Arkansas Nuclear One

Name:	KEY
Login:	N/A
Date:	3/7/2018

ANO-1
Subject:
2018 ILO NRC

Total Points:
75/25/100
Percent Grade:

	T	F		
1	■	B	C	D
2	■	B	C	D
3	A	■	C	D
4	A	B	C	■
5	■	B	C	D
6	A	B	C	■
7	A	■	C	D
8	A	■	C	D
9	A	B	■	D
10	A	■	C	D
11	A	■	C	D
12	A	B	C	■
13	■	B	C	D
14	A	B	C	■
15	■	B	C	D
16	A	B	■	D
17	A	■	C	D
18	A	B	C	■
19	■	B	C	D
20	A	■	C	D
21	A	■	C	D
22	■	B	C	D
23	A	B	C	■
24	A	■	C	D
25	A	B	■	D
26	■	B	C	D
27	A	■	C	D
28	A	■	C	D
29	A	B	C	■
30	A	■	C	D
31	A	■	C	D
32	A	■	C	D
33	A	■	C	D
34	A	B	■	D
35	■	B	C	D
36	■	B	C	D
37	A	B	■	D
38	A	B	C	■
39	A	■	C	D
40	■	B	C	D
41	■	B	C	D
42	■	B	C	D
43	A	B	■	D
44	A	■	C	D
45	A	B	■	D
46	■	B	C	D
47	■	B	C	D
48	A	B	■	D
49	A	B	■	D
50	■	B	C	D
51	■	B	C	D
52	A	B	C	■
53	A	■	C	D
54	A	■	C	D
55	A	■	C	D
56	A	B	■	D
57	■	B	C	D
58	A	B	■	D
59	A	B	C	■
60	A	B	■	D
61	A	■	C	D
62	■	B	C	D
63	A	■	C	D
64	A	B	C	■
65	A	B	C	■
66	A	■	C	D
67	A	■	C	D
68	A	B	C	■
69	A	B	■	D
70	A	B	■	D
71	A	B	C	■
72	A	B	C	■
73	■	B	C	D
74	A	■	C	D
75	A	B	C	■
76	A	B	C	■
77	■	B	C	D
78	■	B	C	D
79	A	B	■	D
80	■	B	C	D
81	■	B	C	D
82	A	■	C	D
83	A	B	C	■
84	■	B	C	D
85	A	■	C	D
86	A	■	C	D
87	■	B	C	D
88	■	B	C	D
89	■	B	C	D
90	A	B	■	D
91	A	■	C	D
92	A	B	■	D
93	A	B	■	D
94	■	B	C	D
95	A	B	■	D
96	A	B	■	D
97	A	B	■	D
98	A	B	C	■
99	A	B	C	■
100	A	B	■	D

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

QID: 0604 **Rev:** 2 **Rev Date:** 1/9/18 **Source:** Bank **Originator:** S.Pullin

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

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

System Number: E02 **System Title:** Vital System Status Verification

Description: Ability to operate and / or monitor the following as they apply to the (Vital System Status Verification): Operating behavior characteristics of the facility.

K/A Number: EA1.2 **CFR Reference:** 41.7 / 45.5 / 45.6

Tier:	1	**RO Imp:**	3.2	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**		**SRO Select:**	No	**Taxonomy:**	H

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

Given:
* Reactor tripped from 100% power
* CRS has entered Reactor Trip (1202.001)

Assume all actions have been performed in sequence as required by system parameters.

FIVE (5) minutes later:
* ATC reports Pressurizer level dropped to 30" and is lowering
* Pressurizer Level Control (CV-1235) in AUTO and fully open

Which of the following is the proper procedure action for the current conditions?

A Initiate HPI per Repetitive Task (RT-2).

B. Reduce Letdown by closing Orifice Bypass (CV-1223).

C. Isolate Letdown by closing Letdown Cooler Outlet (CV-1221).

D. Operate CV-1235 in HAND to control PZR level 90 to 110".

Answer:

A Initiate HPI per Repetitive Task (RT-2).

Notes:

"A" is correct, this is done when level is < 30" per floating step under RCS Inventory/Press. This is also a step 27 contingency action of 1202.001, Reactor Trip.

"B" is incorrect, this is plausible but was done early in the procedure at step 5, shortly after immediate actions.

"C" is incorrect, isolating letdown is plausible but was done earlier when level dropped less than 55".

"D" is incorrect, taking CV-1235 to hand is plausible but was done earlier when level dropped less than 55".

This question matches the K/A since it is part of Vital System Status Verification which are steps 5 through 36 o 1202.001. These steps verify parameters are normal for post-trip conditions and if not, then actions are taken ir response to the off normal indications, low pressurizer level being one of these parameters.

References:

1202.001, Reactor Trip

History:

New for 2005 RO exam, modified as a replacement question.
Selected for 2010 RO/SRO exam.

Rev.1, editorial changes. Slightly revised stem.
Selected for 2018 exam
Rev. 2, revised notes for correct answer per NRC resolution.

QID: 1214 **Rev:** 1 **Rev Date:** 1/9/18 **Source:** New **Originator:** Cork

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

Section: 4.2 **Type:** Generic APEs

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

Description: Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: PORV isolation (block) valve switches and indicators.

K/A Number: AA2.05 **CFR Reference:** 41.10 / 43.5 / 45.13

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

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

Given:
* Unit 1 at 100% power
* CBOT reports Quench Tank level and temperature are rising
* SPDS point T1025 (ERV PSV 1000 OUTLET TEMP) reading 220 °F and rising
* PZR heater Groups 3 and 4 are ON
* STA states RCS leakage is 12.2 gpm and rising
* RCS pressure 2115 and slowly dropping
* CRS enters Pressurizer Systems Failure (1203.015)

Which of the following is required FIRST by Pressurizer Systems Failure (1203.015) for the above conditions?

A. Close ERV Isolation valve (CV-1000)

B. Perform Rapid Plant Shutdown (1203.045)

C. Trip reactor and perform Reactor Trip (1202.001)

D. Verify Pressurizer Level Control (CV-1235) opens in AUTO to maintain PZR level

Answer:

A. Close ERV Isolation valve (CV-1000)

Notes:

"A" is correct per 1203.015 Section 1 - ERV Failure or Leak. The ERV isolation valve should be closed to attempt to stop leak before taking more drastic actions in 1203.015.

"B" is incorrect but plausible since 1203.015 Section 1 directs this action if total RCS leakage exceeds Tech Specs and at 12.2 gpm, identified leakage has been exceeded. However, this action should only be taken if closing CV-1000 does not stop leak.

"C" is incorrect but plausible since additional heater groups are on (RCS pressure must be less than normal) and 1203.015 Section 1 directs this action if ERV leakage with CV-1000 closed exceeds capability to maintain RCS pressure but since that action has not been taken, it would not be prudent to trip the Reactor since no trip setpoints have been exceeded.

"D" is incorrect since this action is not included in 1203.015 Section 1 but plausible since taking CV-1235 to hand to maintain PZR level is an action in 1203.015 Section 7, and if applicant does not recognize the conditions as a steam space leak.

This question matches the K/A since indications are given for a leaking PORV (steam space accident) and use of the PORV isolation valve handswitch is required.

References:

1203.015, Pressurizer Systems Failure

History:

New for 2018 exam
Rev. 1, added "and rising" to T1025 and leakage parameters, re-ordered answer choices short to long, per NRC resolution.

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

QID: 0506 **Rev:** 3 **Rev Date:** 9/14/17 **Source:** Modified **Originator:** NRC

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

Section: 4.1 **Type:** Generic EPE

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

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

K/A Number: 2.4.21 **CFR Reference:** 41.7 / 43.5 / 45.12

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

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

Given:

* Small Break LOCA occurred
* Reactor manually tripped
* Both OTSG pressures 895 psig and stable
* SCM 25 °F
* All RCPs OFF

NOW
ATC states EFW is excessive

Per RT-5, SG levels will be filling to level band of ___(1)___
and control EFW in hand with flow rate ≥ ___(2)___ gpm until level band is achieved.

A. (1) 300 to 340"
(2) 340

B. (1) 370 to 410"
(2) 340

C. (1) 300 to 340"
(2) 570

D. (1) 370 to 410"
(2) 570

Answer:

B. (1) 370 to 410"
(2) 340

Notes:

"B" is correct, with SCM less than 30 °F and RCPs off, then Reflux Boiling setpoint is required which is 370 to 410".
Manual fill rate is determined from step 3.1 of RT-5 where it states to control EFW flow in hand if EFW is either inadequate or excessive and to keep flow rate ≥340 gpm until Reflux Boiling band is reached.

"A" is incorrect but plausible since RCPs are not running and 300 to 340" is the Natural Circulation setpoint but SCM is inadequate so the Reflux Boiling band should be used. The flow rate is correct.

"C" is incorrect but plausible since RCPs are not running and 300 to 340" is the Natural Circulation setpoint but SCM is inadequate so the Reflux Boiling band should be used. The flow rate is a valid flow rate but is for when only one SG is available.

"D" is plausible since this is the correct level band but incorrect since the flow rate is a valid flow rate but is for when only one SG is available.

This question matches the K/A since the conditions state a small break LOCA has occurred and the applicant must have knowledge of how to properly assess the status of the heat removal safety function.

References:

1202.012, Repetitive Tasks, RT5 - Verify Proper EFW Actuation and Control

History:

Developed by NRC.
Used on 2004 RO/SRO Exam
Used on the 2008 RO Exam
Selected for 2011 RO Exam
Rev. 3, revised conditions slightly, too much verbiage, deleted turbine tripped (unnecessary). Added condition that EFW is excessive so it would be controlled in hand.
Revised stem to ask for proper fill band and manual fill rate. Question was too simplistic and too easy to eliminate incorrect answers with system knowledge
Selected for 2018 exam

QID: 0506 **Rev:** 2 **Rev Date:** 12/16/2002 **Source:** Repeat **Originator:** NRC

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

Section: 4.1 **Type:** Generic EPE

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

Description: Knowledge of the interrelationships between the small break LOCA and the following: S/G's.

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

Tier:	1	**RO Imp:**	3.0	**RO Select:**	No	**Difficulty:**	4
Group:	1	**SRO Imp:**	3.3	**SRO Select:**	No	**Taxonomy:**	Ap

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

Given:

- A Small Break LOCA has occurred.
- ATC has tripped the reactor.
- CBOT has tripped the turbine.
- Both OTSG Pressures are at 895 psig and stable
- SCM is 25°F.
- All RCPs are OFF.

Q#3 PARENT

The desired final OTSG levels will be maintained with EFW using ___?___ Level instrument(s) and obtained with a fill rate of ___?___.

A. EFIC Low Range,
2" to 8"/min in Manual or 340gpm/SG in Auto

B. EFIC High Range
2" to 8"/min in Auto or 340gpm/SG in Manual

C. EFIC Low Range
2" to 8"/min in Auto or 340gpm/SG in Manual

D. EFIC High Range
2" to 8"/min in Manual or 340gpm/SG in Auto

Answer:

B. EFIC High Range
2" to 8"/min in Auto or 340gpm/SG in Manual

Notes:

"B" is correct, EFIC High Range, 2" to 8"/min in Auto or 340gpm/SG in manual until Steam Generator level is 370 to 410 inches.. EFIC Low Range does not cover the 370-410" level required for this EOP/RT. EOP/RT specifies the level and rate and the other three distracters are incorrect combinations of other instruments and manual versus auto modes of control.

References:

1202.012 "Repetitive Tasks", RT 5 Change 009

History:

Developed by NRC.
Used on 2004 RO/SRO Exam
Used on the 2008 RO Exam

Selected for 2011 RO Exam, repeat from last two exams

Q#3 PARENT

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

QID: 1215 **Rev:** 0 **Rev Date:** 9/14/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-ARCP **Objective:** 19 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 015 **System Title:** Reactor Coolant Pump Malfunctions

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

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

Tier: 1 **RO Imp:** 3.7 **RO Select:** Yes **Difficulty:** 3

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

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

Given:
* Unit 1 at 65% power
* P-32A, P-32C, and P-32D RCPs running
* Annunciator RCP BLEEDOFF TEMP HIGH (K08-C7) alarms
* P-32A Seal Bleedoff temperature 210 °F
* Seal injections flows are steady at 8 gpm each
* ICW flows are unchanged

(1) Which of the following operator actions are required per Reactor Coolant Pump and Motor Emergencies (1203.031) AND (2) what is the reason for the action?

A. (1) Reduce Reactor power to 50% and stop RCP P-32A
(2) Seal degraded but not failed

B. (1) Trip RCP P-32A and verify proper ICS response
(2) Seal failed, RCP must be tripped immediately

C. (1) Raise monitoring frequency of P-32A seals
(2) Seal off-normal but not degraded

D. (1) Trip reactor and then trip RCP P-32A
(2) RPS trip criteria is met

Answer:

D. (1) Trip reactor and then trip RCP P-32A
(2) RPS trip criteria is met

Notes:

"D" is correct, this meets criteria for seal failure (seal bleedoff temp >200 °F) and requires tripping pump, this leaves no pumps in the B loop (B pump is not running) which would cause a Rx trip, so the Rx must be tripped per 1203.031, section 2, before tripping RCP.

"A" is incorrect, this would be done only if the seal were degraded (>180 °F but <200 °F, not failed) per Section 1 and only if at least one pump was remaining in each loop. The reason given is incorrect, the seal has failed.

"B" is incorrect, this would be done if tripping the RCP would NOT cause an automatic Rx trip per 1203.031, Section 2. The reason is correct, the seal has failed and the RCP must be tripped, not stopped.

"C" is incorrect, this would be done per Section 1 when Seal bleedoff temperature had exceeded 155 °F but temperature is now 210 °F where tripping the RCP is required.

References:

1203.031, Reactor Coolant Pump and Motor Emergencies
1203.012G, Annunciator K08 Corrective Action
1202.001, Reactor Trip

History:

New question for 2018 exam

QID: 0549 **Rev:** 2 **Rev Date:** 1/9/18 **Source:** Bank **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: 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:** 3

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

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

Given:
* Unit 1 at 100% power
* HPI pump discharge pressure oscillating from 1500 to 2500 psig
* Makeup flow rate oscillating from 0 to 70 gpm
* Seal Injection total flow oscillating from 30 to 60 gpm
* Pressurizer level 215" and dropping
* Letdown flow 80 gpm and stable
* Makeup tank level 50" and dropping

Which of the following actions, and reasons for the actions, are procedurally required to be performed in response to these indications?

A. Trip HPI pump and isolate Letdown by closing Letdown Coolers Outlet, CV-1221, due to indications of degraded suction.

B. Take manual control of RC Pumps Total Injection Flow, CV-1207, and maintain 30-40 gpm due to loss of seal injection at power.

C. Take manual control of Pressurizer Level Control, CV-1235, and stabilize Pressurizer level due to automatic valve control malfunction.

D. Trip HPI pump, trip reactor, and go to EOP 1202.001, Reactor Trip, due to excessive RCS leakage.

Answer:

A. Trip HPI pump and isolate Letdown by closing Letdown Coolers Outlet, CV-1221, due to indications of degraded suction.

Notes:

"A" is the correct response due to indications of loss of suction (oscillating discharge pressure, oscillating flow) to HPI pump per section 2 of 1203.026.

"B" is incorrect but plausible since it is important to maintain seal injection. Seal injection flow is given as oscillating but it is more important to trip HPI pump to prevent damage, since RCP seals will still be cooled by ICW.

"C" is incorrect but plausible since makeup flow is oscillating but this is because the pump has lost suction, not a control valve malfunction.

"D" is incorrect but plausible since significant RCS leakage is indicated but mitigating actions from 1203.026 should be attempted first.

This question matches the K/A since it involves a loss of reactor coolant makeup and requires applicant to know action from the AOP and the reasons for those actions.

References:

1203.026, Loss of Reactor Coolant Makeup

History:

New for 2005 RO exam, but not used.
Selected for 2007 RO Exam.
Selected for 2011 RO Exam.
Rev. 1, editorial changes
Rev. 2, took reason given in D and used that in B, and inserted "excessive RCS leakage" as reason in D, per NRC resolution.
Selected for 2018 exam

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

QID: 0033 **Rev:** 2 **Rev Date:** 9/15/17 **Source:** Bank **Originator:** Cork

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

Section: 4.2 **Type:** Generic APEs

System Number: 025 **System Title:** Loss of Residual Heat Removal System (RHRS)

Description: Knowledge of the interrelations between the Loss of Residual Heat Removal System and the following: LPI or Decay Heat Removal/RHR pumps.

K/A Number: AK2.02 **CFR Reference:** 41.7 / 45.7

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

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

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

Given:
* RCS in lowered inventory
* RCS level being lowered for nozzle dam installation
* Annunciator DECAY HEAT VORTEX WARNING (K09-D8) alarms
* DH flow oscillating from 1000 gpm to 3000 gpm

Which of the following actions shall be performed FIRST per Loss of DH Removal (1203.028) for the above conditions?

A. Close at least one DH suction valve from the RCS.

B. Start the other DH pump to makeup to RCS from BWST.

C. Initiate containment closure per Att. G of 1203.028.

D. Stabilize flow by throttling one of the DH discharge flowpath valves.

Answer:

D. Stabilize flow by throttling one of the DH discharge flowpath valves.

Notes:

"D" is correct per 1203.028, section 4, step 2, attempt to stabilize flow by throtlling before taking more drastic measures.

"A" is one of the first four actions in sections 2 & 3, and is thus plausible.

"B" is a follow-up action if "D" does not work and is thus plausible.

"C" is a follow-up action if "D" does not work and is thus plausible.

References:

1203.028, Loss of Decay Heat Removal

History:

Developed for 1998 RO/SRO Exam.
Selected for 2005 RO exam, but not used.
Selected for 2007 RO Exam.
Rev. 2, editorial changes, also revised stem to remove "due to vortexing".
Selected for 2018 exam

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

QID: 1240 **Rev:** 1 **Rev Date:** 2/7/08 **Source:** New **Originator:** Cork

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

Section: 4.2 **Type:** Generic APEs

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

Description: Knowledge of the reasons for the following responses as they apply to the Loss of Component Cooling Water: The conditions that will initiate the automatic opening and closing of the SWS isolation valves to the CCWS coolers.

K/A Number: AK3..01 **CFR Reference:** 41.5, 41.10 / 45.6 / 45.13

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

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

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

(1) W hich ESAS channels close the SW isolations to the ICW coolers (CV-3820/3811) and (2) why is SW isolated to the ICW coolers during ESAS?

A. (1) 1 and 2;
(2) Prevent SW pump runout

B. (1) 1 and 2;
(2) Separate SW trains

C. (1) 5 and 6;
(2) Prevent SW pump runout

D. (1) 5 and 6;
(2) Separate SW trains

Answer:

B. (1) 1 and 2;
(2) Separate SW trains

Notes:

"B" is correct. ESAS Channels 1 and 2 close CV-3820 and CV3811, respectively, and these v alves are closed to separate the two SW trains since they are cross-connected at the ICW coolers via manual valv es SW -5 and SW-6 which provide cooling to the "swing" ICW cooler, E-28B.

"A" is incorrect but plausible since the ESAS channels are correct but SW is not isolated to SW pump runout. This is the correct reason that the ACW isolation is closed on ES signal (this works in conjunction with SW discharge cross-connects closing and B5/B6 aligned electrically with B SW pump) but SW to ICW coolers is isolated to separate the SW trains.

"C" is incorrect but plausible since ESAS channels 5 and 6 are the channels which cause the RB Coolers to swap from Chilled Water to Service Water, so they are associated with a SW system actuation. As stated in the explanation for "A" above, this is the incorrect reason for isolating SW to the ICW coolers.

"D" is incorrect but plausible since ESAS channels 5 and 6 are the channels which cause the RB Coolers to swap from Chilled Water to Service Water, so they are associated with a SW system actuation. This distractor does have the correct reason for isolating SW to the ICW coolers.

This question matches the K/A since it requires knowledge of the conditions that isolate SW to the ICW coolers (CCW S) and the reason for doing so.

References:

1104.029, Service Water & Auxiliary Cooling System

STM 1-65, Engineered Safeguards Actuation System

History:

New for 2018 exam
Rev.1, post-NRC week validation, high miss rate during validation (4/7 ROs missed), validators were of the opinion recalling exactly which channels cause isolation of SW to ICW coolers was difficult since the assignment of the RB isolation valves to specific channles does not follow a discrete methodology, revised first part of C and D distractors to "5 and 6".

QID: 1170 **Rev:** 0 **Rev Date:** 7/18/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-APZR **Objective:** E01 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

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

Description: Ability to determine and interpret the following as they apply to the Pressurizer Pressure Control Malfunctions: Normal values for RCS pressure.

K/A Number: AA2.02 **CFR Reference:** 41.7 / 43.5 / 45.13

Tier: 1 **RO Imp:** 3.8 **RO Select:** Yes **Difficulty:** 2

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

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

Given:
* Unit 1 at 70% power
* "A" MFP trips

In accordance with entry conditions for Section 6 of Pressurizer Systems Failure (1203.015), what is the RCS pressure setpoint when the operator should observe Pressurizer Spray Valve (CV-1008) go from OPEN to CLOSE position?

A. 2205 psig

B. 2155 psig

C. 2080 psig

D. 2030 psig

Answer:

B. 2155 psig

Notes:

"B" is correct, this is the normal operating pressure for the RCS and the pressure at which the PZR spray valve closes and all PZR heaters will be off. If reactor power is greater than 80% then the PZR spray valve will open on a Main Feedwater Pump (MFP) trip, and in this situation the Spray valve will open at 2080 psig and close at 2030 psig to limit the RCS pressure transient from the loss of feedwater at a high power level. The reactor power given is similar to but less than this, therefore, with the given conditions, the PZR Spray valve will open a its normal setpoint of 2205 psig and close at 2155 psig.

"A" is incorrect but plausible as this is the normal opening setpoint for the PZR Spray valve.

"C" is incorrect but plausible as this is the opening setpoint for the PZR Spray valve when power is >80% and a MFP trips.

"D" is incorrect but plausible as this is the closing setpoint for the PZR Spray valve when power is >80% and a MFP trips.

This question matches the K/A and requires the applicant to evaluate the given conditions to determine whether or not the RCS pressure condtion meets the entry condition for ANO-1's Pressurizer Pressure Control Malfunction procedure section for failure of the PZR spray valve.

References:

1203.015, Pressurizer Systems Failure, Section 6

History:

New question for 2018 exam

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

QID: 0509 **Rev:** 2 **Rev Date:** 1/9/18 **Source:** Mod **Originator:** NRC

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

Section: 4.1 **Type:** Generic EPEs

System Number: 029 **System Title:** Anticipated Transient Without Scram (ATWS)

Description: Knowledge of the interrelationships between the ATWS and the following: Breakers, relays, and disconnects.

K/A Number: EK2.06 **CFR Reference:** 41.7 / 45.7

Tier: 1 **RO Imp:** 2.9 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 3.1 **SRO Select:** No **Taxonomy:** F

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

Given:
* Unit 1 at 100% power
* Both MFW pumps trip
* RPS fails to actuate

Disregarding the Main Turbine, which one of the following describes (1) the operation of the AMSAC (ATWS Mitigation Safety Actuation Circuit) and (2) the DSS (Diverse Scram System) in response to the above conditions?

A. (1) AMSAC relays actuate EFW via EFIC B and C Channel Initiate modules;
(2) DSS opens contacts in power supply to gate drives for regulating rods

B. (1) AMSAC relays actuate EFW via EFIC B and C Channel Initiate modules;
(2) DSS energizes shunt trip coils to open CRD A & B breakers for all rods

C. (1) AMSAC relays actuate EFW via EFIC A and D Channel Initiate modules;
(2) DSS opens contacts in power supply to gate drives for regulating rods

D. (1) AMSAC relays actuate EFW via EFIC A and D Channel Initiate modules;
(2) DSS energizes shunt trip coils to open CRD A & B breakers for all rods

Answer:

C. (1) AMSAC relays actuate EFW via EFIC A and D Channel Initiate modules;
(2) DSS opens contacts in power supply to gate drives for regulating rods

Notes:

"C" is correct. AMSAC actuates EFW using relays to send signal to EFIC A and D channel initiate modules thus ensuring both trains of EFIC actuate. DSS relays open contacts in series with the E and F electronic trips to de-energize the gate drives for the regulating rods.

"A" is incorrect, although this would actuate both trains of EFIC, the correct channels are A and D. This is plausible since the answer for DSS is correct.

"B" is incorrect, although this would actuate both trains of EFIC, the correct channels are A and D. DSS trips the regulating rods via the gate drives not the A&B breakers which would drop all rods

"D" is incorrect but plausible since the AMSAC actuation is correct but DSS DSS trips the regulating rods via the gate drives not the A&B breakers which would drop all rods.

This question matches the K/A since it requires the applicant to have knowledge of how DSS and AMSAC function to mitigate an ATWS.

References:

STM 1-59, Diverse Reactor Overpressure Prevention System
STM 1-66, Emergency Feedwater Initiation and Control
STM 1-02, Control Rod Drive System

History:

Developed by NRC.
Used on 2004 RO/SRO Exam.
Rev. 1, corrected error in correct answer, both DSS and AMSAC trip the turbine. Revised all answers to provide a better link to the K/A. Added condtions to a loss of MFW without RPS trip since this is Tier 1. Revised stem to eliminate providing answer to another question. All of these changes cause this to be a modified question.
Rev. 2, changed second part of B and D to tripping all rods via the A&B breakers, per NRC resolution.

QID: 0509 **Rev:** 0 **Rev Date:** 12/8/2003 **Source:** Direct **Originator:** NRC

TUOI: **Objective:** **Point Value:** 1

Section: 4.1 **Type:** Generic EPE

System Number: 029 **System Title:** Anticipated Transient Without Scram (ATWS)

Description: Knowledge of the interrelationships between the ATWS and the following: Breakers, relays, and disconnects.

K/A Number: EK2.06 **CFR Reference:** 41.7/45.7

Tier: 1 **RO Imp:** 2.9 **RO Select:** No **Difficulty:** 2

Group: 2 **SRO Imp:** 3.1 **SRO Select:** No **Taxonomy:** K

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

Which one of the following describes the operation of the AMSAC (ATWS Mitigation Safety Actuation Circuit) and the DSS (Diverse Scram System) during an ATWS with a complete loss of Main Feedwater?

Q#9 PARENT

A. AMSAC trips the main turbine while DSS trips the regulating rods and starts the EFW pumps.

B. AMSAC trips the regulating rods while DSS trips the main turbine and starts EFW pumps.

C. AMSAC trips the main turbine and starts the EFW pumps while DSS trips the regulating rods.

D. AMSAC starts the EFW pumps and trips the regulating rods while DSS trips the main turbine.

Answer:

C. AMSAC trips the main turbine and starts the EFW pumps while DSS trips the regulating rods.

Notes:

"C", AMSAC trips the main turbine and starts the EFW pumps while DSS trips the regulating rods, is the correct answer. The other distracters are incorrect combinations of the 4 items that both sub-systems due as parts of DROPS.

References:

STM 1-59, Rev 1, page 9-10.

History:

Developed by NRC.
Used on 2004 RO/SRO Exam.

QID: 1092 **Rev:** 1 **Rev Date:** 1/9/18 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-EOP06 **Objective:** 10 **Point Value:** 1

Section: 4.1 **Type:** Generic EPE

System Number: 038 **System Title:** Steam Generator Tube Rupture (SGTR)

Description: Knowledge of the operational implications of the following concepts as they apply to the SGTR: Natural circulation.

K/A Number: EK1.03 **CFR Reference:** 41.8 / 41.10 / 45.3

Tier:	1	**RO Imp:**	3.9	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.2	**SRO Select:**	No	**Taxonomy:**	H

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

Given:
* Reactor tripped due to loss of both 6.9KV "H" busses
* Offsite power is available
* "B" OTSG has been isolated due to tube rupture

* Plant is cooling down on "A" OTSG at 70 °F/hr
* Tube to Shell delta T 80 °F tubes colder
* Subcooling Margin is adequate
* ICC display indicates reactor vessel head voiding

Which of the following is required per Tube Rupture (1202.006)?

A. Establish 40-60 °F Tube to Shell delta T.

B. Reduce cooldown rate to ≤ 50 °F/hr.

C. Open A and B Loop High Point Vents and leave open.

D. Maximize RB cooling (RT-9).

Answer:

B. Reduce cooldown rate to ≤ 50 °F/hr.

Notes:

"B" is correct. With natural circulation in progress and cooling down on the "good" SG, the operators would be at step 40 of 1202.006, Tube Rupture, which directs that if reactor vessel head voiding occurs and SCM is adequate, then cooldown rate must be reduced to less than 50°F/hr.

"A" is incorrect but plausible since 1202.006 does have actions for reducing tube to shell DT when it is greater than 60°F but that action is to stop the RCP in the loop. Another action in the Degraded Power EOP (1202.007) will have the operators establish 40-60 °F Primary-toSecondary DT, not tube to shell DT.

"C" is incorrect but plausible since 1202.006 does direct opening the high point vents but it states to open them only as necessary to eliminate voids, not to leave them open.

"D" is incorrect and plausible since a step 33 contingency will maximize RB cooling if primary to secondary heat transfer is NOT in progess but it obviously is in fact, excessive.

This question matches the K/A since it requires the candidate to have knowledge of operational implications (head voiding) during a natural circulation cooldown with a "bad" SG during a SGTR.

References:

1202.006, Tube Rupture

History:

New question for 2016 exam, NOT used due to overlap with SRO exam.
Rev. 1, added cooldown rate of 70 °F/hr, per NRC resolution.
Selected for 2018 exam

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

QID: 1171 **Rev:** 0 **Rev Date:** 7/19/17 **Source:** New **Originator:** Cork

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

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

System Number: E05 **System Title:** Excessive Heat Transfer

Description: Ability to operate and / or monitor the following as they applty to the (Excessive Heat Transfer): Desired operating results during abnormal and emergency situations.

K/A Number: EA 1.3 **CFR Reference:** 41.7 / 45.5 / 45.6

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

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

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

Given:
* Unit 1 tripped from 100% power
* Following immediate actions, CRS entered Overcooling (1202.003)
* ATC reports:
  "A" SG pressure 590 psig
  "B" SG pressure 470 psig

In accordance with Overcooling (1202.003) and RT-6, Verify Proper MSLI and EFW Actuation and Control, the following lights should be verified on the EFIC Remote Switch Matrix on C09:

A. SG-B MSL red lights ON Bus 1 and Bus 2 on both Train A and B Matrices;
RESET lights OFF

B. SG-A and SG-B MSL red lights ON Bus 1 and Bus 2 on both Train A and B Matrices;
RESET lights OFF

C. SG-B MSL red lights ON Bus 1 and Bus 2 on both Train A and B Matrices;
RESET lights ON and FLASHING

D. SG-A and SG-B MSL red lights ON Bus 1 and Bus 2 on both Train A and B Matrices;
RESET lights ON and FLASHING

Answer:

B. SG-A and SG-B MSL red lights ON Bus 1 and Bus 2 on both Train A and B Matrices;
RESET lights OFF

Notes:

"B" is correct, Per step 10 of Overcooling, since both SG pressures are less than 600 psig then an operator should verifiy proper Main Steam Line Isolation (MSLI) actuation and control per RT-6. The Reset lights on th EFIC matrices should be OFF unless there was a "half trip", but indications of a half trip are not given.

"A" is incorrect plausible since both SG pressures are less than 600 psig but B SG is 120 psig less than A and therfore the EFIC Vector signal will not allow EFW to feed B, but both SGs should have an MSL signal. The Reset light indication is correct.

"C" is incorrect, plausible since both SG pressures are less than 600 psig but B SG is 120 psig less than A and therefore the EFIC Vector signal will not allow EFW to feed B, but both SGs should have an MSL signal. Also, the Reset light indications flashing means that a component has failed to actuate, an undesirable condition.

"D" is incorrect, plausible since an MSL signal should be present on both SGs. The Reset light indications flashing means that a component has failed to actuate, an undesirable condition.

This question matches the K/A since it involves an excessive heat transfer condition (overcooling) and the applicant is required to know what the desired indications (operating results) should be for an MSLI.

References:

1202.003, Overcooling
1202.012, Repetitive Tasks, RT-6
STM 1-66, Emergency Feedwater Initiation and Control

History:

New question for 2018 exam

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

QID: 0334 **Rev:** 2 **Rev Date:** 10/23/17 **Source:** Mod **Originator:** Cork

TUOI: A1-LP-RO-AOP **Objective:** 1 **Point Value:** 1

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

System Number: 054 **System Title:** Loss of Main Feedwater

Description: Knowledge of the operational implications of the following concepts as they apply to Loss of Main Feedwater: MFW line break depressurizes the S/G (similar to a steam line break)

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

Tier:	1	**RO Imp:**	4.1	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.3	**SRO Select:**	No	**Taxonomy:**	H

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

PRIOR to ANY automatic or operator actions which set of parameters (other than SG pressure dropping) would indicate a Main Feedwater Line Break inside of the reactor building?

A. Indicated Feedwater flow dropping
RCS temperature rising

B. Indicated Feedwater flow rising
RCS temperature dropping

C. Indicated Feedwater flow dropping
RCS temperature dropping

D. Indicated Feedwater flow rising
RCS temperature rising

Answer:

D. Indicated Feedwater flow rising
RCS temperature rising

Notes:

"D" is correct, a MFW line break inside the RB would be downstream of the MFW flow transmitters, therefore MFW flow would rise. Unlike a steam line break RCS temperature would rise since less MFW flow is being delivered to the SGs for RCS heat removal.

"A" is incorrect but plausible since RCS temperature is moving in the proper direction for a MFW line break, but while MFW flow would be dropping if the break were upstream of the flow transmitters, in the case of a MFW line break in the building MFW flow would go up, not down.

"B" is incorrect but plausible since MFW flow is trending in the proper direction for a break in the RB, but a MFW line break would cause RCS temperature to go up, not down as in a steam line break.

"C" is incorrect since RCS temperature is dropping and plausible if the applicant thinks of a MFW line break causing a cooldown like a steam line break. MFW flow is incorrect also but plausible if the applicant can't recall that the MFW flow transmitters are not in the RB.

This question matches the K/A since the conditions given are a loss of main feedwater event caused by a main feedwater line break and requires the applicant to discern the effects of a break inside the Reactor Building.

References:

1203.027, Loss of Steam Generator Feed
STM 1-19, Feedwater System

History:

Developed for 1999 exam.
Modified for 2018 exam
Rev. 1, Modified by the following: added SG pressure trends to all answers to make a closer tie to K/A. Removed SG level trends. Removed RB pressure trends since a lower RB pressure is not plausible. Added RCS temperature trends.
Rev. 2, Validation resolution: removed SG pressure trends since the only plausible direction is dropping. Addec SG pressure to stem to preserve tie to K/A.
Editorial changes

QID: 0334 **Rev:** 0 **Rev Date:** 9-7-99 **Source:** Direct **Originator:** J. Cork

TUOI: ANO-1-LP-RO-AOP **Objective:** 1 **Point Value:** 1

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

System Number: 054 **System Title:** Loss of Main Feedwater

Description: Knowledge of the operational implications of the following concepts as they apply to Loss of Main Feedwater: MFW line break depressurizes the S/G (similar to a steam line break)

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

Tier: 1 **RO Imp:** 4.1 **RO Select:** No **Difficulty:** 3

Group: 2 **SRO Imp:** 4.3 **SRO Select:** No **Taxonomy:** An

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

Prior to any automatic or operator actions, which set of parameters would indicate a Main Feedwater Line Break inside of the reactor building?

Q#12 PARENT

a. OTSG level dropping
Feedwater flow dropping
RB pressure rising

b. OTSG level rising
Feedwater flow dropping
RB pressure rising

c. OTSG level rising
Feedwater flow rising
RB pressure dropping

d. OTSG level dropping
Feedwater flow rising
RB pressure rising

Answer:

d. OTSG level dropping
Feedwater flow rising
RB pressure rising

Notes:

"a" and "b" are incorrect, the OTSG level and RB conditions are correct but a break inside of the RB would be downstream of the FW flow transmitters and therefore FW flow would be rising.
"c" is incorrect, although FW flow is rising (which would be indicative of a break inside of the RB), a MFW line break inside the RB would cause RB pressure to increase, not decrease. OTSG level would also be dropping and not rising.

References:

1203.027 Rev 10

History:

Developed for 1999 exam.

QID: 1209 **Rev:** 1 **Rev Date:** 1/9/18 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-ELECD **Objective:** 14g **Point Value:** 1

Section: 4.1 **Type:** Generic EPEs

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

Description: Knowledge of the reasons for the following responses as they apply to the Station Blackout: Length of time for which battery capacity is designed

K/A Number: EK3.01 **CFR Reference:** 41.5 / 41.10 / 45.6 / 45.13

Tier: 1 **RO Imp:** 2.7 **RO Select:** Yes **Difficulty:** 2

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

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

Disregarding FLEX strategies, if Unit 1 experiences a Blackout, the full load discharge rating of the 125 VDC Vital Batteries (D06 and D07) is ____(1)____ hours and the reason for the sizing of the batteries to carry emergency DC and vital AC loads for a minimum of ____(2)____ hour(s).

A. (1) eight
(2) two

B. (1) four
(2) two

C. (1) eight
(2) one

D. (1) four
(2) one

Answer:

A. (1) eight
(2) two

Notes:

"A" is the correct answer per the SAR section on the 125 Volt DC System and the electrical system STM The batteries have 58 cells with a discharge rating of eight hours and will last a minimum of two hours powering emergency DC and vital AC loads, as well as supplying momentary loads during the two hour time..

"B" is incorrect, plausible since the reason is correct. The discharge rating given is half of the actual rating.

"C" is incorrect, yet plausible since it has the correct discharge rating but the reason is incorrect. The one hour time is plausible since that is the same as the FLEX implementation time.

"D" is incorrect. The discharge rating given is half of the actual rating. The one hour time is plausible since tha is the same as the FLEX implementation time.

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:

STM 1-32, Electrical Distribution

History:

New question for 2018 exam
Rev. 1, modified stem and answer choices to reduce length of answers, per NRC resolution.

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

QID: 0689 **Rev:** 3 **Rev Date:** 1/9/18 **Source:** Bank **Originator:** Steve Pullin

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

Section: 4.2 **Type:** Generic APEs

System Number: 056 **System Title:** Loss of Offsite Power

Description: Ability to operate and / or monitor the following as they apply to the Loss of Offsite Power: HPI system.

K/A Number: AA1.11 **CFR Reference:** 41.7 / 45.5 / 45.6

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

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

Given:
* Degraded Power event occurred
* Both EDGs supplying associated ES buses
* Pressurizer level 200 inches
* CET's indicate 600 degrees
* RCS pressure 1850 psig

MSLI and EFW have been actuated.

Which of the following actions is procedurally required for the above conditions?

A. Verify EFW level set point is 300 to 340 inches per RT-6, Verifiy Proper MSLI and EFW Actuation and Control.

B. Restore letdown for inventory control per RT-13, Restore Letdown.

C. Initiate HPI cooling per RT- 4, Initiate HPI Cooling.

D. Initiate full HPI per RT-3, Initiate Full HPI.

Answer:

D. Initiate full HPI per RT-3, Initiate Full HPI.

Notes:

"D" is correct. Applicant should recognize a loss of SCM has occurred after analyzing CETs and RCS pressure. Loss of SCM requires initiation of full HPI per RT-3 (step 24 of 1202.007, also a floating step).

"A" is incorrect but plausible since a Degraded Power condition means no RCPs will be running and this is the natural circulation control band. However, RCS temperature and pressure conditions mean Subcooling Margin (SCM) is inadequate and the Reflux Boiling band should be in use.

"B" is incorrect, plausible since letdown is restored in Degraded Power but not until step 77 and only when power has been restored to SU1 transformer.

"C" is incorrect, but plausible since the CET temperature is close to the overheating criteria and HPI cooling would be appropriate for that but CETs have not reached the overheating entry criteria of 610 °F.

Question matches K/A since condtions are given for LOSP with conditions which require full HPI initiation.

References:

1202.007, Degraded Power

History:

New question for the 2008 RO Exam.
Selected for 2018 exam
Rev. 2, editorial changes
Rev. 3, swapped B and C positions so answers are long to short, per NRC resolution.

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

QID: 0624 **Rev:** 1 **Rev Date:** 11/27/17 **Source:** Bank **Originator:** J.Cork

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

Section: 4.2 **Type:** Generic APEs

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

Description: Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: S/G pressure and level meters.

K/A Number: AA2.05 **CFR Reference:** 41.7 / 43.5 / 45.13

Tier: 1 **RO Imp:** 3.5 **RO Select:** Yes **Difficulty:** 3

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

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

If a loss of RS-1 panel occurred, (1) what would occur within the NNI system and (2) what would be the effect on SG pressure and level instruments on C03?

A. (1) Instrument power would automatically transfer to YO-2 by the ABT
(2) NNI-X SG pressure and level instruments would not be effected.

B. (1) NNI-X S1 and S2 switches would open and SASS would transfer to NNI-Y
(2) NNI-X SG pressure and level instruments would fail to mid-scale.

C. (1) NNI-X S1 and S2 switches would open and SASS would transfer to NNI-Y
(2) NNI-X SG pressure and level instruments would not be effected.

D. (1) Instrument power would automatically transfer to YO-2 by the ABT
(2) NNI-X SG pressure and level instruments would fail to mid-scale.

Answer:

A. (1) Instrument power would automatically transfer to YO-2 by the ABT
(2) NNI-X SG pressure and level instruments would not be effected.

Notes:

"A" is correct, a loss of RS-1 would cause NNI-X to be powered from YO-2, -24vDC logic power is auctioneered so there would be no loss of logic power, and instrument power would transfer by the ABT within 0.5 seconds so there would be no effect on NNI-X instruments.

"B" is incorrect, it would take a loss of both RS-1 and YO-2 to cause the S1 and S2 switches to open and thus wrong but if the S1 and S2 switches did open, then the NNI-X instruments would fail to mid-scale.

This question matches the K/A since it involves a loss of a Vital AC Instrument Panel and the applicant must determine how this affects the NNI system and the NNI-X SG pressure and level instrumentation.

"C" is incorrect, it would take a loss of both RS-1 and Y-02 for the S1 and S2 switches to open.and but SG pressure and level would not be effected so answer is thus plausible.

"D" is incorrect, the first part is correct and thus plausible, but the NNI-X instruments will not fail to mid-scale.

References:

STM 1-69, Non-Nuclear Instrument System

History:

New for 2005 RO re-exam.
Selected for the 2010 RO/SRO exam
Rev. 1, revised stem to more closely match K/A, revised "D" so question was a true 2x2, added "NI-X" to beginning of all second halves of answers, editorial changes.

Selected for 2018 exam.

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

QID: 1172 **Rev:** 2 **Rev Date:** 1/9/18 **Source:** New **Originator:** Cork

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 EOP entry conditions and immediate action steps.

K/A Number: 2.4.1 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier: 1 **RO Imp:** 4.6 **RO Select:** Yes **Difficulty:** 3

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

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

Given:
* Unit 1 at 100% power
* SW pumps P-4A and P-4C are in service
* SW Pump P-4B is aligned to Green train
* Annunciator SW BAY LEVEL LO (K10-A4) alarms
* Annunciator TRAV SCREEN SYSTEM TROUBLE (K05-F1) alarms

* SPDS display shows:
  * A SW Bay level 330 ft and dropping
  * B SW Bay level 337.5 ft and dropping
  * C SW Bay level 336.5 ft and dropping

What procedure action(s) are required to be taken FIRST for the above conditions?

A. Trip the reactor and go to 1202.001.

B. Backwash "A" Service Water Bay Strainers.

C. Swap MOD, start P-4B, and stop P-4A.

D. Disable the Electric Fire Pump P-6A.

Answer:

C. Swap MOD, start P-4B, and stop P-4A.

Notes:

"C" is correct since A SW bay level is 330 ft and the Traveling Screen System Trouble alarm is in, then this is a problem with the traveling screen leading to low bay level. Both 1203.030 and 1203.012I direct stopping the A SW pump but only 1203.030, Loss of Service Water, directs starting the standby pump. All SW discharge crosstie valves are normally open so it is simply a matter of swapping the MOD and starting the B SW pump to restore SW flow to normal.

"A" is incorrect but plausible since this is a possible action but this action is only taken in 1203.030 if only one SW pump is running and another pump cannot be started. The applicant might infer from the dropping C SW Bay level that a loss of the P-4C SW pump is imminent but 336.5 ft is only slightly lower than the normal level. This action might be required if conditions worsen but is not applicable for the given conditions.

"B" is incorrect but plausible since this action is specified in 1203.012I for low bay level in A or C bays but level is at 330 ft which requires immediate action to secure the pump to prevent damage.

"D" is incorrect but plausible since a fire pump should be disabled per 1203.012I, but P-6A is in the C SW Bay. The fire pump that should be disabled is Diesel Fire Pump P-6B which is in the A SW bay.

This question matches the KA since it requires knowledge of entry condtions for the Loss of Service Water AOP (low bay level and traveling screen trouble alarms). The Loss of Service Water AOP does not have any immediate actions (only the Reactor Trip EOP has immediate actions) therefore an important follow-up action was tested.

References:

1203.030, Loss of Service Water
1203.012I, Annunciator K10 Corrective Action

History:

New for 2018 exam
Rev. 1, changed B bay trend to dropping and B pump aligned to Green train based on validator comments,
Rev. 2, moved P-4B condition of being aligned to green train to 3rd bullet, corrected annunciator designator, revised stem so it is asking for FIRST action to be taken, per NRC resolution.

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

QID: 0691 **Rev:** 4 **Rev Date:** 1/9/18 **Source:** Bank **Originator:** Steve Pullin

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

Section: 4.2 **Type:** Generic APEs

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

Description: Ability to operate and/or monitor the following as they apply to the Loss of Instrument Air: Components served by instrument air to minimize drain on system.

K/A Number: AA1.02 **CFR Reference:** 41.7 / 45.5 /45.6

Tier: 1 **RO Imp:** 2.6 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 2.8 **SRO Select:** No **Taxonomy:** F

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

Given:
* Instrument Air leak is reported on Unit 2.
* Instrument Air header pressure low annunciator, K12-B3, alarmed.

Instrument Air header pressure drops to 58 psig and stabilizes.

Which of the following actions are required in accordance with 1203.024, Loss of Instrument Air?

A. Isolate Letdown by closing Letdown Coolers Outlet valve CV-1221.

B. Place RCP Seal Injection Block valve CV-1206 in override.

C. Trip the reactor and perform 1202.001 in conjunction with 1203.024.

D. Makeup to ICW Surge Tanks to maintain 1 to 2.7 psid.

Answer:

B. Place RCP Seal Injection Block valve CV-1206 in override.

Notes:

"B" is correct, this action is performed in 1203.024 when Inst Air pressure drops below 60 psig.

"A" is incorrect, this is plausible since this action is performed later in 1203.024, isolation of letdown is not done until IA pressure is less than 35 psig.

"C" is incorrect, this is plausible since this action is performed later in 1203.024, however this not done until IA pressure is less than 35 psig.

"D" is incorrect, this is plausible since this action is performed later in 1203.024, however this not done until IA pressure is less than 35 psig.

This question matches the K/A since the correct answer is to operate CV-1206 by placing it in override to prevent spurious operation and this also minimizes drain on system.

References:

1203.024, Loss of Instrument Air

History:

Modified from QID-0103
Selected for the 2008 RO Exam
Revised for 2016 exam but not used
Selected for 2018 exam

Rev. 3, revised question since it did not match K/A. No component is operated/monitored until IA pressure is less than 60 psig, revised pressure to 58 psig. Added low Inst Air press alarm. Replaced distracter B with placing CV-1206 in override, this is now the correct answer. Added trip the reactor as distracter C.
Rev. 4, deleted "suddenly" from IA pressure condition, and added "and stabilizes", revised D to makeup to Surge Tanks to maintain normal level band, per NRC resolution.

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

QID: 1173 **Rev:** 1 **Rev Date:** 1/9/18 **Source:** New **Originator:** Cork

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

Section: 4.2 **Type:** Generic APEs

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

Description: Knowledge of the interrelations between Generator Voltage and Electric Grid Disturbances and the following: Breakers, relays.

K/A Number: AK2.02 **CFR Reference:** 41.4, 41.5, 41.7, 41.10 / 45.8

Tier:	1	**RO Imp:**	3.1	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.3	**SRO Select:**	No	**Taxonomy:**	H

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

Given:
* Unit 1 at 360 MWE and +150 MVARS due to Dispatcher request
* Thunderstorms are causing grid disturbances

NOW:
* Annunciator GEN VOLTS/HZ RELAY FAILURE (K04-A6) alarms
* Shortly thereafter GENERATOR L.O. RELAY TRIP (K04-A8) alarms

Based on these conditions, the CRS should enter __________(1)__________ and
the CBOTshould verify Generator Output Breakers trip __________(2)___________ .

A. (1) Reactor Trip (1202.001)
(2) after a three second time delay

B. (1) Reactor Trip (1202.001)
(2) immediately

C. (1) Turbine Trip Below 43% Power (1203.018)
(2) after a three second time delay

D. (1) Turbine Trip Below 43% Power (1203.018)
(2) immediately

Answer:

D. (1) Turbine Trip Below 43% Power (1203.018)
(2) immediately

Notes:

"D" is correct, with a Gen Volts/Hz Relay alarm combined with a Generator LO Relay alarm, this means an electrical fault has occurred and the Generator Output Breakers open immediately on an electrical fault. With the Main Generator output at 360MW, the Reactor should be around 40% power and therefore no Reactor trip will occur since power is less than 43% and thus 1203.018 should be entered.

"A" is incorrect but plausible if the applicant does not recognize a Reactor trip will not occur. Also, during a normal trip (without an electrcial fault) there is approximately a 3 second time delay before the Generator Outpu Breakers open to allow for a "slow" transfer of the 6900v/4160v buses from the Unit Aux transformer to a Startup Transformer (normally SU #1).

"B" is incorrect but plausible as stated above for the use of 1202.001 but here the correct answer for the Generator Output Breakers is given.

"C" is incorrect but plausible since the correct procedure is stated but there is no time delay for the Generator Output Breakers to open on an electrical fault.

This question matches the K/A since it requires the applicant to assimiliate knowledge for Main Generator output and reactor power as well as how the Generator Output Breakers respond to electrcial fault relays.

References:

1203.018, Turbine Trip Below 43% Power
1203.012C, Annunciator K04 Corrective Action
STM 1-30, Main Generator and Controls

History:

New for 2018 exam
Rev. 1, changed initial power level to 360 Mwe, per NRC resolution.

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

QID: 0422 **Rev:** 2 **Rev Date:** 1/9/18 **Source:** Bank **Originator:** Cork

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

Section: 4.2 **Type:** Generic APEs

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

Description: Ability to operate and/or monitor the following as they apply to the Continuous Rod Withdrawal: Rod in-out-hold switch.

K/A Number: AA1.02 **CFR Reference:** 41.7 / 45.5 / 45.6

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

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

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

Given:
* Power escalation to 100% power in progress
* Group 7 control rods begin to continuously withdraw at 30 inches per minute without a command signal present

Assuming the Diamond is taken to Manual, which of the following are procedure steps specified by Control Rod Drive Malfunction Action (1203.003) to stop the rod withdrawal?

1. Group/Aux to Aux
2. Seq/Seq Or to Seq.
3. Group Sel to All
4. Run/Jog to Jog
5. Fault Reset to Reset
6. Insert/Withdrawal to Insert

A. 1, 4, and 6

B. 2, 3, and 5

C. 2, 4, and 6

D. 1, 5, and 6

Answer:

A. 1, 4, and 6

Notes:

"A" has the MAJI steps (Manual, Aux, Jog, Insert) to give the CRD system a conflicting signal. The conflicting signal should stop rod motion. These steps are found in 1203.003, Section 9, step 4.

"B", "C", and "D" are incorrect but plausible since these actions are all switch manipulations which exist on the "Diamond" rod control panel however, they are incorrect combinations of steps which will not stop continuous rod motion.

References:

1203.003, Control Rod Drive Malfunction Action

History:

Used QID 5341 from regular exambank.
Modified for use in 2002 RO/SRO exam.
Selected for 2018 exam
Rev. 1, editorial changes, added procedure to stem, deleted Clamp/Clamp Rel since it wasn't used in any answer choices. Changed "C" to use Run/Jog so that more than the correct answer contains it. Moved

"Diamond to Man" to stem since all answer choices contained it.
Rev. 2, deleted #4 (single sel to all) since it was not used in any answer choice which required a re-numbered list and revised answer choices accordingly, per NRC resolution.

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

QID: 0001 **Rev:** 5 **Rev Date:** 1/9/18 **Source:** Bank **Originator:** Giles

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

Section: 4.2 **Type:** Generic APEs

System Number: 005 **System Title:** Inoperable/Stuck Control Rod

Description: Knowledge of the interrelation between the inoperable/stuck control rod and the following: controllers and positioners.

K/A Number: AK2.12 **CFR Reference:** 41.7/45.7

Tier:	1	**RO Imp:**	2.5	**RO Select:**	Yes	**Difficulty:**	4
Group:	2	**SRO Imp:**	2.5	**SRO Select:**	No	**Taxonomy:**	H

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

Given:
* Plant power reduced to 350 MWe
* Group 7 Rod 3 API indication not moving
* Group 7, Rod 3, stator temperature 193 °F and rising
* Stator temperature HI alarm in
* All other control rod stator temperatures are normal

Which action must be performed FIRST per Control Rod Drive Malfunction (1203.003) due to Group 7, Rod 3?

A. Manually trip reactor due to Group 7, Rod 3 stator temperature exceeding 190 °F

B. Transfer Group 7, Rod 3 to the Aux Bus and pull programmer control fuses for the Aux Power Supply.

C. Drop Group 7, Rod 3 by removing the six stator fuses for the rod in the CRD transfer cabinet.

D. Adjust Group 7, Rod 3 RPI to agree with API to prevent sequence inhibit alarm.

Answer:

B. Transfer Group 7, Rod 3 to the Aux Bus and pull programmer control fuses for the Aux Power Supply.

Notes:

"B" is correct. Per 1203.003, Section 7, step 4, if only one CRD stator temperature is high, then the rod is transferred to the aux bus (after ensuring power is reduced to less than 360 Mwe - 40% , so that an ICS runback on a dropped rod will not be initiated) when the programmer fuses are pulled in the Aux Power Supply cabinet to drop the rod.

"A" is incorrect but plausible since a manual reactor trip would be required if more than one CRD stator temperature exceeded180°F.

"C" is incorrect but plausible based on guidance in 1203.003, Section 7, CRD Stator Temperature High. This step is performed after verification of the rod dropping into the core after transfer to the Aux bus and pulling programmer fuses.

"D" is incorrect but plausible since the conditions note that API has stopped moving so there would be a disagreement between API and RPI but it is more important to de-energize the rod to prevent damage to its stator.

This question matches the K/A since it involves a stuck control rod and the equivalent in ANO-1 control rod drive system to a controller, i.e., programmer.

References:

1203.003, Control Rod Drive Malfunction

History:

Developed for 1998 RO/SRO Exam.
Selected for the 2008 RO Exam
Rev. 4, editorial changes, revised distractor D since it could not be physically accomplished.
Selected for 2018 exam
Rev. 5, changed 1st bullet to 350 Mwe, per NRC resolution

QID: 1217 **Rev:** 1 **Rev Date:** 1/9/18 **Source:** Modified **Originator:** Cork

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

Section: 4.2 **Type:** Generic APEs

System Number: 037 **System Title:** Steam Generator Tube Leak

Description: Ability to determine and interpret the following as they apply to the Steam Generator Tube Leak: Comparison of RCS fluid inputs and outputs, to detect leaks.

K/A Number: AA2.04 **CFR Reference:** 41.3 / 43.5 / 45.13

Tier: 1 **RO Imp:** 3.4 **RO Select:** Yes **Difficulty:** 4

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

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

Following a reactor trip, MSLI on ‘A’ SG had to be actuated to correct an Overcooling event.

The following are observed 20 minutes later:
* TAVE 532 °F and stable
* Pressurizer Level 90 inches and lowering at 1 inch/min
* ‘B’ TBVs are being controlled in Manual
* M/U Flow 82 gpm
* L/D Flow 43 gpm
* Seal Inj Flow 40 gpm
* Seal Bleedoff Flow 6 gpm total
* RB Leak Detector RE-7461 80 cpm
* (T-37A) ICW Surge Tank - 0.8 psid and stable
* (T-37B) ICW Surge Tank - 0.8 psid and stable
* Letdown temperature 95 °F

(1) W here is the RCS Leak and (2) what is the rate?

A. (1) L/D Cooler Leak
(2) 85 gpm

B. (1) Steam Generator Tube Leak
(2) 85 gpm

C. (1) L/D Cooler Leak
(2) 61 gpm

D. (1) Steam Generator Tube Leak
(2) 61 gpm

Answer:

B. (1) Steam Generator Tube Leak
(2) 85 gpm

Notes:

"B" is the correct location and leak rate. An overcooling has occurred which puts stresses on the tubes and can cause a tube leak. Calculation is from 1203.039 Exhibit 1 (Makeup flow + seal injection flow) - (Letdown flow + seal bleedoff flow + PZR level change), so (82 + 40 + 12.4) - (43 + 6) = 85 gpm Pressurizer lev el change must be subtracted if rising and added if lowering.

"A" is incorrect but plausible as this is the correct leak rate. One source of leakage is a letdown cooler tube leak but ICW Surge Tank levels are steady so this is not the source of leakage. The leak rate of 85 gpm is correct per the above explanation..

"D" is incorrect but plausible as this is the correct leakage source.

"C" and "D" are incorrect based on leak rate, 61 gpm would be the calculated leak rate if the applicant erroneously subtracted the change in pressurizer level instead of adding.

This question matches the K/A since conditions are given for a tube leak and the applicant must compare RCS inputs and outputs to determine the correct leak rate.

References:

1203.039, Excess RCS Leakage

History:

Modified QID 894 for 2018 exam, changed Makeup flow from 58 gpm to 82 gpm, this made "D" correct (vs.B).
Changed A and B to 85 gpm to make them plausible with change in makeup flow.
Rev. 1, changed PZR level from rising to lowering which makes B the correct answer, per NRC resolution.

QID: 0894 **Rev:** 1 **Rev Date:** 9/5/14 **Source:** Repeat **Originator:** Possage

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

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

System Number: 037 **System Title:** Steam Generator Tube Leak

Description: Ability to determine and interpret the following as they apply to the Steam Generator Tube Leak: S/G tube failure

K/A Number: AA2.06 **CFR Reference:** 43.5 / 45.13

Tier: 1 **RO Imp:** 4.3 **RO Select:** No **Difficulty:** 4

Group: 2 **SRO Imp:** 4.5 **SRO Select:** No **Taxonomy:** Ap

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

Following a reactor trip, MSLI on 'A' SG had to be actuated to correct an Overcooling event.

Q#21 PARENT

The following are observed 20 minutes later:
- TAVE 532 °F and stable
- Pressurizer Level 90 inches and rising at 1 inch/min
- 'B' TBVs are being controlled in Manual
- M/U Flow 58 gpm
- L/D Flow 43 gpm
- Seal Inj Flow 40 gpm
- Seal Bleedoff Flow 6 gpm total
- RB Leak Detector RE-7461 80 cpm
- (T-37A) ICW Surge Tank - 0.8 psid and stable
- (T-37B) ICW Surge Tank - 0.8 psid and stable
- L/D Temperature 95 °F

Where is the RCS Leak and what is the rate?

A. L/D Cooler Leak; 37 gpm

B. Steam Generator Tube Leak; 37 gpm

C. L/D Cooler Leak; 61 gpm

D. Steam Generator Tube Leak; 61 gpm

Answer:

B. Steam Generator Tube Leak ; 37 gpm

Notes:

B is the correct location and leak rate. (See filled out Exhibit 1 from 1203.039)
A is incorrect based on steady ICW Surge Tank levels.
C and D are incorrect based on leak rate, 61 gpm would be the calculated leak rate if the applicant erroneously applied the change in pressurizer level.

References:

1203.039, Excess RCS Leakage

History:

New for 2014 Exam

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

QID: 0951 **Rev:** 1 **Rev Date:** 9/18/17 **Source:** Bank **Originator:** NRC

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

Section: 4.2 **Type:** Generic APEs

System Number: 059 **System Title:** Accidental Liquid Radwaste Release

Description: Knowledge of the interrelations between the Accidental Liquid Radwaste Release and the following: Radioactive-liquid monitors.

K/A Number: AK2.01 **CFR Reference:** 41.7 / 45.7

Tier: 1 **RO Imp:** 2.7 **RO Select:** Yes **Difficulty:** 2

Group: 2 **SRO Imp:** **SRO Select:** No **Taxonomy:** F

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

Given:
* Unit 100% power
* No planned releases are in progress

NOW the following alarms occur:
* PROC MONITOR RADIATION HI (K10-B2)
* Liquid Radwaste Process Monitor (RI-4642), (C25, Bay 2)

Per Liquid Waste Discharge Line High Radiation Alarm (1203.007) which of the following actions is required to be taken FIRST?

A. Verify that no release in progress by monitoring discharge flow to flume (FI-4642) on C19

B. Verify no fault with rad monitor by verifying RADIATION MONITOR TROUBLE (K10-C1) NOT in alarm

C. Ensure Treated Waste Discharge to Circulating Water (CW) Flume (CZ-58) manual valve closed

D. Ensure Filtered Waste Monitoring Tank Discharge to CW Flume (DZ-25) manual valve closed

Answer:

A. Verify that no release is in progress by monitoring discharge flow to flume (FI-4642) on C19

Notes:

"A" is correct, this is the first step in 1203.007. The Liquid Radwaste Process Monitor (RI-4642) will close CV-4642 so this action is to verify that an automatic action occurred.

"B" is incorrect but credible as this could be a normal reaction of an RO. A loss of power to the rad monitor will cause it to close CV-4642 just like a high rad signal.

"C" incorrect but plausible since this is a contingency action in 1203.007.

"D" incorrect but plausible since this is a contingency action in 1203.007.

This question matches the K/A since it involves an accidental liquid radwaste release and the interrelation with the liquid radwaste process monitor.

References:

1203.007, Liquid Waste Discharge Line High Radiation Alarm

History:

New for 2013 Exam
Selected for 2018 exam
Rev. 1, revised B distractor due to lack of credibility, editorial changes

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

QID: 1175 **Rev:** 0 **Rev Date:** 7/24/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-RMS **Objective:** EO7 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 061 **System Title:** Area Radiation Monitoring System Alarms

Description: Knowledge of the interrelations between the Area Radiation Monitoring (ARM) System Alarms and the following: Detectors at each ARM system location.

K/A Number: AK2.01 **CFR Reference:** 41.7 / 45.7

Tier:	1	**RO Imp:**	2.5	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	2.6	**SRO Select:**	No	**Taxonomy:**	F

RO: 23 **SRO:**

Question:

When investigating an Area Radiation Monitor alarm, which of the following area monitors will have a readout in R/hr (vs. mR/hr)?

A. Spent Fuel Filter (RE-8016)

B. Makeup Pump Room (RE-8011)

C. Radio Chem Lab (RE-8006)

D. Fuel Hand Area (RE-8017)

Answer:

D. Fuel Hand Area (RE-8017)

Notes:

"D" is correct, the Fuel Handling Area monitor (RE-2017) is one of only 6 area rad monitors which read in R/hr vs. mR/hr for the other 16 area monitors.

"A" is incorrect but plausible since this is an area of possible very high radiation.

"B" is incorrect but plausible since this is an area of possible very high radiation.

"C" is incorrect but plausible since this is an area of possible very high radiation.

This question matches the K/A since it requires knowledge of a Area Rad Monitor interrelationship with it's location: due to being in an area where a fuel handling accident could occur it has a higher readout than most of the other radiation monitors.

References:

STM 1-62, Radiation Monitoring

History:

New for 2018 exam

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

QID: 1212 **Rev:** 0 **Rev Date:** 9/13/17 **Source:** New **Originator:** Cork

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

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

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

Description: Knowledge of the operational implications of the following concepts as they apply to the (Loss of NNI-X): Normal, abnormal and emergency operating procedures associated with (Loss of NNI-X).

K/A Number: AK1.2 **CFR Reference:** 41.8 / 41.10 / 45.3

Tier: 1 **RO Imp:** 3.4 **RO Select:** Yes **Difficulty:** 3

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

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

Given:
* Unit 1 at 40% power
* ATC reports a loss of NNI X AC power is indicated on C13

Which of the following actions is procedurally required to mitigate this event?

A. Close RCS Makeup Block (CV-1233).

B. Trip the reactor and perform Reactor Trip (1202.001).

C. Operate MFW pumps in HAND.

D. Place RCP Seal Injection Block (CV-1206) in OVRD.

Answer:

B. Trip the reactor and perform Reactor Trip (1202.001).

Notes:

"B" is correct in accordance with 1203.047, step 6 directs tripping of the reactor and performing 1202.001 if NN X AC power is lost.

"A" is incorrect but plausible since this action is also performed on a loss of NNI but this action is for a loss of NNI X DC power only.

"C" is incorrect but plausible since this action is also performed on a loss of NNI but this action is for a loss of NNI Y power only when MFW pumps are on DP control (up to 50%) and power is given as 40% so this is plausible.

"D" is incorrect but plausible since this action is also performed on a loss of NNI but this action is for a loss of NNI X DC power only.

References:

1203.047, Loss of NNI Power

History:

New for 2018 exam

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

QID: 1176 **Rev:** 1 **Rev Date:** 1/9/18 **Source:** New **Originator:** Cork

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

Section: 4.2 **Type:** Generic APEs

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

Description: Knowledge of how abnormal operating procedures are used in conjunction with EOPs.

K/A Number: 2.4.8 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier: 1 **RO Imp:** 3.8 **RO Select:** Yes **Difficulty:** 2

Group: 2 **SRO Imp:** 4.5 **SRO Select:** No **Taxonomy:** F

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

Which of the following is used in conjunction with, and specifically referred to in, Turbine Trip Below 43% Power (1203.018)?

A. ICS Abnormal Operations (1203.001)

B. RT-14, Control RCS Pressure

C. RT-19, Check Proper Electrical Response

D. Load Rejection (1203.020)

Answer:

C. RT-19, Check Proper Electrical Response

Notes:

"C" is correct, step 8 of 1203.018 directs use of RT-19 and possibly RT-21 if EDGs are running but none of the others.

"A" is incorrect but plausible since the applicant could infer that ICS Abnormal Ops would be used since the Main Turbine receives ICS inputs, but 1203.001 is not mentioned in 1203.018.

"B" is incorrect since it is not specified in 1203.018 but plausible since RCS pressure would be changing due to the loss of load.

"D" is incorrect since it is not specified in 1203.018 but plausible since a load rejection will occure with a turbine trip below 43% poower.

This question matches the K/A since it requires the applicant to know which EOP repetitive task from 1202.012 is used in conjunction with the AOP 1203.018.

References:

1203.018, Turbine Trip Below 43% Power

History:

New question for 2018 exam
Rev. 1, replaced RTs in A and D with more plausible AOPS, revised stem for clarity due to this change, per NRC resolution.

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

QID: 1211 **Rev:** 1 **Rev Date:** 10/9/17 **Source:** Modified **Originator:** Cork

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

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

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

Description: Knowledge of the reasons for the following responses as they apply to the (Flooding): Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations.

K/A Number: AK3.3 **CFR Reference:** 41.5 / 41.10, 45.6, 45.13

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

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

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

Given:

* 100% power
* Lake Dardanelle level 346 feet and rising due to heavy rains
* Corps of Engineers predicts peak flood levels will reach 355 feet

(1) What action is required per Natural Emergencies (1203.025) Section 6, Flood, and (2) why?

A. (1) Perform Rapid Plant Shutdown (1203.045) and align one Decay Heat pump for Decay Heat removal;
(2) will be cooling down to lowest possible RCS temp

B. (1) Perform Rapid Plant Shutdown (1203.045) and make preparations to transfer plant auxiliaries to SU1 transformer;
(2) SU1 is designed for site flooding

C. (1) Perform Power Reduction and Plant Shutdown (1102.016) and align one Decay Heat pump for Decay Heat removal;
(2) will be cooling down to lowest possible RCS temp

D. (1) Perform Power Reduction and Plant Shutdown (1102.016) and make preparations to transfer plant auxiliaries to SU1 transformer;
(2) SU1 is designed for site flooding

Answer:

A. (1) Perform Rapid Plant Shutdown (1203.045) and align one Decay Heat pump for Decay Heat removal;
(2) will be cooling down to lowest possible RCS temp

Notes:

"A" is correct since 1203.025 directs one to perform a shutdown per 1203.045, and 1203.025 states to align one DH loop for DH removal and the other loop is ensured to be aligned for ES standby (LPI), which it normally is. The goal is to cool down to the lowest possible RCS temperature in case the flooding is long term.

"B" is incorrect but plausible since 1203.045 is the correct shutdown procedure when lake level is greater than 345 feet, but SU2 transformer is designed for flooding and auxiliaries, not SU1. This is plausible since SU1 transformer is the normal power supply for transfers for non-flooding shutdown.

"C" is incorrect since the procedure does not state to use the normal plant shutdown procedure but is plausible since 1203.025 does direct aligning one DH pump for DH removal. The reason given is correct.

"D" is incorrect since the procedure does not state to use the normal plant shutdown procedure and SU2 transformer is designed for flooding and auxiliaries, not SU1. This is plausible since SU1 transformer is the normal power supply for transfers for non-flooding shutdown.

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:

Modified QID 780 for 2018 exam
Modified question by deleting P-34A DH pump OOS and changing B and D answers to SU1 transformer instead of SU2, this makes "A" the correct answer (vs. B). Also changed B and D answers to say "one" DH pump vs. P-34B since both loops will be available.
Rev. 1, added "reasons" to stem and all answers to match K/A.

QID: 0780 **Rev:** 1 **Rev Date:** 5/19/17 **Source:** Repeat **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:**	No	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.7	**SRO Select:**	No	**Taxonomy:**	H

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

Given:

Q#26 PARENT

* 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.

Q#26 PARENT

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

QID: 1208 **Rev:** 2 **Rev Date:** 2/7/18 **Source:** New **Originator:** Cork

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

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

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:** 3

Group: 2 **SRO Imp:** **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
* All valves are in ESAS actuated positions
* Annunciator A HPI FLOW HI/LO (K11-A4) in alarm
* Annunciator A4 L.O. RELAY TRIP (K02-B7) in alarm

P-36A HPI pump flows:

125 gpm to "A" HPI line
135 gpm to "B" HPI line
125 gpm to "C" HPI line
165 gpm to "D" HPI line

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

(Assume throttling to the values below do not change values on other lines.)

A. Throttling is not allowed with the above conditions.

B. Throttle "D" HPI valve until "D" line flow is 135 gpm.

C. Throttle "D" HPI valve until "D line flow is 150 gpm.

D. Throttle all HPI valves until each line reads 110 gpm.

Answer:

B. Throttle "D" HPI valve until "D" line flow is 135 gpm.

Notes:

"B" is correct. Conditions given are that ESAS has actuated, only HPI pump is running, so in accordance with RT-10 if an alarm is present then HPI flow should be throttled per the ACA. "B" has the throttling criteria for when only one HPI pump is runnning (A4 has a lockout so only one pump will be running) and RCS pressure is >600 psig. This critieria is to throttle the highest flow to within 20 gpm of the next highest flow. Throttling "D" flow to 135 gpm will place it within 20 gpm of "B" so total flow will be 520 gpm which is slightly less than the maximum flow (525 gpm) allowed.

"A" is incorrect but plausible if applicant believes that throttlling of any kind is not allowed with a loss of SCM.

"C" is incorrect but plausible if applicant can recall HPI throttling critieria when only one HPI pump is runnning and RCS pressure is >600 psig. This criteria has one throttling the highest flow (D - 165 gpm) to within 20 gpm of the next highest flow (B - 135 gpm). However, this is not applicable when two pumps are running and throttling D to 150 gpm will still have total flow for this pump at 535 gpm which is in excess of 500 gpm and possibly operating P-36A in runout conditions.

"D" is incorrect. If all are throttled to 110 gpm each, then this will add to a total of 440 gpm which will clear the annunciator and be less than the maximum flow but this is too much throttling for a loss of SCM.

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

References:

1202.012, Repetitive Tasks, RT-10, Verify Proper ESAS Actuation
1203.012J, Annunciator K11 Corrective Action

History:

New question for 2018 exam
Rev. 1, 11/1/17, added A4 lockout so that B would be correct (validation was successful), adjusted B to 135 so total would be slightly less than maximum. Adjusted "D" to 110 gpm (and to throttle all) which adds to 440 which is less than maximum but is too much throttling considering the loss of SCM, this was done to ensure "D" was not a correct answer.
Rev. 2, 2/7/18, discovered during NRC validation week that this question was similar to a simulator JPM. After discussion it was determined this question is indeed testing a different concept than the JPM. Made "D" HPI line flow 165 gpm and "A" HPI line flow 125 gpm so that the line with the highest flow wasn't the same as the JPM.

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

QID: 0053 **Rev:** 1 **Rev Date:** 9/11/17 **Source:** Bank **Originator:** Cork

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

Section: 3.4 **Type:** Heat Removal From Reactor Core

System Number: 003 **System Title:** Reactor Coolant Pump System (RCPS)

Description: Knowledge of the effect of a loss or malfunction on the following will have on the RCPS: Containment isolation valves affecting RCP operation.

K/A Number: K6.04 **CFR Reference:** 41.7 / 45.5

Tier:	2	**RO Imp:**	2.8	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**		**SRO Select:**	No	**Taxonomy:**	F

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

Without operator action, which of the following incidents would have the most detrimental effect on RCP operation?

A. Loss of nuclear ICW to RCP

B. Main steam line rupture inside RB

C. Loss of RCP seal injection

D. RCP bleedoff normal return valve fails closed

Answer:

B. Main steam line rupture inside RB

Notes:

"B" is correct due to isolation of non-nuclear ICW to all RCP motors and nuclear ICW to RCP seal coolers from ESAS. This will require securing RCPs per RT-10 since both motor cooling and seal cooling will be isolated.

"A" is incorrect since a loss of nuclear ICW will not harm RCPs unless seal injection is also lost.

"C" is incorrect since a loss of seal injection will not harm RCPs unless nuclear ICW is also lost.

"D" is incorrect,closing of seal return isolation will only cause seal return to be diverted to Quench Tank.

References:

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

History:

Used in 1998 initial RO exam
Selected for 2005 RO re-exam.
Selected for 2011 RO Exam.
Selected for 2018 exam

QID: 0699 **Rev:** 2 **Rev Date:** 9/19/17 **Source:** Bank **Originator:** Cork

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

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

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

Description: Ability to manually operate and/or monitor in the control room: Letdown pressure and temperature control valves.

K/A Number: A4.05 **CFR Reference:** 41.7 / 45.5 to 45.8

Tier: 2 **RO Imp:** 3.6 **RO Select:** Yes **Difficulty:** 3

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

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

During power operation the Letdown 3-way Valve (CV-1248) is positioned to BLEED with both the degassifier inlet (CZ-8) and bypass (CZ-9) valves closed.

(1) What would you expect letdown pressure to be and (2) where would flow be going?

A. (1) 100 psig
(2) Vacuum Degasifier Moisture Separator Tank (T-76)

B. (1) 150 psig
(2) Vacuum Degasifier Moisture Separator Tank (T-76)

C. (1) 100 psig
(2) Aux Bldg Equip Drain Tank (T-11)

D. (1) 150 psig
(2) Aux Bldg Equip Drain Tank (T-11)

Answer:

D. (1) 150 psig
(2) Aux Bldg Equip Drain Tank (T-11)

Notes:

"D" is correct, this would cause letdown to have no flow path, pressure would excalate until the Letdown Relief (PSV-1236) lifted at 150 psig. The Letdown Relief valve's flow goes to the Aux Bldg Equip Drain Tank (T-11).

A is incorrect, but plausible since the setpoint of 100 psig is the setpoint of the Makeup Tank Relief valve. The Vacuum Degasifier Moisture Separator Tank (T-76) is also plausible since the Makeup Tank Relief Valve relieves to this location.

B is incorrect, but plausible since the setpoint is correct but the relief destination is not. The Vacuum Degasifier Moisture Separator Tank (T-76) is plausible since the Makeup Tank Relief Valve relieves to this location.

C is incorrect but plausible since T-11 is the correct destination, but the pressure is below the Letdown relief valve setpoint.

This question matches the K/A since it concerns CVCS (makeup and purification) and the applicant is theoretically operating the Letdown 3 way valve (ANO-1 does not have a temperature control valve, this is the closest thing) and must know what letdown pressure would be when the relief valve lifts.

References:

STM 1-04, Makeup and Purification

History:

Exam Bank: OpsUnit1 QuestionID: ANO-OPS1-2133
Selected for the 2008 RO Exam
Selected for 2018 RO exam
Rev. 2, Replaced T-20 in answers A and B with T-76 since T-20 was not plausible as it does not receive any relief liquid. Editorial changes.

QID: 0963 **Rev:** 1 **Rev Date:** 1/10/18 **Source:** Bank **Originator:** NRC

TUOI: A1LP-RO-RCS **Objective:** EO21 **Point Value:** 1

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

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

Description: Knowledge of the operational implications of the following concepts as they apply to the CVCS: Response of PRT during bubble formation in PZR: increase in Quench Tank pressure when cycling PORV shows that complete steam bubble does not exist, that significant noncondensable gas is still present.

K/A Number: K5.40 **CFR Reference:** 41.5 / 45.7

Tier: 2 **RO Imp:** 3.0 **RO Select:** Yes **Difficulty:** 3

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

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

Given:
* Plant startup is in progress
* RCS pressure 150 psig
* RCS temperature 225 °F
* Pressurizer level 90 inches
* Pressurizer Operation (1103.005) in use to draw steam bubble in pressurizer

A three minute blow through the ERV resulting in a Quench Tank pressure rise of ____(1)____ psig in conjunction with saturation conditions in the ____(2)____ is indication that a steam bubble has now formed in the pressurizer.

A. (1) two or more
(2) RCS

B. (1) less than or equal to one
(2) pressurizer

C. (1) two or more
(2) pressurizer

D. (1) less than or equal to one
(2) RCS

Answer:

B. (1) less than or equal to one
(2) pressurizer

Notes:

"B" is correct, IAW with 1103.005, Pressurizer Operations, a three minute blow through the ERV resulting in a Quench Tank pressure rise of ≤ 1 psig combined with saturation temperature/pressure conditions in the pressurizer is the indication to the operators that a steam bubble has now formed in the pressurizer.

"A" is incorrect but plausible since the pressure rise given is the same as the acceptance criterion in 1103.005 Supplement 3 for stroke of the Nitrogen supply check valve to the Quench Tank. The second part is plausible since the pressurizer is attached to the RCS.

"C" is incorrect but plausible since the pressure rise given is the same as the acceptance criterion in 1103.005 Supplement 3 for stroke of the Nitrogen supply check valve to the Quench Tank. The second part is correct.

"D" is incorrect but plausible since the time given is correct. The second part is plausible since the pressurizer is attached to the RCS.

This question matches the K/A since the knowledge required to correctly answer the question is the same as the knowledge required to know that significant noncondesabel gases are still present in the PZR.

References:

1103.005, Pressurizer Operations

History:

New for 2013 Exam
Rev.1, editorial changes, changed "A" and "C" first part to "two or more" vs. three for added plausibility.

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

QID: 1178 **Rev:** 0 **Rev Date:** 7/25/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-DHR **Objective:** 19 **Point Value:** 1

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

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

Description: Knowledge of RHRS design feature(s) and/or interlock(s) which provide for the following: Lineup for "piggy-back" mode with high-pressure injection.

K/A Number: K4.08 **CFR Reference:** 41.7

Tier: 2 **RO Imp:** 3.1 **RO Select:** Yes **Difficulty:** 3

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

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

Given:
* LOCA has occurred causing unit trip
* HPI Cooldown (1202.011) in use
* BWST level 6.5 ft and dropping
* CRS directs performance of RT-15

In accordance with RT-15, Shift to RB Sump Suction, which of the following would REQUIRE DH Supply to Makeup Pump Suction valves to be OPEN?

A. CET SCM adequate AND
Both LPI pump flows 2900 gpm per pump

B. CET SCM adequate AND
RCS pressure > 150 psig

C. Makeup Tank level > 86" AND
Both LPI pump flows 2900 gpm per pump

D. Makeup Tank level > 86" AND
RCS pressure > 150 psig

Answer:

B. CET SCM adequate AND
RCS pressure > 150 psig

Notes:

"B" is correct. Per RT-15 if HPI is in service, then HPI may be terninated:
If All of the following satisfied:
* CET SCM adequate
* Any LPI flow exists
* HPI throttle to < 110 gpm/pump
* RCS press and temp are NOT rising
OR BOTH of the following satisified:
* CETs do not indicate superheat
* Both LPI pump ≥ 2800 gpm each OR One LPI pump ≥ 3050 gpm.
Only B contains conditions requiring use of piggy-back mode: CET SCM is adequate (first set of conditions above are satisfied due to SCM) but RCS pressure > 150 psig so no LPI flow would be indicated (both sets of conditions not satisifed due to no LPI flow).

"A" is incorrect but plausible since CET SCM is adequate and LPI pump flows are 2900 gpm each, thus HPI could be secured with the verification of other parameters. Piggy-back mode would not be required in this case

"C" is incorrect but plausible, both LPI pumps flows are one of the criteria for securing HPI piggy-back mode and while the flows listed are less than the flow for one LPI pump running (3050 gpm), they are above the critieria for two pumps running (≥2800 gpm) and thus the piggy-back valves could be closed as long as CETs do not indicate superheat. If the Makeup Tank level were greater than 86", then the piggy-back valves would be closed until Makeup Tank level was 55 to 86".

"D" is incorrect but plausible since CET do not indicate superheat is one of the critieria for closing piggy-back valves but LPI flows cannot be present with an RCS pressure > 150 psig so the valves must remain open. If the Makeup Tank level were greater than 86", then the piggy-back valves would be closed until Makeup Tank level was 55 to 86".

This question matches the K/A since it requires the applicant to know when piggy-back mode is required in conjunction with high pressure injection.

References:

1202.011, HPI Cooldown
1202.012, RT-15

History:

New for 2018 exam

QID: 0091 **Rev:** 1 **Rev Date:** 9/11/17 **Source:** Modified **Originator:** Cork

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

Section: 3.4 **Type:** Heat Removal From Reactor Core

System Number: 005 **System Title:** Residual Heat Removal System (RHRS)

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RHRS controls including: Heatup/cooldown rates.

K/A Number: A1.01 **CFR Reference:** 41.5 / 45.5

Tier:	2	**RO Imp:**	3.5	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.6	**SRO Select:**	No	**Taxonomy:**	H

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

Given:
* Plant cooldown in progress
* P-34A DH Removal pump in service
* DH suction temperature 148 °F

What is the MAXIMUM allowable RCS cooldown rate per Plant Shutdown and Cooldown (1102.010) for the above conditions?

A. 15 °F/hr

B. 25 °F/hr

C. 50 °F/hr

D. 100 °F/hr

Answer:

B. 25 °F/hr

Notes:

"B" is correct, per 1102.010 this cooldown rate is applicable when RCS is less than 150 °F.

"A" is incorrect, but plausible since 1102.010 refers to a 15 °F maximum step change several times.

"C" is incorrect, but plausible since the maximum cooldown rate is 50°F/hr when the RCS is ≥150°F.

"D" is incorrect, plausible since this is allowed per EOPs, but is not allowed by 1102.010 (except for the Pressurizer).

This question matches the K/A since it involves RHR system (DH at ANO-1) and requires applicant to know the procedural limit for cooldown rate.

References:

1102.010, Plant Shutdown and Cooldown

History:

Developed for 1998 RO/SRO exam
Used in 2005 RO Exam
Rev. 1, editorial changes, added procedure to stem, deleted RCPs running, modified by adding DH suction temperature <150 °F so that "A" is correct. Replaced 75 °F/hr with 15 °F/hr since 75 was non-existent, and re-ordered answers.
Selected for 2018 exam

QID: 0091 **Rev:** 0 **Rev Date:** 7/12/98 **Source:** Direct **Originator:** JCork

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

Section: 3.4 **Type:** Heat Removal From Reactor Core

System Number: 005 **System Title:** Residual Heat Removal System (RHRS)

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RHRS controls including: Heatup/cooldown rates.

K/A Number: A1.01 **CFR Reference:** 41.5 / 45.5

Tier:	2	**RO Imp:**	3.5	**RO Select:**	No	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.6	**SRO Select:**	No	**Taxonomy:**	C

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

Given:

- Plant cooldown in progress
- "A" DH Removal pump in service
- "A" and "C" RCPs running

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What is the maximum allowable cooldown rate in this condition?

A. 25°F/hr

B. 50°F/hr

C. 75°F/hr

D. 100°F/hr

Answer:

B. 50°F/hr

Notes:

Per 1102.010, the maximum cooldown rate is 50°F/hr when the RCS is <280°F and >150°F, "b" is correct.
DH cannot be placed in service unless the RC temp is <280°F, therefore "D" is incorrect.
The RCP's are removed from service while a PZR steam bubble is still present and specifically when RC temp is between 166°F and 270°F, therefore "A" is incorrect.
"C" is fictitious.

References:

1102.010, Plant Shutdown and Cooldown, Chg. 053-10-0

History:

Developed for 1998 RO/SRO exam
Used in 2005 RO Exam

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

QID: 1228 **Rev:** 1 **Rev Date:** 1/10/18 **Source:** New **Originator:** Cork

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

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

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

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:**		**SRO Select:**	No	**Taxonomy:**	F

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

Given:
* Unit 1 at 100% power
* CFT T-2A level inadvertently raised to 14.2 ft.
* Level being lowered per 1104.001, Section 9.0, Bleeding and Draining Core Flood Tanks

Of the following which is an acceptable level, per Tech Specs, where draining the "A" CFT can be stopped?

A. 13.5 ft.

B. 12.8 ft.

C. 12.5 ft.

D. 12.0 ft.

Answer:

B. 12.8 ft.

Notes:

"B" is correct, procedure 1104.001 lmit and precaution 5.4.1 contains Tech Spec limits for level which include instrument uncertainty. The low Tech Spec level limit with instrument uncertainty is 12.6 feet, so a level of 12.8 feet is above this and is also above the low level alarm setpoint of 12.74 ft.

"A" is incorrect but plausible since this is close to a design limit but this is above the high Tech Spec level limit of 13.4 ft. including instrument uncertainty.

"C" is incorrect but plausible since this is a design limit but this is below the 12.6 ft low Tech Spec level limit with instrument uncertainty.

"D" is incorrect but plausible since this is just above the Tech Spec low level of 11.95 ft but this value does not include instrument uncertainty and is therefore incorrect.

This question matches the K/A since it involves an ECCS component and the applicant must know the design limits of the Core Flood Tanks to prevent exceeding them during a draining evolution.

References:

1104.001, Core Flood System Operating Procedure
ANO-1 Technical Specifications 3.5.1 bases

History:

New for 2018 exam
Rev. 1, revised stem wording per NRC resolution.

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

QID: 0463 **Rev:** 2 **Rev Date:** 1/10/18 **Source:** Bank **Originator:** Giles

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

Section: 3.5 **Type:** Containment Integrity

System Number: 007 **System Title:** Pressurizer Relief Tank / Quench Tank

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

K/A Number: A2.01 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

Tier: 2 **RO Imp:** 3.9 **RO Select:** Yes **Difficulty:** 3

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

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

Given:
* Unit 1 100% power
* Pressurizer Code Safety leaking by

NOW
* RCS pressure 2000 psig and going down
* Annunciator RELIEF VALVE OPEN (K09-A1) in alarm
* Code Safety Relief Valve Acoustic Monitor VYI-1001A rising
* All Pressurizer heaters ON

Considering the above, which of the following is procedurally required to mitigate these conditions?

A. Perform "RCS Leakage Monitoring" section of RCS Leak Detection (1103.013).

B. Commence plant shutdown per Power Reduction and Plant Shutdown (1102.016).

C. Trip the reactor and go to Reactor Trip (1202.001).

D. Commence rapid plant shutdown per Rapid Plant Shutdown (1203.045).

Answer:

C. Trip the reactor and go to Reactor Trip (1202.001).

Notes:

"C" is correct, per 1203.015, Section 2 - Leaking Pressurizer Code Safety Valve, if code safety valve leakage exceeds capability to maintain RCS pressure, then trip reactor and perform 1202.001.

"A" is incorrect but plausible since this action is in 1203.015, Section 2, step 3, but would only be perfomed for a leaking safety where RCS pressure is not decreasing rapidly.

"B" is incorrect but plausible since this action would be taken if Code Safety leakage were greater than 1 gpm and deemed unsafe.

"D" is incorrect but plausible since this action would be taken if total RCS leakage were exceeding Tech Specs but is inappropriate for RCS pressure decreasing rapidly.

The question matches the K/A since conditions are given for a stuck open PZR code safety and the applicant must assess the severity of the conditions, determine the impact on the plant, and select the appropriate action which would mitigate that impact.

References:

1203.015, Pressurizer Systems Failure

History:

Created for 2002 RO/SRO exam.
Rev. 1, 9/11/17, added condition that code safety was leaking. Replaced A, B, and D distractors since they didn't have anything to do with a code safety and were thus implausible. Revised stem.
Selected for 2018 exam
Rev. 2, 1/10/18, replaced distractor A with RCS Leak Detection procedure, per NRC resolution.

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

QID: 1227 **Rev:** 1 **Rev Date:** 1/10/18 **Source:** New **Originator:** Cork

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

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

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: Effects of shutting (automatically or otherwise) the isolation valves of the letdown cooler.

K/A Number: A2.08 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

Tier: 2 **RO Imp:** 2.5 **RO Select:** Yes **Difficulty:** 3

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

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

Given:
* Unit One at 100% power
* Letdown cooler E-29B being removed from service for maintenance
* Annunciator LETDOWN TEMP HI (K10-A8) alarms
* Letdown temperature 132 °F

The cause of the alarm is the __(1)___ second time delay between ICW inlet (CV-2216) and cooler inlet (CV-1213) malfunctioned and the required ACA action is to __________(2)________________ .

A. (1) 8
(2) reduce letdown flow until temperature lowers

B. (1) 20
(2) reduce letdown flow until temperature lowers

C. (1) 8
(2) cycle letdown coolers outlet (CV-1221) until temperature lowers

D. (1) 20
(2) cycle letdown coolers outlet (CV-1221) until temperature lowers

Answer:

A. (1) 8
(2) reduce letdown flow until temperature lowers

Notes:

"A" is correct. An interlock between the letdown coolers ICW inlet valves and RCS inlet valves prevents hot RCS fluid from entering the coolers before the ICW cooling valves are opened. A single handswitich is used to open both the ICW inlet (CV-2216) and cooler inlet (CV-1213). The ICW inlet valve opens first and after a 20 second time delay the cooler inlet opens. The reverse is true when closing the inlets: the cooler inlet goes closed first and after an 8 second time delay the ICW inlet starts to close. A failure of the time delay could cause the ICW inlet to start going closed immediately and a high letdown temperature condition could occur. The alarm comes in at 130 °F, five degrees before high temperature isolation. ACA 1203.012I states to reduce letdown flow until temperature lowers.

"B" is incorrect but plausible since there is a 20 second time delay when these valves are opened, not closed. The action is correct per the ACA.

"C" is incorrect but plausible since the time delay for these valves is correct. Cycling CV-1221 is a technique used when re-establishing letdown following a high temperature isolation (135 °F setpoint) but is not specified by the ACA as an immediate response to this alarm.

"D" is incorrect but plausible since there is a 20 second time delay but this is when the valves are opening, not closing. Cycling CV-1221 is a technique used when re-establishing letdown following a high temperature isolation (135 °F setpoint) but is not specified by the ACA as an immediate response to this alarm.

The first part of the K/A is to predict the impact of closing the isolation valves of the letdown cooler. This would lead to either two correct answers or no plausible distractors so the question was written to predict the malfunction which caused the impact (the alarm coming in) and matches the second part of the K/A, to use procedures to mitigate the consequences.

References:

1203.012I, Annunciator K10 Corrective Action
1104.002, Makeup & Purification System Operation
STM 1-04, Makeup & Purification

History:

New for 2018 exam
Rev.1, 1/10/18, added condition of Letdown temperature 132 °F, this confirms temp alarm but is less than isolation setpoint of 135, per NRC resolution.

QID: 0562 **Rev:** 1 **Rev Date:** 5/19/17 **Source:** Repeat **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 backup.

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:** 36 **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 from B-11,
P33B from B-12,
P33C from B-13

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

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.
Repeated for 2018 exam

QID: 1213 **Rev:** 0 **Rev Date:** 9/13/17 **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:** 41.7 / 45.5

Tier: 2 **RO Imp:** 3.6 **RO Select:** Yes **Difficulty:** 2

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

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

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

Assuming no malfunctions, what is the pressure setpoint at which ATC will observe Pressurizer Heater Bank 3 automatically turn ON?

A. 2155 psig

B. 2140 psig

C. 2135 psig

D. 2125 psig

Answer:

C. 2135 psig

Notes:

"C" is correct, the Pressurizer Pressure controller sends a signal to Heater Bank 3 to turn ON at 2135 psig decreasing.

"A" is incorrect but plausible, this is the setpoint at which Heater Bank 3 turns OFF.

"B" is incorrect but plausible as Heater Bank 4 turns OFF at this pressure.

"D" is incorrect but plausible as Heater Bank 5 turns OFF at this pressure.

This question matches the K/A since the applicant must know when the presurizer heater bank 3 normally turns on to be able to monitor automatic operation.

References:

1103.005, Pressurizer Operation

History:

New question for 2018 exam

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

QID: 1230 **Rev:** 1 **Rev Date:** 1/10/18 **Source:** New **Originator:** Cork

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

Section: 3.7 **Type:** Instrumentation

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

Description: Knowledge of the physical connections and/or cause-effect relationships between the RPS and the following systems: SDS (Steam Dump System).

K/A Number: K1.07 **CFR Reference:** 41.2 to 41.9 / 45.7 to 45.8

Tier:	2	**RO Imp:**	2.8	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**		**SRO Select:**	No	**Taxonomy:**	F

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

Which of the following is an input signal that originates with the Reactor Protection System and is an input to the Turbine Bypass Valves?

A. SG pressure

B. Reactor power

C. RCS pressure

D. Reactor trip confirm

Answer:

D. Reactor trip confirm

Notes:

"D" is correct. There is no direct link between the RPS and the Turbine Bypass Valves (Steam Dump System) but the RPS does trip the CRD breakers and a Trip Confirm signal (actually from the CRD breaker cabinets) is used to apply a 100 psig bias to the setpoint of the Turbine Bypass Valves as part of the Integrated Master Subsystem in the Integrated Control System. The 100 psig bias limits the opening of the Turbine Bypass Valves following a Reactor Trip to limit the cooldown effect post-trip (prevents PZR from emptying due to cooldown).

"A" is incorrect but plausible since SG pressure is an input to the Turbine Bypass Valves for pressure control but it comes from the NNI system, not RPS. Some NNI signals originate from RPS but not this one.

"B" is incorrect but plausible since power demand is an input to the Turbine Bypass Valves for application of the 50 psig bias but this comes from power demand internal to ICS, not RPS.

"C" is incorrect but plausible since RCS pressure signals originate from RPS but RCS pressure is not an input to the Turbine Bypass Valves.

This matches the K/A since the question requires knowledge of a relationship between RPS and the Turbine Bypass Valves (steam dumps).

References:

STM 1-64, Integrated Control System

History:

New for 2018 exam
Rev. 1, revised distractor C to RCS pressure, per NRC resolution.

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

QID: 1231 **Rev:** 0 **Rev Date:** 9/29/17 **Source:** New **Originator:** Cork

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

Section: 3.7 **Type:** Instrumentation

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

Description: Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

K/A Number: 2.2.25 **CFR Reference:** 41.5 / 41.7 / 43.2

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

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

Given:
* Unit 1 at 90% power
* ICS in manual
* Slow moderator dilution event occurs

Which of the following RPS trips is designed to provide PRIMARY protection for the above accident?

A. High Power

B. High RCS Pressure

C. High RCS Temperature

D. High Power/Imbalance/Flow

Answer:

B. High RCS Pressure

Notes:

"B" is correct per Tech Spec Bases for LCO 3.3.1 and lesson plan.

"A" is incorrect but plausible since the High Power trip provides protection for high reactivity insertion events such as a rod ejection accident.

"C" is incorrect but plausible since pressure and temperature usually trend together but the high pressure trip provides the protection for the slow reactivity event.

"D" is incorrect but plausible since it is a reactor power limiting protective trip but is not credited for protection fo this type of event.

Normally this K/A would not be used for an RO level question but ANO-1 has an RO level lesson plan objective and supporting material to show the bases for RPS trips is taught in the classroom.

References:

Reactor Protection System, A1LP-RO-RPS

History:

New for 2018 exam

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

QID: 0192 **Rev:** 1 **Rev Date:** 9/19/17 **Source:** Bank **Originator:** Haynes

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

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

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

Description: Knowledge of bus power supplies to the following: ESFAS/safeguards equipment control.

K/A Number: K2.01 **CFR Reference:** 41.7

Tier: 2 **RO Imp:** 3.6 **RO Select:** Yes **Difficulty:** 3

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

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

Given:
* Unit 1 at 100%
* Reactor Building Pressure Transmitter (PT-2407) has failed high
* Annunciator ESAS PARTIAL TRIP (K11-F6) alarms
* CBOT reports ESAS Analog Channel 3 tripped

With the above conditions, a power loss to which of the following would cause an ESAS actuation?

A. RS1

B. RA1

C. RS3

D. RA2

Answer:

A. RS1

Notes:

"A" is correct. A loss of power to RS1 will trip Analog Channel 1 which would then complete the 2 out of 3 analog trip logic causing an ESAS actuation.

"B" and "D" are incorrect. RA1 and RA2 are important low voltage power panels but they do not result in loss of ESAS functions.

"C" is incorrect. Analog 3 would be tripped as a result of a power loss to RS3. Analog 3 is already tripped.

References:

1105.003, Engineered Safeguards Actuation Signal

History:

Developed for use in 98 RO Re-exam
Used in 2001 RO/SRO Exam. KA K6.01
Selected for 2007 RO Exam.
Selected for 2013 RO Exam.
Selected for 2018 exam
Rev. 1, replaced B71 and B72 with RA1 and RA2, believe they are more plausible.

QID: 0274 **Rev:** 1 **Rev Date:** 9/19/17 **Source:** Bank **Originator:** D. Slusher

TUOI: A1LP-RO-RBS **Objective:** 5 **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:**	1.5
Group:	1	**SRO Imp:**		**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 41 **SRO:**

What would be the consequences if the Reactor Building Cooler Chilled Water Bypass Dampers remained latched after an ESAS actuation?

A. Inadequate air flow through the Service Water Cooling Coils

B. Excessive heat load on the Chilled Water System

C. Damage to RB ventilation plenum from excessive pressure

D. Excessive current on the cooling fan motors

Answer:

A. Inadequate air flow through the Service Water Cooling Coils

Notes:

"A" is the correct answer, the bypass dampers drop to allow more flow through the Service Water coils by bypassing the Chilled Water coils and thus provide more cooling to the RB atmosphere.

"B" is incorrect, the Chilled Water System is isolated on ESAS and therefore no additional heat load will be placed on it. Plausible if applicant did not recognize this.

"C" is incorrect, plausible due to the high RB pressures possible after a LOCA but the RB ventilation plenum has been analyzed for these conditions and it will withstand the pressures after an ESAS.

"D" is incorrect, plausible if applicant thought fans would do additional work thus current would rise but current on the motors is not a concern in this situation.

This question matches the K/A since the applicant must have knowledge of the interrelationship between the Containment Coolers, the purpose of the bypass damper, and the SW system to correctly answer this question.

References:

1104.033, Reactor Building Ventilation

History:

Developed for 1999 exam.
Used in 2001 RO Exam.
Selected for 2013 RO Exam
Selected for 2018 exam
Rev. 1, editorial changes

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

QID: 1221 **Rev:** 0 **Rev Date:** 9/19/17 **Source:** New **Originator:** Cork

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

Section: 3.5 **Type:** Containment Integrity

System Number: 026 **System Title:** Containment Spray

Description: Knowledge of CSS design feature(s) and/or interlock(s) which provide for the following: Source of water for CSS, including recirculation phase after LOCA.

K/A Number: K4.01 **CFR Reference:** 41.7

Tier: 2 **RO Imp:** 4.2 **RO Select:** Yes **Difficulty:** 3

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

Question: **RO:** 42 **SRO:**

Given:
* LOCA occurred
* ESAS actuated on all channels
* Annunciators RB SPRAY FLOW HI (K11-C4/C5) are in alarm

NOW (two hours later)
* Spray flow alarms are clear
* BWST level 5.5 ft

(1) What is the suction source for the RB spray pumps and (2) what is each train's flow rate?

A. (1) RB sump
(2) 1100 gpm

B. (1) BWST
(2) 1500 gpm

C. (1) RB sump
(2) 1500 gpm

D. (1) BWST
(2) 1100 gpm

Answer:

A. (1) RB sump
(2) 1100 gpm

Notes:

"A" is correct. At a BWST level of 6 ft. the suction for the RB Spray and LPI pumps will be transferred to the RB sump, so the RB Spray pumps will be taking suction from the RB sump. The annunciators for high RB spray flow alarm when spray flow is ≥1700 gpm. The annunciator response, as well as the ESAS procedure, gives direction to throttle spray flow to 1050-1200 gpm if an ES signal is present. The lower flow rate ensures adequate NPSH following transfer to RB sump recirculation.

"B" is incorrect, the RB spray pumps will be transferred to take suction from the RB sump. The flow rate of 1500 gpm is plausible since this is the flow rate at which the pumps are tested on a monthly basis and this flow rate will clear the annunciator.

"C" is incorrect but plausible since the suction source is correct but the flow rate of 1500 gpm is too high. The flow rate of 1500 gpm is plausible since this is the flow rate at which the pumps are tested on a monthly basis and this flow rate will clear the annunciator.

"D" is incorrect but plausible since the flow rate is correct but the suction source will be from the RB sump at this point.

This question matches the K/A since it involves RB Spray design feature of transferring suction to RB sump on low BWST level for recirculation of RB sump contents.

References:

1202.010, ESAS
1203.012J, Annunciator K11 Corrective Action

History:

New for 2018 exam

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

QID: 1155 **Rev:** 1 **Rev Date:** 5/19/17 **Source:** Repeat **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:** 43 **SRO:**

Given:
* 100% power
* Main Steam leak occurs

Control rooom operators will INITIALLY observe a Reactor power ___(1)___ and Main Generator MW ____(2)____.

A. (1) rise
(2) stay the same

B. (1) drop
(2) drop

C. (1) rise
(2) drop

D. (1) drop
(2) stay the same

Answer:

C. (1) rise
(2) 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 1-64, 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.

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

QID: 0537 **Rev:** 1 **Rev Date:** 9/20/17 **Source:** Bank **Originator:** NRC

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

Section: 3.4 **Type:** Heat Removal From Reactor Core

System Number: 059 **System Title:** Main Feedwater System

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MFW controls including: Power level restrictions for operation of MFW pumps and valves.

K/A Number: A1.03 **CFR Reference:** 41.5 / 45.5

Tier:	2	**RO Imp:**	2.7	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	2.9	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 44 **SRO:**

Given:
* Unit 1 just completed refueling outage
* Power escalation in progress

The second Main feed pump is placed in service ______________________________ per Power Operations (1102.004).

A. Prior to exceeding 50% power using the Gamma Metrics Linear Power instrument.

B. Prior to exceeding 350 Mwe on generator output (~35% on ULD).

C. Prior to reaching 40% open on MFW pump Low Load Control Valve Demand.

D. Prior to exceeding 45% setting (~450 Mwe) on each FW Loop Demand.

Answer:

B. prior to exceeding 350 Mwe on generator output (~35% on ULD).

Notes:

"B" is correct. Prior to exceeding 35% setting (~350 Mwe) on the Unit Load Demand (ULD), is the correct answer per step 7.10 in 1102.004.

"A" is incorrect but plausible due to the existence of a procedure step checking MFW flow against linerar power prior to AMSAC being enabled. This is incorrect because the actual procedure statement is "Check MFW flow is >0.90 e6 lbm/hr prior to Gamma Metrics Linear Power rising above 45% power".

"C" is incorrect because step 7.10 states that "WHEN ~350 Mwe is reached OR prior to reaching 90% open on Low Load Control Valve demand, THEN perform the following: 7.10.1 Place second MFWP (P-1A or P-1B) in service." It is plausible since the Low Load demand is one of the key indications for when to place a second MFWP in service.

"D" is incorrect, the ULD demand is what is used to determine when to place a second MFW pump in service, and the value is too high at 45%. Plausible since Feedwater Demand signal goes to the Feedwater Pumps.

References:

1102.004, Power Operations

History:

Developed by NRC.
Used on 2004 RO Exam.
Selected for 2005 RO re-exam.
Selected for 2018 exam
Rev. 1, editorial changes, changed correct answer from 36% and 360 Mwe to 35% and 350 Mwe due to procedure change. Modified "D" to be FW Demand vs. ULD so question does not appear to be a "2x4".

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

QID: 1232 **Rev:** 0 **Rev Date:** 10/2/17 **Source:** New **Originator:** Cork

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

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

System Number: 061 **System Title:** Auxiliary/Emergency Feedwater

Description: Knowledge of the effect of a loss or malfunction of the following will have on the AFW components: Controllers and positioners

K/A Number: K6.01 **CFR Reference:** 41.7 / 45.7

Tier: 2 **RO Imp:** 2.5 **RO Select:** Yes **Difficulty:** 3

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

Question: **RO:** 45 **SRO:**

Given:
* Unit 1 tripped due to a loss of both MFW pumps
* SG "A" pressure 1000 psig and steady
* SG "B" pressure 500 psig and dropping

NOW
* EFW pump (P-7B) Flow Control Valves (CV-2646 and CV-2648) remote valve positioner loses power

This malfunction will cause the P-7B Flow Control Valves to fail _____(1)____ and the flow rate to each SG from P-7B will be _____(2)____.

A (1) open
(2) "A" 100%; "B" 100%

B. (1) closed
(2) "A" 0%; "B" 0%

C. (1) open
(2) "A" 100%; "B" 0%

D. (1) mid-position
(2) "A" 50%; "B" 50%

Answer:

C. (1) open
(2) "A" 100%: "B" 0%

Notes:

"C" is correct. The EFW flow control valves' remote valve positioners are DC powered. On a loss of power the valves fail open. However, the question conditions show there will be a Vector Isolation signal for the "B" SG due to it being less than 600 psig. The Vector Isolation signal goes to the isolation valves as well as the flow control valves, therefore EFW flow will be 100% to "A" and 0% to "B" from P-7B.

"A" is incorrect but plausible since the valves do fail open but the Vector Isolation signal will close the isolation valve to "B" and it will not be at 100%.

"B" is incorrect but plausible since there are valves that fail closed on loss of power but not the EFW flow control valves.

"D" is incorrect but plausible since there are instrument loops that fail mid-position but this does not occur with the EFW flow control valves.

This question matches the K/A since it tests the applicants' ability to recall the failure mode of EFW flow control valve positioners.

References:

STM 1-66, Emergency Feedwater Initiation and Control

History:

New for 2018 exam

QID: 1229 **Rev:** 1 **Rev Date:** 1/10/18 **Source:** New **Originator:** Cork

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

Section: 3.6 **Type:** Electrical

System Number: 062 **System Title:** AC Electrical Distribution

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: 2 **RO Imp:** 4.0 **RO Select:** Yes **Difficulty:** 3

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

Question: **RO:** 46 **SRO:**

A loss of which of the following would cause an entry into T.S. LCO 3.8.1, AC Sources Operating (assume unit is in an applicable mode for this LCO)?

A. Auto Transformer

B. 120V AC Panel RS1

C. 161KV Pleasant Hill line

D. AC supply to Battery Charger D-03A

Answer:

A. Auto Transformer

Notes:

"A" is correct, a loss of the Auto Transformer means a loss of offsite power to Startup Transformer #1 rendering it inoperable. This requires entry into 3.8.1 required action A.1 which has a one hour completion time making this RO level.

"B" is incorrect but plausible since this would cause entry into a similar LCO, 3.8.9, Distribution - Operating.

"C" is incorrect but plausible since this is one of two 161KV sources to SU 2, but only one is required.

"D" is incorrect but plausible since this would cause entry into a similar LCO, 3.8.9, Distribution - Operating but only if the D-03B charger were inoperable also.

This question matches the K/A since it requires the applicant to have knowledge of limiting conditions for operations for the AC electrical distribution system.

References:

Technical Specifications, 3.8.1
1107.001, Electrical System Operations

History:

New for 2018 exam
Rev. 1, revised answers short to long, per NRC resolution.

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

QID: 0384 **Rev:** 3 **Rev Date:** 9/20/17 **Source:** Bank **Originator:** R.Soukup

TUOI: A1LP-RO-ELECD **Objective:** 14.f **Point Value:** 1

Section: 3.6 **Type:** Electrical

System Number: 063 **System Title:** DC Electrical Distribution

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the DC Electrical Systems; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Grounds.

K/A Number: A2.01 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

Tier: 2 **RO Imp:** 2.5 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 3.2 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 47 **SRO:**

Given:
* Unit 1 at 100%
* Annunciator DO1 TROUBLE (K01-D7) alarms
* Inside AO reports local trouble annunciator is "GROUND ALARM"
* Inside AO reports GROUND ALARM with 55 volts on positive bus (V1)

Given these conditions, (1) what is the impact to the DC electrical distribution system and (2) what action per Battery And 125V DC Distribution (1107.004) would mitigate these consequences?

A. (1) Impact: No immediate impact.

(2) Mitigation: Electrical maintenance support required to determine ground location.

B. (1) Impact: No immediate impact.

(2) Mitigation: Transfer D11 to D21 and see if ground still present on D01.

C. (1) Impact: Entry into LCO 3.8.4, DC Sources -Operating, is required.

(2) Mitigation: Electrical maintenance support required to determine ground location.

D. (1) Impact: Entry into LCO 3.8.4, DC Sources -Operating, is required.

(2) Mitigation: Transfer D11 to D21 and see if ground still present on D01.

Answer:

A. (1) Impact: No immediate impact.

(2) Mitigation: Electrical maintenance support required to determine ground location.

Notes:

"A" is correct in accordance with 1107.004, electrical maintenance support is needed to determine ground location and there is no immediate impact due to Unit 1 DC being a floating system (ungrounded). The voltage indication of 50 volts means there is a high resistance ground so this is not a significant ground. If either V1 or V2 indicated less than 20 volts, then this would be a low resistance gound and is a much more severe situation.

"B" is incorrect, this is plausible snce the impact is correct but transferring D11 to D21 is not allowed in modes 1 4. It is, however, allowed in Modes 5 and 6 as a means to determine ground location and is thus plausible.

"C" is incorrect, a single high-resistance ground will not short a floating system and thus there is no immediate impact and the red train DC system is operable. The mitigation method is correct.

"D" is incorrect, transferring D11 to D21 is not allowed in modes 1-4. It is, however, allowed in Modes 5 and 6. A single high-resistance ground will not short a floating system and thus there is no immediate impact and the red train DC system is operable.

This question matches the K/A since it presents the condition of a DC ground and requires the applicant to assess the severity of the ground for impact and to recall the applicable procedural mitigation method for the ground.

References:

1107.004, Battery And 125V DC Distribution
1203.012A, Annunciator K01 Corrective Action

History:

New for 2001 RO/SRO Exam.
Selected for the 2008 RO Exam
Rev. 3, removed "immediately" from distractor D, revised condtions, added "55 volts on V1", changed C and D to state impact is entry into LCO 3.8.4 required since this is "above the line". Added 1107.004 to question stem
Selected for 2018 exam

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

QID: 0849 **Rev:** 1 **Rev Date:** 9/20/17 **Source:** Bank **Originator:** Cork

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

Section: 3.6 **Type:** Electrical

System Number: 064 **System Title:** Emergency Diesel Generators

Description: Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: Fuel oil storage tanks.

K/A Number: K6.08 **CFR Reference:** 41.7 / 45.7

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

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

Question: **RO:** 48 **SRO:**

Given:
* #1 EDG in 8th hour of a 24 hour full load run after maintenance
* #2 EDG in standby
* AO completed a fuel oil tanker truck off-load
* Sediment from tanker entered Fuel Oil Bulk tank T-25 and caused outlet filter F-27 to clog

What would be the result of this condition on the #1 EDG Fuel Oil system?

A. Fuel Oil Day Tank T-30A level low alarm

B. Fuel Transfer Pump P-16A damage

C. Fuel Oil Storage Tank T-57A level low alarm

D. Fuel Transfer Pump P-16A discharge pressure hi alarm

Answer:

C. Fuel Oil Storage Tank T-57A level low alarm

Notes:

"C" is correct, with #1 EDG running fully loaded it would require fuel oil continuously, with the F-27 filter betwee the Bulk tank and storange tank T-57A clogged, then the T-57A tank would no longer "float" on the bulk tank and and level would lower. T-57A is vented to atmosphere to prevent collapse. Components downstream of the storage tank would operate normally.

"A" is incorrect, although a T57A storage tank low alarm would result, the transfer pump P-16A would still be able to fill the Day Tank.

"B" is incorrect, plasuible if the applicant believed the strainer was on the suction side of P-16A.

"D" is incorrect, if the applicant believed the sediment in T-25 would be carried over to T-57A, then the applican would deduce the transfer pump's discharge filter would clog.

This question matches the K/A since it requires knowledge of effect a malfunction of the bulk tank outlet filter will have on the fuel oil storage tanks.

References:

1203.012A, Annunciator K01 Corrective Action

History:

Direct from Unit 2 regular exam bank ANO-OpsUnit2-10285
Selected for 2011 RO Exam.
Rev. 1, editorial changes, replaced B distractor with P-16A damage, changed D to discharge pressure vs. D/P.

Selected for 2018 exam

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

QID: 1235 **Rev:** 1 **Rev Date:** 1/10/18 **Source:** New **Originator:** Cork

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

Section: 3.6 **Type:** Electrical

System Number: 064 **System Title:** Emergency Diesel Generator

Description: Knowledge of ED/G system design feature(s) and/or interlock(s) which provide for the following: Trips while loading the ED/G (frequency, voltage, speed).

K/A Number: K4.01 **CFR Reference:** 41.7

Tier: 2 **RO Imp:** 3.8 **RO Select:** Yes **Difficulty:** 2

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

Question: **RO:** 49 **SRO:**

Given:
* Unit 1 tripped due to loss of offsite power

Which of the following protective trips will trip the EDG while it is being loaded?

1. Generator Differential
2. Overspeed
3. High crankcase pressure

A. 1 ONLY

B. 3 ONLY

C. 1 and 2

D. 2 and 3

Answer:

C. 1 and 2

Notes:

"C" is correct. Generator differential will trip the EDG via the lockout relay which will open the output breaker and energize the emergency trip relay (K-11) to trip the diesel engine. Mechanical overspeed will always trip the EDG. The high crankcase pressure trip has been modified recently this year (2017) by adding a valve to isolate the high crankcase pressure device from the low lube oil pressure switch.

"A" is incorrect but plausible since generator differential will trip the EDG but so will overspeed.

"B" is incorrect but plausible since high crankcase pressure formerly tripped the EDG but is defeated normally, however the high crankcase pressure trip is enabled during surveillance testing.

"D" is incorrect but plausible since overspeed will trip the EDG and high crankcase pressure formerly tripped the EDG but is defeated normally, however the high crankcase pressure trip is enabled during surveillance testing adding to its plausibility.

This question matches the K/A as it requires knowledge of EDG trips during loading.

References:

STM 1-31, Emergency Diesel Generator
1104.036, Emergency Diesel Generator Operation

History:

New question for 2018 exam

Rev. 1, changed B to "3 ONLY" per NRC resolution.

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

QID: 1226 **Rev:** 0 **Rev Date:** 9/20/17 **Source:** New **Originator:** Cork

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

Section: 3.7 **Type:** Instrumentation

System Number: 073 **System Title:** Process Radiation Montioring

Description: Ability to manually operate and/or monitor in the control room: Check source for operability demonstration.

K/A Number: A4.03 **CFR Reference:** 41.7 / 45.5 to 45.8

Tier: 2 **RO Imp:** 3.1 **RO Select:** Yes **Difficulty:** 2

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

Question: **RO:** 50 **SRO:**

When performing a release of a Treated Waste Monitor Tank how does one check the functionality of the Liquic Radwaste Process Monitor (RI-4642) when the display is reading 1000 cpm or less?

A. Select CHECK SOURCE on RI-4642 drawer and verify count rate rises

B. Lower RI-4642 alarm setpoint until HIGH RAD alarm actuates

C. Remove power fuses to check PROC MONITOR RADIATION HI alarms

D. Pull out drawer and adjust THRESHOLD to minimum until count rate rises

Answer:

A. Select CHECK SOURCE on RI-4642 drawer and verify count rate rises

Notes:

"A" is correct. In 1104.020, Atttachment B1, step 1.6, if the count rate is 1000 cpm or less, then the check source is selected and the operator verifies the count rate rises 100 cpm.

"B" is incorrect but plausible as this is how the operator tests that RI-4642 will close CV-4642 on a high rad alarm.

"C" is incorrect but plausible since this will generate a radiation high alarm but is not used to test functionality.

"D" is incorrect but plausible since this will cause the meter to read upscale but this adjustment is made by I&C technicians.

This question matches the K/A as it asks about use of check source for operaility demonstration for a process radiation monitor.

References:

1104.020, Clean Waste System Operation

History:

New question

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

QID: 0271 **Rev:** 2 **Rev Date:** 9/20/17 **Source:** Bank **Originator:** D. Slusher

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

Section: 3.7 **Type:** Instrumentation

System Number: 073 **System Title:** Process Radiation Monitoring System (PRM)

Description: Knowledge of the effect of a loss or malfunction of the PRM system will have on the following: Radioactive effluent releases.

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

Tier: 2 **RO Imp:** 3.6 **RO Select:** Yes **Difficulty:** 2.5

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

Question: **RO:** 51 **SRO:**

Which of the following must be performed to release TWMT T-16A contents with the Liquid Radwaste Process Monitor (RI-4642) inoperable?

A. Chemistry personnel must have independent sample and analysis results as well as independently verified computer input data.

B. Chemistry must obtain grab samples every hour during a release via this pathway.

C. The release flow rate must be estimated at least once every four hours during the release.

D. Discharge Flume process monitor RI-3618 must be checked for operability.

Answer:

A. Chemistry personnel must have independent sample and analysis results as well as independently verified computer input data.

Notes:

The requirements for release when the Liquid Radwaste Process Monitor is inoperable are:

a. An independent sample and analysis of the tank contents,
b. Computer input data independently verified.

"A" is therefore the correct answer.

"B" is incorrect, plausible since grab samples are required when other rad monitors are inoperable, but they are not required here.

"C" is incorrect, this is plausible since this is done if the flow recorder is inoperable.

"D" is incorrect, plausible since this detector also monitors the flume, it is not required for RI-4642 inoperability.

References:

1104.020, Clean Waste System Operation

History:

Used in 1999 exam.
Direct from ExamBank, QID# 2765
Used in 2001 RO/SRO Exam.
Modified for 2005 RO re-exam.
Selected for 2011 RO Exam.
Rev. 2, editorial changes only

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

QID: 1234 **Rev:** 0 **Rev Date:** 10/2/17 **Source:** New **Originator:** Cork

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

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

System Number: 076 **System Title:** Service Water

Description: Knowledge of the effect that a loss or malfunction of the SWS will have on the following: RHR components, controls, sensors, indicators, and alarms, including rad monitors.

K/A Number: K3.05 **CFR Reference:** 41.7 / 45.6

Tier: 2 **RO Imp:** 3.0 **RO Select:** Yes **Difficulty:** 3

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

Question: **RO:** 52 **SRO:**

Given:
* Plant is heating up following refueling
* P-34B DH suction temperature 200 ° F
* P-4B to P-4C SW Crosstie Valve CV-3640 tagged closed for emergent maintenance

NOW
* Service Water pump P-4C trips and won't restart

What is the effect on plant components and what actions are required to be taken?

A. Spent Fuel Cooling has been lost, enter Unit 1 Spent Fuel Emergencies (1203.050)

B. #2 EDG has lost cooling, enter Tech Spec LCO 3.8.2

C. RCP seals will overheat, enter Reactor Coolant Pump and Motor Emergency (1203.031)

D. SW side of B DH cooler will overheat, enter Loss of Decay Heat Removal (1203.028)

Answer:

D. SW side of B DH cooler will overheat, enter Loss of Decay Heat Removal (1203.028)

Notes:

"D" is correct. Per the caution before step 10 in Section 5 of 1203.028 the SW side of the affected DH cooler could reach saturation temperature due to lack of flow, if RCS temperature is >200 °F.

"A" is incorrect but plausible since 1203.050 is entered on a loss of SW but only for a loss of both trains of SW.

"B" is incorrect but plausible since LCO 3.8.2 addresses EDGs but in this mode only one train is required.

"C" is incorrect but plausible since RCP seals could overheat due to ICW being cooled by SW but with these plant conditions RCPs would not be running.

References:

1203.028, Loss of Decay Heat Removal

History:

New for 2018 exam

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

QID: 0673 **Rev:** 1 **Rev Date:** 9/25/17 **Source:** Bank **Originator:** Possage

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: Ability to monitor automatic operation of the IAS, including: Air pressure.

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

Tier: 2 **RO Imp:** 3.1 **RO Select:** Yes **Difficulty:** 2

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

Question: **RO:** 53 **SRO:**

Given:
* Annunciator INST AIR HEADER PRESS LO (K12-B3) in alarm
* Instrument Air header pressure continuing to lower
* Plant shutdown commenced at 10%/minute

What is the pressure setpoint when the service air to instrument air crossover valve (SV-5400) automatically opens?

A. 35 psig

B. 50 psig

C. 60 psig

D. 75 psig

Answer:

B. 50 psig

Notes:

"B" is correct. SV-5400 will automatically open to crosstie SA with IA at less than or equal to 50 psig.

"A" is incorrect, but plausible since at 35 psig the reactor is tripped per 1203.024.

"C" is incorrect, but plausible since at 60 psig a plant shutdown is commenced at 10% / minute per 1203.024.

"D" is incorrect, but plausible since at 75 psig the Instrument Air Header Pressure Low annunciator alarms..

This question matches the K/A since the applicant must know the setpoint for automatic operation of the service air to instrument air crossover valve to be able to monitor it.

References:

1104.024, Instrument Air System
1203.024, Loss of Instrument Air
1203.012K, Annunciator K12 Corrective Action

History:

New for 2007 RO Exam.
Selected for 2018 exam
Rev. 1, editorial changes only, reworded stem for clarity.

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

QID: 1241 **Rev:** 0 **Rev Date:** 1/11/18 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-RBS **Objective:** E08 **Point Value:** 1

Section: 3.5 **Type:** Containment Integrity

System Number: 026 **System Title:** Containment Spray System

Description: Ability to manually operate and/or monitor in the control room: CSS controls.

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:** **SRO Select:** No **Taxonomy:** F

Question: **RO:** 54 **SRO:**

Given:
* Plant in Mode 1
* RB Spray Pump P-35A quarterly test in progress

NOW
* A RB SPRAY FLOW HI (K11-C4) alarms

Which valve is required to be throttled, IAW 1104.005 or the ACA, in response to this alarm?

A. RB Spray Block (CV-2401)

B. DH Test & Recirc Isol (DH-10)

C. RB Spray to Test & Recirc Header (BS-3)

D. P-35A Disch Isol to Recirc & Test Header (BS-2A)

Answer:

B. DH Test & Recirc Isol (DH-10)

Notes:

"B" is correct per 1203.012J and 1104.005, Supplement 3. The RB Spray Block valve (CV-2401) is closed for the quarterly test so the Test & Recirc Header is used which directs flow back to the BWST. The BS-3 and BS-2A valves are opened for this test but flow is maintained with DH-10 (or DH-9 which bypasses DH-10) for this test so flow must be throttle with DH-10.

"A" is incorrect but plausible since the RB Spray Block would be used to throttle flow during an ES actuation.

"C" is incorrect but plausible since this valve is opened for the quarterly test but is no used to throttle flow.

"D" is incorrect but plausible since this valve is opened for the quarterly test but is no used to throttle flow.

This question matches the K/A since it involves the Containment Spray system and requires the applicant to recall how to throttle flow using CSS controls during a surveillance.

References:

1104.005, Reactor Building Spray System Operation
1203.012J, Annunciator K11 Corrective Action

History:

New for 2018 exam

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

QID: 0158 **Rev:** 2 **Rev Date:** 1/10/18 **Source:** Bank **Originator:** JCork

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

Section: 3.5 **Type:** Containment Integrity

System Number: 103 **System Title:** Containment System

Description: Knowledge of the effect that a loss or malfunction of the Containment System will have on the following: Loss of containment integrity under normal operations.

K/A Number: K3.02 **CFR Reference:** 41.7 / 45.6

Tier:	2	**RO Imp:**	3.8	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	4.2	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 55 **SRO:**

Given:

* Unit 1 at 100% power
* Outside door of personnel air lock just had a seal gasket replaced
* All work complete, no more entries required

What is the MAXIMUM time allowed to perform an LLRT on the personnel air lock before the entry conditions are met for Loss of Reactor Building Integrity (1203.005)?

A. 24 hours

B. 7 days

C. 14 days

D. 31 days

Answer:

B. 7 days

Notes:

"B" is the correct answer per 1203.005 entry conditions.

"A" is incorrect, but plausible as this answer would apply to an inoperable valve in a penetration flow path that has only one reactor building isolation valve. This is also the completion time for required action A.2 of 3.6.2.

"C" is incorrect. This is twice the correct time and makes the answer choices logical: 1 day, 1 week, 2 weeks, 1 month.

"D" is incorrect. This answer would be allowed if frequent entries through the air lock were required. This is also the completion time for required action B.3 of 3.6.2. This time allowance is not applicable for a one time repair to the seal gasket.

This question matches the K/A since the applicant must know what constitutes a loss of containment integrity during normal operations.

References:

1203.005, Loss of Reactor Building Integrity

History:

Developed for 1998 RO Re-exam
Used in 1999 exam.
Used in 2001 RO/SRO Exam. Was generic KA 2.4.11

Modified for 2007 RO Exam.
Selected for 2018 exam
Rev. 1, 9/25/17, editorial changes, revised stem to ensure "A" is not also a correct answer. Changed D to 29 days
Rev. 2, deleted 48 hours ago and revised second given condition accordingly, all answers changed per NRC resolution.

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

QID: 1179 **Rev:** 0 **Rev Date:** 8/1/17 **Source:** New **Originator:** Cork

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

Section: 3.1 **Type:** Reactivity Control

System Number: 001 **System Title:** Control Rod Drive

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CRDS controls including: Reactor power.

K/A Number: A1.06 **CFR Reference:** 41.5 / 45.5

Tier:	2	**RO Imp:**	4.1	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	4.4	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 56 **SRO:**

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

Given:
* Unit 1 power escalation is progress
* Power was 45% from 0100 Sunday until 1300 Thursday to repair MFW Pump
* Power currently 56%

NOW
* ATC pulls rods to continue power escalation at 1556
* Rx power 60% at 1600

By procedure, what is the earliest time reactor power can be raised to 90%?

A. 1700

B. 1710

C. 1800

D. 1810

Answer:

C. 1800

Notes:

"C" is correct. Per 1102.004, Power Operation, Attachment L - Reactor Maneuvering Recommendations, if power has been below 50% for more than 96 hours, then Table L1 provides power escalation limits. Between 60 to 90% power the escalation rate is ≤15%/hr. Att. L, step 1.2 provides guidance for step changes in power, dependent on the escalation rate. Since the escalation rate is ≤15%/hr, then a power change of >3.75% in ≤5 minutes requires a 10 minute hold, however the power change occurred before 60% where the power change limit is >5% in ≤5 minutes so no hold is required. So power can be raised 15%/hr, therefore 90% can be achieved in two hours, or 1800.

"A" is incorrect but plausible since a power change of 30%/hr is acceptable if the reactor had been less than 50% for less than 96 hours but the reactor was less than 50% for 109 hours.

"B" is incorrect but plausible since this time also reflects a 10 minute hold which an applicant could deduce was required and the power change of 30%/hr is incorrect per "A" explanation.

"D" is incorrect but plausible since this time reflects the correct 15%/hr change rate but it includes a 10 minute hold which is not required.

This question matches the K/A since the question conditions require the applicant to use procedural guidance for control rod operation to prevent exceeding design maneuvering limits.

References:

1102.004, Power Operation

Include 1102.004 Attachment L in handout.

History:

New for 2018 exam

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

QID: 1180 **Rev:** 0 **Rev Date:** 8/1/17 **Source:** Modified **Originator:** Cork

TUOI: A1-LP-RO-MU **Objective:** 9 **Point Value:** 1

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

System Number: 011 **System Title:** Pressurizer Level Control System

Description: Knowledge of the operational implications of the following concepts as they apply to the PZR LCS: Indicated charging flow: seal flow plus actual charging flow.

K/A Number: K5.06 **CFR Reference:** 41.5 / 41.7

Tier: 2 **RO Imp:** 2.9 **RO Select:** Yes **Difficulty:** 3

Group: 2 **SRO Imp:** 3.2 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 57 **SRO:**

Given:

* 100% power
* Total RCS leakage 0.5 gpm
* Seal injection flow to each RCP 9 gpm
* Controlled bleedoff flow from each RCP 1.5 gpm
* Letdown flow at maximum for one Makeup Filter (F-3s)
* Pressurizer level 220"

Approximately how much flow is being added to the RCS via the makeup line?

A. 50-55 gpm

B 56-61 gpm

C. 90-95 gpm

D. 110-115 gpm

Answer:

A. 50-55 gpm

Notes:

"A" is correct. Pressurizer level is at normal setpoint of 220". The limit for maximum flow through one makeup filter is 80 gpm. Each RCP has bleedoff flow of 1.5 gpm so total will be 6 gpm. Seal Injection flow to each RCP is 9 gpm so total seal injection flow is 36 gpm. Input flow must equal outgoing flow so [80+6+0.5]=36+X, X=50.5. Therefore "A" is the only correct answer and a band is provided to allow for minor setpoint recall errors

"B" is incorrect but plausible, if applicant recalls the max flow of a single letdown cooler (87.5 gpm) instead of a makeup filter, then the calculation will result in a value of 58 gpm.

"C" is incorrect but plausible, if applicant uses the max flow of a single letdown demineralizer (123 gpm) instead of a makeup filter, then the calculation will result in a value of 93.5 gpm.

"D" is incorrect but plausible, if applicant uses the max flow of the makeup pre-filter (140 gpm) instead of a makeup filter, then the calculation will result in a value of 110.5 gpm.

References:

1104.002, Makeup & Purification System Operation

History:

Modified QID 319 for 2018 exam, modified question by changing Letdown flow from maximum for one DI (123 gpm) to maximum for one Letdown filter (80 gpm). This makes the correct answer to be "A" 50-55 gpm, formerly was 90-99 gpm. All answers revised.

QID: 0319 **Rev:** 0 **Rev Date:** 9-5-99 **Source:** Direct **Originator:** J. Cork

TUOI: ANO-1-LP-RO-MU **Objective:** 9 **Point Value:** 1

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

System Number: 011 **System Title:** Pressurizer Level Control System

Description: Ability to manually operate and/or monitor in the control room: Charging pump and flow controls.

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

Tier:	2	**RO Imp:**	3.5	**RO Select:**	No	**Difficulty:**	5
Group:	2	**SRO Imp:**	3.2	**SRO Select:**	No	**Taxonomy:**	Ap

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

Given:

- 100% power
- Total RCS leakage is .5 gpm
- Seal injection flow to each RCP is 9 gpm
- Controlled bleedoff flow from each RCP is 1.5 gpm
- Letdown flow is maximum for one demineralizer
- Pressurizer level is 220"

Q#57 PARENT

Approximately how much flow is being added to the RCS via the makeup line?

a. 70-79 gpm

b. 80-89 gpm

c. 90-99 gpm

d. 100-109 gpm

Answer:

c. 90-99 gpm

Notes:

Pressurizer level is at normal setpoint. Setpoint for maximum flow through one demineralizer is 123 gpm.
[123+6+.5]=36+X,
X=93.5. Therefore "c" is the only correct answer and a band is provided to allow for setpoint recall errors.

References:

1104.002 Rev 051-02-0

History:

Used in 1999 exam.
Modified from ExamBank, QID# 982.

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

QID: 1218 **Rev:** 1 **Rev Date:** 1/10/18 **Source:** Mod **Originator:** Cork

TUOI: A1LP-RO-NNI **Objective:** 19 **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: Relationship between meter readings and actual parameter value.

K/A Number: A3.02 **CFR Reference:** 41.7 / 45.5

Tier:	2	**RO Imp:**	2.9	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**		**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 58 **SRO:**

Given:

- Plant is at 100% power.
- PZR level transmitter LT-1001 selected via HS-1002 on C04.
- PZR temperature element TE-1001A selected via HS-1000 on C04.

The PZR temperature indicator, TI-1000, on C04 rises quickly to 700°F (top of scale).

Without operator action, what will be the effect on the PZR Level Control System?

A. PZR Level Control Valve, CV-1235, will open more to establish a higher actual steady-state PZR level.

B. PZR Level Control Valve, CV-1235, will fully close causing actual PZR level to continuously lower.

C. PZR Level Control Valve, CV-1235, will close more to establish a lower actual steady-state PZR level.

D. PZR Level Control Valve, CV-1235, will fully open to continuously raise actual PZR level.

Answer:

C. PZR Level Control Valve, CV-1235, will close more to establish a lower actual steady-state PZR level.

Notes:

"C" is correct. The key to this question is knowing the effect of temperature compensation on the PZR level indication. The temperature compensation failing high will cause indicated (and controlling) PZR lev el to read higher than actual, this will cause level to be above setpoint for the PZR level control system which will cause CV-1235 to go closed which will establish an actual lower steady-state level when it brings it back to setpoint.

"A" is incorrect but plausible (this was the correct answer in the original question) if the applicant chooses the wrong direction for failure of temperature compensation.

"B" is incorrect, but plausible if the applicant believes the instrument failing high will cause the makeup valve to go fail closed.

"D" is incorrect but plausible The loss of temperature compensation does not produce an indication that is similar to a high off scale indication.

This question matches the K/A since it requires the applicant to understand the difference between indicated PZR level and actual PZR level.

References:

STM 1-69, Non-Nuclear Instrumentation System

History:

Modified QID 169 for 2018 exam. Question was modified by failing temperature indicator high (vs. low on original), this will cause level to indicate higher than actual and make "C" the correct answer (vs. "A"). Added "actual" to all answers for a closer tie to the K/A.
Rev. 1, revised all aswers to be more specific regarding valve travel per NRC resolution.

QID: 0169 **Rev:** 1 **Rev Date:** 11/19/98 **Source:** Repeat **Originator:** J. Cork

TUOI: A1LP-RO-NNI **Objective:** 19 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 016 **System Title:** Non-Nuclear Instrumention System (NNIS)

Description: Ability to monitor automatic operation of the NNIS, including: Relationship between meter readings and actual parameter value.

K/A Number: A3.02 **CFR Reference:** 41.7 / 45.5

Tier: 2 **RO Imp:** 2.9 **RO Select:** No **Difficulty:** 4

Group: 2 **SRO Imp:** 2.9 **SRO Select:** No **Taxonomy:** An

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

Given:

Q#58 PARENT

- Plant is at 100% power.
- PZR level transmitter LT-1001 selected via HS-1002 on C04.
- PZR temperature element TE-1001A selected via HS-1000 on C04.

The PZR temperature indicator, TI-1000, on C04 drops suddenly to 50°F (bottom of scale).

Without operator action, what will be the effect on the PZR Level Control System?

A. PZR Level Control Valve, CV-1235, will open to establish a higher steady-state PZR level.

B. PZR Level Control Valve, CV-1235, will go full closed causing PZR level to continuously lower.

C. PZR Level Control Valve, CV-1235, will close to establish a lower steady-state PZR level.

D. PZR Level Control Valve, CV-1235, will go full open to continuously raise PZR level.

Answer:

A. PZR Level Control Valve, CV-1235, will open to establish a higher steady-state PZR level.

Notes:

[A] is correct. A loss of temperature compensation will result which will appear as a low PZR level. This is the same reason which makes [B] & [C] incorrect.
(D) is incorrect. The loss of temperature compensation does not produce an indication that is similar to a high off scale indication.
Made minor adjustments to the distractors based on feedback from NRC.

References:

STM 1-69, Non-Nuclear Instrumentation System

History:

Developed for 98 exam.
Used in 2001 Exam.
Selected for use in 2002 SRO exam. KA 011 K4.03
Selected for 2014 Exam

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

QID: 0210 **Rev:** 1 **Rev Date:** 9/18/17 **Source:** Bank **Originator:** Cork

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

Section: 3.5 **Type:** Containment Integrity

System Number: 028 **System Title:** Hydrogen Recombiner and Purge Control System

Description: Knowledge of the effect that a loss or malfunction of the HRPS will have on the following: Hydrogen concentration in containment.

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

Tier:	2	**RO Imp:**	3.3	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**		**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 59 **SRO:**

Given:
* LOCA occurred eight hours ago
* Hydrogen Recombiner M-55A placed in service seven hours ago
* Hydrogen Recombiner M-55B in standby

Which of the following would require placing Hydrogen Recombiner M-55B in service?

A. M-55A average thermocouple temperature reaches 1225 °F

B. H2 concentration lowered but is stable at 2%

C. M-55A power indication reaches 60 KW and steady

D. Hydrogen concentration exceeds 3% and rising

Answer:

D. Hydrogen concentration exceeds 3% and rising

Notes:

"D" is correct per 1104.031. When H2 concentration is greater than 3% and rising, then this is indication the in-service recombiner has insufficient capacity, or is malfunctioning, and the spare recombiner must be placed in service.

"A" is incorrect but plausible. The temperature is very high but this is the operating temperature of the recombiner. A temperature of 1450 °F would be cause to place the standby in-service.

"B" is incorrect but plausible. During post-LOCA operation, H2 concentration should lower but it may stabilize a times.

"C" is incorrect but plausible 60 KW is a high power setting but is acceptable. The recombiner should be operated at less than 75 KW.

This question matches the K/A since it involves the Hydrogen Recombiners in a post-LOCA situation and requires the applicant to know how to detect a malfunction of a H2 Recombiner.

References:

1104.031, Containment Hydrogen Control

History:

Developed for use in 98 RO Re-exam
Selected for 2002 RO/SRO exam.
Selected for 2018 exam
Rev. 1, editorial changes

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

QID: 1219 **Rev:** 0 **Rev Date:** 9/18/17 **Source:** New **Originator:** Cork

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

Section: 3.2 **Type:** Inventory Control

System Number: 002 **System Title:** Reactor Coolant System

Description: Knowledge of RCS design feature(s) and/or interlock(s) which provide for the following: Overpressure protection.

K/A Number: K4.10 **CFR Reference:** 41.7

Tier: 2 **RO Imp:** 4.2 **RO Select:** Yes **Difficulty:** 2

Group: 2 **SRO Imp:** **SRO Select:** No **Taxonomy:** F

Question: **RO:** 60 **SRO:**

What is the opening setpoint for the Pressurizer Code Safety Valves (PSV-1001 & 1002)?

A. 2355 psig

B. 2450 psig

C. 2500 psig

D. 2750 psig

Answer:

C. 2500 psig

Notes:

"C" is correct, the PZR code safeties open at 2500 psig.

"A" is incorrect but plausible, this is the RPS high RCS pressure trip setpoint.

"B" is incorrect but plausible, this is the ERV lift setpoint.

"D" is incorrect but plausible, this is the Tech Spec 2.1.2 safety limit. The Safeties are attempting to keep pressure less than this value.

This question matches the K/A as it requires knowledge of an RCS overpressure protection design feature: PZR code safeties.

References:

STM 1-03, Reactor Coolant System

History:

New for 2018 exam

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

QID: 1181 **Rev:** 1 **Rev Date:** 1/10/18 **Source:** New **Originator:** Cork

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

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

System Number: 035 **System Title:** Steam Generator System

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the SG; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Steam flow/feed mismatch.

K/A Number: A2.04 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.5

Tier: 2 **RO Imp:** 3.6 **RO Select:** Yes **Difficulty:** 3

Group: 2 **SRO Imp:** 3.8 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 61 **SRO:**

Given:
* Power escalation in progress
* 2nd MFW pump placed in service and in Auto

NOW
* Power 50%
* Annunciator REACTOR IS FEEDWATER LIMITED (K07-C1) alarms
* MFW pump P-1B speed 3000 rpm

(1) Which of the following is the cause of the alarm and (2) what action is procedurally required for the above conditions?

A. (1) Reactor power is 2% greater than feedwater flow
(2) Trip the reactor and perform Reactor Trip (1202.001)

B. (1) Reactor power is 5% greater than feedwater flow
(2) Trip MFW pump P-1B and open Discharge Crosstie (CV-2827)

C. (1) Reactor power is 2% greater than feedwater flow
(2) Trip MFW pump P-1B and open Discharge Crosstie (CV-2827)

D. (1) Reactor power is 5% greater than feedwater flow
(2) Trip the reactor and perform Reactor Trip (1202.001)

Answer:

B. (1) Reactor power is 5% greater than feedwater flow
(2) Trip MFW pump P-1B and open Discharge Crosstie (CV-2827)

Notes:

"B" is correct. MFW pump P-1B speed at 3000 rpm means the pump has lowered to minimum speed. Since P-1B has not tripped this will mean FW flow will be lower than required, the pump must be manually tripped and the discharge crosstie valve opened to supply both SGs. The Reactor is Feedwater Limited annunciator means that FW is lower than demand by more than 5%. FYI, this alarm is known as an ICS cross limit.

"A" is incorrect but plausible since this is the correct cause but an incorrect setpoint. The 2% value orginates from the NI deviation from thermal power which requires an NI adjustment. The reactor would only be tripped if the discharge crosstie valve did not go open following tripping of the B MFW pump.

"C" is incorrect but plausible since this is the correct cause but an incorrect setpoint. The 2% value orginates from the NI deviation from thermal power which requires an NI adjustment. This choice has the correct actions to take.

"D" is incorrect but plausible since this is the correct cause of the alarm but contains the incorrect actions.

This question matches the K/A as it requires the applicant to predict the impact of the malfunction of the B MFW pump on the Steam Generator System and to recall the proper actions to take for the resulting steam flow/feed flow mismatch.

Alternatives to the first part of the answers are "ICS is placed in track" and "ICS runs back to 40%".

References:

1203.027, Loss of Steam Generator Feed
1203.012F, Annunciator K07 Corrective Action

History:

New for 2018 exam
Rev. 1, revised 1st part of distractors A and C since they were deemed non-plausible, now are just like B and D but with incorrect setpoint, per NRC resolution.

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

QID: 1182 **Rev:** 1 **Rev Date:** 1/10/18 **Source:** New **Originator:** Cork

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/Turbine Bypass Control

Description: Knowledge of the effect of a loss or malfunction will have on the SDS: Controllers and positioners, including ICS, S/G, CRDS.

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

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

Group: 2 **SRO Imp:** 2.9 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 62 **SRO:**

Given:
* Unit at 10% power
* Main Generator just synchronized to grid and block loaded
* TBVs in AUTO

NOW
* Bias for TBV controller applied early

What will be (1) the INITIAL effect of this malfunction on TBVs/GVs, and (2) the initial effect on RCS Tave?

A. (1) TBVs close
(2) rises

B. (1) GVs open
(2) lowers

C. (1) GVs close
(2) rises

D. (1) TBVs open
(2) lowers

Answer:

A. (1) TBVs close
(2) rises

Notes:

"A" is correct. Applying the TBV 50 psig bias early means this will close all TBVs, steam header pressure will rise, GVs will open, RCS temperature will initially rise and then fall back to normal.

"B" is incorrect, the ealy application of the bias will cause the GV's to open but RCS temp initially rises due to TBV closure.

"C" is incorrect, RCS temp does rise initially but the GVs should open, not close.

"D" is incorrect, the TBVs close, not open, and RCS temp rises, not lowers.

This question matches the KA since it involves a failure of a steam dump system controller and requires applicant to know the resultant effects.

References:

1102.002, Plant Startup
CR-ANO-1-2004-01468, TBV Bias applied early due to relay failure

History:

New for 2018 exam
Rev. 1, deleted parenthetical "now", changed temperature to Tave in stem, per NRC resolution.

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

QID: 1237 **Rev:** 0 **Rev Date:** 10/9/17 **Source:** Modified **Originator:** Cork

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

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

System Number: 045 **System Title:** Main Turbine Generator

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: 2 **RO Imp:** 3.9 **RO Select:** Yes **Difficulty:** 3

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

Question: **RO:** 63 **SRO:**

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

Given:

* Thunderstorms in the area

* Due to plant issues, Unit 1 is operating at 820 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 must NOT change and final power factor must be same value as beginning power factor

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

A. 600 MVARs

B. 320 MVARs

C. 280 MVARs

D. 160 MVARs

Answer:

B. 320 MVARs

Notes:

"B" is correct, operating at a PF of .98 under-excited with a load of 820 MW means the Main Generator reactive load must be -160 MVARs. The most reactive load which could be raised is where the 820 MW line intersets the .98 PF line in the over-excited half of Att. N. This would be -160 to +160 for a total of 320 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 820 MW line at .98 PF and goes until the 820 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.

"C" is incorrect but is plausible if the applicant studied a copy of an exam, this was previously the correct answer (see QID 1126).

"D" is incorrect but is plausible if the candidate mistakenly starts at the 820 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.

References:

1102.004, Power Operation

History:

Modified QID 1126 for 2018 exam
Modified by changing "720 Mwe" to "820". Modified B to 320 to make it the correct answer. Modified A to 600 and D to 160 so they are plausible with these changes. Left C "as-is" in case someone is studying previous exam questions.

QID: 1126 **Rev:** 2 **Rev Date:** 5/18/17 **Source:** Repeat **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:** No **Difficulty:** 3

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

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

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

Q#63 PARENT

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.

Q#63 PARENT

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

QID: 0261 **Rev:** 2 **Rev Date:** 1/10/18 **Source:** Bank **Originator:** Slusher

TUOI: ANO-1-LP-RO-COND **Objective:** 15 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal From Reactor Core

System Number: 056 **System Title:** Condensate System

Description: Knowledge of the physical connections and/or cause-effect relationships between the Condensate System and the following systems: (CFR:)
MFW

K/A Number: K1.03 **CFR Reference:** 41.2 to 41.9 / 45.7 to 45.8

Tier: 2 **RO Imp:** 2.6 **RO Select:** Yes **Difficulty:** 3.5

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

Question: **RO:** 64 **SRO:**

Given:

* Unit 1 at 30 % power
* Main Feedwater Pump P-1A in service
* Main Feedwater Pump P-1B shutdown
* Condensate pumps P-2A and P-2C in service

Select the answer below which explains the response of P-2B condensate pump, if P-2C condensate pump trips.

A. Condensate pump P-2C low discharge pressure will auto-start condensate pump P-2B

B. Condensate pump P-2C breaker opening will auto-start condensate pump P-2B

C. Condensate pump P-2B will not start since plant is less than 40% power

D. Condensate pump P-2B will not start since Main Feedwater Pump P-1B unlatched

Answer:

D. Condensate pump P-2B will not start since Main Feedwater Pump P-1B unlatched

Notes:

"D" is correct. Condensate pumps will only auto start if both Main Feedwater pumps are latched.

"A" is incorrect but plausible since this will auto-start P-2B pump, but only if both MFW pumps are latched.

"B" is incorrect but plausible since this will auto-start P-2B pump, but only if both MFW pumps are latched.

"C" is incorrect because 40% power is not an interlocking function. It is plausible since there is an ICS runback to 40% if 2 of 3 condensate pumps trip when > 40% power, but it does not auto-start a condensate pump.

This question matches the K/A since the applicant must know the cause-effect relationship between the condensate pump auto-start and the latching of both MFW pumps.

References:

1203.012E, Annunciator K06 Corrective Action

History:

Developed for 1999 exam.
Selected for 2002 RO exam.
Selected for 2018 exam
Rev. 1, 9/19/17, editorial changes
Rev. 2, changed "remain off" in C and D to "not start", per NRC resolution.

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

QID: 1220 **Rev:** 1 **Rev Date:** 1/11/18 **Source:** New **Originator:** Cork

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

Section: 3.9 **Type:** Radioactivity Release

System Number: 071 **System Title:** Waste Gas Disposal

Description: Ability to manually operate and/or monitor in the control room: WGDS status alarms.

K/A Number: A4.14 **CFR Reference:** 41.7 / 45.5 to 45.8

Tier:	2	**RO Imp:**	2.9	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**		**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 65 **SRO:**

Given:
* Unit 1 at 100% power
* Release of Waste Gas Decay Tank T-18C in progress

NOW
* Annunciator PROC MONITOR RADIATION HI (K10-B2) alarms
* Annunciator RADWASTE GAS PANEL TROUBLE (K09-D5)
* CBOT reports PROC MONITOR RE-4830 (Gaseous Radwaste) high alarm

In accordance with Waste Gas Discharge Line Radiation High (1203.006) which of the following actions will the CRS direct operators to perform?

A. Verify in-service Waste Gas Compressor (C-9A/B) tripped.

B. Verify ABVH Vent Header valve (CV-4806) closes.

C. Reset Gaseous Radwaste Monitor (RE-4830) and continue release.

D. Isolate WGDT T-18C and resubmit release permit request.

Answer:

D. Isolate WGDT T-18C and resubmit release permit request.

Notes:

"D" is correct per 1203.006 step 5. The Waste Gas Decay Tank (WGDT) being released should be isolated and the release permit paperwork re-submitted.

"A" is incorrect since the Waste Gas Compressors do not trip on high radiation but it is plausible since a tripping of the compressor would help stop a release and it is one of the causes of the K09-B5 annunciator.

"B" is incorrect because the ABVH Vent Header (CV-4806) opens to divert to the Waste Gas Surge Tank, but plausible since the Station Vent Discharge Valve (CV-4830) closes on a high rad signal.

"C" is incorrect but plausible since this action would be taken due to a spike and a venting operation were taking place but not a release.

This question matches the K/A since it involves the Waste Gas Disposal system and requires applicant to be able to monitor the alarms as well as know the corrective actions to the alarms.

References:

1203.006, Waste Gas Discharge Line Radiation High

History:

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

QID: 1184 **Rev:** 0 **Rev Date:** 8/14/17 **Source:** Modified **Originator:** Cork

TUOI: ASLP-RO-OPSPR **Objective:** 4 **Point Value:** 1

Section: 2.0 **Type:** Generic Knowledges and Abilities

System Number: 2.1 **System Title:** Conduct of Operations

Description: Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc.

K/A Number: 2.1.4 **CFR Reference:** 41.10 / 43.2

Tier:	3	**RO Imp:**	3.3	**RO Select:**	Yes	**Difficulty:**	2
Group:		**SRO Imp:**	3.8	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 66 **SRO:**

For the purpose of maintaining an NRC operator's license, which of the following should be reported to the NRC within 30 days?

A. A court order of "no contact" filed following spousal separation.

B. A conviction of a felony.

C. A broken leg.

D. A filing for bankruptcy.

Answer:

B. A conviction of a felony.

Notes:

"B" is correct per 1063.008 and 10CFR55.

"A" is incorrect but plausible due to the implications of domestic violence.

"C" is incorrect but plausible due change in medical condition and the condition is temporary

"D" is incorrect but plausible de to the implications of a legal proceeding.

While the original question stem is the same, this question is modified from the original by changing all of the answer choices. Previously the correct answer was a medical condition.

References:

1063.008, Operations Training Sequence

History:

Modified QID 838 for 2018 exam

QID: 0838 **Rev:** 0 **Rev Date:** 5/24/11 **Source:** Repeat **Originator:** J. Cork

TUOI: ASLP-RO-OPSPR **Objective:** 4 **Point Value:** 1

Section: 2.0 **Type:** Generic Knowledge and Abilities

System Number: 2.1 **System Title:** Conduct of Operations

Description: Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc.

K/A Number: 2.1.4 **CFR Reference:** 41.10 / 43.2

Tier: 3 **RO Imp:** 3.3 **RO Select:** No **Difficulty:** 2

Group: **SRO Imp:** 3.8 **SRO Select:** No **Taxonomy:** K

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

For the purpose of maintaining an NRC operator's license, which of the following should be reported to the NRC?

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A. A change in marital status.

B. A traffic citation for speeding.

C. A new diagnosis for high blood pressure.

D. An audit by the IRS of previous year's tax return.

Answer:

C. A new diagnosis for high blood pressure.

Notes:

Only "C" is requred to be reported per EN-NS-112 and 1063.008.
The others are plausible situations which can occur in life that are not required to be reported as part of an operator's license.

This question matches the K/A since it relates to an individual licensed operator responsibility to maintain an active license.

References:

1063.008, Operations Training Sequence

History:

New for 2011 RO Exam.
Selected for 2016 exam.

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

QID: 1185 **Rev:** 0 **Rev Date:** 8/14/17 **Source:** Modified **Originator:** Cork

TUOI: A1LP-AO-VALVE **Objective:** 5 **Point Value:** 1

Section: 2.0 **Type:** Generic Knowledges and Abilities

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:**	F

Question: **RO:** 67 **SRO:**

Given:
* Unit 1 heating up following refueling
* Valve lineup in progress but an MOV inside Aux Building is leaking by
* MOV to be manually seated
* This is a 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 manually 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 wrench.

D. Danger tag MOV breaker open AND manually tighten by hand without using a torque wrench.

Answer:

B. Danger tag MOV breaker open AND manually tighten using a torque wrench.

Notes:

"B" is correct since an MOV in the Aux Building would be danger tagged if manually operated, and a Q MOV requires tightening using a torque wrench.

"C" is incorrect but plausible as this was the previous correct answer. 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. Non-Q MOV do not have torque limits and not use a TAD (torque amplifying device) but this is a Q MOV in the Aux Building.

"A" is incorrect but plausible as this distractor has the correct method of tightening the valve but incorrect method of de-energizing the MOV.

"D" is incorrect but plausible since this has the correct method of de-energizing the MOV but the incorrect method of tightening the valve.

Modified QID 1142 by changing location of MOV from Reactor Building to Aux Building (this requires a danger tag) and changing the MOV from a non-Q to a Q MOV, thus requiring a torque wrench. This changes the correct answer from C to B.

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:

Modified QID 1142 for 2018 exam

QID: 1142 **Rev:** 1 **Rev Date:** 5/21/17 **Source:** Repeat **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:** No **Difficulty:** 2

Group: **SRO Imp:** 4.0 **SRO Select:** No **Taxonomy:** H

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

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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.

Q#67 PARENT

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

QID: 0991 **Rev:** 1 **Rev Date:** 1/11/18 **Source:** Bank **Originator:** NRC

TUOI: A1QC-RO-QUAL **Objective:** 3.1.23 **Point Value:** 1

Section: 2.0 **Type:** Generic K/As

System Number: 2.1 **System Title:** Conduct of Operations

Description: Knowledge of the station's requirements for verbal communications when implementing procedures.

K/A Number: 2.1.38 **CFR Reference:** 41.10 / 45.13

Tier: 3 **RO Imp:** 3.7 **RO Select:** Yes **Difficulty:** 2

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

Question: **RO:** 68 **SRO:**

Per Conduct of Operations (1015.001) who is the operations staff required to contact when any annunciator is removed from service due to a malfunction?

A. Operations Work Liaison

B. Design Engineering

C. Engineering Programs

D. System Engineering

Answer:

D. System Engineering

Notes:

"D" is correct per Section 10.0 of 1015.001, System Engineering (SYE) shall be contacted about any annunciator modified or removed from service due to a malfunctionin.

"A" is incorrect because the Operations Work Liaison (OWL) is not required to be contacted per Section 10.0 of this procedure. The OWL is to be contacted for arranging unplanned non-emergency maintenance activities per Section 8.0 of the procedure and is thus plausible.

"B" is incorrect because the group isn’t referred to in Section 10.0. They are to be contacted to coordinate Category E valve position alignment checks per Attachment D.2, Section 2.0.

"C" is incorrect because the group isn’t referred to in Section 10.0. They are supposed to be contacted when containment isolation valves are inoperable to perform reactor building leakage assessments per Section 15.0 of the procedure.

References:

1015.001, Conduct of Operations

History:

New for 2013 Exam
Rev. 1, editorial changes only. Replacement question per NRC resolution
Selected for 2018 exam

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

QID: 1186 **Rev:** 0 **Rev Date:** 5/14/17 **Source:** New **Originator:** Cork

TUOI: ASLP-OPS-CCT **Objective:** 5 **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:** 41.10 / 43.3 / 45.13

Tier: 3 **RO Imp:** 3.9 **RO Select:** Yes **Difficulty:** 2

Group: **SRO Imp:** 4.3 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 69 **SRO:**

Given:
* Refueling outage in progress
* Ops desires to re-position a component
  * Component position NOT governed by procedure steps
  * Component positioning NOT part of maintenance
  * Component NOT danger tagged
  * Component NOT located inside tagout boundary

How is the positioning of the component to be managed in accordance with procedures?

A. Test and Maintenance tag process

B. Danger tag process

C. Configuration Control Record process

D. Caution tag process

Answer:

C. Configuration Control Record process

Notes:

"C" is correct per 1015.049, Configuration Control Process, Att. B flowchart.

"A" is incorrect but plausible since Test and Maintenance tags can be used to manipulate a component but per 1015.049, a Configuration Control tag and T&M tag can not be used on the same component.

"B" is incorrect but plausible since Danger tags can be issued by Ops but not to re-position a component.

"D" is incorrect but plausible since Caution tags can be used to make off-normal conditions more visible but this is not allowed by 1015.049.

This question matches the K/A since it requires direct knowledg of the ANO process for controlling equipment configuration.

References:

1015.049, Configuration Control Program

History:

New for 2018 exam

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

QID: 0118 **Rev:** 2 **Rev Date:** 1/10/18 **Source:** Bank **Originator:** Cork

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

Section: 2.0 **Type:** Generic K/As

System Number: 2.2 **System Title:** Equipment Control

Description: Knowledge of surveillance procedures.

K/A Number: 2.2.12 **CFR Reference:** 41.10 / 45.13

Tier: 3 **RO Imp:** 3.0 **RO Select:** Yes **Difficulty:** 2

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

Question: **RO:** 70 **SRO:**

Which of the following describes a Channel Check as defined by Tech Specs?

A. The test of logic elements in a protection channel to verify associated trip actions.

B. The adjustment of the channel output such that it responds accurately to known values.

C. The qualitative assessment, by observation, of channel behavior during operation.

D. The injection of a simulated or actual signal into the channel to verify channel operability.

Answer:

C. The qualitative assessment, by observation, of channel behavior during operation.

Notes:

"C" is the correct definition of a Channel Check.

"A" is incorrect but plausible as this defines a Trip Test, an obsolete TS definition.

"B" is incorrect but plausible as this is an Channel Calibration.

"D" is incorrect but plausible as this is a Channel Functional Test.

This question matches the K/A since a channel check is a basic surveillance method.

References:

Technical Specifications, section 1.1

History:

Selected for 2005 RO re-exam.
Rev. 1, 8/15/17
Added first sentence to provide a more direct link to K/A.
Minor revision of notes to bring them up to date.
Selected for 2018 exam
Rev. 2, deleted 1st sentence as it was unnecessary, per NRC resolution.

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

QID: 1239 **Rev:** 1 **Rev Date:** 1/10/18 **Source:** New **Originator:** Cork

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

Section: 2.0 **Type:** Generic Knowledges and Abilities

System Number: 2.2 **System Title:** Equipment Control

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:** **SRO Select:** No **Taxonomy:** F

Question: **RO:** 71 **SRO:**

Which one of the following is required by Unit 1 Technical Specifications in order to consider the reactor in Mode 3?

Average
Reactor Coolant
Temperature (°F)

A. ≤ 200

B. 280 > Tavg > 200

C. < 280

D. ≥ 280

Answer:

D. ≥ 280

Notes:

"D" is correct, Keff < 0.99 and RCS Tavg is ≥ 280°F for Mode 3.

"A" is incorrect, this choice is plausible but has the temperature range for Mode 5.

"B" is incorrect, this choice is plausible but is the Tavg range associated with Mode 4.

"C" is incorrect, this choice is plausible since this has the correct temperature but the wrong sign.

This matches the K/A since it requires knowledge of Tech Spec mode conditions.

References:

Technical Specifications

History:

Different version of QID 458
New for 2018 exam
Rev. 1, 1/10/18, deleted Keff values, changed tavg in A and D, per NRC resolution.

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

QID: 1144 **Rev:** 1 **Rev Date:** 5/21/17 **Source:** Repeat **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:** 72 **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.
Selected for 2018 exam

QID: 0995 **Rev:** 1 **Rev Date:** 10/6/17 **Source:** Bank **Originator:** NRC

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

Section: 2.0 **Type:** Generic Knowledges and Abilities

System Number: 2.3 **System Title:** Radiation Control

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

K/A Number: 2.3.15 **CFR Reference:** 41.12 / 43.4 / 45.9

Tier: 3 **RO Imp:** 2.9 **RO Select:** Yes **Difficulty:** 2

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

Question: **RO:** 73 **SRO:**

At ANO there are several highly sensitive process radiation detectors, such as the Main Steam N-16 Radiation Monitors, that use photomultiplier tubes as part of their radiation detection process.

Process radiation monitors are ____________ type detectors.

A. Scintillation

B. Geiger-Mueller

C. Ion Chamber

D. Proportional

Answer:

A. Scintillation

Notes:

"A" is the correct answer because the N-16 radiation monitors use scintillation detectors and scintillation detectors use photomultipliers as part of the detection process.

"B" is incorrect because while many of the radiation monitors use Geiger-Mueller type detectors, the N-16 monitors do not. Also, Geiger-Mueller detectors do not use photomultipliers.

"C" is incorrect because none of the radiation monitors use an ion chamber type detector. Also, ion Chambers detectors do not use photomultipliers.

"D" is incorrect because none of the radiation monitors use a proportional type detector. Also, Proportional detectors do not use photomultipliers.

References:

STM 1-62, Radiation Monitors

History:

New for 2013 Exam
Selected for 2018 exam
Rev. 1, added "process" before "radiation" in first sentence and stem. Removed "that use photomultiplier tubes" from stem due to reviewer comment.

QID: 1143 **Rev:** 1 **Rev Date:** 5/21/17 **Source:** Repeat **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:** 74 **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.
Selected for 2018 exam

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

QID: 0393 **Rev:** 1 **Rev Date:** 8/16/17 **Source:** Bank **Originator:** R.Soukup

TUOI: ANO-1-LP-RO-AOP **Objective:** 3 **Point Value:** 1

Section: 2 **Type:** Generic K & A's

System Number: 2.4 **System Title:** Emergency Procedures/Plan

Description: Knowledge of operator response to loss of all annunciators.

K/A Number: 2.4.32 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier:	3	**RO Imp:**	3.3	**RO Select:**	Yes	**Difficulty:**	3
Group:		**SRO Imp:**	3.5	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 75 **SRO:**

Both AC and DC "Power Available" lamps have gone out for all Control Room annunciator panels.

Which of the following actions should be taken?

A. Trip the reactor and enter Reactor Trip (1202.001) .

B. Commence power reduction per Rapid Plant Shutdown (1203.045).

C. Commence normal plant shutdown per Power Reduction and Plant Shutdown (1102.016).

D. Notify the Shift Manager to implement Emergency Action Level Classification (1903.010).

Answer:

D. Notify the Shift Manager to implement Emergency Action Level Classification (1903.010),

Notes:

"D" is correct, power should be maintained steady and SM should consult 1903.010.

"A" is incorrect although plausible since this is a common response to other major losses of equipment but annunciators are vital when verifying plant conditions following a Rx trip.

"B" and "D" are incorrect but plausible since a shutdown is often called for in AOPs but steady state power should be maintained while annunciators are inoperable.

This question matches the K/A since it requires knowledge of the AOP actions for a loss of all control room annunciators.

References:

1203.043, Loss of Control Room Annunciators

History:

New question created for 2001 RO/SRO Exam.
Rev. 1, 8/16/17
Editorial changes to notes and question to reflect current formatting and practices.
Selected for 2018 exam

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

QID: 1205 **Rev:** 0 **Rev Date:** 9/7/17 **Source:** New **Originator:** Cork

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

Section: 4.1 **Type:** Generic EPEs

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

Description: Knowledge of EOP mitigation strategies.

K/A Number: 2.4.6 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier:	1	**RO Imp:**		**RO Select:**	No	**Difficulty:**	4
Group:	1	**SRO Imp:**	4.7	**SRO Select:**	Yes	**Taxonomy:**	H

Question: **RO:** **SRO:** 76

Given:
* Unit 1 tripped from 100% power on low RCS pressure
* ESAS Channels 1-6 actuated
* CRS entered ESAS (1202.010)

NOW

* RCS pressure 1200 psig and dropping slowly
* CETs 520°F and dropping slowly
* PZR level 80" and rising
* Both SG pressures 800 psig and steady
* TBVs in AUTO and closed

Which of the following procedures should the CRS use to recover from this event?

A. Transition to Natural Circulation Cooldown (1203.013)

B. Return to Reactor Trip (1202.001)

C. Transition to Loss of Subcooling Margin (1202.002)

D. Transition to Small Break LOCA Cooldown (1203.041)

Answer:

D. Transition to Small Break LOCA Cooldown (1203.041)

Notes:

"D" is correct. The given conditions show that a low RCS pressure condtion caused the event and ESAS channels 1-6 actuated which means RB pressure must have risen above 4 psig to actuate channels 5 and 6. Actuation of 5 and 6 would also mean that RCPs would have been secured since ICW to RCPs isolates on 5 and 6 actuation. Now RCS pressure and CETs show that Subcooling Margin has been restored by HPI flow being greater than break flow but the Small Break LOCA is causing a cooldown since SG pressures are lower than TBV setpoint and the TBVs are closed (no steam demand). CETs dropping slowly is a good sign but RCS pressure dropping slowly means the break was not isolated. Therefore, step 12 of ESAS (1202.010) will have the CRS go to the Contingency Action column and choose a procedure to transition to. The correct procedure is Small Break LOCA Cooldown since the break is causing the cooldown.

"A" is incorrect but plausible since ESAS does have a transition to 1203.013 and the applicant should deduce that RCPs are not running but ESAS (1202.010) only has a transition to this procedure if PZR level is rising without a corresponding rise in RCS temperature or pressure.

"B" is incorrect but plausible since ESAS will return to Reactor Trip in step 12 but only if the cause of the ESAS actuation has been corrected, i.e., the break was isolated by ESAS actuation. This is not the case with the condition of RCS pressure dropping slowly.

"C" is incorrect but plausible since ESAS will transition to Loss of Subcooling Margin if RCS pressure remains greater than 150 psig but only if SCM has not been restored and the pressure/temperature combination shows thatn SCM has been restored.

This question matches the K/A since a small break LOCA scenario is described and it requires detailed knowledge of EOP transition steps.

References:

1202.010, ESAS
1203.041, Small Break LOCA Cooldown

History:

New for 2018 SRO exam

QID: 1189 **Rev:** 2 **Rev Date:** 1/15/18 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-EOP06 **Objective:** 14/15 **Point Value:** 1

Section: 4.1 **Type:** Generic Emergency Plant Evolutions

System Number: 038 **System Title:** Steam Generator Tube Rupture

Description: Ability to determine or interpret the following as they apply to a SGTR: RCP restart criteria.

K/A Number: EA2.17 **CFR Reference:** 43.5 / 45.13

Tier:	1	**RO Imp:**	4.4	**RO Select:**	No	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.4	**SRO Select:**	Yes	**Taxonomy:**	H

Question: **RO:** **SRO:** 77

Given:
* Steam Generator Tube Rupture has occurred
* Offsite power available
* CETs 590 °F
* RCS pressure 1700 psig
* All RCPs tripped
* Full HPI per RT-3 initiated

NOW
* RCS pressure has stabilized at 1650 psig
* CETs 570 °F

(1) What criteria must be met prior to restarting RCPs and (2) what RT will be used?

A. (1) SCM of ≥ 40 °F
(2) RT-11, Start or Bump RCPs

B. (1) SCM of ≥ 40 °F
(2) RT-8, Restore RCP Services

C. (1) PZR temp ≥ CET temp + 10 °F
(2) RT-11, Start or Bump RCPs

D. (1) PZR temp ≥ CET temp + 10 °F
(2) RT-8, Restore RCP Services

Answer:

A. (1) SCM of ≥ 40 °F
(2) RT-11, Start or Bump RCPs

Notes:

"A" is correct. Per step 33 of Tube Rupture one RCP per loop will be re-started using RT-11 if SCM has been restored. RT-11 step 4 states to verify SCM is above minimum adequate (30 °F for 1650 psig) by ≥ 10 °F. RT 11 contains the direction and criteria for re-starting RCPs.

"B" is incorrect but plausible since it contains the correct RCP restart criteria but has the incorrect Repetitive Task. RT-8, Restore RCP Services is used in preparation to restart an RCP after ESAS but ESAS has not occurred in this scenario.

"C" is incorrect but plausible since it contains RCP restart criteria for PZR temperature from RT-11 but this criteria is incorrect since it is the SCM criteria incorrectly applied to the adequate PZR temperature criteria. The correct PZR temperature criteria should be PZR temp ≥ CET + 40 °F for RCS pressures > 1000 psig. "C" is also plausible since it has the correct RT.

"D" is incorrect but plausible since it contains RCP restart criteria for PZR temperature from RT-11 but this criteria is incorrect since it is the SCM criteria incorrectly applied to the adequate PZR temperature criteria. The correct PZR temperature criteria should be PZR temp ≥ CET + 40 °F for RCS pressures > 1000 psig. "D" also has the wrong RT.

This question matches the K/A since it requires the ability to determine applicable RCP restart critieria during a Tube Ruputure.

This question is SRO Only since it requires the SRO applicant to determine the correct selection of procedures, an SRO Only responsibility.

References:

1202.006, Tube Rupture
1202.012, Repetitive Tasks, RT-11, Start or Bump RCPs

History:

New question for 2018 SRO exam
Rev. 1, 9/1/17; Changed conditions by adding RCS pressure and CETs vs. simply stating SCM was lost. Added CET temps under "NOW", also changed "CRS directs RCP restart" to "CRS considering RCP restart".
Rev. 2, Replaced EALs with RT-8 and RT-11. deleted last bullet item to remove cueing since it contains RT-11 and revised stem. All per NRC resolution.

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

QID: 1206 **Rev:** 2 **Rev Date:** 2/5/18 **Source:** Mod **Originator:** Cork

TUOI: A1LP-RO-EOP08 **Objective:** 10 **Point Value:** 1

Section: 4.1 **Type:** Generic EPEs

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

Description: Ability to prioritize and interpret the significance of each annunciator or alarm.

K/A Number: 2.4.45 **CFR Reference:** 41.10 / 43.5 / 45.3 / 45.12

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

Question: **RO:** **SRO:** 78

Given:
* Both units tripped due to degraded offsite power
* SU1 voltage 15.3 KV
* SU2 voltage 60.1 KV
* K01-A2 "EDG1 TRIP" in alarm
* K02-B7 "A4 L.O. RELAY TRIP" in alarm

NOW the CBOT reports these critical parameters:
* CETs 600 °F
* RCS pressure 1850 psig
* All RVLMS indicators are green

Based on the above conditions, which of the following procedure actions are required to be performed?

A. Dispatch operator to perform Att. 1, Blackout Breaker Alignment and UV Relay Defeat, of Blackout (1202.008)

B. Perform rapid cooldown per Blackout (1202.008)

C. Perform RT-4, Initate HPI Cooling

D. Dispatch operator to perform Att. 2, Recovery from Blackout Breaker Alignment and UV Relay Defeat, of Blackout (1202.008)

Answer:

A. Dispatch operator to perform Att. 1, Blackout Breaker Alignment and UV Relay Defeat, of Blackout (1202.008)

Notes:

"A" is correct. The conditions given show that both SU1 and SU2, while energized from offsite power, are degraded. This will cause the undervoltage relays on A3/A4 to pickup, A3/A4 feeder breakers to open, and both EDGs to start. EDG2 will be running but will not tie on to the bus due to the presence of the A4 lockout relay alarm, this combined with the EDG1 trip alarm indicate that no 4160v busses are energized and a Blackout has occurred. Progressing through the Blackout EOP will lead the user to step 47 where an operator is dispatched to perform Att. 1 which will allow energization of buses by defeating undervoltage relays.

"B" is incorrect, but plausible since SCM is inadequate and if head voids were indicated, then a rapid plant cooldown would be required per floating step of 1202.008. RVLMS indicates green which means the sensors are wet and there are no head voids.

"C" is incorrect but plausible since step 8 in RT-4 provides for continuing the RT without HPI pumps but no steps in 1202.008 direct performance of this RT.

"D" is incorrect but plausible since Att. 2 will be performed during a Blackout with degraded voltage indicated or

SU transformers but not until after the buses are energized following performance of Att. 1. 1202.008 step 9 directs performance of Att. 2 but only if Att. 1 has been performed.

This matches the K/A since two very significant alarms indicate that a Blackout has occurred and the applicant should deduce which step is the correct one to perform.

References:

1202.008, Blackout

History:

Modified QID 1026 for 2018 SRO exam by changing the condition RVLMS indicates dry (making the former correct answer "B" incorrect) to RVLMS indicates green and changing "A" from Loss of SCM (1202.002) to Perform Att. 1, Blackout Breaker Alaignment and UV Relay Defeat, which is now the correct answer.
Rev. 1, 11/16/17, changed RVLMS condition to "indicators are green" as this was confusing some validators and causing them to choose an incorrect answer.
Rev. 2, 2/5/18, moved the last 3 parameters down and added "NOW the CBOT reports the following critical parameters" due to suggestion for additional emphasis from SRO validator.

QID: 1026 **Rev:** 0 **Rev Date:** 9/30/14 **Source:** Repeat **Originator:** Cork

TUOI: A1LP-RO-EOP08 **Objective:** 10 **Point Value:** 1

Section: 4.1 **Type:** Generic EPEs

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

Description: Ability to determine or interpret the following as they apply to a Station Blackout: RCS core cooling through natural circulation cooling to S/G cooling.

K/A Number: EA2.02 **CFR Reference:** 43.5 / 45.13

Tier: 1 **RO Imp:** 4.4 **RO Select:** No **Difficulty:** 4

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

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

Q#78 PARENT

Given:
- Both units have tripped due to a loss of offsite power
- SU1 voltage 15.3 KV
- SU2 voltage 60.1 KV
- K01-A2 "EDG1 TRIP" in alarm
- K02-B7 "A4 L.O. RELAY TRIP" in alarm
- CETs 600 °F
- RCS pressure 1850 psig
- RVLMS Level 1 and 2 indicate "Dry"

Based on the above conditions, which of the following procedure actions are required to be performed?

A. Go to 1202.002, Loss of Subcooling Margin

B. Perform rapid cooldown per 1202.008, Blackout

C. Perform RT-4, Initate HPI Cooling

D. Dispatch operator to perform Att. 2, Recovery from Blackout Breaker Alignment and UV Relay Defeat, of 1202.008, Blackout.

Answer:

B. Perform rapid cooldown per 1202.008, Blackout

Notes:

B is correct, with inadequate SCM and head voids indicated, then a rapid plant cooldown is required per floating step of 1202.008.
A is incorrect but plausible since a Loss of Subcooling Margin is indicated but no power exists so 1202.008 should be in use to restore power.
C is incorrect but plausible since step 8 in RT-4 provides for continuing the RT without HPI pumps but no steps in 1202.008 direct performance of this RT.
D is incorrect but plausible since Att. 2 will be performed during a Blackout with degraded voltage indicate on SU transformers but not until after the buses are energized following performance of Att. 1 (which has a similar sounding title).

References:

1202.008, Blackout

History:

New for 2014 SRO Exam

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

QID: 1003 **Rev:** 2 **Rev Date:** 1/15/18 **Source:** Bank **Originator:** NRC

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

Section: 4.2 **Type:** Generic APEs

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

Description: Ability to apply Technical Specifications for a system.

K/A Number: 2.2.40 **CFR Reference:** 41.10 / 43.2 / 43.5 / 45.3

Tier:	1	**RO Imp:**		**RO Select:**	No	**Difficulty:**	4
Group:	1	**SRO Imp:**	4.7	**SRO Select:**	Yes	**Taxonomy:**	H

Question: **RO:** **SRO:** 79

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

Given:
* Unit 1 at 100% power

At 0900 on February 18, 2018, discovered one cell's float voltage at 2.02 V on D06.

At 1000 on February 18, 2018, Electrical Maintenance reports:

* D06 float current 2.3 amps
* D06 terminal voltage 122.4 V

* D07 float current 2.0 amps
* D07 terminal voltage 128.6 V

Based on the above conditions, and assuming that action completion times are NOT met, Technical Specifications require the unit to be in MODE 3 no later than what time?

A. 1600 on February 18, 2018

B. 1700 on February 18, 2018

C 0000 on February 19, 2018

D. 0200 on February 19, 2018

Answer:

C. 0000 on February 19, 2018

Notes:

"C" is correct. To arrive at the correct answer involves referring to two different LCOs: 3.8.4 for DC Sources - Operating which basically requires two different trains of Vital DC be operable, and 3.8.6 for Battery Parameters which contains specific actions for various vital battery parameters.
The question states that D06 has a cell with float voltage which is low. SR 3.8.6.5 is to verify each connected battery cell float voltages are ≥ 2.07 volts. Since D06 is below this value, then 3.8.6 condition A must be entered. Per action A.1 SR 3.8.4.1 for DC Sources must be performed within 2 hours to verify battery terminal voltage is greater than the minimum float voltage. Also SR 3.8.6.1 must be performed per action A.2 within 2 hours to verify battery float current is less than 2.0 amps. Lastly, affected cell voltage must be restored to ≥ 2.07 volts within 24 hours per action A.3. However, condition F states that if a battery has a cell with low float voltage AND battery float current is > 2.0 amps, then the battery must be declared inoperable immediately. Therefore, D06 must be declared inoperable at 1000 on Feb. 18. This will cause the crew to enter 3.8.4 condition A for one DC electrical subsystem inoperable with 8 hours to restore it to operable status, and 6 hours to be in Mode 3 (14 hours total), so the plant must be in Mode 3 by midnight if the battery is not restored to operable.

"A" is incorrect. An applicant would select this if they made the same assumption as stated above for "B" but calculated the LCO 3.0.3 entry time starting at 0900 instead of 1000.

"B" is incorrect but plausible if applicant determined both batteries were inoperable at 1000 on Feb. 18. D07 battery parameters were checked at the same time as D06. D07 float current is in spec but just barely, and D07 terminal voltage is also in spec. If the applicant believed both batteries are inoperable and LCO 3.8.4 does not have a condition for two inoperable DC subsytems, then the applicant would think 3.0.3 is applicable with 7 hours to be in Mode 3, and determined that the unit must be in Mode 3 by 1700 on Feb. 18.

"D" is incorrect. An applicant would select this if they entered 3.8.6.A at 1000 on Feb. 18, allowed 2 hours to restore at least on battery to within limits per 3.8.6.E, then declared D06 inoperable per 3.8.6.F and determined there were 14 hours to be in mode 3 per 3.8.4.A and 3.8.4.B.

This question matches the K/A since the applicant must use the given conditions and apply the supplied Tech Spec references.

References:

[Provide Tech Specs 3.8.6 and 3.8.4 as references]

ANO1 Tech Specs 3.0.3, 3.8.4, & 3.8.6

History:

New for 2013 SRO Exam
Rev. 1, 10/13/17 - formatting changes;
Revised by adding conditions of amps and terminal voltages for both D06 and D07 due to changes in TS 3.8.6 (old table was deleted), modified question following round 1 of validation so only D06 is inoperable, correct answer is now "A". Modified A and B due to changes in the specs.
Selected for 2018 SRO exam
Rev. 2, 1/15/18, re-ordered answers from earliest to latest, correct answer now "C", per NRC resolution.

QID: 0757 **Rev:** 1 **Rev Date:** 8/30/17 **Source:** Bank **Originator:** Pullin

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

Section: 2 **Type:** Generic KA

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

Description: Ability to perform system specific and integrated plant procedures during all modes of plant operation.

K/A Number: 2.1.23 **CFR Reference:** 41.10/43.5/45.2/45.6

Tier: 1 **RO Imp:** **RO Select:** No **Difficulty:** 4

Group: 1 **SRO Imp:** 4.4 **SRO Select:** Yes **Taxonomy:** H

Question: **RO:** **SRO:** 80

Given:
* Unit 1 in Mode 4
* RCS filled and vented
* P-34A Decay Heat Pump in service with 2800 gpm flow
* All RCPs are secured
* RCS pressure 180 psig

NOW the following are observed:
* RCS temperature 220 °F and rising
* RCS pressure 190 psig and rising
* Train A CET TEMP HI (K09-D6) in alarm
* DH PUMP A/B SUCT TEMP HI (K09-C8) in alarm

For these conditions, which operating procedure and actions are required?

A. 1203.028, Loss of Decay Heat section 5, Loss of Service Water Flow;
Close Service Water Inlet to E-35A, CV-3822, and immediately open supply breaker B-5182.

B. 1203.028, Loss of Decay Heat section 5, Loss of Service Water Flow;
Stop ‘A’ DH Pump and close at least one DH Suction Valve.

C. 1203.028, Loss of Decay Heat section 9, Loss of Both DH Systems, RCS Pressure Boundary Intact;
Close Service Water Inlet to E-35A, CV-3822 and immediately open supply breaker B-5182.

D. 1203.028, Loss of Decay Heat section 9, Loss of Both DH Systems, RCS Pressure Boundary Intact;
Stop ‘A’ DH Pump and close at least one DH Suction Valve.

Answer:

A. 1203.028, Loss of Decay Heat section 5, Loss of Service Water Flow;
Close Service Water Inlet to E-35A, CV-3822
and immediately open supply breaker B-5182.

Notes:

"A" is the correct procedure section due to conditions given of adequate DH flow but temperatures rising. The correct action from step 10 of Section 5 is stated. This action is taken per Caution before step 10 that with RCS temps above 200, that it is possible for the SW side of the affected DH cooler to reach saturation temp. This could cause the SW system to see temps and pressures above the design limit of the piping, so the SW inlet to the DH cooler is closed and the breaker opened (to prevent automatic re-opening).

B is the correct procedure section but incorrect action. This action is plausible since stopping the pump and closing at least one suction valve is taken if RCS pressure cannot be reduced below the applicable limit (step 8 of Section 5). However, the applicable limit with RCS loops filled is 250 psig and that limit has not been reache yet.

C is the incorrect procedure section but correct action. This procedure section is plausible if applicant believe the indications are for a loss of DH pump.

D is the incorrect procedure as explained above but plausible since the action given with this distractor supports entry into this section.

References:

1203.028, Loss of Decay Heat

History:

New for the 2009 Retake SRO Exam
Rev. 1,
editorial changes
Added condition of RCS filled and vented.
Added RCS pressure condition of 190 psig and rising.
Selected for 2018 SRO exam

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

QID: 1004 **Rev:** 1 **Rev Date:** 8/30/17 **Source:** Modified **Originator:** NRC

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

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

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

Description: Ability to determine and interpret the following as they apply to the (Inadequate Heat Transfer): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

K/A Number: EA2.1 **CFR Reference:** 43.5 / 45.13

Tier: 1 **RO Imp:** **RO Select:** No **Difficulty:** 3

Group: 1 **SRO Imp:** 4.4 **SRO Select:** Yes **Taxonomy:** H

Question: **RO:** **SRO:** 81

Given:
* Reactor tripped
* CRS using Loss of Subcooling Margin (1202.002)
* P-7A EFW pump tripped
* P-7B EFW pump out for maintenance
* CET temperatures 612 °F and stable
* Both DGs started due to voltage fluctuation
* A1 and A2 powered from SU#1 Transformer
* RCS pressure 1700 psig and stable

Based on the above current conditions, the CRS should _______.

A. Remain in Loss of Subcooling Margin (1202.002)

B. Transition to Overheating (1202.004)

C. Transition to Inadequate Core Cooling (1202.005)

D. Transition to Degraded Power (1202.007)

Answer:

A. Remain in Loss of Subcooling Margin (1202.002)

Notes:

"A" is correct. Transition conditions are met for other EOPs but Loss of SCM is the highest priority and CRS should remain in this procedure.

"B" is incorrect. Even though CETs are above 610 °F with all feedwater lost, loss of subcooling margin has priority and a transition to Overheating should not be made if SCM is inadequate.

"C" is incorrect, although conditions given show that the RCS is saturated and this could cause entry into ICC if actions are not taken but a transition to the ICC EOP is not required yet.

"D" is incorrect but plausible as this was formerly the correct answer, EDGs started but A1 and A2 energized from SU #1 means offsite power is available.

References:

1202.002, Loss of Subcooling Margin
1202.007, Degraded Power

History:

New for 2013 SRO Exam

Rev.1, editorial changes
Modified question by changing conditions to offsite power is available even though DGs are running. This changes the correct answer from "D" to "A". Changed RCS pressure to 1700 psig so that ICC transition is not correct, and added "rising slowly" as a trend.
Also changed answers to "transition to" or "remain in" to avoid possibility of no correct answer.
Left this question at QID 1004 since it was no longer SRO Only in its original form.
Modified for 2018 SRO exam

QID: 1004 **Rev:** 0 **Rev Date:** 2/18/13 **Source:** Direct **Originator:** NRC

TUOI: **Objective:** **Point Value:** 1

Section: **Type:**

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

Description: Ability to determine and interpret the following as they apply to the (Inadequate Heat Transfer): Adherence to appropriate procedures and operation within the limitations in the facility*s license and amendments.

K/A Number: EA2.2 **CFR Reference:**

Tier: 1 **RO Imp:** **RO Select:** No **Difficulty:** 0

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

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

Q#81 PARENT

Given:
- Reactor tripped and the crew is operating in Loss of Subcooling Margin (1202.002).
- P-7A tripped.
- P-7B is out for maintenance.
- CET temperatures are 610°F and rising.
- Offsite power is subsequently lost.
- Both DGs are supplying their emergency buses.
- RCS pressure is 1800 psig.

Based on the above current conditions, the correct procedure transition is to _______.

A. Remain in Loss of Subcooling Margin (1202.002)

B. Overheating (1202.004)

C. Inadequate Core Cooling (1202.005)

D. Degraded Power (1202.007)

Answer:

D. Degraded Power (1202.007)

Notes:

A. (Incorrect) Even though loss of SCM is the highest priority, the lack of offsite power requires entry into 1202.007 which has steps for loss of SCM.
B. (Incorrect) Even though CETs are rising above 610, with all feedwater is lost, 1202.007 has steps for mitigating an overheating condition.
C. (Incorrect) Conditions given could cause entry into ICC if actions are not taken but entry into the ICC EOP is not required yet.
D. (Correct) This is the correct procedure since offsite power is lost.

References:

1202.004

History:

New for 2013 SRO Exam

QID: 1190 **Rev:** 1 **Rev Date:** 1/15/18 **Source:** New **Originator:** Cork

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

Section: 4.3 **Type:** Generic APEs

System Number: 032 **System Title:** Loss of Source Range Nuclear Instrumentation

Description: Ability to perform specific system and integrated plant procedures during all modes of plant operation.

K/A Number: 2.1.23 **CFR Reference:** 41.10 / 43.5 / 45.2 / 45.6

Tier: 1 **RO Imp:** **RO Select:** No **Difficulty:** 3

Group: 2 **SRO Imp:** 4.4 **SRO Select:** Yes **Taxonomy:** H

Question: **RO:** **SRO:** 82

Given:
* Reactor startup in progress following forced outage
* NR-502 is operable
* Reactor is critical
* Escalation of power in progress

NOW:
* ATC reports NI-501 indication dropped to bottom of scale
* ATC stops pulling rods and reports the following:
  * SR NI-502 8e4 cps
  * IR NI-3 4e-11 amps
  * IR NI-4 7e-11 amps

(1) Which of the following procedures would be in use for this evolution and (2) what action is required to be taken for the above conditions?

A. (1) Plant Startup (1102.002)
(2) Continue power escalation

B. (1) Approach to Criticality (1102.008)
(2) Initiate shutdown to Mode 3 and open all CRD breakers

C. (1) Plant Startup (1102.002)
(2) Initiate shutdown to Mode 3 and open all CRD breakers

D. (1) Approach to Criticality (1102.008)
(2) Continue power escalation

Answer:

B. (1) Approach to Criticality (1102.008)
(2) Initiate shutdown to Mode 3 and open all CRD breakers

Notes:

"B" is correct. Approach to Criticality (1102.008) contains the escalation to just below the POAH (point of adding heat) for recording critical data. The bases of Tech Spec 3.3.10 states that if neither IR channel is >10e-10 amps then both are to be considered inoperable until at least one decade of overlap between Source Range and Intermediate Range is achieved. The IR channels should be indicating 10e-10 amps when the SR channels are indicating ~2e3 cps. The fact that both IR channels are less than 10e-10 amps when the sole SR channel is indicating ~8e4 cps shows there is either a problem with the SR, or both of the IR channels.

"A" is incorrect but plausible since Plant Startup (1102.002) will be the procedure used for power escalation AFTER critical data is recorded at 10e-8 amps. This distractor also contains the incorrect action to take but this action is plausible since the Loss of SR (Section 3) of Loss of Neutron Flux Indication (1203.021) states that plant operations may continue as long as one SR channel is operable. However, Section 2 of 1203.021 would direct a plant shutdown if both IR channels are inoperable, which they are.

"C" is incorrect but plausible since Plant Startup (1102.002) will be the procedure used for power escalation AFTER critical data is recorded at 10e-8 amps. This distractor is more plausible since it contains the correct action to take.

"D" is incorrect but plausible since it contains the correct procedure but is incorrect in that it contains the incorrect action. This action is plausible since Section 3 (Loss of SR) of Loss of Neutron Flux Indication (1203.021) states that plant operations may continue as long as one SR channel is operable. However, Section 2 of 1203.021 would direct a plant shutdown if both IR channels are inoperable, which they are.

This question matches the K/A since it involves a loss of a source range channel leaving just one SR channel tc compare with the IR channels.

References:

1102.008, Approach to Criticality
Technical Specifications, 3.3.10 and bases

History:

New for 2018 SRO exam
Rev. 1, deleted "within one hour" from second part of "B" and "C" to eliminate cueing, per NRC resolution.

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

QID: 1201 **Rev:** 1 **Rev Date:** 11/7/17 **Source:** Mod **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 determine and interpret the following as they apply to the Control Room Evacuation: SG pressure.

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

Tier: 1 **RO Imp:** **RO Select:** No **Difficulty:** 2

Group: 2 **SRO Imp:** 4.0 **SRO Select:** Yes **Taxonomy:** H

Question: **RO:** **SRO:** 83

Given:
* Unit 1 100% power
* Unit 1 has a fire in the Control Room
* Heavy smoke has accumulated in the Unit 2 Control Room
* RCS Tavg 555 °F
* RCS pressure 2160 psig

CRS/SM will be using ______(1)________ and SG pressures are being controlled by ______(2)_________.

A. (1) Remote Shutdown (1203.029);
(2) Turbine Bypass Valves

B. (1) Alternate Shutdown (1203.002);
(2) Turbine Bypass Valves

C. (1) Remote Shutdown (1203.029);
(2) MSSVs

D. (1) Alternate Shutdown (1203.002);
(2) MSSVs

Answer:

D. (1) Alternate Shutdown (1203.002);
(2) MSSVs

Notes:

When conditions require the Control Room to be evacuated due to a fire in the Control Room or Cable Spreading Room, then the Alternate Shutdown AOP will be entered. If the Control Room is evacuated for any other reason, then Remote Shutdown will be used. Alternate Shutdown initially has SG pressure being maintained by Main Steam Safety Valves (MSSVs), but later directs the local manual use of Atmospheric Dump Valves (ADVs) when more personnel are available. Remote Shutdown controls SG pressure with Turbine Bypass valves (TBVs).

"D" is correct since Alternate Shutdown will be in use per the above explanation. An RCS Tavg of 555 °F means that SG Tsat is 555 °F (applicant must use steam tables) and therefore SG pressures are ~1070 psig and thus MSSVs are controlling SG pressure. The given Tavg is within the acceptable range for Mode 3 listed in step 32 of section 1A in 1203.002, Alternate Shutdown.

"A" is incorrect since a fire is forcing evacuation of the Control Room and the ADVs, not TBVs will be used to control SG pressures. This is plausible as Remote Shutdown was the correct answer in the previous version when a fire in Unit 2 forced an evac of Unit 1 due to smoke. The range for SG pressure control via the TBVs in Remote Shutdown is 950 to 1020 psig and since SG pressure is 1070 psig, then TBVs are not being used for pressure control. The range for acceptable RCS temperature in Mode 3 for Remote Shutdown is the same as in Alternate Shutdown, 540 to 560.

"B" is incorrect but plausible since Alternate Shutdown is the correct evacuation procedure but incorrectly identifies the use of Turbine Bypass Valves. The range for SG pressure control in Alternate Shutdown is similar to Remote Shutdown at 980 to 1020 psig but this would be with the Atmospheric Dump Valves (ADVs) and then only if extra personnel were available to take control of the ADVs.

"C" is wrong but plausible because it correctly interprets that MSSVs are conrolling SG pressures but lists the incorrect procedure.

Matches the KA because these are CR evacuation procedures and demonstrates the ability to determine SG pressure using RCS Tavg and thus interpret that MSSVs must be in use to control SG pressure.

References:

1203.002, Alternate Shutdown
1203.029, Remote Shutdown

History:

Modified QID 1116 for 2018 SRO exam by changing conditions from a fire in Unit 2 control room to a fire in Unit 1 control room, this changes the correct answer from "A" to "D". Changed "CRS" to "CRS/SM" since the SM will be directing control of ADVs. Also changed "the CBOT" to "an operator" since an NLO will be performing the actual valve ops, not the CBOT. Changed "950" to "980" since this is the lower part of the band in Exhibit A of 1203.002.
Rev. 1, added Tavg condition of 555 °F, changed C and D second part to "MSSVs", changed A and B second part to "Atmospheric Dump Valves", so that applicant must interpret Tavg and determine SG pressure is being controlled by MSSVs to more closely align with K/A

QID: 1116 **Rev:** 1 **Rev Date:** 5/16/17 **Source:** Repeat **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:** No **Taxonomy:** H

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

Q#83 PARENT

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

Q#83 PARENT

QID: 1191 **Rev:** 0 **Rev Date:** 8/23/17 **Source:** New **Originator:** Cork

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

Section: 4.2 **Type:** Generic APEs

System Number: 069 **System Title:** Loss of Containment Integrity

Description: Ability to determine and interpret the following as they apply to the Loss of Containment Integrity: Verification of automatic and manual means of restoring integrity

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

Tier: 1 **RO Imp:** **RO Select:** No **Difficulty:** 2

Group: 2 **SRO Imp:** 4.4 **SRO Select:** Yes **Taxonomy:** F

Question: **RO:** **SRO:** 84

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

Given:
* Unit 1 in Mode 4 and heating up
* Surveillances in progress in preparation for entering Mode 3
* CBOT reports Reactor Building Purge Supply Valve CV-7402 is open with key inserted

The required action to comply with Tech Specs is to verify the valve closed with key removed within the stated maximum allowed completion time of ______(1)________ since the Tech Spec bases states the valve could not demonstrate the ability to close during a ______(2)_________ .

A. (1) 48 hours
(2) LOCA

B. (1) 48 hours
(2) Main Steam Line Break

C. (1) 72 hours
(2) LOCA

D. (1) 72 hours
(2) Main Steam Line Break

Answer:

A. (1) 48 hours
(2) LOCA

Notes:

"A" is the correct answer. TS LCO 3.6.3 is applicable in Modes 1-4 and has a specific surveillance for the RB Purge Isolation valves, SR 3.6.3.1. LCO Condition A is applicable in this case since there are two isolation valves in the RB supply and exhaust flow paths, therefore with one valve inoperable in each flow path, the completion required time is a maximum of 48 hours for Required Action A.1. The bases for SR 3.6.3.1 states the RB Purge isolation valve failed to demonstrate the ability to close during a LOCA.

"B" is incorrect but plausible since it has the correct completion time but has the wrong Design Basis Accident (DBA) of Main Steam Line Break which is plausible since it does cause a challenge to Containment Integrity.

"C" is incorrect but plausible since it has the correct DBA but an incorrect completion time. The time given is for Conditon C, Required Action C.1, and is thus plausible.

"D" is incorrect since it has an incorrect, but plausible, completion time. The DBA of Main Steam Line Break is also incorrect but plausible since it does cause a challenge to Containment Integrity.

References:

Technical Specifications, 3.6.3 and bases for SR 3.6.3.1

History:

New question for 2018 SRO exam

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

QID: 0736 **Rev:** 2 **Rev Date:** 10/17/17 **Source:** Modified **Originator:** Cork

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

Section: 4.2 **Type:** Generic APE's

System Number: 076 **System Title:** High Reactor Coolant Activity

Description: Ability to interpret and execute procedure steps.

K/A Number: 2.1.20 **CFR Reference:** 41.10 / 43.5 / 45.12

Tier: 1 **RO Imp:** **RO Select:** No **Difficulty:** 4

Group: 2 **SRO Imp:** 4.6 **SRO Select:** Yes **Taxonomy:** H

Question: **RO:** **SRO:** 85

Given:
* Reactor at 100% power
* Failed fuel ratio 20.2

NOW
* Failed fuel ratio 12.1
* Chem reports RCS Dose Equivalent I-131 is 10 µCi/gm

According to _______(1)_________, reactor ______________(2)____________________.

A. (1) TS RCS Activity LCO (3.4.12);
(2) must be in Mode 3 within 6 hours

B. (1) High Activity In Reactor Coolant (1203.019);
(2) power must be reduced to 50%

C. (1) TS RCS Activity LCO (3.4.12);
(2) power must be reduced to 50%

D. (1) High Activity In Reactor Coolant (1203.019);
(2) must be in Mode 3 within 6 hours

Answer:

B. (1) High Activity In Reactor Coolant (1203.019);
(2) power must be reduced to 50%

Notes:

"B" is the correct response per 1203.019, Section 2 - Failed Fuel. This section states if failed fuel ratio drops by 40% to reduce Rx power by 50% of the current level using 1102.016, Power Reduction and Plant Shutdown. The Failed Fuel Ratio values provided show a drop of 40%. The Dose Equivalent I-131 value is representative of the type of increase one would see if Failed Fuel Ratio had dropped, and is similar to an action level listed in step 1.C of 1203.019, Section 1 for a Chemistry alternative sampling method.

"A" is incorrect, but plausible since the RCS Activity LCO (3.4.12) is mentioned in 1203.019, Section 1 - High Gross Gamma Activity, and states that if Dose Equivalent I-131 exceeds 60 µCi/gm to be in Mode 3, but the condition given is that this value is 10 µCi/gm which would cause 3.4.12 Condition A to be entered but there would be 48 hours to restore I-131 within limits. Additionally, a similar action level but with different units is stated in Section 1 (High Gross Gamma Activity) of 1203.019.

"C" is incorrect, but plausible since LCO 3.4.12 is a concern here as stated in the explanation for "C" above. The action given is correct but LCO 3.4.12 has no power level reduction percentage.

"D" is incorrect, but highly plausible since 1203.019 Section does state to place the plant in Mode 3 within 6 hours but only if Dose Equivalent I-131 exceeds 60 µCi/gm.

References:

1203.019, High Activity In Reactor Coolant
Technical Specifications, 3.4.12

History:

Used in 1999 exam, J. Haynes originator
Direct from ExamBank, QID# 1816
Selected for use in 2002 SRO exam.
Modified for use in 2007 SRO exam.
Selected for the 2008 SRO Exam (modified version of 342)
Rev. 1,
Changed A and B distractors to use TS 3.4.12 instead of Rapid Plant Shutdown since there is no other AOP similar to 1203.019
editorial changes
Rev. 2, due to concerns of this question not being SRO level, modified question so that it does not statet Failed Fuel ratio has changed by 40%, instead gave two values for Failed Fuel Ratio, and added a value for I-131 to make the TS LCO more plausible. Changed second half of all distractors to be two possible actions.
Modified as stated above for 2018 SRO exam

QID: 0736 **Rev:** 0 **Rev Date:** 06/02/2008 **Source:** Direct **Originator:** J Haynes

TUOI: A1-LP-RO-AOP **Objective:** 5 **Point Value:** 1

Section: 4.2 **Type:** Generic APE's

System Number: 076 **System Title:** High Reactor Coolant Activity

Description: Ability to determine and interpret the following as they apply to the High Reactor Coolant Activity: Corrective actions required for high fission product activity in the RCS.

K/A Number: AA2.02 **CFR Reference:** 43.4 / 43.5 / 45.13

Tier: 1 **RO Imp:** 2.8 **RO Select:** No **Difficulty:** 4

Group: 2 **SRO Imp:** 3.4 **SRO Select:** No **Taxonomy:** Ap

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

Given:

Q#85 PARENT

- Reactor at 100% power.
- Failed fuel ratio, as indicated by the WCO logs, has dropped by 50%.

According to _______?_________, Reactor power must be reduced to ____?____ power.

A. 1203.045 Rapid Plant Shutdown 50%

B. 1203.019 High Activity In Reactor Coolant 50%

C. 1203.045 Rapid Plant Shutdown 60%

D. 1203.019 High Activity In Reactor Coolant 60%

Answer:

B. 1203.019 High Activity In Reactor Coolant 50%

Notes:

"B" is the correct response per 1203.019 which states to reduce Rx power by 50% of the current level with the given conditions.
"A" is incorrect, both an incorrect power level and incorrect procedure are listed.
"D" is incorrect, an incorrect power level is listed with the correct procedure.
"C" is incorrect, the correct power level with an incorrect procedure is listed.

References:

1203.019, Chg. 011-01-0

History:

Used in 1999 exam.
Direct from ExamBank, QID# 1816
Selected for use in 2002 SRO exam.
Modified for use in 2007 SRO exam.
Selected for the 2008 SRO Exam (modified version of 342)

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

QID: 1204 **Rev:** 0 **Rev Date:** 9/6/17 **Source:** New **Originator:** Cork

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

Section: 3. **Type:** RCS Inventory Control

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

Description: Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

K/A Number: 2.2.25 **CFR Reference:** 41.5 / 41.7 / 43.2

Tier: 2 **RO Imp:** **RO Select:** No **Difficulty:** 3

Group: 1 **SRO Imp:** 4.2 **SRO Select:** Yes **Taxonomy:** H

Question: **RO:** **SRO:** 86

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

Given:
* Unit 1 at 60% power
* I&C reports PDT-1209, HPI Flow to P-32C on C18, is out of specification
* Safety Function Determination Program (1015.45), Att. 2, has been performed

(1) Which of the following Tech Spec LCOs must be entered and (2) what does Tech Spec state as the time limit?

A. (1) Enter LCO 3.5.2 ECCS Operating;
(2) Restore to operable status within 72 hours

B. (1) Enter LCO 3.3.15 PAM Instrumentation;
(2) Restore to operable status within 30 days

C. (1) Enter LCO 3.5.2 ECCS Operating;
(2) Restore to operable status within 18 hours

D. (1) Enter LCO 3.3.15 PAM Instrumentation;
(2) Restore to operable status within 7 days

Answer:

B. (1) Enter LCO 3.3.15 PAM Instrumentation;
(2) Restore to operable status within 30 days

Notes:

"B" is correct. Per bases for LCO 3.5.2, if an ECCS train is considered inoperable solely due to HPI flow indication inoperability, TS 3.0.6 can be invoked and LCO 3.3.15 can be used for TS compliance in lieu of LCO 3.5.2 conditions and required actions. The PAM time limit from 3.3.15 Condition A.1 is 30 days.

"A" is incorrect, but plausible since this LCO must be entered for reasons other than flow inoperability. Additionally, if the Safety Function Determination Program was not applied to support use of 3.0.6, then this would be the appropriate LCO to enter with the correct time clock from 3.5.2 action A.1.

"C" is incorrect but plausible due to the above explanation. The time limit listed is a combination of the times fo 3.5.2 actions B.1 and B.2 (6+ 12).

"D" is incorrect but plausible since this is the correct specification but the time limit listed is from 3.3.15 action C.1 when both channels of indication are inoperable.

This question matches the K/A since it involves the CVCS system (HPI flow indication) and requires the applicant to recall and apply the bases of a Tech Spec LCO.

References:

ANO-1 Technical Specifications 3.5.2 and 3.3.15

TS 3.5.2 and 3.3.15 must be in SRO handout!!!!!

History:

New for 2018 SRO exam

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

QID: 1199 **Rev:** 0 **Rev Date:** 8/31/17 **Source:** Mod **Originator:** Cork

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

Section: 3.5 **Type:** Containment Integrity

System Number: 026 **System Title:** Containment Spray

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:** **RO Select:** No **Difficulty:** 3

Group: 1 **SRO Imp:** 4.2 **SRO Select:** Yes **Taxonomy:** H

Question: **RO:** **SRO:** 87

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 complete

NOW

* RCS pressure 100 psig and stable
* RB Sump level dropped
* RB Flood level steady
* Both LPI Pump discharge pressures fluctuating between 100 - 160 psig
* Annunciators RB SPRAY P35A ES FAILURE (K11-C6) and RB SPRAY P35B ES FAILURE (K11-C7)) are coming in and out of alarm
* Dose Assessment reports dose rates at site boundary approaching Protective Action Guideline limits

CRS should mitigate the event using ______(1)_______ and direct the crew to override and stop __(2)__ RB Spray pump(s).

A. (1) 1202.010 (ESAS),
(2) only one (1)

B. (1) 1202.010 (ESAS),
(2) both (2)

C. (1) 1202.011 (HPI Cooldown),
(2) only one (1)

D. (1) 1202.011 (HPI Cooldown),
(2) both (2)

Answer:

A. (1) 1202.010 (ESAS),
(2) only one (1)

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 EOP is plausible since Primary to Secondary heat transfer is ineffective under these conditions. Due to a LOCA (and loss of SCM) the hot legs would be voided, preventing natural circulation flow. Conditions given (sump level dropping, flood level steady, LPI pump discharge pressures fluctuating, Spray failure annunciators alarming) indicate there is sump blockage. Since there is a breach of Containment stopping one train of RB Spray is directed in Attachment 1 of ESAS (1202.010). This will allow RB Spray flow to continue and hopefully lessen the effects of the offsite release.

"A" is correct per the above explanation.

"B" is wrong but plausible because it references the correct procedure and both trains would be stopped if a containment breach were not occurring, but the report from Dose Assessment indicates there is an offsite release in progress.

"C" is wrong but plausible as HPI cooldown is a LOCA based procedure and refers to RT-15 for RB sump recirculation. Stopping a single train of RB Spray is the correct action to take.

"D" is wrong but plausible as HPI cooldown is a LOCA based procedure, and the actions to stop both trains of RB Spray is incorrect.

Meets the KA since the question involves the CSS and requires applicant to verify RB spray failure alarms are consistent with the other indications of sump blockage.

Modified QID 1152 by:
1. Adding annunciator description to next to last bullet under "NOW"
2. Changing last bullet under "NOW" so that an offsite release is in progress, this changes the correct answer from "B" to "A".

References:

1202.010, ESAS
1203.012J, Annunciator K11 Corrective Action

History:

Modified QID 1152 for 2018 SRO exam

QID: 1152 **Rev:** 2 **Rev Date:** 5/17/17 **Source:** Repeat **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:** No **Taxonomy:** H

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

Q#87 PARENT

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 nc 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

Q#87 PARENT

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: 0649 **Rev:** 1 **Rev Date:** 8/31/17 **Source:** Bank **Originator:** D Thompson

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

Section: 3.6 **Type:** Electrical

System Number: 063 **System Title:** DC Electrical Distribution

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the DC electrical systems and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of ventilation during battery charging.

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

Tier:	2	**RO Imp:**	2.3	**RO Select:**	No	**Difficulty:**	2
Group:	1	**SRO Imp:**	3.1	**SRO Select:**	Yes	**Taxonomy:**	F

Question: **RO:** **SRO:** 88

Given:

* Unit 1 at 100% power
* VCH-4A North Emergency Switchgear Room Chiller has failed
* No compensatory actions have been taken

What are the impacts of the above failure (1) and which procedure contains mitigating actions for this failure (2)?

A. (1) Switchgear and Battery Chargers are inoperable;
(2) Battery and Switchgear Emergency Cooling System (1104.027)

B. (1) Switchgear and Battery Chargers are inoperable;
(2) Chilled Water System (1104.026)

C. (1) Switchgear and Battery Chargers are operable;
(2) Battery and Switchgear Emergency Cooling System (1104.027)

D. (1) Switchgear and Battery Chargers are operable;
(2) Chilled Water System (1104.026)

Answer:

A. (1) Switchgear and Battery Chargers are inoperable;
(2) Battery and Switchgear Emergency Cooling System (1104.027)

Notes:

"A" is correct. Previously, the switchgear and battery operability was not contingent on emergency chiller operability but on room temperature. 1104.027 requires entry into TS actions for inoperability of switchgear and battery chargers until compensatory actions are in place for a chiller failure. Therefore the switchgear and chargers are inoperable.

"B" is incorrect but plausible: although a chiller failure is present and 1104.026 is used to operate the chillers, mitigating actions are not contained in 1104.026 for this system. This choice is plausible since the switchgear and battery chargers are inoperable.

"C" is incorrect but plausible since previously the switchgear and battery chargers remained operable unless room temperatures became elevated above design limits, however this is incorrect now. This contains the correct procedure.

"D" is incorrect but plausible since previously the switchgear and battery chargers remained operable unless room temperatures became elevated above design limits, however this is incorrect now. This contains an incorrect procedure but is plausible: although a chiller failure is present and 1104.026 is used to operate the chillers, mitigating actions are not contained in 1104.026 for this system.

This question matches the K/A since the applicant must predict the effect of a chiller failure on the operability of key component of the DC electrical distribution system and select the procedure containing the mitigating actions for this failure.

References:

1104.027, Battery and Switchgear Emergency Cooling System

History:

New for the 2009 Retake SRO Exam
Rev. 1: revised stem and made editorial changes.
Selected for 2018 SRO exam

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

QID: 1200 **Rev:** 1 **Rev Date:** 1/15/18 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-ELECD **Objective:** 12 **Point Value:** 1

Section: 3.6 **Type:** Electrical

System Number: 064 **System Title:** Emergency Diesel Generators

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Consequences of opening auxiliary feeder bus (ED/G sub supply).

K/A Number: A2.13 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

Tier: 2 **RO Imp:** **RO Select:** No **Difficulty:** 4

Group: 1 **SRO Imp:** 2.8 **SRO Select:** Yes **Taxonomy:** H

Question: **RO:** **SRO:** 89

Given:
* Unit 1 at 100% power
* OP HPI pump is P-36C
* B55/56 on green train
* Annunciator B5/B6 LOSS OF VOLTAGE (K02-A8) alarms
* CBOT reports B5 bus voltage 0 volts
* Annunciator EDG1 AUTO START COMMAND (K01-A1) alarms
* DG1 tied onto A3
* CBOT reports B5 is de-energized and feeder breaker B-512 will NOT close
* After arriving at B5, Inside AO reports flag on 51N, Ground Fault Relay

Considering the above conditions:
(1) what is the highest priority impact to the plant, and
(2) what procedure and action will be used to mitigate the consequences of this event?

A. (1) DG1 is running without Service Water and a Fuel Oil Transfer pump
(2) Loss of Loadcenter (1203.046);
Place DG1 Output breaker A-308 in PULL-TO-LOCK and DG1 in LOCKOUT

B. (1) Main Turbine and Main Generator have lost ACW flow
(2) Abnormal ES Bus Voltage and Degraded Offsite Power (1203.037);
Place Main Lube Oil Temp and Gen H2 Temp control valves in MANUAL and throttle open

C. (1) DG1 is running without Service Water and a Fuel Oil Transfer pump
(2) Abnormal ES Bus Voltage and Degraded Offsite Power (1203.037)
Place DG1 Output breaker A-308 in PULL-TO-LOCK and DG1 in LOCKOUT

D. (1) Main Turbine and Main Generator have lost ACW flow
(2) Loss of Loadcenter (1203.046);
Place Main Lube Oil Temp and Gen H2 Temp control valves in MANUAL and throttle open

Answer:

A. (1) DG1 is running without Service Water and a Fuel Oil Transfer pump
(2) Loss of Loadcenter (1203.046);
Place DG1 Output breaker A-308 in PULL-TO-LOCK and DG1 in LOCKOUT

Notes:

"A" is correct. A loss of B5 loadcenter will cause the A3 feeder breaker to trip open and DG1 to auto start. The DG will tie onto the A3 bus and a SW pump will auto start but the SW supply valve to DG1 is powered from a

B5 powered MCC. DG1 fuel oil transfer pump has lost power so DG1 can only run on the fuel oil in it's day tank. Also, if any ECCS pump auto starts, then it will start without suction since BWST Outlet CV-1407 is closed and has no power. B5 cannot be quickly re-energized since it has a ground fault. Therefore, the correct action to perform is to de-energize A3 and place DG1 in lockout to prevent damage to the DG and any red train powered ECCS pumps.

"B" is incorrect but plausible as this is the impact for a loss of B6. The procedure (1203.037) is incorrect but plausible due to ES bus voltage mentioned in the title. The action is correct for a loss of B6 but not for a loss of B5.

"C" is incorrect but plausible as this is the correct impact and action but the wrong procedure.

"D" is incorrect but plausible as this has the correct procedure but the impact and action for a loss of B6, not B5

This question matches the K/A since the applicant must predict the impact on an EDG due to an opened feeder breaker and select the proper procedure and action which will mitigate the consequences of this event.

References:

1203.046, Loss of Loadcenter

History:

New for 2018 SRO exam
Rev. 1, added EDG auto start annunciator to conditions, also added condition of CBOT reporting B5 bus voltage is 0 volts, changed DG1 bullet to simply "tied onto A3", per NRC resolution.

QID: 1202 **Rev:** 1 **Rev Date:** 1/15/18 **Source:** New **Originator:** Cork

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

Section: 3.5 **Type:** Containment Integrity

System Number: 103 **System Title:** Containment System

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: 2 **RO Imp:** **RO Select:** No **Difficulty:** 3

Group: 1 **SRO Imp:** 4.7 **SRO Select:** Yes **Taxonomy:** H

Question: **RO:** **SRO:** 90

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

Given:
* Unit 1 heating up following a refueling outage
* RCS temp 295 °F

CRS Admin informs Control Room the breaker for P-35A RB Spray pump has been racked down and tagged out for PM.
P-35B Spray pump is operable.

(1) What is the action to comply with Tech Specs and (2) what is the MAXIMUM completion time Tech Specs allows for this action?

A. (1) Restore P-35A Spray pump to operable status;
(2) 36 hours

B. (1) Restore P-35A Spray pump to operable status;
(2) 72 hours

C. (1) Close and de-energize CV-2401 RB Spray Block valve;
(2) 48 hours

D. (1) Close and de-energize CV-2401 RB Spray Block valve;
(2) 72 hours

Answer:

C. (1) Close and de-energize CV-2401 RB Spray Block valve;
(2) 48 hours

Notes:

"C" is correct. Only one train of RB Spray is required by LCO 3.6.5 for Mode 3 but RB isolation valves are required in Modes 1-4. If ESAS Channel 7 were to actuate with P-35 A racked down, then the Block valve CV-2401 would still open and a path would this RB penetration would be compromised. LCO 3.6.3 conditions A.1 and A.2 thus apply and the Block valve must be verified closed and de-energized within 48 hours.

"A" is incorrect but plausible since RB Spray is required in Modes 3 and 4 but the note for LCO 3.6.5 states that only one RB Spray train is required in Modes 3 and 4. The completion time for 3.6.5 Condition E.1 is 36 hours.

"B" is incorrect but plausible since RB Spray is required in Modes 3 and 4 but the note for LCO 3.6.5 states that only one RB Spray train is required in Modes 3 and 4. The completion time for 3.6.5 Condition A.1 is 72 hours.

"D" is incorrect but plausible since LCO 3.6.3 for RB Isolation valves is applicable but the completion time is incorrect. The completion time for 3.6.3 Condtion C.1 is 72 hours but Condition C is not applicable since the RB Spray system also has a check valve on the RB side and is an open system as the note for Condition C states it is only applicable to closed sytems.

This question matches the K/A since it concerns containment isolation valve limiting conditions for operations and isolation valves are part of the Containment System.

References:

Tech Specs LCO 3.6.3 and 3.6.5

TS LCO 3.6.3 must be in SRO handout!!

History:

New for 2018 SRO exam
Removed TS 3.6.5 from SRO handout, per NRC resolution

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

QID: 0455 **Rev:** 1 **Rev Date:** 8/24/17 **Source:** Bank **Originator:** Cork

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

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

System Number: 034 **System Title:** Fuel Handling Equipment System

Description: Knowledge of design feature(s) and/or interlock(s) which provide for the following: Overload protection.

K/A Number: K4.03 **CFR Reference:** 41.7 / 43.7

Tier: 2 **RO Imp:** 2.6 **RO Select:** No **Difficulty:** 3

Group: 2 **SRO Imp:** 3.3 **SRO Select:** Yes **Taxonomy:** H

Question: **RO:** **SRO:** 91

You are the SRO in Charge of Fuel Handling and a fuel assembly is being removed from the core.

What is the implication of the Fuel Load Cell reading 2600 pounds?

A. Full weight of fuel assembly is on the hoist.

B. Fuel assembly is hung up on a grid strap.

C. Fuel assembly cannot be moved in fast speed.

D. Fuel hoist cannot be lowered.

Answer:

B. Fuel assembly is hung up on a grid strap.

Notes:

"B" is correct, a reading of 2600 pounds will initiate the overload interlock which prevents damage to an assembly when it is stuck, as in when a grid clip hangs up on another assembly.

"A" is incorrect, a fuel assembly weighs approximately 1800 to 2000 pounds, the fuel load interlock of >1200 pounds means the weight of an assembly is on the grapple and prevents the grapple from disengaging while an assembly is in the hoist.

"C" is incorrect, the hoist is limited to slow speeds only while in 3 different slow zones but the hoist can move in fast speed when not in one of these zones.

"D" is incorrect, the low load interlock (<600 pounds) indicates the weight of the 800 pound grapple tube is resting on top of a fuel assembly and prevents further lowering to prevent damage to the assembly.

This question is SRO-Only since it involves fuel handling facilities and procedures . 10CFR55.43(b)(7)

References:

STM 1-51, Fuel Handling Equipment

History:

Created for 2002 SRO exam.
Rev. 1, 8/24/17
Added notes.
Made minor editorial changes.
Selected for 2018 SRO exam

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

QID: 1192 **Rev:** 1 **Rev Date:** 10/9/17 **Source:** New **Originator:** Cork

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

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

System Number: 075 **System Title:** Ciculating Water

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:** 41.5 / 43.5 / 45.12 / 45.13

Tier: 2 **RO Imp:** **RO Select:** No **Difficulty:** 3

Group: 2 **SRO Imp:** 4.7 **SRO Select:** Yes **Taxonomy:** H

Question: **RO:** **SRO:** 92

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

Given:
* Unit 1 at 100%
* Annunciator SW BAY LEVEL LOW (K10-A4) alarms
* Annunciator DARDANELLE RESERVOIR LEVEL LO (K15-B5) alarms
* U.S. Army Corps of Engineers reports Dardanelle Lock has failed
* Lake level lowering at rate of one foot per hour
* CRS enters Natural Emergencies (1203.025)

NOW
* Circ Water pump discharge pressures fluctuating from 2 to 4 psig
* Inside AO reports P-3A and P-3B Circ Pump amps fluctuating from 150 to 200 amps
* ECP level 5.7 feet

What action is required to be taken and what EAL classification should be declared?

A. Perform plant shutdown at maximum safe rate using Rapid Plant Shutdown (1203.045)
Unusual Event

B. Perform plant shutdown at maximum safe rate using Rapid Plant Shutdown (1203.045)
Alert

C. Trip reactor and perform 1202.001 in conjunction with 1203.025
Unusual Event

D. Trip reactor and perform 1202.001 in conjunction with 1203.025
Alert

Answer:

C. Trip reactor and perform 1202.001 in conjunction with 1203.025
Unusual Event

Notes:

"C" is correct. In accordance with 1203.025, Section 5 - Loss of Dardanelle Reservoir, step 2.I, if low lake level causes degradation in Circ Pump performance (erratic discharge pressure and fluctuating motor amps), then the reactor should be tripped and 1202.001 performed in conjunction with 1203.025. In accordance with 1903.010 the appropriate EAL classification is Unusual Event (HU-6) An upgrade to Alert would be made if the ECP were inoperablem, and ECP level is given as 5.7 ft. which is adequate for a 30 day supply.

"A" is incorrect but plausible since this action is given in Section 5 but the reactor should be tripped based on Circ Pump indications. The EAL classification is correct.

"B" is incorrect but plausible since this action is given in Section 5 but the reactor should be tripped based on

Circ Pump indications. The EAL classification of Alert is incorrect but plausible if applicant cannot recall ECP level which is the threshold for operabiity (66.9" or 5.575 ft to ensure 30 day supply in bases of Tech Specs for 3.7.8).

"D" is incorrect but plausible since this is the correct action to take. The EAL classification of Alert is incorrect but plausible if applicant cannot recall ECP level which is the threshold for operabiity (66.9" or 5.575 ft to ensure 30 day supply in bases of Tech Specs for 3.7.8).

This question matches the K/A snce the applicant is given indications of plant performance and the applicant must evaluate those and select the action which is appropriate for the indications.

References:

1203.025, Natural Emergencies
1903.010, Emergency Action Level Classification

History:

New question for 2018 SRO exam
Rev. 1, added ECP level for added plausibility of Alert

QID: 1193 **Rev:** 1 **Rev Date:** 10/9/17 **Source:** New **Originator:** Cork

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

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

System Number: 086 **System Title:** Fire Protection System

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the Fire Protection System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Inadvertent actuation of the FPS due to circuit failure or welding.

K/A Number: A2.03 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

Tier: 2 **RO Imp:** **RO Select:** No **Difficulty:** 3

Group: 2 **SRO Imp:** 2.9 **SRO Select:** Yes **Taxonomy:** H

Question: **RO:** **SRO:** 93

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

Given:
* Unit 1 100% power
* Annunciator K12-A1 FIRE alarms at 1000
* CBOT reports alarm on C463 panels is red LED on
  B4-6U UNPPR ZONE 79-U
* WCO reports no actual fire in area
* WCO reports actuation was spurious

NOW
* 1045 - Reset of Grinnell A4 Multimatic sprinkler valve in progress and expected to take 45 minutes

Which of the following, when performed within one hour, satisfies the TRM for Zone 79-U?

A. ONLY establish a continuous fire watch.

B. ONLY establish a 1-hour roving fire watch.

C. Establish a continuous fire watch AND run hoses for backup suppression.

D. Establish a 1-hour roving fire watch AND verify alternate smoke/heat detection with control room alarm.

Answer:

C. Establish a continuous fire watch and run hoses for backup suppression.

Notes:

"C" is correct. The Upper North Piping Penetration Room (UNPPR) has a smoke detection system which automatically actuates a 4" multimatic valve (UAV-5654). The smoke detection system being in a "fire" alarm condition would trip the deluge valve. Removal of the smoke detector string would render the detection string inoperable since this would break string continuity. A multimatic sprinkler valve must be disassembled in order to reset it, this will take at least an hour to do (all sprinkler valves require isolation for reset but only multimatics require disassembly). This requires a continuous fire watch per the TRM. Therefore, the sprinkler valve is inoperable and cannot be manually actuated so hoses must be run to the UNPPR as backup fire suppression. With the detection system also inoperable this requires a continuous fire watch per the TRM.

"A" is incorrect but plausible if only the detection were inoperable. In that case the sprinkler valve would still be capable of manual actuation and hoses would not have to be run.

"B" is incorrect but plausible if the applicant thinks of the sprinkler valve as having performed it's function as designed and the only problem is with the detection system.

"D" is incorrect but plausible if the applicant recognizes the detection system has made the sprinkler system inoperable but doesn't know there is not an alternate detection system available for the UNPPR (as in a cross zoned system). With this detection system a single detector malfunction will bring in the trouble alarm for this area and no further alarms will come in, therefore the entire detection for the area is inoperable. A cross zoned system has two separate, independent detector strings and thus the detection could remain operable since a separate detector string would be available but the UNPPR is NOT a cross zoned system.

References:

ANO-1 Technical Requirements Manual 3.3.6, 3.7.9

TRM TRO's 3.3.6 and 3.7.9 must be in SRO handout!!!!

History:

New question for 2018 SRO exam
Rev. 1, revised conditions

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

QID: 1194 **Rev:** 0 **Rev Date:** 8/26/17 **Source:** New **Originator:** Cork

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

Section: 2.0 **Type:** Generic K/As

System Number: 2.1 **System Title:** Conduct of Operations

Description: Knowledge of refueling administrative requirements.

K/A Number: 2.1.40 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier: 3 **RO Imp:** **RO Select:** No **Difficulty:** 2

Group: **SRO Imp:** 3.9 **SRO Select:** Yes **Taxonomy:** F

Question: **RO:** **SRO:** 94

In regards to the following statement choose the answer which meets the requirements of Control of Unit 1 Refueling (1502.004) as well as that of Fuel Handling Process (EN-FAP-OU-108).

During ______(1)______ an SRO or SRO limited to fuel handling shall directly supervise the activity and shall have no other duties. Bridge operators are limited to ___(2)___ hours of continuous duty.

A. (1) alterations of the core;
(2) three

B. (1) spent fuel movement in SFP;
(2) three

C. (1) alterations of the core;
(2) six

D. (1) spent fuel movement in SFP;
(2) six

Answer:

A. (1) alterations of the core;
(2) three

Notes:

"A" is correct. In accordance with both 1502.004 and EN-FAP-OU-108 an SRO or SRO limited to fuel handling shall supervise core alterations with no concurrent duties. In accordance with 1502.004 a bridge operator is limited to ~3 hours of continuous duty to maintain maximum attentivelness.

"B" is incorrect but plausible since the time given for bridge operators is current but in accordance with EN-FAP-OU-108 a Fuel Handling Supervisor who is a first line supervisor and has met the training requirements can supervise the movement of spent fuel in the spent fuel pool.

"C" is incorrect since six hours is too long for a bridge operator's continuous duty but plausible since alterations of the core is correct.

"D" is incorrect since six hours is too long for a bridge operator's continuous duty but plausible since a six hour tour is a common concept (half of a 12 hour watch). "D" is also incorrect since any movement of spent fuel may be supervised by a Fuel Handling Supervisor who is non-licensed, as long as that spent fuel movement takes place in the spent fuel pool area.

This question matches the K/A since it requires recall of refueling administrative requirements.

This question is SRO Only since it involves 55.42(b)(7), fuel handling facilities and procedures.

References:

1502.004, Control of Unit 1 Refueling
EN-FAP-OU-108, Fuel Handling Process

History:

New question for 2018 SRO exam

QID: 1207 **Rev:** 0 **Rev Date:** 9/8/17 **Source:** New **Originator:** Cork

TUOI: **Objective:** **Point Value:** 1

Section: 2.0 **Type:** Generic K/As

System Number: 2.1 **System Title:** Conduct of Operations

Description: Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.

K/A Number: 2.1.5 **CFR Reference:** 41.10 / 43.5 / 45.12

Tier: 3 **RO Imp:** **RO Select:** No **Difficulty:** 2

Group: **SRO Imp:** 3.9 **SRO Select:** Yes **Taxonomy:** F

Question: **RO:** **SRO:** 95

In accordance with Fatigue Management Program (EN-OM-123) the ______(1)_______ can approve a waiver of work hour limits and ___________(2)_______________ is a situation where work hour limits may be waived.

A. (1) General Manager Plant Operations
(2) RO just finished working on unplanned forced outage due to Tech Spec LCO and needs to attend required training

B. (1) Shift Manager
(2) RO just finished working on unplanned forced outage due to Tech Spec LCO and needs to attend required training

C. (1) General Manager Plant Operations
(2) SRO is needed for unplanned equipment outage that changes plant status risk to yellow

D. (1) Shift Manager
(2) SRO is needed for unplanned equipment outage that changes plant status risk to yellow

Answer:

C. (1) General Manager Plant Operations
(2) SRO is needed for unplanned equipment outage that changes plant status risk to yellow

Notes:

"C" is correct per section 5.9 and Attachment 9.7 of EN-OM-123. Definition 3.8, Condition Adverse to Safety or Security, lists an unplanned increase in the plant status risk color assignment as an example.

"A" is incorrect but plausible since the GMPO is one of two persons that may approve a waiver of work hour limits but Step 5.9[4] states that the waiver process is not applicable if a covered worker will exceed work hour limits while performing non-covered work, therefore the RO cannot exceed work hour limits to attend training, even if the training is required by regulations.

"B" is incorrect but plausible since the Shift Manager can request a waiver of work hour limits but he cannot approve it. The RO situation is incorrect as explained above.

"D" is incorrect but plausible since the Shift Manager can request a waiver of work hour limits but he cannot approve it. The SRO situation is correct.

This question matches the K/A since the applicant must know the procedural requirements of overtime limitations.

References:

EN-OM-123, Fatigue Management Program

History:

New for 2018 SRO exam

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

QID: 1197 **Rev:** 1 **Rev Date:** 1/15/18 **Source:** New **Originator:** Cork

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

Section: 2.0 **Type:** Generic K/As

System Number: 2.2 **System Title:** Equipment Control

Description: Ability to determine operability and/or availability of safety related equipment.

K/A Number: 2.2.37 **CFR Reference:** 41.7 / 43.5 / 45.12

Tier: 3 **RO Imp:** **RO Select:** No **Difficulty:** 3

Group: **SRO Imp:** 4.6 **SRO Select:** Yes **Taxonomy:** H

Question: **RO:** **SRO:** 96

Given:
* Unit 1 at 100% power
* CRS discovers an ES pump Quarterly Test (92 days) ,was last performed 95 days ago

Which one of the following correctly completes the following statement to describe the operability of the pump?

The ES pump ______________________________ .

A. Shall be declared INOPERABLE from the time of discovery, and will remain inoperable until the surveillance is successfully completed.

B. May be considered OPERABLE as long as the surveillance is performed within the applicable LCO time period, with a risk evaluation performed if delayed greater than this time period.

C. May be considered OPERABLE for an additional 20 days, provided the surveillance is successfully completed within this additional time.

D. Shall be declared INOPERABLE if a risk evaluation is NOT performed within 24 hours from the time of discovery.

Answer:

C. May be considered OPERABLE for an additional 20 days, provided the surveillance is successfully completed within this additional time.

Notes:

"C" is correct. A quarterly surveillance is performed every 92 days. SR 3.0.2 allows a 25% grace period of the surveillance interval from the previous performance to allow future performance of the surveillance. 25% of 92 is 23 days, 3 days have already elapsed, so there are 20 days left.

"A" is incorrect, this is a paraphrase of a statement in SR 3.0.1, and is plausible since this would be true if the 25% grace period had been exceeded but it has not.

"B" is incorrect, this is plausible since the phrase "LCO time period" is similar but not the same as what SR 3.0.3 states which is "limit of the specified frequency".

"D" is incorrect, this is plausible since it is a re-wording of SR 3.0.3 which allows a delay of 24 hours to perform a surveillance if the frequency has been exceeded but the time stated is per 3.0.2 and this answer only states if a risk evaluation is not performed within 24 hours of discovery, it does not mention surveillance performance.

This question matches the K/A since it requires the applicant to generically determine operability of safety related equipment with respect to surveillance frequency requirements in Tech Specs.

References:

ANO-1 Technical Specifications, SR 3.0.2

History:

New for 2018 SRO exam
Rev. 1, revised "D" to "...if a risk evaluation is NOT performed within 24 hours...), per NRC resolution.

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

QID: 0852 **Rev:** 2 **Rev Date:** 1/15/18 **Source:** Bank **Originator:** Cork

TUOI: ASLPP-SRO-MNTC **Objective:** 2 **Point Value:** 1

Section: 2 **Type:** Generic K&A

System Number: 2.2 **System Title:** Equipment Control

Description: Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator.

K/A Number: 2.2.17 **CFR Reference:** 41.10 / 43.3 / 45.13

Tier: 3 **RO Imp:** 2.6 **RO Select:** No **Difficulty:** 2

Group: G **SRO Imp:** 3.8 **SRO Select:** Yes **Taxonomy:** H

Question: **RO:** **SRO:** 97

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

Given:
* Unit One at 100% power

Following items are on the schedule to be worked this shift:
(1) 1304.205, EFIC Channel A Monthly Test
(2) 1305.036, Unit 1 Power Range Linear Amp Calibration At Power
(3) Chemistry sampling both CFTs
(4) Swapping SW Pumps P4C to P4B for strainer cleaning
(5) 1304.169, Unit 1 Main Steam Line Radiation Monitor Test

NOW
CBOT states NI-7 has a power error of +0.8%

In accordance with COPD013, Operations Maintenance Interface Standards and Expectations, which of the above items are allowed to be worked this shift? (Shift Manager approval has NOT been given to exceed normal limits.)

A. 2, 3, 5

B. 2, 3, 4

C. 3, 4, 5

D. 1, 4, 5

Answer:

C. 3, 4, 5

Notes:

Per COPD013 Att. M the activities listed have the following point values:
2 for (1) 1304.205, EFIC Channel A Monthly Test
4 for (2) 1305.036, Unit 1Power Range Linear Amp Calibration At Power (high due to adjustment)
1 for (3) Chemistry sampling both CFTs
2 for (4) Swapping SW Pumps P4C to P4B for strainer cleaning
1 for (5) 1304.169, Unit 1 Main Steam Line Radiation Monitor Test
Per the same attachment the maximum sum of all activities must be 4 or less for normal control room shifts, i.e., non-outage.
Therefore only "C" meets this requirement with a total of 4.

References:

COPD013, Operations Maintenance Interface Standards and Expectations

Attachment M (Table Only) must be in SRO handout.

History:

New for 2011 SRO Exam.
Rev. 1, 11/9/17, editiorial changes, added "CBOT states NI-7 has a power error of +0.8%" so that applicant must recognize this requires an adjustment vs. simply stating that an adjustment is needed, replaced item 1 with EFIC monthly test, item 3 to sampling CFTS, and item 5 to Main Steam Rad Monitor monthly test. The last 3 changes were needed due to COPD revision.
Rev. 2, 1/15/18, revised "A" to be "2,3,5", removed all but the table of Att. M from SRO handout, per NRC resolution.

QID: 1242 **Rev:** 0 **Rev Date:** 2/6/18 **Source:** New **Originator:** Cork

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

Section: 2.0 **Type:** Generic K/As

System Number: 2.3 **System Title:** Radiation Control

Description: Knowledge of radiation exposure limits under normal or emergency conditions.

K/A Number: 2.3.4 **CFR Reference:** 41.12 / 43.4 / 45.10

Tier:	3	**RO Imp:**		**RO Select:**	No	**Difficulty:**	2
Group:		**SRO Imp:**	3.7	**SRO Select:**	Yes	**Taxonomy:**	F

Question: **RO:** **SRO:** 98

What is the TEDE dose limit (1) in REM allowed for Emergency Workers and what is the MINIMUM emergency classification (2) at which the operations personnel log onto the Emergency RWP?

A. (1) 10
(2) Alert

B. (1) 5
(2) Site Area Emergency

C. (1) 10
(2) Site Area Emergency

D. (1) 5
(2) Alert

Answer:

D. (1) 5
(2) Alert

Notes:

"D" is correct. Position Guide A for the Shift Manager/ Emergency Directors states when the minimum classification of Alert is declared, to ensure Operations personnel are instructed to log on to the Emergency RWP. The Emergency RWP automatically imposes a dose limit of 5 Rem (Emergency dose limit equivalent to 10CFR20 annual dose limit).

"A" is incorrect but plausible since it contains the correct minimum emergency classification, however it has a dose limit of 10 Rem which is also plausible since this is the dose limit for Emergency Repair Team personnel to repair equipment.

"B" is incorrect but plausible since it contains the correct emergency dose limit but lists the next highest EAL classification. An SAE would require logging on to the Emergency RWP but it is not the minimum classification

"C" is incorrect but plausible since it has a dose limit of 10 Rem which is the dose limit for Emergency Repair Team personnel to repair equipment.

References:

1903.064, Emergency Response Facillity - Control Room
EP-4-ALL, Exposure Authorization Form

History:

New question for 2018 SRO exam, this question replaces QID 1196 which had a 100% miss rate during all phases of validation.

QID: 1198 **Rev:** 0 **Rev Date:** 8/29/17 **Source:** New **Originator:** Cork

TUOI: A1SPG-SRO-EAL **Objective:** 1 **Point Value:** 1

Section: 2.0 **Type:** Generic K/As

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:** 41.10 / 41.12 / 43.5 / 45.11

Tier: 3 **RO Imp:** **RO Select:** No **Difficulty:** 4

Group: **SRO Imp:** 4.4 **SRO Select:** Yes **Taxonomy:** H

Question: **RO:** **SRO:** 99

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

Given:
* General Emergency declared five minutes ago as first EAL declaration
* HPI flow ~850 gpm total
* CNTMT High Range Rad Monitors reading 5280 R/hr
* Indications of leakage past RB Purge isolations
* Wind direction 348.2
* No dose assessment is available yet

What PAR is required to be made for this event?

A. Evacuate zones G U, shelter zones S T, zones H I J K L M N O P Q R to go indoors

B. Evacuate zones G U, zones H I J K L M N O P Q R S T to go indoors

C. Evacuate zones G H U, zones I J K L M N O P Q R S T to go indoors

D. Evacuate zones G H U, shelter zones I S T, zones J K L M N O P Q R to go indoors

Answer:

D. Evacuate zones G H U, shelter zones I S T, zones J K L M N O P Q R to go indoors

Notes:

"D" is correct. With a large break LOCA indicated (HPI flow of 850 gpm) and CNTMT high range radiation monitors reading 5280 R/hr, this would be a "rapidly progressing severe accident" in accordance with 1903.011, Att. 6, PAR flow chart page 1. This would direct the person with Emergency Direction and Control to choose PAR 7. With a wind direction of 348.2, this would be the last line of the table.

"A" is incorrect but plausible if applicant uses correct PAR 7 but gets wind direction incorrect.

"B" is incorrect but plausible if applicant uses PAR 1 instead of PAR 7 and gets wind direction wrong.

"C" is incorrect but plausible if applicant uses PAR 1 (or PAR 3) instead of PAR 7 with correct wind direction.

References:

1903.011, Emergency Response/Notifications

1903.011 pages 46 thru 56 must be in SRO handout

History:

New for 2018 SRO exam

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

QID: 0998 **Rev:** 2 **Rev Date:** 1/15/18 **Source:** Bank **Originator:** NRC

TUOI: ASLP-RO-EPLAN **Objective:** 4 **Point Value:** 1

Section: 2.0 **Type:** Generic K/As

System Number: 2.4 **System Title:** Emergency Procedures/Plan

Description: Knowledge of SRO responsibilities in emergency plan implementation.

K/A Number: 2.4.40 **CFR Reference:** 41.10 / 43.5 / 45.11

Tier:	3	**RO Imp:**	2.7	**RO Select:**	No	**Difficulty:**	2
Group:		**SRO Imp:**	4.5	**SRO Select:**	Yes	**Taxonomy:**	F

Question: **RO:** **SRO:** 100

Given:
* ANO-1 in Mode 3
* Alert declared on Unit One at 1200
* TSC is operational
* Unit 1 Shift Manager passes out at 1210, and medical assistance is requested
* Replacement Shift Manager has been contacted at home and will report onsite to take watch within the hour

During the time period before the replacement takes watch, per Emergency Action level Classification (1903.010), who has the FIRST responsibility of Emergency Direction and Control for Unit One?

A. Unit 1 Shift Technical Advisor

B. Unit 2 Shift Manager

C. Unit 1 Control Room Supervisor

D. TSC Emergency Plant Manager

Answer:

C. Unit 1 Control Room Supervisor

Notes:

"C" is correct, per procedure 1903.010, Section 5.2, if the unit Shift Manager is not available to assume his/her Emergency Direction and Control responsibility, the unit Control Room Supervisor (CRS) will assume this responsibility until a replacement Shift Manager arrives.

"A" is plausible because the person filling this position assists the Shift Manager in EAL classifications and notifications, and may be an individual who is licensed as a SRO. However, per procedure 1903.010, Section 5.2, this is incorrect.

"B" is plausible because the unit Shift Managers both implement actions in procedure 1903.010 during events, so it may be perceived that they are an acceptable replacement while the incoming Shift Manager heads to the site. However, this is incorrect. Additional credibility is bestowed on this distractor due to FLEX iimplementatioı there are events where the emergency is on one unit while the opposite unit's Shift Manager directs FLEX actions to Unit One operators.

"D" is plausible because this is one of the positions that is filled when the Emergency Response Organization (ERO) is activated, and this individual may assume Emergency Direction and Control responsibilities during the course of an event. However, this individual is not manning the TSC on a continuous basis to perform this role whenever an event occurs. Therefore, the responsibility falls to the on shift CRS, as directed in procedure 1903.010.

This question matches the K/A since the applicant must recall a responsibility of the CRS (SRO) during an emergency event.

This question is SRO Only since it is linked to 10CFR55.42(b)(5) and is a specific responsibility of the SRO licensed CRS on duty in the Control Room.

References:

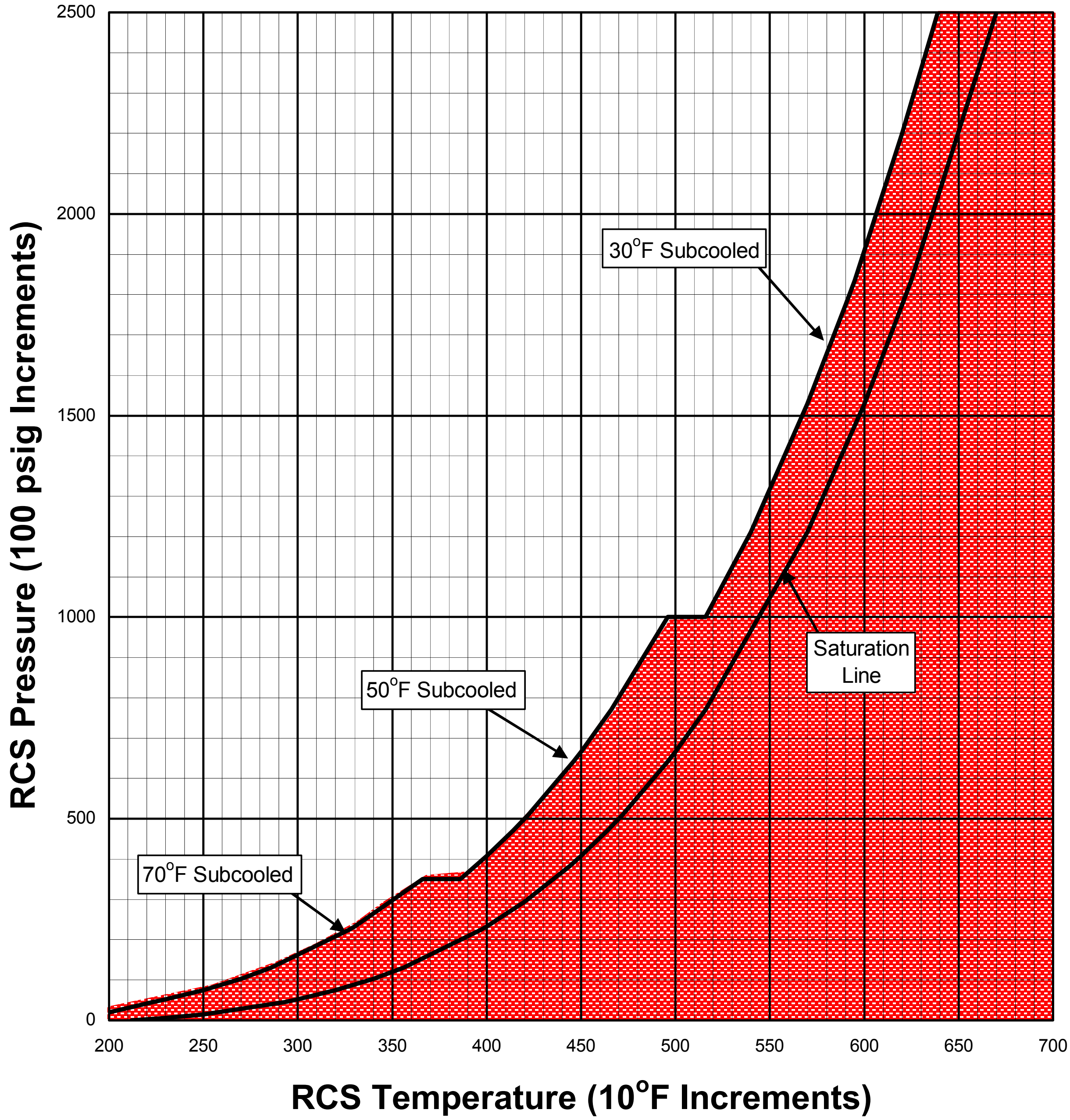
1903.010, Emergency Action level Classification

History:

New for 2013 Exam
Rev. 1, 8/29/17
Editorial changes
Added that Alert has been declared on Unit One
Added times so applicant could determine that TSC would not be staffed.
Selected for 2018 SRO exam
Rev. 2. 1/15/18, changed times to day shift, added condition of TSC operational, added info about FLEX to notes, per NRC resolution.

2018
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

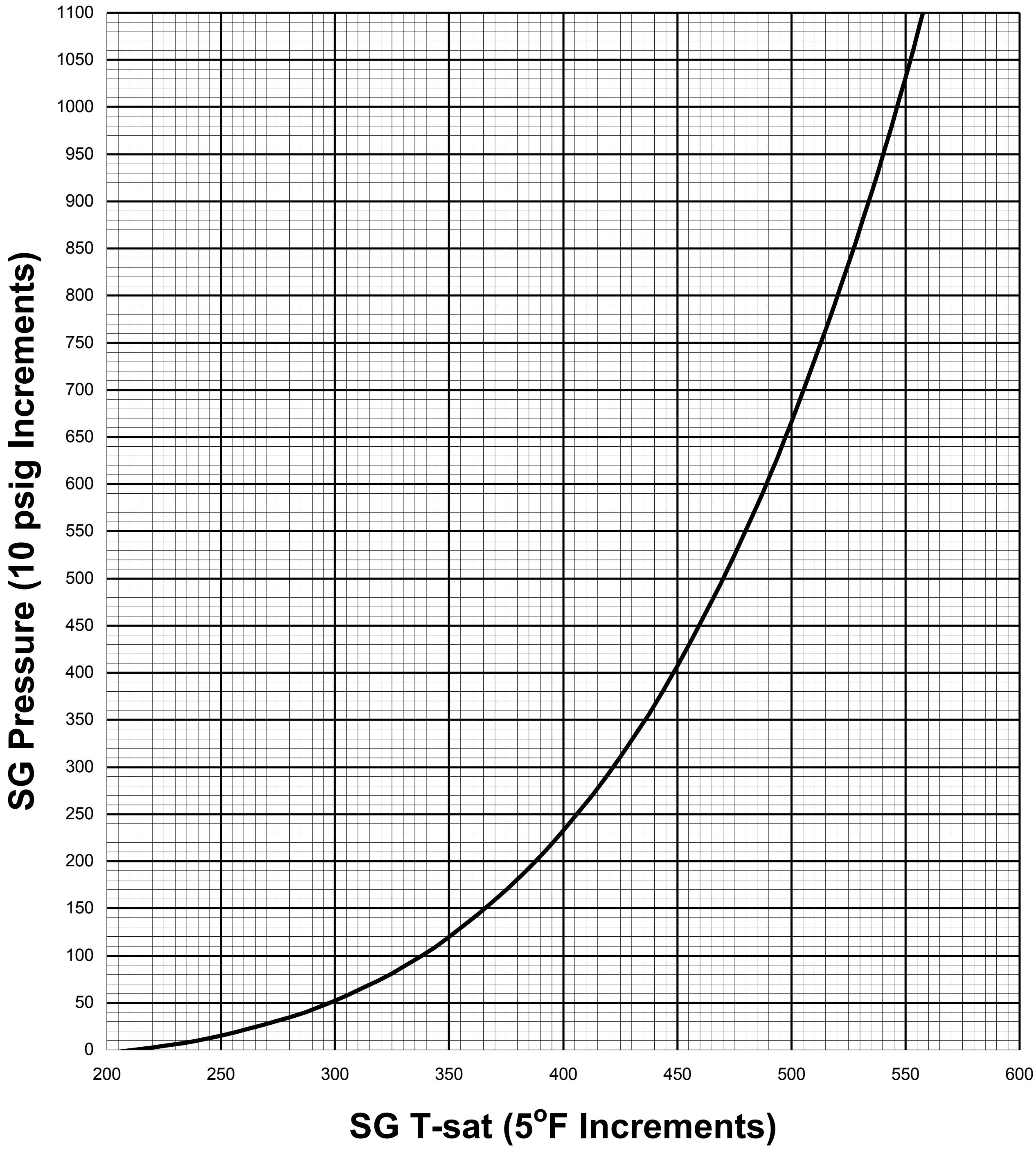


FIGURE 3
RCS Pressure vs Temperature Limits

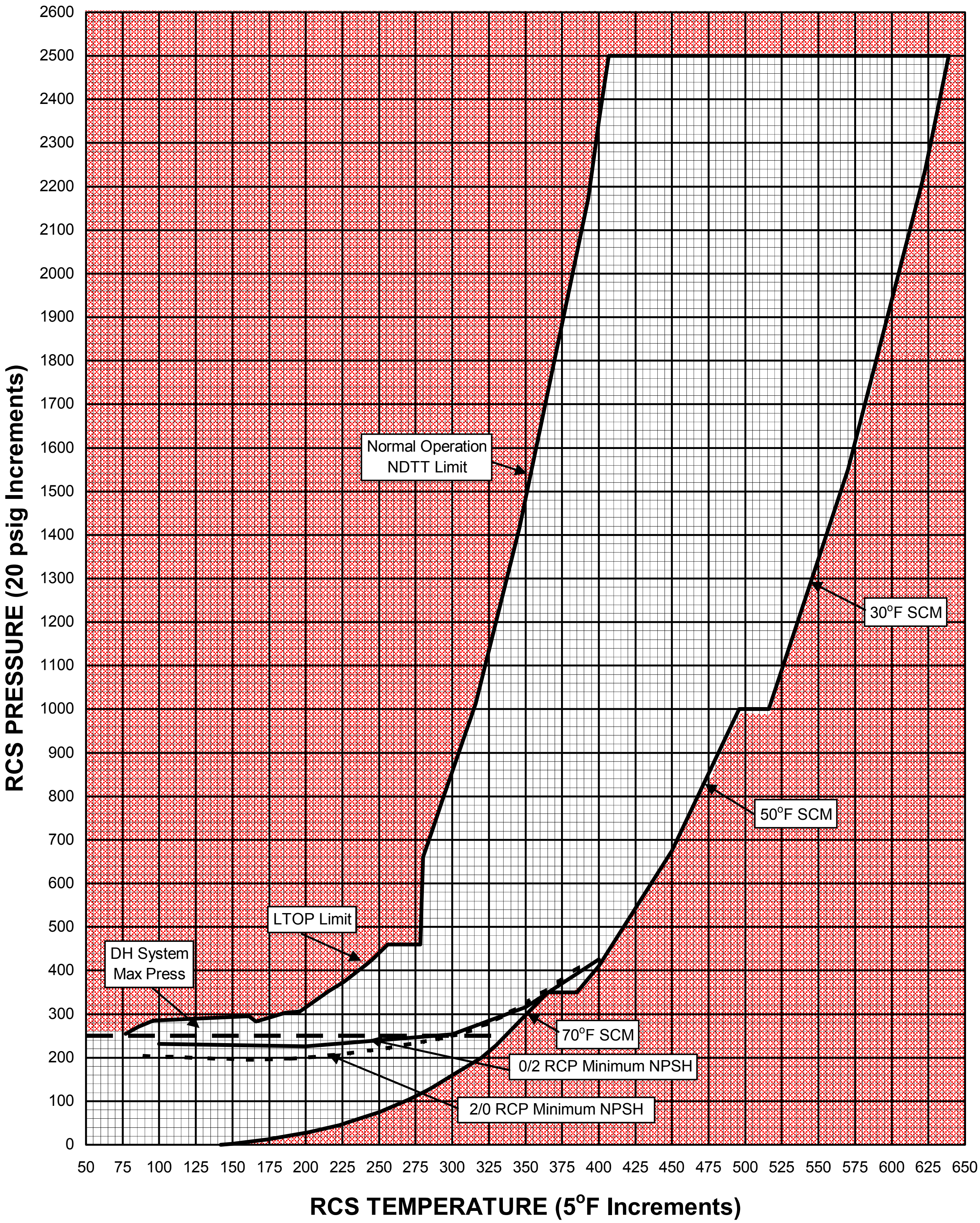


FIGURE 4
Core Exit Thermocouple for Inadequate Core Cooling

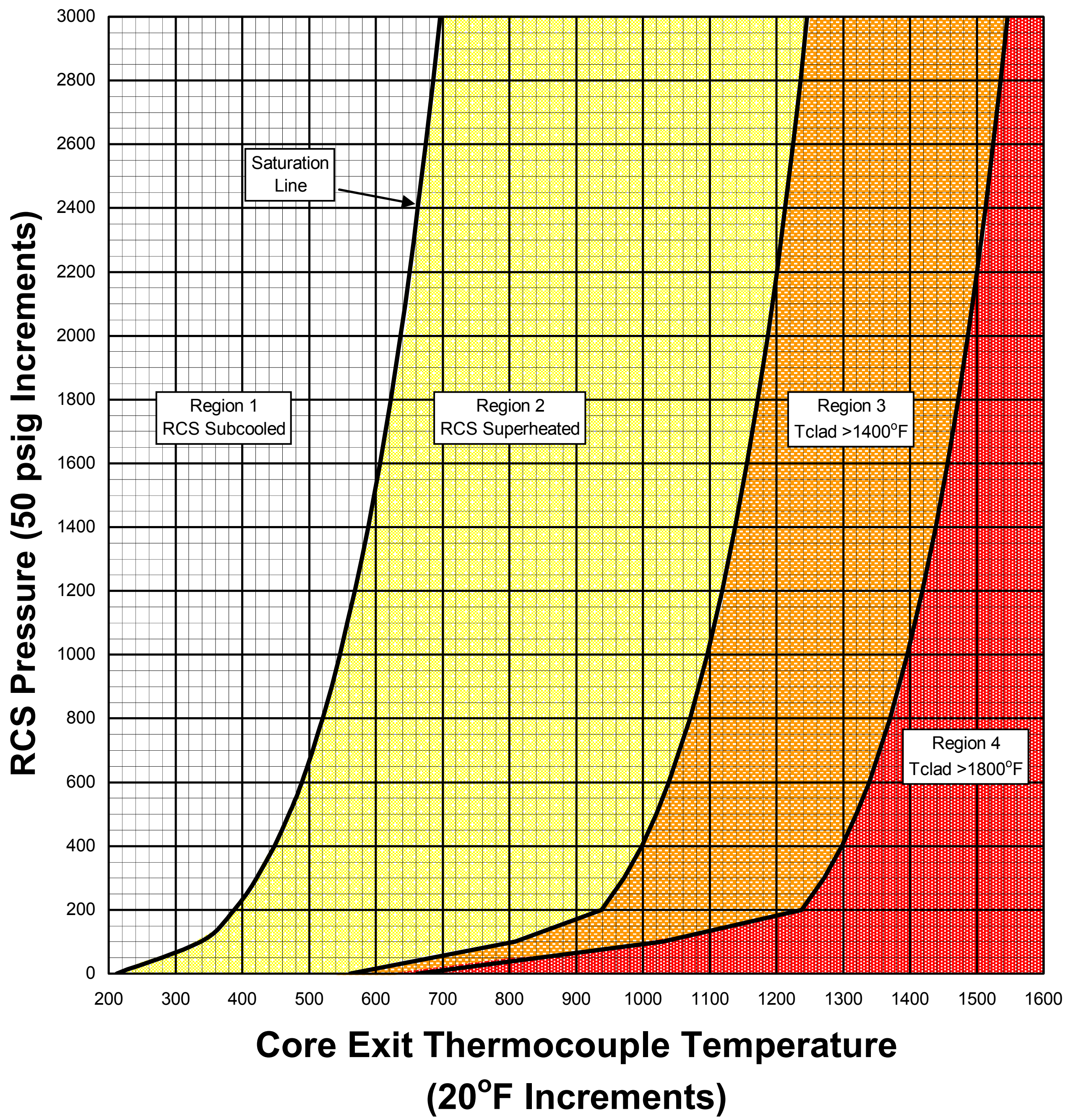


FIGURE 5

SG Pressure to Establish 40° to 60°F Primary to Secondary ΔT

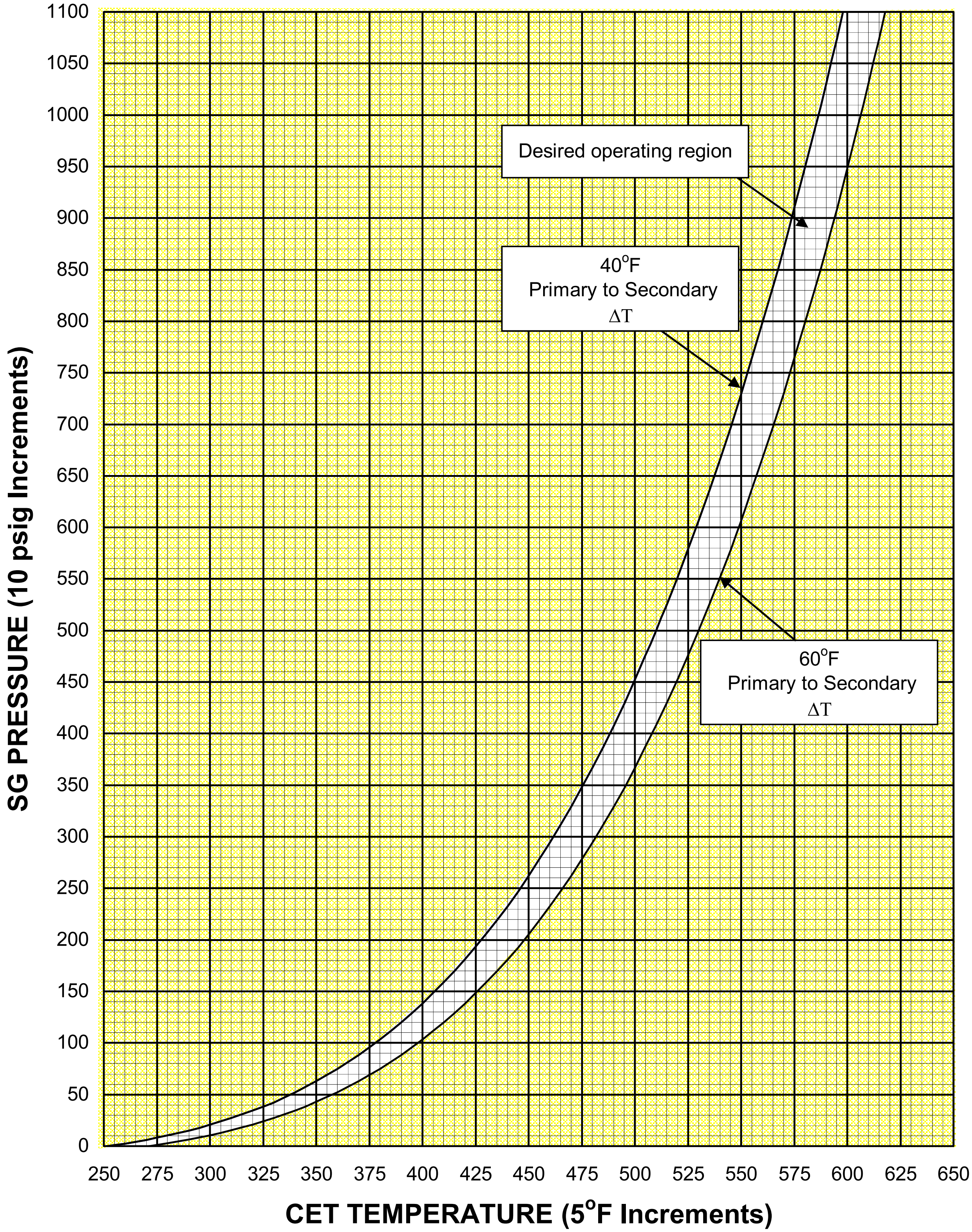
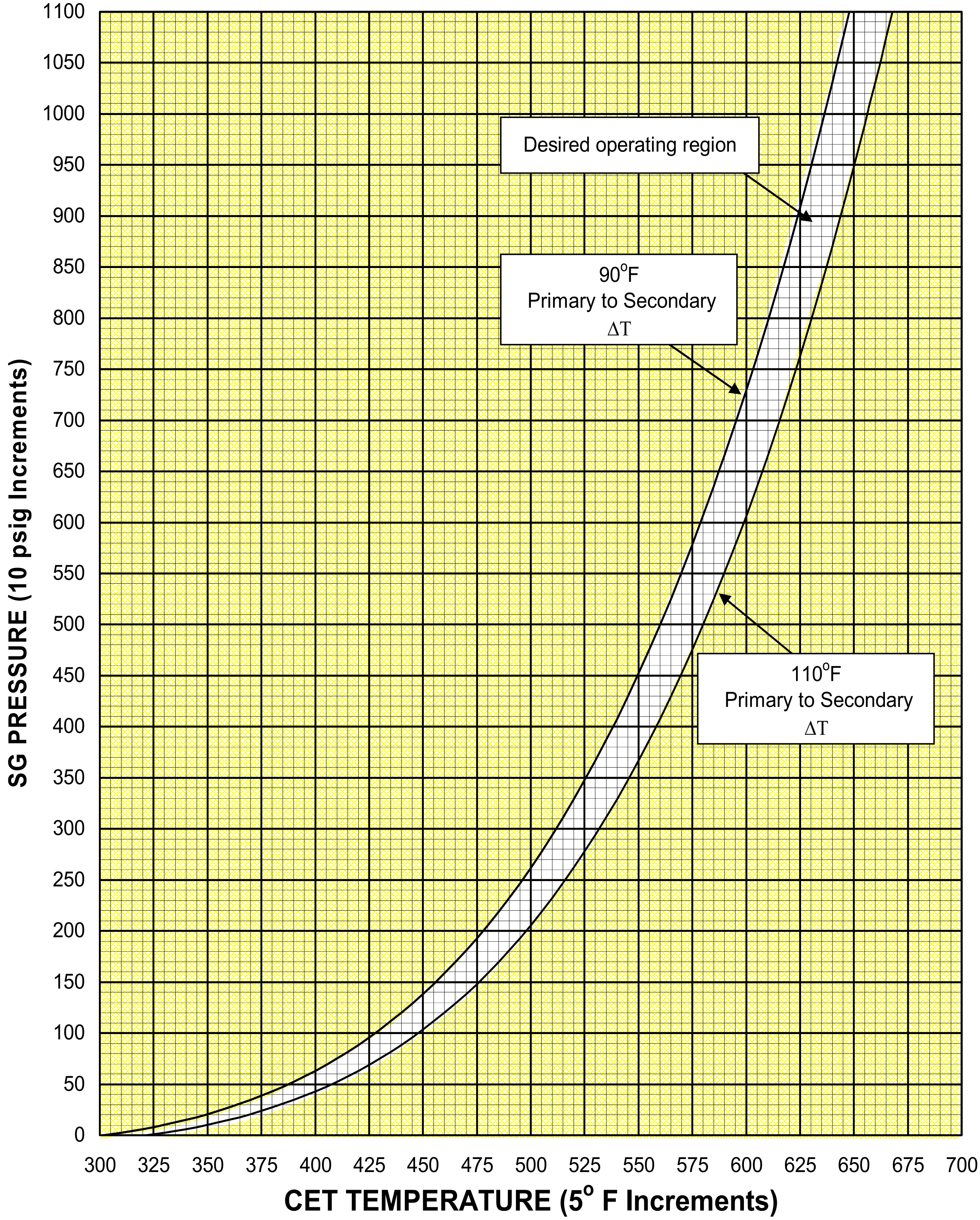


FIGURE 6

SG Pressure to Establish 90° to 110°F Primary to Secondary ΔT



ATTACHMENT L

REACTOR MANEUVERING RECOMMENDATIONS

Reactor Engineering personnel may be consulted as necessary for further recommendations not covered in this attachment. Power maneuvers can be performed using rods, boration and dilution.

1.0 POWER ESCALATION

During a startup with a significant xenon concentration in the core, imbalance will be positive. Group 7 rods may be inserted to ~50% withdrawn during the startup to aid in imbalance control to ~40% FP.

1.1 Power Escalation Limits

- Table L1 shows the maximum rates for power escalation.
- For power histories not listed below, the “Below 50% power for less than 96 hours" rates may be used.
- Power levels listed in the table below assume 4-RCP operation.
  For 3-RCP operation, use 75% of the listed power level bands
  (example: 0%-40% becomes 0%-30%, 40%-60% becomes 30%-45%, etc.)

TABLE L1 – POWER ESCALATION LIMITS					
Power History	0%-40% Power	40%-60% Power	60%-90% Power	90%-98% Power	98%-100% Power
Below 50% power for less than 96 hours (1)	≤30%/hr	≤30%/hr	≤30%/hr	≤30%/hr	≤5%/hr
Below 50% power for more than 96 hours (1)	≤30%/hr	≤30%/hr	≤15%/hr	≤5%/hr	≤5%/hr
Initial startup after refueling	≤30%/hr	≤5%/hr	≤5%/hr	≤3%/hr	≤3%/hr
Dropped rod recovery less than 8 hrs after rod drop	≤30%/hr	≤30%/hr	≤30%/hr	≤30%/hr	≤5%/hr
Dropped rod recovery 8 to 24 hrs after rod drop	≤30%/hr	≤30%/hr	≤15%/hr	≤5%/hr	≤5%/hr
Dropped rod recovery greater than 24 hrs after rod drop	≤3%/hr	≤3%/hr	≤3%/hr	≤3%/hr	≤3%/hr
Note 1: 96 hours applies only to time the Reactor is critical and below 50%. Subcritical time is not included in the 96 hours.					

ATTACHMENT L

1.2 Step Changes in Power

- Although the power escalation rates of Table L1 are expressed in % full power per hour, the operator should strive to control the reactor power change at a smooth and constant rate per minute as is practical. For example, if the allowed power escalation rate is 30%FP/hr and power is to be raised 15%, the operator should strive to accomplish the power change at a constant rate over at least 30 minutes.

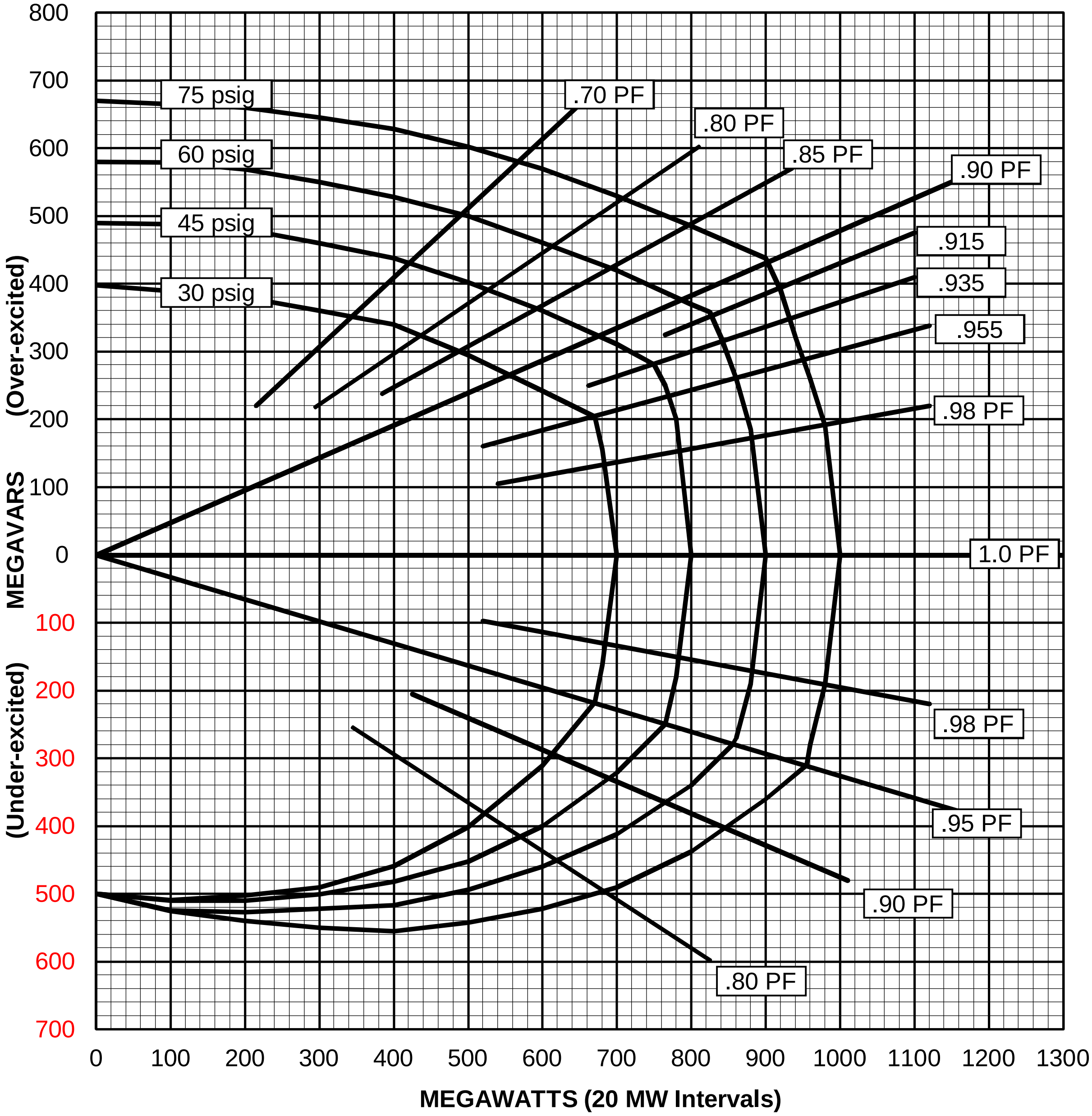
- Step changes in reactor power are measured in any continuous time period of five minutes or less. Step changes in power that meet the Step Change Definition of Table L2 below must be followed by a 10-minute hold at constant power level before further power escalation. Although step changes are allowed as defined, they should be minimized.

TABLE L2 – STEP CHANGE DEFINITION	
Allowable Rate of Escalation from Table L1	Step Change Requiring a 10-minute hold
30%/hour	Power escalation of >5% in ≤5 minutes
15%/hour	Power escalation of >3.75% in ≤5 minutes
5%/hour	Power escalation of >1.25% in ≤5 minutes
3%/hour	Power escalation of >0.75% in ≤5 minutes

ATTACHMENT N

PAGE 1 OF 1

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

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NRC INITIAL LICENSE EXAMINATION REFERENCE MATERIAL

<u>SRO</u>

3.3 INSTRUMENTATION

3.3.15 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.15 The PAM instrumentation for each Function in Table 3.3.15-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one required channel inoperable.	A.1 Restore required channel to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action to prepare and submit a Special Report.	Immediately
C. One or more Functions with two required channels inoperable.	C.1 Restore one channel to OPERABLE status.	7 days
D. Required Action and associated Completion Time of Condition C not met.	D.1 Enter the Condition referenced in Table 3.3.15-1 for the channel.	Immediately

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. As required by Required Action D.1 and referenced in Table 3.3.15-1.	E.1 Be in MODE 3. <u>AND</u> E.2 Be in MODE 4.	6 hours 12 hours
F. As required by Required Action D.1 and referenced in Table 3.3.15-1.	F.1 Initiate action to prepare and submit a Special Report.	Immediately

SURVEILLANCE REQUIREMENTS

---NOTE---
These SRs apply to each PAM instrumentation Function in Table 3.3.15-1.

SURVEILLANCE		FREQUENCY
SR 3.3.15.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	31 days
SR 3.3.15.2	--------------------------NOTE------------------------------------ Neutron detectors are excluded from CHANNEL CALIBRATION. --- Perform CHANNEL CALIBRATION.	 18 months

Table 3.3.15-1
Post Accident Monitoring Instrumentation

	FUNCTION	REQUIRED CHANNELS	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1
1.	Wide Range Neutron Flux	2	E
2.	RCS Hot Leg Temperature	2	E
3.	RCS Hot Leg Level	2	F
4.	RCS Pressure (Wide Range)	2	E
5.	Reactor Vessel Water Level	2	F
6.	Reactor Building Water Level (Wide Range)	2	E
7.	Reactor Building Pressure (Wide Range)	2	E
8.	Penetration Flow Path Automatic Reactor Building Isolation Valve Position	2 per penetration flow path[(a)(b)]	E
9.	Reactor Building Area Radiation (High Range)	2	F
10.	Deleted		
11.	Pressurizer Level	2	E
12.	a. SG “A” Water Level – Low Range	2	E
	b. SG “B” Water Level – Low Range	2	E
	c. SG “A” Water Level – High Range	2	E
	d. SG “B” Water Level – High Range	2	E
13.	a. SG “A” Pressure	2	E
	b. SG “B” Pressure	2	E
14.	Condensate Storage Tank Level	2	E
15.	Borated Water Storage Tank Level	2	E
16.	Core Exit Temperature (CETs per quadrant)	2	E
17.	a. Emergency Feedwater Flow to SG “A”	2	E
	b. Emergency Feedwater Flow to SG “B”	2	E
18.	High Pressure Injection Flow	2	E
19.	Low Pressure Injection Flow	2	E
20.	Reactor Building Spray Flow	2	E

(a) Not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

(b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS - Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with Reactor Coolant System (RCS) temperature > 350 °F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more trains inoperable.	A.1 Restore train(s) to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Reduce RCS temperature to ≤ 350°F.	6 hours 12 hours
C. Less than 100% of the ECCS flow equivalent to a single OPERABLE train available.	C.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.2.2	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE		FREQUENCY
SR 3.5.2.3	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.5.2.4	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	18 months
SR 3.5.2.5	Verify, by visual inspection, each ECCS train reactor building sump suction inlet is not restricted by debris and screens show no evidence of structural distress or abnormal corrosion.	18 months

3.6 REACTOR BUILDING SYSTEMS

3.6.3 Reactor Building Isolation Valves

LCO 3.6.3 Each reactor building isolation valve shall be OPERABLE.

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

ACTIONS

--NOTES--

1. Penetration flow paths, except for purge valve penetration flow paths, may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for system(s) made inoperable by reactor building isolation valves.

4. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Reactor Building," when isolation valve leakage results in exceeding the overall reactor building leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. --------------NOTE------------- Only applicable to penetration flow paths with two reactor building isolation valves. ----------------------------------- One or more penetration flow paths with one reactor building isolation valve inoperable.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. AND	48 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 --------------NOTES------------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. ------------------------------------ Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside the reactor building <u>AND</u> Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside the reactor building
B. -------------NOTE-------------- Only applicable to penetration flow paths with two reactor building isolation valves. ------------------------------------ One or more penetration flow paths with two reactor building isolation valves inoperable.	B.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. -------------NOTE-------------- Only applicable to penetration flow paths with only one reactor building isolation valve and a closed system. ------------------------------------ One or more penetration flow paths with one reactor building isolation valve inoperable.	C.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange. <u>AND</u> C.2 --------------NOTES------------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. ------------------------------------ Verify the affected penetration flow path is isolated.	72 hours Once per 31 days
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. <u>AND</u> D.2 ---------------NOTE-------------- LCO 3.0.4.a is not applicable when entering Mode 4. ------------------------------------ Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1	Verify each reactor building purge isolation valve is closed.	31 days
SR 3.6.3.2	------------------------------NOTE--------------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. --- Verify each reactor building isolation manual valve and blind flange that is located outside the reactor building and not locked, sealed, or otherwise secured, and is required to be closed during accident conditions is closed, except for reactor building isolation valves that are open under administrative controls.	31 days
SR 3.6.3.3	------------------------------NOTE--------------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. --- Verify each reactor building isolation manual valve and blind flange that is located inside the reactor building and not locked, sealed, or otherwise secured, and required to be closed during accident conditions is closed, except for reactor building isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days
SR 3.6.3.4	Verify the isolation time of each automatic power operated reactor building isolation valve is within limits.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.3.5	Verify each automatic reactor building isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	18 months

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources - Operating

LCO 3.8.4 Both DC electrical power subsystems shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One DC electrical power subsystem inoperable.	A.1 Restore DC electrical power subsystem to OPERABLE status.	8 hours
B. Required Action and Associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 ---------------NOTE--------------- LCO 3.0.4.a is not applicable when entering Mode 4. -------------------------------------- Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage is greater than or equal to the minimum established float voltage.	7 days
SR 3.8.4.2	Verify each battery charger supplies ≥ 300 amps at greater than or equal to the minimum established float voltage for ≥ 8 hours. <u>OR</u> Verify each battery charger can recharge the battery to the fully charged state within 24 hours while supplying the largest combined demands of the various continuous steady state loads, after a battery discharge to the bounding design basis event discharge state.	18 months
SR 3.8.4.3	---------------------------------NOTE------------------------------ This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this SR. --- Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test or a modified performance discharge test.	18 months

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Battery Parameters

LCO 3.8.6 Battery parameters for the Train A and Train B electrical power subsystem batteries shall be within limits.

APPLICABILITY: When associated DC electrical power subsystems are required to be OPERABLE.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One battery with one or more battery cells float voltage < 2.07 V.	A.1 Perform SR 3.8.4.1.	2 hours
	AND	
	A.2 Perform SR 3.8.6.1.	2 hours
	AND	
	A.3 Restore affected cell voltage ≥ 2.07 V.	24 hours
B. One battery with float current > 2 amps.	B.1 Perform SR 3.8.4.1	2 hours
	AND	
	B.2 Restore battery float current to ≤ 2 amps.	12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
------------------NOTE---------------- Required Action C.2 shall be completed if electrolyte level was below the top of the plates. --	--------------------NOTE------------------ Required Actions C.1 and C.2 are only applicable if electrolyte level was below the top of the plates. --	
C. One battery with one or more cells electrolyte level less than minimum established design limits.	C.1 Restore electrolyte level to above top of plates.	8 hours
	<u>AND</u>	
	C.2 Verify no evidence of leakage.	12 hours
	<u>AND</u>	
	C.3 Restore electrolyte level to greater than or equal to minimum established design limits.	31 days
D. One battery with pilot cell electrolyte temperature less than minimum established design limits.	D.1 Restore battery pilot cell temperature to greater than or equal to minimum established design limits.	12 hours
E. Two batteries with battery parameters not within limits.	E.1 Restore at least one battery to within limits.	2 hours
F. Required Actions and associated Completion Times of Condition A, B, C, D, or E not met. <u>OR</u> One battery with one or more battery cells float voltage < 2.07 V and float current > 2 amps.	F.1 Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.6.1	------------------------------NOTE------------------------------ Not required to be met when battery terminal voltage is less than the minimum established float voltage of SR 3.8.4.1. -- Verify each battery float current is ≤ 2 amps.	7 days
SR 3.8.6.2	Verify each battery pilot cell float voltage is ≥ 2.07 V.	31 days
SR 3.8.6.3	Verify each battery connected cell electrolyte level is greater than or equal to minimum established design limits.	31 days
SR 3.8.6.4	Verify each battery pilot cell temperature is greater than or equal to minimum established design limits.	31 days
SR 3.8.6.5	Verify each battery connected cell float voltage is ≥ 2.07 V.	92 days
SR 3.8.6.6	------------------------------NOTE------------------------------ This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this SR. -- Verify battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.	60 months <u>AND</u> 12 months when battery shows degradation, or has reached 85% of the expected life with capacity < 100% of manufacturer's rating <u>AND</u> 24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating

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. <u>AND</u>	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

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. <u>OR</u> A.1.2 Verify FUNCTIONAL smoke and/or heat detection for the affected area with control room alarm. <u>AND</u>	1 hour 1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 Establish backup fire suppression equipment for the affected area.	1 hour
	<u>AND</u>	
	A.3 Restore the non-functional Fire Suppression Sprinkler System to FUNCTIONAL status.	14 days
B. Required Actions and associated Completion Time for Condition A not met.	B.1 Initiate a condition report.	Immediately
	<u>AND</u>	
	B.2 Determine any limitations for continued operation of the plant.	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

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.

5.0 Generic Control Room Activity Values (Unit 1)

Minimal Impact	
Activity < 15 minutes (i.e. Starting P-6A Elec Firewater Pump, starting P-66 BW Recirc Pump, Aligning HP LD sample)	
Walkdowns and inspections (i.e. Fire extinguisher checks, SCBA inspections, etc.)	
Print or procedure updates	
CSG minor computer maintenance	
Evacuation alarm and pager test	
NI Cal no adjustment	
Placing / removing control room on emergency recirc (when Unit 2 has the lead)	
CBO Weekly Logs	
Low Impact	
Chart recorder maintenance	**0/1**
Sigma / Dixson maintenance	**0/1**
Maintenance that causes 1-2 control room alarm(s)	**0/1**
Cabinet PMs	**0/1**
Operations PMT valve strokes	**0/1**
PZR Water Space samples, CFT samples	**1**
Main Steam Rad monitor monthly	**1**
Containment Hi Range Rad monitor monthly	**1**
RB Leak Detector paper change out	**1**
Control Room Rad Monitor monthly	**1**
Starting/Stopping RB Purge	**1**
Refueling/Defueling	**1**
Medium Impact	
Fire Drill	**2**
RPS Monthly Test (non-trip initiator portion)	**2**
EFIC Monthly Test	**2**
Maintenance that causes >2 control room alarm(s)	**2**
Selecting alternate channels/instruments for control	**2**
I&C DROPS/AMSAC calibrations	**2**
Operations RPS bypass operations	**2**
RCS Heatup	**2**
RCS Cooldown in Mode 5	**2**
Starting/Stopping ES pumps (EFW, SW, HPI, etc.)	**2**

ATTACHMENT M

5.0 Generic Control Room Activity Values (Unit 1)(continued)

High Impact	
Operations pump surveillances	**3**
RPS Quarterly (Breaker trip test)	**3**
EDG Surveillance	**3**
Placing / removing control room on emergency recirc (Unit 1 has lead)	**3**
Fire Panel maintenance	**3**
Fill Fuel Transfer Canal or RCS	**3**
Shifting Electrical Loads	**3**
RCS Cooldown to Mode 5	**3**
RCS Dilution	**3**
Starting or Stopping DHR or RCPs	**3**
Undervoltage Monitor Relay Testing	**3**
NI Calibration with an adjustment	**4**
ICS to Auto/Manual	**4**
CRD exercise	**4**
I &C Semi-annual DROPS/AMSAC Testing	**4**
TV/GV Stroke Testing	**4**
Plant maneuver	**5**
Physics Testing	**5**
Reactor Startup	**8**
Draining the RCS with fuel in vessel	**8**
Integrated ES Test	**8**

ATTACHMENT 6
PROTECTIVE ACTION RECOMMENDATIONS (PARs)
FOR GENERAL EMERGENCY

Table of Contents

PAGE No.

ATTACHMENT 6
PROTECTIVE ACTION RECOMMENDATIONS (PARs)
FOR GENERAL EMERGENCY

Discussion

This attachment provides instructions for the assessment and initiation of Protective Action Recommendations (PARs) following the declaration of a General Emergency classification. Offsite response agencies shall be notified of Protective Action Recommendation within 15 minutes. Revisions to Protective Action Recommendations may be based upon:

- Current plant conditions
- Projected offsite dose assessment
- Forecasted/actual wind shifts

Evacuation is the preferred method for protecting the public within the ANO 10-mile Emergency Planning Zone (EPZ) as a result of a radiological emergency event at ANO. However, some circumstances may warrant a protective action of “shelter” when evacuation cannot be performed due to impediments and/or severe weather conditions. Individuals responsible for determining PARs at ANO should consider all circumstances when developing protective actions.

In the event of a “shelter” PAR, coordinate with ADH to develop a plan for transitioning out of this protective action as soon as possible. This is especially of concern during weather extremes since the public is advised to shut down ventilation systems.

The Arkansas Department of Health (ADH) will be notified of the ANO protective action recommendations and are responsible for determining and issuing a Protective Action Advisory (PAA) to the County Judges (Conway, Johnson, Logan, Pope and Yell counties). Arkansas law places the responsibility for issuing protective actions to the public with the County Judges which will have both a Protective Action Recommendation and a Protective Action Advisory available for decision making. At a General Emergency classification, the Arkansas Department of Health, at a minimum, will issue a default Protective Action Advisory of “evacuate a 5-mile radius and evacuate 5-10 miles downwind and the remaining EPZ to remain indoors and listen to emergency broadcasts”. At a General Emergency classification, ANO, at a minimum, will issue a default Protective Action Recommendation (PAR) of “evacuate a 2-mile radius and evacuate 2-5 miles downwind and the remaining EPZ to remain indoors and listen to emergency broadcasts”. The ADH Protective Action Advisory encompasses a larger area than that recommended by federal guidance and the ANO General Emergency classification PAR. Be aware of this difference between the ANO protective action recommendation and the ADH protective action advisory should a question arise. ANO PARs meet all of the EPA/NRC recommended regulatory guidance and are consistent with the rest of the nuclear industry.

Guidance Involving Wind Shifts within the 10-mile EPZ

If wind shifts are occurring or are predicted to occur within the 10-mile EPZ, guidance is provided on PAR No. 6 within this attachment.

Use of the PAR Flowchart in Attachment 6

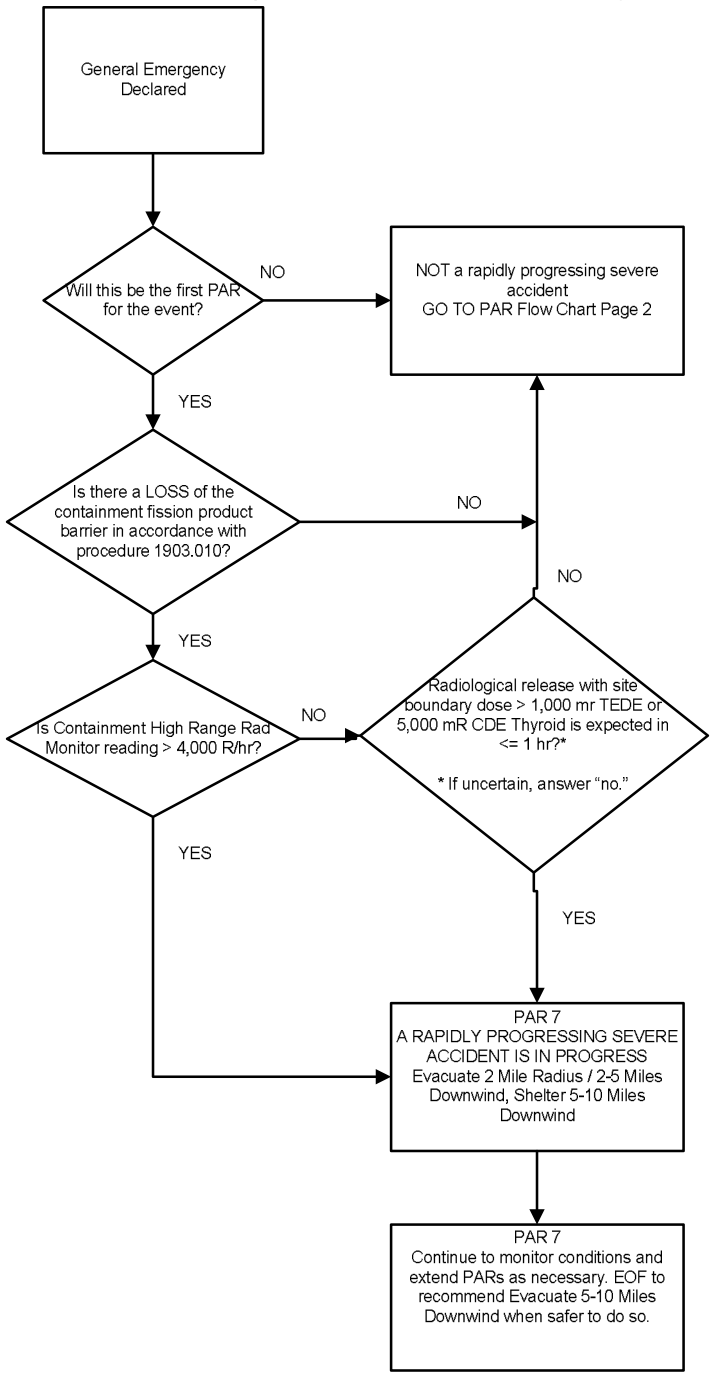
A PAR Flowchart is included on Pages 3 and 4 of this attachment. This flowchart should be used initially starting on Page 3 and at the beginning of each subsequent PAR evaluation (page 4) to help determine the correct PAR to issue based on plant conditions, release status, evacuation impediments and offsite dose assessment.

ATTACHMENT 6

PROTECTIVE ACTION RECOMMENDATIONS (PARs) FOR GENERAL EMERGENCY

PAR Flow Chart – Page 1

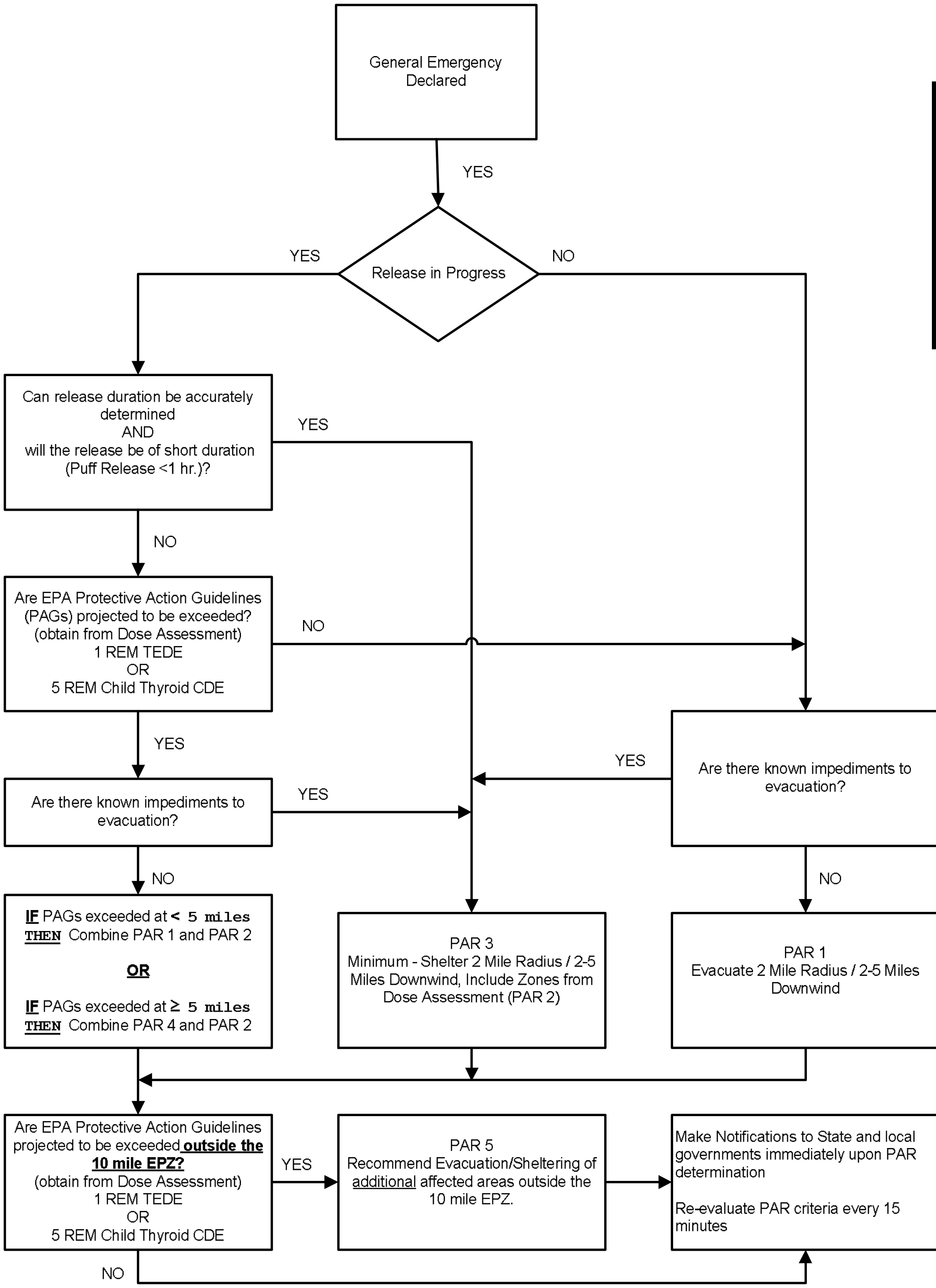
(A Guide for Determining PARs)



ATTACHMENT 6
PROTECTIVE ACTION RECOMMENDATIONS (PARs)
FOR GENERAL EMERGENCY

PAR Flow Chart – Page 2

(A Guide for Determining PARs)



NOTE

<u>IF</u> Wind Shifts are forecasted, <u>THEN</u> PAR No. 6 shall be reviewed to assist with final PAR determination.

ATTACHMENT 6
PROTECTIVE ACTION RECOMMENDATIONS (PARs)
FOR GENERAL EMERGENCY

PAR No. 1
EVACUATE

NOTE
State and local governments must be notified within **15 minutes** of PARs or changes to PARs using Form 1903.011-Y.

1. Entry Conditions

   General Emergency Declared

2. Recommend the following Protective Action Recommendations:

   Recommend **evacuation** of 2 mile radius and 2-5 miles downwind. Recommend the remainder of the 10 mile EPZ to go indoors and listen to the emergency broadcast for this event. Include any previously evacuated zones with this PAR. **DO NOT** change any previously evacuated zones to "shelter" or "go indoors" on this PAR.

   Determine the affected zones for the PAR from the chart given below.

Wind Direction (from)	Evacuate Zones	Zones "to go indoors"
348.75 to 11.25	G U	H I J K L M N O P Q R S T
11.25 to 33.75	G R U	H I J K L M N O P Q S T
33.75 to 56.25	G R U	H I J K L M N O P Q S T
56.25 to 78.75	G R U	H I J K L M N O P Q S T
78.75 to 101.25	G N O R	H I J K L M P Q S T U
101.25 to 123.75	G N O R	H I J K L M P Q S T U
123.75 to 146.25	G K N O	H I J L M P Q R S T U
146.25 to 168.75	G K N O	H I J L M P Q R S T U
168.75 to 191.25	G K N	H I J L M O P Q R S T U
191.25 to 213.75	G K	H I J L M N O P Q R S T U
213.75 to 236.25	G K	H I J L M N O P Q R S T U
236.25 to 258.75	G H K	I J L M N O P Q R S T U
258.75 to 281.25	G H K	I J L M N O P Q R S T U
281.25 to 303.75	G H K U	I J L M N O P Q R S T
303.75 to 326.25	G H U	I J K L M N O P Q R S T
326.25 to 348.75	G H U	I J K L M N O P Q R S T

3. Reassess PARs every **15 minutes** until downgrade or recovery phase is entered.

ATTACHMENT 6
PROTECTIVE ACTION RECOMMENDATIONS (PARs)
FOR GENERAL EMERGENCY

PAR No. 2
EVACUATE

NOTE
State and local governments must be notified within **15 minutes** of PARs or changes to PARs using Form 1903.011-Y.

1. Entry Conditions

   General Emergency declared

   AND

   Dose Assessment projects EPA Protective Action Guidelines (PAGs) exceeded

   1 Rem TEDE OR 5 Rem Child Thyroid CDE

2. Recommend the following Protective Action Recommendation:

NOTE
If there are known impediments to evacuation, then consider "sheltering" of the affected zones versus evacuation.

2.1 **IF** PAGs are exceeded at ≥ 5 miles
THEN recommend the following PAR:
- **EVACUATE** zones from **PAR 4**
- **EVACUATE** any additional [1]ZONES projected by dose assessment to exceed the EPA PAGs (obtain from dose assessment).
- Remainder of the 10 mile EPZ to go indoors and listen to the Emergency Broadcasts

2.2 **IF** PAGs are exceeded at < 5 miles,
THEN recommend the following PAR:
- **EVACUATE** zones from **PAR 1**
- **EVACUATE** any additional [1]ZONES projected by dose assessment to exceed the EPA PAGs (obtain from dose assessment).
- Remainder of the 10 mile EPZ to go indoors and listen to the Emergency Broadcasts

3. Include any previously evacuated zones on this PAR. **DO NOT** change any previously evacuated zones to "shelter" or "go indoors" on this PAR.

4. Reassess PARs every **15 minutes** until downgrade or recovery phase is entered.

[1]Dose assessment PARs will be initially provided by the Initial Dose Assessor in the Control Room. When the Dose Assessors becomes operational in the EOF, they will provide this information.

ATTACHMENT 6
PROTECTIVE ACTION RECOMMENDATIONS (PARs)
FOR GENERAL EMERGENCY

PAR No. 3
Shelter

NOTE
State and local governments must be notified within **15 minutes** of PARs or changes to PARs using Form 1903.011-Y.

1. Entry Conditions

General Emergency declared
AND
Known Impediments to Evacuation exist
OR
Offsite Release is a Puff Release (< 1 hour in duration)

2. Recommend the following Protective Action Recommendation:

Recommend **sheltering** a 2 mile radius and 2-5 miles downwind. Recommend the remainder of the 10-mile EPZ to go indoors and listen to the emergency broadcast for this event. Determine the affected zones for the PAR from the chart given below. **Include any zones recommended for evacuation by Dose Assessment.** **DO NOT** change any previously evacuated zones to "shelter" or "go indoors" on this PAR.

Determine the affected zones for the PAR from the chart given below.

Wind Direction (from)	Shelter Zones	Zones "to go indoors"
348.75 to 11.25	G U	H I J K L M N O P Q R S T
11.25 to 33.75	G R U	H I J K L M N O P Q S T
33.75 to 56.25	G R U	H I J K L M N O P Q S T
56.25 to 78.75	G R U	H I J K L M N O P Q S T
78.75 to 101.25	G N O R	H I J K L M P Q S T U
101.25 to 123.75	G N O R	H I J K L M P Q S T U
123.75 to 146.25	G K N O	H I J L M P Q R S T U
146.25 to 168.75	G K N O	H I J L M P Q R S T U
168.75 to 191.25	G K N	H I J L M O P Q R S T U
191.25 to 213.75	G K	H I J L M N O P Q R S T U
213.75 to 236.25	G K	H I J L M N O P Q R S T U
236.25 to 258.75	G H K	I J L M N O P Q R S T U
258.75 to 281.25	G H K	I J L M N O P Q R S T U
281.25 to 303.75	G H K U	I J L M N O P Q R S T
303.75 to 326.25	G H U	I J K L M N O P Q R S T
326.25 to 348.75	G H U	I J K L M N O P Q R S T

3. PARs must be reassessed every **15 minutes** until downgrade or recovery phase is entered.

ATTACHMENT 6
PROTECTIVE ACTION RECOMMENDATIONS (PARs)
FOR GENERAL EMERGENCY

PAR No. 4
EVACUATE

NOTE

State and local governments must be notified within **15 minutes** of PARs or changes to PARs using Form 1903.011-Y.

1. Entry Conditions

   General Emergency Declared
   AND
   EPA Protective Action Guidelines (PAGs) are projected to be exceeded **5-10 miles downwind**.

   1 Rem TEDE
   OR
   5 Rem Child Thyroid CDE

2. Recommend the following Protective Action Recommendation:

   Recommend **evacuation** of 2 mile radius and 2-10 miles downwind. Recommend that the remainder of the 10-mile EPZ go indoors and listen to the emergency broadcasts for this event. Include any previously evacuated zones with this PAR. **DO NOT** change any previously evacuated zones to "shelter" or "go indoors" on this PAR.

   Determine the affected zones for the PAR from the chart given below.

Wind Direction (from)	Evacuate Zones	Zones "to go indoors"
348.75 to 11.25	G U S T	H I J K L M N O P Q R
11.25 to 33.75	G Q R S U	H I J K L M N O P T
33.75 to 56.25	G Q R S U	H I J K L M N O P T
56.25 to 78.75	G Q R S U	H I J K L M N O P T
78.75 to 101.25	G N O P Q R	H I J K L M S T U
101.25 to 123.75	G N O P Q R	H I J K L M S T U
123.75 to 146.25	G K M N O P	H I J L Q R S T U
146.25 to 168.75	G K M N O P	H I J L Q R S T U
168.75 to 191.25	G K M N O P	H I J L Q R S T U
191.25 to 213.75	G K L M	H I J N O P Q R S T U
213.75 to 236.25	G J K L M	H I N O P Q R S T U
236.25 to 258.75	G H I J K L M	N O P Q R S T U
258.75 to 281.25	G H I J K L	M N O P Q R S T U
281.25 to 303.75	G H I J K U	L M N O P Q R S T
303.75 to 326.25	G H I J S T U	K L M N O P Q R
326.25 to 348.75	G H I S T U	J K L M N O P Q R

3. Reassess PARs every **15 minutes** until downgrade or recovery phase is entered.

ATTACHMENT 6
PROTECTIVE ACTION RECOMMENDATIONS (PARs)
FOR GENERAL EMERGENCY

PAR No. 5
Outside the 10 Mile EPZ

NOTE
Protective Action Recommendations beyond the 10-mile EPZ shall be coordinated with State and local government officials.

1. Entry Conditions

General Emergency declared
AND
EPA Protective Action Guidelines (PAGs) are projected to be exceeded **outside the 10-mile EPZ**.

1 Rem TEDE
OR
5 Rem Child Thyroid CDE

2. Recommend the following Protective Action Recommendation:

Recommend **evacuation** of the affected areas. If known impediments to evacuation exist consider sheltering of the affected area.

Use dose assessment personnel to determine the affected sector(s) and downwind distances and then use the chart below to determine the affected area(s) to evacuate.

Affected Sector(s)	Evacuate/Shelter Sectors	Distance from Site
1	16, 1, 2	10 miles to (Determined by Dose Assessment)
2	1, 2, 3	10 miles to (Determined by Dose Assessment)
3	2, 3, 4	10 miles to (Determined by Dose Assessment)
4	3, 4, 5	10 miles to (Determined by Dose Assessment)
5	4, 5, 6	10 miles to (Determined by Dose Assessment)
6	5, 6, 7	10 miles to (Determined by Dose Assessment)
7	6, 7, 8	10 miles to (Determined by Dose Assessment)
8	7, 8, 9	10 miles to (Determined by Dose Assessment)
9	8, 9, 10	10 miles to (Determined by Dose Assessment)
10	9, 10, 11	10 miles to (Determined by Dose Assessment)
11	10, 11, 12	10 miles to (Determined by Dose Assessment)
12	11, 12, 13	10 miles to (Determined by Dose Assessment)
13	12, 13, 14	10 miles to (Determined by Dose Assessment)
14	13, 14, 15	10 miles to (Determined by Dose Assessment)
15	14, 15, 16	10 miles to (Determined by Dose Assessment)
16	15, 16, 1	10 miles to (Determined by Dose Assessment)

3. Reassess PARs every **15 minutes** until downgrade or recovery phase is entered.

ATTACHMENT 6
PROTECTIVE ACTION RECOMMENDATIONS (PARs)
FOR GENERAL EMERGENCY

PAR No. 6
Wind Shift PAR Determination

NOTE
A wind shift is defined as any change in 15-minute averaged wind direction that affects new offsite protective action zones that are 2-5 or 5-10 miles downwind.

1. Entry Conditions

   General Emergency Declared

   AND

   Previous PAR has been issued

   AND

   Actual/Forecasted Wind Shift

2. **IF** the conditions in 2.1 through 2.3 below are met,
   THEN revise PARs based on dose assessment results only. Go to Step 4.

   2.1 Plant conditions are well understood and changes can be reasonably predicted.

   2.2 Radiological releases have a high degree of predictability in terms of isotopic composition, release pathway, and release rate.

   2.3 Meteorological conditions for the projected duration of the release are well understood.

3. **IF** the conditions described in 2.1 through 2.3 above are not met
   AND an actual wind shift occurs **OR** is forecasted to occur within 6 hours,
   THEN

   STEP 1 – Wind Direction Transition Area: Evacuate any additional zones projected to exceed the EPA PAGs (obtain from dose assessment).

   STEP 2 – Final Wind Direction: Revise the current PAR to include any downwind zones using the table below. If conditions warrant, evacuation out to 10 miles may be necessary. Refer to PAR 5, as needed, to determine those areas located outside of the 10-mile EPZ.

Wind Direction (from)	2-5 Miles Downwind Zones	5-10 Miles Downwind Zones
348.75 to 11.25	U	S T
11.25 to 33.75	R U	Q S
33.75 to 56.25	R U	Q S
56.25 to 78.75	R U	Q S
78.75 to 101.25	N O R	P Q
101.25 to 123.75	N O R	P Q
123.75 to 146.25	K N O	M P
146.25 to 168.75	K N O	M P
168.75 to 191.25	K N	M P
191.25 to 213.75	K	L M
213.75 to 236.25	K	J L M
236.25 to 258.75	H K	I J L M
258.75 to 281.25	H K	I J L
281.25 to 303.75	H K U	I J
303.75 to 326.25	H U	I J S T
326.25 to 348.75	H U	I S T

4. Reassess PARs every **15 minutes** until downgrade or recovery phase is entered.

ATTACHMENT 6
PROTECTIVE ACTION RECOMMENDATIONS (PARs)
FOR GENERAL EMERGENCY

PAR No. 7
EVACUATE

NOTE
State and local governments must be notified within **15 minutes** of PARs or changes to PARs using Form 1903.011-Y.

1. Entry Conditions

   General Emergency Declared
   AND
   A rapidly progressing severe accident is in progress

2. Recommend the following Protective Action Recommendation:

   Recommend **evacuation** of 2 mile radius and 2-5 miles downwind. Recommend shelter for 5-10 miles downwind. Recommend that the remainder of the 10-mile EPZ go indoors and listen to the emergency broadcasts for this event.

   Determine the affected zones for the PAR from the chart given below.

Wind Direction (from)	Evacuate Zones	Shelter Zones	Zones “to go indoors”
348.75 to 11.25	G U	S T	H I J K L M N O P Q R
11.25 to 33.75	G R U	Q S	H I J K L M N O P T
33.75 to 56.25	G R U	Q S	H I J K L M N O P T
56.25 to 78.75	G R U	Q S	H I J K L M N O P T
78.75 to 101.25	G N O R	P Q	H I J K L M S T U
101.25 to 123.75	G N O R	P Q	H I J K L M S T U
123.75 to 146.25	G K N O	M P	H I J L Q R S T U
146.25 to 168.75	G K N O	M P	H I J L Q R S T U
168.75 to 191.25	G K N O	M P	H I J L Q R S T U
191.25 to 213.75	G K	L M	H I J N O P Q R S T U
213.75 to 236.25	G K	J L M	H I N O P Q R S T U
236.25 to 258.75	G H K	I J L M	N O P Q R S T U
258.75 to 281.25	G H K	I J L	M N O P Q R S T U
281.25 to 303.75	G H K U	I J	L M N O P Q R S T
303.75 to 326.25	G H U	I J S T	K L M N O P Q R
326.25 to 348.75	G H U	I S T	J K L M N O P Q R

NOTE
Changing the recommendation for areas 5-10 miles downwind from shelter to evacuate is the responsibility of the EOF and will not be performed in the Control Room.

3. A recommendation of evacuation of 5-10 miles downwind should only be considered when safer to do so (when the EOF and state and local EOCs are staffed and operational AND the release source term has significantly reduced (i.e., a reduction of 25% or more))

   a. A change in recommendation may be considered based on a change in wind direction with site wind variability taken into account.

   b. The decision to change the recommendation relies ultimately upon the judgment of decision makers at the time of the event.

4. Reassess PARs every **15 minutes** until downgrade or recovery phase is entered.

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

GENERAL EMERGENCY

CG1 5 6

Loss of RCS / reactor vessel inventory affecting fuel clad integrity with containment challenged

<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 will likely exceed the applicable time.

1. a. Core exit thermocouples indicate superheat for ≥ 30 minutes.

   <u>AND</u>

   b. Any of the following containment challenge indications:

   - CONTAINMENT CLOSURE not established
   - Explosive mixture inside containment
   - UNPLANNED rise in containment pressure

   <u>OR</u>

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:

SITE AREA EMERGENCY

CS1 5 6

Loss of RCS / reactor vessel inventory affecting core decay heat removal capability

<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 will likely exceed the applicable time.

1. With CONTAINMENT CLOSURE **<u>not</u>** 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

   <u>OR</u>

2. With CONTAINMENT CLOSURE established, core exit thermocouples indicate superheat.

   <u>OR</u>

ALERT

CA1 5 6

Loss of RCS / reactor vessel inventory

<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 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>

UNUSUAL EVENT

CU1 5

RCS leakage

<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 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 6

UNPLANNED loss of RCS / reactor vessel Inventory

<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 will likely exceed the applicable time.

1. UNPLANNED RCS / reactor vessel level drop as indicated by either of the following:

<table>
<tr><th>GENERAL EMERGENCY</th><th>SITE AREA EMERGENCY</th><th>ALERT</th><th>UNUSUAL EVENT</th></tr>
<tr><td colspan="4">COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION – Loss of RCS / Reactor Vessel Inventory</td></tr>
<tr>
<td>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
<u>AND</u>
b. Any of the following containment challenge indications:
• CONTAINMENT CLOSURE not established
• Explosive mixture inside containment
• UNPLANNED rise in containment pressure</td>
<td>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</td>
<td>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.</td>
<td>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.
<u>OR</u>
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.
<u>OR</u>
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.</td>
</tr>
</table>

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 **<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 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 **<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 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. **<u>AND</u>** 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 **<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 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 **<u>Emergency Action Level(s):</u>** 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				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

TAB E

Independent Spent Fuel Storage Installation (ISFSI) Malfunction

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
ISFSI MALFUNCTION – Cask Damage			
			E-HU1 1 2 3 4 5 6 D **Note:** Security Events are bounded by the Hazards EALs. Damage to a loaded cask CONFINEMENT BOUNDARY **<u>Emergency Action Level(s):</u>** 1. Damage to a loaded cask CONFINEMENT BOUNDARY.

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. <u>Primary Coolant Activity Level (FCB1)</u>		**1. <u>RCS Leak Rate (RCB1)</u>**		**1. <u>Containment Pressure (CNB1)</u>**	
1. Coolant activity > 300 µCi/gm dose equivalent I-131 activity by Chemistry sample **<u>OR</u>** 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 **<u>OR</u>** 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 **<u>OR</u>** 2. Explosive mixture exists inside Containment **<u>OR</u>** 3. a. Containment Pressure > containment spray actuation setpoint **Unit 1:** 44.7 PSIA (30 PSIG) **Unit 2:** 23.3 PSIA **<u>AND</u>** 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 monitors 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
Emergency Action Level(s):	**Emergency Action Level(s):**	**Emergency Action Level(s):**	**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.	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.

<table>
<tr><th>GENERAL EMERGENCY</th><th>SITE AREA EMERGENCY</th><th>ALERT</th><th>UNUSUAL EVENT</th></tr>
<tr><td colspan="4">HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY – Control Room Evacuation</td></tr>
<tr><td></td><td>HS3 1 2 3 4 5 6 D

Control Room evacuation has been initiated and plant control cannot be established

<u>Emergency Action Level(s):</u>

1. a. Control Room evacuation has been initiated.

<u>AND</u>

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”</td><td>HA3 1 2 3 4 5 6 D

Control Room evacuation has been initiated

<u>Emergency Action Level(s):</u>

1. Alternate Shutdown procedure requires Control Room evacuation:
Unit 1: 1203.002, “Alternate Shutdown”
Unit 2: 2203.014, “Alternate Shutdown”</td><td></td></tr>
</table>

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 **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:	**[1]HU4** 1 2 3 4 5 6 D FIRE within the PROTECTED AREA not extinguished within 15 minutes of detection **OR** EXPLOSION within the PROTECTED AREA **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. FIRE in any **Table H1** structure or area not extinguished: a. within 15 minutes of Control Room notification **OR** b. within 15 minutes of [2]verification of a Control Room FIRE alarm (i.e. Alarm valid until disproved) **OR** 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>**

<table>
<tr><th>GENERAL EMERGENCY</th><th>SITE AREA EMERGENCY</th><th>ALERT</th><th>UNUSUAL EVENT</th></tr>
<tr><td colspan="4">HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY</td></tr>
<tr><td colspan="2" rowspan="2"></td><td colspan="2">Natural or Destructive Phenomena</td></tr>
<tr><td>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</td><td>HU6 (continued)

5. Lake Dardanelle level < 335 feet.

<u>OR</u>

6. Lake Dardanelle level > 345 feet.</td></tr>
</table>

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>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 • 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 Prolonged loss of all offsite and all onsite AC power to Vital 4.16 KV busses **<u>Emergency Action Level(s):</u>** 1. a. Loss of all offsite and all onsite AC power to Vital 4.16 KV busses. **<u>AND</u>** b. Either of the following: • Restoration of at least one Vital 4.16 KV bus in < 4 hours is not likely. **<u>OR</u>** • Continuing degradation of core cooling based on Fission Product Barrier monitoring as indicated by CETs ≥ 700 °F.	**SS1** 1 2 3 4 Loss of all offsite and all onsite AC power to Vital 4.16 KV busses ≥ 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. Loss of all offsite and all onsite AC power to Vital 4.16 KV busses ≥ 15 minutes.	**SA1** 1 2 3 4 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 **<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. AC power capability to Vital 4.16 KV busses reduced to a single power source ≥ 15 minutes. **<u>AND</u>** b. Any additional single power source failure will result in station blackout.	**SU1** 1 2 3 4 Loss of all offsite AC power to Vital 4.16 KV busses ≥ 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. 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 **<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. < 105 volts on all Vital DC busses ≥ 15 minutes.		

<table>
<tr><th>GENERAL EMERGENCY</th><th>SITE AREA EMERGENCY</th><th>ALERT</th><th>UNUSUAL EVENT</th></tr>
<tr><td colspan="4">SYSTEM MALFUNCTION – Loss of Annunciators</td></tr>
<tr>
<td></td>
<td>
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.
</td>
<td>
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
</td>
<td>
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.
</td>
</tr>
</table>

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 **<u>Emergency Action Level(s):</u>** 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 **<u>OR</u>** 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 **<u>OR</u>** • **Unit 1:** ≥ 60 uCi/gm Dose Equivalent I-131 **Unit 2:** > 60 uCi/gm Dose Equivalent I-131 **<u>OR</u>**

<table>
<tr><th>GENERAL EMERGENCY</th><th>SITE AREA EMERGENCY</th><th>ALERT</th><th>UNUSUAL EVENT</th></tr>
<tr><td colspan="4">SYSTEM MALFUNCTION – Fuel Clad Degradation</td></tr>
<tr><td></td><td></td><td></td><td>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</td></tr>
<tr><td colspan="4">SYSTEM MALFUNCTION – Inadvertant Criticality</td></tr>
<tr><td></td><td></td><td></td><td>SU10 3 4
Inadvertent criticality
<u>Emergency Action Level(s):</u>
1. An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.</td></tr>
<tr><td colspan="4">SYSTEM MALFUNCTION – Failure to Shutdown</td></tr>
<tr><td></td><td></td><td></td><td>SU11 1 2 3 4
Inability to reach required operating mode within Technical Specification limits
<u>Emergency Action Level(s):</u>
1. A Plant is not brought to required operating mode within Technical Specifications LCO action statement time.</td></tr>
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