AC trip breakers will not close. This is to prevent energizing the CRDM's with inadequate cooling water flow.

"A" is incorrect, although plausible since a clogged filter could be indicative of low flow but the prefilters have bypass valves which open when this alarm comes in to prevent a low flow condition. This alarm has no interlock with the AC trip breakers.

"C" is incorrect, although plausible since a low flow condition could cause this alarm but this alarm could also be caused by inadequate Service Water flow to the Non-Nuclear ICW cooler. This alarm has no interlock with the AC trip breakers.

"D" is incorrect, although plausible since a low flow condition could cause this alarm but this alarm could also be caused by inadequate Service Water flow to the Non-Nuclear ICW cooler. This alarm has no interlock with the AC trip breakers.

This question matches the K/A since it requires knowledge of the cause/effect relationship (interlock) between the CRD system and the ICW (CCWS) system.

References:

1105.009, CRD System Operating Procedure

History:

New question for 2017 RO Re-take exam.
Rev. 1, 5/21/17
Editorial changes.

QID: 1135 **Rev:** 0 **Rev Date:** 3/9/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-ICS **Objective:** 30 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal from Reactor Core

System Number: 002 **System Title:** Reactor Coolant System

Description: Knowledge of the effect that a loss or malfunction of the RCS will have on the following: Fuel.

K/A Number: K3.02 **CFR Reference:** 41.7

Tier: 2 **RO Imp:** 4.2 **RO Select:** Yes **Difficulty:** 2

Group: 2 **SRO Imp:** 4.5 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 57 **SRO:**

What feature in the Integrated Control System (ICS) is designed specifically to respond to malfunctions which will have an effect on the nuclear fuel?

A. Feedwater Cross-Limits

B. BTU Limit Alarm

C. Delta Tc Controller

D. Rapid Feedwater Reduction

Answer:

C. Delta Tc Controller

Notes:

"C" is correct. The Delta Tc Controller adjusts FW flow to each SG based on the amount of RCS flow through that SG in order to maintain Delta Tc at minimum. Maintaining Delta Tc at minimum ensures that Quadrant Tilt Limits are not challenged and thus challenges to fuel are minimized.

"A" is incorrect yet plausible since Feedwater Cross-Limits can limit Reactor power based on available Feedwater. The purpose of this however is to ensure heat production and heat removal are balanced so that system transients are minimized and power production can continue.

"B" is incorrect yet plausible since BTU Limits calculations are greatly affected by RCS flow and a reduction in RCS flow will bring in the BTU Limits alarm(s). However, the BTU Limits alarms are to ensure 35°F superheat is available, not nuclear fuel considerations.

"D" is incorrect yet plausible since the RFR rapidly reduces Feedwater following a Reactor Trip which reduces the overcooling of the primary from the secondary. Lowering RCS pressure from overcooling does place the fuel closer to DNB but the primary concern for overcooling the RCS is stresses to the Reactor vessel, not the fuel.

This question matches the K/A since it asks the applicant what ICS feature is designed to keep RCS malfunctions from affecting the nuclear fuel. The applicant must know that a difference in Tcold temperatures caused by a reactor coolant flow malfunction (i.e., a Reactor Coolant Pump trip) will cause reactor core quadrant tilt to become worse and that the purpose of the Delta Tc Controller is to minimize differences in Tcold temperatures between the loops which, in turn, minimizes Quadrant Power Tilts in the reactor core.

References:

STM 1-64, Integrated Control System

History:

New question for 2017 RO Re-exam.

QID: 1136 **Rev:** 1 **Rev Date:** 5/21/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-CRD **Objective:** 14 **Point Value:** 1

Section: 3.1 **Type:** Reactivity Control

System Number: 014 **System Title:** Rod Position Indication

Description: Knowledge of the operational implications of the following concepts as they apply to the RPIS: RPIS independent of demand position

K/A Number: K5.02 **CFR Reference:** 41.5 / 45.7

Tier: 2 **RO Imp:** 2.8 **RO Select:** Yes **Difficulty:** 3

Group: 2 **SRO Imp:** 3.3 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 58 **SRO:**

Given:

* 100% power

* 1105.009, Supplement 1, Absolute and Relative Position Indication Comparison, in progress

* CBOT reports Control Rod 6-5 is indicating 95% RPI and 100% API via PI panel

* All Group 6 Out Limit lights are ON

Which of the following actions are required for Control Rod 6-5 per 1105.009 for the above conditions?

A. Realign rod with rest of Group 6.

B. Place S-2 toggle switch for rod in bypass.

C. Adjust rod's RPI to agree with API at PI panel.

D. Declare rod inoperable and enter T.S. LCO 3.1.4.

Answer:

C. Adjust rod's RPI to agree with API at PI panel.

Notes:

"C" is correct, RPI stands for Relative Position Indication and corresponds to rod demand. API stands for Absolute Position Indication and is determined from actual rod position. With all Group 7 Out Limit lights on, a validation of API position being accurate exists since the 100% zone indication comes from the out limit switch, thus RPI is not accurate and should be adjusted per Supplement 1 step 3.1.3.A, third bullet.

"A" is incorrect, yet plausible since this is performed for rod misalignments per Supplement 1 step 3.1.3.A (first bullet) but in this case the rod is not misaligned.

"B" is incorrect, yet plausible as this is done when API is determined to be faulty but it is the RPI which is invalid.

"D" is incorrect, yet plausible if applicant cannot recall threshold for operability (6.5%) per TS 3.1.4 and believes API to be invalid.

This question matches the K/A since it involves the Rod Position Indication and requires applicant to have knowledge of the operational implication (adjust RPI vs. actions for API) of rod position indication independent of demand (RPI).

References:

1105.009, CRD System Operating Procedure
1203.003, Control Rod Drive Malfunction Action

History:

New question for 2017 RO Re-exam
Rev. 1, 5/21/17
Changed group and rod to 6-5 (from7-5).
Incorporated rod into stem.
Re-ordered answers from short to long, correct answer unchanged.
Editorial changes.

QID: 0464 **Rev:** 2 **Rev Date:** 5/21/17 **Source:** Bank **Originator:** Cork

TUOI: A1LP-RO-NI **Objective:** 10 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 015 **System Title:** Nuclear Instrumentation System

Description: Ability to predict and/or monitor changes in parameters to prevent exceeding design limits associated with operating the NIS controls including: SUR.

K/A Number: A1.02 **CFR Reference:** 41.5 / 45.5

Tier:	2	**RO Imp:**	3.5	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	3.6	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 59 **SRO:**

Given:

* Reactor startup in progress

* Excore NI's indicating:
  * NI-1 & NI-2 1 E3 CPS
  * NI-3 & NI4 2 x 10 -11 amps

During the next rod pull, HI SUR ROD HOLD (K08-B2) alarms.
Simultaneously CRD WITHDRAWAL INHIBITED (K08-A2) alarms.

The cause of alarms is excessive rod pull causing SUR to reach setpoint of:

A. 1 DPM on SR monitors

B. 2 DPM on SR monitors

C. 1 DPM on IR monitors

D. 2 DPM on IR monitors

Answer:

B. 2 DPM on SR monitors

Notes:

"B" is the correct answer, a SUR reaching or exceeding 2 DPM on the source range monitors will cause the HI SUR and CRD withdrawal inhibited alarms.

"A" is an incorrect answer, a SUR of 1 DPM on the source range monitors will not cause the HI SUR alarm to come in. This distractor is plausible since this is the maximum SUR allowed per limit and precaution 6.3 in 1102.008, Approach to Criticality.

"C" is an incorrect answer, a SUR of 1 DPM on the intermediate range monitors will not cause the HI SUR alarm to come in. Setpoint for the IR high SUR rod hold is 3 DPM. This distractor is plausible since this is the maximum SUR allowed per limit and precaution 6.3 in 1102.008, Approach to Criticality.

"D" is an incorrect answer, a SUR of 2 DPM on the intermediate range monitors will not cause the HI SUR alarm to come in. Setpoint for the IR high SUR rod hold is 3 DPM. This is plausible since this is the setpoint for the Hi SUR rod hold from the source range monitors.

This question matches the K/A since it test the ability of the applicant to monitor alarms and NI parameters (SUR) of automatic action which prevent exceeding design limits.

References:

1203.012G, Annunciator K08 Corrective Action

History:

Direct from regular exambank QID 1789.
Selected for use in 2002 RO/SRO exam.
Selected for 2011 RO Exam.
Selected for 2017 RO Re-exam
Rev. 2, 5/21/17
Added IR readings.
Editorial changes.

QID: 1137 **Rev:** 1 **Rev Date:** 5/21/17 **Source:** Modified **Originator:** Cork

TUOI: A1LP-RO-NNI **Objective:** 5 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 016 **System Title:** Non-Nuclear Instrumentation

Description: Ability to monitor automatic operation of the NNIS, including: Automatic selection of NNIS inputs to control systems.

K/A Number: A3.01 **CFR Reference:** 41.7 / 45.5

Tier:	2	**RO Imp:**	2.9	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	2.9	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 60 **SRO:**

Given:

* 100% power

* Loop A RCS flow 65 E6 lbm/hr

* Loop B RCS flow 70 E6 lbm/hr

* Loop A Tave 580 °F

* Loop B Tave 578 °F

* Unit Tave 579 °F

What causes SASS to select a specific Tave for control (vs. Unit Tave) and which Tave will be selected by the SASS Auto/Manual transfer switch?

A. Loop A flow low, Loop B Tave

B. Loop A Tave high, Loop B Tave

C. Loop A Tave high, Loop A Tave

D. Loop A flow low, Loop A Tave

Answer:

A. Loop A flow low, Loop B Tave

Notes:

"A" is correct. SASS will automatically select the Loop Tave for the Loop with the highest RCS flow should either flow drop below 95%. Normally Unit Tave is used for control. Normal RCS loop flow is ~70 E6 lbm/hr, therefore Loop A flow is <95% and SASS will select Loop B for Tave control. This control function protects the core from excessive heat transfer based upon flux to flow.

"B" is incorrect, SASS does not select Tave control based upon which is the highest, it is based on highest RCS flow. This is plausible since most control selections are based upon the highest indication.

"C" is incorrect, SASS does not select Tave control based upon which is the highest, it is based on highest RCS flow. This is plausible since most control selections are based upon the highest indication.

"D" is incorrect, this is the correct cause for Tave control to change from Unit Tave to a specific loop Tave but it will swap to the loop with the highest flow, not the lowest.

This is a modified version of QID 77. All answers were changed to a 2x2 format. The loop with the lower flow was changed to A. The order of the reason and signal selection were reversed in the stem. Added 100% power condition. Changed RCS flow for the lower loop to be closer to the 95% threshold.

This question matches the K/A due to subject is NNI and the knowledge required is what will cause automatic selection of NNIS input (Tave) and which will be selected (monitor).

References:

STM 1-69, Non-Nuclear Instrumentation System

History:

Modified QID 77 for 2017 RO Re-exam
Rev. 1, 5/21/17
Editorial changes.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1138 **Rev:** 1 **Rev Date:** 5/21/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-RBVEN **Objective:** 12 **Point Value:** 1

Section: 3.8 **Type:** Plant Service Systems

System Number: 029 **System Title:** Containment Purge System

Description: Ability to verify that the alarms are consistent with the plant conditions.

K/A Number: 2.4.46 **CFR Reference:** 41.10 / 43.5 / 45.3 / 45.12

Tier: 2 **RO Imp:** 4.2 **RO Select:** Yes **Difficulty:** 3

Group: 2 **SRO Imp:** 4.2 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 61 **SRO:**

Given:

* Unit 1 in Mode 5
* Fuel movement NOT in progress
* Commencing a Reactor Building Purge per 1104.033, Attachment B

* CBOT starts RB Purge Supply Fan (VSF-2)
* Annunciator CONT BLDG EXHAUST FAN AUTO TRIP (K15-A3) alarms

Which of the following is an action required per Reactor Building Ventilation (1104.033) and is consistent with the above alarms?

A. Place RB Purge Supply Fan (VSF-2) handswitch in OVERRIDE.

B. Place RB Purge Supply Fan (VSF-2) handswitch in MAN.

C. Place RB Purge Exhaust Fan (VEF-15) handswitch in MAN.

D. Place RB Purge Exhaust Fan (VEF-15) handswitch in OVERRIDE.

Answer:

C. Place RB Purge Exhaust Fan (VEF-15) handswitch in MAN.

Notes:

"C" is correct, if VEF-15 fails to start when VSF-2 starts, then an attempt is made to start VEF-15 by going to MAN. This alarm could come in if VEF-15 handswitch was not placed in AUTO prior to starting VSF-2.

"A" is incorrect but plausible since placing VSF-2 in "override" will prevent it from tripping and a note in the procedure states this, the procedure does not have a step allowing this action.

"B" is incorrect but plausible since the required action is to place VEF-15 handswitch in MAN.

"D" is incorrect but plausible since VSF-2 has an override feature but VEF-15 does not.

This question matches the K/A as it pertains to the containment purge system (RB purge) and requires the applicant to recall which action is necessary in response to the alarm (i.e., the applicant must know how the system works).

References:

1104.033 Reactor Building Ventilation

History:

New question for 2017 RO Re-exam
Rev. 1, 5/21/17
Changed B distractor to place VSF-2 in MAN to make question a 2x2.
Editorial changes.

QID: 0296 **Rev:** 1 **Rev Date:** 5/21/17 **Source:** Bank **Originator:** D. Slusher

TUOI: A1LP-RO-ICS **Objective:** 17 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal From Reactor Core

System Number: 041 **System Title:** Steam Dump System and Turbine Bypass Control

Description: Ability to manually operate and/or monitor in the control room: Steam dump valves

K/A Number: A4.08 **CFR Reference:** 41.7 / 45.5 to 45.8

Tier:	2	**RO Imp:**	3.0	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.1	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 62 **SRO:**

Given:

* Startup in progress
* Turbine-Generator in Integrated Control
* Generator load 175 megawatts
* Turbine header pressure 890 psig
* Header pressure setpoint set at 49%

With the above conditions at what pressure will the Turbine Bypass Valves open and close?

A. Open at 905 psig;
Close < 895 psig

B. Open at 945 psig;
Close < 945 psig

C. Open at 995 psig;
Close < 985 psig

D. Open at 895 psig;
Close < 885 psig

Answer:

B. Open at 945 psig;
Close < 945 psig

Notes:

"B" is correct, at above 15% MW output (~135MWe) a +50 psig bias is added to the Turbine Bypass Valve setpoint (895 psig, 49% on header pressure setpoint station) to prevent the Turbine Bypass Valves (TBVs) from competing with the turbine governor valves. For the given conditions the Turbine Bypass Valves would open if header pressure reaches 945 psig (setpoint + 50 psig) and close when header pressure falls below 945 psig.

"A" is incorrect, although plausible since this is how the TBVs will operate if generator load dropped to less than 15%, the 50 psig bias would be removed and the TBVs would open at 10 psig above setpoint and close at setpoint. But the given load is above 15%.

"C" is incorrect, although plausible since this is how the TBVs operate following a turbine trip when a 100 psig bias is applied.

"D" is incorrect, although plausible if there was no bias applied to the TBVs and they opened at the normal setpoint of 895 psig.

This question matches the K/A since it concerns the Steam Dump/Turbine Bypass Control system (ICS) and for the applicant to monitor operation in the control room the applicant must know when to expect the TBVs to open.

References:

1105.004, Integrated Control System
STM 1-64, Integrated Control System

History:

Used in 1999 exam, modified from ExamBank, QID# 345.
Selected for 2017 RO Re-exam
Rev. 1, 5/21/17
Simplified answer choices by removing repetitive wording.
Editorial changes

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1139 **Rev:** 2 **Rev Date:** 6/8/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-AOP **Objective:** 4 **Point Value:** 1

Section: 3.9 **Type:** Radioactivity Release

System Number: 068 **System Title:** Liquid Radwaste

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the Liquid Radwaste System ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure of automatic isolation. (CFR: 41.5 / 43.5 / 45.3 / 45.13)

K/A Number: A2.04 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

Tier:	2	**RO Imp:**	3.3	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.3	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 63 **SRO:**

Given:

* 100% power

* Treated Waste Monitor Tank T-16A release is in progress

* Annunciator PROC MONITOR RADIATION HI (K10-B2) alarms

CBOT reports Liquid Radwaste Process Monitor (RI-4642) in alarm.

CBOT reports Liquid Waste to Flume valve (CV-4642) failed to close, CV-4642 was closed using handswitch.

Which of the following actions are required by Liquid Waste Discharge Line High Radiation Alarm (1203.007) in response to the above conditions?

A. Stop Treated Waste Monitor Pump (P-47A)

B. Close Treated Waste Discharge to Circ Water Flume (CZ-58)

C. Reset Liquid Radwaste Process Monitor RI-4642 and re-establish release

D. Ensure Plant Computer tabular log activated for RI-4642 and re-establish release

Answer:

A. Stop Treated Waste Monitor Pump (P-47A)

Notes:

"B" is correct, even though Rad Monitor RI-4642 only experienced a momentary "spike" the applicant should deduce that CV-4642 is inoperable since it did not close when RI-4642 alarmed, and that in addition to verifying CV-4642 goes closed, the Treated Waste Monitor Pump (P-47A) should be stopped per 1203.007. This ensures the release is terminated.

"B" is incorrect, but plausible as this action would be taken if CV-4642 could not be closed. CV-4642 was closed so this step is not applicable.

"C" is incorrect, but plausible since this is allowed due to spikes on RI-4642 but in this instance CV-4642 did not automatically close and the release permit does not contain contingencies for CV-4642 being inoperable.

"D" is incorrect, but plausible since, per the release permit in 1104.020, the tabular log is activated if RR-4830 is unavailable.

This question matches the K/A since it involves a liquid radwaste release where automatic isolation failed to occur. The prediction of the impact is the applicant must realize CV-4642 will not perform it's function and deduce which of the choices given must be performed for CV-4642's failure.

References:

1104.020, Clean Waste System Operation
1203.007, Liquid Waste Discharge Line High Radiation Alarm

History:

New question for 2017 RO Re-exam
Rev. 1, 5/21/17
Changed distractor D to "Ensure Plant Computer tabular log activated for RI-4642 and re-establish release."
Deleted "but is now indicating pre-release value" from RI-4642 condition, changed was to is.
Changed last condtion for CV-4642 to failed to close (vs. is open).
Re-sequenced A and B so answer choices are short to long.
Editorial changes.
Rev. 2, 6/8/17
Changed answer field to correct letter.
Corrected Notes filed to reflect changes to distractor D from Rev. 1.

QID: 1140 **Rev:** 1 **Rev Date:** 5/21/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-MSSS **Objective:** 5 **Point Value:** 1

Section: 3.8 **Type:** Plant Service Systems

System Number: 079 **System Title:** Station Air System

Description: Knowledge of SAS design feature(s) and/or interlock(s) which provide for the following: Cross-connect with IAS.

K/A Number: K4.01 **CFR Reference:** 41.7

Tier: 2 **RO Imp:** 2.9 **RO Select:** Yes **Difficulty:** 2

Group: 2 **SRO Imp:** 3.2 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 64 **SRO:**

In accordance with Service Air System (1104.025), SA to IA X-Over Valve (SV-5400) ___________ when Instrument Air pressure lowers to _____ psig.

A. Opens;
60

B. Opens;
50

C. Closes;
50

D. Closes;
60

Answer:

B. Opens;
50

Notes:

"B" is correct, to support a lowering Instrument Air header pressure, SV-5400 opens when Inst. Air pressure drops down to 50 psig as a means to prevent Inst. Air from dropping further.

"A" is incorrect, but plausible as this is the correct position for SV-5400 on a dropping Inst. Air header pressure but the setpoint is incorrect. This setpoint is the identical to the setpoint used in 1203.024, Loss of Instrument Air, to determine if the cross-connect between Unit 1 and Unit 2 Inst. Air systems should be closed.

"C" is incorrect, but plausible as this is the correct setpoint but the position is incorrect. This is further plausible if the applicant thinks of the Service Air System as a source of leakage which should be isolated on lowering Inst. Air header pressure but SV-5400 is normally closed.

"D" is incorrect, but plausible if the applicant thinks of the Service Air System as a source of leakage which should be isolated on lowering Inst. Air header pressure but SV-5400 is normally closed. SV-5400 opens automatically to try to hold Instrument Air pressure up. The setpoint in this distractor is the identical to the setpoint used in 1203.024, Loss of Instrument Air, to determine if the cross-connect between Unit 1 and Unit 2 Inst. Air systems should be closed.

This question matches the K/A since it involves the Service Air (Station Air) cross-connect with instrument air and requires applicant knowledge of the interlock between the two.

References:

1104.025, Service Air System
1203.024, Loss of Instrument Air

History:

New for 2017 RO Re-exam
Rev. 1, 5/21/17

Revised stem to incorporate beginning statement so question is now a fill in the blank.
Added 1203.024 to references.
Simplified answers accordingly

QID: 0619 **Rev:** 2 **Rev Date:** 5/21/17 **Source:** Bank **Originator:** J.Cork

TUOI: A1LP-RO-FPS **Objective:** 6 **Point Value:** 1

Section: 3.8 **Type:** Plant Service Systems

System Number: 086 **System Title:** Fire Protection System (FPS)

Description: Knowledge of the effect of a loss or malfunction of the Fire Protection System will have on the following:
Fire, smoke, and heat detectors.

K/A Number: K6.04 **CFR Reference:** 41.7 / 45.7

Tier: 2 **RO Imp:** 2.6 **RO Select:** Yes **Difficulty:** 3

Group: 2 **SRO Imp:** 2.9 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 65 **SRO:**

The smoke detector string for Cable Spreading Room has a trouble alarm on it's respective Zone Indicating Unit (ZIU).

If a fire occurs in Cable Spreading Room which of the following is a functional method of actuating the deluge system for this area?

A. Manually via Man Trip switch on C463

B. Automatic actuation via protectowire detector

C. Manually by taking Inhibit switch to "Normal" on C463

D. Automatic actuation via smoke detector and protectowire detector

Answer:

A. Manually via Man Trip switch on C463

Notes:

"A" is correct, a Trouble alarm means a de-energized or open smoke detector string, therefore automatic operation is non-functional for the Cable Spreading Room. The Cable Spreading Room is a cross-zoned system, meaning it takes both a smoke detector and a protectowire detector in alarm to actuate the sprinkler valve. Cross-zoned systems have an Inhibit switch to disable automatic actuation when either detector string has a malfunction, or trouble alarm. Operating the Man Trip switch on C463 bypasses the automatic actuation (and Inhibit) contacts to manually trip the sprinkler valve.

"B" is incorrect,but plausible if the applicant cannot recall how a cross-zoned system works and thinks either string will automatically actuate the system. Automatic actuation is not available on a cross zoned system without both detection strings.

"C" is incorrect, although the Inhibit switch is taken to Inhibit on cross zoned systems when a string is inoperable, taking it out of inhibit will enable automatic actuation but automatic actuation is inoperable due to the inoperable smoke detector string.

"D" is incorrect, but plausible if the applicant does not understand the significance of the Trouble alarm on the smoke detector string. This is how the system automatically actuates without any malfunction.

This question matches the K/A since it involves a malfunction of a smoke detector string and the applicant must have knowledge of the effect of how this malfunction affects automatic and manual actuation of the sprinkler system.

References:

1203.009, Fire Protection System Annunciator Corrective Action
1104.032, Fire Protection System

History:

New for 2005 RO exam, replacement question.
Selected for 2017 RO Re-exam
Rev. 2, 5/21/17
Changed "operable" to "functional" in stem to be more accurate.
Re-ordered answers short to long.
Editorial changes.

QID: 1141 **Rev:** 1 **Rev Date:** 5/21/17 **Source:** New **Originator:** Cork

TUOI: A1QC-RO-QUAL **Objective:** 2.1 **Point Value:** 1

Section: 2.0 **Type:** Generic KA's

System Number: 2.1 **System Title:** Conduct of Operations

Description: Knowledge of shift or short-term relief turnover practices.

K/A Number: 2.1.3 **CFR Reference:** 41.10 / 45.13

Tier:	3	**RO Imp:**	3.7	**RO Select:**	Yes	**Difficulty:**	2
Group:		**SRO Imp:**	3.9	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 66 **SRO:**

Given:

* 100% power

* CRS, ATC and CBOT are only licensed operators in control room.

* ATC needs to go to computer room to perform procedure step.

Per EN-OP-115-03, Shift Turnover and Relief, which ONE of the following describes the requirements for this activity?

The ATC:

A. Must be relieved by a licensed operator other than the current CBOT.

B. May leave as long as a currently on-shift RO or SRO monitors the panels until the ATC returns.

C. Must remain in the control room at all times until formally relieved by another licensed operator.

D. May leave the control room, with CRS permission, without being relieved.

Answer:

C. Must remain in the control room at all times until formally relieved by another licensed operator.

Notes:

"C" is correct, a formal relief is required for the ATC to leave the control room.

"A" is incorrect, but plausible since the shift manning requirement is for two licensed operators to be in the control room .

"B" is incorrect, but plausible as this is the requirement during log taking while the ATC goes to the back panels.

"D" is incorrect, the ATC may not leave the control room without being relieved.

This question matches the K/A since it concerns short term relief turnover practices.

References:

EN-OP-115-03, Shift Turnover and Relief
COPD-001, Operations Expectations and Standards

History:

New for 2017 RO-rexam.
Rev. 1, 5/21/17
Editorial changes.

QID: 1142 **Rev:** 1 **Rev Date:** 5/21/17 **Source:** New **Originator:** Cork

TUOI: A1LP-AO-VALVE **Objective:** 5 **Point Value:** 1

Section: 2.0 **Type:** Generic KA's

System Number: 2.1 **System Title:** Conduct of Operations

Description: Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.

K/A Number: 2.1.29 **CFR Reference:** 41.10 / 45.1 / 45.12

Tier:	3	**RO Imp:**	4.1	**RO Select:**	Yes	**Difficulty:**	2
Group:		**SRO Imp:**	4.0	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 67 **SRO:**

Given:
* Unit 1 heating up following refueling
* Valve lineup in progress but an MOV inside Reactor Building is leaking by
* MOV to be manually seated
* This is a non-Q MOV

What are two of the procedural requirements in Conduct of Operations (1015.001) for manually seating this MOV?

A. Verify MOV breaker open AND tighten using a torque wrench.

B. Danger tag MOV breaker open AND manually tighten using a torque wrench.

C. Verify MOV breaker open AND manually tighten by hand without using a torque amplifying device.

D. Danger tag MOV breaker open AND manually tighten by hand without using a torque amplifying device.

Answer:

C. Verify MOV breaker open AND manually tighten by hand without using a torque amplifying device.

Notes:

"C" is correct. MOVs inside the Reactor Building are not danger tagged due to the inability to leave these tags inside the building when closing out the building prior to heatup, therefore no danger tags should be used during heatup. Additionally, the applicant should deduce since this is a non-Q MOV that torque limits do not apply and therefore the valve is to be hand tightened without the use of a TAD (torque amplifying device).

"A" is incorrect but plausible in this distractor has the correct method of de-energizing the MOV but incorrect method of tightening valve.

"B" is incorrect but plausible since MOVs are usually danger tagged if they are to be manually operated, but not if they are in the Reactor Building. Non-Q MOVs additionally do not have to be tightened using a torque wrench.

"D" is incorrect since MOVs in Reactor Building are not danger tagged but it has the correct method of tightening the valve and is thus plausible.

This question matches the K/A since this is a situation requiring knolwedge of how to perform a valve lineup using an MOV breaker and how to manually close it.

References:

1015.001, Conduct of Operations

History:

New for 2017 RO Re-exam
Rev. 1, 5/21/17
Swapped A and C positions to make choices short to long.
Editorial changes.

QID: 1084 **Rev:** 2 **Rev Date:** 5/21/17 **Source:** Repeat **Originator:** Cork

TUOI: ASLP-RO-COPD **Objective:** DD **Point Value:** 1

Section: 2.0 **Type:** Generic K&A

System Number: 2.1 **System Title:** Conduct of Operations

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

K/A Number: 2.1.37 **CFR Reference:** 41.1 / 43.6 / 45.6

Tier:	3	**RO Imp:**	4.3	**RO Select:**	Yes	**Difficulty:**	2
Group:		**SRO Imp:**	4.6	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 68 **SRO:**

In accordance with ANO Reactivity Management Program (COPD-030), which of the following activities would require a Reactivity Management Brief prior to performance of the activity?

A. Adding nitrogen to Makeup Tank T-4.

B. Bypassing E-3/4A Feedwater Heaters.

C. Raising seal injection flow rate to RCP P-32A.

D. Adjusting reactive loading on Main Generator.

Answer:

B. Bypassing the E-3/4A Feedwater Heaters.

Notes:

"B" is correct based on Att. 9.3 in COPD-030. Changing Feedwater flow rate or temperature will affect secondary power and thus will affect reactor power.

"A" is incorrect but plausible since the Makeup Tank is part of Makeup and Purification which makes up to the RCS but changing Makeup Tank pressure will not affect reactivity.

"C" is incorrect but plausible since this evolution will increase the amount of fluid going into the RCS, but seal injection is coming from the Makeup Tank so reactivity will not be affected.

"D" is incorrect but plausible as this contains the word "reactive" but changing reactive load will not change secondary power so there is no reactivity affect.

This question matches the K/A since it requires knowledge of what activity requires a RM brief per ANO's procedure for reactivity management.

References:

COPD-030, ANO Reactivity Management Program

History:

New question for 2016 exam
Repeated for 2017 RO Re-exam
Rev. 2, 5/21/17
Swapped A and C to make choices short to long.
Editorial changes.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1082 **Rev:** 1 **Rev Date:** 5/21/17 **Source:** Repeat **Originator:** Cork

TUOI: ASLP-RO-PRCON **Objective:** 1 **Point Value:** 1

Section: 2.0 **Type:** Generic K&A

System Number: 2.2 **System Title:** Equipment Control

Description: Knowledge of the process for making changes to procedures.

K/A Number: 2.2.6 **CFR Reference:** 41.10 / 43.3 / 45.13

Tier: 3 **RO Imp:** 3.0 **RO Select:** Yes **Difficulty:** 2

Group: **SRO Imp:** 3.6 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 69 **SRO:**

You are assigned to the 4th floor as a procedure writer and are making a procedure change.

Which of the following would be regarded as a change to the INTENT of a procedure?

A. Adding text to clarify the purpose of a procedure step.

B. Adding a step to close an open configuration control loop.

C. Deleting a QC hold point in a procedure section for a filter change.

D. Changing the title of a position to correspond to corporate heirarchy.

Answer:

C. Deleting a QC hold point in a procedure section for a filter change.

Notes:

"C" is the correct answer per 1000.006, definition 4.9.2.

"A", "B", and "D" are common procedure changes and thus plausible, but none of these constitute intent changes per 1000.006.

Changed "C" to "B" to change order of correct answer.

This question matches the K/A since it requires the candidate to recall part of the process of making a procedure change: the definition of an intent change.

References:

1000.006, Procedure Control

History:

New question for 2016 exam
Repeated for 2017 RO Re-exam
Rev. 1, 5/21/17
Re-sequenced answers short to long.

QID: 1161 **Rev:** 1 **Rev Date:** 5/21/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-TS **Objective:** 3 **Point Value:** 1

Section: 2.0 **Type:** Generic K/As

System Number: 2.2 **System Title:** Equipment Control

Description: Knowledge of limiting conditions for operations and safety limits.

K/A Number: 2.2.22 **CFR Reference:** 41.5 / 43.2 / 45.2

Tier:	3	**RO Imp:**	4.0	**RO Select:**	Yes	**Difficulty:**	2
Group:		**SRO Imp:**	4.7	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 70 **SRO:**

Which of the following are addressed in Technical Specification Safety Limits?

A. RCS Leakage

B. Quadrant Power Tilt

C. Reactor Building Pressure

D. Fuel Centerline Temperature

Answer:

D. Fuel Centerline Temperature

Notes:

"D" is correct, fuel pin centerline temperature is Safety Limit 2.1.1.1.

"A" is incorrect, but plausible since RCS leakage is indicative of a failure of the RCS fission product barrier but is not addressed in safety limits.

"B" is incorrect, but plausible since there is a Tech Spec LCO for quadrant power tilt and QPT could affect fuel pin centerline temperature but it is not specifically stated in the safety limits.

"C" is incorrect, but plausible since RB pressure is an important parameter with respect to challenges of a fission product barrier, the Reactor Building, but is not specifically addressed in the safety limits.

References:

Technical Specifications, Section 2.0

History:

New for 2017 RO Re-exam
Rev. 1, 5/21/17
Re-sequenced answers short to long.

QID: 0458 **Rev:** 1 **Rev Date:** 5/21/17 **Source:** Bank **Originator:** S.Pullin

TUOI: A1LP-RO-TS **Objective:** 2 **Point Value:** 1

Section: 2 **Type:** Generic Knowledges and Abilities

System Number: 2.1 **System Title:** Conduct of Operations

Description: Ability to determine Technical Specification Mode of Operation.

K/A Number: 2.2.35 **CFR Reference:** 41.7 / 41.10 / 43.2 / 45.13

Tier:	3	**RO Imp:**	3.6	**RO Select:**	Yes	**Difficulty:**	2
Group:		**SRO Imp:**	4.5	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 71 **SRO:**

Which one of the following conditions is required by Unit 1 Technical Specifications in order to consider the reactor in Mode 4?

	Reactivity Condition (Keff)	Average Reactor Coolant Temperature (°F)
A.	> 0.99	280 > Tavg > 200
B.	< 0.99	280 > Tavg > 200
C.	> 0.99	≤ 200
D.	< 0.99	≤ 200

Answer:

B. < 0.99 280 > Tavg > 200

Notes:

"B" is correct, Keff < 0.99 and RCS Tavg is >200°F and <280°F for Mode 4.

"A" is incorrect, this choice is plausible since this has the correct temperature range but the Keff is for Modes 1 or 2.

"C" is incorrect, this choice is plausible since this has the temperature for the next lowest mode and the Keff is for Modes 1 or 2.

"D" is incorrect, this choioe is plausible since this has the proper Keff for Mode 4 but the Tavg associated with Mode 5.

This matches the K/A since it requires knowledge of Tech Spec mode conditions.

References:

Technical Specifications

History:

Direct from regular exambank QID 39.
Selected for use in 2002 SRO exam.
Selected for 2011 RO/SRO Exam.
Selected for 2017 RO Re-exam.
Rev. 1, 5/21/17
Revised question to be a 2x2 formatted to be similar to Mode table in Tech Specs.
Editorial changes.

QID: 1081 **Rev:** 2 **Rev Date:** 5/21/17 **Source:** Repeat **Originator:** Cork

TUOI: ESLP-GET-RWT01.07 **Objective:** 44 **Point Value:** 1

Section: 2.0 **Type:** Generic K&A

System Number: 2.3 **System Title:** Radiation Control

Description: 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.

K/A Number: 2.3.12 **CFR Reference:** 41.12 / 45.9 / 45.10

Tier:	3	**RO Imp:**	3.2	**RO Select:**	Yes	**Difficulty:**	3
Group:		**SRO Imp:**	3.7	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 72 **SRO:**

You have been directed to perform a task in Makeup Tank Room, a Locked High Radiation Area (LHRA).

Dose rates in Makeup Tank Room are 1.1 R/hr.

Which of the following is a requirement per EN-RP-101, Access Control for Radiologically Controlled Areas, SPECIFICALLY for entry into the LHRA?

A. Red trip ticket

B. Double PC garments

C. Continuous RP coverage

D. Approval by on-watch Shift Manager

Answer:

C. Continuous RP coverage

Notes:

"C" is the correct answer per EN-RP-101, continuous RP coverage is required for workers in a field dose rate greater than or equal to 1R/hr which is the definition of an LHRA.

"A" is incorrect but plausible as this is required for HRA (high radiation area) as well as LHRA. The trip ticket with a red border or red background are used by RP but these tickets are not specifically described in a procedure.

"B" is incorrect but plausible, this may be required for highly contaminated areas but is not peculiar to LHRA entry.

"D" is incorrect but plausible as this is required for entry into VHRA (very high radiation area).

This question matches the K/A since it requires the candidate to recall an essential and unique requirement for entry into a locked high radiation area.

Revised C & D, and stem per NRC examiner request. JWC 7/14/16

References:

EN-RP-101, Access Control for Radiologically Controlled Areas

History:

New for 2016 exam
Repeat for 2017 RO Re-exam
Rev. 2, 5/21/17
Re-sequenced choices short to long.

Editorial changes.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1144 **Rev:** 1 **Rev Date:** 5/21/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-EOP06 **Objective:** 14 **Point Value:** 1

Section: 2.0 **Type:** Generic KA's

System Number: 2.3 **System Title:** Radiation Control

Description: Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

K/A Number: 2.3.14 **CFR Reference:** 41.12 / 43.4 / 45.10

Tier: 3 **RO Imp:** 3.4 **RO Select:** Yes **Difficulty:** 2

Group: **SRO Imp:** 3.8 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 73 **SRO:**

Given:

* Unit 1 shutting down due to "A" SG tube leak

* Control of Secondary System Contamination (1203.014), in progress

Which Condensate Polishers are preferred to be left in service and why these specific two?

A. E & F, to limit contamination to two polishers

B. C & D, to limit contamination to two polishers

C. E & F, to reduce personnel dose rates

D. C & D, to reduce personnel dose rates

Answer:

D. C & D, to reduce personnel dose rates

Notes:

"D" is correct, two polishers are left in service, C & D are preferred as an ALARA practice since they are in the middle which will increase distance from the polishers to the operator at the polisher panel and increase distance from personnel in the train bay.

"A" is incorrect but plausible since reducing the number of polishers to two is to limit contamination but these are the wrong two and the incorrect reason for the specific two polishers.

"B" is incorrect but plausible since these are the correct two polishers and the reason for reducing the number of polishers to two is to limit contamination but the reason C & D are used is due to their central location.

"C" is incorrect, this is plausible since using E & F as the in-service polishers will reduce dose rates to the operator at the polisher panel but will raise dose rates for personnel in the train bay.

This question matches the K/A since a tube leak is an abnormal condition which introduces radiation hazards, and the question requires the knowledge of why a particular action is taken, i.e., to reduce personnel exposure to a radiation hazard.

References:

1203.014, Control of Secondary System Contamination

History:

New for 2017 RO Re-exam
Rev. 1, 5/21/17
Editorial changes.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0128 **Rev:** 4 **Rev Date:** 6/8/17 **Source:** Bank **Originator:** JCork

TUOI: ASLP-RO-EPLAN **Objective:** 2 **Point Value:** 1

Section: 2.0 **Type:** Generic K/As

System Number: 2.4 **System Title:** Emergency Procedures/Plan

Description: Knowledge of the emergency plan.

K/A Number: 2.4.29 **CFR Reference:** 41.10 / 43.5 / 45.11

Tier:	3	**RO Imp:**	3.1	**RO Select:**	Yes	**Difficulty:**	2
Group:		**SRO Imp:**	4.0	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 74 **SRO:**

Which of the following would be classified as a loss, or a potential loss, of a fission product barrier per Emergency Action Level Classification (1903.010)?

A. RCS leakage indicates greater than 50 gpm with letdown isolated.

B. RCS pressure 2450 psig with ERV controlling pressure.

C. CNTMT pressure indicates 20 psia.

D. Engineering Assessment of core damage indicates 0.5% fuel cladding failure.

Answer:

A. RCS leakage indicates greater than 50 gpm with letdown isolated.

Notes:

"A" is correct, leakage is >50 gpm with letdown isolated is a potential loss of the RCS barrier per 1903.010.

"B" is incorrect, this is a challenge to, but not a loss of, the RCS pressure boundary.

"C" is incorrect, CNTMT pressure given is greater than ESAS actuation setpoint but less than RB Spray actuation setpoint of 44.7 psia, which makes this NOT a challenge to the RB boundary.

"D" is incorrect, failed fuel must reach 5.0% to be a breach.

References:

1903.010, Emergency Action Level Classification
ASLP-RO-EPLAN, enabling objective EO-2

History:

Developed for 1998 SRO exam.
Revised after 9/98 exam analysis review.
Used in 2001 SRO Exam.
Modified for 2005 RO exam as replacement question.
Selected for 2017 RO Re-exam.
Rev. 3, 5/21/17
Changed C parameter to 20 psia.
Changed D parameter to 0.5 %.
Editorial changes.
Rev. 4, 6/8/17
Added CFR number.

QID: 1143 **Rev:** 1 **Rev Date:** 5/21/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-EOP **Objective:** **Point Value:** 1

Section: 2.0 **Type:** Generic KA's

System Number: 2.4 **System Title:** Emergency Procedures/Plan

Description: Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, and severe accident management guidelines.

K/A Number: 2.4.16 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier: 3 **RO Imp:** 3.5 **RO Select:** Yes **Difficulty:** 2

Group: **SRO Imp:** 4.4 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 75 **SRO:**

What is the only EOP which may be directly entered from an AOP without first entering Reactor Trip (1202.001)?

A. ESAS (1202.010)

B. Tube Rupture (1202.006)

C. Degraded Power (1202.007)

D. Loss of Subcooling Margin (1202.002)

Answer:

B. Tube Rupture (1202.006)

Notes:

"B" is correct. Per 1015.043, ANO-1 EOP/AOP User Guide, 1202.006 Tube Rupture may be entered directly from AOP 1203.023 Small Generator Tube Leak without first entering 1202.001 so that off-site releases may be limited by performing a controlled shutdown in 1202.006.

"A" is incorrect, yet plausible since this EOP's entry conditions are obvious from the ESAS annunicators, yet 1202.001 Reactor Trip is still entered first.

"C" is incorrect, yet plausible since this EOP contains several sections designed to mitigate Loss of Subcooling Margin, Overcooling, and Overheating. It's entry conditions are also quite obvious, yet 1202.001 Reactor Trip is still entered first.

"D" is incorrect, yet plausible since this EOP has the highest priority per the EOP User's Guide. Yet it is still entered only after diagnosis is made in 1202.001. It is even entered from 1202.006, Tube Rupture, if problems other than a tube rupture are diagnosed.

This question matches the K/A since it requires knowledge of EOP hierarchy and how certain AOPs are used with the EOPs.

References:

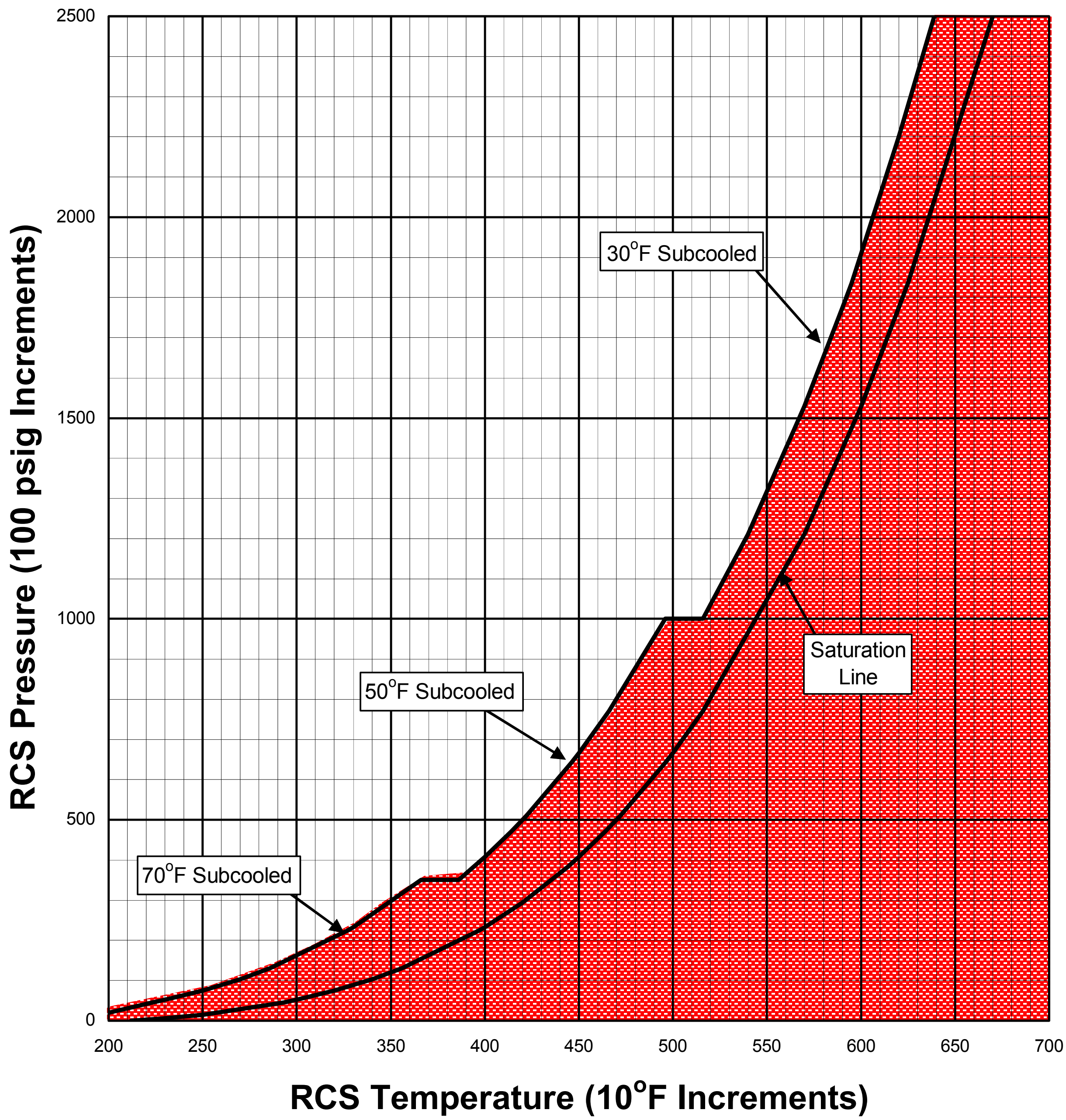
1015.043, ANO-1 EOP/AOP User Guide

History:

New question for 2017 RO Re-exam
Rev. 1, 5/21/17
Swapped positions of A and D to make choices short to long.
Editiorial changes.

2017
ANO UNIT 1
NRC INITIAL LICENSE EXAMINATION REFERENCE MATERIAL
<u>RO</u>

FIGURE 1
Saturation and Adequate SCM



RCS Pressure	Adequate SCM
>1000 psig	≥30°F
350 to 1000 psig	≥50°F
<350 psig	≥70°F

FIGURE 2
SG Pressure vs T-sat

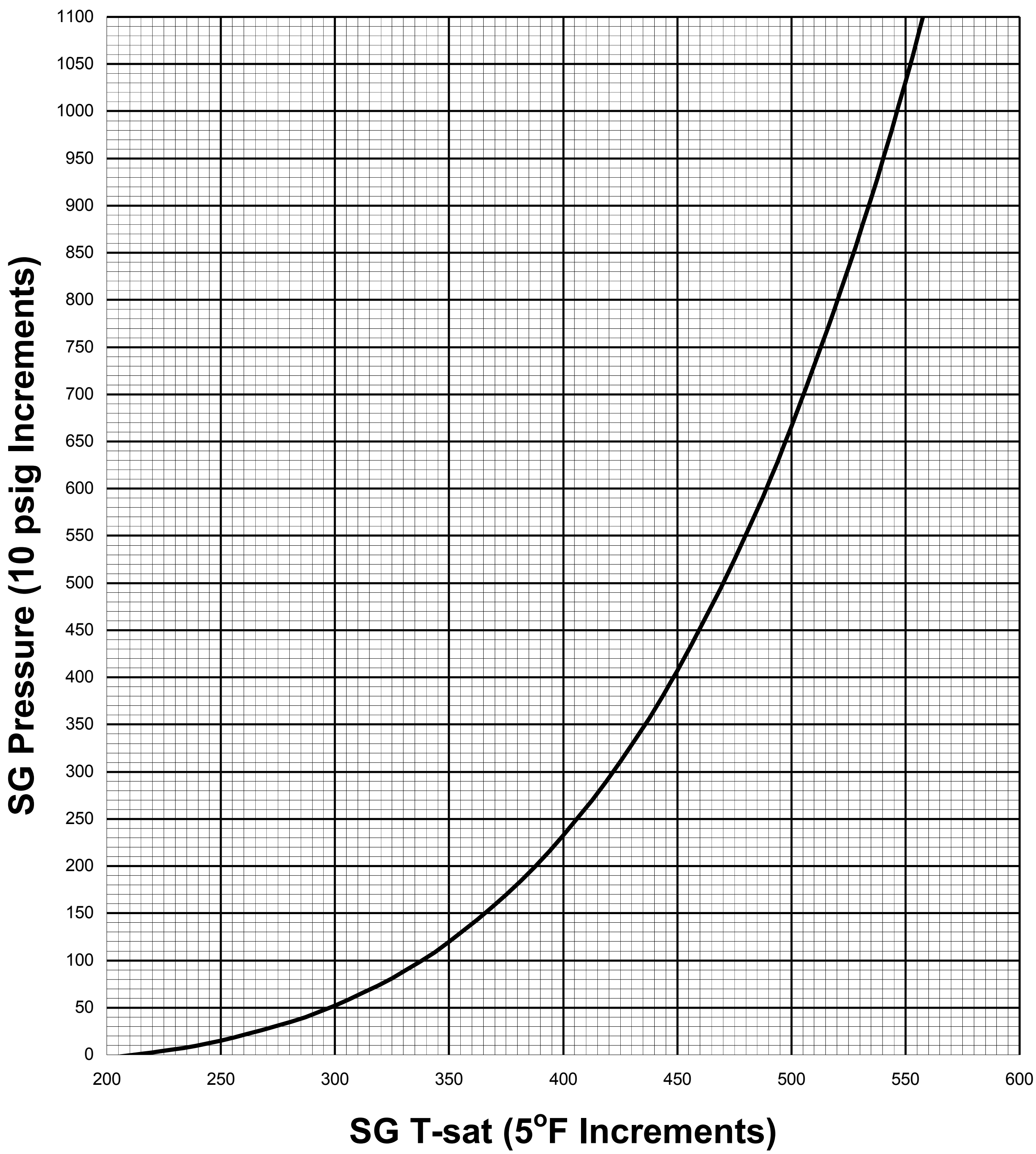


FIGURE 3
RCS Pressure vs Temperature Limits

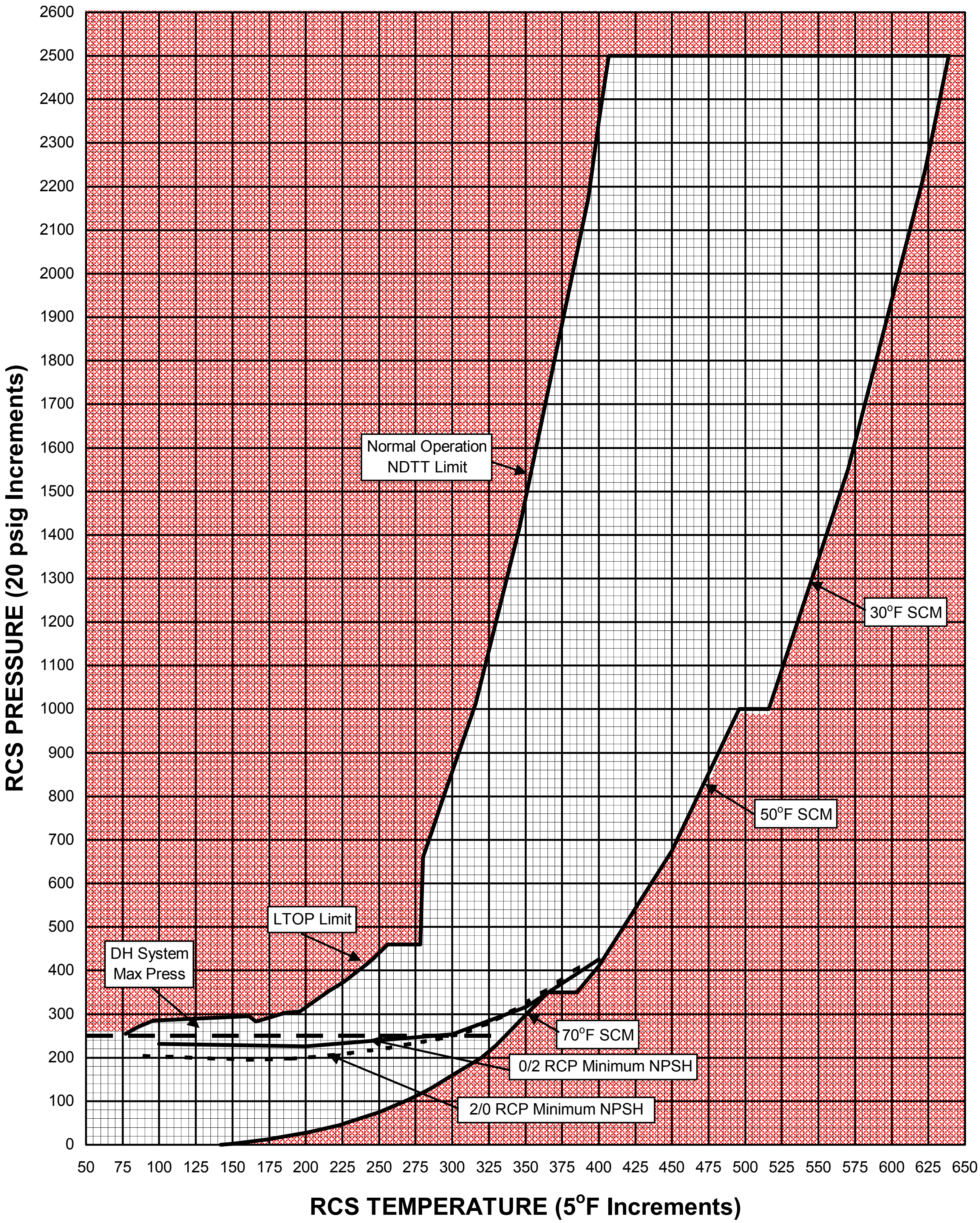


FIGURE 4
Core Exit Thermocouple for Inadequate Core Cooling

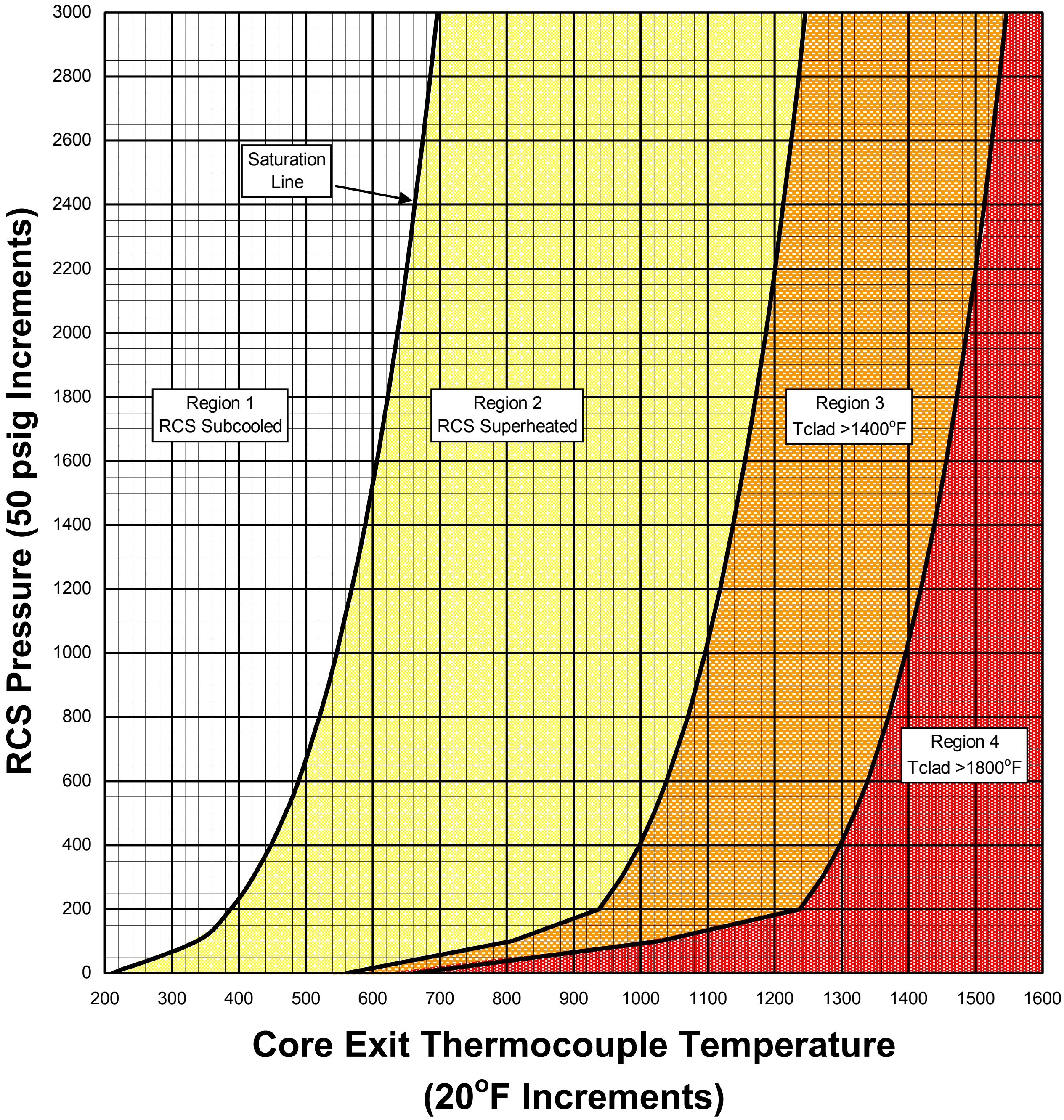


FIGURE 5

SG Pressure to Establish 40° to 60°F Primary to Secondary ΔT

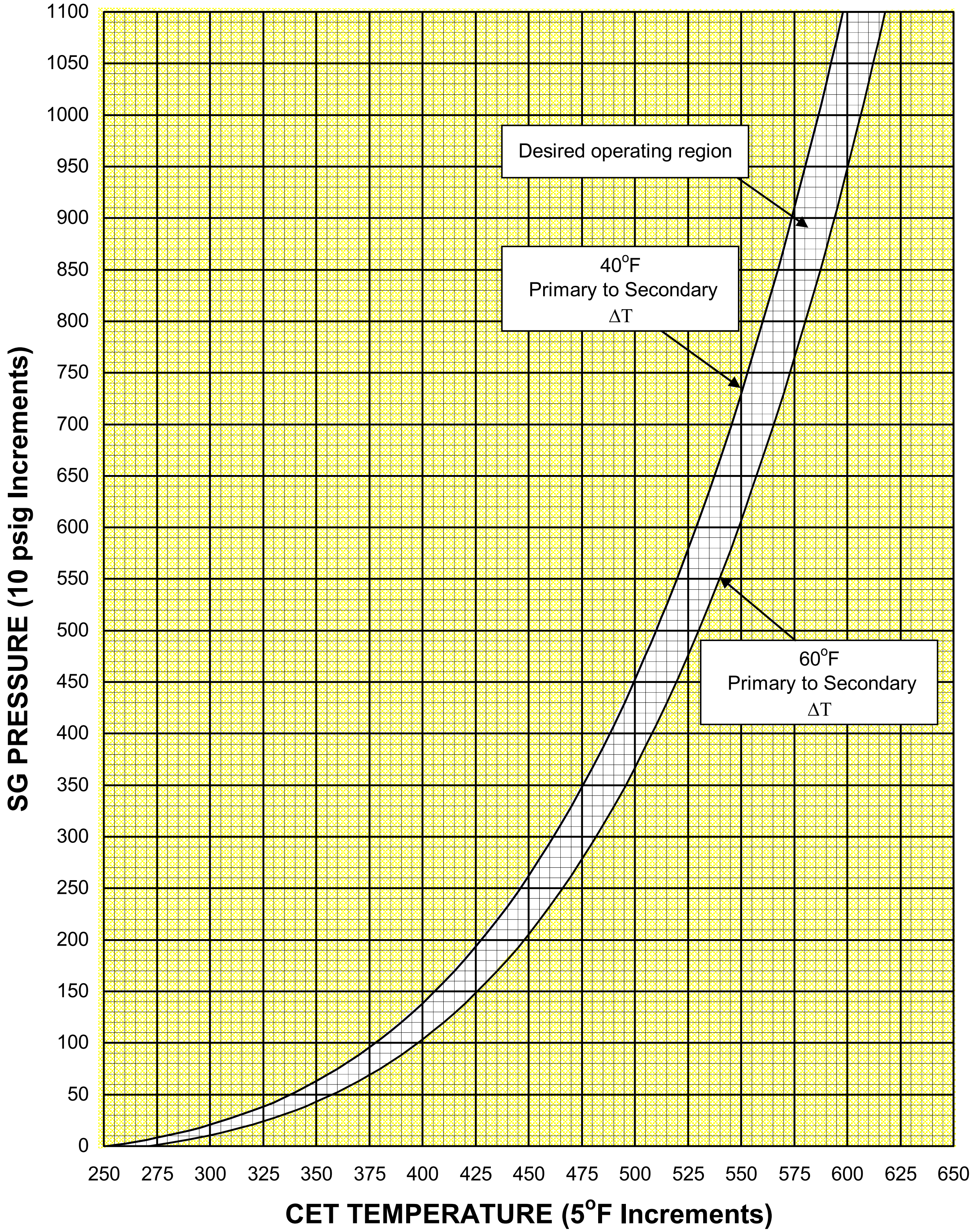
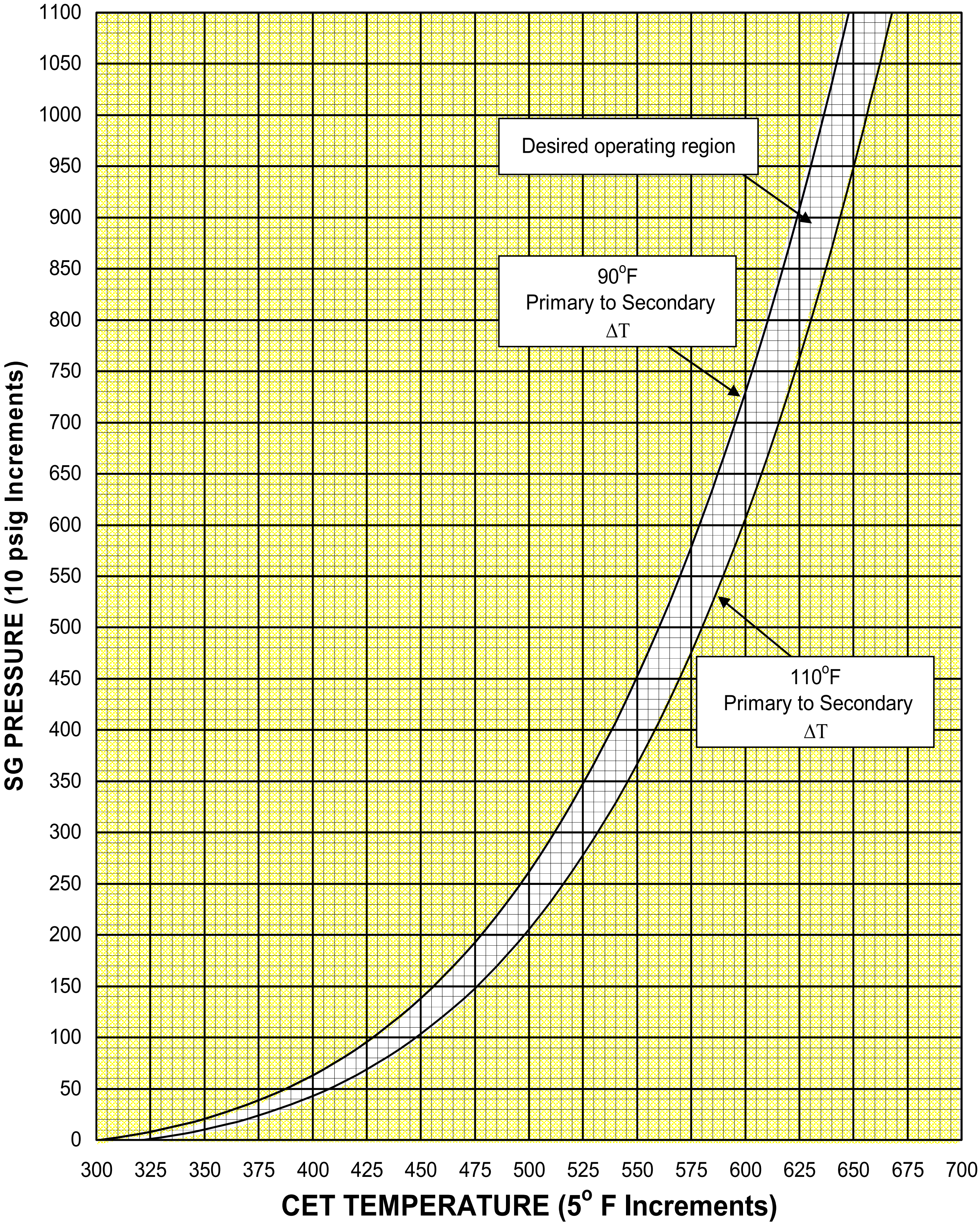


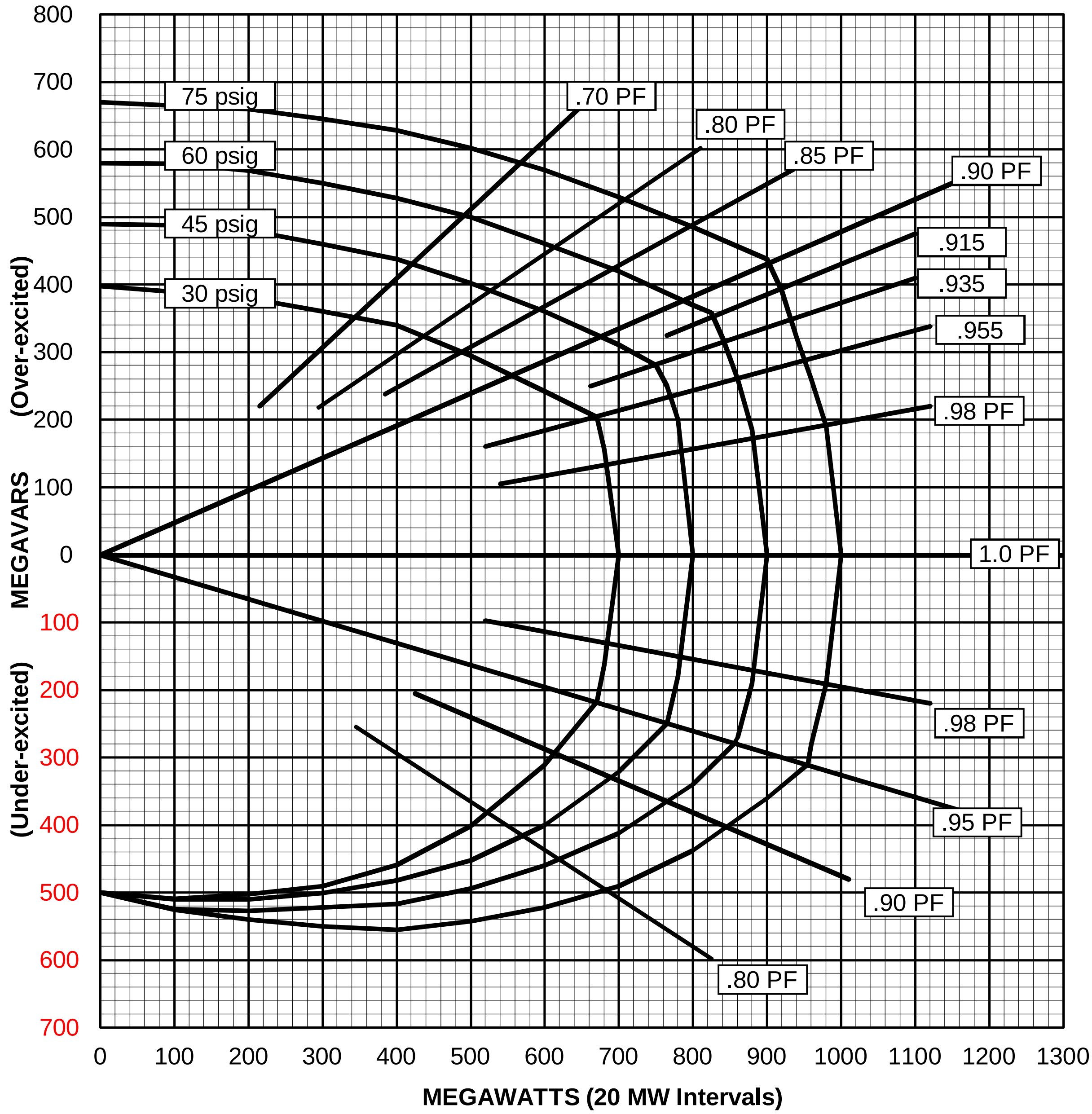
FIGURE 6

SG Pressure to Establish 90° to 110°F Primary to Secondary ΔT



ATTACHMENT N

Hydrogen Inner-Cooled Turbine Generator Calculated Capability Curve at Rated Voltage



Basis	1002.6 MVA 0.90 PF 22 KV 0.58 SCR	3 Phase 60 Hz 1800 RPM 75 PSIG

QID: 1119 **Rev:** 3 **Rev Date:** 5/16/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-MU **Objective:** 10 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 022 **System Title:** Loss of Reactor Coolant Makeup

Description: Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup: Whether charging line leak exsts.

K/A Number: AA2.01 **CFR Reference:** 43.5

Tier: 1 **RO Imp:** **RO Select:** No **Difficulty:** 3

Group: 1 **SRO Imp:** 3.8 **SRO Select:** Yes **Taxonomy:** H

Question: **RO:** **SRO:** 76

Given:
* Unit 1 90% power
* Pressurizer level 215" and dropping
* Makeup Tank level 78" and dropping
* Total Seal Injection Flow 25 gpm and lowering
* Seal Bleed Off Temperatures 130 °F and slowly rising
* MU flow 30 gpm and lowering
* AREA MONITOR RADIATION HI (K10-B1) alarms due to RE-8011, Makeup Pump Area

Which of the following is the correct procedure and mitigating action?

A. Loss of Reactor Coolant Makeup (1203.026);
Stop the running HPI pump.

B. Loss of Reactor Coolant Makeup (1203.026);
Isolate RCP Seal Bleedoff for all RCPs.

C. Reactor Coolant Pump and Motor Emergencies (1203.031);
Stop the running HPI pump.

D. Reactor Coolant Pump and Motor Emergencies (1203.031);
Isolate RCP Seal Bleedoff for all RCPs.

Answer:

A. Loss of Reactor Coolant Makeup (1203.026);
Stop the running HPI pump

Notes:

For the given conditions the event in progress is a line break downstream of the HPI pump but prior to the flow branching off to Seal Injection and Makeup flow as indicated by the radiation monitor in alarm and lowering Seal Injection and Makeup flow with Pzr level dropping.

"A" is Correct, seal injection flow and makeup flow have lowered due to a leak and the (RCS) leak has caused the area radiation levels to rise enough to cause an alarm. With the leak downstream of the HPI Pump, the procedure directs stopping the pump to allow for isolating the leak.

"B" is wrong but plausible since it is the correct procedure and a lowering flow might indicate a pump issue, if a HPI pump is stopped then isolating Seal Bleedoff is a correct action.

"C" is wrong but plausible since there are RCP seal cooling issues indicated RCP and Motor Emergencies is plausible a clogged seal injection filter would cause a lowering seal injection flow and stopping the HPI pump is the correct action.

"D" is wrong but a plausible procedure as stated in "C". Seal bleedoff isolation is a step found within the RCP and Motor Emergency procedure.

This question matches the K/A since conditions are given for a charging line leak and candidate must

differentiate between RCP seal problem and charging line leak.

References:

1203.026, Loss of Reactor Coolant Makeup
1203.031, Reactor Coolant Pump and Motor Emergencies

History:

New question for 2017 SRO Re-exam.
Rev 1
1. Added location of rad monitor to stem
2. Removed higher flow contradiction/ flow will be lowering due to break location
3, Removed reference to P-32B
4. Total number of bullets lowered and all together

Rev. 2 5/4/17
1. Added Seal Bleed Off Temperature - 130 F and rising to the given information to add to the plausability of the distractors
2. Removed the parenthesis from (Makeup Pump Area) since it is a formal noun name for the monitor
3. Additional supporting information / explanation to the notes for distractors
4. Sump level increasing not required but can be added if you like
5. Question changed to which procedure and action required. Applicant must determine that a leak exists to select the appropriate procedure and action.
6. Caps removed
7. Explanation provided.
8. Removed reference to ACAs 10. NO total flow alarm
9. 2x2 format
10. Changed wording

Rev 3, 5/16/17
Editorial changes
Added "Bleedoff" back to B distractor
Removed "ATC announces that" from 2nd condition

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1165 **Rev:** 2 **Rev Date:** 5/16/17 **Source:** New **Originator:** Burton

TUOI: A1LP-RO-AOP **Objective:** 2 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 027 **System Title:** Pressurizer Pressure Control System Malfunction

Description: Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

K/A Number: 2.2.25 **CFR Reference:** 43.5

Tier: 1 **RO Imp:** 3.2 **RO Select:** No **Difficulty:** 3

Group: 1 **SRO Imp:** 4.2 **SRO Select:** Yes **Taxonomy:** F

Question: **RO:** **SRO:** 77

Given:
* Unit 1 trips from 100% power due to a loss of switchyard from a severe storm
* Pressurizer Spray valve fails open
* ATC operator was able to close Spray isolation valve and stop RCS pressure lowering trend
* Annunciator PZR HEATER GROUND FAULT (K09-E3) alarms
* RCS pressure response abnormally slow with available Pressurizer heaters energized
* Efforts to restore ES backed pressurizer heaters have been unsuccessful.

An alternate way to maintain subcooling margin listed in TS Bases 3.4.9 is to _________ ?

A. Backfeed bus A4 to A2 to energize additional pressurizer heaters

B. Align Unit 2 Charging pumps for makeup to Unit 1 pressurizer

C. Use of available HPI pumps to makeup to pressurizer

D. Connect AACG (2K-9) diesel power to A1 bus to energize additional pressurizer heaters

Answer:

C. Use of available HPI pumps to makeup to pressurizer

Notes:

"C" is correct per TS Bases 3.4.9 which states RCS pressure control via high head pressure injection pumps is an alternate method of maintaining subcooling.

"A" is the incorrect but plausible since there are non-vital powered PZR heaters on A2, however, in Degraded Power the CRS would not direct backfeeding from A4 for the sole purpose of restoring additional PZR heaters.

"B" is incorrect but plausible since the installation of FLEX modifications installed this capability, but this is only used during an extended Blackout.

"D" is incorrect but plausible since there are non-vital powered PZR heaters on A2, however, in Degraded Power the CRS would not direct aligning the AACG for the sole purpose of restoring additional PZR heaters.

Matches the KA since the applicant must recall from TS Bases available means to restore SCM.

References:

T.S. 3.4.9 Bases
1202.007, Degraded Power
1FSG-001, Long Term RCS Inventory Control
STM 1-03, Reactor Coolant System

History:

New for the SRO 2017 re-exam
Rev. 1 5/4/17

1) Modified the wording of the question as recommended by the NRC
2)Added a second part to Answer A to make it look more like the distractors
3)Corrected a typo in Distractor D
4)Replaced Distractor C
Rev. 2, 5/16/17
Reverted back to Distractor C to use FLEX
Deleted bullet about Maintenance performing 1307.009
Made C the correct answer to break up string of A's.
Editorial changes

QID: 1112 **Rev:** 3 **Rev Date:** 5/16/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-EOP03 **Objective:** 12 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 040 **System Title:** Steam Line Rupture

Description: Ability to determine and interpret the following as they apply to the Steam Line Rupture: Difference between steam line rupture and LOCA

K/A Number: AA2.03 **CFR Reference:** 43.5

Tier:	1	**RO Imp:**		**RO Select:**	No	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.7	**SRO Select:**	Yes	**Taxonomy:**	H

Question: **RO:** **SRO:** 78

Given:
* Unit 1 tripped from 100% power.

CBOT reports following critical parameters:
* RCS pressure 1620 psig and lowering slowly
* Pressurizer level 60 inches and lowering slowly
* Thot 545 °F and lowering slowly
* "A" SG pressure 850 psig and lowering slowly
* "B" SB pressure 1000 psig and steady
* RB pressure 15 psia and rising

Which of the following is the correct procedure and a mitigating action for these conditions?

A. Overcooling (1202.003); Isolate Letdown.

B. Overcooling (1202.003); Trip both MFW pumps.

C. Loss of Subcooled Margin (1202.002); Isolate Letdown.

D. Loss of Subcooled Margin (1202.002); Trip both MFW pumps.

Answer:

A. Overcooling (1202.003); Isolate Letdown.

Notes:

The applicant is given conditions of lowering RCS pressure and level and rising RB pressure which can be either an
RCS break or Steam line break. Now they add in the lowering RCS temperature and SG pressures to determine that
a small steam line break has occurred and Overcooling is the correct procedure. Once they identify the correct procedure then determine which logic step within Overcooling applies.

"A" is correct, Overcooling is entered based on A SG pressure <900 psig, and Isolating Letdown is directed when they transition to RT-2.

"B" is wrong but plausible since it is the correct procedure and a contingency action related to RB pressure to stop both MFW pumps can be found within the procedure .

"C" is wrong but plausible because for the given RCS conditions a LOCA could be occurring which can result in a loss of subcooled margin, however current SCM is 50 F which is adequate for the given RCS pressure. Isolating Letdown is plausible since this action is located in both procedures.

"D" is wrong but plausible because for the given RCS conditions a LOCA could be occurring which can result in a loss of subcooled margin, however current SCM is 50 F which is adequate for the given RCS pressure. Stopping both MWP is plausible since this action is contained within both procedures.

This question matches the K/A since it requires the applicant to evaluate conditions and choose between a

steam line break procedure (Overcooling) and a LOCA procedure (Loss of Subcooling)

References:

1202.003, Overcooling
1202.002, Loss of Subcooled Margin
1202.013, EOP Figure 1

History:

New question for 2017 SRO Re-exam
Rev 1
1. Choosing the correct procedure demonstrates LOCA/MSLB
2. Changed temp to T-hot
3. Better explanations provided
4. Referenced this step in support document
5, EOP figure 1 defines adequate SCM (provided)
6. Question has been modified, SG pressure and trend is needed.
7. Reactor is tripped
8. Changed dropping to lowering for entire exam

Rev 2 5/4/17
1. Added SCM value to the notes for C and D
2. Changed the stem by adding an "a" before mitigating actions

Rev. 3, 5/16/17
Corrected explanation for A being correct
Editorial changes
Added EOP Figure 1 to references

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0840 **Rev:** 2 **Rev Date:** 5/16/17 **Source:** Bank **Originator:** Cork

TUOI: A1LP-RO-TS **Objective:** 5 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 058 **System Title:** Loss of DC Power

Description: Ability to determine the operability and/or availability of safety related equipment.

K/A Number: 2.2.37 **CFR Reference:** 43.5

Tier:	1	**RO Imp:**	3.6	**RO Select:**	No	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.6	**SRO Select:**	Yes	**Taxonomy:**	F

Question: **RO:** **SRO:** 79

Unit 1 has entered Technical Specification 3.8.4 due to a loss of one required DC electrical power subsystem.

In addition to the Battery, Battery Charger and associated cables, which of the following component(s) are considered a part of the DC subsystem per Tech Spec Bases?

1. 125V DC panel
2. Inverter
3. Static Switch

A. 1 only

B. 2 only

C. 1 and 3 only

D. 2 and 3 only

Answer:

A. 1 only

Notes:

"A" is correct per the TS bases for 3.8.4.
The 125 VDC electrical power system consists of two independent redundant safety related subsystems. Each subsystem consists of one 125 VDC battery, the associated battery charger and all the associated control equipment and interconnecting cables, which includes the DC panel.

"B", "C" and "D" contain incorrect components which are inter-related to the DC system and are thus plausible, and those components are actually part of the 120v Vital AC system. Inverter is incorrect in B and D. Static switch is incorrect in C and D.

This matches the KA because the applicant must identify the components that must be Operable to support the
Operability of DC sub-system.

References:

Technical Specifications 3.8.4 and Bases

History:

New for 2011 SRO Exam.
Selected for 2017 SRO Re-take exam.
Modified the stem changing from 7 choices to 3 to components to satisfy the Bases.

Rev 1 5/4/17
1. Made minor grammatical changes based on NRC comments

Rev 2, 5/16/17

Editorial changes
Corrected explanatory notes

QID: 1114 **Rev:** 3 **Rev Date:** 5/16/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-ALOIA **Objective:** 2 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 065 **System Title:** Loss of Instrument Air

Description: Ability to determine and interpret the following as they apply to the Loss of Instrument Air: When to commence plant shutdown if instrument air pressure is decreasing.

K/A Number: AA2.05 **CFR Reference:** 43.5

Tier:	1	**RO Imp:**		**RO Select:**	No	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.1	**SRO Select:**	Yes	**Taxonomy:**	H

Question: **RO:** **SRO:** 80

Given:
* Unit 1 100% power
* INST AIR HEADER PRESS LO (K12-B3) alarms
* Loud hissing sounds on mezzanine in Turbine Building
* Loss of Instrument Air (1203.024) is entered
* Pressurizer level 175" and lowering slowly
* Instrument Air pressure 55 psig and lowering slowly

Which of the following is the correct mitigating action and procedure?

A. Trip the reactor and perform Reactor Trip (1202.001).

B. Commence plant shutdown at ≥ 10% per minute using Rapid Plant Shutdown (1203.045).

C. Commence plant shutdown at ≤ 5% per minute using Power Reduction and Plant Shutdown (1102.016).

D, Take manual control of Pressurizer level using Makeup and Purification System Operation (1104.002)

Answer:

B. Commence plant shutdown at ≥ 10% per minute using Rapid Plant Shutdown (1203.045).

Notes:

"B" is correct, step 14 of 1203.024 contingency action will direct CRS to step 18 which directs a plant shutdown per Rapid Plant Shutdown AOP at ≥10% per minute. FYI, low instrument air pressure alarm comes in at 75 psig.

"A" is wrong but plausible step 25 directs a Reactor Trip when Instrument Air pressure drops to 35 psig as checked in step 19 with a contingency to transition to step 25.

"C" is wrong but plausible, for most plant shutdowns the normal shutdown procedure is used but if Instrument Air pressure drops to less than 60 psig, then Rapid Plant Shutdown is used.

"D" is wrong but plausible because this procedure refers to pressurizer level and has multiple contingency actions in regards to level, this requires the applicant to consider this condition and if an action should be taken.

This question matches the K/A since conditions are given for a loss of instrument air which requires commencement of a plant shutdown.

References:

1203.024, Loss of Instrument Air
1102.016, Power Reduction and Plant Shutdown

History:

New question for 2017 SRO Re-exam
Rev. 1
1. Removed "all previous action completed" statement
2. Changed distractor" so x-connect should not be an issue
3 4 & 5. Provided steps as requested
6. 1102.016 is provided
Rev. 2 5/4/17
1. Made grammatical changes based on NRC comment
Rev. 2 5/4/17
1. Made grammatical changes based on NRC comment
7. Changed distractor "D"
Rev. 3, 5/16/17
Editorial changes

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1166 **Rev:** 2 **Rev Date:** 5/16/17 **Source:** Modified **Originator:** Cork

TUOI: A1LP-RO-AOP **Objective:** 3 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 077 **System Title:** Generator Voltage and Electric Grid Disturbances

Description: Knowledge of abnormal condition procedures.

K/A Number: 2.4.11 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier: 1 **RO Imp:** **RO Select:** No **Difficulty:** 4

Group: 1 **SRO Imp:** 4.2 **SRO Select:** Yes **Taxonomy:** H

Question: **RO:** **SRO:** 81

Given:
* Unit 1 in Mode 1.
* Ambient outside temperature 103 °F
* CBOT reports:
    * A3 and A4 bus voltages ~3750 volts
    * SU1 Transformer voltage 22.1 KV
    * SU2 Transformer voltage 160KV
* Dispatcher reports:
    * Voltage regulators are in service and working properly
    * A major capacitor bank is out of service
    * Grid disturbances are causing grid voltage and frequency to oscillate

The above conditions have not improved after several hours

Procedure 1203.037, Abnormal ES Bus Voltage and Degraded Offsite Power, has been entered.

Which of the following procedure sections should be transitioned to and which procedurally required actions are warranted for the above conditions?

A. Section 3, Offsite Voltage Abnormal;
Start and parallel one DG to grid then separate associated ES bus by opening its feeder breaker.

B. Section 1, ES Bus Voltage Low;
Start and parallel one DG to grid then separate associated ES bus by opening its feeder breaker.

C. Section 3, Offsite Voltage Abnormal;
Start one DG, de-energize associated ES bus by opening its feeder breaker, then verify DG output breaker closes.

D. Section 1, ES Bus Voltage Low;
Start one DG, de-energize associated ES bus by opening its feeder breaker, then verify DG output breaker closes.

Answer:

D. Section 1, ES Bus Voltage Low;
Start one DG, de-energize associated ES bus by opening its feeder breaker, then verify DG output breaker closes.

Notes:

"D" is correct; bus voltages are low but not low enough to autostart the DGs, and since grid disturbances are occuring per 1203.037 section 1 the ES bus should be de-energized to allow the DG to automatically re-energize it.

"B" is incorrect, plausible as this would be the correct answer if the grid was stable but grid disturbances are occuring and an EDG should not be paralleled with an unstable grid since this could cause damage to the EDG.

"A" is incorrect since Section 3 does not contain this action, instead major loads are secured to reduce voltage but offsite voltages are not low enough to require this section to be used. Section 4, Offsite Frequency Low, contains this action. "A" contains the correct action but the wrong procedure section.

"C" is incorrect since Section 3 does not contain this action, instead major loads are secured to reduce voltage but offsite voltages are not low enough to require this section to be used. Section 4, Offsite Frequency Low, contains this action. "C" contains the wrong procedure section and the wrong action but completes the 2x2 format.

Modified QID 1047 by changing the last bullet from "No grid disturbances are expected." to grid disturbances are occuring. This changes the correct answer from "C" to "D".

This question matches the K/A since a grid disturbance is ongoing and it requires knowledge of the proper action to take per the proper Abnormal Operating Procedure (1203.037).

References:

1203.037, Abnormal ES Bus Voltage and Degraded Offsite Power

History:

Modified QID 1047 for 2017 SRO Re-exam
Rev. 1 5/4/17
1. Broke out the CBOT and Dispatcher reports into bulleted items
2. Deleted unneccessary words from the answer choices based on NRC comment
Rev. 2, 5/16/17
Editorial changes.
Changed status from New to Modified due to statement in Notes section.

QID: 1115 **Rev:** 3 **Rev Date:** 5/16/17 **Source:** New **Originator:** Cork

TUOI: ASCBT-EP-A0081 **Objective:** 2 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 036 **System Title:** Fuel Handling Accident

Description: Ability to determine and interpret the following as they apply to the Fuel Handling Incidents: Magnitude of potential radioactive release

K/A Number: AA2.03 **CFR Reference:** 43.4

Tier:	1	**RO Imp:**		**RO Select:**	No	**Difficulty:**	2
Group:	2	**SRO Imp:**	4.2	**SRO Select:**	Yes	**Taxonomy:**	H

Question: **RO:** **SRO:** 82

*****************REFERENCE PROVIDED************************************

Given:
* Unit 1 in Refueling Outage, core reload in progress
* Main Fuel Bridge reports a spent fuel assembly was dropped
* Spent fuel assembly confirmed to be damaged
* RE-8017, Fuel Handling Area, in alarm at 0.5 R/hr and rising slowly
* RX-9820, Containment Purge (SPING Channel 7), in alarm for 17 minutes at 1.25E -2 µCi/cc and rising slowly

The correct EAL classification is _________ and the order in which authorities will be notified is __________.

A. Unusual Event; NRC then State and Local Authorities

B. Alert; NRC then State and Local Authorities

C. Unusual Event; State and Local Authorities then NRC

D. Alert; State and Local Authorities then NRC

Answer:

D. Alert; State and Local Authorities then NRC

Notes:

"D" is correct, with a damaged fuel assembly and RE-8017 in alarm the EAL is Alert (AA2 in Abnormal Radiation Levels tab). There are no spurious or input failure related alarms associated with the Rad monitors. With fuel damage and a rising trend on the monitors the SM can determine that the readings are valid. The order given for notifications is correct per 1903.011-Y.

"A" is wrong but plausible if candidate does not read all of AU2 in the Abnormal Radiation Levels tab and merely uses the fact that RE-8017 is in alarm as justification for the Unusual Event. The order is the opposite of the correct order in 1903.011-Y.

"B" is wrong but plausible since this is the correct EAL classification. The order given is the opposite of the correct order in 1903.011-Y.

"C" is wrong but plausible if candidate does not read all of AU2 in the Abnormal Radiological Effluents tab and merely uses the fact that RX-9820 is in alarm as justification for the Unusual Event. The order given is correct.

This question matches the K/A since a fuel handling accident has occurred and the magnitude of the potential radioactive release determines the EAL classification.

References:

1903.010, Emergency Action Level Classification
1903.011-Y, Emergency Class Initial Notification Message

1903.010, EAL Classification, pages 18-68 and 74-180 (EAL tabs plus bases, except Tab E) will be provided as an applicant reference.

History:

New question for 2017 SRO Re-exam
Rev 1
1. With fuel damage and a rising trend on the monitors the SM can determine that the readings are valid. The order is correct per 1903.011-Y.
2. LOD and Taxonomy revised
4, Trends provided
5. SPING 7 needs a time element for declaration
7/8. Changed wording as suggested
9. Removed time element since it was common and only requesting correct order of notification
Rev. 2, 5/4/17 (NRC comments)
1. Distractors A and B, corrected typo, changed State or Local to State and Local.
2. Revised References field to indicate what is supplied vs. what is not supplied.
3. Made various changes per comments under "Also".
4. Corrected explanation for correct answer D and distractor C to refer to Abnormal Radiation Levels tab.
Rev. 3, 5/16/17
Editorial changes.
Changed RX-9820 reading to be just above AU1 criteria in Abnormal Radiological Effluents tab.
Deleted "into the core" from 2nd bullet since it didn't matter.
Corrected explanations in Notes section.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1116 **Rev:** 1 **Rev Date:** 5/16/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-AOP **Objective:** 3 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 068 **System Title:** Control Room Evacuation

Description: Ability to perform specific system and integrated plant procedures during all modes of plant operation.

K/A Number: 2.1.23 **CFR Reference:** 43.5

Tier: 1 **RO Imp:** **RO Select:** No **Difficulty:** 2

Group: 2 **SRO Imp:** 4.4 **SRO Select:** Yes **Taxonomy:** H

Question: **RO:** **SRO:** 83

Given:
* Unit 1 100% power
* Unit 2 has a fire in their Control Room
* Heavy smoke has accumulated in the Unit 1 Control Room

CRS will enter ______________ and direct the CBOT to control SG pressures
950 to 1020 psig using _______________.

A. Remote Shutdown (1203.029);
Turbine Bypass Valves

B. Alternate Shutdown (1203.002);
Turbine Bypass Valves

C. Remote Shutdown (1203.029);
Atmospheric Dump Valves

D. Alternate Shutdown (1203.002);
Atmospheric Dump Valves

Answer:

A. Remote Shutdown (1203.029);
Turbine Bypass Valves

Notes:

When conditions require the Control to be evacuated Remote Shutdown will be entered except in the case of a Fire
in the CR or Cable Spreading Room in which case Alternate Shutdown will be entered. Remote Shutdown controls
SG pressure with Turbine Bypass valves while Alternate Shutdown directs the local manual use of ADVs

"A" is correct since a fire is not forcing evacuation of the Control Room and the Turbine Bypass Valves will be used to control SG pressures.

"B" is wrong but plausible because Alternate Shutdown is a CR evacuation procedure and correctly identifies the use Turbine Bypass Valves

"C" is wrong but plausible because it identifies the correct procedure and local manual operation of the ADVs is available.

"D" is wrong but plausible because Alternate Shutdown is a CR evacuation procedure and local manual operation of the ADVs is available.

Matches the KA because these are CR evacuation procedures and demonstrates the ability to use of plant and system
procedures from Modes 1-3.

References:

1203.029, Remote Shutdown
1203.002, Alternate Shutdown

History:

New question for 2017 SRO Re-exam
Rev. 1, 5/16/17
Editorial changes only

QID: 1167 **Rev:** 2 **Rev Date:** 5/16/17 **Source:** New **Originator:** Burton

TUOI: ASLP-RO-EPLAN **Objective:** 6 **Point Value:** 1

Section: 4.3 **Type:** B&W EPEs

System Number: E08 **System Title:** LOCA Cooldown

Description: Knowledge of the emergency action level thresholds and classifications.

K/A Number: 2.4.41 **CFR Reference:** 43.5

Tier:	1	**RO Imp:**	2.3	**RO Select:**	No	**Difficulty:**	3
Group:	2	**SRO Imp:**	4.1	**SRO Select:**	Yes	**Taxonomy:**	H

Question: **RO:** **SRO:** 84

*****************REFERENCE PROVIDED***********************************

Given:
* Unit 1 tripped from 100% power 3 hours ago due to LOCA
* RCS pressure 50 psig
* Total HPI flow 480 gpm
* CETs 720 °F and lowering
* RE-8060, Containment High Range Rad Monitor 3700 R/hr
* RE-8061, Containment High Range Rad Monitor 4250 R/hr

Which of the following is the correct classification and barrier status?

A. SAE: Loss of 2 barriers only

B. SAE: Loss of 1 barrier and Potential Loss of 2nd barrier

C. GE: Loss of 2 barriers and Potential Loss of 3rd barrier

D. GE: Loss of 3 barriers

Answer:

C. GE: Loss of 2 barriers and Potential Loss of 3rd barrier

Notes:

For the given conditions "C" is the correct answer.
Potential Loss of Containment due to rad levels > 4000 R/hr (CNB5)
Loss of RCS due to rad levels > 100 R/hr (RCB3)
Loss of Fuel Clad due to rad levels 1000 R/hr (FCB4)
Potential Loss of Fuel Clad (CET temp) and HPI flow is for plausibility

"A" is wrong but plausible because 2 barriers are LOST.

"B" is wrong but plausible because all 3 barriers have met at least the Potential Loss criteria.

"D" is wrong both RCS and Fuel Clad are Lost and Containment is challenged.

Rev. 1 (NRC comments)
1. Re-formatted way rad monitors are described, and HPI flow.

Rev. 2, 5/16/17
Editorial changes only

References:

1903.010, Classification

1903.010, EAL Classification, pages 18-68 and 74-180 (EAL tabs plus bases, except Tab E) will be provided as an applicant reference.

History:

Developed for the 2017 SRO Re-take Exam.

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INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1118 **Rev:** 2 **Rev Date:** 5/16/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-EOP07 **Objective:** 14 **Point Value:** 1

Section: 4.3 **Type:** B&W EPE/APE

System Number: E09 **System Title:** Natural Circulation Cooldown

Description: Ability to determine and interpret the following as they apply to the (Natural Circulation Cooldown):
Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

K/A Number: EA2.1 **CFR Reference:** 43.5

Tier:	1	**RO Imp:**	2.8	**RO Select:**	No	**Difficulty:**	4
Group:	2	**SRO Imp:**	4.2	**SRO Select:**	Yes	**Taxonomy:**	H

Question: **RO:** **SRO:** 85

Given:
* A severe thunderstorm caused a loss of offsite power several hours ago
* Crew entered Degraded Power (1202.007)
* Stuck open MSSV on "A" SG resulted in this SG being dry and isolated
* Unit 1 cooling down on "B" SG only
* Natural Circulation Cooldown (1203.013) in use
* CETs 410 °F and lowering slowly
* Offsite power NOT expected to return for at least 12 hours
* EDGs providing power to A3 and A4

NOW

* During RCS de-pressurization ERV sticks open
* RCS pressure 390 psig and lowering

Which procedure shall be selected for the above conditions?

A. Go to HPI Cooldown (1202.011)

B. Return to Degraded Power (1202.007)

C. Go to Loss of Subcooling Margin (1202.002)

D. Stay in Natural Circulation Cooldown (1203.013)

Answer:

B. Return to Degraded Power (1202.007)

Notes:

"B" is correct. Section 1, Degraded Power, of 1203.013 would be in use. Step 4 of this section states to go to 1202.007, Degraded Power, if SCM is lost. SCM is > 30 but less than 50 degrees so it is not met.

"A" is wrong although 1202.011, HPI Cooldown, would be in use if there was an unisolable steam leak on BOTH SGs but only only the "A" SG has this problem.

"C" is wrong although 1202.002 would be entered for loss of SCM if the Natural Circulation was being performed with Section 2, Offsite Power Available. However, Section 1, Degraded Power, is applicable and therefore 1202.007 should be entered.

"D" is wrong. 1203.013 does not contain any mitigating actions for loss of SCM. Other EOPs contain those actions so this distractor is plausible if the candidate is not familiar with the procedure.

This question matches the K/A since it involves a Natural Circulation Cooldown condtion with conditions that require selection and transition to another procedure.

References:

1203.013, Natural Circulation Cooldown

History:

New question for 2017 SRO Re-exam
Rev 1 removed Loss of SCM statement and provided temp and pressure

Rev. 2, 5/16/17
Editorial changes.
4th bullet, added "Transition made to" at beginning.
Added another bullet stating crew has entered 1202.007.
Re-arranged answer choices short to long.
Added lesson plan reference.

QID: 0809 **Rev:** 3 **Rev Date:** 5/16/17 **Source:** Bank **Originator:** S Pullin

TUOI: A1LP-RO-AOP **Objective:** 6 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal from Reactor Core

System Number: 003 **System Title:** Reactor Coolant Pump System (RCPs)

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

K/A Number: 2.1.7 **CFR Reference:** 43.5

Tier:	2	**RO Imp:**		**RO Select:**	No	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.7	**SRO Select:**	Yes	**Taxonomy:**	H

Question: **RO:** **SRO:** 86

Given:
* 100% Power
* P-32B RCP seal bleedoff temperature 210 °F and slowly rising
* P-32B RCP motor bearing temperature 172 °F and stable
* P-32B RCP motor inboard vibration alert alarm
* P-32B RCP seal cavity pressure fluctuating between 650 and 1250 psig

Which of the following is the correct section and actions of
1203.031, Reactor Coolant Pump and Motor Emergency, to perform?

A. Section 2, Seal Failure;
Trip the Reactor, then trip affected RCP.

B. Section 5, Motor / Bearing Trouble;
Trip the Reactor, then trip affected RCP.

C. Section 2, Seal Failure;
Reduce reactor power to within the capacity of unaffected RCP combination,
then stop affected RCP.

D. Section 5, Motor / Bearing Trouble;
Reduce reactor power to within the capacity of unaffected RCP combination,
then stop affected RCP.

Answer:

A. Section 2, Seal Failure;
Trip the Reactor, then trip affected RCP.

Notes:

This question requires the SRO applicant to assess conditions and determine one of seven sections of 1203.031 to enter, and based upon that section, what action(s) are appropriate to take for the given conditions. It is thus still applicable to an SRO applicant, even though there are no transitions to other procedures.

"A" is correct, a seal bleedoff temperature of greater than 200 °F with no change in cooling (seal injection or ICW flow) meets the requirements to trip the RCP due to Section 2, Seal Failure.

"B" is wrong. The given conditions do not indicate a bearing problem that warrants stopping the RCP.

"C" is wrong. The given conditions require a trip of the RCP instead of a normal shutdown of an RCP.

"D" is wrong. The given conditions require a trip of the RCP instead of a normal shutdown of an RCP.

Matches the KA because the question requires evaluation and application of the RCP and Motor Emergency procedure and requires operational judgment based on instrument interpretation.

References:

1203.031, Reactor Coolant Pump and Motor Emergency
1203.012G, Annunciator K08 Corrective Action

History:

New selected for 2010 SRO exam
Question 1000 is a modified version of this question and was used in 2017 Re-exam Audit
Selected for 2017 SRO Re-exam

Bank question with the following changes for 2017 Re-take:
Modified question stem,
Affected RCP from "C" to "B",
RCP Motor bearing temperature from 185 to 172 °F,
Changed order making A correct instead of B.

Rev. 1 (NRC comments)
Made editorial changes per comments.
Added "and slowly rising" to seal bleedoff temp, and added "and stable" to motor bearing temp to ensure there only A is correct and that D is not also a correct answer.

Rev. 2, incorporated minor validation comments.

Rev. 3, 5/16/17
Editorial changes.
Deleted per 1103.006 from C and D.
Added 1203.012G to references

QID: 0594 **Rev:** 3 **Rev Date:** 5/16/17 **Source:** Modified **Originator:** Cork

TUOI: A1LP-RO-DH **Objective:** 27 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal from Reactor Core

System Number: 005 **System Title:** Residual Heat Removal

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the RHRS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Pressure transient protection during cold shutdown.

K/A Number: A2.02 **CFR Reference:** 43.5

Tier: 2 **RO Imp:** 3.5 **RO Select:** No **Difficulty:** 3

Group: 1 **SRO Imp:** 3.7 **SRO Select:** Yes **Taxonomy:** H

Question: **RO:** **SRO:** 87

Given:
* RCS pressure 200 psig and stable
* RCS temperature 150 °F and stable
* "A" DH pump is in service
* RCS pressure SASS handswitch hard selected to "X" due to failure of "Y" instrument

If I&C were to perform a pressure instrument calibration on RPS channel "A" channel, which of the following would occur and what actions would the CRS perform?

A. RCS pressure will drop rapidly, enter Pressurizer Systems Failure (1203.015) and isolate ERV.

B. DH isolation will close, enter Pressurizer Systems Failure (1203.015) and stop "A" DH pump.

C. RCS pressure will drop rapidly, enter Loss of Decay Heat Removal (1203.028) and isolate ERV.

D. DH isolation will close, enter Loss of Decay Heat Removal (1203.028) and stop "A" DH pump.

Answer:

A. RCS pressure will drop rapidly, enter Pressurizer Systems Failure (1203.015) and isolate ERV.

Notes:

"A" is correct, even though the ERV will be in LTOP mode with a setpoint of 400 psig and the pressure input for the low pressure control is wide range RCS pressure from ESAS Analog Channel 1, the high pressure setpoint is still active. The "A" RPS narrow range RCS pressure instrument inputs into high pressure ERV control. The calibration of "A" RPS pressure instrument will lift the ERV and entry into 1203.015 will lead the operators to isolate the ERV.

"B" is wrong but plausible since this is the correct procedure but the Decay Heat pump would not be isolated by this pressure channel calibration, however it would if it was the ESAS channel which inputs into the DH suction valve closure logic.

"C" is wrong but plausible since this is the correct action but the wrong procedure. This would be the correct procedure if it was the "C" RPS channel being calibrated.

"D" is wrong this is the wrong action and the wrong procedure. This would be the correct action and procedure if it was the ESAS channel being calibrated.

This question matches the K/A since it requires the applicant to "predict" the result of the inappropriate testing of the shutdown pressure protection and then determine the correct procedure and actions to perform.

References:

1203.015, Pressurizer Systems Failure
1203.028, Loss of Decay Heat Removal
STM 1-03, Reactor Coolant System

History:

New for 2005 SRO exam.
Modified for 2017 SRO Re-take exam

Rev. 1 - Modified this question to be in line with current SRO level questions by converting it to a 2x2 format and adding
actions to the choices. The two procedure choices are the most logical choices and the two actions correspond
to "A" or "C" RPS RCS pressure channel actions.

Rev. 2 (NRC comments)
1. Revised stem per NRC suggestion, combined calibration of channel with question and posed it as a hypothetical situation.
2. Added fourth bullet about RCS pressure selected to "X" since lifting the ERV would not occur otherwise.

Rev. 3, 5/16/17
Editorial changes.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1009 **Rev:** 2 **Rev Date:** 5/17/17 **Source:** Bank **Originator:** NRC

TUOI: A1LP-RO-TS **Objective:** 13 **Point Value:** 1

Section: 3.2 **Type:** RCS Inventory Control

System Number: 006 **System Title:** Emergency Core Cooling System (ECCS)

Description: Knowledge of surveillance procedures.

K/A Number: 2.2.12 **CFR Reference:** 43.2

Tier:	2	**RO Imp:**		**RO Select:**	No	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.1	**SRO Select:**	Yes	**Taxonomy:**	H

Question: **RO:** **SRO:** 88

Given:
* Unit 1 is operating at 100% power 60 days following a Refueling
* Control Room is informed that one valve in an ECCS train was not tested during the outage as required by its 18 month Surveillance Requirement 3.5.2.3
* This valve was last tested 20 months ago
* This valve is inaccessible with the reactor at power
* A mid-cycle outage is scheduled to occur in 8 months

To comply with Tech Specs, the CRS __________________.

A. Is required to enter TS 3.5.2.A upon notification.

B. Can delay entering TS 3.5.2.A for a maximum of 24 hours from the time of discovery if a risk assessment is performed and the risk is managed.

C. Is required to enter TS 3.5.2.A when the 25% grace period expires.

D. can delay entering TS 3.5.2.A until the mid-cycle outage if a risk assessment is performed and the risk is managed.

Answer:

D. can delay entering TS 3.5.2.A until the mid-cycle outage if a risk assessment is performed and the risk is managed.

Notes:

"D" is correct. Per TS SR 3.0.3, if a required surveillance is missed, entry into the applicable TS can be delayed for an additional surveillance frequency (in this case 18 months per SR 3.5.2.3) from the original due date if a risk assessment is done and the risk is managed. Because the mid-cycle outage is scheduled to fall within this window, distractor D is correct.

"A" is wrong based upon the above discussion on SR 3.0.3 but is plausible if the candidate does not recall the nuances of SR 3.0.3.

"B" is wrong but plausible based upon the above discussion on SR 3.0.3 but SR3.0.3 states "24 hours or the frequency, whichever is greater" but this is incorrect since 18 months is obviously greater than 24 hours.

"C" is wrong based upon the above discussion on SR 3.0.3 but is plausible if the candidate applies SR 3.0.2 which discusses a 25% grace period.

This question matches the K/A since a situation requiring the use of Tech Specs is given and the candidate must apply the specifications as well as knowledge of general surveillance requirement applicability (the latter is not part of the handout).

References:

TS 3.5.2, SR 3.0.1, SR 3.0.2, and SR 3.0.3

History:

New for 2013 SRO Exam
Selected for 2017 SRO Re-exam
Changed look of question slightly by changing from 3 valves to a single valve and midcycle from 12 months to 8
Rev. 1 5/5/17
1. Made recommended enhancements based on NRC comments
Rev. 2, 5/17/17
Editorial changes.
Shuffled answer choices.

QID: 1117 **Rev:** 3 **Rev Date:** 5/17/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-ASDCD **Objective:** 12 **Point Value:** 1

Section: 3.2 **Type:** RCS Inventory Control

System Number: 013 **System Title:** Engineered Safety Features Actuation System

Description: Knowledge of EOP mitigation strategies.

K/A Number: 2.4.6 **CFR Reference:** 43.5

Tier: 2 **RO Imp:** **RO Select:** No **Difficulty:** 4

Group: 1 **SRO Imp:** 4.7 **SRO Select:** Yes **Taxonomy:** H

Question: **RO:** **SRO:** 89

Given:
* Reactor has tripped due to a LOCA
* ESAS (1202.010) has been entered
* Crew has not initiated a cooldown
* Five (5) hours after a Reactor trip, the following conditions are observed
  * RCS pressure 800 psig and lowering
  * Full HPI flow
  * T-hot 450 °F and lowering
  * RB pressure peaked at 48 psia, now 30 psia and lowering slowly
  * BWST level 8 feet and lowering

Based on these conditions the crew should _______________ and verify that _________________.

A. remain in ESAS (1202.010);
ERV is isolated

B. remain in ESAS (1202.010);
RB Spray flow throttled to maintain 1050 - 1200 gpm per train

C. transition to Small Break LOCA Cooldown (1203.041);
ERV is isolated

D. transition to Small Break LOCA Cooldown (1203.041);
RB Spray flow throttled to maintain 1050 - 1200 gpm per train

Answer:

D. transition to Small Break LOCA Cooldown (1203.041);
RB Spray flow throttled to maintain 1050 - 1200 gpm per train

Notes:

"D" is correct because ESAS directs the transition to SBLOCA if cause of ESAS actuation is NOT corrected and an uncontrolled RCS cooldown is occurring due to HPI/break flow. Action is correct because RB Spray initiated at 44 psia and will need to be throttled prior to BWST reaching 6 feet.

"A" is wrong but plausible as entry into 1202.010 was appropriate, and if the uncontrolled cooldown was NOT occurring then the crew would be directed to remain in 1202.010 (incorrect procedure choice). The action is plausible because both the ESAS and SBLOCA procedures address positioning of this valve and if HPI cooling was not in progress then the ERV isolation should be closed, but this is not the correct action per the correct procedure (1203.041).

"B" is wrong but plausible: if the uncontrolled cooldown was NOT occurring, then the crew would be directed to remain in 1202.010. Throttling RB flow is the correct action.

"C" is wrong but plausible because ESAS directs the transition to SBLOCA if ESAS actuation is NOT corrected and an uncontrolled RCS cooldown is occurring due to HPI/break flow (correct transition). The action is plausible because both the ESAS and SBLOCA procedures address positioning of this valve and if HPI cooling was not in progress then the ERV isolation should be closed, but this is .

Matches the KA because the correct answer addresses both ESFAS actuations (RB Spray) and EOP mitigating strategies

References:

1203.041 Small Break LOCA Cooldown
1202.010, ESAS

History:

New question for 2017 SRO Re-exam
Rev 1- Removed reference to SCM, applicant must determine. Also changed to a 2x2 format as suggested.
Correct answer is now - transtition to another procedure
Rev. 2
1. Deleted unnecessary words from A & C choices no change to intent
2. Corrected explanation notes in A & B

Rev. 3, 5/17/17
Editorial changes.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1152 **Rev:** 2 **Rev Date:** 5/17/17 **Source:** New **Originator:** Burton

TUOI: A1LP-RO-EOP02 **Objective:** 14 **Point Value:** 1

Section: 3.5 **Type:** Containment Integrity

System Number: 026 **System Title:** Containment Spray System (CSS)

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the CSS; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Loss of containment spray pump suction when in recirculation mode, possibly caused by clogged sump screen, pump inlet high temperature exceeded cavitation, voiding or sump level below cutoff (interlock) limit.

K/A Number: A2.07 **CFR Reference:** 43.5

Tier: 2 **RO Imp:** 3.6 **RO Select:** No **Difficulty:** 3

Group: 1 **SRO Imp:** 3.9 **SRO Select:** Yes **Taxonomy:** H

Question: **RO:** **SRO:** 90

Given:
* Unit 1 tripped from 100% power due to LOCA
* Loss of Subcooling Margin (1202.002) in use
* RB Spray actuated
* Transfer to RB sump recirculation is complete

NOW

* RCS pressure 100 psig and stable
* RB Sump level dropped
* RB Flood level is steady
* Both LPI Pump discharge pressures fluctuating between 100 - 160 psig
* Both RB Spray P-35A/B ES Failure annunciators are coming in and out of alarm
* Dose Assessment reports no offsite release in progress

CRS should mitigate the event using ________________ and direct the crew to override and stop _____ RB Spray pump(s).

A. 1202.010 (ESAS), only one (1)

B. 1202.010 (ESAS), both (2)

C. 1202.011 (HPI Cooldown), only one (1)

D. 1202.011 (HPI Cooldown), both (2)

Answer:

B. 1202.010 (ESAS), both (2)

Notes:

It is stated that Loss of SCM is in use, however, conditions require transition to ESAS due to RCS pressure being less than 150 psig.. HPI Cooldown is plausible since Primary to Secondary is not effective under these conditions since with a LOCA (and loss of SCM) the hot legs would be voided, preventing natural circulation flow. Since there is no breach of Containment stopping both trains of RB Spray is directed in Attachment 1 of ESAS (1202.010).

"B" is correct per the above explanation.

"A" is wrong but plausible because it references the correct procedure and a single train would be stopped if containment breach had occurred, but due to the lack of offsite release, no breach is indicated.

"C" is wrong but plausible as HPI cooldown is a LOCA based procedure and refers to RT-15 for RB sump recirculation. A single train of RB Spray would be stopped if containment breach had occurred, but there are no indications of a breach.

"D" is wrong but plausible as HPI cooldown is a LOCA based procedure, and the actions to stop both trains of RB Spray is correct.

Meets the KA since the question involves the CSS and requires the use of a procedure to mitigate the event in progress.

References:

1202.010, ESAS
1202.011, HPI Cooldown
1202.012, Repetitive Task 15

History:

Selected for 2017 SRO Exam.
Rev. 1 5/5/17
1. Changed the stem to inform the applicant that LOSM EOP was entered due to a LOCA
2. Moved the information that RCS press is 100 psig down under the current status information
3. Modified the explaination notes to reflect the changes to the question
Rev. 2, 5/17/17
Editorial changes.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1154 **Rev:** 2 **Rev Date:** 6/8/17 **Source:** New **Originator:** Burton

TUOI: A1LP-RO-CRD **Objective:** 12/13 **Point Value:** 1

Section: 3.1 **Type:** Reactivity Control

System Number: 014 **System Title:** Rod Position Indication

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

K/A Number: A2.03 **CFR Reference:** 43.6

Tier:	2	**RO Imp:**		**RO Select:**	No	**Difficulty:**	3
Group:	2	**SRO Imp:**	4.1	**SRO Select:**	Yes	**Taxonomy:**	F

Question: **RO:** **SRO:** 91

Given:
* Unit 1 100% power
* Group 5 rods are 100% withdrawn when Group 5 Rod 5 drops to 0% withdrawn
* Plant has runback to 40% power

(1) How would the dropped rod affect the Rod Position Indication System (RPIS)?
(2) If Control Rod alignment and power recovery to 60% begins 30 hours after event, then Control Rod Drive Malfunction Action (1203.003) limits power escalation to a maximum of _____ /hr.

A. (1) RPI indicates 0%, API indicates 100%
(2) 3%

B. (1) RPI indicates 0%, API indicates 100%
(2) 30%

C. (1) RPI indicates 100%, API indicates 0%
(2) 3%

D. (1) RPI indicates 100%, API indicates 0%
(2) 30%

Answer:

C. (1) RPI indicates 100%, API indicates 0%
(2) 3%

Notes:

"C" is correct - API position is by use of reed switches so where the rod is what API uses RPI uses demand signal of the motor therefore in the case of a dropped rod RPI will indicate the last position. Per 1203.033 power rate of change is 3%/hr after 24 hours no matter what current power level. Less than 24 hours the applicant must apply time and power level to determine the correct rate of change for this power change which is 30%/hr.

"A" is wrong but plausible since one set of indicators will read 100% and the other 0% withdrawn. Also 3% is correct.

"B" is wrong but plausible since one set of indicators will read 100% and the other 0% withdrawn, 30% is valid.

"D" is wrong but plausible since the indicators are correct and 30% is a valid limit.

Matches the KA because the question requires the applicant to correctly predict the effect on RPIS due to a dropped Control Rod.

This is SRO Only as only the CRS is required to make the decision as to when and how fast to change power. The SRO directs the Ros in setting final power level and the rate of power change.

References:

1203.003, Control Rod Drive Malfunction Action
STM 1-02, Control Rod Drive System
SOER 84-2, Control Rod Mispositioning
SER 84-83, Control Rod Misalignment
INPO 91-008

History:

Selected for 2017 SRO Re-exam
Rev. 1, 5/17/07
Editorial changes.
Rev. 2, 6/8/17
Changed taxonomy to F.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1041 **Rev:** 2 **Rev Date:** 5/17/17 **Source:** Bank **Originator:** Possage

TUOI: A1LP-RO-FH **Objective:** 4 **Point Value:** 1

Section: 3.8 **Type:** Plant Service Systems

System Number: 034 **System Title:** Fuel Handling Equipment

Description: Ability to explain and apply systen limits and precautions

K/A Number: 2.1.32 **CFR Reference:** 43.2

Tier:	2	**RO Imp:**	3.8	**RO Select:**	No	**Difficulty:**	3
Group:	2	**SRO Imp:**	4.0	**SRO Select:**	Yes	**Taxonomy:**	F

Question: **RO:** **SRO:** 92

Given:
* Unit 1 in Mode 6
* SRO in Charge of Refueling verifying limits and precautions contained in Control of Unit 1 Refueling (1502.004) are met prior to core off-load.

In accordance with 1502.004, which of the following conditions would prevent moving fuel in the Reactor Building?

A. Tornado Watch in effect for Conway County.

B. Reactor was tripped and Mode 3 entered 92 hours ago.

C. One Control Room Emergency Air Conditioning System (CREACS) inoperable for the past 4 days.

D. RB Fuel Handling Area radiation monitor RE-8017 inoperable, and portable survey instrument is being used to monitor the fuel handling bridge.

Answer:

B. Reactor was tripped and Mode 3 entered 92 hours ago.

Notes:

"B" is correct, the reactor must be subcritical for greater than 100 hours prior to fuel movement

"A" is wrong. Pope, Johnson, Yell and Logan counties in a tornado watch would require stopping fuel movement. Conway county is immediately east of Pope county.

"C" is wrong. With one CREACS channel inoperable we have 30 days to repair prior to stopping fuel movement.

"D" is wrong. RE-8017 is desired to be operable for monitoring radiation levels on the bridge, however if it becomes inoperable any portable survey instrument is allowed for monitoring rad levels and continue fuel movement.

Matched the KA because it is knowledge of the limitations and precautions allowing the use of Fuel Handling Equipment
for core off-load.

References:

1502.004, Control of Unit 1 Refueling
TS 3.7.10, CREACS
TRM 3.9.1, Refueling Operations

History:

Selected for 2014 SRO Exam
Repeated for 2017 SRO Exam
modified an NRC bank question (2014). Modified stem for 2017 exam, (0 hours subcritical to reactor tripped

and
mode 3 entered 92 hours ago. Changed correct answer from position "A" to "B"
Rev. 1
1. Made editorial corrections based on NRC comments

Rev. 2, 5/17/17
Editorial changes.
Corrected noun name of RE-8017.

QID: 1110 **Rev:** 2 **Rev Date:** 5/17/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-FPS **Objective:** 10/11 **Point Value:** 1

Section: 3.8 **Type:** Plant Service Systems

System Number: 086 **System Title:** Fire Protection

Description: Ability to apply Technical Specifications for a system.

K/A Number: 2.2.40 **CFR Reference:** 43.5

Tier: 2 **RO Imp:** 3.4 **RO Select:** No **Difficulty:** 3

Group: 2 **SRO Imp:** 4.7 **SRO Select:** Yes **Taxonomy:** H

Question: **RO:** **SRO:** 93

****************REFERENCE PROVIDED***********************

Given:
* Unit 1 100% power
* Annunciator K12-D1 FIRE PROT SYSTEM TROUBLE alarms
* CBOT reports alarm on C463 panels is yellow LED on
  C4-4U EFW PUMP ROOM SMOKE DET ZONE 38-Y
* Alarm will not reset
* No other new alarms are indicated on C463 panels
* Unit 1 Fire Impairment Database lists no impairments for this zone

Which of the following satisfies the TRM for Zone 38-Y and can be completed within this shift?

A. Establish a continuous fire watch.

B. Establish a 1-hour roving fire watch only.

C. Run fire hoses to establish backup fire suppression only.

D. Verify alternate smoke and/or heat detection with control room alarm.

Answer:

A. Establish a continuous fire watch for Zone 38-Y.

Notes:

"A" is correct, with the yellow "trouble" LED in alarm on the P-7A EFW Pump Room, the detection is non-functional but the sprinkler system also is non-functional, therefore 3.7.9 required action A.1.1, establishing a continuous fire watch, must be completed within one hour.

"B" is wrong but plausible if the candidate considers only the non-functional smoke detection (3.3.6 action A.1) and not the suppression system whose required actions are more stringent.

"C" is wrong but plausible if the candidate does not recognize that the sprinkler system can still be manually actuated and running fire hoses is unnecessary, and thus is not required.

"D" is incorrect but plausible if the candidate merely reads the 3.7.9 required action A.1.2 statement and does not recognize that there is no other smoke/heat detection in the area.

This question matches the K/A as it requires the candidate to use the provided TRM specifications and apply them to the non-functional fire protection components.

References:

ANO-1 Technical Requirements Manual

TRM TRO's 3.3.6 and 3.7.9 must be in SRO handout!!!!

History:

New for 2017 SRO Re-exam
Rev. 1 (NRC comments)
1. Made "action" singular in stem.
2. Removed "for Zone 38-Y" from all answer choices and placed it in stem.
3. Swapped A and B.
4. Removed quotes from alarm descriptions.
5. Changed "one hour" to 1-hour" in B.

Rev. 2, 5/17/17
Revised stem.
Added "only" to B and C.
Editorial changes.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1109 **Rev:** 3 **Rev Date:** 5/17/17 **Source:** New **Originator:** Cork

TUOI: ASLP-RO-REACT **Objective:** 6 **Point Value:** 1

Section: 2.0 **Type:** Generic Knowledges and Abilities

System Number: 2.1 **System Title:** Conduct of Operations

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

K/A Number: 2.1.37 **CFR Reference:** 43.6

Tier: 3 **RO Imp:** **RO Select:** No **Difficulty:** 2

Group: **SRO Imp:** 4.6 **SRO Select:** Yes **Taxonomy:** F

Question: **RO:** **SRO:** 94

Per COPD-030, ANO Reactivity Management Program, which of the listed positions determines the Risk Level Classification for an activity determined to have a Reactivity Management Impact?

A. Operations SRO

B. Work Week Manager

C. Shift Technical Advisor

D. Reactor Engineering Supervisor

Answer:

A. Operations SRO

Notes:

"A" is correct per step 6.1.2 of COPD-030.

"C" is wrong but plausible since the STA can be the second individual verifying borations or dilutions are performed per step 5.5[29] of COPD-030 but they do not have this responsibility.

"D" is wrong but plausible since this position is consulted for power maneuvers, startups, etc., but this is not one of Rx Engineering's responsibilities.

"B" is wrong but plausible since this position is involved with many parts of work activities but this is beyond his responsibility in COPD-030. In addition per the On-line Work Management Process (EN-WM-101) he signs the emergent addition/deletion form which assesses both Risk and Reactivity Impact.

This question matches the K/A since it directly asks about responsibilities in ANO's specific Operations Directive for Reactivity Management.

References:

COPD-030, Reactivity Management

History:

New question for 2017 SRO Retake exam.

Rev 1 - Did not make any changes to question, stated why Work Week Manager is plausible in justification pages.

Rev 2
1. Added "Supervisor" to end of B.
2. Deleted "..and Minimum Required Defenses" from stem.
3. Combined given condition into stem to simplify question.

Rev. 3, 5/17/17
Re-ordered answer choices short to long.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1157 **Rev:** 1 **Rev Date:** 5/17/17 **Source:** New **Originator:** Burton

TUOI: A1LP-RO-TS **Objective:** 13 **Point Value:** 1

Section: 2.0 **Type:** Generic Knowledges and Abilities

System Number: 2.1 **System Title:** Conduct of Operations

Description: Knowledge of primary and secondary plant chemistry limits.

K/A Number: 2.1.34 **CFR Reference:** 43.5

Tier: 3 **RO Imp:** **RO Select:** No **Difficulty:** 3

Group: **SRO Imp:** 3.5 **SRO Select:** Yes **Taxonomy:** F

Question: **RO:** **SRO:** 95

Given:
* Unit 1 100% power

(1) Per LCO 3.4.12, RCS Specific Activity, the maximum limit for DOSE EQUIVALENT I-131 is ≤ ____ µCi/gm;
(2) Per Bases 3.4.12, the resulting offsite and control room dose will not exceed the applicable 10 CFR 50.67 requirements following a Main Steam Line Break or ______ .

A. (1) 0.1
(2) Loss of Coolant Accident

B. (1) 0.1
(2) Steam Generator Tube Rupture

C. (1) 1.0
(2) Loss of Coolant Accident

D. (1) 1.0
(2) Steam Generator Tube Rupture

.

Answer:

D. (1) 1.0
(2) Steam Generator Tube Rupture

Notes:

"D" is correct because it identifies both the correct I-131 limit for the RCS and the correct accidents that the limit is based.

"A" is wrong but plausible because 0.1 is the secondary limit for I-131and LOCA is a break in one of the 3 barriers and is identified in numerous bases as a limiting accident.

"B" is wrong but plausible because 0.1 is the secondary limit for I-131 and identifies the correct accident.

"C" is wrong but plausible because it correctly identifies the RCS limit for I-131and LOCA is a break in one of the 3 barriers and is identified in numerous bases as a limiting accident.

This question matches the K/A since it requires the applicant to demonstrate knowledge of primary chemistry limits.

References:

TS 3.4.12, RCS Specific Activity and Bases
TS 3.7.4, Secondary Specific Activity

History:

New question for 2017 SRO Retake exam.

Rev. 1, 5/17/17
Editorial changes

QID: 1158 **Rev:** 3 **Rev Date:** 5/17/17 **Source:** New **Originator:** Burton

TUOI: ASLP-SRO-OPSPR **Objective:** 6 **Point Value:** 1

Section: 2 **Type:** Generic Knowledges and Abilities

System Number: 2.2 **System Title:** Equipment Control

Description: Knowledge of the process for conducting special or infrequent tests.

K/A Number: 2.2.7 **CFR Reference:** 43.3

Tier:	3	**RO Imp:**	2.9	**RO Select:**	No	**Difficulty:**	2
Group:		**SRO Imp:**	3.6	**SRO Select:**	Yes	**Taxonomy:**	F

Question: **RO:** **SRO:** 96

Given:
* Unit 100% power
* Crew has been directed to perform a task which requires a new procedure in conjunction with existing procedures
* Potential to cause a significant plant transient exists

Which of the following is correct in regards to required briefs that should be prepared before the task can be allowed to proceed?

A. Pre-Job and Reactivity

B. Refocus and Reactivity

C. Pre-Job and Infrequently Performed Tests or Evolutions

D. Refocus and Infrequently Performed Tests or Evolutions

Answer:

C. Pre-Job and Infrequently Performed Tests or Evolutions

Notes:

"C" is correct per the IPTE and COPD001 procedures both a pre-job and IPTE brief is required because this is a special or infrequent tests, in this case, the applicant must recognize that any first time use procedure with a potential for a plant transient require IPTE controls be established including a specific brief.

"A" is wrong but plausible since both Pre-job is correct and Reactivity briefs are identified in COPD001 also there is a potential for a reactivity excursion due to this task.

"B" is wrong but plausible since both the Refocus and Reactivity briefs are identified in COPD001 and there is a potential for a reactivity excursion due to this task.

"D" is wrong but plausible since IPTE is correct and a Refocus brief is identified in COPD001

This question matches the K/A as it tests the knowledge of performing special or infrequent tests, in this case, the applicant must recognize that any first time use procedure with a potential for a plant transient require IPTE controls be established including a specific brief.

References:

EN-OP-116, Infrequently Performed Test or Evolutions
EN-OP-115-04, Operations Briefs
COPD001, Operations Expectations and Standards
COPD030, ANO Reactivity Management Program

History:

New question for 2017 SRO Retake exam

Rev. 3, 5/17/17

Editorial changes.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1107 **Rev:** 2 **Rev Date:** 5/17/17 **Source:** New **Originator:** Burton

TUOI: A1LP-SRO-MNTC **Objective:** 9 **Point Value:** 1

Section: 2.0 **Type:** Generic Knowledges and Abilities

System Number: 2.2 **System Title:** Equipment Control

Description: Knowledge of the process for controlling equipment configuration or status.

K/A Number: 2.2.14 **CFR Reference:** 43.3

Tier:	3	**RO Imp:**		**RO Select:**	No	**Difficulty:**	2
Group:		**SRO Imp:**	4.3	**SRO Select:**	Yes	**Taxonomy:**	F

Question: **RO:** **SRO:** 97

Given:
* Planned maintenance requires connecting contaminated and non-contaminated systems with a rubber hose
* System Engineer determined this is a Temporary Modification

Before the hose is connected between these two systems, who must authorize installation AND what is the MINIMUM number of check valve(s) which must be installed to meet requirements of EN-DC-136, Temporary Modifications?

A. Shift Manager; 1 check valve

B. Shift Manager; 2 check valves in series

C. System Engineering Manager; 1 check valve

D. System Engineering Manager; 2 check valves in series

Answer:

B. Shift Manager; 2 check valves in series

Notes:

"B" is correct, the Shift Manager authorizes and the T-mod procedure requires 2 check valves in series.

"A" is wrong but plausible since Shift Manager is correct and a single check valve is allowed as a Clearance Boundary and per B3.6.3 can be credited as a Containment Isolation Boundary.

"C" is wrong but plausible since the Engineering Manager is identified and has numerous responsibilities within the T-mod process and procedure. Also a single check valve is allowed as a Clearance Boundary and per B3.6.3 can be credited as a Containment Isolation Boundary.

"D" is wrong but plausible since the Engineering Manager is identified and has numerous responsibilities within the T-mod process and procedure. 2 check valves in series is correct.

This question meets the K/A since it requires the applicant to demonstrate knowledge of the T-mod process which is controlling equipment configuration.

References:

EN-DC-136, Temporary Modifications
EN-OP-102, Protective and Caution Tagging
TS bases 3.6.3 Reactor Building Isolation Valves

History:

New question for 2017 SRO Retake exam.
Rev 1 - Changed to a different question we were having trouble with just 1 correct answer that could not be disputed.

Rev. 2, 5/17/17

Editorial changes

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1168 **Rev:** 3 **Rev Date:** 5/17/17 **Source:** Modified **Originator:** Burton

TUOI: A1LP-SRO-RAD **Objective:** 4 **Point Value:** 1

Section: 2.0 **Type:** Generic K & A's

System Number: 2.3 **System Title:** Radiation Control

Description: 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. .

K/A Number: 2.3.12 **CFR Reference:** 43.4

Tier:	3	**RO Imp:**	3.2	**RO Select:**	No	**Difficulty:**	3
Group:		**SRO Imp:**	3.7	**SRO Select:**	Yes	**Taxonomy:**	F

Question: **RO:** **SRO:** 98

Given:
* Unit 1 tripped from 100% due to LOCA
* TSC activated
* An operator has 1.4 Rem accumulative dose for calendar year
* NRC form 4 on file
* Access required to High Radiation Area to align a filter
* Expected exposure will be 1.7 Rem

Per Entergy administrative procedures, which of the SPECIFIC recommendations and authorizations listed below are required to extend the workers TEDE exposure limit?

A. Supervisor AND Radiation Protection Manager ONLY.

B. Supervisor, Radiation Protection Manager AND Plant General Manager.

C. Radiation Protection Manager AND Emergency Plant Manager ONLY.

D. Radiation Protection Manager, Plant General Manager AND Site Vice President.

Answer:

B. Supervisor, Radiation Protection Manager, AND Plant General Manager

Notes:

"B" is correct. Dose is > 3R but less than 4R (Supervisor, RP Manager and Plant GM approves). All of the remaining choices are plausible as they are found within the procedure table or with a LOCA could be an emergecny dose OP-1903.033.

"A" is wrong, this is the extension required for doses >2 R but <3 R.

"C" is wrong this is the authorization if there is an emergency and an equipment or lifesaving circumstance existed. (10R/25R)

"D" is wrong this is the authorization for doses >4 R but <4.5 R.

Changed stem to an Operator task and order of the distractor so that correct answer is "D" instead of "C"
Changed the numbers slightly so they add up to 3.1 instead of 4.4

References:

EN-RP-201,Dosimetry Administration
OP-1903.033, PAG for Rescue/Repair & Damage Control Team

History:

Modified QID 931 from the 2014 SRO exam for 2017 SRO Re-take Exam.
Rev. 2
1. Removed "EOF" from in front of Emergency Plant Manager in Distractor C
2. Reworded explanation note for Distractor A and C

Rev. 3, 5/17/17
Editorial changes.

QID: 1160 **Rev:** 2 **Rev Date:** 5/17/17 **Source:** New **Originator:** Burton

TUOI: ASCBT-EP-A0081 **Objective:** 5 **Point Value:** 1

Section: 2.0 **Type:** Generic Knowledges and Abilities

System Number: 2.4 **System Title:** Emergency Procedures / Plan

Description: Knowledge of procedures relating to a security event (non-safeguards information).

K/A Number: 2.4.28 **CFR Reference:** 43.5

Tier:	3	**RO Imp:**	3.2	**RO Select:**	No	**Difficulty:**	3
Group:		**SRO Imp:**	4.1	**SRO Select:**	Yes	**Taxonomy:**	H

Question: **RO:** **SRO:** 99

*****************REFERENCE PROVIDED***********************************

Given:
* OAO reports an on-going gun fight with multiple intruders and Security officers by Intake Structure traveling screen (F-7A)
* Security confirms report and adds there has been an explosion in a manhole adjacent to the Intake Structure
* OAO also reports she is exiting the area and there is no visible damage to the Service Water system

Per 1903.010, Emergency Action Level Classification, what is the appropriate event classification?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Answer:

C. Site Area Emergency

Notes:

"C" contains the correct classification per 1903.010 since this is a security event involving gun fire inside the protected area HS1.

"A" is wrong but plausible because an explosion on site meets the criteria of HU4.

"B" is wrong but plausible because an explosion in the manholes near the Intake Structure is HA4

"D" is wrong but plausible because HG1 would be met if the security event involved a vital area

This matches the K/A since 1903.010 is one of the procedures which would be used during a security event.

References:

1903.010, Emergency Action Level Classification

1903.010 EAL Classification (MODIFIED) will be provided as a reference. Information pages 1-18 and 69-73 have been removed.

History:

New question for 2017 SRO Retake exam.
Rev. 1
1. Made editorial changes based on feedback from NRC
Rev.2, 5/17/17
Editorial changes.

QID: 1162 **Rev:** 3 **Rev Date:** 5/17/17 **Source:** New **Originator:** Burton

TUOI: ASCBT-EP-A0082 **Objective:** 15 **Point Value:** 1

Section: 2 **Type:** Generic Knowledges and abilities

System Number: 2.4 **System Title:** Emergency Procedures/Plan

Description: Knowledge of emergency plan protective action recommendations.

K/A Number: 2.4.44 **CFR Reference:** 43.5

Tier: 3 **RO Imp:** 2.4 **RO Select:** No **Difficulty:** 3

Group: **SRO Imp:** 4.4 **SRO Select:** Yes **Taxonomy:** F

Question: **RO:** **SRO:** 100

Per Emergency Response/Notifications (1903.011):

(1) Protective Action Recommendations (PARs) are required to be provided after a declaration of ___________.
(2) If there has been a change in wind direction, then PAR must be re-assessed within a MINIMUM of ____________.

A. (1) ONLY General Emergencies
(2) 15 minutes

B. (1) ONLY General Emergencies
(2) 30 minutes

C. (1) BOTH General and Site Area Emergencies
(2) 15 minutes

D. (1) BOTH General and Site Area Emergencies
(2) 30 minutes

Answer:

A. (1) ONLY General Emergencies
(2) 15 minutes

Notes:

"A" is correct. The Emergency Class Initial Notification Message form 1903.011-Y states that PARs are required for GEs but not for NUE, Alert or SAE classifications. 15 minutes is correct requirement for making the notification for the GE which includes the PAR.

"B" is wrong. Only General Emergency is correct, 30 minutes is wrong but plausible because this is the maximum allowed time to complete notifications from the time conditions are available in the Control Room. (15 minutes + 15 to complete notifications)

"C" is wrong as PARs are not required for SAEs but plausible as plant or localized evacuations could be required for Ges or SAEs based on the plant conditions. 15 minutes is correct.

"D" is wrong as PARs are not required for SAEs as stated in "C". 30 minutes is plausible as stated in "B".

This question matches the K/A since the applicant must have knowledge of when to provide Emergency Plan PARs and how often PARs are re-assessed.

References:

1903.010, Emergency Action Level Classification
1903.011, Emergency Response/Notifications

History:

Selected for 2017 SRO Retake exam.
Rev. 2 5/6/17

1. Revised stem and Notes based on NRC comment
2. Added "BOTH" in front of Distractors C & D to make them appear more like A & B which state "ONLY"
Rev. 3, 5/17/17
Editorial changes.

2017
ANO UNIT 1
NRC INITIAL LICENSE EXAMINATION REFERENCE MATERIAL
SRO

TRM 3.3 INSTRUMENTATION

TRM 3.3.6 Fire Detection System Instrumentation

TRO 3.3.6

---NOTE---
1. Reactor Building smoke detectors are not required to be FUNCTIONAL during Type A Integrated Leak Rate Testing.
2. All non-functional detectors specified in TRM Table 3.3.6-1 will be tracked.
3. TRO entry not required solely due to maintenance or testing activities where FUNCTIONALITY is expected to be restored within one hour.

The following heat/smoke detectors in the locations specified in TRM Table 3.3.6-1 shall be FUNCTIONAL:

1. A minimum of 50% of the heat/smoke detectors in locations outside the Reactor Building, and,

2. All heat/smoke detectors located inside the Reactor Building.

APPLICABILITY: At all times

ACTIONS

---NOTE---
1. Separate Condition entry is allowed for each location specified in TRM Table 3.3.6-1.
2. In lieu of Required Actions establishing a fire watch or requiring equipment restoration, the licensee may choose to establish compensatory measures commensurate with the evaluated risk for continued operation with non-functional detectors. All other Required Actions are applicable regardless of compensatory measures established.
3. Entry into Condition A or C requires documentation of a Fire System Impairment, except when the non-functional detector is a result of maintenance or testing lasting less than 12 hours.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -------------NOTE-------------- Not applicable to Reactor Building fire detectors. ----------------------------------- Less than 50% of the detectors in the locations specified in TRM Table 3.3.6-1 FUNCTIONAL.	A.1 Establish a 1-hour roving fire watch. AND	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
Condition A (continued)	A.2	Restore at least 50% of the detectors in the locations specified in TRM Table 3.3.6-1 to FUNCTIONAL status.	14 days
B. One or more detectors in the locations specified in TRM Table 3.3.6-1 non-functional that result in complete loss of automatic actuation function of a fire suppression system.	B.1	Declare the associated Fire Suppression Sprinkler/Halon System non-functional and enter applicable Conditions and Required Actions of TRO 3.7.9 and/or 3.7.10.	Immediately
C. One or more Reactor Building fire detectors non-functional.	C.1	-------------NOTE-------------- Only required in Mode 1 and 2, or when Required Action C.2 cannot be performed. ----------------------------------	
		Monitor and record Reactor Building temperature.	Once per hour
	<u>AND</u>		
	C.2	-------------NOTE-------------- Only required in Modes 3, 4, 5, 6 and defueled when environmental and radiological conditions permit unescorted entry. ----------------------------------	
		Verify fire watch patrol of the affected area.	Once per 8 hours
D. Required Actions and associated Completion Time for Condition A, B, or C not met.	D.1	Initiate a condition report.	Immediately
	<u>AND</u>		
	D.2	Determine any limitations for continued operation of the plant.	24 hours

TEST REQUIREMENTS

TEST		FREQUENCY
TR 3.3.6.1	Perform a CHANNEL CHECK of required fire detection instrumentation.	12 hours
TR 3.3.6.2	------------------------------NOTE------------------------------ Not required for Reactor Building detectors. -- Perform a visual inspection of the required fire detection instrumentation.	 6 months
TR 3.3.6.3	Perform a visual inspection of the required Reactor Building fire detection instrumentation.	18 months
TR 3.3.6.4	Perform a CHANNEL FUNCTIONAL TEST of the required fire detection instrumentation located outside the Reactor Building.	12 months
TR 3.3.6.5	Perform a CHANNEL FUNCTIONAL TEST of the required fire detection instrumentation located inside the Reactor Building.	18 months

TRM Table 3.3.6-1

AREAS PROTECTED BY HEAT/SMOKE DETECTORS

Protected Area Description	Fire Zone	Elevation	Controls Suppression System
Spent Fuel Area	159-B	404'	N/A
Computer Room (under floor detection only)	160-B	404’	N/A
Computer Transformer Room	167-B	404'	N/A
Upper North Reactor Building Cable Spreading Area	32-K	401'	FS-5643
Upper South Reactor Building Cable Spreading Area	33-K	401'	FS-5644
North Emergency Diesel Generator Exhaust Fans (#2)	1-E	386’	N/A
South Emergency Diesel Generator Exhaust Fans (#1)	2-E	386’	N/A
Controlled Access Area	128-E	386'	N/A
Main Control Room Ceiling	129-F	386'	Halon System #3
Auxiliary Control Room Ceiling	129-F	386'	Halon System #2
Auxiliary Control Room Floor	129-F	386'	Halon System #1
Upper South Electrical Penetration Room	144-D	386'	UAV-5616
Upper North Electrical Penetration Room	149-E	386'	UAV-5615
Lower South Electrical Penetration Room	105-T	374'	UAV-5626
Lower North Electrical Penetration Room	112-I	373'	UAV-5625
Lower North Reactor Building Cable Spreading Area	32-K	373'	FS-5642
Lower South Reactor Building Cable Spreading Area	33-K	373'	FS-5645
Main Chiller Room (detection in Black Battery Room)	75-AA	372’	N/A
North Battery Room	95-O	372’	N/A
Cable Spreading Room	97-R	372'	UAV-5638
Hallway	98-J	372'	UAV-5639
North Switchgear Room (A-4)	99-M	372'	N/A
South Switchgear Room (A-3)	100-N	372'	N/A
South Inverter Room	110-L	372'	N/A
South Battery Room	110-L	372'	N/A
4160 VAC Switchgear Area	197-X	372	N/A
West Heater Deck Area	197-X	372	N/A
North Emergency Diesel Generator Room (#2)	86-G	369'	UAV-5602
South Emergency Diesel Generator Room (#1)	87-H	369'	UAV-5601
Electrical Equipment Room (Lower South)	104-S	368'	N/A
North Upper Piping Penetration Room	79-U	360'	UAV-5654
South Upper Piping Penetration Room	77-V	356'	N/A
Tank Room	68-P	354'/374'	N/A
Intake Structure	INTAKE	354'/366'	N/A
Lube Oil Storage Tank Room (Heat Detection)	175-CC	354'	UAV-5620

TRM Table 3.3.6-1 (continued)

Protected Area Description	**Fire Zone**	**Elevation**	**Controls Suppression System**
Laboratory And Demineralizer Access Area	67-U	354'	N/A
Condensate Demineralizer Area	73-W	354'	N/A
Compressor Room.	76-W	354'	N/A
Bowling Alley (Near Train Bay)	197-X	354	N/A
Pipe Area	40-Y	341'	N/A
Storage And Pipe Area	34-Y	335'/341'	N/A
Radwaste Processing Area	20-Y	335'	N/A
EFW Pump Room	38-Y	335'	UAV-5607
South Lower Piping Penetration Room	46-Y	335'	N/A
Penetration Ventilation Room	47-Y	335'	N/A
North Lower Piping Penetration Room	53-Y	335'	N/A
East Decay Heat Removal Pump Room (B Vault)	10-EE	317'	N/A
West Decay Heat Removal Pump Room (A Vault)	14-EE	317'	N/A

TRM 3.7 PLANT SYSTEMS

TRM 3.7.9 Fire Suppression Sprinkler System

TRO 3.7.9

--NOTE--
Fire Suppression Water System sectionalized, loop, or sprinkler system valves may be closed to support system testing provided an individual is stationed at the valve with direct communication with the control room, such that the valve can be re-opened without delay if needed.

The Fire Suppression Sprinkler Systems specified in TRM Table 3.7.9-1 shall be FUNCTIONAL.

APPLICABILITY: At all times

ACTIONS

--NOTE--
1. Separate Condition entry is allowed for each sprinkler system specified in TRM Table 3.7.9-1.
2. In lieu of Required Actions establishing a fire watch, verifying FUNCTIONAL smoke and/or heat detection for the affected areas, establishing backup suppression equipment, or returning non-functional fire suppression sprinkler systems to FUNCTIONAL status, the licensee may choose to establish compensatory measures commensurate with the evaluated risk for continued operation with non-functional Fire Suppression Sprinkler Systems. All other Required Actions are applicable regardless of compensatory measures established.
3. Entry into Condition A requires documentation of a Fire System Impairment, except when non-functionality is a result of maintenance or testing lasting less than 12 hours.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Fire Suppression Sprinkler Systems specified in TRM Table 3.7.9-1 non-functional.	A.1.1 Establish a continuous fire watch in the affected area.	1 hour
	<u>OR</u>	
	A.1.2 Verify FUNCTIONAL smoke and/or heat detection for the affected area with control room alarm.	1 hour
	<u>AND</u>	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 Establish backup fire suppression equipment for the affected area. <u>AND</u> A.3 Restore the non-functional Fire Suppression Sprinkler System to FUNCTIONAL status.	1 hour 14 days
B. Required Actions and associated Completion Time for Condition A not met.	B.1 Initiate a condition report. <u>AND</u> B.2 Determine any limitations for continued operation of the plant.	Immediately 24 hours

TEST REQUIREMENTS

TEST		FREQUENCY
TR 3.7.9.1	Verify each Fire Suppression Sprinkler System manual, power operated, or automatic valve in the flow paths specified in TRM Table 3.7.9-1 that is not locked, sealed, or otherwise secured in position, is correctly aligned and capable of transporting water from the system main to the sprinkler heads.	31 days
TR 3.7.9.2	Deleted	
TR 3.7.9.3	Cycle through at least one complete cycle each valve in the Fire Suppression Sprinkler System flow path located outside the Reactor Building specified in TRM Table 3.7.9-1.	12 months

TEST REQUIREMENTS (continued)

TEST		FREQUENCY
TR 3.7.9.4	Deleted	
TR 3.7.9.5	Perform visual integrity inspection of spray nozzles and headers associated with the Fire Suppression Sprinkler Systems specified in TRM Table 3.7.9-1.	18 months

TRM Table 3.7.9-1

AREAS PROTECTED BY SPRINKLER SYSTEMS

Suppression Sprinkler Systems	**Fire Zone**	**Elevation**	**Control Valve / Flow Switch**
Reactor Building Purge Room*	163-B	404’	UAV-5631
Boric Acid Addition Tank & Pump Room*	120-E	386’	UAV-3202
Respirator Storage Room*	125-E	386’	FS-5632
Decon Room and Hot Mechanic Shop*	149-E	386’	FS-5630
Upper South Electrical Penetration Room	144-D	386'	UAV-5616
Upper North Electrical Penetration Room	149-E	386'	UAV-5615
Lower South Electrical Penetration Room	105-T	374'	UAV-5626
Lower North Electrical Penetration Room	112-I	373'	UAV-5625
Cable Spreading Room	97-R	372'	UAV-5638
Hallway	98-J	372'	UAV-5639
Controlled Access	128-E	372’	UAV-3202
North Emergency Diesel Generator Room	86-G	369'	UAV-5602
South Emergency Diesel Generator Room	87-H	369'	UAV-5601
Laboratory and Demineralizer Access Area*	67-U	354’	UAV-5628
Condensate Demineralizer Area	73-W	354'	UAV-5627
Main Chiller Room	75-AA	354’	FS-5625
Upper North Piping Penetration Room	79-U	354’	UAV-5654
T-27 Lube Oil Storage Tank Room	175-CC	354’	UAV-5620
Turbine Building (below Operating Floor west of turbine centerline	197-X	354’	UAV-5624
Intake Structure	INTAKE	354'	FS-5600
Radwaste Processing Room*	20-Y	335’	UAV-5628**
EFW Pump Room, P7A	38-Y	335'	UAV-5607
Clean & Dirty Lube Oil Storage Tank Room*	187-DD	335’	FS-5626

* Area is covered by a Sprinkler system without a corresponding Detection System.

** Suppression from 67-U provides suppression to BWST valve area in 20-Y.

ATTACHMENT 1

INDEX OF EMERGENCY ACTION LEVELS

INDEX OF EMERGENCY ACTION LEVELS

SECTION **PAGE**

TAB A: Abnormal Radiation Levels / Radiological Effluents

TAB C: Cold Shutdown / Refueling System Malfunction

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

SECTION **PAGE**

TAB A

Abnormal Radiation Levels / Radiological Effluents

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

ABNORMAL RADIOLOGICAL EFFLUENTS

AG1

1	2	3	4	5	6	D

Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity > 1000 mR TEDE or 5000 mR child thyroid CDE for the actual or projected duration of the release using actual meteorology

Emergency Action Level(s):

NOTE:

The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, the classification should be based on EAL #2 instead of EAL #1. Do not delay declaration awaiting dose assessment results.

1. VALID reading on Channel 9 on any of the following radiation monitors > the reading shown for ≥ 15 minutes:

MONITORS – Unit 1		LIMIT
RX-9820	Containment Purge	1.18E+2 µCi/cc
RX-9825	Radwaste Area	1.07E+2 µCi/cc
RX-9830	Fuel Handling Area	9.08E+1 µCi/cc
RX-9835	Emerg. Penetration Room	1.91E+3 µCi/cc
MONITORS – Unit 2		**LIMIT**
2RX-9820	Containment Purge	8.92E+1 µCi/cc
2RX-9825	Radwaste Area	6.64E+1 µCi/cc
2RX-9830	Fuel Handling Area	8.92E+1 µCi/cc
2RX-9835	Emerg. Penetration Room	1.77E+3 µCi/cc
2RX-9840	PASS Building	8.84E+2 µCi/cc
2RX-9845	Aux. Building Extension	2.53E+2 µCi/cc

OR

AS1

1	2	3	4	5	6	D

Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity > 100 mR TEDE or 500 mR child thyroid CDE for the actual or projected duration of the release

Emergency Action Level(s):

NOTE:

The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, the classification should be based on EAL #2 instead of EAL #1. Do not delay declaration awaiting dose assessment results.

1. VALID reading on Channel 9 on any of the following radiation monitors > the reading shown for ≥ 15 minutes:

MONITORS – Unit 1		LIMIT
RX-9820	Containment Purge	1.18E+1 µCi/cc
RX-9825	Radwaste Area	1.07E+1 µCi/cc
RX-9830	Fuel Handling Area	9.08E0 µCi/cc
RX-9835	Emerg. Penetration Room	1.91E+2 µCi/cc
MONITORS – Unit 2		**LIMIT**
2RX-9820	Containment Purge	8.92E0 µCi/cc
2RX-9825	Radwaste Area	6.64E0 µCi/cc
2RX-9830	Fuel Handling Area	8.92E0 µCi/cc
2RX-9835	Emerg. Penetration Room	1.77E+2 µCi/cc
2RX-9840	PASS Building	8.84E+1 µCi/cc
2RX-9845	Aux. Building Extension	2.53E+1 µCi/cc

OR

AA1

1	2	3	4	5	6	D

Any release of gaseous or liquid radioactivity to the environment > 200 times the ODCM limits for ≥ 15 minutes

Emergency Action Level(s):

NOTE:

The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

1. VALID reading on Channel 7 on any of the following radiation monitors > the reading shown for ≥ 15 minutes:

MONITORS – Unit 1		LIMIT
RX-9820	Containment Purge	1.18E0 µCi/cc
RX-9825	Radwaste Area	1.07E0 µCi/cc
RX-9830	Fuel Handling Area	9.08E-1 µCi/cc
RX-9835	Emerg. Penetration Room	1.91E+1 µCi/cc
MONITORS – Unit 2		**LIMIT**
2RX-9820	Containment Purge	8.92E-1 µCi/cc
2RX-9825	Radwaste Area	6.64E-1 µCi/cc
2RX-9830	Fuel Handling Area	8.92E-1 µCi/cc
2RX-9835	Emerg. Penetration Room	1.77E+1 µCi/cc
2RX-9840	PASS Building	8.84E0 µCi/cc
2RX-9845	Aux. Building Extension	2.53E0 µCi/cc
2RX-9850	LLRW Storage Building	3.54E0 µCi/cc

OR

AU1

1	2	3	4	5	6	D

Any release of gaseous or liquid radioactivity to the environment > 2 times the ODCM limits for ≥ 60 minutes

Emergency Action Level(s):

NOTE:

The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

1. VALID reading on Channel 7 on any of the following radiation monitors > the reading shown for ≥ 60 minutes:

MONITORS – Unit 1		LIMIT
RX-9820	Containment Purge	1.18E-02 µCi/cc
RX-9825	Radwaste Area	1.07E-02 µCi/cc
RX-9830	Fuel Handling Area	9.08E-03 µCi/cc
RX-9835	Emerg. Penetration Room	1.91E-01 µCi/cc
MONITORS – Unit 2		**LIMIT**
2RX-9820	Containment Purge	8.92E-3 µCi/cc
2RX-9825	Radwaste Area	6.64E-3 µCi/cc
2RX-9830	Fuel Handling Area	8.92E-3 µCi/cc
2RX-9835	Emerg. Penetration Room	1.77E-1 µCi/cc
2RX-9840	PASS Building	8.84E-2 µCi/cc
2RX-9845	Aux. Building Extension	2.53E-2 µCi/cc
2RX-9850	LLRW Storage Building	3.54E-2 µCi/cc

OR

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

ABNORMAL RADIOLOGICAL EFFLUENTS

AG1 *(continued)*

2. Dose assessment using actual meteorology indicates doses > 1000 mR TEDE or 5000 mR child thyroid CDE at or beyond the site boundary.

OR

3. Field survey results indicate closed window dose rates > 1000 mR/hr expected to continue for ≥ 60 minutes; or analyses of field survey samples indicate child thyroid CDE > 5000 mR for one hour of inhalation, at or beyond the site boundary.

AS1 *(continued)*

2. Dose assessment using actual meteorology indicates doses > 100 mR TEDE or 500 mR child thyroid CDE at or beyond the site boundary.

OR

3. Field survey results indicate closed window dose rates > 100 mR/hr expected to continue for ≥ 60 minutes; or analyses of field survey samples indicate child thyroid CDE > 500 mR for one hour of inhalation, at or beyond the site boundary.

AA1 *(continued)*

2. **EITHER** VALID reading on any of the following radiation monitors > 200 times the alarm setpoint established by a current release permit for ≥ 15 minutes **OR** VALID reading greater than the value listed for ≥ 15 minutes:

MONITORS – Unit 1		LIMIT
RX-9820	Cont. Purge (Ch. 7 or 9)	N/A
RX-4830	Waste Gas Monitor	9.5E7 cpm
RX-4642	Liquid Radwaste Monitor	9.5E7 cpm
RX-9835	Emerg. Penetration Room	N/A
MONITORS – Unit 2		**LIMIT**
2RX-9820	Cont. Purge (Ch. 7 or 9)	N/A
2RX-2429	Waste Gas Monitor	9.5E5 cpm
2RX-2330	BMS Discharge Monitor	9.5E5 cpm
2RX-4423	LRW Discharge Monitor	9.5E5 cpm
2RX-4425	SG BD to Flume Monitor	9.5E5 cpm

OR

3. Confirmed grab sample analyses for gaseous or liquid releases indicates concentrations or release rates > 200 times the applicable values of the ODCM for ≥ 15 minutes.

AU1 *(continued)*

2. VALID reading on any of the following radiation monitors > 2 times the alarm setpoint established by a current release permit for ≥ 60 minutes:

MONITORS – Unit 1	
RX-9820	Cont. Purge (Ch. 7 or 9)
RX-4830	Waste Gas Monitor
RX-4642	Liquid Radwaste Monitor
RX-9835	Emerg. Penetration Room
MONITORS – Unit 2	
2RX-9820	Cont. Purge (Ch. 7 or 9)
2RX-2429	Waste Gas Monitor
2RX-2330	BMS Discharge Monitor
2RX-4423	LRW Discharge Monitor
2RX-4425	SG BD to Flume Monitor

OR

3. Confirmed grab sample analyses for gaseous or liquid releases indicates concentrations or release rates > 2 times the applicable values of the ODCM for ≥ 60 minutes.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

ABNORMAL RADIATION LEVELS

ALERT

AA2 1 2 3 4 5 6 D

Damage to irradiated fuel or loss of water level that has resulted or will result in the uncovering of irradiated fuel outside the reactor vessel

Emergency Action Level(s):

1. A water level drop in the refueling canal or spent fuel pool that will result in irradiated fuel becoming uncovered.

   OR

2. VALID alarm on any of the following radiation monitors due to damage to irradiated fuel or loss of water level:

MONITORS – Unit 1	
RX-9820	Containment Purge (Channel 7 or 9)
RX-9825	Radwaste Area (Channel 7 or 9)
RX-9830	Fuel Handling Area (Channel 7 or 9)
RE-8060	Containment High Range Monitor
RE-8061	Containment High Range Monitor
RE-8009	Spent Fuel Area
RE-8017	Fuel Handling
MONITORS – Unit 2	
2RX-9820	Containment Purge (Channel 7 or 9)
2RX-9825	Radwaste Area (Channel 7 or 9)
2RX-9830	Fuel Handling Area (Channel 7 or 9)
2RE-8905	Containment Equipment Hatch Area
2RE-8909	Containment Personnel Hatch Area
2RE-8925-1/2	Containment High Range Monitors
2RE-8914/15/16	Spent Fuel Area Monitors
2RE-8912	Containment Incore Instruments

UNUSUAL EVENT

AU2 1 2 3 4 5 6 D

UNPLANNED rise in plant radiation levels

Emergency Action Level(s):

1. a. UNPLANNED lowering of water level in the refueling canal or spent fuel pool as indicated by:

   - Personnel observation, refueling crew report, indication on area security camera, borated water source (BWST or RWT) level drop due to makeup demands.

   AND

   b. VALID Area Radiation Monitor reading rise on any of the following:

MONITORS – Unit 1	
RE-8009	Spent Fuel Area
RE-8017	Fuel Handling Area
MONITORS – Unit 2	
2RE-8914	Spent Fuel Area
2RE-8915	Spent Fuel Area
2RE-8916	Spent Fuel Area
2RE-8912	Containment Incore Instrumentation

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
ABNORMAL RADIATION LEVELS			
		AA3 1 2 3 4 5 6 D Rise in radiation levels within the facility that impedes operation of systems required to maintain plant safety functions. **Emergency Action Level(s):** 1. Dose rate > 15 mR/hr in any of the following areas requiring continuous occupancy to maintain plant safety functions: • Unit 1 Control Room • Unit 2 Control Room • Central Alarm Station	**OR** **AU2** *(continued)* 2. UNPLANNED VALID Area Radiation Monitor readings or survey results indicate a rise by a factor of 1000 over normal* levels. **NOTE:** *For area radiation monitors with ranges incapable of measuring 1000 times normal* levels, classification shall be based on VALID full scale indication unless surveys confirm that area radiation levels are below 1000 times normal* within 15 minutes of the Area Radiation Monitor indications going to full scale indication.* * Normal can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

TAB C

Cold Shutdown / Refueling System Malfunction

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION – Loss of RCS / Reactor Vessel Inventory

CG1 (Modes: 5 6)

Loss of RCS / reactor vessel inventory affecting fuel clad integrity with containment challenged

Emergency Action Level(s):

NOTE:

The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

1. a. Core exit thermocouples indicate superheat for ≥ 30 minutes.

   AND

   b. Any of the following containment challenge indications:

   - CONTAINMENT CLOSURE not established
   - Explosive mixture inside containment
   - UNPLANNED rise in containment pressure

   OR

2. a. RCS / reactor vessel level cannot be monitored for ≥ 30 minutes with a loss of RCS/ reactor vessel inventory as indicated by any of the following:

CS1 (Modes: 5 6)

Loss of RCS / reactor vessel inventory affecting core decay heat removal capability

Emergency Action Level(s):

NOTE:

The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

1. With CONTAINMENT CLOSURE **not** established:

   Loss of RCS / reactor vessel level as indicated by:

   Unit 1: RVLMS Levels 1 through 9 indicate DRY

   Unit 2: RVLMS Levels 1 through 6 indicate DRY

   OR

2. With CONTAINMENT CLOSURE established, core exit thermocouples indicate superheat.

   OR

CA1 (Modes: 5 6)

Loss of RCS / reactor vessel inventory

Emergency Action Level(s):

NOTE:

The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

1. Loss of RCS / reactor vessel inventory as indicated by:

   Unit 1: RVLMS Levels 1 through 8 indicate DRY

   Unit 2: RVLMS Levels 1 through 5 indicate DRY

   OR

   Unit 1: Reactor vessel level <368 ft., 0 in. (bottom of the hot leg)

   Unit 2: Reactor vessel level < 369 ft., 1.5 in. (bottom of the hot leg)

   OR

CU1 (Modes: 5)

RCS leakage

Emergency Action Level(s):

NOTE:

The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

1. RCS leakage results in the inability to maintain or restore level within Pressurizer or RCS level target band for ≥ 15 minutes.

CU2 (Modes: 6)

UNPLANNED loss of RCS / reactor vessel Inventory

Emergency Action Level(s):

NOTE:

The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

1. UNPLANNED RCS / reactor vessel level drop as indicated by either of the following:

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION – Loss of RCS / Reactor Vessel Inventory			
CG1 *(continued)* • Containment High Range Radiation Monitor reading >10 R/hr • Erratic source range monitor indication • Unexplained level rise in Reactor Building Sump, Reactor Drain Tank, Quench Tank, Aux. Building Equipment Drain Tank, or Aux. Building Sump **AND** b. Any of the following containment challenge indications: • CONTAINMENT CLOSURE not established • Explosive mixture inside containment • UNPLANNED rise in containment pressure	**CS1** *(continued)* 3. RCS / reactor vessel level cannot be monitored for ≥ 30 minutes with a loss of RCS / reactor vessel inventory as indicated by any of the following: • Containment High Range Radiation Monitor reading > 10 R/hr • Erratic source range monitor indication • Unexplained level rise in Reactor Building Sump, Reactor Drain Tank, Quench Tank, Aux. Building Equipment Drain Tank, or Aux. Building Sump	**CA1** *(continued)* 2. RCS / reactor vessel level cannot be monitored for ≥ 15 minutes with a loss of RCS / reactor vessel inventory as indicated by an unexplained level rise in the Reactor Building Sump, Reactor Drain Tank, Aux. Building Equipment Drain Tank, Aux. Building Sump, or Quench Tank.	**CU2** *(continued)* a. RCS / reactor vessel water level drop below the reactor vessel flange for ≥15 minutes when the RCS / reactor vessel level band is established above the reactor vessel flange. **OR** b. RCS / reactor vessel water level drop below the RCS / reactor vessel level band for ≥ 15 minutes when the RCS / reactor vessel level band is established below the reactor vessel flange. **OR** 2. RCS / reactor vessel level cannot be monitored with a loss of RCS / reactor vessel inventory as indicated by an unexplained level rise in (as applicable) the Reactor Building Sump, Reactor Drain Tank, Aux. Building Equipment Drain Tank, Aux. Building Sump, or Quench Tank.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION – Loss of Decay Heat Removal

ALERT

CA3 5 6

Inability to maintain plant in Cold Shutdown

Emergency Action Level(s):

1. An UNPLANNED event results in RCS temperature > 200 °F > the specified duration in Table C1.

Table C1
RCS Reheat Duration Thresholds

RCS	Containment Closure	Duration
Intact (but not RCS lowered inventory)	N/A	60 minutes*
Not intact or RCS lowered inventory	Established	20 minutes*
	Not Established	0 minutes

*If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.

OR

NOTE: *EAL #2 does not apply in solid plant conditions.*

2. An UNPLANNED event results in RCS pressure rise > 10 psi due to a loss of RCS cooling.

UNUSUAL EVENT

CU3 5 6

UNPLANNED loss of decay heat removal capability with irradiated fuel in the reactor vessel

Emergency Action Level(s):

NOTE:

The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

1. UNPLANNED event results in RCS temperature exceeding 200 °F.

OR

2. Loss of all RCS temperature and RCS / reactor vessel level indication for ≥ 15 minutes.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION – Loss of AC Power			
		CA5 5 6 D Loss of all offsite and all onsite AC power to Vital 4.16 KV busses ≥ 15 minutes **Emergency Action Level(s):** **NOTE:** *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.* 1. Loss of all offsite and all onsite AC power to Vital 4.16KV busses ≥ 15 minutes.	**CU5** 5 6 AC power capability to Vital 4.16 KV busses reduced to a single power source ≥ 15 minutes such that any additional single power source failure would result in station blackout **Emergency Action Level(s):** **NOTE:** *The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.* 1. a. AC power capability to Vital 4.16 KV busses reduced to a single power source ≥ 15 minutes. **AND** b. Any additional single power source failure will result in station blackout.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION – Loss of DC Power			
			CU6 5 6 Loss of required DC power ≥ 15 minutes **Emergency Action Level(s):** **NOTE:** *The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.* 1. < 105 volts on required Vital DC bus ≥ 15 minutes.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION – Inadvertant Criticality			
			CU7 5 6 Inadvertent criticality **Emergency Action Level(s):** 1. UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION – Loss of Communications			
			CU8 (Modes: 5, 6, D) Loss of all onsite or offsite communications capabilities **Emergency Action Level(s):** 1. Loss of all Table C2 onsite communication methods affecting the ability to perform routine operations. **Table C2 Onsite Communications Equipment**: Station radio system; Plant paging system; In-plant telephones; Gaitronics **OR** 2. Loss of all Table C3 offsite communication methods affecting the ability to perform offsite notifications. **Table C3 Offsite Communications Equipment**: All telephone lines (commercial and microwave); ENS

Table C2
Onsite Communications Equipment

Station radio system
Plant paging system
In-plant telephones
Gaitronics

Table C3
Offsite Communications Equipment

All telephone lines (commercial and microwave)
ENS

TAB F

Fission Product Barrier Degradation

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
FISSION PRODUCT BARRIER DEGRADATION – Barriers			
FG1 1 2 3 4	**FS1** 1 2 3 4	**FA1** 1 2 3 4	**FU1** 1 2 3 4
Loss of ANY two barriers AND loss or potential loss of third barrier	Loss or potential loss of ANY two barriers	ANY loss or ANY potential loss of EITHER fuel clad or RCS	ANY loss or ANY potential loss of containment

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, multiple events could occur which result in the conclusion that exceeding the loss or potential loss EALs is IMMINENT. In this IMMINENT loss situation use judgment and classify as if the EALs are exceeded.

Fuel Clad Barrier EALs		RCS Barrier EALs		Containment Barrier EALs	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
1. Primary Coolant Activity Level (FCB1)		**1. RCS Leak Rate (RCB1)**		**1. Containment Pressure (CNB1)**	
1. Coolant activity > 300 µCi/gm dose equivalent I-131 activity by Chemistry sample **OR** 2. Radiation levels > 1000 MR/hr **Unit 1:** at SA-229 **Unit 2:** at 2TCD-19	None	RCS leak rate > available makeup capacity as indicated by: **Unit 1:** Loss of adequate subcooling margin **Unit 2:** RCS subcooling (MTS) can NOT be maintained at least 30 °F	**Unit 1:** UNISOLABLE RCS leak > 50 gpm with Letdown isolated **Unit 2:** UNISOLABLE RCS leak > 44 gpm with Letdown isolated	1. Rapid unexplained drop in containment pressure following an initial rise in containment pressure **OR** 2. Containment pressure or sump level response not consistent with LOCA conditions	1. **Unit 1:** Containment pressure 73.7 PSIA (59 PSIG) and rising **Unit 2:** Containment pressure 73.7 PSIA and rising **OR** 2. Explosive mixture exists inside Containment **OR** 3. a. Containment Pressure > containment spray actuation setpoint **Unit 1:** 44.7 PSIA (30 PSIG) **Unit 2:** 23.3 PSIA **AND** b. LESS THAN one full train of spray operating

Fuel Clad Barrier EALs		RCS Barrier EALs		Containment Barrier EALs	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
2. Core Exit Thermocouple Readings (FCB2)		**2. SG Tube Rupture (RCB2)**		**2. Core Exit Thermocouple Readings (CNB2)**	
> 1200 °F CET temperature	**Unit 1:** ICC exists as evidenced by CETs indicating superheated conditions **Unit 2:** Average CETs indicate superheat for current RCS pressure	SGTR that results in an ECCS (SI) actuation	None	None	1. a. CETs indicate > 1200 °F **AND** b. Restoration procedures not effective within 15 minutes **OR** 2. a. CETs indicate > 700 °F **AND** b. RVLMS indicates **Unit 1:** Levels 1 through 9 DRY **Unit 2:** Levels 1 through 7 DRY **AND** c. Restoration procedures not effective within 15 minutes

Fuel Clad Barrier EALs		RCS Barrier EALs		Containment Barrier EALs	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
3. Reactor Vessel Water Level (FCB3)		**3. Containment Radiation Monitoring (RCB3)**		**3. SG Secondary Side Release With Primary-to-Secondary Leakage (CNB3)**	
None	**Unit 1:** RVLMS Levels 1 through 9 indicate DRY **Unit 2:** RVLMS Levels 1 through 7 indicate DRY	Containment high range radiation monitor reading > 100 R/hr	None	1. RUPTURED steam generator is also FAULTED outside of containment **OR** 2. a. Primary to secondary leakrate > 10 gpm **AND** b. UNISOLABLE steam release from affected steam generator to the environment	None
4. Containment Radiation Monitoring (FCB4)		**4. Emergency Director Judgment (RCB4)**		**4. Containment Isolation Failure or Bypass (CNB4)**	
Containment high range radiation monitor reading > 1000 R/hr	None	Any condition in the opinion of the SM / ED that indicates Loss or Potential Loss of the RCS barrier		1. UNISOLABLE breach of containment **AND** 2. Direct downstream pathway to the environment exists after containment isolation signal	None

Fuel Clad Barrier EALs		RCS Barrier EALs		Containment Barrier EALs	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
5. Core Damage Assessment (FCB5)				**5. Containment Radiation Monitoring (CNB5)**	
At least 5% fuel clad damage as determined from core damage assessment	None			None	Containment high range radiation monitor reading > 4000 R/hr
6. Emergency Director Judgment (FCB6)				**6. Other Indications (CNB6)**	
Any condition in the opinion of the SM/ED that indicates Loss or Potential Loss of the fuel clad barrier				Elevated readings on the following radiation monitors that indicate loss or potential loss of the Containment barrier: (see monitor table below)	
				7. Emergency Director Judgment (CNB7)	
				Any condition in the opinion of the SM / ED that indicates Loss or Potential Loss of the containment barrier	

MONITORS – Unit 1	
RX-9820	Containment Purge
RX-9825	Radwaste Area
RX-9830	Fuel Handling Area
RX-9835	Emergency Penetration Room
MONITORS – Unit 2	
2RX-9820	Containment Purge
2RX-9825	Radwaste Area
2RX-9830	Fuel Handling Area
2RX-9835	Emergency Penetration Room
2RX-9840	Post Accident Sampling Building
2RX-9845	Auxiliary Building Extension

TAB H

Hazards and Other Conditions Affecting Plant Safety

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY – Security			
HG1 1 2 3 4 5 6 D	**HS1** 1 2 3 4 5 6 D	**HA1** 1 2 3 4 5 6 D	**HU1** 1 2 3 4 5 6 D
HOSTILE ACTION resulting in loss of physical control of the facility	HOSTILE ACTION within the PROTECTED AREA	HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat	Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant
Emergency Action Level(s):	**Emergency Action Level(s):**	**Emergency Action Level(s):**	**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.	1. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by ANO Security Shift Supervision.	1. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by ANO Security Shift Supervision. **OR** 2. A validated notification from NRC of an airliner attack threat within 30 minutes of the site.	1. A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by ANO 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.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY – Discretionary			
HG2 1 2 3 4 5 6 D	**HS2** 1 2 3 4 5 6 D	**HA2** 1 2 3 4 5 6 D	**HU2** 1 2 3 4 5 6 D
Other conditions exist which in the judgment of the SM / ED warrant declaration of General Emergency	Other conditions exist which in the judgment of the SM / ED warrant declaration of a Site Area Emergency	Other conditions exist which in the judgment of the SM / ED warrant declaration of an Alert	Other conditions exist which in the judgment of the SM warrant declaration of an NUE
<u>Emergency Action Level(s):</u>	**<u>Emergency Action Level(s):</u>**	**<u>Emergency Action Level(s):</u>**	**<u>Emergency Action Level(s):</u>**
1. Other conditions exist which in the judgment of the SM / ED indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	1. Other conditions exist which in the judgment of the SM / ED indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.	1. Other conditions exist which in the judgment of the SM / ED indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	1. Other conditions exist which in the judgment of the SM indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY – Control Room Evacuation			
	HS3 1 2 3 4 5 6 D Control Room evacuation has been initiated and plant control cannot be established **Emergency Action Level(s):** 1. a. Control Room evacuation has been initiated. **AND** b. Control of the plant cannot be established in accordance with the following procedures within 15 minutes: **Unit 1:** 1203.002, "Alternate Shutdown" **Unit 2:** 2203.014, "Alternate Shutdown"	**HA3** 1 2 3 4 5 6 D Control Room evacuation has been initiated **Emergency Action Level(s):** 1. Alternate Shutdown procedure requires Control Room evacuation: **Unit 1:** 1203.002, "Alternate Shutdown" **Unit 2:** 2203.014, "Alternate Shutdown"	

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY			
		Fire	
		[1]HA4 1 2 3 4 5 6 D FIRE or EXPLOSION affecting the operability of plant safety systems required to establish or maintain safe shutdown **<u>Emergency Action Level(s):</u>** 1. FIRE or EXPLOSION resulting in VISIBLE DAMAGE to any **Table H1** structure or area containing safety systems or components <u>or</u> Control Room indication of degraded performance of those safety systems:	**[1]HU4** 1 2 3 4 5 6 D FIRE within the PROTECTED AREA not extinguished within 15 minutes of detection **<u>OR</u>** EXPLOSION within the PROTECTED AREA **<u>Emergency Action Level(s):</u>** **NOTE:** *The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.* 1. FIRE in any **Table H1** structure or area not extinguished: a. within 15 minutes of Control Room notification **<u>OR</u>** b. within 15 minutes of [2]verification of a Control Room FIRE alarm (i.e. Alarm valid until disproved) **<u>OR</u>** 2. EXPLOSION within the PROTECTED AREA.

[1]The HA4 and HU4 EALs apply to any Table H1 structure or area whether in service or tagged out for maintenance.

[2]Verification of a fire detection system alarm/actuation includes actions that can be taken within the Control Room or other nearby site specific location to ensure that it is not spurious.

Table H1

Unit 1

Reactor Building

All Elevations

Aux Building

All Elevations Including Penthouse/MSIV Room
Exceptions: Boric Acid Mix Tank Room (Chem Add Area) 404’ (157-B)
EDG Exhaust Fan area on 386’ (1-E and 2-E)

Turbine Building

All Elevations
Including:
Pipechase under ICW Coolers
CRD Pump Pit / T-28 Room / Area under ICW Pumps

Outside Areas

Manholes adjacent to Startup #2 XFMR (MH-03/MH-04)
Manholes adjacent to Intake Structure (MH-05/MH-06)
Intake Structure (354’ and 366’)
Diesel Fuel Vault
Diesel Fuel Vault Pump Manholes MH-09 and MH-10 (Manhole, MH-09, is located approximately 15 feet northeast of the Unit 1 QCST , Manhole, MH-10, is located approximately 5 feet west of Unit 2 Condensate Storage Tank, 2T-41A)

Table H1

Unit 2

Reactor Building

All Elevations

Aux Building

All Elevations including Aux Extensions

Turbine Building

All Elevations

Outside Areas

Intake Structure (354’ and 366’)
Concrete Manhole East, NE of intake
Concrete Manhole East of Turbine building next to train bay
Diesel Fuel Vault
Diesel Fuel Vault Pump Manholes MH-09 and MH-10 (Manhole, MH-09, is located approximately 15 feet northeast of the Unit 1 QCST , Manhole, MH-10, is located approximately 5 feet west of Unit 2 Condensate Storage Tank, 2T-41A)

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY			
		Toxic Gas	
		HA5 1 2 3 4 5 6 D Access to a VITAL AREA is prohibited due to toxic, corrosive, asphyxiant, or flammable gases which jeopardize operation of operable equipment required to maintain safe operations or safely shutdown the reactor **Emergency Action Level(s):** **NOTE:** *If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.* 1. Access to a VITAL AREA is prohibited due to toxic, corrosive, asphyxiant, or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shutdown the reactor.	**HU5** 1 2 3 4 5 6 D Release of toxic, corrosive, asphyxiant, or flammable gases deemed detrimental to NORMAL PLANT OPERATIONS **Emergency Action Level(s):** 1. Toxic, corrosive, asphyxiant, or flammable gases in amounts that have or could adversely affect NORMAL PLANT OPERATIONS. **OR** 2. Report by Local, County or State officials for evacuation or sheltering of site personnel based on an offsite event.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

Toxic Gas

HA5 *(continued)* 1 2 3 4 5 6 D

Unit 1

VITAL AREA	APPLICABLE MODES
A-4 Switchgear Room	3, 4
Upper North Electrical Penetration Room	3, 4
Lower South Electrical Equipment Room	3, 4
Control Room	ALL

Unit 2

VITAL AREA	APPLICABLE MODES
Auxiliary Building 317' Emergency Core Cooling Rooms	3, 4
Auxiliary Building 317' Tendon Gallery Access	3, 4
Auxiliary Building 335' Charging Pumps/ 2B-52	3, 4
Auxiliary Building 354' 2B-62 Area	3, 4
Emergency Diesel Generator Corridor	3, 4
Lower South Piping Penetration Room	3, 4
Auxiliary Building 386' Containment Hatch	3, 4
Control Room	ALL

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY			
		Natural or Destructive Phenomena	
		HA6 1 2 3 4 5 6 D Natural or destructive phenomena affecting VITAL AREAS **<u>Emergency Action Level(s):</u>** 1. a. Seismic event > Operating Basis Earthquake (OBE) as indicated by annunciation of the 0.1g acceleration alarm. **<u>AND</u>** b. Earthquake confirmed by ANY of the following: • Earthquake felt in plant • National Earthquake Center • Control Room indication of degraded performance of systems required for the safe shutdown of the plant **<u>OR</u>**	**HU6** 1 2 3 4 5 6 D Natural or destructive phenomena affecting the PROTECTED AREA **<u>Emergency Action Level(s):</u>** 1. Seismic event identified by any 2 of the following: • Seismic event confirmed by annunciation of the 0.01g acceleration alarm • Earthquake felt in plant • National Earthquake Center **<u>OR</u>** 2. Tornado striking within PROTECTED AREA boundary or high winds > 67 mph. (**2 minute average**) **<u>OR</u>** 3. Internal flooding that has the potential to affect safety related equipment required by Technical Specifications for the current operating mode in any of the structures or areas in **Table H1. (Page 47)** **<u>OR</u>** 4. Turbine failure resulting in casing penetration or damage to turbine or generator seals. **<u>OR</u>**

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY			
		Natural or Destructive Phenomena	
		HA6 *(continued)* 2. Tornado striking or winds > 67 mph (2 minute average) resulting in VISIBLE DAMAGE to any of the following structures/equipment containing safety systems or components <u>or</u> Control Room indication of degraded performance of those safety systems: • Reactor Building • Intake Structure • Ultimate Heat Sink • BWST/RWT • Auxiliary Building • Turbine Building • QCST • Control Room • Startup Transformers • Diesel Fuel Vault **<u>OR</u>** 3. Internal flooding in any of the following areas resulting in an electrical shock hazard that precludes access to operate or monitor safety equipment <u>or</u> Control Room indication of degraded performance of those safety systems: • Intake Structure • Auxiliary Building • Turbine Building	**HU6** *(continued)* 5. Lake Dardanelle level < 335 feet. **<u>OR</u>** 6. Lake Dardanelle level > 345 feet.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY			
		Natural or Destructive Phenomena	
		HA6 *(continued)* **<u>OR</u>** 4. Turbine failure-generated PROJECTILES resulting in VISIBLE DAMAGE to or penetration of any of the structures/equipment containing safety systems or components <u>or</u> Control Room indication of degraded performance of those safety systems: • Auxiliary Building • Turbine Building • Control Room • Startup Transformers **<u>OR</u>** 5. Lake Dardanelle level < 335 feet and Emergency Cooling Pond inoperable. **<u>OR</u>**	

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY			
		Natural or Destructive Phenomena	
		HA6 *(continued)* 6. Vehicle crash resulting in VISIBLE DAMAGE to any of the structures/equipment containing safety systems or components <u>o</u>r Control Room indication of degraded performance of those safety systems: • Reactor Building • Intake Structure • Ultimate Heat Sink • BWST/RWT • Auxiliary Building • Turbine Building • QCST • Startup Transformers • Diesel Fuel Vault	

TAB S

System Malfunction

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
SYSTEM MALFUNCTION – Loss of AC Power			
SG1 1 2 3 4	**SS1** 1 2 3 4	**SA1** 1 2 3 4	**SU1** 1 2 3 4
Prolonged loss of all offsite and all onsite AC power to Vital 4.16 KV busses **Emergency Action Level(s):** 1. a. Loss of all offsite and all onsite AC power to Vital 4.16 KV busses. **AND** b. Either of the following: • Restoration of at least one Vital 4.16 KV bus in < 4 hours is not likely. **OR** • Continuing degradation of core cooling based on Fission Product Barrier monitoring as indicated by CETs ≥ 700 °F.	Loss of all offsite and all onsite AC power to Vital 4.16 KV busses ≥ 15 minutes **Emergency Action Level(s):** **NOTE:** *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.* 1. Loss of all offsite and all onsite AC power to Vital 4.16 KV busses ≥ 15 minutes.	AC power capability to Vital 4.16 KV busses reduced to a single power source ≥ 15 minutes such that any additional single power source failure would result in station blackout **Emergency Action Level(s):** **NOTE:** *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.* 1. a. AC power capability to Vital 4.16 KV busses reduced to a single power source ≥ 15 minutes. **AND** b. Any additional single power source failure will result in station blackout.	Loss of all offsite AC power to Vital 4.16 KV busses ≥ 15 minutes **Emergency Action Level(s):** **NOTE:** *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.* 1. Loss of all offsite AC power to Vital 4.16 KV busses ≥ 15 minutes.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
SYSTEM MALFUNCTION – Failure of Reactor Protection System			
SG3 1 2 Automatic trip and all manual actions fail to shutdown the reactor and indication of an extreme challenge to the ability to cool the core exists **Emergency Action Level(s):** 1. a. An automatic trip failed to shutdown the reactor. **AND** b. All manual actions do not shutdown the reactor as indicated by reactor power ≥ 5%. **AND** c. Either of the following exist or have occurred due to continued power generation: • CET temperatures at or approaching 1200 °F. **OR** • Feedwater flow rate less than: **Unit 1:** 430 gpm **Unit 2:** 485 gpm	**SS3** 1 2 Automatic trip fails to shutdown the reactor and manual actions taken from the reactor control console are not successful in shutting down the reactor **Emergency Action Level(s):** 1. a. An automatic trip failed to shutdown the reactor. **AND** b. Manual actions taken at panel C03 (Unit 1) or panels 2C03/2C14 (Unit 2) do not shutdown the reactor as indicated by reactor power ≥ 5%.	**SA3** 1 2 Automatic trip fails to shutdown the reactor and the manual actions taken from the reactor control console are successful in shutting down the reactor **Emergency Action Level(s):** 1. a. An automatic trip failed to shutdown the reactor as indicated by reactor power ≥ 5%. **AND** b. Manual actions taken at panel C03 (Unit 1) or panels 2C03/2C14 (Unit 2) successfully shutdown the reactor as indicated by reactor power < 5%.	

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
SYSTEM MALFUNCTION – Loss of DC Power			
	SS4 1 2 3 4 Loss of all Vital DC power ≥ 15 minutes **Emergency Action Level(s):** **NOTE:** *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.* 1. < 105 volts on all Vital DC busses ≥ 15 minutes.		

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
SYSTEM MALFUNCTION – Loss of Annunciators			
	SS6 1 2 3 4 Inability to monitor a SIGNIFICANT TRANSIENT in progress **<u>Emergency Action Level(s):</u>** **NOTE:** *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.* 1. a. UNPLANNED loss of > approximately 75% of the following ≥ 15 minutes: • Control Room annunciators associated with safety systems. **<u>OR</u>** • Control Room safety system indication. **<u>AND</u>** b. A SIGNIFICANT TRANSIENT in progress. **<u>AND</u>** c. Compensatory indications are unavailable.	**SA6** 1 2 3 4 UNPLANNED loss of safety system annunciation or indication in the Control Room with either (1) a SIGNIFICANT TRANSIENT in progress, or (2) compensatory indicators unavailable **<u>Emergency Action Level(s):</u>** **NOTE:** *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.* 1. a. UNPLANNED loss of > approximately 75% of the following ≥ 15 minutes: • Control Room annunciators associated with safety systems. **<u>OR</u>** • Control Room safety system indication. **<u>AND</u>** b. Either of the following: • A SIGNIFICANT TRANSIENT is in progress **<u>OR</u>** • Compensatory indications are unavailable	**SU6** 1 2 3 4 UNPLANNED loss of safety system annunciation or indication in the Control Room for ≥ 15 minutes **<u>Emergency Action Level(s):</u>** **NOTE:** *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.* 1. UNPLANNED loss of > approximately 75% of the following ≥ 15 minutes: a. Control Room annunciators associated with safety systems. **<u>OR</u>** b. Control Room safety system indication.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
SYSTEM MALFUNCTION – RCS Leakage			
			SU7 1 2 3 4 RCS leakage **Emergency Action Level(s):** 1. Unidentified or pressure boundary leakage > 10 gpm. **OR** 2. Identified leakage > 25 gpm.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
SYSTEM MALFUNCTION – Loss of Communications			
			SU8 1 2 3 4 Loss of all onsite or offsite communications capabilities **Emergency Action Level(s):** 1. Loss of all Table M1 onsite communications methods affecting the ability to perform routine operations. **Table M1 Onsite Communications Methods** Station radio system Plant paging system In-plant telephones Gaitronics **OR** 2. Loss of all Table M2 offsite communications methods affecting the ability to perform offsite notifications. **Table M2 Offsite Communications Methods** All telephone lines (commercial and microwave) ENS

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
SYSTEM MALFUNCTION – Fuel Clad Degradation			
			SU9 1 2 3 4 Fuel clad degradation **Emergency Action Level(s):** 1. Failed Fuel Iodine radiation monitor reading indicates fuel clad degradation > Technical Specification allowable limits: **Unit 1:** RI-1237S reads > 1.3 x 10^5 cpm **Unit 2:** 2RITS-4806B reads > .65 x 10^5 cpm **OR** 2. RCS sample activity value indicating fuel clad degradation > Technical Specification allowable limits: • > 1.0 uCi/gm Dose Equivalent I-131 for more than 48 hours **OR** • **Unit 1:** ≥ 60 uCi/gm Dose Equivalent I-131 **Unit 2:** > 60 uCi/gm Dose Equivalent I-131 **OR**

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
SYSTEM MALFUNCTION – Fuel Clad Degradation			
			SU9 *(continued)* • **Unit 1:** > 2200 µCi/gm Dose Equivalent Xe-133 for more than 48 hours • **Unit 2:** > 3100 µCi/gm Dose Equivalent Xe-133 for more than 48 hours
SYSTEM MALFUNCTION – Inadvertant Criticality			
			SU10 3 4 Inadvertent criticality **Emergency Action Level(s):** 1. An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.
SYSTEM MALFUNCTION – Failure to Shutdown			
			SU11 1 2 3 4 Inability to reach required operating mode within Technical Specification limits **Emergency Action Level(s):** 1. A Plant is not brought to required operating mode within Technical Specifications LCO action statement time.

Attachment 2

Arkansas Nuclear One

EAL Basis Document

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS
AU1

Initiating Condition – NOTIFICATION OF UNUSUAL EVENT

Any release of gaseous or liquid radioactivity to the environment >2 times the ODCM limits for ≥60 minutes

Operating Mode Applicability: All

Example Emergency Action Level(s): (1 or 2 or 3)

Note: *The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.*

1. VALID reading on Channel 7 on any of the following radiation monitors > the reading shown for ≥ 60 minutes:

MONITORS – UNIT 1		LIMIT
RX-9820	Containment Purge	1.18E-02 µCi/cc
RX-9825	Radwaste Area	1.07E-02 µCi/cc
RX-9830	Fuel Handling Area	9.08E-03 µCi/cc
RX-9835	Emergency Penetration Room	1.91E-01 µCi/cc
MONITORS – UNIT 2		**LIMIT**
2RX-9820	Containment Purge	8.92E-03 µCi/cc
2RX-9825	Radwaste Area	6.64E-03 µCi/cc
2RX-9830	Fuel Handling Area	8.92E-03 µCi/cc
2RX-9835	Emergency Penetration Room	1.77E-01 µCi/cc
2RX-9840	Post Accident Sampling Building	8.84E-02 µCi/cc
2RX-9845	Aux. Building Extension	2.53E-02 µCi/cc
2RX-9850	Low Level Radwaste Storage Bldg.	3.54E-02 µCi/cc

OR

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS
AU1

2. VALID reading on any of the following radiation monitors >2 times the alarm setpoint established by a current release permit for ≥60 minutes.

EFFLUENT MONITORS – Unit 1	
RX-9820	Containment Purge (Channel 7 or 9)
RE-4830	Waste Gas Radiation Monitor
RE-4642	Liquid Radwaste Monitor
EFFLUENT MONITORS – Unit 2	
2RX-9820	Containment Purge (Channel 7 or 9)
2RE-2429	Waste Gas Decay Tank Vent Line Radiation Monitor
2RE-2330	BMS Liquid Discharge Monitor
2RE-4423	Regenerative Waste Discharge Monitor
2RE-4425	SG Blowdown to Flume Radiation Monitor

OR

3. Confirmed grab sample analyses for gaseous or liquid releases indicates concentrations or release rates >2 times the applicable values of the ODCM for ≥60 minutes.

Basis:

The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This IC addresses a potential reduction in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

ANO 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. 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.

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 IC.

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS
AU1

This Initiating Condition includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit.

<u>EAL #1</u>

This EAL addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed the threshold identified in the EAL.

This EAL is intended for sites that have established effluent monitoring on non-routine release pathways for which a discharge permit would not normally be prepared.

<u>EAL #2</u>

This EAL addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed the threshold identified in this Initiating Condition established by the release permit. This value may be associated with a planned batch release, or a continuous release path.

<u>EAL #3</u>

This EAL addresses uncontrolled releases that are detected by sample analyses, particularly on unmonitored pathways, e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, lake, etc.

EAL #1 and #2 directly correlate with the IC since annual average meteorology is required to be used in showing compliance with the ODCM and is used in calculating the alarm setpoints.

<u>Reference Documents</u>:

1. 1604.051, “Eberline Radiation Monitor System”
2. Offsite Dose Calculation Manual

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS
AU2

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

UNPLANNED rise in plant radiation levels

Operating Mode Applicability: All

Example Emergency Action Level(s): (1 or 2)

1. a. UNPLANNED lowering of water level in the refueling canal or spent fuel pool as indicated by:

   - Personnel observation, refueling crew report, indication on area security camera, borated water source (BWST or RWT) level drop due to makeup demands.

   AND

   b. VALID Area Radiation Monitor reading rise on any of the following:

Unit 1	
RE-8009	Spent Fuel Area
RE-8017	Fuel Handling Area
Unit 2	
2RE-8914	Spent Fuel Area
2RE-8915	Spent Fuel Area
2RE-8916	Spent Fuel Area
2RE-8912	Containment Incore Instrumentation

OR

2. UNPLANNED VALID Area Radiation Monitor readings or survey results indicate a rise by a factor of 1000 over normal* levels.

Note: *For area radiation monitors with ranges incapable of measuring 1000 times normal* levels, classification shall be based on VALID full scale indication unless surveys confirm that area radiation levels are below 1000 times normal* within 15 minutes of the Area Radiation Monitor indications going to full scale indication.*

* Normal can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS
AU2

Basis:

This IC addresses elevated radiation levels as a result of lowered water level above irradiated fuel or events that have resulted, or may result, in UNPLANNED rises in radiation dose rates within plant buildings. These radiation rises represent a loss of control over radioactive material and represent a potential degradation in the level of safety of the plant.

EAL #1

The refueling pathway is a site specific combination of cavities, tubes, canals and pools. While a radiation monitor could detect a rise 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, a refueling bridge ARM reading may rise due to planned evolutions such as head lift, or even a fuel assembly being raised in the manipulator mast. Also, a 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. Generally, elevated 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 RPV 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

This EAL addresses rises in plant radiation levels that represent a loss of control of radioactive material resulting in a potential degradation in the level of safety of the plant.

This EAL excludes radiation level rises that result from planned activities such as use of radiographic sources and movement of radioactive waste materials. A specific list of ARMs is not required as it would restrict the applicability of the Threshold. The intent is to identify loss of control of radioactive material in any monitored area.

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS
AA1

Initiating Condition - ALERT

Any release of gaseous or liquid radioactivity to the environment >200 times the ODCM limits for ≥15 minutes

Operating Mode Applicability: All

Example Emergency Action Level(s): (1 or 2 or 3)

Note: *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.*

1. VALID reading on Channel 7 on any of the following radiation monitors > the reading shown for ≥ 15 minutes:

MONITORS – UNIT 1		LIMIT
RX-9820	Containment Purge	1.18E+00 µCi/cc
RX-9825	Radwaste Area	1.07E+00 µCi/cc
RX-9830	Fuel Handling Area	9.08E-01 µCi/cc
RX-9835	Emergency Penetration Room	1.91E+01 µCi/cc
MONITORS – UNIT 2		**LIMIT**
2RX-9820	Containment Purge	8.92E-01 µCi/cc
2RX-9825	Radwaste Area	6.64E-01 µCi/cc
2RX-9830	Fuel Handling Area	8.92E-01 µCi/cc
2RX-9835	Emergency Penetration Room	1.77E+01 µCi/cc
2RX-9840	Post Accident Sampling Building	8.84E+00 µCi/cc
2RX-9845	Aux. Building Extension	2.53E+00 µCi/cc
2RX-9850	Low Level Radwaste Storage Bldg.	3.54E+00 µCi/cc

<u>OR</u>

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS
AA1

2. **EITHER** VALID reading on any of the following radiation monitors > 200 times the alarm setpoint established by a current release permit for ≥ 15 minutes **OR** VALID reading greater than the value listed for ≥ 15 minutes.

MONITORS – UNIT 1		LIMIT
RX-9820	Containment Purge (Channel 7 or 9)	N/A
RE-4830	Waste Gas Radiation Monitor	9.5E7 cpm
RE-4642	Liquid Radwaste Monitor	9.5E7 cpm
MONITORS – UNIT 2		**LIMIT**
2RX-9820	Containment Purge (Channel 7 or 9)	N/A
2RE-2429	Waste Gas Monitoring System	9.5E5 cpm
2RE-2330	BMS Liquid Discharge Monitor	9.5E5 cpm
2RE-4423	Regenerative Waste Discharge Monitor	9.5E5 cpm
2RE-4425	SG Blowdown to Flume Radiation Monitor	9.5E5 cpm

OR

3. Confirmed grab sample analyses for gaseous or liquid releases indicates concentrations or release rates > 200 times the applicable values of the ODCM for ≥ 15 minutes.

Basis:

The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This IC addresses an actual or substantial potential reduction in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time. ANO 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. 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.

Releases should <u>not</u> be prorated or averaged. For example, a release exceeding 600 times ODCM limits for 5 minutes does not meet the threshold for this IC.

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS

AA1

This Initiating Condition includes any release for which a release permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit.

EAL #1

This EAL addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed the threshold identified in the Initiating Condition.

This EAL is intended for sites that have established effluent monitoring on non-routine release pathways for which a discharge permit would not normally be prepared.

EAL #2

This EAL addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed the threshold identified in this Initiating Condition established by the radioactivity discharge permit. This value may be associated with a planned batch release, or a continuous release path. The limit values provided are for those cases in which the maximum monitor range is less than the release permit value multiplied by 200.

EAL #3

This EAL addresses uncontrolled releases that are detected by sample analyses, particularly on unmonitored pathways, e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, lake, etc.

EAL #1 and #2 directly correlate with the IC since annual average meteorology is required to be used in showing compliance with the ODCM and is used in calculating the alarm setpoints.

Reference Documents:

1. 1604.051, “Eberline Radiation Monitor System”
2. Offsite Dose Calculation Manual

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS
AA2

Initiating Condition - ALERT

Damage to irradiated fuel or loss of water level that has resulted or will result in the uncovering of irradiated fuel outside the reactor vessel

Operating Mode Applicability: All

Example Emergency Action Level(s): (1 or 2)

1. A water level drop in the refueling canal or spent fuel pool that will result in irradiated fuel becoming uncovered.

   <u>OR</u>

2. VALID alarm on any of the following radiation monitors due to damage to irradiated fuel or loss of water level.

Unit 1	
RX-9820	Containment Purge (Channel 7 or 9)
RX-9825	Radwaste Area (Channel 7 or 9)
RX-9830	Fuel Handling Area (Channel 7 or 9)
RE-8060	Containment High Range Radiation Monitors
RE-8061	Containment High Range Radiation Monitors
RE-8009	Spent Fuel Area
RE-8017	Fuel Handling
Unit 2	
2RX-9820	Containment Purge (Channel 7 or 9)
2RX-9825	Radwaste Area (Channel 7 or 9)
2RX-9830	Fuel Handling Area (Channel 7 or 9)
2RE-8905	Containment Equipment Hatch Area
2RE-8909	Containment Personnel Access Area
2RE-8925-1	Containment High Range Radiation Monitors
2RE-8925-2	Containment High Range Radiation Monitors
2RE-8914	Spent Fuel Area
2RE-8915	Spent Fuel Area
2RE-8916	Spent Fuel Area
2RE-8912	Containment Incore Inst.

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS
AA2

Basis:

This IC addresses rises 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 an actual or substantial potential 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. This IC applies to spent fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage.

<u>EAL #1</u>

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 drop in applicable borated water storage tank level. Video cameras (Security or outage-related) may allow remote observation of level.

<u>EAL #2</u>

This EAL addresses radiation monitor indications of fuel uncovery and/or fuel damage.

Elevated ventilation monitor readings may be indication of a radioactivity release from the fuel, confirming that damage has occurred. Elevated background at the ventilation monitor due to water level drop may mask elevated ventilation exhaust airborne activity and needs to be considered.

While a radiation monitor could detect a rise in dose rate due to a drop in the water level, it might <u>not</u> be a reliable indication of whether or <u>not</u> the fuel is covered.

For example, a refueling bridge ARM reading may rise due to planned evolutions such as head lift, or even a fuel assembly being raised in the manipulator mast. Also, a 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. Generally, elevated radiation monitor indications will need to be combined with another indicator (or personnel report) of water loss.

Escalation of this emergency classification level, if appropriate, would be based on AS1 or AG1.

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS
AA3

Initiating Condition - ALERT

Rise in radiation levels within the facility that impedes operation of systems required to maintain plant safety functions

Operating Mode Applicability: All

Example Emergency Action Level(s):

Dose rate > 15 mR/hr in any of the following areas requiring continuous occupancy to maintain plant safety functions:

- Unit 1 Control Room
- Unit 2 Control Room
- Central Alarm Station

Basis:

This IC addresses elevated radiation levels that impact continued operation in areas requiring continuous occupancy to maintain safe operation or to perform a safe shutdown.

The cause and/or magnitude of the rise in radiation levels is <u>not</u> a concern of this IC. The SM/ED must consider the source or cause of the elevated radiation levels and determine if any other IC may be involved.

This IC is not meant to apply to rises in the containment dome radiation monitors as these are events which are addressed in the fission product barrier matrix EALs.

Areas requiring continuous occupancy include the Control Rooms and the Central Alarm Station.

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS
AS1

Initiating Condition -- SITE AREA EMERGENCY

Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity >100 mR TEDE or 500 mR child thyroid CDE for the actual or projected duration of the release

Operating Mode Applicability: All

Example Emergency Action Level(s): (1 or 2 or 3)

Note: *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, the classification should be based on EAL #2 instead of EAL #1. Do not delay declaration awaiting dose assessment results.*

1. VALID reading on Channel 9 on any of the following radiation monitors > the reading shown for ≥ 15 minutes:

MONITORS – UNIT 1		**LIMIT**
RX-9820	Containment Purge	1.18E+01 µCi/cc
RX-9825	Radwaste Area	1.07E+01 µCi/cc
RX-9830	Fuel Handling Area	9.08E+00 µCi/cc
RX-9835	Emergency Penetration Room	1.91E+02 µCi/cc
MONITORS – UNIT 2		**LIMIT**
2RX-9820	Containment Purge	8.92E+00 µCi/cc
2RX-9825	Radwaste Area	6.64E+00 µCi/cc
2RX-9830	Fuel Handling Area	8.92E+00 µCi/cc
2RX-9835	Emergency Penetration Room	1.77E+02 µCi/cc
2RX-9840	Post Accident Sampling Building	8.84E+01 µCi/cc
2RX-9845	Aux. Building Extension	2.53E+01 µCi/cc

OR

2. Dose assessment using actual meteorology indicates doses > 100 mR TEDE or 500 mR child thyroid CDE at or beyond the site boundary.

OR

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS
AS1

3. Field survey results indicate closed window dose rates >100 mR/hr expected to continue for ≥60 minutes; or analyses of field survey samples indicate child thyroid CDE >500 mR for one hour of inhalation, at or beyond the site boundary.

Basis:

This IC addresses radioactivity releases that result in doses at or beyond the site boundary that exceed 10% 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.

<u>EAL #1</u>

The monitor list in EAL #1 includes monitors on all potential release pathways (plant stack, primary-secondary leak, fuel handling accident).

<u>EAL #2</u>

Since dose assessment in EAL #2 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, emergency implementing procedures should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EALs.

<u>EAL #3</u>

Field team surveys in EAL #3 should be performed at or beyond the SITE BOUNDARY 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. 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.

<u>Reference Documents:</u>

1. 1604.051, “Eberline Radiation Monitor System”
2. Offsite Dose Calculation Manual

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS
AG1

Initiating Condition -- GENERAL EMERGENCY

Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity >1000 mR TEDE or 5000 mR child thyroid CDE for the actual or projected duration of the release using actual meteorology

Operating Mode Applicability: All

Example Emergency Action Level(s): (1 or 2 or 3)

Note: *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, the classification should be based on EAL #2 instead of EAL #1. Do not delay declaration awaiting dose assessment results.*

1. VALID reading on Channel 9 on any of the following radiation monitors > the reading shown for ≥ 15 minutes:

MONITORS – UNIT 1		LIMIT
RX-9820	Containment Purge	1.18E+02 (µCi/cc)
RX-9825	Radwaste Area	1.07E+02 (µCi/cc)
RX-9830	Fuel Handling Area	9.08E+01 (µCi/cc)
RX-9835	Emergency Penetration Room	1.91E+03 (µCi/cc)
MONITORS – UNIT 2		**LIMIT**
2RX-9820	Containment Purge	8.92E+01 (µCi/cc)
2RX-9825	Radwaste Area	6.64E+01 (µCi/cc)
2RX-9830	Fuel Handling Area	8.92E+01 (µCi/cc)
2RX-9835	Emergency Penetration Room	1.77E+03 (µCi/cc)
2RX-9840	Post Accident Sampling Building	8.84E+02 (µCi/cc)
2RX-9845	Aux. Building Extension	2.53E+02 (µCi/cc)

OR

2. Dose assessment using actual meteorology indicates doses >1000 mR TEDE or 5000 mR child thyroid CDE at or beyond the site boundary.

OR

3. Field survey results indicate closed window dose rates >1000 mR/hr expected to continue for ≥60 minutes; or analyses of field survey samples indicate child thyroid CDE >5000 mR for one hour of inhalation, at or beyond the site boundary.

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS
AG1

Basis:

This IC addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage.

EAL #1

The monitor list in EAL #1 includes monitors on all potential release pathways (plant stack, primary-secondary leak, fuel handling accident).

EAL #2

Since dose assessment in EAL #2 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, emergency implementing procedures should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EALs.

EAL #3

Field team surveys in EAL #3 should be performed at or beyond the SITE BOUNDARY 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. 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.

Reference Documents:

1. 1604.051, “Eberline Radiation Monitor System”

2. Offsite Dose Calculation Manual

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)

Example Emergency Action Level(s):

Note: *The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. RCS leakage results in the inability to maintain or restore level within Pressurizer or RCS level target band for ≥15 minutes.

Basis:

This IC is considered to be a potential degradation of the level of safety of the plant. The inability to maintain or restore level is indicative of loss of RCS inventory.

Relief valve normal operation should be excluded from this IC. However, a relief valve that operates and fails to close per design should be considered applicable to this IC if the relief valve cannot be isolated.

Prolonged loss of RCS Inventory may result in escalation to the Alert emergency classification level via either CA1 or CA3.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CU2

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

UNPLANNED loss of RCS / reactor vessel inventory

Operating Mode Applicability: Refueling (Mode 6)

Example Emergency Action Level(s): (1 or 2)

Note: *The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. UNPLANNED RCS / reactor vessel level drop as indicated by either of the following:

   a. RCS / reactor vessel water level drop below the reactor vessel flange for ≥15 minutes when the RCS / reactor vessel level band is established above the reactor vessel flange

   OR

   b. RCS / reactor vessel water level drop below the RCS / reactor vessel level band for ≥15 minutes or longer when the RCS / reactor vessel level band is established below the reactor vessel flange.

OR

2. RCS / reactor vessel level cannot be monitored with a loss of RCS / reactor vessel inventory as indicated by an unexplained level rise in (as applicable) the Reactor Building Sump, Reactor Drain Tank, Aux. Building Equipment Drain Tank, Aux. Building Sump, or Quench Tank.

Basis:

This IC is a precursor of more serious conditions and considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that lower RCS water level below the reactor vessel flange are carefully planned and procedurally controlled. An UNPLANNED event that results in water level dropping below the reactor vessel flange, or below the planned RCS water level for the given evolution (if the planned RCS water level is already below the reactor vessel flange), warrants declaration of an NUE due to the lowered RCS 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 RCS Inventory will result in escalation to the Alert emergency classification level via either CA1 or CA3.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CU2

EAL #1

This EAL involves a drop 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 drops in flooded reactor cavity level, which is addressed by AU2 EAL1, until such time as the level drops to the level of the vessel flange.

If reactor vessel level continues to drop and reaches the Bottom ID of the RCS Loop then escalation to CA1 would be appropriate.

EAL #2

This EAL addresses conditions in the refueling mode when 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 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 rises 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 the Alert emergency classification level would be via either CA1 or CA3.

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)

Example Emergency Action Level(s): (1 or 2)

Note: *The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. UNPLANNED event results in RCS temperature exceeding 200 °F.

   OR

2. Loss of all RCS temperature and RCS/reactor vessel level indication for ≥15 minutes.

Basis:

This IC is 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. Since the RCS usually remains intact in the cold shutdown mode a large inventory of water is available to keep the core covered.

During refueling the level in the reactor vessel will normally be maintained above the reactor vessel flange. Refueling evolutions that lower water level below the reactor vessel flange are carefully planned and procedurally controlled. Loss of forced decay heat removal at lowered inventory may result in more rapid rises in RCS/reactor vessel temperatures depending on the time since shutdown.

Normal means of core temperature indication and RCS level indication may not be available in the refueling mode. Redundant means of reactor vessel level indication are therefore procedurally installed 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 of refueling modes, EAL 2 would result in declaration of an NUE if both temperature and 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 its temperature criteria.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CU5

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

AC power capability to Vital 4.16 KV busses reduced to a single power source ≥15 minutes such that any additional single power source failure would result in station blackout

Operating Mode Applicability: Cold Shutdown (Mode 5)
Refueling (Mode 6)

Example Emergency Action Level(s):

Note: *The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. a. AC power capability to Vital 4.16 KV busses reduced to a single power source ≥ 15 minutes.

   AND

   b. Any additional single power source failure will result in station blackout.

Basis:

The condition indicated by this IC is the degradation of the offsite and onsite AC power systems such that any additional single power source failure would result in a station blackout. This condition could occur due to a loss of offsite power with a concurrent failure of all but one emergency generator to supply power to its emergency busses. The subsequent loss of this single power source would escalate the event to an Alert in accordance with CA5.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of power.

The EAL allows credit for operation of the Alternate AC Diesel Generator.

Reference Documents:

1. 1202.007, "*Degraded Powe*r"
2. 1202.008, "*Blackout"*
3. 2202.007, "*Loss of Off-Site Power"*
4. 2202.008, "*Station Blackout"*
5. 2104.037, "*Alternate AC Diesel Generator Operations"*

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CU6

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

Loss of required DC power ≥15 minutes

Operating Mode Applicability: Cold Shutdown (Mode 5)
Refueling (Mode 6)

Example Emergency Action Level(s):

Note: *The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. < 105 volts on required Vital DC bus ≥15 minutes.

Basis:

The purpose of this IC and its associated EALs is to recognize a loss of DC power compromising the ability to monitor and control the removal of decay heat during Cold Shutdown or Refueling operations.

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, the escalation to an Alert will be per CA3.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CU7

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

Inadvertent criticality

Operating Mode Applicability: Cold Shutdown (Mode 5)
Refueling (Mode 6)

Example Emergency Action Level(s):

1. UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

Basis:

This IC addresses criticality events that occur in Cold Shutdown or Refueling modes such as fuel mis-loading events and inadvertent dilution events. This IC indicates a potential degradation of the level of safety of the plant, warranting an NUE classification.

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 fuel bundle or control rod movements during core alteration. These short term positive startup rates are the result of the rise in neutron population due to subcritical multiplication.

Escalation would be by SM judgment.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CU8

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

Loss of all onsite or offsite communications capabilities

Operating Mode Applicability: Cold Shutdown (Mode 5)
Refueling (Mode 6)
Defueled

Example Emergency Action Level(s): (1 or 2)

1. Loss of all Table C2 onsite communication methods affecting the ability to perform routine operations.

   OR

2. Loss of all Table C3 offsite communication methods affecting the ability to perform offsite notifications.

Table C2 Onsite Communications Methods
Station radio system
Plant paging system
In-plant telephones
Gaitronics

Table C3 Offsite Communications Methods
All telephone lines (commercial and microwave)
ENS

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 issues with offsite authorities. The loss of off-site 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 federal, state, and local authorities of plant issues. 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.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CA1

Initiating Condition - ALERT

Loss of RCS / reactor vessel inventory

Operating Mode Applicability: Cold Shutdown (Mode 5)
Refueling (Mode 6)

Example Emergency Action Level(s): (1 or 2)

Note: *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. Loss of RCS / reactor vessel inventory as indicated by:

   Unit 1: RVLMS Levels 1 through 8 indicate DRY

   Unit 2: RVLMS Levels 1 through 5 indicate DRY

   <u>OR</u>

   Unit 1: Reactor vessel level < 368 ft., 0 in. (bottom of the hot leg)

   Unit 2: Reactor vessel level < 369 ft., 1.5 in. (bottom of the hot leg)

   <u>OR</u>

2. RCS / reactor vessel level cannot be monitored for ≥15 minutes with a loss of RCS / reactor vessel inventory as indicated by an unexplained level rise in (as applicable) the Reactor Building Sump, Reactor Drain Tank, Aux. Building Equipment Drain Tank, Aux. Building Sump, or Quench Tank.

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 lowering and potential core uncovery. This condition will result in a minimum emergency classification level of an Alert.

<u>EAL #1</u>

The bottom of the RCS hot leg penetration into the reactor vessel is approximately RLVMS Level 8 (Unit 1) or RVLMS Level 5 (Unit 2). However, RVLMS may not be available in mode 6. Redundant means level indication is provided in this mode and included in EAL #1. The bottom of the RCS hot leg penetration into the reactor vessel is 368 ft., 0 in. (Unit 1) or 369 ft., 1.5 in. (Unit 2). Below this level, reactor vessel level indication will be lost and loss of suction to decay heat removal systems will occur. The inability to restore and maintain level after reaching this setpoint would be indicative of a failure of the RCS barrier.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CA1

EAL #2

In the cold shutdown mode, normal RCS level and reactor vessel level instrumentation systems will usually be available. In the refueling mode, normal means of reactor vessel level indication may not be available. Redundant means of reactor vessel level indication will usually 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 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 rises 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.

If reactor vessel level continues to lower then escalation to Site Area Emergency will be via CS1.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CA3

Initiating Condition - ALERT

Inability to maintain plant in Cold Shutdown

Operating Mode Applicability: Cold Shutdown (Mode 5)
Refueling (Mode 6)

Example Emergency Action Level(s): (1 or 2)

1. An UNPLANNED event results in RCS temperature >200 °F > the specified duration in Table C1.

Table C1 RCS Reheat Duration Thresholds		
RCS	**Containment Closure**	**Duration**
Intact (but not RCS Lowered Inventory)	N/A	60 minutes*
Not intact or RCS Lowered Inventory	Established	20 minutes*
	Not Established	0 minutes
* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.		

<u>OR</u>

Note: *EAL #2 does not apply in solid plant conditions.*

2. An UNPLANNED event results in RCS pressure rise >10 psi due to a loss of RCS cooling.

Basis:

<u>EAL #1</u>

The RCS Reheat Duration Threshold table addresses complete loss of functions required for core cooling for greater than 60 minutes during refueling and cold shutdown modes when RCS integrity is established. RCS integrity should be considered to be in place when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle dams). The 60 minute time frame should allow sufficient time to restore cooling without there being a substantial degradation in plant safety.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CA3

The RCS Reheat Duration Threshold table also addresses the complete loss of functions required for core cooling for greater than 20 minutes during refueling and cold shutdown modes when CONTAINMENT CLOSURE is established but RCS integrity is not established or RCS inventory is lowered (e.g., mid-loop operation). As discussed above, RCS integrity should be assumed to be in place when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (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.

Finally, the EAL addresses complete loss of functions required for core cooling during refueling and cold shutdown modes when neither CONTAINMENT CLOSURE nor RCS integrity are established.

The (*) indicates that this EAL is not applicable if actions are successful in restoring an RCS heat removal system to operation and RCS temperature is being reduced within the specified time frame.

<u>EAL #2</u>

The 10 psi pressure rise addresses 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 chosen should be 10 psi or the lowest pressure that the site can read on installed Control Board instrumentation that is equal to or greater than 10 psi.

Escalation to Site Area Emergency would be via CS1 should boiling result in significant reactor vessel level loss leading to core uncovery.

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.

The SM / ED must remain alert to events or conditions that lead to the conclusion that exceeding the EAL is IMMINENT. If, in the judgment of the SM / ED, an IMMINENT situation is at hand, the classification should be made as if the threshold has been exceeded.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CA5

Initiating Condition - ALERT

Loss of all offsite and all onsite AC power to Vital 4.16KV busses ≥15 minutes

Operating Mode Applicability: Cold Shutdown (Mode 5)
Refueling (Mode 6)
Defueled

Example Emergency Action Level(s):

Note: *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. Loss of all offsite and all on-site AC power to Vital 4.16KV busses ≥ 15 minutes.

Basis:

Loss of all AC power compromises all plant safety systems requiring electric power including DHR/shutdown cooling, emergency core cooling, containment cooling, spent fuel pool cooling and the ultimate heat sink.

The event can be classified as an Alert when in cold shutdown, refueling, or defueled mode because of the significantly reduced decay heat and lower temperature and pressure, which allow raising the time to restore one of the emergency busses, relative to that specified for the Site Area Emergency EAL.

Escalating to Site Area Emergency, if appropriate, is by Abnormal Radiation Levels/ Radiological Effluent (TAB A) ICs.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CS1

Initiating Condition - SITE AREA EMERGENCY

Loss of RCS / reactor vessel inventory affecting core decay heat removal capability

Operating Mode Applicability: Cold Shutdown (Mode 5)
Refueling (Mode 6)

Example Emergency Action Level(s): (1 or 2)

Note: *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. With CONTAINMENT CLOSURE **not** established:

   Unit 1: RVLMS Levels 1 through 9 indicate DRY

   Unit 2: RVLMS Levels 1 through 6 indicate DRY

   OR

2. With CONTAINMENT CLOSURE established, core exit thermocouples indicate superheat.

   OR

3. RCS / reactor vessel level cannot be monitored for ≥ 30 minutes with a loss of RCS / reactor vessel inventory as indicated by any of the following:

   - Containment High Range Radiation Monitor reading > 10 R/hr
   - Erratic source range monitor indication
   - Unexplained level rise in Reactor Building Sump, Reactor Drain Tank, Quench Tank, Aux. Building Equipment Drain Tank, or Aux. Building Sump.

Basis:

Under the conditions specified by this IC, continued lowering in RCS / reactor vessel level is indicative of a loss of inventory control. Inventory loss may be due to an RCS breach, pressure boundary leakage, or continued boiling in the RPV. Thus, declaration of a Site Area Emergency is warranted.

Escalation to a General Emergency is via CG1 or AG1.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CS1

EAL #3

In the cold shutdown mode, normal RCS level and reactor vessel level instrumentation systems will usually be available. In the refueling mode, normal means of reactor vessel level indication may not be available. Redundant means of reactor vessel level indication will usually 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 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 rises 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 30-minute duration allows sufficient time for actions to be performed to recover inventory control equipment.

As water level in the reactor vessel lowers, the dose rate above the core will rise. The dose rate due to this core shine should result in site specific monitor indication and possible alarm.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CG1

Initiating Condition - GENERAL EMERGENCY

Loss of RCS / reactor vessel inventory affecting fuel clad integrity with containment challenged

Operating Mode Applicability: Cold Shutdown (Mode 5)
Refueling (Mode 6)

Example Emergency Action Level(s):

Note: *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. a. Core exit thermocouples indicate superheat for ≥ 30 minutes.

   AND

   b. Any of the following containment challenge indications:
   - CONTAINMENT CLOSURE not established
   - Explosive mixture inside containment
   - UNPLANNED rise in containment pressure

   OR

2. a. RCS / reactor vessel level cannot be monitored for ≥ 30 minutes with a loss of RCS / reactor vessel inventory as indicated by any of the following:
   - Containment High Range Radiation Monitor reading > 10 R/hr
   - Erratic source range monitor indication
   - Unexplained level rise in Reactor Building Sump, Reactor Drain Tank, Quench Tank, Aux. Building Equipment Drain Tank, or Aux. Building Sump

   AND

   b. Any of the following containment challenge indications:
   - CONTAINMENT CLOSURE not established
   - Explosive mixture inside containment
   - UNPLANNED rise in containment pressure

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CG1

Basis:

This IC represents the inability to restore and maintain reactor vessel level to above the top of active fuel with containment challenged. Fuel damage is probable if reactor vessel level cannot be restored, as available decay heat will cause boiling, further reducing the reactor vessel level. 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. The GE is declared on the occurrence of the loss or IMMINENT loss of function of all three barriers.

A number of variables can have a significant impact on heat removal capability challenging the fuel clad barrier. Examples include mid-loop, reduced level / flange level, head in place, cavity flooded, RCS venting strategy, decay heat removal system design, vortexing pre-disposition, and steam generator U-tube draining.

Analysis indicates that core damage may occur within an hour following continued core uncovery therefore, 30 minutes was conservatively chosen.

If CONTAINMENT CLOSURE is re-established prior to exceeding the 30 minute core uncovery time limit then escalation to GE would not occur.

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.

Sump and tank level rises 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.

In the cold shutdown mode, normal RCS level and reactor vessel level instrumentation systems will usually be available. In the refueling mode, normal means of reactor vessel level indication may not be available. Redundant means of reactor vessel level indication will usually 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 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 rises 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.

As water level in the reactor vessel lowers, the dose rate above the core will rise. The dose rate due to this core shine should result in site specific monitor indication and possible alarm.

<u>Reference Documents</u>:

1. ULD-1-SYS-24, *"Unit 1 Inadequate Core Cooling"*
2. ULD-2-SYS-24, *"Unit 2 Inadequate Core Cooling"*

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 (See Sections 3.4 and 3.8). NUE 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 off-site dose assessments, would require continual assessments of radioactive inventory and containment integrity. Alternatively, if both Fuel Clad and RCS Barrier "Potential Loss" EALs existed, the SM / ED 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.
- The Containment Barrier should not be declared lost or potentially lost based on exceeding Technical Specification action statement criteria, unless there is an event in progress requiring mitigation by the Containment barrier. When no event is in progress (Loss or Potential Loss of either Fuel Clad and/or RCS) the Containment Barrier status is addressed by Technical Specifications.

FISSION PRODUCT BARRIER
FUEL CLAD

Fuel Clad Barrier Emergency Action Levels: FCB1 **OR** FCB2 **OR** FCB3 **OR** FCB4 **OR** FCB5 **OR** FCB6

The Fuel Clad barrier consists of the zircalloy or stainless steel fuel bundle tubes that contain the fuel pellets.

1. Primary Coolant Activity Level (FCB1)

Loss:

1. Coolant activity > 300 µCi/gm dose equivalent I-131 activity by Chemistry sample

OR

2. Radiation levels > 1000 MR/hr

   Unit 1: at SA-229

   Unit 2: at 2TCD-19

Potential Loss: None

Basis:

Loss

The site specific value corresponds to 300 µCi/gm I-131 equivalent. Assessment by the EAL Task Force indicates that 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.

A reading of greater than 1000 mR/hr within at one foot from the RCS sample lines (SA-229 for Unit 1, 2TCD-19 for Unit 2) has been determined to correspond to fuel clad failure of approximately 2-5%, and thus the fuel clad barrier is considered lost. This reading is well above that expected for iodine spikes and thus indicates significant clad damage and thus the fuel clad barrier is considered lost.

Potential Loss

There is no Potential Loss EAL associated with this item.

Reference Documents

1. ANO Calculation 03-E-0002-01, *"Radiation Monitor EAL Setpoints for Fission Product Barrier Degradation"*

FISSION PRODUCT BARRIER
FUEL CLAD

2. Core Exit Thermocouple Readings (FCB2)

Loss: > 1200 °F CET temperature.

Potential Loss:

Unit 1: ICC exists as evidenced by CETs indicating superheated conditions

Unit 2: Average CETs indicate superheat for current RCS pressure

Basis:

Loss

The Loss EAL of > 1200 °F is consistent with NEI 99-01 and corresponds to significant superheating of the coolant.

Potential Loss

The Potential Loss EAL corresponds to a loss of subcooling margin.

Note that the loss or potential loss EAL for this category will occur after a loss of adequate sub-cooling margin, which represents a loss of the RCS barrier in EAL RCB1, and therefore represents the loss of two barriers, resulting in a Site Area Emergency per FS1. Any loss or potential loss of the containment barrier at that point would escalate to a General Emergency.

Reference Documents

1. Unit 1 EOP 1202.005, “Inadequate Core Cooling”
2. Unit 1 EOP 1202.013, “EOP Figures”
3. Unit 2 OP 2202.009, “Functional Recovery”
4. ANO Procedure OP 1302.022, “Core Damage Assessment”
5. CE-NPSD-241, “Development of the Comprehensive Procedure Guideline for Core Damage Assessment,” Task 467
6. BWOG EOP Technical Bases Document, Vol. 3, Chapter III.F

FISSION PRODUCT BARRIER
FUEL CLAD

3. Reactor Vessel Water Level (FCB3)

Loss: None

Potential Loss:

Unit 1: RVLMS Levels 1 through 9 indicate DRY

Unit 2: RVLMS Levels 1 through 7 indicate DRY

Basis:

Loss

There is no Loss EAL associated with this item.

Potential Loss

The Reactor Vessel Level Monitoring Systems at ANO do not provide positive indication of core uncovery. The above core level indication provided is used to monitor the approach to and recovery from ICC conditions, but the CETs are used to identify core uncovery, and are the only positive indication of core uncovery.

Per reference document #1, the reactor vessel level indicators installed in Unit 1 extend from the top of the reactor vessel to the fuel alignment plate, and information in reference document #2 indicates that the lowest sensor is greater than 2 feet above the top of active fuel. If any of the 4 RCPs are running, flow induced turbulence produced by the pumps renders the reactor vessel level indicator readings invalid.

Per reference document #3, only the reactor vessel level indicators above the core are considered part of the ICC monitoring system. Per reference document #4, the lowest sensor above the core, RVLMS LVL 6 on the ICC monitoring panel 2C388, is 47 inches above the top of the core. If any of the 4 RCPs are running, flow induced turbulence produced by the pumps renders the reactor vessel level indicator readings invalid.

For either unit then, should CET indication be unavailable and reactor vessel level indication be unavailable due to RCP operation or any other cause, a degraded ability to monitor the barrier would exist.

Reference Documents:

1. ULD-1-SYS-24, *"Unit 1 Inadequate Core Cooling System"*
2. Calculation 84-EQ-0080-02, *"Loop Error Analysis for Reactor Vessel Level Monitoring System"*
3. ULD-2-SYS-24, *"Unit 2 Inadequate Core Cooling Monitoring System"*
4. Calculation 90-E-0116-01, *"Unit 2 EOP Setpoint Document,"* Setpoint R.3

FISSION PRODUCT BARRIER
FUEL CLAD

4. Containment Radiation Monitoring (FCB4)

Loss: Containment high range radiation monitor reading > 1000 R/hr

Potential Loss: None

Basis:

Loss

The 1000 R/hr reading on the containment high range radiation monitors (RE-8060 or RE-8061 for Unit 1, 2RE-8925-1 or 2RE-8925-2 for Unit 2) is a value which indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the containment.

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 RCS barrier that appropriately escalates the emergency classification to a Site Area Emergency per FS1.

Potential Loss

There is no Potential Loss EAL associated with this item.

Reference Documents:

1. NUREG 1228, *"Source Term Estimation During Incident Response to Severe Nuclear Power Plant Accidents"*
2. ANO Calculation 03-E-0002-01, *"Radiation Monitor EAL Setpoints for Fission Product Barrier Degradation"*

FISSION PRODUCT BARRIER
FUEL CLAD

5. Core Damage Assessment (FCB5)

Loss: At least 5% fuel clad damage as determined from core damage assessment

Potential Loss: None

Basis:

Loss

This level is consistent with other fuel clad barrier loss EALs indicative of significant fuel clad damage, but uses core damage assessment evaluations by Technical Support personnel. The fuel clad barrier is considered lost.

If this determination is made from the high range containment radiation monitor readings, or if accompanied by other indications of a loss or potential loss of the RCS barrier, this EAL condition represents a Site Area Emergency per FS1.

Potential Loss

There is no potential loss EAL associated with this item.

Reference Documents:

1. ANO Procedure OP-1302.022, *"Core Damage Assessment"*

6. Emergency Director Judgment (FCB6)

Any condition in the opinion of the SM / ED 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 SM / ED in determining whether the Fuel Clad barrier is lost or potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in SM / ED judgment that the barrier may be considered lost or potentially lost.

FISSION PRODUCT BARRIER
RCS

RCS Barrier EALs: RCB1 **OR** RCB2 **OR** RCB3 **OR** RCB4

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.

1. RCS Leak Rate (RCB1)

Loss: RCS leak rate > available makeup capacity as indicated by:

Unit 1: Loss of adequate subcooling margin

Unit 2: RCS subcooling (MTS) can NOT be maintained at least 30 °F

Potential Loss:

Unit 1: UNISOLABLE RCS leak > 50 gpm with Letdown isolated

Unit 2: UNISOLABLE RCS leak > 44 gpm with Letdown isolated

Basis:

Loss

This 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.

Potential Loss

This EAL is based on the apparent inability to maintain normal liquid inventory within the Reactor Coolant System (RCS) by normal operation of the Makeup and Purification System (Unit 1) or the Chemical and Volume Control System (Unit 2).

Isolating letdown is a standard abnormal operating procedure action and may prevent unnecessary classifications when a non-RCS leakage path such as a Makeup and Purification System or CVCS leak exists. The intent of this condition is met if attempts to isolate Letdown are NOT successful. Additional charging pumps being required is indicative of a substantial RCS leak.

Reference Documents:

1. Unit 1 EOP 1202.013, Figure 1, *"Saturation and Adequate SCM"*
2. Unit 1 EOP Setpoint Document, Calculation 90-E-0116-07, Setpoint B.19
3. Unit 2 EOP 2202.009, *"Functional Recovery"*
4. Unit 2 EOP Setpoint Document, Calculation 90-E-0116-01
5. Unit 2 SAR Table 9.3-14, Charging Pumps Design Data

FISSION PRODUCT BARRIER
RCS

2. SG Tube Rupture (RCB2)

Loss: SGTR that results in an ECCS (SI) actuation

Potential Loss: None

Basis:

Loss

This EAL addresses the full spectrum of Steam Generator (SG) tube rupture events in conjunction with Containment barrier Loss EALs. It 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 leak rate barrier Potential Loss EAL.

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 per Containment barrier Loss EAL CNB3.

Potential Loss

There is no Potential Loss EAL associated with this item.

3. Containment Radiation Monitoring (RCB3)

Loss: Containment high range radiation monitor reading > 100 R/hr.

Potential Loss: None

Basis

Loss

The 100 R/hr reading on the containment high range radiation monitors (RE-8060 or RE-8061 for Unit 1, 2RE-8925-1 or 2RE-8925-2 for Unit 2) is a value which indicates the release of reactor coolant to the containment.

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 rose to that specified by Fuel Clad barrier EAL, fuel damage would also be indicated.

During the initial fifteen minutes after a thermal event inside containment, the high range radiation monitor readings are considered invalid due to possibility of a transient thermally-induced current.

FISSION PRODUCT BARRIER
RCS

Potential Loss

There is no Potential Loss EAL associated with this item.

Reference Documents:

1. ANO Calculation 03-E-0002-01**, "***Radiation Monitor EAL Setpoints for Fission Product Barrier Degradation"*

4. Emergency Director Judgment (RCB4)

Any condition in the opinion of the SM / ED that indicates Loss or Potential Loss of the RCS Barrier.

Basis:

This EAL addresses any other factors that are to be used by the SM / ED in determining whether the RCS barrier is lost or potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in SM / ED judgment that the barrier may be considered lost or potentially lost.

FISSION PRODUCT BARRIER
CONTAINMENT

Containment Barrier EALs: CNB1 **OR** CNB2 **OR** CNB3 **OR** CNB4 **OR** CNB5 **OR** CNB6 **OR** CNB7

The Containment Barrier includes the containment building and 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.

1. Containment Pressure (CNB1)

Loss:

1. Rapid unexplained drop in containment pressure following an initial rise in containment pressure

   OR

2. Containment pressure or sump level response not consistent with LOCA conditions

Potential Loss:

1. **Unit 1:** Containment pressure > 73.7 PSIA (59 PSIG) and rising

   Unit 2: Containment pressure > 73.7 PSIA (59 PSIG) and rising

   OR

2. Explosive mixture exists inside containment.

   OR

3. a. Containment Pressure > containment spray actuation setpoint

      UNIT 1: 44.7 PSIA (30 PSIG)

      UNIT 2: 23.3 PSIA (8.6 PSIG)

      AND

   b. LESS THAN one full train of spray operating

FISSION PRODUCT BARRIER
CONTAINMENT

Basis:

Loss

Rapid unexplained loss of pressure (i.e., not attributable to containment spray or condensation effects) following an initial pressure rise from a primary or secondary high energy line break indicates a loss of containment integrity. Containment pressure and sump levels should rise as a result of mass and energy release into containment from a LOCA. Thus, sump level or pressure not rising indicates containment bypass and a loss of containment integrity.

This indicator relies on operator recognition of an unexpected response for the condition and therefore, does not have a specific value associated with it. The unexpected response is important because it is the indicator for a containment bypass condition.

Potential Loss 1.

The site specific pressure is based on the containment design pressure.

Potential Loss 2.

Existence of an explosive mixture means a hydrogen and oxygen concentration of at least the lower deflagration limit curve exists. The hydrogen concentration of 4% has been recognized by the NRC staff as a well-established lower flammability limit in air or steam-air atmospheres that is adequately conservative for protecting against an H2 explosion. Hydrogen control systems at ANO are designed and operated as to maintain the containment hydrogen concentration below this level, so that indications of hydrogen concentrations above this are considered a potential challenge to the containment integrity.

Potential Loss 3.

This EAL represents a potential loss of containment in that the containment heat removal/depressurization system (e.g., containment sprays, ice condenser fans, etc., 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.

Reference Documents:

1. Unit 1 OP-1105.003, *“Engineering Safeguards Actuation System”*
2. Unit 1 SAR Sections 1.4.43, 5.2.1.2.1, 14.2.2.5.5.1 (reactor building design pressure)
3. Unit 1 SAR Section 6.6 (Post-Loss of Coolant Accident Hydrogen Control)
4. Unit 1 TS Table 3.3.5-1
5. Unit 2 SAR Section 6.2.5 (Combustible Gas Control In Containment)
6. Unit 2 SAR Section 3.8.1.3.1.D (Containment Design Pressure)
7. Unit 2 TS Table 3.3-4
8. Regulatory Guide 1.7, *“Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident, Rev. 2 1978”*

FISSION PRODUCT BARRIER
CONTAINMENT

2. Core Exit Thermocouple Readings (CNB2)

Loss: None

Potential Loss:

1. a. CETs indicate > 1200 °F

   AND

   b. Restoration procedures not effective within 15 minutes.

OR

2. a. CETs indicate > 700 °F

   AND

   b. RVLMS indicates:

   Unit 1: Levels 1 through 9 DRY

   Unit 2: Levels 1 through 7 DRY

   AND

   c. Restoration procedures not effective within 15 minutes.

Basis:

Loss

There is no Loss EAL associated with this item.

Potential Loss

The conditions in these EALs represent an IMMINENT core melt sequence which, if not corrected, could lead to vessel failure and a higher potential for containment failure. In conjunction with the Core Cooling and RCS Leakage criteria in the Fuel and RCS barrier columns, this threshold would result in the declaration of a General Emergency, i.e., loss of two barriers and the potential loss of a third. If the function restoration procedures are ineffective, there is no "success" path.

The function 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 dropping or if the vessel water level is rising.

FISSION PRODUCT BARRIER
CONTAINMENT

Whether or not the procedures will be effective should be apparent within 15 minutes. The SM / ED should make the declaration as soon as it is determined that the procedures have been, or will be ineffective.

3. SG Secondary Side Release With Primary-to-Secondary Leakage (CNB3)

Loss:

1. RUPTURED steam generator is also FAULTED outside of containment

   OR

2. a. Primary-to-secondary leakrate > 10 gpm

      AND

   b. UNISOLABLE steam release from affected steam generator to the environment

Potential Loss: None

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.

This EAL results in a NUE for smaller breaks that; (1) do not exceed the Normal Makeup Capacity for Unit 1 or the capacity of one charging pump in the normal charging lineup for Unit 2 EAL in RCS leak rate barrier Potential Loss , or (2) do not result in ECCS actuation in RCS SG tube rupture barrier Loss. For larger breaks, RCS barrier threshold criteria would result in an Alert. For SG tube ruptures which may involve multiple steam generators or UNISOLABLE secondary line breaks, this condition would exist in conjunction with RCS barrier conditions and would result in a Site Area Emergency. Escalation to General Emergency would be based on "Potential Loss" of the Fuel Clad Barrier.

Loss 1.

This EAL addresses the condition in which a RUPTURED steam generator is also FAULTED. This condition represents a bypass of the RCS and containment barriers and is a subset of the second threshold. In conjunction with RCS leak rate barrier loss EAL RCB2, this would always result in the declaration of a Site Area Emergency.

FISSION PRODUCT BARRIER
CONTAINMENT

Loss 2.

This EAL addresses SG tube leaks that exceed 10 gpm in conjunction with an UNISOLABLE release path to the environment from the affected steam generator. The threshold for establishing the UNISOLABLE 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., SG tube rupture with concurrent loss of off-site power and the RUPTURED steam generator is required for plant cooldown or a stuck open relief valve). The time it takes to isolate a SG with tube leakage > 10 gpm in accordance with plant specific EOPs is not considered a prolonged release. In this case the SG with tube leakage > 10 gpm with a concurrent loss of offsite power is normally steamed to the environment in a controlled manner to achieve and maintain a RCS Hot Leg temperature below that which corresponds to the Main Steam Safety Valve relief settings. However, if the SG cannot be isolated or if both SGs have tube leakage > 10 gpm, a prolonged release will likely be necessary to support plant cooldown. If the main condenser is available, there may be releases via air ejectors, gland seal exhausters, and other similar controlled, and often monitored, pathways. These pathways do not meet the intent of an UNISOLABLE release path to the environment. These minor releases are assessed using Abnormal Radiation Levels / Radiological Effluent ICs (TAB A).

Potential Loss

There is no Potential Loss EAL associated with this item.

4. Containment Isolation Failure or Bypass (CNB4)

Loss:

1. UNISOLABLE breach of containment

   AND

2. Direct downstream pathway to the environment exists after containment isolation signal

Potential Loss: None

FISSION PRODUCT BARRIER
CONTAINMENT

Basis:

Loss

This EAL addresses incomplete containment isolation that allows a direct release to the environment. A breach of containment has also occurred if an inboard and outboard pair of isolation valves fails to close on an automatic actuation signal or from a manual action in the Control Room and opens a release path to the environment.

The breach is not isolable from the Control Room if an attempt for isolation from the Control Room has been made and was unsuccessful. An attempt for isolation should be made prior to the accident classification. If isolable upon identification, then this Initiating Condition is not applicable.

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 product 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.

Potential Loss

There is no Potential Loss EAL associated with this item.

FISSION PRODUCT BARRIER
CONTAINMENT

5. Containment Radiation Monitoring (CNB5)

Loss: None

Potential Loss:

Containment high range radiation monitor reading > 4000 R/hr

Basis:

Loss

There is no Loss EAL associated with this item.

Potential Loss

The 4000 R/hr reading on the containment high range radiation monitors (RE-8060 or RE-8061 for Unit 1, 2RE-8925-1 or 2RE-8925-2 for Unit 2) 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 off-site 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 SM/ED should declare a General Emergency when this value on the Containment High Range Rad Monitor is exceeded as a loss of two barriers (fuel clad and RCS) and potential loss of the third (containment).

Reference Documents:

1. ANO Calculation 03-E-0002-01, *“Radiation Monitor EAL Setpoints for Fission Product Barrier Degradation”*
2. NUREG 1228, *“Source Term Estimation During Incident Response to Severe Nuclear Power Plant Accidents”*

FISSION PRODUCT BARRIER
CONTAINMENT

6. Other Indications (CNB6)

Elevated readings on the following radiation monitors that indicate loss or potential loss of the Containment barrier:

MONITORS – UNIT 1	
RX-9820	Containment Purge
RX-9825	Radwaste Area
RX-9830	Fuel Handling Area
RX-9835	Emergency Penetration Room
MONITORS – UNIT 2	
2RX-9820	Containment Purge
2RX-9825	Radwaste Area
2RX-9830	Fuel Handling Area
2RX-9835	Emergency Penetration Room
2RX-9840	Post Accident Sampling Building
2RX-9845	Aux. Building Extension

Basis:

This EAL covers other indications that may unambiguously indicate the loss or potential loss of the containment barrier.

7. Emergency Director Judgment (CNB7)

Any condition in the opinion of the SM / ED that indicates Loss or Potential Loss of the Containment Barrier.

Basis:

This EAL addresses any other factors that are to be used by the SM / ED in determining whether the Containment barrier is lost or potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in SM / ED judgment that the barrier may be considered lost or potentially lost.

The Containment barrier should not be declared lost or potentially lost based on exceeding Technical Specification action statement criteria, unless there is an event in progress requiring mitigation by the Containment barrier. When no event is in progress (Loss or Potential Loss of either Fuel Clad and/or RCS) the Containment barrier status is addressed by Technical Specifications

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HU1

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant

Operating Mode Applicability: All

Example Emergency Action Level(s): (1 or 2 or 3)

1. A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by ANO 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 EAL is based on the Safeguards Contingency Plan. The Safeguards Contingency Plan is based on guidance provided in NEI 03-12.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HU1

EAL #2

This EAL 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 NUE.

The determination of “credible” is made through use of information found in the Safeguards Contingency Plan.

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 NUE.

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

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Other conditions exist which in the judgment of the SM warrant declaration of an NUE

Operating Mode Applicability: All

Example Emergency Action Level(s):

1. Other conditions exist which in the judgment of the SM indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

Basis:

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the SM to fall under the NUE emergency classification level.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HU4

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

FIRE within the PROTECTED AREA not extinguished within 15 minutes of detection or EXPLOSION within the PROTECTED AREA

Operating Mode Applicability: All

Example Emergency Action Level(s): (1 or 2)

Note: *The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the duration has exceeded, or will likely exceed, the applicable time.*

1. FIRE in any **Table H1** structure or area not extinguished:

   1) within 15 minutes of Control Room notification

   OR

   2) within 15 minutes of verification of a Control Room FIRE alarm (i.e. Alarm valid until disproved).

OR

2. EXPLOSION within the PROTECTED AREA.

Basis:

This IC addresses the magnitude and extent of FIRES or EXPLOSIONS that may be potentially significant precursors of damage to safety systems. It addresses the FIRE / EXPLOSION, and not the degradation in performance of affected systems that may result.

As used here, detection is visual observation and report by plant personnel or sensor alarm indication.

EAL #1

The 15-minute time period begins with a credible notification that a FIRE is occurring or indication of a fire detection system alarm/actuation. Verification of a fire detection system alarm/actuation includes actions that can be taken within the Control Room or other nearby site specific location to ensure that it is not spurious. An 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).

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HU4

EAL #2

This EAL addresses only those EXPLOSIONS of sufficient force to damage permanent structures or equipment within the PROTECTED AREA.

No attempt is made to assess the actual magnitude of the damage. The occurrence of the EXPLOSION is sufficient for declaration.

The SM also needs to consider any security aspects of the EXPLOSION, if applicable.

Escalation of this emergency classification level, if appropriate, would be based on HA4.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HU4

Table H1
Unit 1

Reactor Building

All Elevations

Aux Building

All Elevations Including Penthouse/MSIV Room
Exceptions: Boric Acid Mix Tank Room (Chem Add Area) 404’ (157-B)
EDG Exhaust Fan area on 386’ (1-E and 2-E)

Turbine Building

All Elevations
Including:
Pipechase under ICW Coolers
CRD Pump Pit / T-28 Room / Area under ICW Pumps

Outside Areas

Manholes adjacent to Startup #2 XFMR (MH-03/MH-04)
Manholes adjacent to Intake Structure (MH-05/MH-06)
Intake Structure (354’ and 366’)
Diesel Fuel Vault
Diesel Fuel Vault Pump Manholes MH-09 and MH-10 (Manhole, MH-09, is located approximately 15 feet northeast of the Unit 1 QCST , Manhole, MH-10, is located approximately 5 feet west of Unit 2 Condensate Storage Tank, 2T-41A)

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HU4

Table H1
Unit 2

Reactor Building

All Elevations

Aux Building

All Elevations including Aux Extensions

Turbine Building

All Elevations

Outside Areas

Intake Structure (354’ and 366’)
Concrete Manhole East, NE of intake
Concrete Manhole East of Turbine building next to train bay
Diesel Fuel Vault
Diesel Fuel Vault Pump Manholes MH-09 and MH-10 (Manhole, MH-09, is located approximately 15 feet northeast of the Unit 1 QCST , Manhole, MH-10, is located approximately 5 feet west of Unit 2 Condensate Storage Tank, 2T-41A)

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HU5

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Release of toxic, corrosive, asphyxiant, or flammable gases deemed detrimental to NORMAL PLANT OPERATIONS.

Operating Mode Applicability: All

Example Emergency Action Level(s): (1 or 2)

1. Toxic, corrosive, asphyxiant or flammable gases in amounts that have or could adversely affect NORMAL PLANT OPERATIONS.

   OR

2. Report by Local, County or State officials for evacuation or sheltering of site personnel based on an offsite event.

Basis:

This IC is based on the release of toxic, corrosive, asphyxiant or flammable gases of sufficient quantity to affect NORMAL PLANT OPERATIONS.

The fact that SCBAs may be worn does not eliminate the need to declare the event.

This IC is not intended to require significant assessment or quantification. It assumes an uncontrolled process that has the potential to affect plant operations. This would preclude small or incidental releases, or releases that do not impact structures needed for plant operation.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death.

Escalation of this emergency classification level, if appropriate, would be based on HA5.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HU6

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Natural or destructive phenomena affecting the PROTECTED AREA

Operating Mode Applicability: All

Example Emergency Action Level: (1 or 2 or 3 or 4 or 5 or 6)

1. Seismic event identified by any 2 of the following:
   - Seismic event confirmed by annunciation of the 0.01g acceleration alarm
   - Earthquake felt in plant
   - National Earthquake Center

   OR

2. Tornado striking within PROTECTED AREA boundary or high winds > 67 mph (**2 minute average**).

   OR

3. Internal flooding that has the potential to affect safety related equipment required by Technical Specifications for the current operating mode in any of the structures or areas in **Table H1 (see Table H1 located in HU4).**

   OR

4. Turbine failure resulting in casing penetration or damage to turbine or generator seals.

   OR

5. Lake Dardanelle level < 335 feet.

   OR

6. Lake Dardanelle level > 345 feet.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HU6

Basis:

These EALs are categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

EAL #1

Damage may be caused to some portions of the site, but should not affect ability of safety functions to operate.

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 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.*

The National Earthquake Center can confirm if an earthquake has occurred in the area of the plant.

EAL #2

This EAL is based on a tornado striking (touching down) or high winds within the PROTECTED AREA.

The high wind value in EAL #2 is conservatively based on the SAR design basis for Unit 1 of 67 mph. Unit 2 Design basis is 80 mph.

Escalation of this emergency classification level, if appropriate, would be based on VISIBLE DAMAGE, or by other in plant conditions, via HA6.

EAL #3

This EAL addresses the effect of internal flooding caused by events such as component failures, equipment misalignment, or outage activity mishaps.

Escalation of this emergency classification level, if appropriate, would be via HA6, or by other plant conditions.

EAL #4

This EAL addresses main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. 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.

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.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HU6

This EAL is consistent with the definition of an NUE while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment.

Escalation of this emergency classification level, if appropriate, would be to HA6 based on damage done by PROJECTILES generated by the failure or in conjunction with a steam generator tube rupture. These latter events would be classified by the radiological (A) ICs or Fission Product Barrier (F) ICs.

EALs #5 and #6

EALs #5 and #6 are based on the levels of Lake Dardanelle at which the site will take specific action to reduce the impact of the lake level on plant safety by initiating plant shutdown.

Reference Documents:

1. OP-1203.025, *"Natural Emergencies"*
2. OP-2203.008, *"Natural Emergencies"*
3. Unit 1 FSAR
4. Unit 2 FSAR

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA1

Initiating Condition - ALERT

HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat

Operating Mode Applicability: All

Example Emergency Action Level(s): (1 or 2)

1. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by ANO 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.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA1

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 Organizations 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 exist which in the judgment of the SM / ED warrant declaration of an Alert

Operating Mode Applicability: All

Example Emergency Action Level(s):

1. Other conditions exist which in the judgment of the SM / ED indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Basis:

This EAL addresses unanticipated conditions not addressed explicitly elsewhere, but that warrant declaration of an emergency because conditions exist which are believed by the SM / ED to fall under the Alert emergency classification level.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HA3

Initiating Condition - ALERT

Control room evacuation has been initiated

Operating Mode Applicability: All

Example Emergency Action Level(s):

1. Alternate Shutdown procedure requires Control Room evacuation:

   Unit 1: 1203.002, “Alternate Shutdown”

   Unit 2: 2203.014, “Alternate Shutdown”

Basis:

With the Control Room evacuated, additional support, monitoring and direction through the Technical Support Center and/or other emergency response facilities may be 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

Example Emergency Action Level(s):

1. FIRE or EXPLOSION resulting in VISIBLE DAMAGE to any Table H1 structure or area containing safety systems or components or Control Room indication of degraded performance of those safety systems.

Basis:

VISIBLE DAMAGE is used to identify the magnitude of the FIRE or EXPLOSION and to discriminate against minor FIRES and EXPLOSIONS.

The reference to structures or areas containing safety systems or components is included to discriminate against FIRES or 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 or EXPLOSION was large enough to cause damage to these systems.

The use of VISIBLE DAMAGE should not be interpreted as mandating a lengthy damage assessment prior to classification. The declaration of an Alert and the activation of the Technical Support Center will provide the SM/ED with the resources needed to perform detailed damage assessments.

The SM / ED also needs to consider any security aspects of the EXPLOSION.

Escalation of this emergency classification level, if appropriate, will be based on System Malfunction (S), Fission Product Barrier Degradation (F) or Abnormal Radiation Levels / Radiological Effluent (A) ICs.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HA4

Table H1
Unit 1

Reactor Building

All Elevations

Aux Building

All Elevations Including Penthouse/MSIV Room
Exceptions: Boric Acid Mix Tank Room (Chem Add Area) 404’ (157-B)
EDG Exhaust Fan area on 386’ (1-E and 2-E)

Turbine Building

All Elevations
Including:
Pipechase under ICW Coolers
CRD Pump Pit / T-28 Room / Area under ICW Pumps

Outside Areas

Manholes adjacent to Startup #2 XFMR (MH-03/MH-04)
Manholes adjacent to Intake Structure (MH-05/MH-06)
Intake Structure (354’ and 366’)
Diesel Fuel Vault
Diesel Fuel Vault Pump Manholes MH-09 and MH-10 (Manhole, MH-09, is located approximately 15 feet northeast of the Unit 1 QCST , Manhole, MH-10, is located approximately 5 feet west of Unit 2 Condensate Storage Tank, 2T-41A)

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HA4

Table H1
Unit 2

Reactor Building

All Elevations

Aux Building

All Elevations including Aux Extensions

Turbine Building

All Elevations

Outside Areas

Intake Structure (354’ and 366’)
Concrete Manhole East, NE of intake
Concrete Manhole East of Turbine building next to train bay
Diesel Fuel Vault
Diesel Fuel Vault Pump Manholes MH-09 and MH-10 (Manhole, MH-09, is located approximately 15 feet northeast of the Unit 1 QCST , Manhole, MH-10, is located approximately 5 feet west of Unit 2 Condensate Storage Tank, 2T-41A)

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HA5

Initiating Condition - ALERT

Access to a VITAL AREA is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of operable equipment required to maintain safe operations or safely shutdown the reactor.

Unit 1

VITAL AREA	APPLICABLE MODES
A-4 Switchgear Room	3, 4
Upper North Electrical Penetration Room	3, 4
Lower South Electrical Equipment Room	3, 4
Control Room	ALL

Unit 2

VITAL AREA	APPLICABLE MODES
Auxiliary Building 317’ Emergency Core Cooling Rooms	3, 4
Auxiliary Building 317’ Tendon Gallery Access	3, 4
Auxiliary Building 335’ Charging Pumps / 2B-52	3, 4
Auxiliary Building 354’ 2B-62 Area	3, 4
Emergency Diesel Generator Corridor	3, 4
Lower South Piping Penetration Room	3, 4
Auxiliary Building 386’ Containment Hatch	3, 4
Control Room	ALL

Operating Mode Applicability: As stated in above tables.

Example Emergency Action Level(s):

Note: *If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.*

1. Access to a VITAL AREA is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shutdown the reactor.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HA5

Basis:

Gases in a VITAL AREA can affect the ability to safely operate or safely shutdown the reactor. The fact that SCBAs may be worn does not eliminate the need to declare the event.

Declaration should not be delayed for confirmation from atmospheric testing if the atmosphere poses an immediate threat to life and health or an immediate threat of severe exposure to gases. This could be based upon documented analysis, indication of personal ill effects from exposure, or operating experience with the hazards.

If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death.

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. 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 assumes concentrations of flammable gasses which can ignite/support combustion.

Escalation of this emergency classification level, if appropriate, will be based on System Malfunction (S), Fission Product Barrier Degradation (F) or Abnormal Radiation Levels / Radioactive Effluent (A) ICs.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HA6

Initiating Condition - ALERT

Natural or destructive phenomena affecting VITAL AREAS

Operating Mode Applicability: All

Example Emergency Action Level(s): (1 or 2 or 3 or 4 or 5 or 6)

1. a. Seismic event > Operating Basis Earthquake (OBE) as indicated by annunciation of the 0.1g acceleration alarm.

   AND

   b. Earthquake confirmed by any of the following:

   - Earthquake felt in plant
   - National Earthquake Center
   - Control Room indication of degraded performance of systems required for the safe shutdown of the plant

   OR

2. Tornado striking or high winds > 67 mph (2 minute average) resulting in VISIBLE DAMAGE to any of the following structures/equipment containing safety systems or components or Control Room indication of degraded performance of those safety systems:

Reactor Building	Turbine Building
Intake Structure	Q Condensate Storage Tank (QCST)
Ultimate Heat Sink	Control Room
Startup Transformers	Auxiliary Building
Diesel Fuel Vault	Borated Water Storage Tank (BWST)
Refueling Water Tank (RWT)	

   OR

3. Internal flooding in any of the following areas resulting in an electrical shock hazard that precludes access to operate or monitor safety equipment or Control Room indication of degraded performance of those safety systems:

   Intake Structure
   Turbine Building
   Auxiliary Building

   OR

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HA6

4. Turbine failure-generated PROJECTILES resulting in VISIBLE DAMAGE to or penetration of any of the structures/equipment containing safety systems or components or Control Room indication of degraded performance of those safety systems:

Control Room	Turbine Building
Startup Transformers	Auxiliary Building

OR

5. Lake Dardanelle level < 335 feet and Emergency Cooling Pond inoperable.

OR

6. Vehicle crash resulting in VISIBLE DAMAGE to any of the structures/equipment containing safety systems or components or Control Room indication of degraded performance of those safety systems:

Reactor Building	Turbine Building
Intake Structure	QCST
Ultimate Heat Sink	RWT
Startup Transformers	Auxiliary Building
Diesel Fuel Vault	BWST

Basis:

These EALs escalate from 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 Room 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 of this emergency classification level, if appropriate, would be based on System Malfunction (S) ICs.

EAL #1

Seismic events of this magnitude can result in a VITAL AREA being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to plant safety systems.

The National Earthquake Center can confirm if an earthquake has occurred in the area of the plant.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HA6

EAL #2

This EAL is based on a tornado striking (touching down) or high winds that have caused VISIBLE DAMAGE to structures containing functions or systems required for safe shutdown of the plant. The high wind value in EAL #2 is conservatively based on the SAR design basis for Unit 1 of 67 mph. Unit 2 Design basis is 80 mph.

EAL #3

This EAL addresses the effect of internal flooding caused by events such as component failures, equipment misalignment, or outage activity mishaps. It is based on the degraded performance of systems, or has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to access, operate or monitor safety equipment represents an actual or substantial potential degradation of the level of safety of the plant.

Flooding as used in this EAL describes a condition where water is entering the room faster than installed equipment is capable of removal, resulting in a rise of water level within the room. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source.

EAL #4

This EAL addresses the threat to safety related equipment imposed by PROJECTILEs generated by main turbine rotating component failures. Therefore, this EAL is consistent with the definition of an ALERT in that the potential exists for actual or substantial potential degradation of the level of safety of the plant.

EAL #5

EAL #5 addresses site specific phenomena which has the potential for the loss of primary and secondary heat sink.

EAL #6

This EAL addresses vehicle crashes within the PROTECTED AREA that result in VISIBLE DAMAGE to VITAL AREAS or indication of damage to safety structures, systems, or components containing functions and systems required for safe shutdown of the plant.

Reference Documents:

1. OP-1203.025, “Natural Emergencies”
2. OP-2203.008, “Natural Emergencies”
3. Unit 1 FSAR
4. Unit 2 FSAR

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HS1

Initiating Condition - SITE AREA EMERGENCY

HOSTILE ACTION within the PROTECTED AREA

Operating Mode Applicability: All

Example Emergency Action Level(s):

1. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by ANO 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 EAL 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 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 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 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 exist which in the judgment of the SM / ED warrant declaration of a Site Area Emergency

Operating Mode Applicability: All

Example Emergency Action Level(s):

1. Other conditions exist which in the judgment of the SM / ED indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

Basis:

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the SM / ED to fall under the emergency classification level description for Site Area Emergency.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HS3

Initiating Condition - SITE AREA EMERGENCY

Control Room evacuation has been initiated and plant control cannot be established

Operating Mode Applicability: All

Example Emergency Action Level(s):

1. a. Control room evacuation has been initiated

   AND

   b. Control of the plant cannot be established in accordance with the following procedures within 15 minutes:

   Unit 1: 1203.002, “Alternate Shutdown”

   Unit 2: 2203.014, “Alternate Shutdown”

Basis:

The intent of this IC is to capture those events where control of the plant cannot be reestablished in a timely manner. In this case, expeditious transfer of control of safety systems has not occurred (although fission product barrier damage may not yet be indicated).

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).

The determination of whether or not control is established is based on SM / ED judgment. The SM / ED is expected to make a reasonable, informed judgment within 15 minutes that the plant staff has control of the plant .

Escalation of this emergency classification level, if appropriate, would be by Fission Product Barrier Degradation (F) or Abnormal Radiation Levels/Radiological Effluent (A) EALs.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HG1

Initiating Condition - GENERAL EMERGENCY

HOSTILE ACTION resulting in loss of physical control of the facility

Operating Mode Applicability: All

Example Emergency Action Level(s): (1 or 2)

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 cannot 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 secondary heat removal (ability to maintain a heat sink).

Loss of physical control of the Control Room or remote shutdown/alternate shutdown capability alone may not prevent the ability to maintain safety functions per se. Design of the remote shutdown/alternate shutdown capability and the location of the transfer switches 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 threshold 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 ANO, 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 exist which in the judgment of the SM / ED warrant declaration of a General Emergency

Operating Mode Applicability: All

Example Emergency Action Level(s):

1. Other conditions exist which in the judgment of the SM / ED indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Basis:

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the SM / ED to fall under the emergency classification level description for General Emergency.

SYSTEM MALFUNCTION

SYSTEM MALFUNCTION
SU1

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Loss of all offsite AC power to Vital 4.16 KV busses ≥ 15 minutes

Operating Mode Applicability: Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Example Emergency Action Level(s):

Note: *The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.*

1. Loss of all offsite AC power to Vital 4.16 KV busses ≥ 15 minutes.

Basis:

Prolonged loss of offsite 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 to emergency busses.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of off-site power.

Reference Documents:

1. 1202.007, “Degraded Power”
2. 1202.008, “Blackout”
3. 2202.007, “Loss of Off-Site Power”
4. 2202.008, “Station Blackout”

SYSTEM MALFUNCTION
SU6

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

UNPLANNED loss of 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)

Example Emergency Action Level(s):

Note: *The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.*

1. UNPLANNED Loss of > approximately 75% of the following $\geq$ 15 minutes:

   a. Control Room annunciators associated with safety systems.

   OR

   b. Control Room safety system indication.

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.

"Planned" loss of annunciators or indicators includes scheduled maintenance and testing activities.

Quantification is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel 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 further recognized that most plant designs provide redundant safety system indication powered from separate uninterruptible power supplies. 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 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, the NUE is based on SU11 "Inability to reach required operating mode within Technical Specification limits."

SYSTEM MALFUNCTION
SU6

Indicators associated with safety systems are those indicators for reactivity control, core cooling, maintaining reactor coolant system integrity or maintaining containment integrity.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

This NUE will be escalated to an Alert based on a concurrent loss of compensatory indications or if a SIGNIFICANT TRANSIENT is in progress during the loss of annunciation or indication (SA6).

Reference Documents:

1. 1203.043, "Loss Control Room Annunciators"
2. 2203.042, "Loss of Control Room Annunciators"

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)

Example Emergency Action Level(s): (1 or 2)

1. Unidentified or pressure boundary leakage > 10 gpm.

   OR

2. Identified leakage > 25 gpm.

Basis:

With respect to this IC, RCS leakage is defined as a loss of RCS inventory due to a leak in the RCS or a supporting system that is not or cannot be isolated within 10 minutes. For example, isolation of the RCS Letdown (purification) system is a standard abnormal operating procedure action and may prevent unnecessary classifications when a non-RCS leakage path leak exists. However, the intent of this condition is met if attempts to isolate the RCS leak are NOT successful.

This IC is included as an NUE 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. The 10 gpm value for the unidentified or 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).

Relief valve normal operation should be excluded from this IC. However, a relief valve that operates and fails to close per design should be considered applicable to this IC if the relief valve cannot be isolated.

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. In either case, escalation of this IC to the Alert level is via Fission Product Barrier Degradation (F) ICs.

SYSTEM MALFUNCTION
SU8

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

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)

Example Emergency Action Level(s): (1 or 2)

1. Loss of all Table M1 onsite communications methods affecting the ability to perform routine operations.

   OR

2. Loss of all Table M2 offsite communications methods affecting the ability to perform offsite notifications.

Table M1 Onsite Communications Methods
Station radio system
Plant paging system
In-plant telephones
Gaitronics

Table M2 Offsite Communications Methods
All telephone lines (commercial and microwave)
ENS

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 issues with offsite authorities.

The availability of one method of ordinary offsite communications is sufficient to inform federal, state, and local authorities of plant problems. This EAL is intended to be used only when extraordinary means (e.g., relaying of information from non-routine radio transmissions, individuals being sent to off-site locations, etc.) are being used to make communications possible.

Reference Documents:

1. 1903.062, “Communications System Operating Procedure”

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)

Example Emergency Action Level(s): (1 or 2)

1. Failed Fuel Iodine radiation monitor reading indicates fuel clad degradation > Technical Specification allowable limits:

   Unit 1:

   RI-1237S reads > 1.3 x 10^5 counts per minute

   Unit 2:

   2RITS-4806B reads > .65 x 10^5 counts per minute

   OR

2. RCS sample activity value indicating fuel clad degradation > Technical Specification allowable limits:

   - > 1.0 uCi/gm Dose Equivalent I-131 for more than 48 hours

     OR

   - **Unit 1:**

     ≥ 60 uCi/gm Dose Equivalent I-131

     Unit 2:

     > 60 uCi/gm Dose Equivalent I-131

     OR

   - **Unit 1:**

     > 2200 µCi/gm Dose Equivalent Xe-133 for more than 48 hours

     Unit 2:

     > 3100 µCi/gm Dose Equivalent Xe-133 for more than 48 hours

SYSTEM MALFUNCTION
SU9

Basis:

This IC is included because it is a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant.

EAL #1

This threshold addresses the Letdown Radiation Monitor readings that provide indication of a degradation of fuel clad integrity.

EAL #2

This EAL addresses coolant samples exceeding coolant technical specifications for transient iodine spiking limits and coolant samples exceeding coolant Technical Specifications for nominal operating limits for the time period specified in the Technical Specifications.

Escalation of this IC to the Alert level is via the Fission Product Barriers (F).

Reference Documents:

1. ANO1 Technical Specifications
2. ANO2 Technical Specifications

SYSTEM MALFUNCTION
SU10

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Inadvertent criticality

Operating Mode Applicability: Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Example Emergency Action Level(s):

1. UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

Basis:

This IC addresses inadvertent criticality events. This IC indicates a potential degradation of the level of safety of the plant, warranting an NUE classification. This IC excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups (e.g., criticality earlier than estimated).

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 for (such as shutdown bank withdrawal). These short term positive startup rates are the result of the rise in neutron population due to subcritical multiplication.

Escalation would be by the Fission Product Barrier Table (F), as appropriate to the operating mode at the time of the event.

Reference Documents:

1. 1203.012G, “Annunciator K08 Corrective Action”
2. 2203.012D, “Annunciator 2K04 Corrective Action”

SYSTEM MALFUNCTION
SU11

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Inability to reach required operating mode within Technical Specification limits

Operating Mode Applicability: Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Example 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 operating 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 four 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 NUE 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 NUE is based on the time at which the LCO-specified action statement time period elapses under the site Technical Specifications and is not related to how long a condition may have existed.

Reference Documents:

1. ANO2 Technical Specifications
2. ANO1 Technical Specifications

SYSTEM MALFUNCTION
SA1

Initiating Condition - ALERT

AC power capability to Vital 4.16 KV busses reduced to a single power source ≥ 15 minutes such that any additional single power source failure would result in station blackout

Operating Mode Applicability: Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Example Emergency Action Level(s):

Note: *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.*

1. a. AC power capability to Vital 4.16 KV busses reduced to a single power source ≥ 15 minutes.

   AND

   b. Any additional single power source failure will result in station blackout.

Basis:

The condition indicated by this IC is the degradation of the offsite and onsite AC power systems such that any additional single power source failure would result in a station blackout. This condition could occur due to a loss of offsite power with a concurrent failure of all but one emergency generator to supply power to its emergency busses. Another related condition could be the loss of all offsite power and loss of onsite emergency generators with only one train of emergency busses being backfed from the unit main generator, or the loss of onsite emergency generators 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**.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of power.

The EAL allows credit for operation of the Alternate AC Diesel Generator.

Reference Documents:

1. 1202.007, “Degraded Power”
2. 1202.008, “Blackout”
3. 2202.007, “Loss of Off-Site Power”
4. 2202.008, “Station Blackout”
5. 2104.037, “Alternate AC Diesel Generator Operations”

SYSTEM MALFUNCTION
SA3

Initiating Condition - ALERT

Automatic trip fails to shutdown the reactor and the manual actions taken from the reactor control console are successful in shutting down the reactor

Operating Mode Applicability: Power Operations (Mode 1)
Startup (Mode 2)

Example Emergency Action Level(s):

1. a. An automatic trip failed to shutdown the reactor as indicated by reactor power ≥ 5%.

   AND

   b. Manual actions taken at the reactor control console successfully shutdown the reactor as indicated by reactor power < 5%.

Basis:

Manual trip actions taken at the reactor control console are any set of actions by the Reactor Operator(s) which causes or should cause control rods to be rapidly inserted into the core and shuts down the reactor. Any action taken to trip the reactor from any location other than panel C03 (Unit 1) or 2C03/2C14 (Unit 2) constitutes a failure of the manual trip function. Failure of manual trip would escalate the event to a Site Area Emergency (**SS3**).

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. Thus the plant safety has been compromised because design limits of the fuel may have been exceeded. An Alert is indicated because conditions may exist that lead to potential loss of fuel clad or RCS and because of the failure of the Reactor Protection System to automatically shutdown the plant. This EAL applies whether or not a mode change has occurred. (Reference "**Operating Mode Applicability"** page 72)

If manual actions taken at the reactor control console fail to shutdown the reactor, the event would escalate to a Site Area Emergency.

SYSTEM MALFUNCTION
SA6

Initiating Condition - ALERT

UNPLANNED loss of safety system annunciation or indication in the Control Room with either (1) a SIGNIFICANT TRANSIENT in progress, or (2) compensatory indicators unavailable

Operating Mode Applicability: Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Example Emergency Action Level(s):

Note: *The SM/ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.*

1. a. UNPLANNED loss of > approximately 75% of the following ≥ 15 minutes:

   - Control Room annunciators associated with safety systems

     OR

   - Control Room safety system indication

   AND

   b. Either of the following:

   - A SIGNIFICANT TRANSIENT is in progress

     OR

   - Compensatory indications are unavailable.

Basis:

This IC is 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 SIGNIFICANT TRANSIENT.

Recognition of the availability of computer based indication equipment is considered (e.g., SPDS, plant computer, etc.).

"Planned" loss of annunciators or indicators includes scheduled maintenance and testing activities.

SYSTEM MALFUNCTION
SA6

Quantification is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel 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 most plant designs provide redundant safety system indication powered from separate uninterruptible power supplies. 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 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, the NUE is based on SU11 "Inability to reach required operating mode within Technical Specification limits."

Indicators associated with safety systems are those indicators for reactivity control, core cooling, maintaining reactor coolant system integrity or maintaining containment integrity.

"Compensatory indications" in this context includes computer based information such as SPDS, QSPDS, COLSS, etc. If both a major portion of the annunciation system and all computer monitoring are unavailable, the Alert is required.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

This Alert will be escalated to a Site Area Emergency if the operating crew cannot monitor the transient in progress due to a concurrent loss of compensatory indications with a SIGNIFICANT TRANSIENT in progress during the loss of annunciation or indication.

<u>Reference Documents</u>:

1. 1015.037, “Post Transient Review”
2. 1203.043, “Loss of Control Room Annunciators”
3. 2203.042, “Loss of Control Room Annunciators”

SYSTEM MALFUNCTION
SS1

Initiating Condition - SITE AREA EMERGENCY

Loss of all offsite and all onsite AC power to Vital 4.16 KV busses ≥ 15 minutes

Operating Mode Applicability: Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Example Emergency Action Level(s):

Note: *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.*

1. Loss of all offsite and all onsite AC power to Vital 4.16 KV busses ≥ 15 minutes.

Basis:

Loss of all AC power to emergency busses 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 to emergency busses will lead to loss of Fuel Clad, RCS, and Containment, thus this event can escalate to a General Emergency.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of offsite power.

Escalation to General Emergency is via Fission Product Barrier Degradation (F) or IC SG1, "Prolonged loss of all offsite and all onsite AC power to Vital 4.16 KV busses."

Reference Documents:

1. 1202.007, “Degraded Power”
2. 1202.008, “Blackout”
3. 2202.007, “Loss of Off-Site Power”
4. 2202.008, “Station Blackout”
5. 2104.037, “Alternate AC Diesel Generator Operations”

SYSTEM MALFUNCTION
SS3

Initiating Condition - SITE AREA EMERGENCY

Automatic trip fails to shutdown the reactor and manual actions taken from the reactor control console are not successful in shutting down the reactor

Operating Mode Applicability: Power Operations (Mode 1)
Startup (Mode 2)

Example Emergency Action Level(s):

1. a. An automatic trip failed to shutdown the reactor.

   AND

   b. Manual actions taken at the reactor control console do not shutdown the reactor as indicated by reactor power ≥ 5%.

Basis:

Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed and efforts to bring the reactor subcritical are unsuccessful. A Site Area Emergency is warranted because conditions exist that lead to IMMINENT loss or potential loss of both fuel clad and RCS.

Manual trip actions taken at the reactor control console are any set of actions by the Reactor Operator(s) which causes or should cause control rods to be rapidly inserted into the core and shuts down the reactor.

Manual trip actions are not considered successful if action away from panel C03 (Unit 1) or panels 2C03/2C14 (Unit 2) is required to trip the reactor. This EAL is still applicable even if actions taken away from panel C03 (Unit 1) or panels 2C03/2C14 (Unit 2) are successful in shutting the reactor down because the design limits of the fuel may have been exceeded or because of the gross failure of the Reactor Protection System to shutdown the plant. This EAL applies whether or not a mode change has occurred. (Reference "**Operating Mode Applicability"** page 72)

Escalation of this event to a General Emergency would be due to a prolonged condition leading to an extreme challenge to either core-cooling or heat removal.

SYSTEM MALFUNCTION
SS4

Initiating Condition - SITE AREA EMERGENCY

Loss of all vital DC power ≥15 minutes

Operating Mode Applicability: Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Example Emergency Action Level(s):

Note: *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.*

1. <105 volts on all Vital DC busses ≥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.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Escalation to a General Emergency would occur by Abnormal Radiation Levels/Radiological Effluent (A), Fission Product Barrier Degradation (F).

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)

Example Emergency Action Level(s):

Note: *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.*

1. a. Loss of > approximately 75% of the following ≥ 15 minutes:

   - Control Room annunciators associated with safety systems

     OR

   - Control Room safety system indication

   AND

   b. A SIGNIFICANT TRANSIENT is in progress.

   AND

   c. Compensatory indications are unavailable.

Basis:

This IC is intended to recognize the threat to plant safety associated with the complete loss of capability of the control room staff to monitor plant response to a SIGNIFICANT TRANSIENT.

"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.

Quantification is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel 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.

SYSTEM MALFUNCTION
SS6

It is further recognized that most plant designs provide redundant safety system indication powered from separate uninterruptible power supplies. 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 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, the NUE is based on SU11 "Inability to reach required operating mode within Technical Specification limits."

A Site Area Emergency is considered to exist if the Control Room staff cannot monitor safety functions needed for protection of the public while a significant transient is in progress.

Site specific indications needed to monitor safety functions necessary for protection of the public must include Control Room indications, computer generated indications and dedicated annunciation capability.

Indicators associated with safety systems are those indicators for reactivity control, core cooling, maintaining reactor coolant system integrity or maintaining containment integrity.

"Compensatory indications" in this context includes computer based information such as SPDS, QSPDS, COLSS, etc. This should include all computer systems available for this use depending on specific plant design and subsequent retrofits.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Reference Documents:

1. 015.037, “Post Transient Review”

2. 1203.043, “Loss of Control Room Annunciators”

3. 2203.042, “Loss of Control Room Annunciators”

SYSTEM MALFUNCTION
SG1

Initiating Condition - GENERAL EMERGENCY

Prolonged loss of all offsite and all onsite AC power to Vital 4.16 KV busses

Operating Mode Applicability: Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Example Emergency Action Level(s):

1. a. Loss of all offsite and all onsite AC power to Vital 4.16 KV busses.

   AND

   b. Either of the following:

   - Restoration of at least one Vital 4.16 KV bus in <4 hours is not likely.

     OR

   - Continuing degradation of core cooling based on Fission Product Barrier monitoring as indicated by CETs ≥700 °F.

Basis:

Loss of all AC power to Vital 4.16 KV busses 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 to Vital 4.16 KV busses will lead to loss of fuel clad, RCS, and containment, thus warranting declaration of a General Emergency.

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 Vital 4.16 KV 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.

In addition, under these conditions, fission product barrier monitoring capability may be degraded.

SYSTEM MALFUNCTION
SG1

Although it may be difficult to predict when power can be restored, it is necessary to give the SM / ED 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, 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 SM / ED judgment as it relates to IMMINENT loss or potential loss of fission product barriers and degraded ability to monitor fission product barriers.

Reference Documents:

1. Unit 1 Calculation 85-E-0072-02, “Time from Loss of All AC Power to Loss of Subcooling”

2. Unit 2 Calculation 85-E-0072-01, “Time from Loss of All AC Power to Loss of Subcooling”

SYSTEM MALFUNCTION
SG3

Initiating Condition - GENERAL EMERGENCY

Automatic trip and all manual actions fail to shutdown the reactor and indication of an extreme challenge to the ability to cool the core exists

Operating Mode Applicability: Power Operations (Mode 1)
Startup (Mode 2)

Example Emergency Action Level(s):

1. a. An automatic trip failed to shutdown the reactor

    AND

   b. All manual actions do not shutdown the reactor as indicated by reactor power ≥5%.

    AND

   c. Either of the following exist or have occurred due to continued power generation:

    - CET temperatures at or approaching 1200 °F

      OR

    - Feedwater flow rate less than:

      Unit 1: 430 gpm

      Unit 2: 485 gpm

Basis:

Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed and efforts to bring the reactor subcritical are unsuccessful.

In the event either of these challenges exists at a time that the reactor has not been brought below the power associated with the safety system design 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 table declaration to permit maximum off-site intervention time.