Test Sheet 100

Form No. 243532

Revised from #95679

KEY ID Ⓐ Ⓑ Ⓒ Ⓓ

SCORING & PRINTING OPTIONS: RESCORE · MULTIPLE ANSWER SCORING · CORRECT ANSWER · MARK X · TOTAL ONLY
MARK ONLY ONE

FEED IN THIS DIRECTION

1	B	26	C	51	A
2	C	27	D	52	A
3	A	28	C	53	A
4	B	29	A	54	B
5	C	30	B	55	D
6	C	31	B	56	A
7	A	32	D	57	B
8	D	33	C	58	B
9	D	34	B	59	B
10	D	35	D	60	D
11	D	36	B	61	B
12	A	37	D	62	B
13	B	38	A	63	D
14	D	39	B	64	C
15	D	40	D	65	A
16	A	41	B	66	B
17	C	42	A	67	A
18	B	43	D	68	D
19	B	44	C	69	A
20	D	45	A	70	C
21	C	46	C	71	D
22	D	47	B	72	C
23	D	48	B	73	C
24	C	49	A	74	D
25	D	50	B	75	C

Questions 76–100: blank.

ANSWER KEY INFO. — # OF KEYS: 0 — ITEM COUNT: 075

PERFORMANCE ASSESSMENT — % OF TOTAL SCORE (00 = 100%) — POINTS EARNED — EQUAL PT VALUE

CERTIFICATION: I have reviewed all questions which were missed, have had an opportunity to ask questions, and understand the correct answer to each question. All work on this examination is my own, I have neither given or received help.

SIGNATURE ______ DATE ______

NUMBER CORRECT	75
PERCENT CORRECT	100
ROSTER NUMBER	KEY
SCORE	
RESCORE	

J 3/16/12
SW 3/16/12

COMBINED POINTS EARNED	
COMBINED PERCENT CORRECT	
LETTER GRADE	
SCORE	
RESCORE	

ID/SSN

NAME RO Key 2012

COURSE EXAM LOIT

DATE 3/16/2012

MARKING INSTRUCTIONS

USE NO. 2 PENCIL ONLY

Use a No. 2 Pencil or blue or black ink pen only.

Fill oval completely

Erase cleanly

PVNGS 2012 Reactor Operator NRC Exam

1.

This Exam Level	RO
Appears on:	RO EXAM 2012
	Tier 1 Group 1
K/A #	4.1 007 EA2.03
Importance Rating:	4.2

Which ONE of the following describes **ALL** the available locations that **ALL** (4) RTSG breaker positions can be verified after a Reactor Trip?

(1) PPS Status Panel
(2) Supplemental Protection Logic Actuation (SPLA) Cabinets
(3) B05 Phase Current Lights
(4) Locally at the Breaker

A. 1 and 4 Only

B. 1, 2 and 4 Only

C. 2, 3 and 4 Only

D. 1, 2, 3 and 4

Answer: B

Reference Id:	Q43923	
Difficulty:	2.50	
Time to complete:	2	
10CFR Category:	CFR 55.41 (7)	55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Cognitive Level:	Memory	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: LOIT Lesson Plan

K&A: Ability to determine or interpret the following as they apply to a reactor trip: Reactor trip breaker position.

Learning Objective: L80279 Explain the operation of the RTSG (Reactor Trip Switchgear) Breakers.
Justification:
A. Incorrect: Each SPLA Cabinet has indication of their respective RTSG Breaker,
B. Correct: RTSG Breaker position can be verified at these 3 locations.
C. Incorrect: PPS Status Panels do provide indication and the Phase Current Lights on B05 only show the status of C and D legs not individual breakers.
D. Incorrect:Phase Current Lights on B05 only show the status of C and D legs not individual breakers.

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2.

This Exam Level	RO
Appears on:	RO EXAM 2012 RO EXAM 2005 Tier 1 Group 1
K/A #	4.2 008 AK2.01
Importance Rating:	2.7

Given the following conditions:

- Unit 1 RCS pressure is at 2000 psia.
- A Pressurizer safety/relief valve is leaking to the RDT.
- The RDT is at 10 psig.

Which ONE of the following describes the temperature of the fluid downstream of the relief valve?

A. 215°F

B. 230°F

C. 240°F

D. 280°F

Answer: C

Reference Id:	4083	
Difficulty:	3.00	
Time to complete:	4	
10CFR Category:	CFR 55.41 (14)	55.41 (14) Principles of heat transfer thermodynamics and fluid mechanics.
Cognitive Level:	Comprehension / Anal	
Question Source:	PV Bank Not Modified	
Comment:		

Proposed reference to be provided to applicant during examination: Steam Tables

Technical Reference: Steam Tables, 40EP-9EO03. (LOCA)

K&A: Knowledge of the interrelations between the Pressurizer Vapor Space Accident and the following: Valves

Learning Objective: Given PZR Safety Valve tailpipe temperatures and the steam tables, analyze the data to determine the status of the PZR safety valve in accordance with 40EP-9EO03.

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Justification:

Directions on how to use Mollier Diagram and Steam Tables to determine tailpipe temperature of a leaking PSV.

1. Find the enthalpy of the saturated vapor using Mollier diagram or Table 2.
2. Plot this on the Saturation Line.
3. Draw a horizontal (constant h) line to the pressure that corresponds to where the device is relieving to.
4. If this point lies below the saturation line, follow the pressure line up the saturation line to determine the temperature. If above, compare the point to the Constant Temperature lines.

Any choice is plausible if the examinee does not obtain the specific enthalpy for 2000 psia or is off on drawing the lines to the correct values.

A. Incorrect: 215 ^{0}F corresponds to a RDT pressure of 15 psig if you go down on the curve.
B. Incorrect: 230 ^{0}F corresponds to a RDT pressure of 20 psig if you don't move on the curve.
C. Correct: Steam Tables diagram for a RCS press of 2000 psia and a RDT pressure at 10 psig is 240 ^{0}F.
D. Incorrect: 280 ^{0}F corresponds to a RDT pressure of 50 psig.

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3.

This Exam Level RO
Appears on: RO EXAM 2012
Tier 1 Group 1
K/A # 4.1 009 EK3.28
Importance Rating: 4.5

Given the following conditions:

- Unit 1 has tripped from 100% power.
- Sub-Cooled Margin is 36°F and lowering slowly.
- Containment Pressure is 2.7 psig and rising slowly.
- Pressurizer level is 20% and lowering slowly.
- RCS Pressure is 1780 psia and lowering slowly.
- SG #1 level is 28% WR and rising slowly.
- SG #2 level is 30% WR and rising slowly.
- SPTAs are in progress.
- **NO** ESFAS Actuations have occurred.

Which ONE of the following describes the ESFAS Actuations the RO must manually initiate?

A. SIAS and CIAS ONLY due to exceeding the low pressurizer pressure setpoint.

B. SIAS and CIAS ONLY due to exceeding the high containment pressure setpoint

C. SIAS, CIAS and MSIS ONLY due to exceeding the low pressurizer pressure setpoint.

D. SIAS, CIAS and MSIS ONLY due to exceeding the high containment pressure setpoint.

Answer: A

Reference Id: Q43924
Difficulty: 3.00
Time to complete: 3
10CFR Category: CFR 55.41 (7) 55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Cognitive Level: Comprehension / Anal
Question Source: New
Comment:

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: EOP Setpoint Document and LOIT Lesson Plan

K&A: Knowledge of the reasons for the following responses as the apply to the small break LOCA: Manual ESFAS initiation requirements

Learning Objective: List the parameters and setpoints that will cause PPS actuation.

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Justification:

A. Correct: SIAS, CIAS and MSIS setpoint is > 3.0 psig in CTMT. SIAS and CIAS setpoint < 1837 psia PZR Pressure.
B. Incorrect: SIAS/ CIAS setpoint is > 3.0 psig in CTMT.
C. Incorrect: SIAS and CIAS setpoint < 1837 psia PZR Pressure. MSIS is on High Cntmt pressure or Low SG pressure.
D. Incorrect: SIAS/CIAS and MSIS setpoint is > 3.0 psig in CTMT.

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4.

This Exam Level	RO
Appears on:	RO EXAM 2010
	RO EXAM 2012
	Tier 1 Group 1
K/A #	4.1 011 EK2.02
Importance Rating:	2.6

Given the following conditions:

- A LOCA event results in a Reactor trip.
- Containment Pressure is 3.5 psig and rising.
- The SPTAs are in progress.
- RCS Subcooling indicates 20 °F.

Which ONE of the following describes the guidance regarding the operation of the RCPs?

A. Trip Two RCPs now (in SPTAs).

B. Trip Four RCPs now (in SPTAs).

C. The CRS shall not direct tripping of RCPs until an ORP is entered.

D. The running RCPs shall remain operating until saturation conditions exist (0 ^{o}F subcooling).

Answer: B

Reference Id:	Q6331	
Difficulty:	2.00	
Time to complete:	2	
10CFR Category:	CFR 55.41 (10)	55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.
Cognitive Level:	Comprehension / Anal	
Question Source:	PV Bank Not Modified	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

TECHNICAL REFERENCE: 40EP-9EO01 SPTAs

KA STATEMENT: 4.1 011EK2.02 Knowledge of the interrelations between the pumps and the following: Large break LOCA: Pumps.

Learning Objective:

JUSTIFICATION:

A. Incorrect – All RCPs are to be secured with subcooling < 24 ^{0}F. Candidate may confuse the trip 2 leave 2 strategy with RCS pressure remaining below the SIAS setpoint.

B. Correct – This is the SPTA contingency for loss of subcooling. RCPs should not be operated without adequate subcooling.

C. Incorrect – The expectation is that these pumps will be secured prior to exiting the SPTAs. Candidate may think that this is an early step of the LOCA EOP.

D. Incorrect – This does not meet the standards set by the EOP Technical Guideline. Candidate may understand loss of subcooling as < 0 ^{0}F subcooling, not the procedurally directed < 24 ^{0}F.

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5.

This Exam Level	RO
Appears on:	RO EXAM 2012 Group 1 Tier 1
K/A #	4.2 077 AA1.05
Importance Rating:	3.9

Given the following conditions:

- Unit 1 is operating at 100% power.
- East and West switchyard voltage dropped to 516 kV.
- East and West Bus switchyard Low-Low voltage alarms are locked in.

Which ONE of the following auto/manual action(s) is taken at this time to protect Engineered Safety Function (ESF) equipment from a "double sequencing" event?

A. Water Reclamation Facility supply breakers will trip open.

B. Start both DGs and maintain them paralleled with offsite power

C. Block the NAN-S01/S02 to NAN-S03/S04 fast bus transfer capability.

D. Ensure that the Main Generator's gross MVAR output is greater than zero.

Answer: C

Reference Id:	Q44017	
Difficulty:	0.00	
Time to complete:	0	
10CFR Category:	CFR 55.41 (10)	55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.
Cognitive Level:	Memory	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: None

Technical Reference: 41ST-1ZZ02, Inoperable Power Sources

K&A: Ability to operate and/or monitor the following as they apply to Generator Voltage and Electric Grid Disturbances: Engineered safety features

Learning Objective: Explain the operation of Switchgear NAN-S05 and NAN-S06 under normal operating conditions.

Justification:

A. Incorrect. WRF breakers will auto trip with a low voltage and SIAS actuation
B. Incorrect. An option is to start load DG and isolate class buses from offsite power.
C. Correct. actions directed by 41ST-1ZZ02, appendix G
D. Incorrect. ST directs that gross MVAR be less than 0 such that PVNGS is not supporting switchyard voltage.

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6.

This Exam Level	RO
Appears on:	RO EXAM 2007
	RO EXAM 2012
	Tier 1 Group 2
K/A #	4.2 022 AK3.02
Importance Rating:	3.5

Given the following conditions:

Initial Conditions:

- Unit 1 is operating at 100% power.
- Charging has been secured due to a leak downstream of the Charging Pumps.
- 40AO-9ZZ04, RCP Emergencies, has been entered.

Subsequently:

- The Unit trips due to a LOCA.
- Pressurizer pressure is currently 1500 psia and stable.
- Containment pressure is 2.1 psig and slowly increasing.
- Pressurizer level is 20% and stable.
- RCS T-cold is 560°F.
- RCS T-hot is 563°F.
- RCP 1A seal 2 outlet temperature is 260°F.
- RCP 2A seal 2 outlet temperature is 252°F.
- Safety Injection flow is adequate.
- RCPs 1A/2A have been secured.

Which ONE of the following actions is procedurally required?

A. Trip the 1B/2B RCPs to prevent pump cavitation.

B. Initiate CIAS, containment pressure is greater than setpoint.

C. Isolate Seal bleedoff to the 1A/2A RCPs to prevent seal damage.

D. Override and energize the class pressurizer heaters to restore pressurizer pressure.

Answer: C

Reference Id:	Q10375	
Difficulty:	3.00	
Time to complete:	2	
10CFR Category:	CFR 55.41 (10)	55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.
Cognitive Level:	Comprehension / Anal	
Question Source:	PV Bank Not Modified	

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Comment:

Proposed reference to be provided to applicant during examination: Steam tables and Appendix 2 pump curves

Technical Reference: 40AO-9ZZ04 (RCP emergences)

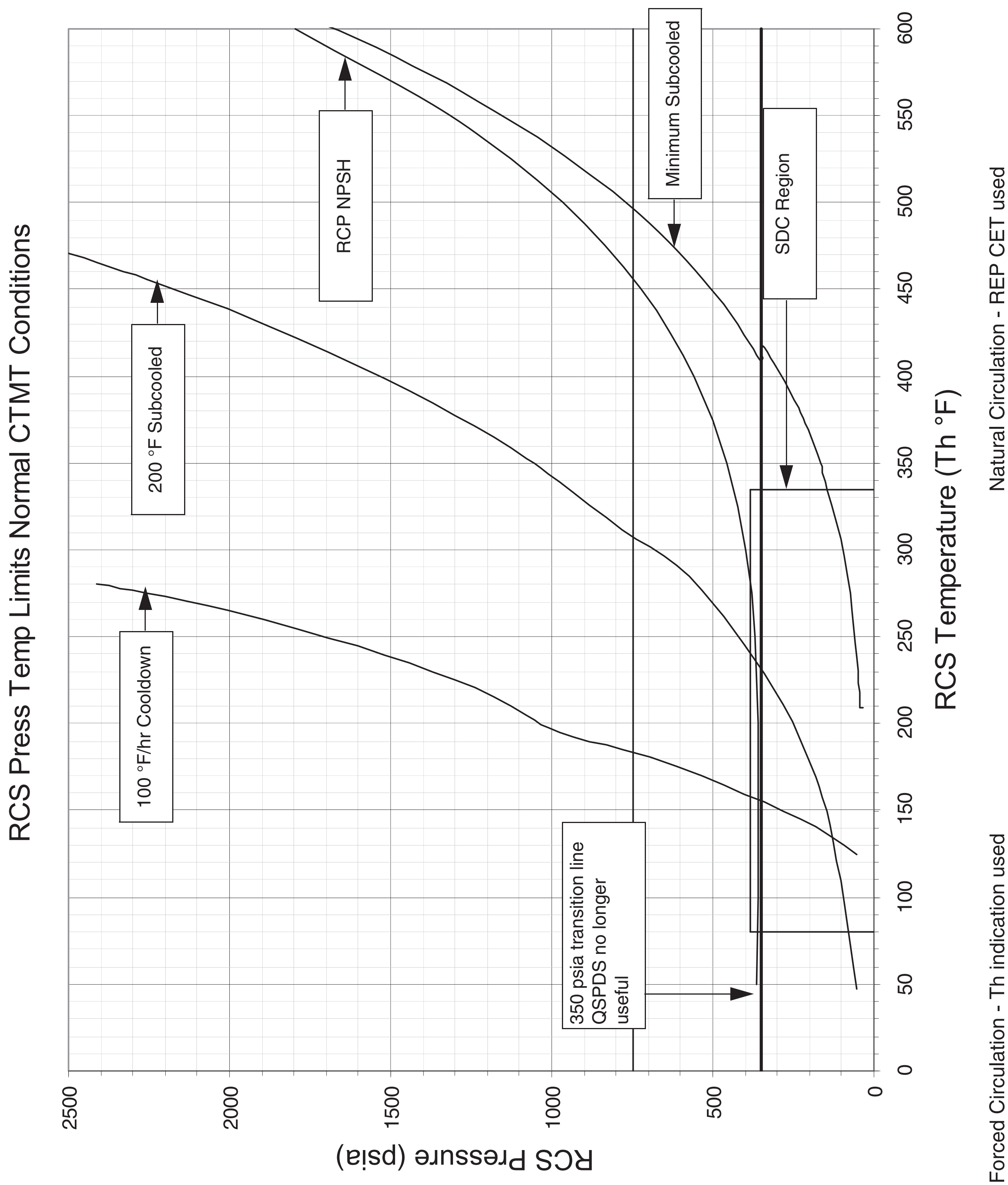
K&A: Knowledge of the reasons for the following responses as they apply to the Loss of Reactor Coolant Makeup: Actions contained in SOPs and EOPs for RCPs, loss of makeup, loss of charging, and abnormal charging

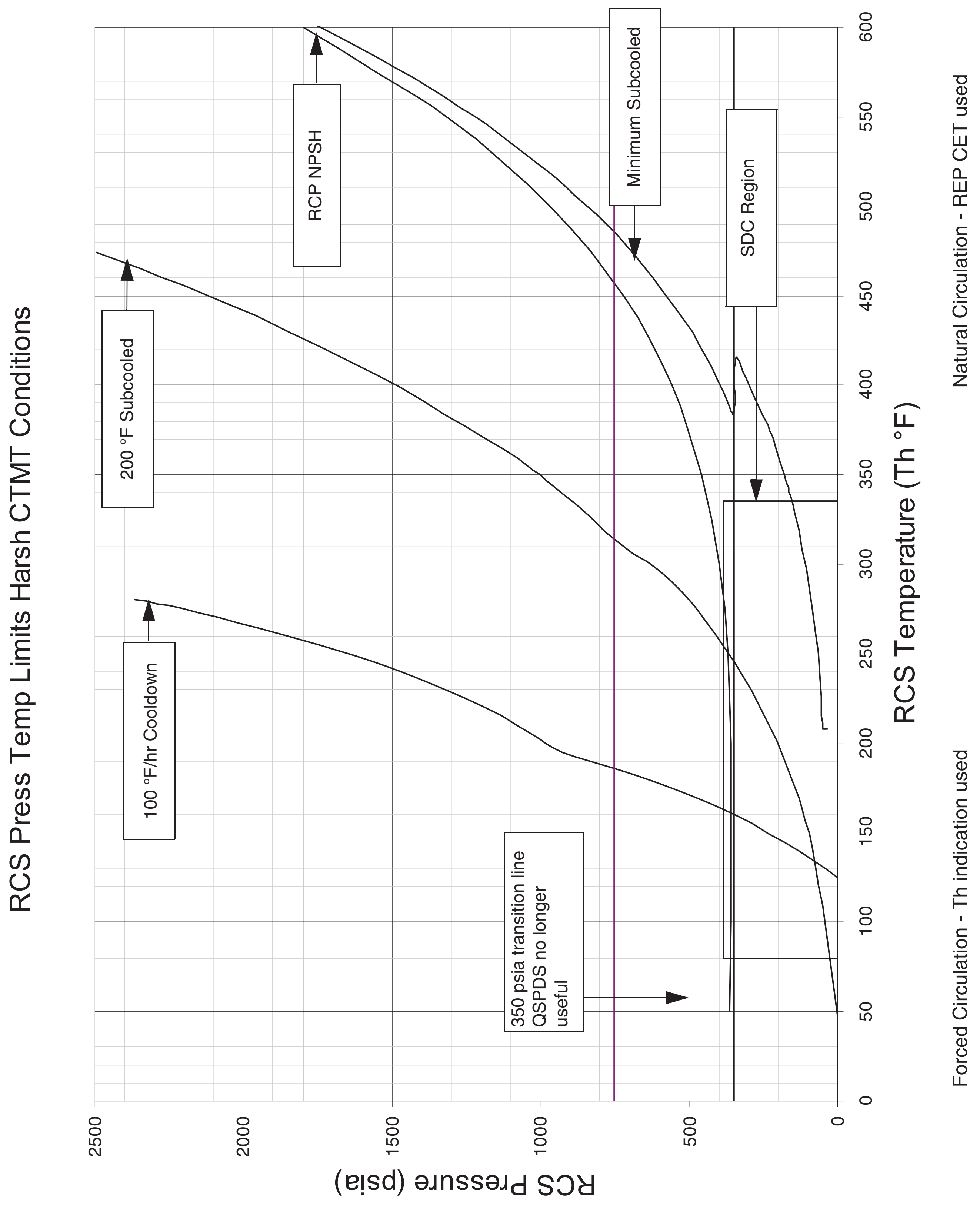
Learning Objective: Given RCP motor amps and Upper Thrust Bearing Temperature determine the appropriate action to take based on RCP motor amps and thrust bearing temperature in accordance with 40AO-9ZZ04.

Justification:

A. Incorrect: subcooled margin and NPSH requirements are met
B. Incorrect: containment pressure is less than setpoint of 3.0 psig
C. Correct: RCP in stby with no seal injection requires that the Bleed Off valve be closed prior to exceeding 250 degrees on Seal 2 outlet temperature
D. Incorrect: PZR level is less than 25%, heater cutout

Appendix 2,
Figures




RCS Press Temp Limits Harsh CTMT Conditions
RCS Temperature (Th °F)
RCS Pressure (psia)
0
50
100
150
200
250
300
350
400
450
500
550
600
2500
2000
1500
1000
500
0
100 °F/hr Cooldown
200 °F Subcooled
RCP NPSH
Minimum Subcooled
SDC Region
350 psia transition line QSPDS no longer useful
Natural Circulation - REP CET used
Forced Circulation - Th indication used

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7.

This Exam Level	RO
Appears on:	RO EXAM 2007
	RO EXAM 2012
	Tier 1 Group 1
K/A #	42 025 AA2.07
Importance Rating:	3.4

Given the following conditions:

- Unit 1 is in Mode 4.
- LPSI pump "B" is providing SDC flow.
- RCS temperature 325°F.

Auxiliary Spray valve "B" fails open and the following conditions are observed.

- LPSI pump "B" amps and flow are oscillating.
- Window 2B06A, SDC TRAIN A/B FLOW LO is alarming.

This is an indication of...

A. LPSI pump B "cavitating".

B. LPSI pump B in a "runout" condition.

C. CHB-HV-530 (RWT to Train B SI Pumps) closing.

D. an inadvertant B train Recirculation Actuation Signal (RAS).

Answer: A

Reference Id:	Q10357	
Difficulty:	2.00	
Time to complete:	3	
10CFR Category:	CFR 55.41 (10)	55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.
Cognitive Level:	Comprehension / Anal	
Question Source:	PV Bank Not Modified	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40EP-9EO11 40AL-9RK2B

K&A: Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: Pump cavitation

Learning Objective: Given the LMFRP HR-2 is being performed, and SDC is in service describe how adequate SDC flow is determined and what actions may be taken if adequate flow cannot be maintained in accordance with 40EP-9EO11.

JUSTIFICATION:

A. Correct: these are classic cavitation indications with lowering PZR pressure and stable temperature
B. Incorrect: run out would be high amps and high flow
C. Incorrect: SDC suction is thru SI-HV-655 and LPSI suction valve SI-HV-692 is closed isolating SDC flow from RWT
D. Incorrect: RAS would trip the LPSI pump

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8.

This Exam Level	RO
Appears on:	RO EXAM 2012 Tier 1 Group 1
K/A #	4.2 026 AK3.03
Importance Rating:	4.0

Given the following initial conditions:

- Unit 1 has tripped from 100% power.
- SIAS/CIAS have automatically initiated.
- Pressurizer pressure is 1800 psia and stable.
- RCS T-cold is 562 degrees and stable.
- Containment pressure is 0.5 psig and stable.
- Containment temperature is 115 degrees and stable.
- RU-6, Nuclear Cooling Water Radiation Monitor is alarming.

Subsequently the CRS enters 40EP-9EO03, LOCA and directs the following actions.

- Stop all RCPs.
- Close the Nuclear Cooling Water Containment Isolation valves.

Why were the RCPs secured?

A. Loss of Seal Injection

B. Loss of RCS subcooling.

C. Harsh containment conditions.

D. Loss of cooling water to the RCPs.

Answer: D

Reference Id:	Q44018	
Difficulty:	3.00	
Time to complete:	3	
10CFR Category:	CFR 55.41 (10)	55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.
Cognitive Level:	Comprehension / Anal	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: Steam Tables

Technical Reference: 40EP-9EO03, LOCA

K&A: Knowledge of the reasons for the following responses as they apply to the Loss of Component Cooling Water: Guidance actions contained in EOP for Loss of CCW

Learning Objective: Given RCS pressure and temperature during performance of an EOP analyze these conditions to decide if the RCPs can be operated.

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JUSTIFICATION:

A. Incorrect. Seal Injection has an automatic isolation on low temperature, examinee may also believe that it isolates due to a SIAS/CIAS as numerous other CVCS valves.
B. Incorrect. Examinee will have to verify the status of subcooling with the reduced RCS pressure. 564 degrees + 24 = 588 = 1409 psi
C. Incorrect. Examinee will have to verify containment conditions
D. Correct. Examinee will have to know the cooling source of the RCPs and the requirement to secure them.

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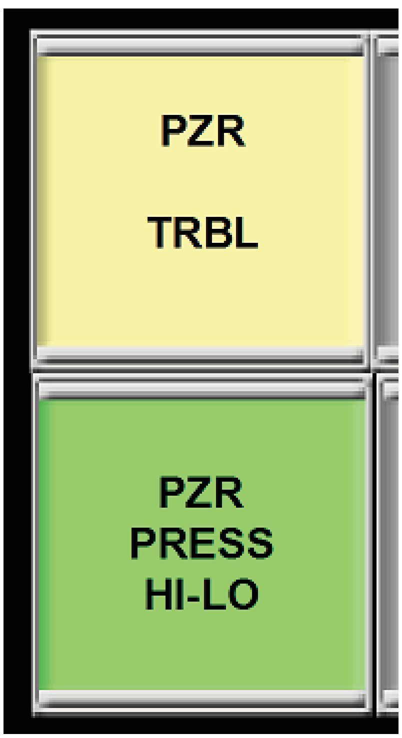
9.

This Exam Level	RO
Appears on:	RO EXAM 2012 Tier 1 Group 1
K/A #	2.4.45
Importance Rating:	4.1

Given the following conditions:

- RCN-PIC-100 (PZR Press Master Controller), is in AUTO.
- RCN-HS-100 (PZR Press Control Channel X/Y selector), is selected to channel X .
- Pressure transmitter RCN-PT-100X fails low.

The following annunciators alarm on B04:



Which ONE of the following describes the appropriate response by the RO?

The RO will **FIRST** address the PZR ...

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A. TRBL Alarm and Stop PZR Heaters.

B. PRESS HI-LO Alarm and Stop PZR Heaters.

C. TRBL Alarm and select 100Y on RCN-HS-100 (Pressurizer Pressure Control Selector).

D. PRESS HI-LO Alarm and select 100Y on RCN-HS-100 (Pressurizer Pressure Control Selector)

Answer: D

Reference Id:	Q43926	
Difficulty:	2.00	
Time to complete:	2	
10CFR Category:	CFR 55.41 (10)	55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.
Cognitive Level:	Comprehension / Anal	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: LOIT Lesson Plan

K&A: Ability to prioritize and interpret the significance of each annunciator or alarm. PPCS Malfunction

Learning Objective: Describe the conditions required to generate the following annunciators: PZR TRBL, PZR PRES HI-LO.

Justification:

A. Incorrect: PZR TRBL Alarm is Amber, so the priority shall be given to the Green PZR Press Hi-Lo alarm. Stop PZR heaters is a correct action ONLY if both pressure instruments Fail Low.
B. Incorrect: PZR Press Hi-Lo alarm is Green, this is the correct Alarm to address. Stop PZR heaters is a correct action ONLY if both pressure instruments Fail Low.
C. Incorrect: PZR TRBL Alarm is Amber color so the priority shall be given to the Green PZR Press Hi-Lo alarm. Selecting the other instrument is the correct response per the ARP.
D. Correct: PZR Press Hi-Lo alarm is Green, this is the correct Alarm to address. Selecting the other instrument is the correct response per the ARP.

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10.

This Exam Level:	RO
Appears on:	RO EXAM 2009
	RO EXAN 2012
	Tier 1 Group 1
K/A #	4.1 029 EK1.03
Importance Rating:	3.6

Given the following conditions:

- Unit 1 is at 30% power while shutting down in preparations for a refueling outage.
- Reactor Coolant pump 1A has tripped.
- The reactor did not automatically trip.
- All attempts to trip the reactor from the Control Room have failed.

Assuming NO other operator actions, initiating an 80 gpm boration would add...

A. positive reactivity to the core and cause RCS temperature to increase.

B. positive reactivity to the core and cause RCS temperature to decrease.

C. negative reactivity to the core and cause RCS temperature to increase.

D. negative reactivity to the core and cause RCS temperature to decrease.

Answer: D

Reference Id:	Q22491	
Difficulty:	3.00	
Time to complete:	3	
10CFR Category:	CFR 55.41 (1)	55.41 (1) Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.
Cognitive Level:	Comprehension / Anal	
Question Source:	PV Bank Not Modified	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40DP-9AP06 (SPTA tech guideline)

K&A: Knowledge of the operational implications of the following concepts as they apply to the ATWS: Effects of boron on reactivity

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Learning Objective: Given plant conditions following a reactor trip analyze whether the Reactivity Control Safety Function is met and what contingency actions are required if it is not in accordance with 40EP-9EO01.

Justification: The examinee may confuse the purpose of boron and dilution as to which will add negative reactivity. Another consideration is that there is a time in core life (BOL, high boron concentration and low power) when a positive MTC could exist where the effects of temperature change don't follow the normal core dynamics.

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

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11.

This Exam Level	RO
Appears on:	RO EXAM 2012 Tier 1 Group 1
K/A #	038 2.2.44
Importance Rating:	4.2

Given the following conditions:

- Unit 2 was tripped due to a Steam Generator Tube Rupture.
- RCS pressure is 895 psia.
- RCS subcooling is 55°F.
- Steam Generator #1 pressure is 890 psia.
- RU-4 in high alarm.
- Steam generator #1 is isolated.
- Steam generator #1 level is 78% NR and rising slowly.
- Steam generator #2 level is 50% NR and steady.

Which ONE of the following is the preferred method to control level in the isolated steam generator with a ruptured tube?

A. Steam the #1 steam generator to atmosphere via the ADVs.

B. Bypass the MSIV and steam the #1 steam generator to the condenser.

C. Line-up high rate blowdown to the condenser from #1 steam generator.

D. Lower RCS pressure below #1 steam generator pressure and allow backflow to the RCS.

Answer: D

Reference Id:	Q44015	
Difficulty:	3.00	
Time to complete:	3	
10CFR Category:	CFR 55.41 (10)	55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.
Cognitive Level:	Comprehension / Anal	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE.

Technical Reference: 40EP-9EO04 (SGTR) 40DP-9AP09 (SGTR Tech Guide)

K&A: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions. SGTR

Learning Objective: L11218 Given that the SGTR EOP is being implemented describe the SGTR EOP mitigation strategy in accordance with 40EP-9EO04.

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Justification:

A. Incorrect: This will lower SG pressure to further below RCS pressure which will increase SG level and spread more contamination.
B. Incorrect: This will lower SG pressure to further below RCS pressure which will increase SG level, steaming to the condenser would minimize the chance of release to the environment, but still spread the contamination to the secondary.
C. Incorrect: Blowdown will lower level, but spread contamination to the secondary.
D. Correct: This will lower RCS pressure and reduce level of the SG by moving water into the RCS. Contamination will be limited by putting the contaminated water back in the RCS.

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12.

This Exam Level	RO
Appears on:	RO EXAM 2009
	RO EXAM 2012
	Tier 1 Group 1
K/A #	4.1 055 EK3.01
Importance Rating:	2.7

If there is a station "Blackout" the class (PK) batteries are designed to maintain rated voltage for UP TO ...

A. 2 hours to provide continuous DC during a Design Basis Event.

B. 4 hours to provide continuous DC during a Design Basis Event.

C. 2 hours to provide sufficient power for the protection and control of transformers and switchgear.

D. 4 hours to provide sufficient power for the protection and control of transformers and switchgear.

Answer: A

Reference Id: Q22493
Difficulty: 2.00
Time to complete: 2
10CFR Category: CFR 55.41 (5) 55.41 (5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.
Cognitive Level: Memory
Question Source: PV Bank Not Modified
Comment:
Proposed reference to be provided to applicant during examination: NONE

PRA SIGNIFICANT QUESTION

Technical Reference: FSAR, LOIT Lesson plans

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

Learning Objective: Discuss the purpose and conditions under which the 125 VDC Class IE Power System is designed to function.

Justification:

A. Correct: 2 hours and concurrent DBE-LOCA concurrent with BO as found in FSAR
B. Incorrect: 4 hours is the old rating for the non-lass NK batteries
C. Incorrect: power for the protection and control of transformers is for the non-class NK batteries, examinee may choose this believing that the ESF transformers use class power
D. Incorrect: 4 hours is the old rating for the non-lass NK batteries

13.

This Exam Level	RO
Appears on:	RO EXAM 2012 Tier 1 Group 1
K/A #	4.2 056 AA2.17
Importance Rating:	3.4

Given the following conditions:

- Unit 1 has tripped from 100% power due to a LOOP.
- EDG 'A' Tripped on Low Lube Oil Pressure.

Which ONE of the following describes the operational status the the PZR Backup Heaters?

________ Backup Heater Banks(s) is/are available.

A. ZERO (0)

B. ONE (1)

C. TWO (2)

D. FOUR (4)

Answer: B

Reference Id:	Q43929	
Difficulty:	2.00	
Time to complete:	3	
10CFR Category:	CFR 55.41 (7)	55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Cognitive Level:	Memory	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: LOIT lesson plan

K&A: Ability to determine and interpret the following as they apply to the Loss of Offsite Power: Operational status of PZR backup heaters

Learning Objective: Describe the interrelationship between the Pressurizer Pressure Control System and the following systems: 480 VAC Class IE Power • 480 VAC Non-Class IE Power

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Justification: Non-class heaters (prop and backup) NGN-L11 & 12, Class backups PGA-L33 & 34

A. Incorrect: NBN-S01/S02 and PBA-S04 are de-energized due to the LOOP and the 'A' EDG tripped. These provide power to all other Backup and Proportional Heaters. B class heaters powered from DG "B" are still available.

B. Correct: Non class heaters come from NGN-L11/12 which are lost on the LOOP. PBB-S04 is the only energized 4160V bus. There are only two banks of class backup heaters. "A" train lost power with the DG tripping. The B Class Backup Heater bank is the only bank with power.

C. Incorrect: NBN-S01/S02 and PBA-S04 are de-energized due to the LOOP and the 'A' EDG tripped. These provide power to all other Backup and Proportional Heaters.

D. Incorrect: NBN-S01/S02 and PBA-S04 are de-energized due to the LOOP and the 'A' EDG tripped. These provide power to all other Backup and Proportional Heaters.

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14.

This Exam Level RO
Appears on: RO EXAM 2012
Tier 1 Group 1
K/A # 057 2.4.50
Importance Rating: 4.2

Given the following conditions:

- **Unit 1** is operating at 100% power.
- 120VAC IE PNL D27 Inverter C Trouble Alarm was received in the Control Room.
- The area operator reports that DC power to 120VAC Class IE Inverter PNC-N13 has been lost.

Which ONE of the following describes the restoration of power to PNC?

PNC 120VAC power is restored by...

A. an auto shift to the battery.

B. a manual shift to the battery.

C. an auto shift to the voltage regulator.

D. a manual shift to the voltage regulator.

Answer: D

Reference Id: Q43931
Difficulty: 2.00
Time to complete: 2
10CFR Category: CFR 55.41 (7) 55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Cognitive Level: Comprehension / Anal
Question Source: New
Comment:
Proposed reference to be provided to applicant during examination: NONE

Technical Reference: LOIT Lesson Plan, 41AL-9RK1A (Unit 1 B01A ARP)

PRA SIGNIFICANT QUESTION
UNIT DIFFERENCES QUESTION

K&A: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.Loss of Vital AC instrument bus (00057)

Learning Objective: Describe the conditions required to generate the following annunciators: • 120VAC IE PNL D25 INV A • 120VAC IE PNL D26 INV B • 120VAC IE PNL D27 INV C • 120VAC IE PNL D28 INV D

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Justification:

A. Incorrect: The battery is the normal supply to the inverter. Unit 1 is not equipped with a static transfer switch.
B. Incorrect: The battery is the normal supply to the inverter. If the normal power supply was the voltage regulator, a manual transfer to the battery would be required.
C. Incorrect: This would be correct in Unit 2 or 3 which is equipped with a Static Transfer switch that would automatically transfer to the voltage regulator.
D. Correct: Unit 1 is NOT supplied with a Static Transfer switch as in Unit 2 and Unit 3. Therefore on a loss of Power to the Inverter the operator must manually transfer the power supply from the inverter to the voltage regulator.

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15.

This Exam Level	RO
Appears on:	RO EXAM 2012
	Tier 1 Group 1
K/A #	4.2 058 AK1.01
Importance Rating:	2.8

Given the following conditions:

- Unit 1 tripped from 100% power due to a Loss of Offsite Power, LOOP.
- No ESFAS actuations have been initiated.
- AFN-P01 (Essential Steam Driven AFW pump) is supplying feed to the SGs.
- The Control Power Transfer Switch (PBA-U01) is selected to it's alternate feed source from PKA-H11.
- SG pressure is being controlled with ADVs SGA-HV-179/184.
- Both DGs are supplying their respective buses.

With no other Operator actions, what is the **immediate** operational implications of a loss of the "A" battery charger, PKA-H11?

A. ADVs (SGA-HV-179/184) will fail closed.

B. DG "A" output breaker (PBA-S03B) can not be tripped from the control room.

C. Downcomer containment isolation valves SGA-UV-172 and SGA-UV-175 fail closed.

D. AFN-P01 (Non Essential Motor Driven Aux Feed Pump) can not be tripped from the control room.

Answer: D

Reference Id:	Q44020	
Difficulty:	3.50	
Time to complete:	3	
10CFR Category:	CFR 55.41 (7)	55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Cognitive Level:	Memory	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40AO-9ZZ13 (Loss of Class Instrument and Control Power), 40EP-9EO05 (LOCA)

PRA SIGNIFICANT QUESTION

K&A: Knowledge of the operational implications of the following concepts as they apply to Loss of DC Power: Battery charger equipment and instrumentation.

Learning Objective: Given a loss of PN or PK describe the availability of Auxiliary Feedwater in accordance with 40AO-9ZZ13.

Justification:

A. ADV will lose their permissive as the "A" battery dies and fail closed, NOT IMMEDIATELY
B. DG "A" output breaker can not be tripped on a loss of PKA-M41/D21. NOT IMMEDIATELY
C. Downcomer Isolation fail open on a loss of power. The economizer isolations would close immediately if aligned to their alternate source PKA-M41.
D. The alternate supply for AFN-P01 is directly from the battery charger.

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16.

This Exam Level	RO
Appears on:	RO EXAM 2012 Tier 1 Group 1
K/A #	4.4 E05 EK2.2
Importance Rating:	3.7

Given the following conditions:

Initial Conditions:

- Unit 2 has tripped from 100% power.
- SG #1 is 1000 psia and lowering.
- SG #1 is 40% WR and lowering.
- SG #2 is 800 psia and lowering.
- SG #2 is 10% WR and lowering.
- PZR level is at 30% and slowly lowering.
- Containment Pressure is 1 psig and rising.

At the time that the ORP is entered the conditions are as follows:

- Containment pressure peaked and is stable at 9.8 psig.
- Containment temperature is 185°F.
- PZR level is 18% and rising.
- RVUH level is 67%.
- RCS subcooling is 98°F.
- SG #1 is at 34% WR (rising) and being fed from AFW at 500 gpm.
- SG #2 is below the indicated level.
- Both HPSI pumps are injecting into the RCS.

Based on these conditions, you should obtain CRS concurrence and throttle HPSI...

A. immediately.

B. when PZR level reaches 33%.

C. when RVUH is equal to 100%.

D. when SG #1 Level is 45%-60% NR.

Answer: A

Reference Id:	Q43934	
Difficulty:	3.00	
Time to complete:	2	
10CFR Category:	CFR 55.41 (10)	55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.
10CFR Category:	CFR 55.41 (10)CFR 55.41 (8)	55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.55.41 (8) Components, capacity, and functions of emergency systems.
Cognitive Level:	Comprehension / Anal	
Question Source:	Modified PV Bank	

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Comment:
Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40EP-9EO05, Excess Steam Demand, 40EP-9EO10 Appendix 2 SI Throttle Criteria

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

Learning Objective: Given conditions of an ESD describe the mitigating strategy outlined in the ESD EOP in accordance with 40EP-9EO05.

Justification:

A. Correct – PZR level requirement is > 15% for Harsh CTMT conditions.
B. Incorrect – PZR level requirement for throttling HPSI is > 15% level when in Harsh CTMT conditions. 33% is the normal PZR Level Band per SPTAs
C. Incorrect – RVUH level must be greater than 16% to throttle HPSI, which it is. Candidate may not understand RVUH and Plenum relationship.
D. Incorrect – The SG requirement is RESTORING to 45-60% NR level. Candidate may believe that SG levels must be in the band.

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17.

This Exam Level:	RO
Appears on:	RO EXAM 2012
	Tier 1 Group 1
K/A #	4.4 E06 EA1.2
Importance Rating:	3.4

Given the following conditions:

- Unit 1 is tripped from 100% power.
- Containment Pressure is 1.7 psig and rising.
- Containment Temperature is 120 ^{0}F and rising.
- Containment Humidity is rising.
- Containment sump levels are rising.
- PZR Pressure is 2250 psia and rising.
- PZR Level is 58% and rising.
- Tcold is 568 ^{0}F and rising.
- Subcooled Margin is 58 ^{0}F and lowering.
- SG 1 and 2 levels are 30% WR and lowering.

Which ONE of the following describes the ongoing event?

A. RCS Cold Leg LOCA.

B. PZR Steam Space LOCA.

C. Feedline Break (ESD) inside containment.

D. Steam Line Break (ESD) inside containment.

Answer: C

Reference Id:	Q43935	
Difficulty:	3.00	
Time to complete:	3	
10CFR Category:	CFR 55.41 (5)	55.41 (5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

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Cognitive Level: Comprehension / Anal
Question Source: New
Comment:
Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40EP-9EO05, ESD

K&A: Ability to operate and / or monitor the following as they apply to the (Loss of Feedwater) Operating behavior characteristics of the facility.

Learning Objective: Given conditions of an ESD analyze whether or not entry into the ESD EOP is appropriate
in accordance with 40EP-9EO05.

Justification:

A. Incorrect: CTMT parameters changing are indicative of a LOCA inside the CTMT, Subcooling lowering is indicative of a LOCA. Tc and PZR parameters would lower.
B. Incorrect: CTMT parameter and PZR level rising support the PZR Steam Space LOCA as does lowering subcooling. PZR Pressure would be lowering.
C. Correct: All of these parameters support the Feedline Break inside CTMT.
D. Incorrect: CTMT parameters support the Steam Line Break inside CTMT. Subcooling would rise, PZR Pressure and Level would lower. ESD procedure will mitigate both the Feedline and Steam Line breaks.

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18.

This Exam Level	RO
Appears on:	RO EXAM 2012
	Tier 1 Group 1
K/A #	4.2 065 AA1.03
Importance Rating:	2.9

Given the following conditions:

- Unit 1 has experienced a Loss of Instrument Air (IA) to the Containment.
- The CRS is implementing 40AO-9ZZ06 (Loss of Instrument Air).

Which ONE of the following valve handswitches must be taken to CLOSE prior to restoring IA to Containment per 40AO-9ZZ06?

A. CHA-HV-507 (RCP Bleedoff Isolation to RDT)

B. CHA-UV-516 (Letdown to Regen Hx Isolation)

C. WCB-UV-61 (CHW Return HDR Inside CNTMT Isol VLV)

D. NCB-UV-403 (NCW CNTMT Downstream Return Isol VLV)

Answer: B

Reference Id:	Q43990	
Difficulty:	3.00	
Time to complete:	2	
10CFR Category:	CFR 55.41 (7)	55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Cognitive Level:	Memory	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40AO-9ZZ06 (Loss of Instrument Air)

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K&A: Ability to operate and / or monitor the following as they apply to the Loss of Instrument Air: Restoration of systems served by instrument air when pressure is regained

Learning Objective: Determine the mitigating strategies of the Loss of Instrument air AOP.

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Justification:

A. Incorrect: This is an IA operated valve inside the CTMT that fails open to allow Seal Bleed Off to the RDT, it is not to be closed.
B. Correct: Per step 4 of section 3.0, this valve will fail closed but if the handswitch is not taken to close the valve will open upon restoration of IA and possibly lead to damage of the letdown IXs.
C. Incorrect: This valve is a Motor Operated Valve that will not be affected by the loss of IA, it is the inside CTMT isolation valve for WC.
D. Incorrect: This valve is a Motor Operated Valve that will not be affected by the loss of IA, it is the inside CTMT isolation valve for NC.

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19.

This Exam Level	RO
Appears on:	RO EXAM 2008
	RO EXAM 2012
	Tier 1 Group 2
K/A #	4.2 001 AA1.07
Importance Rating:	3.3

Given the following conditions:

- Unit 3 is operating at 80%.
- Group 5 CEAs at 120 inches withdrawn.
- All others CEAs at UEL.
- Selected CEA is # 14.
- Selected CEA Group is # 5.
- A malfunction causes CEA 15 to move 12 steps out before STANDBY is selected and motion stops.

Based on this event the pulse counter selected **Group** position reads...

A. 120 inches.

B. 122.25 inches.

C. 124.5 inches.

D. 129 inches.

Answer: B

Reference Id: Q43936
Difficulty: 2.00
Time to complete: 4
10CFR Category: CFR 55.41 (7) 55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Cognitive Level: Comprehension / Anal
Question Source: PV Bank Not Modified
Comment:

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: LOIT lesson plan

K&A: Ability to operate and / or monitor the following as they apply to the Continuous Rod Withdrawal: RPI

Learning Objective: Describe the required actions addressing a continuous rod motion accident.

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Justification: 12 steps times 3/4 inch equals 129 inches withdrawn

A. Incorrect: examinee may believe that that the pulse counter uses lowest CEA position (CPCs)
B. Correct: group position is the average position. 129 + 120 + 120 + 120 = 489/4 = 122.25
C. Incorrect: examinee may believe that the pulse counter uses average of high/low
D. Incorrect: examinee may believe that pulse counter uses highest cea position (CPCs)

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20.

This Exam Level	RO
Appears on:	RO EXAM 2012 Tier 1 Group 2
K/A #	4.2 005 AK3.06
Importance Rating:	3.9

Given the following conditions:

- Multiple channels of CPCs (Lo DNBR) have trip lights illuminated on B05.
- The reactor failed to automatically trip.
- The CRS has directed the RO to open the supply breakers for L03 and L10 for a minimum of 5 seconds.

Which ONE of the following describes the reason for this action?

The 5 seconds allows time for the...

A. motor generator stop contacts to close.

B. CEAs to drop to the bottom of the core.

C. trip coils to actuate to open L03 and L10 breakers.

D. effects of the motor generator flywheel to taper off interrupting power to the CEAs.

Answer: D

Reference Id:	Q43938	
Difficulty:	2.00	
Time to complete:	2	
10CFR Category:	CFR 55.41 (6)	55.41 (6) Design, components, and functions of reactivity control mechanisms and instrumentation.
Cognitive Level:	Memory	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: EOP OPERATIONS EXPECTATIONS

K&A: Knowledge of the reasons for the following responses as they apply to the Inoperable / Stuck Control Rod: Actions contained in EOP for inoperable/stuck control rod.

Learning Objective: Given plant conditions following a reactor trip analyze whether the Reactivity Control Safety Function is met and what contingency actions are required if it is not in accordance with 40EP-9EO01.

Justification:

A. Incorrect: MG stop contact does not get a signal to actuate, these actions remove power from the MG set input, therefore no output.
B. Incorrect: CEAs do require to be inserted within 4 seconds per Tech Specs, but this is not the reason for the 5 second wait.
C. Incorrect: Trip coils inside the breaker have no time delay associated with them, they open instantaneously.
D. Correct: As the Load Center supplying power to the MG sets is de-energized, the MG set flywheels will maintain the MG set output as inertial energy is dissipated.

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21.

This Exam Level	RO
Appears on:	RO EXAM 2012 Tier 1 Group 2
K/A #	4.2 028 AK1.01
Importance Rating:	2.8

Given the following conditions:

- Unit 3 operating at 100% power.
- RCN-LIC-110 (Pressurizer Level Master Controller) is in "REMOTE-AUTO".
- RCN-HS-110 (Level Control Selector Channel X/Y) is selected to channel 'Y'.
- RCN-HS-100-3 (Pressurizer Heater Control Selector Level Trip Channel) is selected to 'X'.
- A leak develops on the reference leg of RCN-LT-110Y (Level Transmitter 110Y). This leak exceeds the capacity of the condensing chamber's ability to keep the reference leg full.

Assuming NO operator action, which ONE of the following describes the plant response?

Pressurizer level indicates ....

A. low, presssurizer heaters will cut-out.

B. low, the standby charging pump will start.

C. high, letdown flow will be lost.

D. high, letdown flow will lower and stabilize at approximately 30 gpm.

Answer: C

Reference Id:	Q43992	
Difficulty:	2.00	
Time to complete:	2	
10CFR Category:	CFR 55.41 (7)	55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Cognitive Level:	Comprehension / Anal	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: LOIT Lesson Plan

K&A: AK1.01 PZR reference leak abnormalities. Knowledge of the operational implications of the following concepts as they apply to Pressurizer Level Control Malfunctions: PZR reference leak abnormalities

Learning Objective: Describe the response of the Pressurizer Level Control System to a failure of a Pressurizer Level Transmitter.

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Justification:

A. Incorrect: PLCS senses a high level, the heaters cut out at 27% indicated level.
B. Incorrect:PLCS senses a high level, charging pumps will stop not start on deviation from setpoint.
C. Correct: The level control system will sense a high level. Letdown flow increases to maximum. "Normally running" charging pump stops. Letdown will isolate due to the automatic closure of CHB-UV-0515 upon receipt of a hi-hi regenerative heat exchanger outlet temperature.
D. Incorrect: The level control system will sense a high level. Letdown flow decrease initially to 30 gpm, but not stabilize.

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22.

This Exam Level	RO
Appears on:	RO EXAM 2012
	Tier 1 Group 2
K/A #	4.2 060 AK2.01
Importance Rating:	2.6

Given the following conditions:

- Unit 1 is operating at 100% power.
- A waste gas system discharge is in progress.
- RU-12 (Waste Gas Decay Tank Discharge) readings are stable.
- RU-14 (Radwaste Building Ventilation Exhaust Filter Inlet) is in ALERT.
- RU-15 (Waste Gas Area Combined Ventilation Exhaust) is in ALERT.
- RU-143 (Plant Vent) readings are rising.

Which ONE of the following describes the event that is occurring?

A. Gas system discharge is exceeding limits, GRN-UV34A/B will auto CLOSE.

B. Gas system discharge is exceeding limits, GRN-UV34A/B will remain OPEN.

C. Waste Gas Decay Tank Leak is in progress, GRN-UV34A/B will auto CLOSE.

D. Waste Gas Decay Tank Leak is in progress, GRN-UV34A/B will remain OPEN.

Answer: D

Reference Id:	Q44021	
Difficulty:	4.00	
Time to complete:	3	
10CFR Category:	CFR 55.41 (11)	55.41 (11) Purpose and operation of radiation monitoring systems, including alarms and survey equipment.
10CFR Category:	CFR 55.41 (11)CFR 55.41 (7)	55.41 (11) Purpose and operation of radiation monitoring systems, including alarms and survey equipment.55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Cognitive Level:	Comprehension / Anal	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 74RM-9EF41, Rad Monitoring ARP

K&A: Knowledge of the interrelations between the Accidental Gaseous Radwaste Release and the following: ARM system, including the normal radiation-level indications and the operability status.

Learning Objective: Explain the operation of the Area Radiation Monitors under normal operating conditions.

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Justification:

a. Incorrect. RU-12 and RU-143 would both be rising or in alarm if a discharge was exceeding limits. GRN-UV-34A/B auto close on RU-12 high alarm only.
b. Incorrect. RU-12 and RU-143 would both be rising or in alarm if a discharge was exceeding limits. GRN-UV-34A/B auto close on RU-12 high alarm only.
c. Incorrect. ARM RU-14/15 and RU-143 would be rising or in alarm for GR system leak. GRN-UV-34A/B auto close on RU-12 high alarm only.
d. Correct. ARM RU-14/15 and RU-143 would be rising or in alarm for GR system leak. GRN-UV-34A/B auto close on RU-12 high alarm only.

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23.

This Exam Level:	RO
Appears on:	RO EXAM 2012 Tier 1 Group 2
K/A #	4.2 068 AK2.02
Importance Rating:	3.7

Given the following conditions:

- Unit 2 Control Room is experiencing a fire.
- The CRS has directed an evacuation of the Control Room.
- 40AO-9ZZ19 (Control Room Fire) has been entered.

Which ONE of the following describes the appropriate actions per the AOP?

A. Initiate a RPCB Loss of Feed Pump from B04.

B. Initiate a boration from the Remote Shutdown Panel.

C. Trip the Reactor by opening the RTSG breakers locally.

D. Trip the Reactor by depressing the RTSG Pushbuttons on B05.

Answer: D

Reference Id:	Q43941	
Difficulty:	2.00	
Time to complete:	2	
10CFR Category:	CFR 55.41 (10)	55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.
Cognitive Level:	Memory	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40AO-9ZZ19 (Control Room Fire)

K&A: Knowledge of the interrelations between the Control Room Evacuation and the following: Reactor trip system.

Learning Objective: State the operator actions that are required to be performed prior to evacuation in the event of a Control Room fire.

Justification:

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A. Incorrect: Numerous AOPs use the RPCB Loss of Feed Pump as a means of a rapid downpower.
B. Incorrect: This is the correct action if after the trip is initiated from the CR, and a CEA doesn't fully insert into the core.
C. Incorrect: Tripping the Reactor is the correct direction, just not the location. Candidate may think that due to the CR Fire that all actions must be taken outside of the control room.
D. Correct: Per the not prior to and including Step 2a of the AOP, Steps 2-5 are expected to be performed in the control room and 2a. states Trip the Reactor.

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24.

This Exam Level	RO
Appears on:	RO EXAM 2012 Tier 1 Group 2
K/A #	4.2 069 AA2.01
Importance Rating:	3.7

Which of the following combination of valves failing **OPEN** would cause a loss of Containment Integrity per Technical Specifications?

- CHB-UV-515 (LETDOWN TO REGEN HX ISOL VLV UV-515)
- CHA-UV-516 (LETDOWN TO REGEN HX ISOL VLV UV-516)
- CHB-UV-523 (REGEN HX OUTLET ISOLATION VLV UV-523)
- CHA-HV-524 (CHARGING PUMPS DSCH HDR TO REGEN HX VLV HV-524)

A. CHB-UV-515 and CHB-UV-523.

B. CHB-UV-515 and CHA-HV-524.

C. CHA-UV-516 and CHB-UV-523.

D. CHA-UV-516 and CHA-HV-524.

Answer: C

Reference Id:	Q43942	
Difficulty:	2.00	
Time to complete:	2	
10CFR Category:	CFR 55.41 (10)	55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.
Cognitive Level:	Memory	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: LOIT Lesson Plan, Tech Specs/TRM

K&A: Ability to determine and interpret the following as they apply to the Loss of Containment Integrity: Loss of containment integrity

Learning Objective: Explain the operation of the following Letdown Isolation Valves, including their Control Room controls, under normal operating conditions:

- Letdown Line to Regen Heat Exch Vlv (CHB-UV-515)
- Letdown Line to Regen Heat Exch Vlv Containment Isolation Vlv (CHA-UV-516)
- Regen Heat Exch to Letdown Heat Exch Containment Isolation Vlv (CHB-UV-523)

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Justification: It requires 2 valves in a penetration to have a loss of Containment Integrity. As long as one valve in a penetration is closed CI is maintained.

a) Incorrect: CHB-UV-515 does not receive a close signal on CIAS and is not listed in the TRM T7.0.300 table .
b) Incorrect: CHB-UV-515 does not receive a close signal on CIAS and CHA-HV-524 and is not listed in the TRM T7.0.300 table
c) Correct: CHA-UV-516 and CHB-UV-523 are the Only Tech Spec Isolation valves in the letdown system. These valves receive a close signal on CTMT ISOL CIAS) and are listed in the TRM T7.0.300 table.
d) Incorrect: CHB-UV-516 is a TS CTMT ISOL VLV, CHA-HV-524 and is not listed in the TRM T7.0.300 table

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25.

This Exam Level	RO
Appears on:	RO EXAM 2008 RO EXAM 2012 Tier 1 Group 2
K/A #	4.4 A16 AK3.3
Importance Rating:	3.3

Given the following conditions:

- Unit 1 is operating at 100% power.
- Pressurizer level is slowly lowering.
- RCS temperature is stable.
- The in-service letdown control valve CHN-110P is slowly closing.
- The CRS implements 40AO-9ZZ02, Excessive RCS Leakrate.
- All available charging pumps are running.
- Pressurizer level continues to lower.

40AO-9ZZ02 (Excessive RCS Leakrate) now directs...

A. isolating letdown to quantify leakage for E-plan classification.

B. an immediate reactor trip to minimize dose rates at the site boundary.

C. an immediate reactor trip due to leakage is excess of Tech Spec limits.

D. isolating letdown to determine if leakage exceeds CVCS makeup capacity.

Answer: D

Reference Id:	Q22453	
Difficulty:	3.00	
Time to complete:	4	
10CFR Category:	CFR 55.41 (10)	55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.
Cognitive Level:	Comprehension / Anal	
Question Source:	PV Bank Not Modified	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40AO-9ZZ02, Excessive RCS Leakrate

K&A: Knowledge of the reasons for the following responses as they apply to the (Excess RCS Leakage) Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations.

Learning Objective: Given indications of RCS or a Steam Generator Tube Leak, describe the basic procedure methodology, including Reactor Trip is thresholds, in accordance with 40AO-9ZZ02.

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Justification:

A. Incorrect: The E-plan numbers are determined by performing appendix A/B of 40AO-9ZZ02.
B. Incorrect: Tripping the Reactor is determined as thresholds are exceeded after completing the next step to isolate letdown then trip if Pzr level continues to lower.
C. Incorrect: TS limits are defined and if not met to be in mode 3 within 6 hours, not to trip immediately. Candidate may think the TS limits are trip thresholds. The next step is to isolate letdown then trip if Pzr level continues to lower.
D. Correct: Isolating letdown eliminates the Letdown system as a possible location of the leak, Plant operation is allowed if the leak is isolated as exhibited by the restoration of Pzr Level. The step of the procedure is to isolate letdown and determine if CVCS makeup capability is exceeded if so then trip reactor

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26.

This Exam Level	RO
Appears on:	RO EXAM 2012 Tier 1 Group 2
K/A #	4.4 A13 AK1.2
Importance Rating:	3.2

Given the following conditions:

- Unit 1 has been in a Blackout condition for 3 hours.
- The crew is performing actions of 40EP-9EO08 (Blackout).
- PBA-S03 has been energized by ONE Station Blackout Generator (SBOG) per Standard Appendix 80.
- Attempts to restore power from other sources have been unsuccessful.
- Natural circulation flow **CAN NOT** be verified.

In accordance with the Blackout procedure, which ONE of the following describes the action(s) that will be taken by the crew?

A. Use Auxiliary Spray to lower RCS pressure.

B. Commence a cooldown to shutdown cooling entry conditions.

C. ENSURE Train "A" ADVs are throttled adequately to maintain RCS subcooling.

D. OVERRIDE and ENERGIZE Train "A" class backup heater to stabilize RCS pressure.

Answer: C

Reference Id:	Q43811	
Difficulty:	4.00	
Time to complete:	3	
10CFR Category:	CFR 55.41 (10)	55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.
Cognitive Level:	Memory	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: Steam Tables

Technical Reference: 40EP-9EO08, BLACKOUT / 40DP-9AP13. BO Tech Guideline

K&A: Knowledge of the operational implications of the following concepts as they apply to the (Natural Circulation Operations) Normal, abnormal and emergency operating procedures associated with (Natural Circulation Operations).

Learning Objective: Given conditions of a Blackout state the action necessary to maintain subcooling margin in accordance with 40EP-9EO08.

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Justification:

A. Incorrect - Lowering RCS pressure will cause subcooled margin to lower, which will not promote natural circulation conditions.
B. Incorrect - This step is not required be performed unless AC power is not restored. PBA-S03 has been energized with a SBOG.
C. Correct - Per Step 21 Blackout EOP, if the conditions are met, ENSURE proper control of steam generator steaming and feeding.
D. Incorrect - Raising pressure would improve subcooling and promote natural circulation conditions. But Pressurizer Level is below the heater cutout setpoint, therefore Heaters are not available.

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27.

This Exam Level:	RO
Appears on:	RO EXAM 2007 RO EXAM 2012 Tier 1 Group 1
K/A #:	4.4 E09 EA1.1
Importance Rating:	4.2

Given the following conditions:

- The Unit 2 CRS has entered the Functional Recovery procedure.
- RWT level is 6.4%.
- You have been directed to verify proper Recirculation Actuation Signal (RAS).

Which ONE of the following actions must be manually performed given a proper "A" train RAS actuation?

A. Stop SIA-P01, LPSI pump A

B. Close SIA-UV-666, HPSI A pump Recirc valve

C. Open SIA-UV-674, Cntmt Sump to Safety Injection Valve

D. Close CHA-HV-531, RWT to Train A Safety Injection Valve

Answer: D

Reference Id:	Q10333	
Difficulty:	3.00	
Time to complete:	2	
10CFR Category:	CFR 55.41 (7)	55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Cognitive Level:	Memory	
Question Source:	PV Bank Not Modified	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40EP-9EO09 (FRP) 40AO-9ZZ17 (Inadvertant PPS actuations)

K&A: Ability to operate and / or monitor the following as they apply to the (Functional Recovery) Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Learning Objective: Given the FRP is being performed and IC is in progress describe how the FRP will maintain or recover the Inventory Control Safety Function in accordance with 40EP-9EO09.

Justification:

A. Incorrect: LPSI pump are tripped on a RAS actuation.
B. Incorrect: All SI miniflow valves close on RAS actuation.
C. Incorrect: RAS sump isolation valves open on RAS actuation.
D. Correct: RWT isolation valves must be manually operated on RAS actuation.

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28.

This Exam Level	RO
Appears on:	RO EXAM 2008
	RO EXAM 2012
	Tier 2 Group 1
K/A #	3.4 003 A1.05
Importance Rating:	3.4

Given the following conditions:

- Unit 1 is operating at 100% power.
- RCP 1A experiences a failure causing it to slow down at 1% per minute.

Assuming that all other input parameters remained the same, the CPC calculated value of DNBR will ...

A. not change until RCP speed reaches 95% of rated speed, then a DNBR trip will occur.

B. not change until RCP speed reaches 95% of rated speed, then an Auxiliary trip will occur.

C. gradually lower until RCP speed reaches 95% of rated speed, then a DNBR trip will occur.

D. gradually lower until RCP speed reaches 95% of rated speed, then an Auxiliary trip will occur.

Answer: C

Reference Id: Q44016
Difficulty: 3.00
Time to complete: 3
10CFR Category: CFR 55.41 (7) 55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Cognitive Level: Comprehension / Anal
Question Source: PV Bank Not Modified
Comment:
Proposed reference to be provided to applicant during examination: NONE

Technical Reference: LOIT lesson plan

K&A: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RCPs controls including: RCS flow

Learning Objective: L77427 Describe the function of the Reactor Coolant Pump Speed inputs to the Core Protection Calculators.

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Justification:

A. Incorrect: Pump Speed is input to the Flow calculation which is used in the DNBR calculation. DNBR will reduce as speed drops. A DNBR trip will be generated when RCP speed reaches 95%.
B. Incorrect: Pump Speed is input to the Flow calculation which is used in the DNBR calculation. DNBR will reduce as speed drops. The auxiliary trip monitoring RCPs is generated when less than 2 RCPs are running.
C. Pump Speed is input to the Flow calculation which is used in the DNBR calculation. DNBR will reduce as speed drops. A DNBR trip will be generated when RCP speed reaches 95%.
D. Incorrect: DNBR will reduce as speed drops then generate a DNBR trip.The auxiliary trip monitoring RCPs is generated when less than 2 RCPs are running.

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29.

This Exam Level:	RO
Appears on:	RO EXAM 2005 RO EXAM 2012
K/A #:	3.4 003 K6.04
Importance Rating:	2.8

Given the following conditions:

- Nuclear Cooling Water (NC) has been lost due to a pipe rupture.
- Train **'B'** Essential Cooling Water (EW) has been cross-connected to NC.

Which ONE of the following describes a condition that will isolate **'B'** Essential Cooling Water to the RCPs?

A. Containment pressure rises to 9.0 psig.

B. Pressurizer pressure drops to 1800 psia.

C. Instrument air header pressure drops to 60 psig.

D. 'B' EW Surge Tank level drops to LO LEVEL setpoint.

Answer: A

Reference Id: Q43945
Difficulty: 3.00
Time to complete: 2
10CFR Category: CFR 55.41 (8) 55.41 (8) Components, capacity, and functions of emergency systems.
Cognitive Level: Comprehension / Anal
Question Source: PV Bank Not Modified
Comment:
Proposed reference to be provided to applicant during examination: NONE

Technical Reference: LOIT Lesson Plans, 40AO-9ZZ17 (Inadvertant PPS-ESFAS Actuations)

K&A: Knowledge of the effect of a loss or malfunction on the following will have on the RCPs: Containment isolation valves affecting RCP operation.
Learning Objective: Describe the automatic features associated with the NC Containment Isolation Valves.
Justification:

A. Correct: Containment Spray Actuation Signal (CSAS) at 8.5 psig will close the CTMT Isolation Valves for the NC system which are downstream of the EW cross tie valves.
B. Incorrect: EW 'A' will isolate on SIAS EW 'B' cross tie valves are manually operated valves with no automatic features.
C. Incorrect: NC and EW valves are Motored Operated valves, the degraded Instrument Air Header pressure will not effect EW to RCPs. 40AO-9ZZ06 (Loss of IA) describes hundreds of components that are effected by the lowering IA header pressure.
D. Incorrect: EW 'A' will isolate on LO 'A' EW Surge Tank Level. EW 'B' cross tie valves are manually operated valves with no automatic features.

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30.

This Exam Level:	RO
Appears on:	RO EXAM 2012
	Tier 2 Group 1
K/A #:	3.2 004 K1.04
Importance Rating:	3.4

Given the following conditions:

- Unit 3 is operating at 100% power.
- All RCP seal injection controllers (CHN-FIC-241-244) are in automatic.
- The output **SIGNAL** of CHN-FIC-241, 1A RCP controller, is rising.
- Disregard the response of the remaining Seal Injection controllers.

Which ONE of the following describes the cause?

A. NNN-D11 is de-energized.

B. Inadvertent CSAS actuation.

C. Actual Seal Injection flow is below setpoint.

D. Regenerative Heat Exchanger outlet temperature has exceeded 413ºF.

Answer: B

Reference Id:	Q10468	
Difficulty:	3.00	
Time to complete:	3	
10CFR Category:	CFR 55.41 (7)	55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Cognitive Level:	Comprehension / Anal	
Question Source:	PV Bank Not Modified	

Comment:

Proposed reference to be provided to applicant during examination: NONE
Technical Reference: LOIT Lesson Plans

K&A: Knowledge of the physical connections and/or cause-effect relationships between the CVCS and the following systems: RCPS, including seal injection flows.

Learning Objective: L68108 Explain the operation of the RCP Seal Injection Flow Control Valves (CHE-FV-241,242,243, and 244), including their Control Room controls, under normal operating conditions.

Justification:

A. Incorrect: Loss of NNN-D11 will de-energize the controller therefore the output will be failed as is.

B. Correct: CSAS actuation will isolate IA to the Containment and valves will slowly open, therefore controller will try to lower flow by raising output. These controllers are reverse acting.

C. Incorrect: Actual Flow less than setpoint will cause the controller output to lower. Reverse Acting Controller.

D. Incorrect: This will provide a close signal to CHB-UV-515, This Loss of Letdown will not effect seal injection flow.

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31.

This Exam Level: RO
Appears on: RO EXAM 2012
Tier 2 Group 1
K/A #: 3.4 005 K3.07
Importance Rating: 3.2

Given the following plant conditions:

- Refueling pool level is 137' 6' (>23 ft above the vessel flange).
- Core RE-LOAD is in progress.
- An irradiated fuel assembly is grappled and in the hoist box.
- Train 'B' is under clearance for maintenance.
- Train 'A' LPSI pump is gas bound.

Which ONE of the following complies with Technical Specifications 3.9.4 (Shutdown Cooling (SDC) and Coolant Circulation - High Water Level) required actions ?

A. Core re-load may continue.

B. Immediately stop core re-load, leave the fuel assembly in the hoist box.

C. Complete placing the fuel assembly in its designated core location, then suspend core re-load.

D. Immediately stop core re-load until you have verification that all activities that could result in boron dilution have been suspended.

Answer: B

Reference Id: Q43947
Difficulty: 4.00
Time to complete: 2
10CFR Category: CFR 55.41 (10) 55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.
Cognitive Level: Comprehension / Anal
Question Source: New
Proposed reference to be provided to applicant during examination: ~~NONE~~ Technical Specification 3.9.4 [Approved as closed-reference but administered as open-reference. See Exam Report for details.]
Technical Reference: Technical Specifications 3.9.4 (Shutdown Cooling (SDC) and Coolant Circulation - High Water Level) and Basis.
K&A: Knowledge of the effect that a loss or malfunction of the RHRS will have on the following: Refueling operations.
Learning Objective: L94060 Given a set of plant conditions identify whether or not LCO 3.9.4 is satisfied and any actions or surveillance requirements that would prevent core alterations per Tech Spec 3.9 and its Basis.
Justification:
A. Incorrect: Core Off Load would be permitted in this instance but Core Re Load would add energy to the core.
B. Correct: Per TS 3.9.4 One SDC Cooling Loop shall be operable and in operation. The fact that B has no power and A is gas bound Condition A is not met and loading irradiated fuel must be suspended immediately.
C. Incorrect: The fuel assembly would be placed back in its original position in the Spent Fuel Pool not the Core.
D. Incorrect: Immediately suspending core reload is correct but once the boron concentration reduction is verified to not exist you may not restart the core re load.

3.9 REFUELING OPERATIONS

3.9.4 Shutdown Cooling (SDC) and Coolant Circulation — High Water Level

LCO 3.9.4 One SDC loop shall be OPERABLE and in operation.

---------------------------NOTE---------------------------
The required SDC loop may be removed from operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause reduction of the Reactor Coolant System boron concentration.

APPLICABILITY: MODE 6 with the water level ≥ 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDC loop requirements not met.	A.1 Suspend operations involving a reduction in reactor coolant boron concentration.	Immediately
	AND	
	A.2 Suspend loading irradiated fuel assemblies in the core.	Immediately
	AND	
	A.3 Initiate action to satisfy SDC loop requirements.	Immediately
	AND	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.4	Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify one SDC loop is operable and in operation.	12 hours

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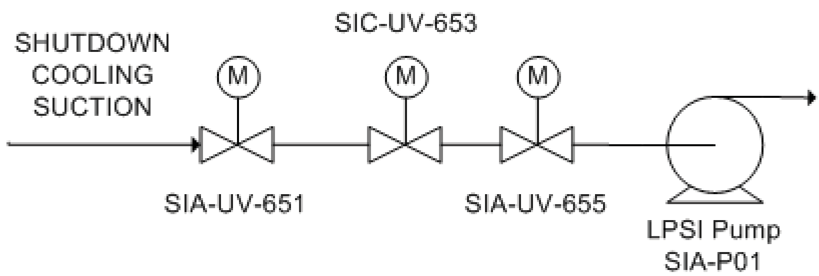
32.

This Exam Level:	RO
Appears on:	RO EXAM 2012
	Tier 2 Group 1
K/A #:	3.2 006 K1.03
Importance Rating:	4.2

Given the following:



RCS Low Temperature Overpressure protection is provided by the LTOPs. Which ONE of the following describes the Train 'A' LTOP?

Train 'A' LTOP is located between valves ____(1)____ and ____(2)____, and lifts at ____(3)____ psig.

A. (1) SIA-UV-651 (2) SIC-UV-653 (3) 410

B. (1) SIC-UV-653 (2) SIA-UV-655 (3) 410

C. (1) SIA-UV-651 (2) SIC-UV-653 (3) 467

D. (1) SIC-UV-653 (2) SIA-UV-655 (3) 467

Answer: D

Reference Id: Q22618
Difficulty: 3.00
Time to complete: 2
10CFR Category: CFR 55.41 (3) 55.41 (3) Mechanical components and design features of the reactor primary system.
Cognitive Level: Comprehension / Anal
Question Source: PV Bank Not Modified
Comment:

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: LOIT Lesson Plan

K&A: Knowledge of the physical connections and/or cause-effect relationships between the ECCS and the following systems: RCS

Learning Objective: Describe the design characteristics of SI system valves.

Justification:

A. Incorrect: Wrong Location and setpoint. 410 psia is the setpoint for the SIT isolation valves.
B. Incorrect: Correct Location wrong setpoint. 410 psia is the setpoint for the SIT isolation valves.
C. Incorrect: Wrong Location and correct setpoint
D. Correct: This is the Correct Location and setpoint.

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33.

This Exam Level	RO
Appears on:	RO EXAM 2012 Tier 2 Group 1
K/A #	3.5 007 A2.05
Importance Rating:	3.2

Given the following conditions:

- Unit 2 is at 100% power.
- PSV-203 (PZR safety valve) has seat leakage.
- Reactor Drain Tank (RDT) level is rising.
- RDT pressure is 7.5 psig and rising slowly.
- Window 3A07A (REAC DRN LOOP TRBL) is alarming.
- Window 3A07B (REAC DRN TK PRESS HI) is alarming.

Which one of the following conditions is correct?

The Alarm Response procedure (40AL-9RK3A) directs the crew to vent the RDT to ....

A. containment before it isolates at 10 psig.

B. containment before it ruptures at 10 psig.

C. the gas surge header before it isolates at 10 psig.

D. the gas surge header before it ruptures at 10 psig.

Answer: C

Reference Id: Q43950
Difficulty: 2.00
Time to complete: 52
10CFR Category: CFR 55.41 (3) 55.41 (3) Mechanical components and design features of the reactor primary system.
Cognitive Level: Comprehension / Anal
Question Source: New
Comment:
Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40AL-9RK4A (B04 ARP)

K&A: 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: Exceeding PRT high-pressure

Learning Objective: Describe automatic functions associated with the following Reactor Drain Tank Valves:• CHA-UV-560 (Reactor Drain Tank Outlet Isolation Valve)• CHB-UV-561 (Reactor Drain Tank Outlet Isolation Valve)• CHN-UV-540 (Reactor Drain Tank Vent Valve)• CHA-UV-580 (Reactor Drain Tank Makeup Supply Isolation Valve).

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Justification:

A. Incorrect: Between 5 and 10 psig the ARP directs venting to the Gas Surge header, the RDT does isolate at 10 psig.
B. Incorrect: Between 5 and 10 psig the ARP directs venting to the Gas Surge header, and the RDT isolates at 10 psig not ruptures.
C. Correct: Between 5 and 10 psig the ARP directs venting to the Gas Surge header, and the RDT isolates at 10 psig. CHN-UV-540 (Gas Surge Header) and CHA-UV-560 (containment isolation) both close.
D. Incorrect: Between 5 and 10 psig the ARP directs venting to the Gas Surge header, but the RDT isolates at 10 psig not ruptures.

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34.

This Exam Level	RO
Appears on:	RO EXAM 2007
	RO EXAM 2012
	Tier 2 Group 1
K/A #	3.8 008 K2.02
Importance Rating:	3.0

Given the following conditions:

- Unit 1 is operating at 100% power.
- NCN-P01A (NCW PUMP A) is in operation with NCN-P01B (NCW PUMP B) in standby.
- The A Emergency Diesel Generator is under permit for maintenance.
- NBN-X03 ESF Service Transformer fails.
- This loss does NOT result in a Reactor Trip.

Based on these conditions, the Nuclear Cooling Water system will...

A. have no pumps running.

B. be unaffected (no change in pump operation).

C. remain in operation, NCN-P01B running and NCN-P01A off.

D. remain in operation, with both NCN-P01A and NCN-P01B running.

Answer: B

Reference Id:	Q5794	
Difficulty:	3.00	
Time to complete:	3	
10CFR Category:	CFR 55.41 (7)	55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Cognitive Level:	Comprehension / Anal	
Question Source:	PV Bank Not Modified	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: LOIT Lesson Plan

K&A: Knowledge of bus power supplies to the following: CCW Pump, including emergency backup.

Learning Objective: Explain the operation of the NC Pumps under normal operating conditions.

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Justification:

A. Incorrect: Candidate may think that the NCW pumps are powered from PB buses and may think this situation has resulted in a loss of power to both.

B. Correct: NCW pumps are powered from non-class 4160v busses NBN-S01 and NBN-S02. Losing transformer NBN-X03 with the A Diesel Generator tagged out will result in a loss of Class 4160v power on the A train, but will not affect power to the NCW pumps.

C. Incorrect: May think that PBA has lost power and NCW A with it, NCW B would start on low header pressure.

D. Incorrect: May think that the power transfer from off site to the EDG would result in both pumps running.

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35.

This Exam Level:	RO
Appears on:	RO EXAM 2012 Tier 2 Group 1
K/A #:	3.3 010 K5.01
Importance Rating:	3.5

Given the following conditions:

- Unit 2 has experienced a LOCA.
- PZR Temperature is 470°.
- PZR Pressure is 300 psia.

Which ONE of the following describes the state of the PZR fluid?

A. Saturated water

B. Saturated steam

C. Subcooled water

D. Superheated steam

Answer: D

Reference Id:	Q43952	
Difficulty:	2.00	
Time to complete:	2	
10CFR Category:	CFR 55.41 (14)	55.41 (14) Principles of heat transfer thermodynamics and fluid mechanics.
Cognitive Category:	Comprehensive/Anal	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: Steam Tables

Technical Reference: Steam Tables

K&A: Knowledge of the operational implications of the following concepts as the apply to the PZR PCS: Determination of condition of fluid in PZR, using steam tables

Learning Objective: Given conditions of LOCA analyze RCS Heat Removal to determine if the SFSC acceptance criteria is satsified in accordance with 40EP-9EO03.

Justification:

A. Incorrect: The temperature at which water first begins to evaporate into steam, while under a given pressure, such as 14.7 psia (atmospheric pressure), is known as the saturation temperature.Water at this temperature and pressure configuration is said to be “saturated water.”
B. Incorrect: The mixture is defined as saturated vapor (or dry vapor) at the point when the last of the liquid has boiled off and becomes vapor. The further addition of heat will now increase the temperature of the vapor.

C. Incorrect: A subcooled liquid or compressed liquid is defined as a liquid existing to the left of the saturated liquid line.
D. Correct: Superheated vapor is defined as vapor heated beyond the saturated Vapor State

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36.

This Exam Level:	RO
Appears on:	RO EXAM 2012 Tier 2 Group 1
K/A #:	37 012 K4.06
Importance Rating:	3.2

The DNBR/LPD Reactor Protection System Operational Bypass is inserted ____(1)____ when the Excore NI Power decreases below ____(2)____ %

A. (1) manually (2) 1E-2%.

B. (1) manually (2) 1E-4%.

C. (1) automatically (2) 1E-2%.

D. (1) automatically (2) 1E-4%.

Answer: B

Reference Id:	Q43995	
Difficulty:	3.00	
Time to complete:	2	
10CFR Category:	CFR 55.41 (7)	55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Cognitive Level:	Memory	
Question Source:	PV Bank Not Modified	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: LOIT Lesson Plan

K&A: Knowledge of RPS design feature(s) and/or interlock(s) which provide: Automatic or manual enable/disable of RPS trips

Learning Objective: L77084 Plant Protection System, Describe the RPS operating bypasses.

Justification:

A. Incorrect: It is inserted manually but is enabled below 1E-4%. 1E-2% is the Log Power Bypass.
B. Correct:The bypass must be manually inserted from key switches at the remote CPC modules on B05 when ex-core safety channel NI power is less than 10-4% power.
C. Incorrect: It is inserted manually. 1E-2% is the Log Power Bypass.
D. Incorrect: It is inserted manually.

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37.

This Exam Level:	RO
Appears on:	RO EXAM 2012
	Tier 2 Group 1
K/A #:	3.7 012 A4.04
Importance Rating:	3.3

Given the following conditions:

- Unit 1 is operating at 100% power
- Channel 'D' PPS HI PZR PRESS is **BYPASSED** due to a failed high RCS pressure (Narrow Range) transmitter.
- Channel 'B' PPS SG-2 level low has **TRIPPED** due to failed transmitter.
- Channel 'A' RCS pressure (Narrow Range) transmitter now **FAILS HIGH**.

Based on these conditions, which ONE of the following is correct?

A. The operator can NOT physically bypass channel 'A' HI PZR PRESS bistable.

B. The reactor would have tripped when the channel 'A' pressure transmitter failed.

C. 2 Reactor Trip Circuit Breakers (RTCBs) would open when the channel 'A' RCS pressure transmitter failed, but the reactor would not trip.

D. If the operator bypasses the 'A' HI PZR PRESS bistable, that channel would go into bypass, while removing the channel 'D' HI PZR PRESS bistable from bypass.

Answer: D

Reference Id:	Q43953	
Difficulty:	2.00	
Time to complete:	2	
10CFR Category:	CFR 55.41 (5)	55.41 (5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.
Cognitive Level:	Comprehension / Anal	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: LOIT lesson plan

K&A: Ability to manually operate and/or monitor in the control room: Bistable, trips, reset and test switches.

Learning Objective: L77088 Describe the RPS Trip Channel bypass interlock.

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Justification:
An electrical interlock prevents the operator from bypassing more than one trip channel at a time for any one type of trip.
Different type trips may be bypassed simultaneously, either in one channel or in different channels.
Attempting to insert a trip channel bypass in a second channel for the same type of trip will result in only the Highest priority channel being in bypass, with A being the highest, and D the lowest priority. If “C” channel Pressurizer pressure had tripped and was bypassed and “A” or “B” channel was subsequently bypassed, “C” would come out of bypass and trip.

A. Incorrect: The operator CAN bypass the A RCS Press Transmitter.
B. Incorrect: In this case the coincidence is 2/3 with the D channel bypassed. 2/4 is the normal coincidence which would result in a trip.
C. Incorrect: This will not result in any RTSG breakers opening. RTSG breakers do not open on the specific parameter, only the Channel trip.
D. Correct: Per the explanation above, the hierarchy of the system would cause the D channel to come out of bypass when the A channel is placed in bypass.

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38.

This Exam Level:	RO
Appears on:	RO EXAM 2007
	RO EXAM 2012
	Tier 2 Group 1
K/A #:	3.2 013 A4.03
Importance Rating:	3.9

Given the following conditions:

- Unit 1 is operating at 100% power.
- The CRS directs an RO to initiate a MSIS from the Aux Relay Cabinets.
- The RO performs the following actions:
  - Depresses the 1-3 and 2-4 MSIS trip pushbuttons simultaneously on the "A" train.
  - Depresses the 1-3 and 2-4 MSIS trip pushbuttons sequentially (push then release) on the "B" train.

Assuming that SG pressures remains above the MSIS setpoint, you would expect an "A" train MSIS full initiation with...

A. no initiation of the "B" train, "A" MSIS can be reset by depressing either reset pushbutton.

B. a half leg initiation of the "B" train, "A" MSIS can be reset by depressing either reset pushbutton.

C. no initiation of the "B" train, "A" MSIS can only be reset by depressing both reset pushbuttons simultaneously.

D. a half leg initiation of the "B" train, "A" MSIS can only be reset by depressing both reset pushbuttons simultaneously.

Answer: A

Reference Id:	Q44012	
Difficulty:	4.00	
Time to complete:	3	
10CFR Category:	CFR 55.41 (7)	55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Cognitive Level:	Memory	
Question Source:	PV Bank Not Modified	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 73ST-9DG01(ISG testing)

K&A: Knowledge of ESFAS design feature(s) and/or interlock(s) which provide for the following: Main Steam Isolation System.

Learning Objective: Describe how an ESFAS subsystem can be manually actuated and manually reset from the Aux Relay Cabinets.

Justification:

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A. Correct: To initiate an ESFAS actuation both buttons must be pushed sim. pushing and releasing gives no initiation half leg or otherwise, power is still available to all relays. Resetting requires that either reset button on the train be depressed.
B. Incorrect: No initiation of the B train will occur, the MSIS can be reset by pushing either Aux Relay Cabinet Pushbutton.
C. Incorrect: No initiation is correct for the B train, but you don't have to press both Aux Relay Cabinet Pushbuttons to reset.
D. Incorrect: No initiation of the B train will occur, but you don't have to press both Aux Relay Cabinet Pushbuttons to reset.

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39.

This Exam Level:	RO
Appears on:	RO EXAM 2012 Tier 2 Group 1
K/A #:	3.5 022 A4.01
Importance Rating:	3.6

Given the following conditions:

- Unit 3 is operating at 100% power.
- An Inadvertent SIAS has occurred.

Which ONE of the following describes the status of the Containment Normal ACUs?

The Containment Normal ACUs...

A. continue to run.

B. are load shed and must be manually started by an operator.

C. are load shed and will sequence back on after 120 seconds.

D. shift to take suction on elevations 100' and below in containment.

Answer: B

Reference Id:	Q43955	
Difficulty:	4.00	
Time to complete:	3	
10CFR Category:	CFR 55.41 (7)	55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Cognitive Level:	Memory	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE
Technical Reference: 40AO-9ZZ17(Inadvertent PPS ESFAS), LOIT Lesson Plans

K&A: Ability to manually operate and/or monitor in the control room: CCS fans

Learning Objective: Describe the automatic functions associated with the Containment Building Normal ACU Fans (HCN-A01-A, B, C, & D) .
Justification:

A. Incorrect: Not all HVAC system respond to a SIAS, the AUX Building HVAC system does not respond to a SIAS.
B. Correct: This is correct, the Containment Normal ACUs will receive a Load Shed signal on the SIAS and need to be manually restarted by a operator.
C. Incorrect: The Load Shed portion is correct but the 120 Seconds is the time delay associated with the CEDM ACUs.
D. Incorrect: On a SIAS the Fuel Building HVAC system will shift suctions to the Aux Building 100 foot elevation and below.

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40.

This Exam Level: RO
Appears on: RO EXAM 2012
Tier 2 Group 1
K/A #: 3.5 026 A3.01
Importance Rating: 4.3

Given the following conditions:

- Unit 1 has tripped from 100% power due to a LOCA.
- PZR Pressure is 1500 psia and lowering.
- CTMT Pressure is 6.5 psia and rising.

Which ONE of the following describes the **CURRENT** status of the Containment Spray System?

The CTMT Spray Pumps are ____(1)____ and the CTMT Spray Header Isolation Valves are ____(2)____.

A. (1) off (2) open

B. (1) off (2) closed

C. (1) running (2) open

D. (1) running (2) closed

Answer: D

Reference Id: Q43956
Difficulty: 4.00
Time to complete: 5
10CFR Category: CFR 55.41 (8) 55.41 (8) Components, capacity, and functions of emergency systems.
Cognitive Level: Comprehension / Anal
Question Source: New
Comment:
Proposed reference to be provided to applicant during examination: NONE
Technical Reference: LOIT Lesson Plans
K&A: Ability to monitor automatic operation of the CSS, including: Pump starts and correct MOV positioning.

Learning Objective: Describe the design characteristics of SI system valves.

Justification:

A. Incorrect: Candidate may think the valves open on SIAS and the pump starts on CSAS.The CS pumps start on SIAS when the RCS pressure is <1837 psia or the CTMT > 3.0 psig. The CTMT Spray Header isolation valves do not open until the CSAS setpoint of 8.5 psig is met.
B. Incorrect: Candidate may think the CS Pump and Valves actuate at the CSAS setpoint.
C. Incorrect: Candidate may not know the CSAS setpoint of 8.5 psia and therefore the full CS actuation has occurred.
D. Correct: he CS pumps start on SIAS when the RCS pressure is <1837 psia or the CTMT > 3.0 psig. The CTMT Spray Header isolation valves do not open until the CSAS setpoint of 8.5 psig is met.

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41.

This Exam Level	RO
Appears on:	RO EXAM 2012
	Tier 2 Group 1
K/A #	3.4 039 K3.03
Importance Rating:	3.2

Given the following conditions:

- Unit 2 has tripped from 100% power.
- S/G #1 level is 23% WR and lowering rapidly.
- S/G #1 pressure is 900 psia and lowering rapidly.
- S/G #2 level is 45% WR and lowering slowly.
- S/G #2 pressure is 1050 psia and stable.

Subsequently

- S/G #1 level is 19% WR and lowering rapidly.
- S/G #1 pressure is 780 psia and lowering rapidly.
- S/G #2 level is 43% WR and lowering slowly.
- S/G #2 pressure is 1050 psia and stable.

Assuming NO operator action, AFA-P01 (Essential Turbine Driven Aux Feed Pump) is...

A. still in standby.

B. operating and aligned to receive steam from BOTH SGs.

C. operating and aligned to receive steam from SG #1 ONLY.

D. operating and aligned to receive steam from SG #2 ONLY.

Answer: B

Reference Id:	Q43957	
Difficulty:	3.00	
Time to complete:	3	
10CFR Category:	CFR 55.41 (4)	55.41 (4) Secondary coolant and auxiliary systems that affect the facility.
Cognitive Level:	Comprehension / Anal	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: LOIT Lesson Plan

K&A: Knowledge of the effect that a loss or malfunction of the MRSS will have on the following: AFW pumps.

Learning Objective: Explain the operation of the AFW Pump Turbine Main Steam Supply Valves (SGA-UV-134 and -138) under normal operating conditions.

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Justification:

a. Incorrect: Both MOVs will open on the AFAS signal that was received at 25.8% WR on the #1 SG. Candidate may not know the AFAS setpoint. Also, Candidate may think the D/P lockout of 185 psid will not allow the lower pressure SG to supply steam to AFA-P01.

b. Correct: Both Main Steam Supply valves AUTO open on an AFAS actuation, regardless of which SG has experienced the low level. In addition, the D/P lockout does NOT impact the operation of the steam supply valves.

c. Incorrect: Candidate may think only the SG that is below the AFAS setpoint will supply steam to AFA-P01.

d. Incorrect: Candidate may think only the SG that is INTACT will supply steam to AFA-P01 due to the D/P lockout.

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42.

This Exam Level: RO
Appears on: RO EXAM 2012
Tier 2 Group 1
K/A #: 3.4 059 A1.03
Importance Rating: 2.7

Which ONE of the following describes the operation of the Main Feedwater Pumps during a power ascension above 20% Power.

In accordance with 40OP-9ZZ05 (Power Operations) the second Main Feedwater Pump must be started prior to...

A. exceeding 60% reactor power.

B. placing 2nd stage reheat in service.

C. MFWP suction pressure lowering below 300 psia.

D. MFWP discharge pressure and SG pressure delta P dropping below 100 psid.

Answer: A

Reference Id: Q43958
Difficulty: 3.00
Time to complete: 2
10CFR Category: CFR 55.41 (4) 55.41 (4) Secondary coolant and auxiliary systems that affect the facility.
Cognitive Level: Memory
Question Source: New
Comment:

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40OP-9ZZ05 (Power Operations)

K&A: 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.

Learning Objective: L82548 Explain the operation of the MFWPs under normal operating conditions.

Justification:

A. Correct: Per NOTE after 4.3.43 the 2nd MFWP must be started to prevent damage to the 1st MFWP turbine.
B. Incorrect: The minimum suction pressure for the MFWP is 300 psig. This threshold has you start the 3rd condensate pump.
C. Incorrect:Placing the 2nd stage reheat is done after reaching 15% power. This is not a milestone for placing the 2nd MFWP in service. Starting a second MFWP would cause suction pressure to lower.
D. Incorrect: 100 psid is the lower limit at 100% power and is not a parameter used for starting a 2nd MFWP.

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43.

This Exam Level	RO
Appears on:	RO EXAM 2012
	RO EXAM 2005
	Tier 2 Group 1
K/A #	3.4 005 K5.03
Importance Rating:	2.9

Given the following conditions:

- Unit 2 is in Mode 5 following refueling.
- Shutdown Cooling in service using LPSI 'A'.
- It is desired to place SDC Train B in a "standby" SDC lineup.

Which ONE of the following describes what must be done prior to placing SDC train B in a standby lineup?

Recirculate SDC train B from the ....

A. RCS thru only SIB-UV-668 (LPSI Pump B Recirc valve).

B. RWT thru only SIB-UV-668 (LPSI Pump B Recirc valve).

C. RCS thru SIB-HV-690 (S/D Cooling Warmup Bypass valve) and SIB-UV-668 (LPSI Pump B Recirc valve).

D. RWT thru SIB-HV-690 (S/D Cooling Warmup Bypass valve) and SIB-UV-668 (LPSI Pump B Recirc valve).

Answer: D

Reference Id:	Q10202	
Difficulty:	3.00	
Time to complete:	3	
10CFR Category:	CFR 55.41 (10)	55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.
Cognitive Level:	Comprehension / Anal	
Question Source:	PV Bank Not Modified	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40OP-9SI01 (Shutdown Cooling Initiation)

K&A: Knowledge of the operational implications of the following concepts as they apply the RHRS: Reactivity effects of RHR fill water.

Learning Objective: L79915 Discuss the concerns with boron concentration associated with the Shutdown Cooling System.

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Justification: per the SI01 procedure prior to placing a SDC loop in standby it must first be recirculated with the RWT to equalize boron concentration and not cause SDM concern

a) Incorrect: This would recirculate the SDC loop with the RWT thru the miniflow. This is a valid alignment that is used to maintain RCS level during reduced inventory operations. This alignment is only used to maintain an already established level such as midloop.

b) Incorrect: This would recirculate the SDC loop with the RWT thru the miniflow but would not equalize the entire train and is not directed by the procedure.

c) Incorrect: This is a valid alignment for temperature control using the LP injection valves. This alignment is used prior to placing the SDC train in service NOT placing it in a standby alignment.

d) Correct: The Precautions and Limitations of the OP describe the fact that an Idle SDC loop may have a different boron concentration. This lineup is directed by the procedure to equalize boron concentration.

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44.

This Exam Level:	RO
Appears on:	RO EXAM 2012 Tier 2 Group 1
K/A #:	3.4 061 K4.02
Importance Rating:	4.5

Given the following conditions:

Initial Conditions:

- Unit 1 is in Mode 3 following an automatic reactor trip..
- AFN-P01 (Non-Essential Motor Driven Aux Feed Pump) is feeding both SGs at 350 gpm.
- AFB-P01 (Essential Motor Driven Aux Feed Pump) is in standby.
- AFA-P01 (Essential Turbine Driven Aux Feed Pump) is in standby.

Subsequently:

- Pressurizer pressure lowers to 1700 psia.

Which ONE of the following describes the status of the Auxiliary Feedwater System One minute after the Pzr Pressure reaches 1700 psia?

AFN-P01...

A. has tripped, AFB-P01 starts and feeds the SGs.

B. is running and feeding both SGs. AFB-P01 is in standby status.

C. has tripped, AFB-P01 starts but must be manually aligned to feed the SGs.

D. is running with its feedpath isolated, AFB-P01 starts but must be manually aligned to feed the SGs.

Answer: C

Reference Id:	Q44004	
Difficulty:	3.00	
Time to complete:	3	
10CFR Category:	CFR 55.41 (7)	55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Cognitive Level:	Comprehension / Anal	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 01-M-AFP-001 (Auxiliary Feedwater System Print)

K&A: Knowledge of AFW design feature(s) and/or interlock(s) which provide for the following: AFW automatic start upon loss of MFW pump, S/G level, blackout, or safety injection.

Learning Objective: Describe the Control Room controls associated with the Essential Auxiliary Feedwater Pump AFB-P01 including it's indications.

Justification:

A. Incorrect: AFN will trip on the load shed stop signal initiated by the SIAS, AFB does automatically start on the SIAS, only AFAS will open the Feed valves therefore AFB will not be feeding the SGs.
B. Incorrect: AFN will trip on the load shed stop signal initiated by the SIAS, therefore the no feed will be supplied to the SGs.
C. Correct: AFN will trip on the load shed stop signal initiated by the SIAS, AFB does automatically start on the SIAS, only AFAS will open the Feed valves therefore AFB will have to be manually aligned to feed the SGs.
D. Incorrect: AFN will trip on the load shed stop signal initiated by the SIAS, the downcomer isolations will remain open so the AFN feedpath is not isolated. AFB does automatically start on the SIAS, only AFAS will open the Feed valves therefore AFB will have to be manually aligned to feed the SGs.

45.

This Exam Level:	RO
Appears on:	RO EXAM 2012 Tier 2 Group 1
K/A #:	3.6 062 K4.03
Importance Rating:	2.8

Given the following list of conditions:

1. The BUS XFR SWITCH must be in AUTO.
2. A generator trip must have occurred.
3. The synchroscope must be on.
4. The synch check relay must be satisfied.
5. A Unit Aux Transformer trip must have occurred.
6. A lockout of the Normal Supply breaker must have occurred.

Which ONE of the following describes the conditions that must be met for an automatic Fast Bus Transfer of NAN-S01 to NAN-S03 to occur? This is not an all inclusive list.

A. 1, 2 and 4

B. 1, 4 and 6

C. 2, 3 and 6

D. 3, 4 and 5

Answer: A

Reference Id:	Q43959	
Difficulty:	3.00	
Time to complete:	2	
10CFR Category:	CFR 55.41 (7)	55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Cognitive Level:	Memory	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: LOIT Lesson Plans

K&A: Knowledge of ac distribution system design feature(s) and/or interlock(s) which provide for the following: Interlocks between automatic bus transfer and breakers

Learning Objective: Explain the operation of Switchgear NAN-S01 and NAN-S02 under normal operating conditions.

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Justification:

The NAN-S01 and NAN-S02 buses are designed with the ability to allow a fast bus transfer from the unit auxiliary transformer source to the NAN-S03 and NAN-S04 source. The feature allows the 13.8 kV bus loads to remain energized in the event of a loss of the main generator, the normal in-house supply. If the main turbine/generator trips at 100% power, the reactor can remain critical following the load rejection as the reactor coolant pumps remain powered. The sequence of events and associated interlocks that initiate a 13.8 kV NA fast bus transfer is listed below.
In order to allow a NA fast bus transfer, the manual/auto transfer switch on the control room B01 panel must be in auto.
The initiating event for a NA fast bus transfer is always a main generator trip . The activation of this lockout initiates the opening of the unit auxiliary transformer supply breakers, NAN-S01A and NAN-S02A.
An automatic sync check is performed between the NAN-S01 to NAN-S03 and NAN-S02 to NAN-S04 bus. If the two sources are in sync, this contact is closed.
Buses NAN-S03/S04 are checked for normal voltage and frequency.
Both the unit auxiliary supply breaker and the bus tie breakers are checked for tripped 86 lockout relays. If both are reset, the close signal is allowed to pass on to the bus tie breakers, NAN-S03B/S04B.

A. Correct: These 3 are required to have an automatic Fast Bus Transfer.
B. Incorrect: 1 and 4 are correct but 6 is not. Lockout on the normal supply breaker would prevent the FBT from occurring. Candidate may believe a UAT Trip is required vice a Main Turbine Trip.
C. Incorrect: 2 is correct, but 3 and 6 are not. Synch Check is automatically performed the synchroscope is not required for this check. Lockout on the normal supply breaker would prevent the FBT from occurring.
D. Incorrect: 4 is correct but 3 and 5 are not. Synch Check is automatically performed the synchroscope is not required for this check. UAT trip may be confused for the Main Turbine Trip requirement.

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46.

This Exam Level	RO
Appears on:	RO EXAM 2012
	Tier 2 Group 1
K/A #	3.6 062 A3.01
Importance Rating:	3.0

Given the following conditions:

- Unit 2 has tripped from 100% power.
- NAN-X01 (S/U XFMR #1) has faulted.
- SIAS has actuated.
- EDG 'A' is at 60.1 Hz and 4200 VAC.
- No 86 Lockouts on PBA-S03.
- Normal/Alternate Supply Breakers to PBA-S03 have operated as designed.

Which ONE of the following describes the status of the...

(1) EDG 'A' output breaker?
(2) NHN-M71 Energized/Not Energized?

A. (1) OPEN (2) ENERGIZED

B. (1) CLOSED (2) ENERGIZED

C. (1) CLOSED (2) NOT ENERGIZED

D. (1) OPEN (2) NOT ENERGIZED

Answer: C

Reference Id: Q43962
Difficulty: 3.00
Time to complete: 2
10CFR Category: CFR 55.41 (7) 55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Cognitive Level: Comprehension / Anal
Question Source: New
Comment:

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: LOIT Lesson Plan, Electrical Distribution Drawing

K&A: Ability to monitor automatic operation of the ac distribution system, including: Vital ac bus amperage

Learning Objective: Describe the Local and Control Room indications associated with the Class IE AC Electrical Distribution System.

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Justification:

NAN-X01 (Startup Transformer #1) is the normal supply to NAN-S05 which supplies PBA-S03 thru its associated ESF Transformer. This fault will cause an undervoltage condition on PBA-S03. Candidate may not know the S/U XFMR arrangement and believe that PBA-S03 is still being powered from off site power. EDG 'A' meets the requirements to automatically close in on PBA-S03 and power the bus. These requirements are Frequency between 59.9 and 60.5 Hz. Voltage between 4080 and 4300 Volts. No lockouts on the bus. Normal and Alternate supply breakers are open. Due to the EDG 'A' supplying PBA-S03 Amps will be indicated. NHN-M71 is a SIAS Load Shed Panel that will be de-energized due to the SIAS.

A. Incorrect: (1) EDG 'A' meets the requirements to automatically close in on PBA-S03 and power the bus. (2) Due to the EDG 'A' supplying PBA-S03 Amps will be indicated. (3) NHN-M71 is a SIAS Load Shed Panel that will be de-energized due to the SIAS.
B. Incorrect: (1) EDG 'A' meets the requirements to automatically close in on PBA-S03 and power the bus. (2) Due to the EDG 'A' supplying PBA-S03 Amps will be indicated. (3) NHN-M71 is a SIAS Load Shed Panel that will be de-energized due to the SIAS.
C. Correct: These are all correct.
D. Incorrect: (1) EDG 'A' meets the requirements to automatically close in on PBA-S03 and power the bus. (2) Due to the EDG 'A' supplying PBA-S03 Amps will be indicated. (3) NHN-M71 is a SIAS Load Shed Panel that will be de-energized due to the SIAS.

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47.

This Exam Level	RO
Appears on:	RO EXAM 2012 Tier 2 Group 1
K/A #	3.6 063 K2.01
Importance Rating:	2.9

Which ONE of the following valves are powered from a vital 125 VDC control center?

A. SIA-UV-644, SIT Isolation

B. SID-UV-654, Shutdown Cooling Isolation

C. SIE-HV-661, Combined SIT Drain to RDT

D. SIB-HV-690, Shutdown Cooling Loop 1 Warm-up Bypass

Answer: B

Reference Id:	Q43972	
Difficulty:	2.00	
Time to complete:	2	
10CFR Category:	CFR 55.41 (7)	55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Cognitive Level:	Memory	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: LOIT Lesson Plan

K&A: Knowledge of bus power supplies to major DC loads.

Learning Objective: Knowledge of major DC loads

Justification:

A. Incorrect: The SIT Isolation valves are powered by class 480v MCCs.
B. Correct: Class DC electrical distribution trains "C" and "D" provide power to the Shutdown Cooling Isolation Valves through inverters PKC-N43 and PKD-N44.
C. Incorrect:The SIT Drains are air operated.
D. Incorrect: The Shutdown Cooling Loop Warm-up Bypasses are powered class 480 v MCCs.

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48.

This Exam Level:	RO
Appears on:	RO EXAM 2007 RO EXAM 2012 Tier 2 Group 1
K/A #:	063 2.1.27
Importance Rating:	3.9

Given the following conditions:

- Unit 1 is in Mode 5.
- Battery Charger "A" (PKA-H11) has tripped.
- Battery Charger "AC" (PKA-H15) is connected to the "C" Battery bus (PKC-M43).

Can the "AC" Battery Charger be aligned to both PKA-M41 and PKC-M43 at this time?

A. YES, provided the Unit remains in Mode 5.

B. NO, a mechanical interlock prevents this alignment.

C. YES, provided that the "A" battery is disconnected from PKA-M41.

D. NO, this action may only occur while restoring the MVDC safety functions as implemented by the Lower Mode Functional Recovery Procedure.

Answer: B

Reference Id:	Q44002	
Difficulty:	2.00	
Time to complete:	2	
10CFR Category:	CFR 55.41 (7)	55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Cognitive Level:	Memory	
Question Source:	PV Bank Not Modified	

Comment:

Proposed reference to be provided to applicant during examination: NONE
Technical Reference: LOIT Lesson Plan
K&A: Knowledge of system purpose and or function: DC Electrical Distribution
Learning Objective: L74205 Explain the operation of the Class IE 125 VDC Battery Chargers under normal operating conditions.
PRA SIGNIFICANT QUESTION
Justification:

A. Incorrect: Tech Specs 3.8.1 do not allow for the class busses to be cross tied in Modes 1-4. Candidate may think that since the unit is in mode 5 this may not apply.
B. Correct: PVNGS has a mechanical interlock that prevents the Swing chargers from connecting to multiple DC buses simultaneously
C. Incorrect: Batteries are not allowed to be crosstied to the same bus, if the A battery was disconnected this would remove that obstacle to crosstying the busses, but the mechanical interlock is not disabled when the battery is disconnected from the bus.
D. Incorrect: LMFRP has many instances where DC busses are restored. Candidate may believe that the crosstying is one of them.

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49.

This Exam Level	RO
Appears on:	RO EXAM 2012
	Tier 2 Group 1
K/A #	3.6 064 K6.07
Importance Rating:	2.7

Given the following list of conditions:

- Unit 1 is operating at 100% power.
- The DG A right bank Starting Air Receiver is tagged out.
- There was an Inadvertent Containment Spray System Actuation.

The remaining left bank receiver and starting air subsystem will apply air to ____(1)____ diesel cylinder bank(s) and the diesel starts in the ____(2)_____ mode.

A. (1) both (2) Test Run

B. (1) both (2) Emergency

C. (1) only the left (2) Test Run

D. (1) only the left (2) Emergency

Answer: A

Reference Id:	Q43971	
Difficulty:	3.00	
Time to complete:	2	
10CFR Category:	CFR 55.41 (7)	55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Cognitive Level:	Comprehension / Anal	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: LOIT Lesson Plan

PRA SIGNIFICANT QUESTION

K&A: K6.07 Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: Air receivers

Learning Objective: Describe the operation of the Diesel Generator Air Starting Sub-system under normal conditions.

Justification:

A. Correct: Crossover piping allows starting air to to be supplied to both banks of diesel cylinders. The diesel starts in the test run mode of operation on an inadvertent Containment Spray System actuation.

B. Incorrect:Crossover piping allows starting air to to be supplied to both banks of diesel cylinders. The diesel does not start in the Emergency run mode of operation on an inadvertent Containment Spray System actuation.

C. Incorrect:Crossover piping allows starting air to to be supplied to both banks of diesel cylinders. The diesel starts in the test run mode of operation on an inadvertent Containment Spray System actuation.

D. Incorrect:Crossover piping allows starting air to to be supplied to both banks of diesel cylinders. The diesel does not start in the Emergency run mode of operation on an inadvertent Containment Spray System actuation.

50.

This Exam Level	RO
Appears on:	RO EXAM 2012 Tier 2 Group 1
K/A #	3.6 064 A1.08
Importance Rating:	3.1

While setting up a Diesel Generator to be paralleled with off-site power the following parameters are noted just before the output breaker is closed;

- The synchroscope is moving slowly in the fast direction.
- Grid frequency 59.9 Hz
- Diesel RPM 600
- Bus Voltage 4160v
- Generator Voltage 4150v

Upon closure of the Diesel Generator output breaker the operator must immediately raise ____(1)____ to avoid a ____(2)____ trip.

A. (1) speed (2) over current

B. (1) speed (2) reverse power

C. (1) voltage (2) over current

D. (1) voltage (2) reverse power

Answer: B

Reference Id: Q43968
Difficulty: 3.00
Time to complete: 2
10CFR Category: CFR 55.41 (7) 55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Cognitive Level: Comprehension / Anal
Question Source: New
Comment:
Proposed reference to be provided to applicant during examination: NONE

Technical Reference: LOIT Lesson Plan, 40OP-9DG01 Emergency Diesel Generator

PVNGS OPERATING EXPERIENCE

K&A: Ability to predict and/or monitor changes in parameters(to prevent exceeding design limits) associated with operating the ED/G systems controls including: Maintaining minimum load on ED/G (to prevent reverse power)

Learning Objective: Manually start, load, and unload the 'A' Diesel Generator

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Justification:

A. Incorrect: Going to raise on the speed controller with the generator output breaker closed will raise load and is directed by procedure however, this will also raise output current.
B. Correct: Going to raise on the speed controller with the generator output breaker closed will raise load and is directed by procedure. The basis for this step is to avoid a reverse power trip.
C. Incorrect: Raising voltage setpoint will change reactive loading however, under the conditions stated an overcurrent condition will not be approached.
D. Incorrect: Raising voltage setpoint will change reactive loading however, raising voltage will not mitigate a reverse power condition.

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51.

This Exam Level	RO
Appears on:	RO EXAM 2012
	Tier 2 Group 1
K/A #	073 2.2.39
Importance Rating:	3.9

Given the following conditions:

- Unit 1 is in Mode 6.
- Fuel movement is in progress.
- It is discovered that both channels of CREFAS (RU-29 and RU-30) are INOPERABLE.

Which of the following is the **MINIMUM** actions required to comply with Tech Spec 3.3.9, Control Room Essential Filtration Actuation Signal?

A. Immediately place one train of CREFS in operation **OR** suspend movement of irradiated fuel assemblies, positive reactivity additions and core alterations.

B. Immediately place one train of CREFS in operation **AND** suspend movement of irradiated fuel assemblies, positive reactivity additions and core alterations.

C. Within 1 hour place one train of CREFS in operation **OR** suspend movement of irradiated fuel assemblies, positive reactivity additions and core alterations.

D. Within 1 hour place one train of CREFS in operation **AND** suspend movement of irradiated fuel assemblies, positive reactivity additions and core alterations.

Answer: A

66739 Describe the administrative requirements associated with system Radiation Monitors

Reference Id:	Q43960	
Difficulty:	3.00	
Time to complete:	3	
10CFR Category:	CFR 55.41 (10)	55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.
Cognitive Level:	Memory	
Question Source:	New	

Comment:

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: Tech Specs

PVNGS OPERATING EXPERIENCE

K&A: Knowledge of less than or equal to one hour Technical Specification action statements for systems: PRMS

Learning Objective:

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Justification:

Tech Spec 3.3.9 condition C states that the requirement during movement of irradiated fuel is to **Immediately** either place a train of CREFAS in operation **OR** suspend movement of, positive reactivity additions and Core Alterations

A. Correct – Immediately and OR meet the minimum requirements.
B. Incorrect – Immediately is right but AND is not required to meet the action.
C. Incorrect – 1 hour is wrong Mode 1 action w/o movement of irradiated fuel.
D. Incorrect – 1 hour is wrong Mode 1 action w/o movement of irradiated fuel.

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52.

This Exam Level:	RO
Appears on:	RO EXAM 2012
	Tier 2 Group 1
K/A #:	3.4 076 A2.01
Importance Rating:	3.5

Given the following conditions:

- Unit 1 is operating at 100% power.
- The Plant Cooling Water System develops a large unisolable leak in the common pump discharge header.
- Plant Cooling Water Header Pressure Low Alarm Annunciates in the Control Room.
- Essential Cooling Water train "A" is crosstied and supplying Nuclear Cooling Water priority loads.
- 40AO-9ZZ03 Loss Of Cooling Water has been entered.

Which ONE of the following systems are affected and what actions should the crew take?

A. Turbine Cooling Water System, Trip the Reactor.

B. Essential Cooling Water System, Trip the Reactor.

C. Turbine Cooling Water System, Trip the Main Turbine.

D. Essential Cooling Water System, Trip the Main Turbine.

Answer: A

Reference Id:	Q43961	
Difficulty:	3.00	
Time to complete:	2	
10CFR Category:	CFR 55.41 (10)	55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.
Cognitive Level:	Comprehension / Anal	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE
Technical Reference: 40AO-9ZZ03, Loss of Cooling Water
K&A: Ability to (a) predict the impacts of the following malfunctions or operations on the SWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of SWS.
Learning Objective: Given plant conditions determine if the Loss of Cooling Water AOP should be executed in accordance with 40AO-9ZZ03.
Justification:

A. Correct: Plant Cooling Water System cools the Turbine Cooling Water Heat Exchanger and, 40AO-9ZZ03 Loss of Cooling Water requires a Reactor Trip.

B. Incorrect: The loss of Essential Cooling Water System in this case would require a Reactor Trip however, the loss of Plant Cooling Water will not affect Essential Cooling Water.

C. Incorrect: It is true that the Plant Cooling Water System cools the Turbine Cooling Water Heat Exchanger however, 40AO-9ZZ03, Loss of Cooling Water requires a Reactor Trip.

D. Incorrect: The loss of Essential Cooling Water System in this case would require a Reactor Trip however, the loss of Plant Cooling Water will not affect Essential Cooling Water. 40AO-9ZZ03, Loss of Cooling Water requires a Reactor Trip.

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53.

This Exam Level:	RO
Appears on:	RO EXAM 2012 Tier 2 Group 1
K/A #:	3.4 076 K1.19
Importance Rating:	3.6

Given the following conditions:

- Unit 1 is operating at 100% power.
- Both Nuclear Cooling Water Pumps are unavailable.
- Essential Cooling Water (EW) is cross tied to supply Nuclear Cooling Water (NC).

Which ONE of the following describes the NC priority heat load that will be supplied from EW?

A. Normal Chillers.

B. Letdown heat exchanger.

C. Waste Gas Compressors.

D. Containment Normal AHUs.

Answer: A

Reference Id:	Q43965	
Difficulty:	2.00	
Time to complete:	2	
10CFR Category:	CFR 55.41 (4)	55.41 (4) Secondary coolant and auxiliary systems that affect the facility.
Cognitive Level:	Memory	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: LOIT Lesson Plan

K&A: Knowledge of the physical connections and/or cause- effect relationships between the SWS and the following systems: SWS emergency heat loads.

Learning Objective: L65468 Describe the Nuclear Cooling Water Priority loads that can be supplied by the Essential Cooling Water system.

Justification:

A. Correct: Normal Chillers are a Priority Heat Load.
B. Incorrect: Waste Gas compressors are not a Priority Heat Load.
C. Incorrect: Letdown heat exchanger are not a Priority Heat Load.
D. Incorrect: Containment Normal AHUs are not a Priority Heat Load.

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54.

This Exam Level	RO
Appears on:	RO EXAM 2007
	RO EXAM 2012
	Tier 2 Group 1
K/A #	3.8 078 K3.02
Importance Rating:	3.6

Which of the following is true regarding an Instrument Air pipe rupture in the Main Steam Support Structure (MSSS)?

A. Service Air will supply all loads.

B. Accumulator will provide ADV operation.

C. Low Pressure Nitrogen will supply all loads.

D. Economizer Feedwater Isolation valves fast closure and slow mode of operation are available via the accumulator.

Answer: B

Reference Id: Q44003
Difficulty: 3.00
Time to complete: 3
10CFR Category: CFR 55.41 (7) 55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Cognitive Level: Comprehension / Anal
Question Source: PV Bank Not Modified
Comment:

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40AO-9ZZ06 (Loss of Instrument Air)

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

Learning Objective: Determine the major effects on plant operation as instrument air pressure degrades.

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Justification:

MSSS is mentioned because the ADVs and Economizer valves are located within.

A. Incorrect:The break will prevent backup sources supplying loads, Service Air no longer is a backup.
B. Correct: Accumulator will allow ADV operation for up to 8 hours.
C. Incorrect: Nitrogen backup may open on low pressure but the pipe break makes this useless.
D. Incorrect: Accumulator provides fast closure but not slow mode of operation.

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55.

This Exam Level	RO
Appears on:	RO EXAM 2007 RO EXAM 2012 Tier 2 Group 1
K/A #	3.5 103 A1.01
Importance Rating:	3.7

Given the following conditions:

- Unit 1 has tripped due to a LOCA inside Containment.
- SIAS/CIAS/MSIS/CSAS have initiated.
- Both Containment Spray trains have failed to actuate.
- The CRS has entered the Functional Recovery procedure.
- CTPC-2 is being implemented to supply CS flow using LPSI pump A.

Which ONE of the below listed sets of parameters will be monitored to satisfy CPTC-2?

Containment...

A. humidity and CS flow.

B. pressure and CS flow.

C. humidity and LPSI pump amps

D. pressure and LPSI pump amps.

Answer: D

Reference Id:	Q43989	
Difficulty:	3.00	
Time to complete:	2	
10CFR Category:	CFR 55.41 (10)	55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.
Cognitive Level:	Memory	
Question Source:	PV Bank Not Modified	
Comment:		

Proposed reference to be provided to applicant during examination: NONE
Technical Reference: 40EP-9EO09, CTPC-2
K&A: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the containment system: Containment pressure, temperature, and humidity.
Learning Objective: L65087 Describe the design basis associated with the Containment system.
Justification:

A. Incorrect: When LPSI is cross tied to CS, CS header flow is not available. (40EP-9EO09, CTPC-2, note by step 3).
B. Incorrect: When LPSI is cross tied to CS, CS header flow is not available. (40EP-9EO09, CTPC-2, note by step 3).
C. Incorrect: Humidity will be high initially from the LOCA, so a change would not be seen.
D. Correct: 40EP-9EO09, CTPC-2 step 3.1.f limits amps to ensure continued operation of the LPSI pump. Containment pressure will drop if the section is performed correctly.

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56.

This Exam Level	RO
Appears on:	RO EXAM 2008
	RO EXAM 2012
	Tier 2 Group 2
K/A #	3.1 001 A4.03
Importance Rating:	4.0

Given the following conditions:

- Unit 3 is operating at 55% power following a Large Load Reject event.
- The CRS has implemented 40AO-9ZZ08 (Load Rejection).
- CEDMCS has been placed in standby.
- Reg. Group 3 CEAs are at 135 inches withdrawn.
- Reg. Group 4 CEAs are fully inserted.

In accordance with 40AO-9ZZ08, proper CEA group overlap will be restored by ...

A. withdrawing Reg group 4 CEAs in manual group mode.

B. withdrawing Reg group 4 CEAs in manual sequential mode.

C. withdrawing Reg. group 4 CEAs in manual individual mode while maintaining CEAs within 6.6 inches.

D. lowering the load limit pot until the "Load Limiting" light illuminates then allow the Reg group 4 CEAs to withdraw in auto sequential mode.

Answer: A

Reference Id: Q22484
Difficulty: 3.00
Time to complete: 3
10CFR Category: CFR 55.41 (7) 55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Cognitive Level: Memory
Question Source: PV Bank Not Modified
Comment:
Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40AO-9ZZ08 (Large Load reject), 40OP-9SF01 (CEDMCS operations)

K&A: Ability to manually operate and/or monitor in the control room: CRDS mode control

Learning Objective: Describe the CEDMCS Remote Operator Module located in the Control Room to include all switches and the meaning of each switch position.

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Justification:

A. Correct: RPCB LLR procedure directs withdraw in manual group.
B. Incorrect: Manual Sequential would cause group 3 to withdraw to UGS while moving group 4
C. Incorrect: this would work but not directed by procedure, 6.6 inches is the CWP/CEDMCS Alarm limit.
D. Incorrect: Lowering the pot is procedurally directed but to clear the RPCB signal not to withdraw CEAs.

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57.

This Exam Level	RO
Appears on:	RO EXAM 2012 Tier 2 Group 2
K/A #	3.2 002 K3.03
Importance Rating:	4.2

Given the following conditions:

- Unit 1 has tripped due to a Large Break LOCA.

Which ONE of the following correctly identifies the threshold values for HARSH containment conditions?

CTMT Temperature > ____(1)____ ^{0}F OR CTMT Radiation level > ____(2)____ mR/hr.

A. (1) 170 (2) 10^5

B. (1) 170 (2) 10^8

C. (1) 235 (2) 10^5

D. (1) 235 (2) 10^8

Answer: B

Reference Id:	Q43966	
Difficulty:	2.00	
Time to complete:	2	
10CFR Category:	CFR 55.41 (10)	55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.
Cognitive Level:	Memory	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40EP-9EO03 (LOCA), 40DP-9AP08 (Tech Guide)

K&A: Knowledge of the effect that a loss or malfunction of the RCS will have on the following: Containment

Learning Objective: Given conditions of LOCA analyze Containment Temperature and Pressure Control to determine if the SFSC acceptance criteria is satisfied in accordance with 40EP-9EO03.

Justification:

A. Incorrect: 170 ^{0}F is correct but 10 ^{5} is the Rem value, the procedure specifically state mR/hr.
B. Correct: 170 ^{0}F is correct and 10 ^{8} is correct.
C. Incorrect: 235 ^{0}F is the temperature that the CSAS pressure corresponds to. 10 ^{5} is the Rem value, the procedure specifically state mR/hr.
D. Incorrect: 235 ^{0}F is the temperature that the CSAS pressure corresponds to. 10 ^{8} is correct.

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58.

This Exam Level:	RO
Appears on:	RO EXAM 2008
	RO EXAM 2012
	Tier 2 Group 2
K/A #:	3.7 016 A3.01
Importance Rating:	2.9

Given the following conditions:

- Unit 1 is operating at 100% power.
- SG #1 level transmitter LT-1111 is within the normal band.
- SG #1 level transmitter LT-1112 is within the normal band.

Which ONE of the following describes the level transmitter signal(s)?

SG #1 DFWCS automatically uses the...

A. lower output of LT-1111 and LT1112.

B. higher output of LT-1111 and LT-1112.

C. average output of LT-1111 and LT-1112.

D. output of LT-1111, unless it is out of range then LT-1112 will be selected.

Answer: B

Reference Id:	Q43967	
Difficulty:	3.00	
Time to complete:	2	
10CFR Category:	CFR 55.41 (7)	55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Cognitive Level:	Memory	
Question Source:	PV Bank Not Modified	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: LOIT lesson plan

K&A: Ability to monitor automatic operation of the NNIS, including: Automatic selection of NNIS inputs to control systems

Learning Objective: L82151 Describe the NR steam generator level inputs to DFWCS and their function.

Justification:

A. Incorrect: DFWCS uses the higher output, candidate may think that the system uses the lower.
B. Correct: DFWCS uses the higher output.
C. Incorrect: DFWCS uses the higher output, candidate may think that the system uses the average.
D. Incorrect: This would be true if LT-1111 is placed in maintenance.

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59.

This Exam Level	RO
Appears on:	RO EXAM 2010
	RO EXAM 2012
	Tier 2 Group 2
K/A #	3.7 017 K1.01
Importance Rating:	3.2

Core Exit Thermocouples (CETs) provide a DIRECT input to which ONE of the following?

A. COLSS.

B. QSPDS.

C. ERFDADS.

D. B02 Post Accident Meters.

Answer: B

Reference Id: Q43753
Difficulty: 3.00
Time to complete: 3
10CFR Category: CFR 55.41 (7) 55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Cognitive Level: Memory
Question Source: New
Comment:

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: LOIT Lesson Plan

K&A: Knowledge of the physical connections and/or cause effect relationships between the ITM system and the following systems: Plant computer.

Learning Objective: Explain the operation of the Core Exit Thermocouples (CETs) associated with the Incore Instrumentation System.

Justification:

A. Incorrect: COLSS receives inputs from the Incore detectors which are on the same instrument string as the CETs.
B. Correct: CET detectors are connected to the QSPDS cabinet by a chromel aluminum lead which removes the need for a temperature controlled environment junction box.
C. Incorrect: ERFDADS receives CET data from QSPDS.
D. Incorrect: B02 Post Accident Monitors receive data from QSPDS to display Core Exit Temps and Saturation Margins.

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60.

This Exam Level	RO
Appears on:	RO EXAM 2012
	Tier 2 Group 2
K/A #	3.5 028 A1.02
Importance Rating:	3.4

Given the following conditions:

- Unit 2 has experienced a small LOCA resulting in a containment pressure of 2 psig.
- PZR pressure is steady at 2100 psia.
- The Hydrogen Recombiners are in operation.
- Containment hydrogen concentration is 3.5%.
- The break suddenly propagates resulting in dropping PZR pressure and containment pressure rising to 7 psig.

Which ONE of the following describes the impact on the Hydrogen Recombiners?

The Hydrogen Recombiners...

A. will still be aligned and may remain so.

B. must be isolated to prevent exceeding their design pressure.

C. must be isolated to prevent exceeding their design hydrogen concentration.

D. have isolated and can be realigned from the control room using its override feature.

Answer: D

Reference Id:	Q44009	
Difficulty:	3.00	
Time to complete:	3	
10CFR Category:	CFR 55.41 (7)	55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Cognitive Level:	Comprehension / Anal	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: System Technical Manual, LOCA Procedure Technical Guide

K&A: Ability to predict and/or monitor changes in parameter (to prevent exceeding design limits) associated with operating the HRPS controls including: Containment pressure.

Learning Objective: Describe the automatic functions associated with the Hydrogen Control System Containment Isolation Valves.

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Justification:

A. Incorrect: The containment isolation valves for the hydrogen control system close on a Containment Isolation Signal actuated at 3.0 psig.

B. Incorrect: The Hydrogen Recombiners can withstand maximum design containment pressure. In the LOCA procedure there is a limit imposed to ensure containment pressure is less than < 8.5 psig before aligning the hydrogen recombiners. The Hydrogen Control operating procedure has a maximum containment pressure of 10 psig.

C. There is a hydrogen concentration lower limit of operation for the PURGE Units of at least 2.8%. The hydrogen control procedure does not have an upper limit on hydrogen concentration however, there is a caution to assume an explosive mixture is present when placing the hydrogen control system in operation.

D. The containment isolation valves for the hydrogen control system close on a Containment Isolation Signal actuated at 3.0 psig and will be overriden and opened to re-establish hydrogen control.

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61.

This Exam Level	RO
Appears on:	RO EXAM 2012 Tier 2 Group 2
K/A #	3.8 029 A1.03
Importance Rating:	3.0

Which ONE of the following describes the interlock associated with Power Access Purge Containment Inlet Isolation valves.

Containment ____(1)____ must be ____(2)____ the setpoint before the dampers will OPEN.

A. (1) pressure (2) above

B. (1) pressure (2) below

C. (1) temperature (2) above

D. (1) temperature (2) below

Answer: B

Reference Id:	Q43969	
Difficulty:	2.00	
Time to complete:	2	
10CFR Category:	CFR 55.41 (7)	55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Cognitive Level:	Memory	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: LOIT Lesson Plan

K&A: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Containment Purge System controls including: Containment pressure, temperature, and humidity.

Learning Objective: Describe the automatic functions and interlocks associated with the Power Access Purge Containment Isolation Dampers (CPA-UV-4A & 4B, and CPB-UV-5A & 5B).

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Justification:

A. Incorrect: Candidate may confuse the inlet isolation valve with the vent valve CPN-PV-43 which has an interlock to remain closed so that flow will be directed through the vent orifice when pressure is above .5 psig

B. Correct: The Power Access Purge Containment Inlet Isolation Valves are interlocked such that Containment Pressure must be below 0.03 psig as measured by HC-PT-493, before the dampers will open.

C. Incorrect: Temperature provides interlocks to the CTMT Purge AHUs to determine if the Heaters or Chill Water will be used to adjust the temperature. CTMT Temperature is a Tech Spec monitored parameter. Candidate may confuse the inlet isolation valve with the vent valve CPN-PV-43 which has an interlock to remain closed so that flow will be directed through the vent orifice when pressure is above .5 psig

D. Incorrect: Temperature provides interlocks to the CTMT Purge AHUs to determine if the Heaters or Chill Water will be used to adjust the temperature. CTMT Temperature is a Tech Spec monitored parameter.

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62.

This Exam Level: RO
Appears on: RO EXAM 2012
Tier 2 Group 2
K/A #: 3.8 033 K4.01
Importance Rating: 2.7

Given the following conditions:

- Unit 1 is at 100% power
- Spent Fuel Pool (SFP) level is 137' 10" and has been noted by the AO to be slowly losing level over the past several shifts.
- Chemistry has just reported SFP Boron Concentration at 1900 ppm.
- The crew is investigating the loss of level at this time.
- You are directed by the CRS to add water to the SFP.

Which ONE of the following is the appropriate source of makeup water to the SFP?

A. Recycle Monitor Tank.

B. Refueling Water Tank.

C. Condensate Storage Tank.

D. Reactor Makeup Water Tank.

Answer: B

Reference Id:	Q43970	
Difficulty:	2.00	
Time to complete:	2	
10CFR Category:	CFR 55.41 (10)	55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.
Cognitive Level:	Comprehension / Anal	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE
Technical Reference: 40AO-9ZZ23 (Loss of SFP Level), Tech Spec 3.714 & 3.7.15
K&A: Knowledge of design feature(s) and/or interlock(s) which provide for the following: Maintenance of spent fuel level.
Learning Objective: Explain the operation of the Spent Fuel Pool under normal operating conditions.
Justification:
Tech Spec 3.7.15 states that SFP Boron Concentration must be > 2150 ppm. Therefore a Borated source must be used for make up. Normal losses from the SFP are from evaporation, therefore the normal makeup is a NON Borated Source. Candidate must know the Tech Spec Limit and that the loss is due to a leak which is not evaporation. These conditions require a Borated Makeup.

A. Incorrect: RMT is a source of make up to the SFP, but it is NOT Borated.
B. Correct: RWT is borated to >4400 ppm and is the correct source.
C. Incorrect: CST is the normal source of make up for losses due to evaporation. It is NOT borated.
D. Incorrect: RMWT is an available makeup source to the SFP, but it is NOT Borated.

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63.

This Exam Level	RO
Appears on:	RO EXAM 2012 Tier 2 Group 2
K/A #	3.4 041 K6.03
Importance Rating:	2.7

Given the following conditions:

- Unit 1 is operating at 100% power.
- 120 VAC NNN-D11 loses power.
- Two minutes later, power is restored using 40AO-9ZZ14, Loss Of Non-Class Control Power.
- Prior to any further operator actions the Main Turbine trips.

The Steam Bypass Control System will respond to the Turbine Trip with ____(1)____, ____(2)____.

A. (1) a Quick Open (2) and Modulation

B. (1) a Quick Open (2) but NO Modulation

C. (1) NO Quick Open (2) but will Modulate

D. (1) NO Quick Open (2) and NO Modulation

Answer: D

Reference Id:	Q43973	
Difficulty:	3.00	
Time to complete:	2	
10CFR Category:	CFR 55.41 (7)	55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Cognitive Level:	Comprehension / Anal	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40AO-9ZZ14, Loss Of Class Instrument Or Control Power

K&A: Knowledge of the effect of a loss or malfunction on the following will have on the SDS: Controller and positioners, including ICS, S/G, CRDS

Learning Objective: Steam Bypass Control System controller/logic loss of power

Justification:

A. Incorrect: SBCS in auto and remote is a Quick Open interlock
B. Incorrect: SBCS in auto and remote is a Quick Open interlock
C. Incorrect:When energized, SBCS will come back in manual with zero output.
D. Correct: On a loss of NNN-D11 SBCS loses logic power. The bypass valves fail closed and cannot be operated in manual or auto. When energized, SBCS will come back in manual with zero output.

PVNGS 2012 Reactor Operator NRC Exam

64.

This Exam Level:	RO
Appears on:	RO EXAM 2012 Tier 2 Group 2
K/A #:	3.7 072 K5.01
Importance Rating:	2.7

Given the following conditions:

- Unit 1 operating at 100% power.
- The core is at 250 EFPD.
- A containment purge is in progress.
- The reactor trips with indications of a large break LOCA.
- A CIAS fails to actuate.
- Core damage is indicated.

The Power Access Purge Area Monitors, SQA-RU-37 and SQB-RU-38 will sense rising _______ radiation levels.

A. beta

B. alpha

C. gamma

D. neutron

Answer: C

Reference Id:	Q44013	
Difficulty:	2.00	
Time to complete:	2	
10CFR Category:	CFR 55.41 (11)	55.41 (11) Purpose and operation of radiation monitoring systems, including alarms and survey equipment.
Cognitive Level:	Comprehension / Anal	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: STM
K&A: Knowledge of the operational implications of the following concepts as they apply to the ARM system: Radiation theory, including sources, types, units, and effects

Learning Objective: L66723 Given a Area Radiation Monitor number and name describe the purpose
Justification:

A. Incorrect: Beta radiation will not be able to penetrate the piping
B. Incorrect: Alpha radiation will not be able to penetrate the piping
C. Correct: The radiation levels sensed by this detector would be coming from inside the purge lines, gamma being the most penetrating.
D. Incorrect: There would be no significant neutron radiation levels due to the trip, containment shielding, and detector design.

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65.

This Exam Level	RO
Appears on:	RO EXAM 2012 Tier 2 Group 2
K/A #	3.8 075 K2.03
Importance Rating:	2.6

EWB-P01, Essential Cooling Water Pump 'B' is powered by which ONE of the following sources?

A. PBB-S04

B. PGB-L32

C. PHB-M32

D. PKD-N44

Answer: A

Reference Id:	Q43976	
Difficulty:	2	
Time to complete:	1	
10CFR Category:	CFR 55.41 (4)	55.41 (4) Secondary coolant and auxiliary systems that affect the facility.
Cognitive Level:	Memory	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40AO-9ZZ12, Degraded Electrical

K&A:.Emergency/essential SWS pumps Knowledge of bus power supplies to the following: Emergency/essential SWS pumps.

Learning Objective: Describe how the Essential Cooling Water system is supported.

Justification:

A. Correct: EW is as system required for safe shutdown. The 4160V pump motor is powered by class switchgear.
B. Incorrect:PGB-L32 is a class 480V load center.
C. Incorrect:PHB-M32 is a 480V MCC.
D. Incorrect:PKD-N44 is the inverter for SDC isolation valve SIB-UV-654

PVNGS 2012 Reactor Operator NRC Exam

66.

This Exam Level: RO
Appears on: RO EXAM 2012
Tier 3
K/A #: 2.0 2.1 2.1.26
Importance Rating: 3.4

Which ONE of the following is the lower oxygen concentration limit which establishes confined space entry requirements?

A. 16.0%

B. 19.5%

C. 21.0%

D. 23.5%

Answer: B

Reference Id:	Q43977	
Difficulty:	3.00	
Time to complete:	2	
10CFR Category:	CFR 55.41 (10)	55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.
Cognitive Level:	Memory	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: LOIT Lesson Plan

K&A:.Conduct of Operations: Knowledge of non-nuclear safety procedures (e.g. rotating equipment, electrical, high temperature, high pressure, caustic, chlorine, oxygen and hydrogen).

Learning Objective: L62991 From memory state the required oxygen levels in a confined space

Justification:

A. Incorrect: This is the lethal limit.
B. Correct:An oxygen deficient atmosphere exists when the oxygen concentration is less than 19.5%.
C. Incorrect:This is the normal concentration in air.
D. Incorrect:This is the upper limit.

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67.

This Exam Level	RO
Appears on:	RO EXAM 2012 Tier 3
K/A #	2.1.37
Importance Rating:	4.3

Which ONE of the following describes the control room personnel that **MUST** attend a reactivity brief for a normal shiftly dilution per ODP-1 (Operations Principles and Standards)?

The CRS, RO...

A. and CO.

B. and STA.

C. CO and SM.

D. STA and SM.

Answer: A

Reference Id:	Q43988	
Difficulty:	2.00	
Time to complete:	2	
10CFR Category:	CFR 55.41 (10)	55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.
Cognitive Level:	Memory	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: ODP-1 (Operations Principles and Standards)

K&A: Conduct of Operations: Knowledge of procedures, guidelines or limitations associated with Reactivity Management

Learning Objective: ODP-1 Reactivity Management

Justification:

A. Correct: Per ODP-1 The CRS, RO and CO WIll attend the Reactivity Brief. The SM and STA(s) should attend but are not required per the ODP-1 guidance.
B. Incorrect: SM should attend but is not required.
C. Incorrect: SM and STA should attend but are not required.
D. Incorrect: CO is required to attend but the SM is not.

PVNGS 2012 Reactor Operator NRC Exam

68.

This Exam Level:	RO
Appears on:	RO EXAM 2012 Tier 3
K/A #:	2.1.29
Importance Rating:	4.1

Given the following conditions:

- Unit 1 is operating at 100% power.
- The operating crew is performing a lineup to Drain the Safety Injection Tank (SIT) 1A .
- SIE-V463 (SIT Fill and Drain Line Containment Isolation Valve) is to be opened to support the evolution.
- The CRS has verified this to be a normally locked closed Containment Isolation valve.

Per guidance found in 40DP-9OP19 (Locked Valve, Breaker, and Component Tracking), this valve ...

A. is prohibited from being operated while in Mode 1.

B. may be opened provided the the four hour action for an inoperable containment penetration is entered when the valve is opened.

C. may be opened provided an Operator is identified in the Control Room log with the responsibility to close the valve with in 1 (ONE) hour.

D. may be opened provided a dedicated Operator is stationed at the valve who must be in constant communication with the Control Room.

Answer: D

Reference Id:	Q5219	
Difficulty:	4.00	
Time to complete:	3	
10CFR Category:	CFR 55.41 (10)	55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.
Cognitive Level:	Memory	
Question Source:	PV Bank Not Modified	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40DP-9OP19 (Locked Valve, Breaker, and Component Tracking)

K&A: Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.

Learning Objective: describe the administrative controls required when intermittently opening of locked closed manual containment isolation valves in accordance with 40DP-9OP19.

Justification:

A. Incorrect: Candidate may think that this is a containment penetration that can not be opened in Mode 1. This may be performed in Mode 1.
B. Incorrect: Entering the 4 hour action of 3.6.3 is not required to entered. Also, this will not eliminate the need for a dedicated operator or 60 second operation.
C. Incorrect: The designated operator will be identified in the control room log, but this does not meet the requirements to close the valve with in 60 seconds. Tech specs has many instances of one hour requirements.
D. Correct: This is correct per 40DP-9OP19

69.

This Exam Level:	RO
Appears on:	RO EXAM 2012 Tier 3
K/A #:	2.2.35
Importance Rating:	3.6

Given the following conditions:

- Unit 1 is in Mode 5 coming out of an outage.

Which ONE of the following describes when entry into Mode 4 from Mode 5 occurs?

A. 210 ^{0}F.

B. 250 psia.

C. 350 ^{0}F.

D. 385 psia.

Answer: A

Reference Id:	Q44008	
Difficulty: 3.00		
Time to complete:	3	
10CFR Category:	CFR 55.41 (10)	55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.
Cognitive Level:	Memory	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: Technical Specifications

K&A: Ability to determine Technical Specification Mode of Operation.

Learning Objective: Given plant conditions determine the MODE of operation in accordance with the Technical Specifications.

Justification:

A. Correct: 210 ^{0}F is the transition from Mode 5 to 4.
B. Incorrect: Pressure is a condition change that is controlled per 40OP-9ZZ11 (Mode Change Checklist App F), 250 psia is the Maximum Pressure that SDC can be operated with a Containment Spray pump.
C. Incorrect: 350 ^{0}F is the temperature that marks the transition from Mode 4 to Mode 3.
D. Incorrect: Pressure is a condition change that is controlled per 40OP-9ZZ11 (Mode Change Checklist App F) 385 psia is the Maximum Pressure that SDC can be operated with a LPSI pump.

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70.

This Exam Level	RO
Appears on:	RO EXAM 2012 Tier 3
K/A #	2.2.12
Importance Rating:	3.0

Given the following conditions:

- A Surviellance Test (ST) is being performed on numerous valves.
- The ST is being performed at the 90% due date.
- The CO strokes one particular valve several times prior to beginning the ST due to his experience with this valve sticking.
- Upon completion of the ST several pages have no other entries than an N/A.
- One valve had to be re-tested due to a failure of M&TE.
- The CRS recognizes the pre-conditioning condition during his administrative review of the ST.
- The STA is NOT licensed.

Which one of the following statements is correct concerning this situation?

A. The ST pages whose only entry is an N/A shall be discarded.

B. The acceptance review should be completed by the STA prior to the end of the shift.

C. Replacement pages may be added to complete testing on the valve with the failed M&TE.

D. The pre-conditioned valve must be declared inoperable at the time of CRS recognition.

Answer: C

Reference Id:	Q44022	
Difficulty:	3.00	
Time to complete:	3	
10CFR Category:	CFR 55.41 (10)	55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.
Cognitive Level:	Memory	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 73DP-9ZZ14, Surveillance Testing

K&A: Knowledge of surveillance procedures.

Learning Objective: Given that an ST is being performed and ST fails a step or data is out of tolerance describes what must be done if an ST fails.

Justification:

A. Incorrect. per step 3.8.1 any pages with entries including N/A shall be retained.
B. Incorrect. per step 3.8.1 acceptance reviewer must be qualified to perform the test.
C. Correct. per step 3.7.1 re-performance may occur and replacement pages are allowed.
D. Incorrect. the valve has not an ST nor exceeded any required dates.

PVNGS 2012 Reactor Operator NRC Exam

71.

This Exam Level:	RO
Appears on:	RO EXAM 2012 Tier 3
K/A #:	2.3.4
Importance Rating:	3.2

Which ONE of the following describes the 10CFR20 **NON-Emergency** radiation dose limit for Extremities (Elbows and below, Knees and Below)?

A. 5 Rem per year.

B. 15 Rem per year.

C. 25 Rem per year.

D. 50 Rem per year.

Answer: D

Reference Id:	Q43982	
Difficulty: 3.00		
Time to complete:	2	
10CFR Category:	CFR 55.41 (12)	55.41 (12) Radiological safety principles and procedures.
Cognitive Level:	Memory	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 10CFR20, PVNGS Radworker Training Manual

K&A: Radiation Control: Knowledge of radiation exposure limits under normal or emergency conditions.

Learning Objective: State the Federal radiation dose limits for Total Effective Dose Equivalent (TEDE), for the skin, extremities, and lens of the eye and the plant administrative limits/guidelines for radiation exposure.

Justification:

A. Incorrect: This is the limit for whole body (head and trunk)
B. Incorrect: This is the limit for Lens of the eye.
C. Incorrect: This is the Emergency limit for TEDE
D. Correct: This is the limit for Extremities.

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72.

This Exam Level	RO
Appears on:	RO EXAM 2012 Tier 3
K/A #	2.3.5
Importance Rating:	2.9

Given the following conditions:

- You are preparing to enter the RCA on an approved Radiological Exposure Permit (REP).
- Electronic Personnel Dosimeter (EPD) dose alarm setting is 500 mrem.
- Electronic Personnel Dosimeter (EPD) dose rate alarm setting is 1000 mrem/hr.
- The expected RP work area dose rate is 200 mr/hr.
- The actual work area dose rate is 1000 mr/hr.

Based on the conditions above, which ONE of the following describes when you would be required to exit the Radiological Control Area (RCA)?

A. Immediately due to an EPD dose alarm.

B. In 30 minutes due to an EPD dose alarm.

C. Immediately due to an EPD dose rate alarm.

D. In 30 minutes due to an EPD dose rate alarm.

Answer: C

72284 Radiological questions

Reference Id:	Q43985	
Difficulty:	2.00	
Time to complete:	2	
10CFR Category:	CFR 55.41 (11)	55.41 (11) Purpose and operation of radiation monitoring systems, including alarms and survey equipment.
Cognitive Level:	Comprehension / Anal	
Question Source:	Industry Bank	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: Radworker Training Handout

K&A: Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Learning Objective:

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Justification:

A. Incorrect. A dose alarm would be received in 30 minutes. Dose = 500 mrem/1000 mr/hr.
B. Incorrect. A dose rate alarm would be received immediately since the work area dose rate is 1000 mr/hr; which is equal to the rate alarm setting.
C. Correct. A dose rate alarm would be received immediately since the work area dose rate is 1000 mr/hr; which is equal to the rate alarm setting. You are required by the ALARA program to exit the RCA upon receiving an ED alarm.
D. Incorrect. A dose rate alarm would be received immediately since the work area dose rate is 1000 mr/hr; which is equal to the rate alarm setting.

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73.

This Exam Level	RO
Appears on:	RO EXAM 2012
	Tier 3
K/A #	2.4.16
Importance Rating:	3.5

Given the following conditions:

- Unit 1 has tripped from 100% power due to an RCS leak.
- A Loss of Offsite Power (LOOP) occurred on the trip.
- Standard Post Trip Actions (SPTAs) are in progress.
- Bus Plus criteria is not currently being met.

In accordance with the EOP Users guide (40DP-9AP16) which ONE of the following actions is appropriate?

A. The Functional Recovery Procedure may be entered directly if the Entry Conditions are met.

B. The use of AOPs is **NOT** allowed in conjunction with the EOPs, the Standard Appendices shall be implemented as required.

C. Upon completion of the Reactivity Safety Function the CRS may direct actions in an AOP that would recover the required electrical bus(s).

D. If the CRS determines that the MVAC safety function is not met during the SPTAs he may immediately implement the Functional Recovery Procedure.

Answer: C

Reference Id:	Q44023	
Difficulty:	2.50	
Time to complete:	3	
10CFR Category:	CFR 55.41 (10)	55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.
Cognitive Level:	Memory	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40DP-9AP18 (AOP Users Guide) / 40DP-9AP16 (EOP Users Guide)

K&A: 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.

Learning Objective: Given indications for entry into an Abnormal Operating Procedure define the required actions for the conditions given in accordance with the applicable Abnormal Operating Procedure.

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Justification:

a. Incorrect. This may be done if the event initiates in Mode 3 or 4, not Mode 1.
b. Incorrect. The use of AOPs is allowed with CRS concurrence.
c. Correct. Once the Reactivity SF is completed then an AOP may be implemented. (section 17 of 9AP18).
d. Incorrect. SPTAs must be completed prior to entering the FRP.

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74.

This Exam Level	RO
Appears on:	RO EXAM 2012
	Tier 3
K/A #	2.4.31
Importance Rating:	4.2

GIven the following conditions:

- Unit 3 has tripped.
- SPTAs are in progress.

Which ONE of the following describes the use of Alarm Response Procedures during the EOPs?

Use of Alarm Response Procedures should...

A. resume only after the EOPs are exited.

B. resume only after the SPTAs are exited.

C. be used concurrently with the SPTAs at all times.

D. resume only after the plant stabilizes and when directed by the CRS.

Answer: D

Reference Id:	Q43987	
Difficulty:	3.00	
Time to complete:	3	
10CFR Category:	CFR 55.41 (10)	55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.
Cognitive Level:	Memory	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: None

Technical Reference: 40DP-9AP16 (EOP users guide)

K&A: Knowledge of annunciator alarms, indications, or response procedures

Learning Objective: Given that an ORP is being implemented describe the use of an AL when the reactor trips or when performing an EOP in accordance with 40DP-9AP16.

Justification:

A. Incorrect: There is no requirement to resume Alarm Response based on procedure entry or exit. Candidate may assume that changing procedures may mean stable plant conditions.
B. Incorrect: There is no requirement to resume Alarm Response based on procedure entry or exit. Candidate may assume that changing procedures may mean stable plant conditions.
C. Incorrect: The CRS may direct using ARPs as practical, they are not required to be used at all times.
D. Correct: This is described in 40DP-9AP16 (EOP users guide) step 28.

75.

This Exam Level:	RO
Appears on:	RO EXAM 2008
	RO EXAM 2012
	Tier 3
K/A #:	2.4.5
Importance Rating:	3.7

A plant perturbation is in progress that if not properly addressed could result in a manual or automatic Unit trip.

Which ONE of the following sets of procedures would be used to mitigate this event?

A. Normal Operating Procedures.

B. General Operating Procedures.

C. Abnormal Operating Procedures.

D. Emergency Operating Procedures.

Answer: C

Reference Id:	Q22410	
Difficulty:	2.00	
Time to complete:	1	
10CFR Category:	CFR 55.41 (10)	55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.
Cognitive Level:	Memory	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: None

Technical Reference: AOP/EOP Users Guides

K&A: Emergency Procedures / Plan Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions.

Learning Objective: Given that an ORP is being implemented describe the use of an AO or OP when the reactor trips or when performing an EOP

Justification:

A. Incorrect: Intended normal conditions not transients
B. Incorrect: For general operations, not transients
C. Correct: AOPs restore normal conditions following a transient
D. Incorrect: Place the plant in a safe condition after a Reactor trip event

KEY ID

SCORING & PRINTING OPTIONS: RESCORE / MULTIPLE ANSWER SCORING / CORRECT ANSWER / MARK X / TOTAL ONLY — MARK ONLY ONE

Test Sheet 100

Form No. 243532

Revised from #95679

FEED IN THIS DIRECTION

1	C
2	B
3	D
4	A
5	B
6	B
7	B
8	C
9	B
10	B
11	B
12	C
13	C
14	D
15	B
16	C
17	D
18	D
19	A
20	D
21	A
22	D
23	A
24	C
25	B

ANSWER KEY INFO. — # OF KEYS — ITEM COUNT: 025

PERFORMANCE ASSESSMENT — % OF TOTAL SCORE (00 = 100%) — POINTS EARNED — EQUAL PT VALUE

CERTIFICATION: I have reviewed all questions which were missed, have had an opportunity to ask questions, and understand the correct answer to each question. All work on this examination is my own, I have neither given or received help.

SIGNATURE

DATE

NUMBER CORRECT	25
PERCENT CORRECT	100
ROSTER NUMBER	KEY
SCORE	
RESCORE	

f 3/16/12

AM 3/16/12

COMBINED POINTS EARNED	
COMBINED PERCENT CORRECT	
LETTER GRADE	
SCORE	
RESCORE	

MARKING INSTRUCTIONS

USE NO. 2 PENCIL ONLY

Use a No. 2 Pencil or blue or black ink pen only.

Fill oval completely

Erase cleanly

ID/SSN

NAME SRO Key 2012

COURSE EXAM LOIT

DATE 3/16/2012

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1.

This Exam Level	SRO
Appears on:	SRO EXAM 2012 Tier 1 Group 1
K/A #	011 2.4.41
Importance Rating:	4.6

Given the following conditions:

- Unit 3 has tripped from 100% power.
- Containment hydrogen concentration per HPA-AI-9 indicates 3.8%.
- Containment hydrogen concentration per HPB-AI-10 indicates 4.2%.
- Estimated reactor coolant system leakage is 500 gpm.
- Highest Rep CET reading is 587°F.
- RCS chemistry sample dose equivalent Iodine 131 indicates 308 uCi/gm.
- Containment pressure - 37 psig and slowly lowering.
- Pressurizer pressure - 610 psia.
- RVLMS - upper head level - 16%.
- All equipment has properly actuated.

Which ONE of the following describes the appropriate classification and code for this event?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Answer: C

Reference Id:	Q43902
Difficulty:	3.00
Time to complete:	3
10CFR Category:	CFR 55.43 (5) 55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Cognitive Level:	Comprehension / Anal
Question Source:	New
Comment:	

Proposed reference to be provided to applicant during examination: NEI 99-01 HOT/COLD EAL CHART

Technical Reference: NEI99-01 HOT EAL CHART

K&A: Large Break LOCA; Knowledge of the emergency action level thresholds and classifications.

Learning Objective: use the EAL tables and basis document to determine the emergency plan classification

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Justification:

A. Incorrect: NUE is met but it is not the highest EAL classification of the event. Candidate may confuse any of the indications and not properly apply them to the EAL chart.
B. Incorrect: Alert is met but it is not the highest EAL classification of the event. Candidate may confuse any of the indications and not properly apply them to the EAL chart.
C. Correct: SAE is met and is the highest EAL classification of the event.
D. Incorrect: GE is not met. Candidate may confuse any of the indications and not properly apply them to the EAL chart.

HOT INITIATING CONDITIONS – MODES 1 – 2 – 3 – 4

GENERAL EMERGENCY

RADIOLOGICAL – EFFLUENTS

RG1 - Off-site dose resulting from an actual or IMMINENT release of gaseous radioactivity greater than 1000 mrem TEDE or 5000 mrem Thyroid CDE for the actual or projected duration of the release using actual meteorology.

Note: The EC 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, declaration should be based on dose assessment instead of radiation monitor values. Do not delay declaration awaiting dose assessment results.

1. VALID reading on **ANY** of the following radiation monitors greater than the value for 15 minutes or longer:
   - Plant Vent RU-144 CH-1 >1.04E+00 uCi/cc
   - Fuel Building RU-146 CH-2 >3.50E+01 uCi/cc

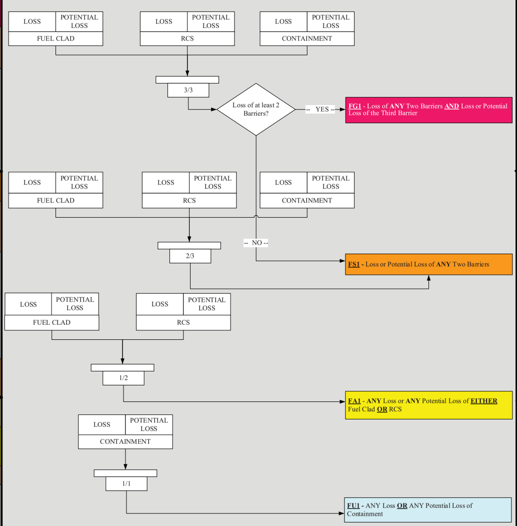
OR

2. Dose assessment using actual meteorology indicates doses greater than 1000 mrem TEDE OR 5000 mrem thyroid CDE at or beyond the site boundary.

OR

3. Field survey results indicate closed window dose rates greater than 1000 mR/hr expected to continue for 60 minutes or longer; or analyses of field survey samples indicate thyroid CDE greater than 5000 mrem for one hour of inhalation, at or beyond site boundary.

FISSION PRODUCT BARRIERS



SYSTEM MALFUNCTIONS – RX and CORE

MG2 - 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. **Modes 1 & 2**

1.a. Plant Protection System failed to shutdown the reactor.
AND
b. **All** Manual actions do NOT shutdown the reactor as indicated by:
- Reactor power is NOT dropping to less than 5% power
- **All** full strength CEAs are NOT inserted

AND
c. Rep CET greater than 1200 °F.

SYSTEM MALFUNCTIONS – AC/DC POWER

MG1 - Prolonged loss of **all** Off-site and **all** On-Site AC power to emergency busses.

1.a. Loss of **all** off-site and **all** on-site AC power to PBA-S03 and PBB-S04.
AND
b. EITHER of the following:
- Restoration of at least one emergency bus in less than 4 hours is not likely.
- RCS and Core Heat Removal Safety Function Acceptance Criteria NOT Satisfied per 40EP-9EO08, BLACKOUT.

HAZARDS – SECURITY

HG1 - HOSTILE ACTION resulting in loss of physical control of the facility.

1. A HOSTILE ACTION has occurred such that plant personnel are unable, either remotely or locally, 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.

HAZARDS – EC DISCRETION

HG2 - Other conditions exist which in the judgment of the EC warrant declaration of a General Emergency.

1. Other conditions exist which in the judgment of the EC 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 off-site for more than the immediate site area.

SITE AREA EMERGENCY

RADIOLOGICAL – EFFLUENTS

RS1 - Off-site dose resulting from an actual or IMMINENT release of gaseous radioactivity greater than 100 mrem TEDE or 500 mrem Thyroid CDE for the actual or projected duration of the release.

Note: The EC 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, declaration should be based on dose assessment instead of radiation monitor values. Do not delay declaration awaiting dose assessment results.

1. VALID reading on **ANY** of the following radiation monitors greater than the value for 15 minutes or longer:
   - Plant Vent RU-144 CH-1 >1.04E-01 uCi/cc
   - Fuel Building RU-146 CH-1 >3.50E+00 uCi/cc

OR

2. Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE OR 500 mrem thyroid CDE at or beyond the site boundary.

OR

3. Field survey results indicate closed window dose rates greater than 100 mR/hr expected to continue for 60 minutes or longer; or analyses of field survey samples indicate thyroid CDE greater than 500 mrem for one hour of inhalation, at or beyond the site boundary

SYSTEM MALFUNCTIONS – RX and CORE

MS2 - Automatic Trip fails to shutdown the reactor and the manual actions taken at the reactor control console are not successful in shutting down the reactor. **Modes 1 & 2**

1.a. Plant Protection System failed to shutdown the reactor.
AND
b. Manual actions taken on Panels B05 and B01 do NOT shut down the reactor as indicated by:
- Reactor power is NOT dropping to less than 5% power
- **All** full strength CEAs are NOT inserted

SYSTEM MALFUNCTIONS – AC/DC POWER

MS1 - Loss of **all** Off-site and **all** On-Site AC power to emergency busses for 15 minutes or longer.

Note: The EC 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** Off-Site and **all** On-Site AC power to PBA-S03 and PBB-S04 for 15 minutes or longer.

MS3 - Loss of **all** Vital DC Power for 15 minutes or longer.

Note: The EC 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. Less than 112 VDC on **all** PKA-M41, PKB-M42, PKC-M43, and PKD-M44 for 15 minutes or longer.

SYSTEM MALFUNCTIONS – ALARMS / COMMUNICATIONS

MS6 - Inability to monitor a significant transient in Progress.

Note: The EC should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time

1. a. Loss of annunciators on **ANY** 4 of the following B01, B02, B04, B05, B06
or
SESS for 15 minutes or longer.
OR
Loss of either PNA-D25 or PNB-D26 for 15 minutes or longer.
AND
b. **ANY** of the following:
- Automatic turbine setback/runback greater than 25% thermal reactor power
- Reactor Trip
- VALID ESFAS Actuation

AND
c. Plant computer indications are unavailable.

HAZARDS – SECURITY

HS4 - HOSTILE ACTION within the PROTECTED AREA.

1. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Team.

HAZARDS – CR EVACUATION

HS2 - Control room evacuation has been initiated and plant control cannot be established.

1.a. Control Room evacuation has been initiated.
AND
b. Control of the plant cannot be established at the Remote Shutdown Panel within 15 minutes.

HAZARDS – EC DISCRETION

HS3 - Other conditions exist which in the judgment of the EC warrant declaration of a Site Area Emergency.

1. Other conditions exist which in the judgment of the EC 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.

ALERT

RADIOLOGICAL – EFFLUENTS

RA1 - **ANY** release of gaseous radioactivity to the environment greater than 20 times the ODCM for 15 minutes or longer.

Note: The EC 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 **ANY** of the following radiation monitors greater than the value for 15 minutes or longer:
   - Plant Vent RU-143 CH-1 > 1.22E-02 uCi/cc
   - Fuel Bldg RU-146 CH-1 >1.13E-01 uCi/cc

OR

2. Confirmed sample analyses for gaseous releases indicates concentrations or release rates greater than 20 times the ODCM Section 3.0 limits for 15 minutes or longer.

RADIOLOGICAL – RAD LEVELS

RA2 - 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.

Note: This EAL does not apply to the cask loading pit during cask loading operations.

1. A water level drop in the reactor refueling cavity, spent fuel pool, cask loading pit, or fuel transfer canal that will result in uncovering irradiated fuel.

OR

2. A VALID High Alarm on **ANY** of the following due to damage to irradiated fuel or loss of water level:
   - RU-16 Containment Operating Level Area
   - RU-17 Incore Instrument Area
   - RU-19 New Fuel Area
   - RU-31 Spent Fuel Pool Area
   - RU-33 Refueling Machine Area
   - RU-143 Plant Vent
   - RU-145 Fuel Building Vent

RA3 - Rise in radiation levels within the facility that impedes operation of systems required to maintain plant safety functions.

1. Dose rate greater than 15 mR/hr in the Control Room Area OR Secondary Alarm Station.

FISSION PRODUCT BARRIERS

Note: Multiple events could occur which result in the conclusion that exceeding the loss or potential loss thresholds is IMMINENT.
In this IMMINENT loss situation use judgment and classify as if the thresholds are exceeded.

The Containment Barrier should not be declared lost or potentially lost based on exceeding Technical Specification action statement criteria, unless there is an event in progress requiring mitigation by the Containment barrier. When no event is in progress (Loss or Potential Loss of either Fuel Clad and/or RCS) the Containment Barrier status is addressed by Technical Specifications.

Fuel Clad Barrier – Loss	Fuel Clad Barrier – Potential Loss	RCS Barrier – Loss	RCS Barrier – Potential Loss	Containment Barrier – Loss	Containment Barrier – Potential Loss
1. A. Coolant activity greater than 300 µCi/gm Dose Equivalent I-131.		1. A. RCS leak rate greater than available makeup capacity as indicated by a loss of RCS subcooling to saturation (0 °F).	1. A. RCS leak rate greater than charging capacity with Letdown isolated. **OR** B. RCS Pressure Control Safety Function Status Not Satisfied. **OR** C. RCS and Core Heat Removal Safety Function Status Not Satisfied.	1. A. A containment pressure rise followed by a rapid unexplained drop in containment pressure. **OR** B. Containment pressure or sump level response not consistent with LOCA or MSLB conditions.	1. A. Containment pressure greater than 60 psig and rising. **OR** B. 4.5% H_2 inside containment. **OR** C. a. Pressure greater than 8.5 psig. **AND** b. Less than one full train of Containment Spray operating.
2. A. Rep CET reading currently or previously greater than 1200 °F	2. A. Rep CET reading currently or previously greater than 700 °F				2. A. a. Rep CET greater than 1200°F. **AND** b. Restoration not effective within 15 minutes. **OR** B. a. Rep CET greater than 700 °F. **AND** b. RVLMS less than 21% plenum. **AND** c. Restoration not effective within 15 minutes.
	3. A. RVLMS level currently or previously less than 21% plenum.	3. A. RUPTURED SG results in an SIAS.		3. A. RUPTURED SG is also FAULTED outside of containment. **OR** B.a. Primary-to-Secondary leakrate greater than 10 gpm. **AND** b. UNISOLABLE steam release from the affected SG to the environment.	
				4. A. a. Failure of **all** valves in any one line to close **AND** b. Direct downstream pathway to the environment exists after containment isolation signal.	
5. A. Containment radiation monitor RU-148 > 2.1E+05 mR/hr **OR** RU-149 > 2.4E+05 mR/hr		5. A. Containment radiation monitor RU-148 > 5.0E+04 mR/hr **OR** RU-149 > 5.6E+04 mR/hr.			5. A. Containment radiation monitor RU-148 > 6.8E+06 mR/hr **OR** RU-149 > 7.8E+06 mR/hr
6. A. Any condition in the opinion of the EC that indicates Loss or Potential Loss of the Fuel Clad Barrier.		6. A. Any condition in the opinion of the EC that indicates Loss or Potential Loss of the RCS Barrier.		6. A. Any condition in the opinion of the EC that indicates Loss or Potential Loss of the Containment Barrier.	

SYSTEM MALFUNCTIONS – RX and CORE

MA2 - Automatic Trip fails to shutdown the reactor and the manual actions taken from the reactor control console are successful in shutting down the reactor. **Modes 1 and 2**

1.a. Plant Protection System failed to shutdown the reactor.
AND
b. Manual shutdown actions taken on Panels B05 or B01 are successful as indicated by **all** of the following:
- Reactor Power is dropping to less than 5% power
- Negative Startup rate
- **All** full strength CEAs are inserted or Boration in progress

SYSTEM MALFUNCTIONS – AC/DC POWER

MA5 - AC power capability to emergency busses reduced to a single power source for 15 minutes or longer such that **ANY** additional single failure would result in station blackout.

Note: The EC 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 PBA-S03 and PBB-S04 reduced to a single power source for 15 minutes or longer.
AND
b. Any additional single power source failure will result in station blackout.

SYSTEM MALFUNCTIONS – ALARMS / COMMUNICATIONS

MA4 - UNPLANNED Loss of safety system annunciation or indication in the Control Room with EITHER (1) a significant transient in progress, or (2) compensatory indicators are unavailable.

Note: The EC 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 annunciators on **ANY** 4 of the following B01, B02, B04, B05, B06 or SESS for 15 minutes or longer
OR
UNPLANNED Loss of either PNA-D25 or PNB-D26 for 15 minutes or longer.
AND
b. **ANY** of the following:
- Automatic turbine setback/runback greater than 25% thermal reactor power
- Reactor Trip
- VALID ESFAS Actuation
- Plant computer unavailable

HAZARDS – NATURAL / DESTRUCTIVE

HA1 - Natural or destructive phenomena affecting VITAL AREAS.

1.a. Seismic event greater than Operating Basis Earthquake (OBE) as indicated by **ANY** Force Balance Accelerometer reading greater than 0.10g.
AND
b. Earthquake confirmed by **ANY** of the following:
- Earthquake felt in plant
- National Earthquake Center
- Control Room indication of degraded performance of systems required for the safe shutdown of the plant.

OR

2. Tornado touching down or high winds reaching 100 mph resulting in VISIBLE DAMAGE to **ANY** POWER BLOCK structure OR Control Room indication of degraded performance of safety systems.

OR

3. Internal flooding in **ANY** POWER BLOCK structure resulting in an electrical shock hazard that precludes access to operate or monitor safety equipment OR Control Room indication of degraded performance of those safety systems.

OR

4. Vehicle crash resulting in VISIBLE DAMAGE to **ANY** POWER BLOCK structure OR Control Room indication of degraded performance of safety systems

HAZARDS – FIRE / EXPLOSION

HA2 - FIRE or EXPLOSION affecting the operability of plant safety systems required to establish or maintain safe shutdown.

1. FIRE or EXPLOSION resulting in VISIBLE DAMAGE to **ANY** POWER BLOCK structure or Control Room indication of degraded performance of safety systems.

HAZARDS – TOXIC / FLAMMABLE

HA3 - Access to a VITAL AREA is prohibited due to release of toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shutdown the reactor.

Note: If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.

1. Access to a VITAL AREA is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shutdown the reactor.

HAZARDS – SECURITY

HA4 - HOSTILE ACTION within the Owner Controlled Area or airborne attack threat

1. A HOSTILE ACTION is occurring or has occurred within the Owner Controlled Area as reported by the Security Team.

OR

2. A validated notification from NRC of an airliner attack threat within 30 minutes of the site.

OR

3. A HOSTILE ACTION directed toward the ISFSI.

HAZARDS – CR EVACUATION

HA5 - Control Room evacuation has been initiated.

1. Control Room evacuation is required by:
40AO-9ZZ18, Shutdown Outside Control Room
OR
40AO-9ZZ19, Control Room Fire.

HAZARDS – EC DISCRETION

HA6 - Other conditions exist which in the judgment of the EC warrant declaration of an Alert.

1. Other conditions exist which in the judgment of the EC 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.

UNUSUAL EVENT

RADIOLOGICAL – EFFLUENTS

RU1 – **ANY** release of gaseous radioactivity to the environment greater than 2 times the ODCM for 60 minutes or longer.

Note: The EC 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 **ANY** of the following radiation monitors greater than the value for 60 minutes or longer:
   - Plant Vent RU-143 CH-1 >1.22E-03 uCi/cc
   - Fuel Bldg RU-145 CH-1 >1.13E-02 uCi/cc

OR

2. Confirmed sample analyses for gaseous releases indicates concentrations or release rates greater than 2 times the ODCM Section 3 limits for 60 minutes or longer.

RADIOLOGICAL – RAD LEVELS

RU2 – UNPLANNED rise in plant radiation levels.

1. a. A VALID Alert Alarm on **ANY** of the following:
   - RU-16 Containment Operating Level Area
   - RU-17 Incore Instrument Area
   - RU-19 New Fuel Area
   - RU-31 Spent Fuel Pool Area
   - RU-33 Refueling Machine Area

AND
b. UNPLANNED water level drop in the reactor refueling cavity, fuel transfer canal, cask loading pit, or spent fuel pool as indicated by **ANY** of the following:
- Visual observation
- SFP LEVEL HI – LOW (EO204A) on PCN-E02
- RWLIS
- Pressurizer level

OR

2. UNPLANNED VALID Area Radiation Monitor readings or survey results indicate a rise by a factor of 1000 over normal* levels.

*Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

SYSTEM MALFUNCTIONS – RX and CORE

MU2 - Inability to reach required shutdown within Technical Specification limits.

1. Plant is not brought to required operating mode within Technical Specifications LCO Action Statement Time.

MU5 - RCS Leakage.

1. Unidentified or pressure boundary LEAKAGE greater than 10 gpm.
OR
2. Identified LEAKAGE greater than 25 gpm.

MU4 - Fuel Clad Degradation.

1. RU-155D High Alarm
OR
2.a. DOSE EQUIVALENT I-131 greater than 1.0 µCi/gm for 48 hours.
OR
b. Coolant Gross Specific Activity greater than 100/Ē µCi/gm.

MU8 - Inadvertent Criticality. **Mode 3 or 4**

1. UNPLANNED sustained source range count rise observed on nuclear instrumentation.

SYSTEM MALFUNCTIONS – AC/DC POWER

MU1 - Loss of **all** Off-site AC power to emergency busses for 15 minutes or longer.

Note: The EC 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** off-site AC power to PBA-S03 and PBB-S04 for 15 minutes or longer.

SYSTEM MALFUNCTIONS – ALARMS / COMMUNICATIONS

MU3 - UNPLANNED loss of safety system annunciation or indication in the Control Room for 15 minutes or longer.

Note: The EC 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 annunciators on **ANY** 4 of the following B01, B02, B04, B05, B06 or SESS for 15 minutes or longer.
OR
UNPLANNED Loss of either PNA-D25 or PNB-D26 for 15 minutes or longer.

MU6 - Loss of **all** On-site or Off-site communications capabilities

1. Loss of **all** of the following on-site communication methods affecting the ability to perform routine operations.
   - PBX
   - Plant Page System
   - Two-Way Radio

OR

2. Loss of **all** of the following off-site communication methods affecting the ability to perform off-site notifications.
   - PBX
   - FTS
   - Cellular Phones

HAZARDS – NATURAL / DESTRUCTIVE

HU1 - Natural or destructive phenomena affecting the PROTECTED AREA.

1. Seismic event identified by **ANY** 2 of the following:
   - VALID Seismic Event alarm
   - Earthquake felt in plant
   - National Earthquake Center

OR

2. Tornado touching down within the PROTECTED AREA or high winds reaching 100 mph.

OR

3. Internal flooding in the POWER BLOCK that has the potential to affect safety related equipment required by Technical Specifications for the current operating mode.

OR

4. Main Turbine failure resulting in casing penetration or damage to turbine or Main Generator seals.

ISFSI

E-HU1 - Damage to a loaded cask CONFINEMENT BOUNDARY

1. Damage to a loaded cask CONFINEMENT BOUNDARY.

HAZARDS – FIRE / EXPLOSION

HU2 - FIRE within the PROTECTED AREA not extinguished within 15 minutes of detection or EXPLOSION within the PROTECTED AREA.

Note: The EC should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the duration has exceeded, or will likely exceed the applicable time.

1. FIRE in the POWER BLOCK or Turbine Building not extinguished within 15 minutes of a FIRE alarm or Control Room notification.

OR

2. EXPLOSION within the PROTECTED AREA.

HAZARDS – TOXIC / FLAMMABLE

HU3 - Release of toxic, corrosive, asphyxiant, or flammable gases deemed detrimental to NORMAL PLANT OPERATIONS.

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 off-site event.

HAZARDS – SECURITY

HU4 - Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant.

1. A SECURITY CONDITION that does NOT involve a HOSTILE ACTION as reported by the Security Team.

OR

2. A credible PVNGS security threat notification.

OR

3. A validated notification from NRC providing information of an aircraft threat.

HAZARDS – EC DISCRETION

HU5 - Other conditions exist which in the judgment of the EC warrant declaration of a UE.

1. Other conditions exist which in the judgment of the EC 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 off-site response or monitoring are expected unless further degradation of safety systems occurs.

Definitions

CONTAINMENT CLOSURE: The procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release in Mode 6.

CONFINEMENT BOUNDARY: The dry storage cask barriers between areas containing radioactive substances and the environment.

EXPLOSION: A rapid, violent, unconfined combustion, or catastrophic failure of pressurized/energized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components.

FAULTED: in a steam generator, the existence of secondary side leakage that results in an uncontrolled drop in steam generator pressure or the steam generator being completely depressurized

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

HOSTILE ACTION: An act toward a NPP or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the NPP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

HOSTILE FORCE: One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

IMMINENT: Mitigation actions have been ineffective, additional actions are not expected to be successful, and trended information indicates that the event or condition will occur.

LEAKAGE shall be:

a. Identified LEAKAGE
   1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;
   2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or
   3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System (primary to secondary LEAKAGE).

b. Unidentified LEAKAGE
   All LEAKAGE that is not identified LEAKAGE;

c. Pressure Boundary LEAKAGE
   LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

NORMAL PLANT OPERATIONS: Activities at the plant site, excluding the Water Reclamation Facility, associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from NORMAL PLANT OPERATIONS.

POWER BLOCK: Structures, systems or components listed below that contain equipment necessary for safe operation and/or shutdown of the reactor.

A. Containment
B. Auxiliary Building
C. Refueling Water Tank (RWT)
D. Diesel Generator Building
E. Diesel Generator Fuel Oil Storage Tanks
F. Fuel Building
G. Spray Pond
H. Condensate Storage Tank (CST)
I. Control Building
J. Corridor Building
K. MSSS

PROTECTED AREA: The area which encompasses all controlled areas within the security PROTECTED AREA fence.

RUPTURED: in a steam generator, existence of primary-to-secondary leakage of a magnitude sufficient to require or cause a reactor trip and safety injection

SECURITY CONDITION: Any Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.

UNISOLABLE: A breach or leak that cannot be isolated from the Control Room.

UNPLANNED: A parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

VALID: An indication, report, or condition, is considered to be VALID when it is verified by (1) an instrument channel check, (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed.

VISIBLE DAMAGE: Damage to equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of the affected structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, and paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

VITAL AREAS: Areas, within the PROTECTED AREA, that contains equipment vital to the operations of the plant.

PVNGS 2012 Senior Reactor Operator NRC Exam

2.

This Exam Level:	SRO
Appears on:	SRO EXAM 2012 Tier 1 Group 1
K/A #:	025 2.4.6
Importance Rating:	4.7

Given the following conditions:

Initial Conditions:

- Unit 1 is in Mode 6.
- Core off-load is in progress.
- SDC is in service using LPSI pump "B".
- "A" EW heat exchanger is tagged out for tube leak repair.

Subsequently:

- A large piece of tarp has lodged in the "B" train SDC suction piping.
- LPSI pump "B" has been secured.

The CRS should restore SDC flow by use of which ONE of the following?

A. CS pump "A" with "A" train auxiliaries per Lower Mode Functional Recovery (40EP-9EO11).

B. LPSI pump "A" with "B" train auxiliaries per Lower Mode Functional Recovery (40EP-9EO11).

C. CS pump "A" with "A" train auxiliaries per Recovery from Shutdown Cooling to Normal Operating Lineup (40OP-9SI02).

D. LPSI pump "A" with "B" train auxiliaries per Recovery from Shutdown Cooling to Normal Operating Lineup (40OP-9SI02).

Answer: B

Reference Id:	Q43900	
Difficulty:	3.00	
Time to complete:	3	
10CFR Category:	CFR 55.43 (5)	55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Cognitive Level:	Comprehension / Anal	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40EP-9EO11 (LMFRP)

K&A: Knowledge of EOP mitigation strategies: Loss of RHR

Learning Objective: L56595 Given the LMFRP HR-2 is being performed, and SDC is in service describe how adequate SDC flow is determined and what actions may be taken if adequate flow cannot be maintained in accordance with 40EP-9EO11.

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Justification:

A. Incorrect: Appendix 241 allows the use of either LPSI or CS pump. Candidate must understand that the 'A' Auxiliaries are unavailable due to the A EW HX being out of service.
B. Correct: For the current lineup, Appendix 241 directs per step 2 to use LPSI A as the SDC pump.
C. Incorrect: Appendix 241 allows for the use of the CS pump and 40OP-9SI02 addresses the use of CS pumps for emergency operations from a SDC Train B lineup.
D. Incorrect: This is the correct action, but 40OP-9SI01 does not address the cross tie for LPSI pumps only CS pumps.

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3.

This Exam Level:	SRO
Appears on:	SRO EXAM 2012 Tier 1 Group 1
K/A #:	4.1 038 EA2.15
Importance Rating:	4.4

Given the following conditions:

- Unit 2 has tripped from 100% power.
- SG 1 AFW Flow is 0 gpm.
- SG 1 pressure is 1165 psia and stable.
- SG 1 level is 10% NR and rising.
- SG 2 AFW Flow is 150 gpm.
- SG 2 pressure is 1170 psia and stable.
- SG 2 level is 60% WR and rising.
- Pressurizer level is 35% and stable.
- RCS pressure is 1300 psia and stable.
- RCPs 1A & 2A are operating.
- Thot is 500°F and stable.
- Tcold is 497°F and stable.
- HPSI has been throttled.
- SPTAs are complete.

Which ONE of the following describes the appropriate procedure and action needed to mitigate this event?

The CRS will enter ____(1)____ **AND** reduce RCS pressure to less than ____(2)____ psia.

A. (1) 40EP-9EO03 (LOCA) (2) 960

B. (1) 40EP-9EO04 (SGTR) (2) 960

C. (1) 40EP-9EO03 (LOCA) (2) 1135

D. (1) 40EP-9EO04 (SGTR) (2) 1135

Answer: D

Reference Id:	Q43905	
Difficulty:	3.00	
Time to complete:	3	
10CFR Category:	CFR 55.43 (5)	55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Cognitive Level:	Comprehension / Anal	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40EP-9EO04 (SGTR)

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K&A: Ability to determine or interpret the following as they apply to a SGTR: Pressure at which to maintain RCS during S/G cooldown.

Learning Objective: L11226 Given the SGTR EOP is being used and given plant conditions determine an appropriate pressure target for depressurization and state the basis for this value.

Justification:

A. Incorrect: LOCA is the incorrect procedure due to the indications of SGTR. Candidate may select LOCA based on the Low PZR Pressure and Level. 960 psia is the MSIS setpoint pressure but the correct pressure is < 1135 psia and 1165 +/- 50 psia.

B. Incorrect: SGTR is the correct procedure but 960 psia is the MSIS setpoint pressure but the correct pressure is < 1135 psia and 1165 +/- 50 psia.

C. Incorrect: LOCA is the incorrect procedure due to the indications of SGTR. Candidate may select LOCA based on the Low PZR Pressure and Level.correct pressure is < 1135 psia and 1165 +/- 50 psia.

D. Correct: Per Step 12 of 40EP-9EO04 (SGTR), Maintain pressurizer pressure within **ALL** of the following criteria: • Less than 1135 psia • Approximately equal to the pressure of the Steam Generator with the tube rupture (± 50 psi) correct pressure is < 1135 psia and 1165 +/- 50 psia.

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4.

This Exam Level	SRO
Appears on:	SRO EXAM 2008 SRO EXAM 2012 Tier 1 Group 1
K/A #	4.2 056 AA2.09
Importance Rating:	2.9

Given the following conditions:

- Unit 2 is operating at 100% power.
- NAN-S02 Fast Bus Transfer is blocked due to SWYD maintenance.
- The "A" and "C" Containment Normal ACUs are running.
- The "B" and "D" Containment Normal ACUs are in standby.
- The "A" and "B" Normal Chillers are running.
- NAN-S01 bus faults and de-energizes.
- All equipment actuates as expected.

Which ONE of the following describes the appropriate procedure the CRS should implement?

A. 40EP-9EO07 (LOOP) due to a loss of 4 RCPs. Normal containment cooling can be restored by energizing NAN-S02 from Offsite power.

B. 40EP-9EO07 (LOOP) due to a loss of 4 RCPs. Normal containment cooling will be restored by the auto start of the "B" and "D" ACU units.

C. 40EP-9EO02 (Reactor Trip) due to a loss of 2 RCPs. Normal containment cooling can be restored by energizing NAN-S02 from Offsite power.

D. 40EP-9EO02 (Reactor Trip) due to the loss of 2 RCPs. Normal containment cooling will be restored by the auto start of the "B" and "D" ACU units.

Answer: A

Reference Id:	Q43903	
Difficulty:	3.00	
Time to complete:	3	
10CFR Category:	CFR 55.43 (5)	55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Cognitive Level:	Comprehension / Anal	
Question Source:	PV Bank Not Modified	
Comment:		

Proposed reference to be provided to applicant during examination: None

Technical Reference: 40EP-0EO07 (LOOP)

K&A: Ability to determine and interpret the following as they apply to the Loss of Offsite Power: Operational status of reactor building cooling unit.

Learning Objective: 74452 Describe the automatic functions associated with the Containment Building Normal ACU Fans (HCN-A01-A, B, C, & D)

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Justification:

A. Correct: LOOP/LOFC due to the loss of 4 RCPs and both NAN-S01/S02 de-energized even though the switchyard is still energized. The operators have the ability to manually energize S02 following the event to restore normal containment cooling.

B. Incorrect:LOOP is correct but the B/D units have no power for the auto start and the "A" (PBA-S03) normal chiller will have to be manually started in addition NC pumps have no power till S02 is energized.

C. Incorrect: all 4 RCPs trip due the loss of NAN-S01 tripping 2 RCPs causing a Rx trip/Turbine trip and a subsequent loss of NAN-S02 due fast bus transfer blocked on the 2 side. Core Heat Removal Safety Function will not be met due to Natural Circulation Delta T being > 10 ^{0}F.

D. Incorrect: 4 RCPs trip and the B/D units have no power for the auto start and the "A" (PBA-S03) normal chiller will have to be manually started in addition NC pumps have no power till S02 is energized. Core Heat Removal Safety Function will not be met due to Natural Circulation Delta T being > 10 ^{0}F.

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5.

This Exam Level:	SRO
Appears on:	SRO EXAM 2012 Tier 1 Group 1
K/A #:	4.2 057 AA2.19
Importance Rating:	4.3

Given the following conditions:

- Unit 1 is operating at 100% power.
- PPS TRBL/GRND alarm on B05.
- All initiation relay lights are extinguished on Channel A and Channel C.
- PKA, PKB, PKC, and PKD are energized.
- All initiation relays on Channels B and D are energized.

Which ONE of the following describes the impact on the plant and the procedure entry required?

A. No RTSG breakers have tripped, enter 40AO-9ZZ13(Loss of Class Control Power).

B. Two RTSG breakers are tripped, enter 40AO-9ZZ13(Loss of Class Control Power).

C. No RTSG breakers have tripped, enter 40AO-9ZZ17(Inadvertent ESFAS Initiation).

D. Two RTSG breakers are tripped, enter 40AO-9ZZ17(Inadvertent ESFAS Initiation).

Answer: B

Reference Id:	Q43887	
Difficulty:	2.00	
Time to complete:	2	
10CFR Category:	CFR 55.43 (5)	55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Cognitive Level:	Comprehension / Anal	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40AO-9ZZ13 (Loss of Class Instrument and Control Power)

K&A: L11089 Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: The plant automatic actions that will occur on the loss of a vital ac electrical instrument bus

Learning Objective: L11089 Given a loss of PK and/or PN describe how the RPS responds to the power loss in accordance with 40AO-9ZZ13.

Justification:

A. Incorrect: PNA and PNC have tripped which will result in RTSGs 1 and 3 opening due to the loss of PNA and PNC. 40AO-9ZZ13 Loss of Class instrument or control power is the correct procedure.

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B. Correct: De-energizing initiation relays on Channel A and C will result in RTSGs 1 and 3 opening due to the loss of PNA OR PNC—either loss will send the crew to 40AO-9ZZ13 Loss of Class instrument or control power.Entry conditions for Inadvertent ESFAS are not met and will not correct this condition.

C. Incorrect: PNA and PNC have tripped which will result in RTSGs 1 and 3 opening due to the loss of PNA and PNC.

D. Incorrect: De-energizing initiation relays on Channel A and C will result in RTSGs 1 and 3 opening due to the loss of PNA OR PNC. Entry conditions for Inadvertent ESFAS are not met and will not correct this condition.

6.

This Exam Level	SRO
Appears on:	SRO EXAM 2009
	SRO EXAM 2012
	Tier 1 Group 1
K/A #	4.4 E05 EA2.2
Importance Rating:	4.2

Given the following conditions:

- Pressurizer pressure is 1600 psia and stable.
- RCS temperature is being controlled with SG 2
- Loop 1 T-cold is 362°F and stable.
- Loop 1 T-hot is 390°F and stable.
- Loop 2 T-cold is 380°F and stable..
- Loop 2 T-hot is 395°F and stable.
- REP CET is 397°F and stable.
- SIAS, CIAS, MSIS, and CSAS have automatically actuated.
- Safety Injection flow is adequate.
- There is no activity present in the steam plant or containment.
- SG 1 WR level is 0%.
- SG 2 WR level is 65% and rising.

The CRS should implement ____(1)____ **AND** ____(2)____.

A. (1) 40EP-9EO05 (ESD) (2) equalize loop T-colds at 362 °F then initiate a cooldown.

B. (1) 40EP-9EO05 (ESD) (2) lower RCS pressure to within Pressure/Temperature limits.

C. (1) 40EP-9EO09 (FRP) HR is jeopardized (2) equalize loop T-colds at 362 °F then initiate a cooldown.

D. (1) 40EP-9EO09 (FRP) HR is jeopardized (2) lower RCS pressure to within Pressure/Temperature limits.

Answer: B

Reference Id:	Q43904	
Difficulty:	4.00	
Time to complete:	4	
10CFR Category:	CFR 55.43 (5)	55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Cognitive Level:	Comprehension / Anal	
Question Source:	PV Bank Not Modified	
Comment:		

Proposed reference to be provided to applicant during examination: 40OP-9EO010 (Standard Appendix) 2 Pages 1 and 2

Technical Reference: ESD, 40EP-9EO06 / Tech guide and standard appendices

K&A: Adherence to appropriate procedures and operation within the limitations in the facility*s license and amendments. <u>Excess Steam Demand</u>

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Learning Objective: L11210 Given that the EOPs are being performed and specific plant conditions are given, determine whether or not the plant is over subcooled, and if it is what actions must be taken in accordance with the appropriate procedure.

Justification:

A. Incorrect: Equalizing Loop Tcolds is required per step 14 of ESD, but a 2 hour soak is required.

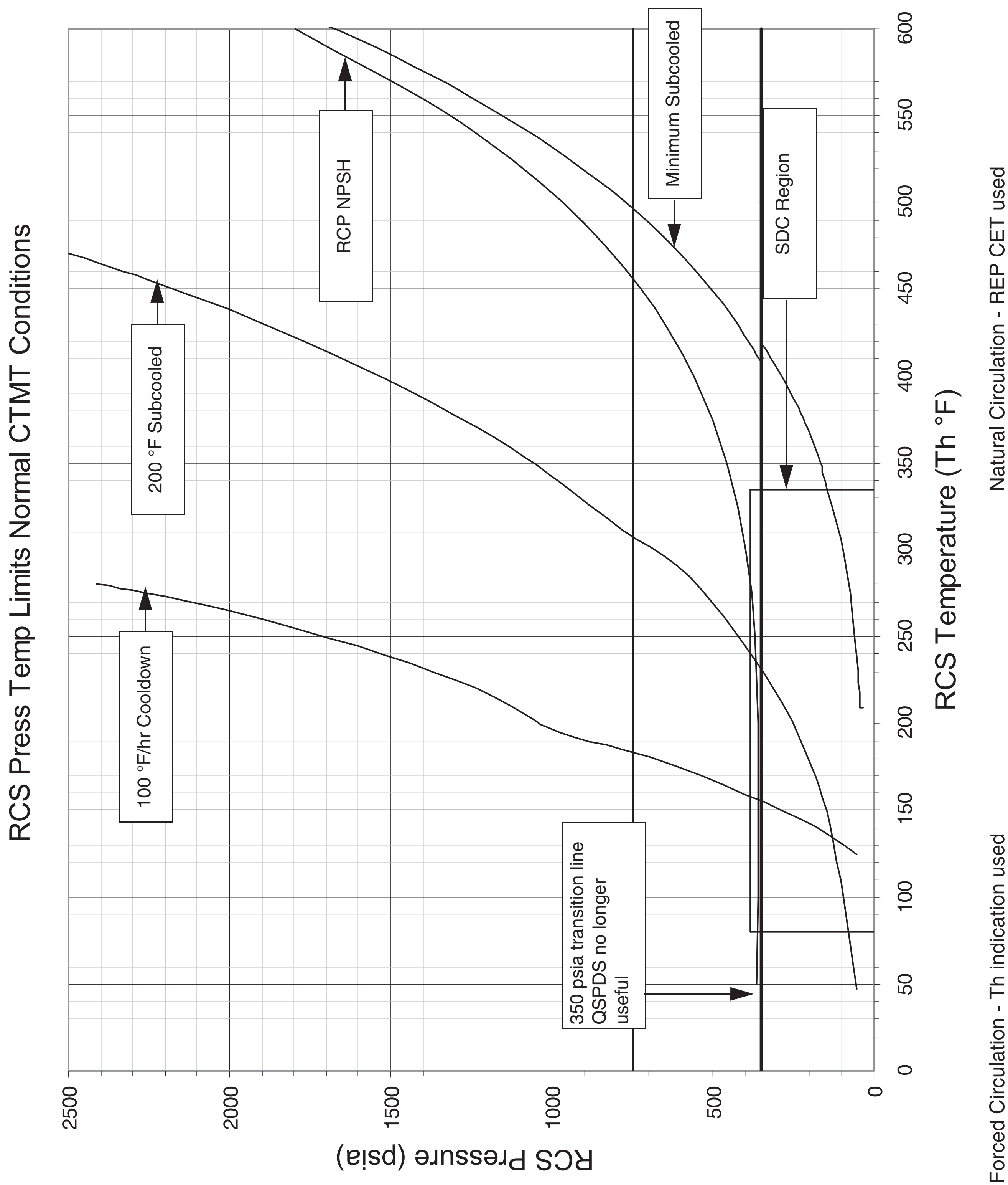
B. Correct: Per Step 14 a and Step 14 e. of ESD Maintain Tc within the P/T limits of App 2 and if PT limits were exceeded and RCPs are secured then a 2 hour soak is required at current conditions.

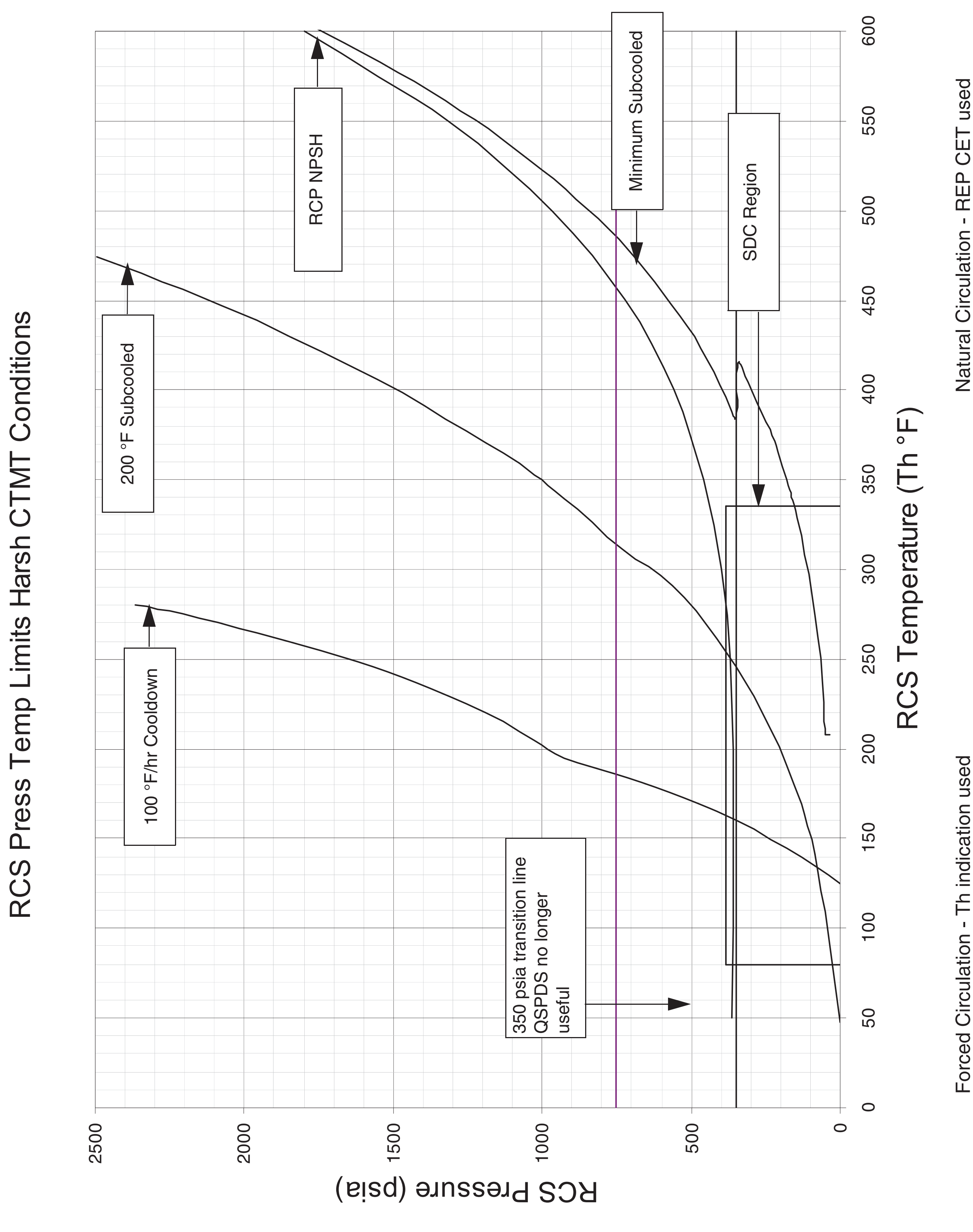
C. Incorrect: Equalizing Loop Tcolds is required per step 14 of ESD, but a 2 hour soak is required. The FRP is not the appropriate ORP due to single event in progress and HR is not jeopardized.

D. Incorrect: Action is correct but the FRP is not the appropriate ORP due to single event in progress and HR is not jeopardized.

Appendix 2,

Figures





RCS Press Temp Limits Harsh CTMT Conditions
RCS Pressure (psia)
2500
2000
1500
1000
500
0
RCS Temperature (Th °F)
0
50
100
150
200
250
300
350
400
450
500
550
600
200 °F Subcooled
100 °F/hr Cooldown
RCP NPSH
Minimum Subcooled
SDC Region
350 psia transition line QSPDS no longer useful
Natural Circulation - REP CET used
Forced Circulation - Th indication used

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7.

This Exam Level	SRO
Appears on:	SRO EXAM 2012 Tier 1 Group 2
K/A #	003 2.2.38
Importance Rating:	4.5

Given the following conditions:

- Unit 2 is operating at 100% ARO.
- A Regulating Group 5 CEA has dropped completely into the core.
- All required actions are complete.

Which ONE of the following describes Technical Specification 3.1.5 (CEA Alignment)?

CEA alignment must be restored within a maximum of ______ hour(s).

A. 1

B. 2

C. 6

D. 12

Answer: B

Reference Id:	Q43907	
Difficulty:	3.00	
Time to complete:	3	
10CFR Category:	CFR 55.43 (1)	55.43 (1) Conditions and limitations in the facility license.
Cognitive Level:	Memory	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: Technical Specification 3.1.5

K&A: Knowledge of conditions and limitations in the facility license. Dropped Control Rod

Learning Objective: Given plant conditions and Technical Specification action statements that are greater than one hour apply the action statements that are greater than one hour for T.S. 3.1 in accordance with Tech Spec 3.1.

Justification:

A. Incorrect: 1 hour applies to reducing THERMAL POWER in accordance with the COLR.
B. Correct: TS 3.1.5 Condition A.2 requires CEA alignment to be restored within 2 hours.
C. Incorrect: 6 hours applies to being in Mode 3 within 6 hours if the CEA alignment or Power limit if condition A can not be met.
D. Incorrect: 12 hours applies to the frequency that CEAs with inoperable position indicators be verified.

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8.

This Exam Level	SRO
Appears on:	SRO EXAM 2012 Tier 1 Group 2
K/A #	4.2 024 AA2.06
Importance Rating:	3.7

Given the following conditions:

Initial Conditions:

- Unit 1 is operating at 100%.
- CEDMCS is in Automatic.
- A New Purification Letdown Ion Exchanger was just placed in service at the end of last shift.

Subsequently:

- Tavg is 591^{0}F and rising slowly.
- A Low Rate CEA insertion demand exists.
- CEAs begin inserting.

Which ONE of the following would cause this condition and what procedure will be used to respond?

A. RWT to CVCS gravity feed isolation (CHE-HV-536) is leaking by, manually **isolate** per 40OP-9CH02 (Purification System).

B. New letdown IX not appropriately borated prior to placing in service, manually **isolate** per 40OP-9CH02 (Purification System).

C. New letdown IX not appropriately borated prior to placing in service, **borate** the RCS per 40OP-9CH01 (CVCS Normal Operations).

D. RWT to CVCS gravity feed isolation (CHE-HV-536) is leaking by, **borate** the RCS per 40OP-9CH01 (CVCS Normal Operations).

Answer: C

Reference Id: Q43890
Difficulty: 3.00
Time to complete: 3
10CFR Category: CFR 55.43 (5) 55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Cognitive Level: Comprehension / Anal
Question Source: New
Comment:
Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40OP-9CH01 (CVCS Normal Operations).

K&A: Ability to determine and interpret the following as they apply to the Emergency Boration: When boron dilution is taking place.

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Learning Objective: Identify how the dilution will be mitigated

Justification:

A. Incorrect: CHE-HV-536 is the RWT gravity feed isolation valve, the RWT is a borated source of water that if it were to leak by RCS Temperature would lower. 40OP-9CH02 is the procedure that provides direction to borate the IX prior to placing in service, but doesn't provide direction to isolate and borate the RCS to remedy to situation.

B. Incorrect: An IX that has not been appropriately borated will resulting in the RCS temperature rise and the CEA insertion, but 40OP-9CH02 is the procedure that provides direction to borate the IX prior to placing in service, but doesn't provide direction to isolate and borate the RCS to remedy to situation.

C. Correct: An IX that has not been appropriately borated will result in the RCS temperature rise and the CEA insertion. 40OP-9CH01 is the procedure that directs borating the RCS to maintain Tc on program.

D. Incorrect: CHE-HV-536 is the RWT gravity feed isolation valve, the RWT is a borated source of water that if it were to leak by RCS Temperature would lower. 40OP-9CH01 is the procedure that directs borating the RCS to maintain Tc on program.

9.

This Exam Level	SRO
Appears on:	SRO EXAM 2012 Tier 1 Group 2
K/A #	2.4.4
Importance Rating:	4.7

Given the following conditions:

- Unit 1 is operating at 100% power.
- PZR level is 47% and trending down.
- PBA-S03 is de-energized.
- RCS T-cold is 557°F and stable.
- RCS T-hot is 612°F and stable.
- CHB-P01 is the only Charging Pump available and is operating.
- Letdown has been isolated.
- Containment temperature and humidity are rising.

Subsequently

- The reactor is manually tripped.
- SPTAs are complete
- Containment temperature and humidity are rising.
- PZR level is 28% and trending down.
- RCS T-cold is 564°F and stable.
- RCS T-hot is 567°F and stable.

The operating crew is required to trip the Reactor, perform SPTAs and implement ...

A. 40EP-9EO05 (ESD) and initiate an MSIS.

B. 40EP-9EO03 (LOCA) and initiate a SIAS/CIAS.

C. 40EP-9EO09 (FRP), CI-1 to restore safety functions.

D. 40EP-9EO09 (FRP), MVAC-1 to restore safety functions.

Answer: B

Reference Id:	Q6849	
Difficulty:	3.00	
Time to complete:	4	
10CFR Category:	CFR 55.43 (5)	55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Cognitive Level:	Comprehension / Anal	
Question Source:	PV Bank Not Modified	
Comment:		

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Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40EP-9EO03 (LOCA)

K&A: Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures. Excess RCS Leakage

Learning Objective: Determine if the Excessive Leakage AOP should be executed in accordance with 40AO-9ZZ02.

Justification: SRO level for this is a diagnostic of the plant post SPTAs which requires the candidate to assess plant conditions and then selecting a procedure to mitigate, recover or with which to proceed.

A. Incorrect: ESD is not the appropriate ORP due to RCS Tcold not lowering; the CTMT parameters and lowering PZR level are indication of a possible ESD. MSIS will not mitigate the event due to being inside the CTMT.
B. Correct: LOCA is correct with the indications provided. PZR level lowering with letdown isolated and one all available charging pumps running. SIAS/CIAS will be required due to the degrading conditions resulting from the LOCA into the CTMT.
C. Incorrect: FRP CI-1 Condition 2 is met; therefore the Safety Function is not jeopardized.
D. Incorrect: FRP MVAC-1 is met; therefore the Safety Function is not jeopardized.

NOTE: Same as Q44010

10.

This Exam Level	SRO
Appears on:	SRO EXAM 2007
	SRO EXAM 2012
	Tier 1 Group 2
K/A #	4.4 E09 EA2.2
Importance Rating:	4.0

Given the following conditions:

Radiation Monitor status just prior to Reactor trip is as follows:

- RU-139 (Main Steam Line SG 1) is in ALERT alarm.
- RU-140 (Main Steam Line SG 2) is in HIGH alarm.
- RU-142 (Main Steam Line N-16) channels 1/2 are ALERT alarm.
- RU-142 (Main Steam Line N-16) channels 3/4 are in HIGH alarm.

Current plant conditions:

- SG 1level is 51% WR and rising.
- SG 1 pressure is 1200 psi and stable.
- SG 2 level is 28% WR and lowering.
- SG 2 pressure 1070 psi and lowering.
- Containment temperature is 195°F.
- Containment pressure 9.0 psig.
- RCPs have been tripped.
- All expected ESFAS actuations have initiated.
- RU-16, Containment Operating Level Monitor, is in ALERT alarm.

Which ONE of the following mitigation strategies would the CRS direct?

A. Implement 40EP-9EO04 (SGTR), feed SG 1 to 45% NR, secure feed to SG 2.

B. Implement 40EP-9EO09 (FRP), feed SG 1 to 45% NR, secure feed to SG 2.

C. Implement 40EP-9EO04 (SGTR), maintain flow to SG 1 and feed SG 2 to 45% NR at 1360 - 1600 gpm.

D. Implement 40EP-9EO09 (FRP), maintain flow to SG 1 and feed SG 2 to 45% NR at 1360 - 1600 gpm.

Answer: B

Reference Id:	Q10294	
Difficulty:	3.00	
Time to complete:	3	
10CFR Category:	CFR 55.43 (5)	55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Cognitive Level:	Comprehension / Anal	
Question Source:	PV Bank Not Modified	

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Comment:

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40EP-9EO09 (FRP)

K&A: Ability to determine and interpret the following as they apply to the (Functional Recovery): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments

Learning Objective: L90459 Diagnose FRP event in progress

Justification: This is a dual event SGTR and ESD (inside contmt) implement the FRP. The strategy is to restore the good SG to 45% NR without overcooling the RCS.

A. Incorrect: SGTR is the wrong procedure, however the strategy is correct.
B. Correct: FRP directs these actions. SG 1 is not faulted so it should be restored to 45 -60% NR, we are not expected feed a faulted SG with another SG available for Heat Removal.
C. Incorrect: SGTR is the wrong procedure and 1360-1600 gpm is the strategy for a SGTR with steam releasing to atmosphere.
D. Incorrect: FRP is correct but the strategy is wrong for a SG ruptured to cntmt.

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11.

This Exam Level	SRO
Appears on:	SRO EXAM 2012 Tier 2 Group 1
K/A #	3.2 004 A2.26
Importance Rating:	3.0

Given the following conditions:

- Unit 1 is operating at 100% power.
- VCT Level is 40% and stable.
- The RO reports that VCT Pressure is 3 psig and lowering.

Which ONE of the following describes the impact of this condition on the CVCS system and what action will mitigate the event?

Primary Coolant (RCS) will have ....

A. increased Hydrogen concentrations, **ISOLATE** the VCT from gaseous radwaste per 40AO-9RK3A (B03 ARP).

B. increased Oxygen concentrations, **ISOLATE** the VCT from gaseous radwaste per 40AO-9RK3A (B03 ARP).

C. increased Hydrogen concentrations, **VENT** the VCT to gaseous radwaste per 40OP-9CH01 (CVCS Normal Operations).

D. increased Oxygen concentrations, **VENT** the VCT to gaseous radwaste per 40OP-9CH01 (CVCS Normal Operations).

Answer: B

Reference Id:	Q43909	
Difficulty:	3.00	
Time to complete:	3	
10CFR Category:	CFR 55.43 (5)	55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Cognitive Level:	Comprehension / Anal	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40AL-9RK3A (B03A Alarm Response Procedure)

K&A:
Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Low VCT pressure

Learning Objective: Explain the operation of the Volume Control Tank under normal operating conditions.

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Justification:

A. Incorrect – Excessive H2 concentrations are not a concern for this event, When Degassing the RCS, VCT pressure is lowered to promote gasses coming out of solution. VCT TRBL alarm actuates at 5 psig, the ARP for high pressure directs venting to the gaseous radwaste system.
B. Correct – With the lower VCT pressure the H2 will come out of solution which will eventually lead to higher than desired O2 concentrations. VCT TRBL alarm actuates at 5 psig, the ARP directs isolating the VCT from the gaseous radwaste system.
C. Incorrect – Excessive H2 concentrations are not a concern for this event, When Degassing the RCS, VCT pressure is lowered to promote gasses coming out of solution. 40OP-9CH01 is the procedure used to establish and maintain VCT H2 pressures during normal conditions, this is an alarm condition. VCT TRBL alarm actuates at 5 psig.
D. Incorrect – With the lower VCT pressure the H2 will come out of solution which will eventually lead to higher than desired O2 concentrations. 40OP-9CH01 is the procedure used to establish and maintain VCT H2 pressures during normal conditions, this is an alarm condition. VCT TRBL alarm actuates at 5 psig.

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12.

This Exam Level	SRO
Appears on:	SRO EXAM 2012 Tier 2 Group 1
K/A #	3.2 006 A2.04
Importance Rating:	3.8

Given the following initial conditions:

- Unit 1 automatically tripped from 100% power.
- SPTAs are in progress.
- The crew has manually initiated SIAS/CIAS.
- Adequate SI flow has been verified.
- 525 KV East and West Bus Voltage meters indicate 0 Vac.
- Pressurizer pressure is 1450 psia and lowering.
- Pressurizer level is 20% and lowering.
- SG 1 & 2 pressures being controlled at 1180 psia with ADVs.
- PBA-S03 is energized by DG "A".
- DG "B" has tripped on "overspeed".

Subsequently:

- HPSI pump "A" discharge pressure degrades to 1000 psig.

Which ONE of the following describes the impact on Safety Injection and the appropriate procedure required to mitigate these events?

HPSI flow lowers to ..

A. zero (0) gpm, utilize 40EP-9EO03 (LOCA).

B. half its original value, utilize 40EP-9EO03 (LOCA).

C. zero (0) gpm, utilize 40EP-9EO09 (FRP) MVAC-2 DGs.

D. half its original value, utilize 40EP-9EO09 (FRP) MVAC-2 DGs.

Answer: C

Reference Id:	Q44014	
Difficulty:	3.00	
Time to complete:	3	
10CFR Category:	CFR 55.43 (5)	55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Cognitive Level:	Comprehension / Anal	
Question Source:	New	

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Comment:
Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40OP-9EO09 (FRP)

K&A: Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Improper discharge pressure.

Learning Objective: Describe how the FRP will maintain or recover the Maintenance of Vital Auxiliaries.

Justification:

A. Incorrect: Per Standard Appendix 2, HPSI Pump Delivery Curves, 0 gpm does not meet the acceptable region of the curve, due to the Loss of Offsite Power (LOOP) and the HPSI A degraded condition, HPSI A is not available, the Loss of the EDG B results in HPSI B not being available. The LOCA procedure does not provide direction to Crosstie PB busses to restore electrical power to the undamaged HPSI B.

B. Incorrect: HPSI Flow will drop to 0 not half. HPSI B is not available and HPSI A is operating below the pressure of the RCS. Due to the Loss of Offsite Power (LOOP) and the HPSI A degraded condition, HPSI A is not available, the Loss of the EDG B results in HPSI B not being available. The LOCA procedure does not provide direction to Crosstie PB busses to restore electrical power to the undamaged HPSI B.

C. Correct: Due to the Loss of Offsite Power (LOOP) and the loss of PBA-S03 along with the HPSI A degraded condition HPSI A is not available, the Loss of the EDG B results in HPSI B not being available. FRP MVAC-2 will provide direction to restore electrical power to PBB-S04 and start HPSI B to restore adequate HPSI delivery.

D. Incorrect: HPSI Flow will drop to 0 not half. HPSI B is not available and HPSI A is operating below the pressure of the RCS Per Standard Appendix 2, HPSI Pump Delivery Curves, 0 gpm does not meet the acceptable region of the curve. FRP MVAC-2 is the correct procedure.

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13.

This Exam Level	SRO
Appears on:	SRO EXAM 2012 Tier 2 Group 1
K/A #	3.3 010 A2.02
Importance Rating:	3.9

Given the following conditions:

- Unit 1 is operating at 100% power.
- PZR pressure was reported as 2230 psia and lowering.
- Main spray valves 100E & 100F indicate full open.
- All attempts to close Main Spray valves have failed.
- Pressurizer pressure is 2050 psia and continuing to lower.

This will cause the RCN-PIC-100 (PPCS master controller) output to go to _____(1)____ and the CRS should trip the reactor, ____(2)_____.

A. (1) minimum, (2) close IAA-UV-2 and enter 40EP-9EO02 (Reactor Trip).

B. (1) maximum, (2) close IAA-UV-2 and enter 40EP-9EO02 (Reactor Trip).

C. (1) minimum, (2) stop all 4 RCPs and enter 40EP-9EO07 (LOOP/LOFC).

D. (1) maximum, (2) stop all 4 RCPs and enter 40EP-9EO07 (LOOP/LOFC).

Answer: C

Reference Id:	Q43920	
Difficulty:	2.00	
Time to complete:	2	
10CFR Category:	CFR 55.43 (5)	55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Cognitive Level:	Comprehension / Anal	
Question Source:	New	
Comment:		

PVNGS Operating Experience

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40AL-9RK4A (Panel B04A ARP), 40EP-9EO07 (LOOP/LOFC)

K&A: Ability to (a) predict the impacts of the following malfunctions or operations on the PZR PCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Spray valve failures

Learning Objective: Describe the response of the Pressurizer Pressure Control System to a failure of an input transmitter.

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Justification:

A. Incorrect: The Pressurizer Pressure Master Controller is a reverse acting controller, minimum is correct. **(OE)** Shutting IAA-UV-2 was previously an option in the B04A Alarm Response procedure. PVNGS experienced a plant event where IA was isolated to CTMT and IA pressure maintained Spray Valves open well past the expected response time. Reactor Trip is not the appropriate procedure.

B. Incorrect: The Pressurizer Pressure Master Controller is a reverse acting controller, maximum is wrong.**(OE)** Shutting IAA-UV-2 was previously an option in the B04A Alarm Response procedure. PVNGS experienced a plant event where IA was isolated to CTMT and IA pressure maintained Spray Valves open well past the expected response time.

C. Correct:The Pressurizer Pressure Master Controller is a reverse acting controller. A decrease in controller output results in an increase in system pressure. The B04A Alarm Response procedure directs stopping all RCPs. The Crew must trip the reactor to stop all 4 RCPs. Tripping all 4 RCPs will result in a LOFC.

D. Incorrect: The Pressurizer Pressure Master Controller is a reverse acting controller, maximum is wrong. The Crew must trip the reactor to stop all 4 RCPs. Tripping all 4 RCPs will result in a LOFC.

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14.

This Exam Level	SRO
Appears on:	SRO EXAM 2012
	Tier 2 Group 1
K/A #	059 2.4.11
Importance Rating:	4.2

Given the following initial conditions:

- Unit 1 is operating at 48% power following a Reactor Power Cutback due to a trip of the B main feedwater pump.
- CEA Subgroups 4, 5, and 22 drop to the bottom of the core.
- CEA Reg group 3 CEAs are automatically inserting as expected to maintain RCS program temperature.
- Turbine Load is approximately 850 MW and stable.

During plant stabilization the following conditions are observed:

- The CRS has implemented 40AO-9ZZ09 (RPCB - Loss of Feedpump).
- CEDMCS has been placed in Manual Sequential (MS).
- CEA 42 (Regulating Group 3, 4 Finger CEA) is at 140 inches, the remaining CEAs in Gp 3 are at 132 inches.
- CPC Pt. ID 0187 (ASI) average value is minus (-) .38 and slowly trending to the top of the core.

Which ONE of the following actions and conditions are both correct?

A. Withdraw Gp 3 CEAs to control ASI per 40AO-9ZZ09 (RPCB).

B. Withdraw Gp 3 CEAs to within 6.6 inches of CEA 42 per 40AO-9ZZ09 (RPCB).

C. Trip the reactor, perform SPTAs and enter 40EP-9EO01 (Rx Trip) due to exceeding ASI limits.

D. Trip the reactor, perform SPTAs and enter 40EP-9EO01 (Rx Trip) due to the CEA mis-alignment.

Answer: D

Reference Id:	Q43913	
Difficulty:	3.00	
Time to complete:	3	
10CFR Category:	CFR 55.43 (5)	55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Cognitive Level:	Comprehension / Anal	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40AO-9ZZ09,RPCB (loss of Feedpump)

K&A: Main Feedwater; Knowledge of abnormal condition procedures.

Learning Objective: Describe the contingency action(s) that the operator would be required to take if RPCB does not operate properly.

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Justification:

A. Incorrect: ASI is negative and trending to to top (more Negative) withdrawing CEAs would drive ASI even more negative and possibly exceed limits.
B. Incorrect: Below insertion limits, a greater than 6.6 inch deviation requires a reactor trip. Examinees may confuse this with the CEA Technical Specification condition reguarding group deviation and not recognize the AOP requirement trip.
C. Incorrect: ASI limits are not currently being exceeded.
D. Correct:During a reactor power cutback an 8 inch deviation, no matter what direction, requires a reactor trip and entry into SPTAs.

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15.

This Exam Level	SRO
Appears on:	RO EXAM 2012
	Tier 2 Group 1
K/A #	3.8 078 A2.01
Importance Rating:	2.4

Given the following initial conditions:

- Unit 1 is operating at 100% power.
- Instrument Air compressor "A" is operating and maintaining IA pressure at 115 psig.
- Instrument Air Dryer IAN-M01C is in service.

Subsequently

- Window 7B01A (INST AIR SYSTEM TROUBLE) is in alarm.
- Alarm point IAPDS238 (Instrument Air Filter After Filter "C" Differential Press Hi) is alarming.
- IA-PI-32 (IA Header Pressure) is 98 psig and dropping slowly.
- Instrument Air compressors "B and C" have started.

Which ONE of the following describes the impact to the IA system and the appropriate procedural action?

The IA header pressure will continue to LOWER ...

A. to approximately 85 psig, implement 40AO-9ZZ06 (Loss of Instrument Air), to perform leak isolation.

B. to approximately 85 psig, implement 40AO-9ZZ06 (Loss of Instrument Air), to valve in another air dryer.

C. requiring a reactor trip, implement 40EP-9EO01 (Reactor Trip) and maintain Heat Removal with Auxiliary Feedwater pumps and Atmospheric Dump Valves.

D. requiring a reactor trip, implement 40EP-9EO01 (Reactor Trip) and maintain Heat Removal with Main Feedwater pumps and the Steam Bypass Control Valves.

Answer: B

Reference Id:	Q44024	
Difficulty:	0.00	
Time to complete:	0	
10CFR Category:	CFR 55.43 (5)	55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Cognitive Level:	Comprehension / Anal	
Question Source:	New	

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Comment:
Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40AO-9ZZ06 (Loss of Instrument Air) 40AL-9RK7B (B07B ARP)

K&A: Ability to (a) predict the impacts of the following malfunctions or operations on the IAS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Air dryer and filter malfunction

Learning Objective: Determine the mitigating strategies of the Loss of Instrument air AOP.

Justification: From the stem the examinee should be able to determine that the in-service dryer "C" has a high dp and is causing the low pressure in the IA header. The Nitrogen backup valve opens at 85 psig and will maintain IA header pressure until the standby dryer can be placed in service iaw 40AO-9ZZ06 (Loss of Instrument Air)

A. Incorrect: the lowering IA pressure and hi dp could be attributed to a large leak but from the indications given leak isolation is not required.

B. Correct: the automatic features associated with the IA system is the N2 Backup alignment and Stby IA comp starting. 40AO-9zz06 has actions to place the standby dryer in service.

C. Incorrect: N2 Backup valve automatically opens at 85 psig to maintain system pressure. A loss of IA would require the use of AFW and ADVs.

D. Incorrect: N2 Backup valve automatically opens at 85 psig to maintain system pressure. If pressure stayed at 85 psig then MFW and SBCVs would be available.

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16.

This Exam Level	SRO
Appears on:	SRO EXAM 2012 Tier 2 Group 1
K/A #	002 2.2.40
Importance Rating:	4.7

Given the following conditions:

- Unit 1 is in Mode 4 during a refueling outage.
- RCS Pressure is 450 psia and stable.
- RCA-HV-106 (PZR/RV HEAD VENT TO CTMT) was declared INOPERABLE 5 hours ago.

Given LCO 3.4.12 (Pressurizer Vents) and appendix C of 40OP-9RC04 (RV Head and Pressurizer Vent System) which ONE of the following describes the required action (if any)?

A. No action required due to only ONE (1) path is INOPERABLE.

B. Restore ONE (1) additional pressurizer vent path to OPERABLE within 1 hour.

C. Restore TWO (2) additional pressurizer vent paths to OPERABLE status within 67 hrs.

D. Lower RCS pressure to < 385 psia within 19 hours.

Answer: C

Reference Id:	Q43813	
Difficulty:	3.00	
Time to complete:	3	
10CFR Category:	CFR 55.43 (2)	55.43 (2)Facility operating limitations in the technical specifications and their bases.
Cognitive Level:	Comprehension / Anal	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: Tech Spec 3.4.12, Diagram of PZR Vents from 40OP-9RC04 (RCGVS)

Technical Reference: Tech Specs

OPERATING EXPERIENCE QUESTION

K&A: Ability to apply Technical Specifications for a system: RCS

Learning Objective: Given conditions when an LCO is not met, apply Tech Spec Section 3.4.12 (PZR Vents) in accordance with Tech Spec 3.4.12.

Justification:

A. Incorrect - Candidate may read the Tech Spec as No Action due to only one vent path INOPERABLE.
B. Incorrect - Incorrect because 2 paths are inoperable when RCA-106 is declared.
C. Correct - This will ensure that all 4 vent paths are INOPERABLE and the LCO can be exited.
D. Incorrect - This could be chosen if it is determined that conditions A and B can not be met.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Pressurizer Vents

LCO 3.4.12 Four pressurizer vent paths shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.
MODE 4 with RCS pressure ≥ 385 psia.

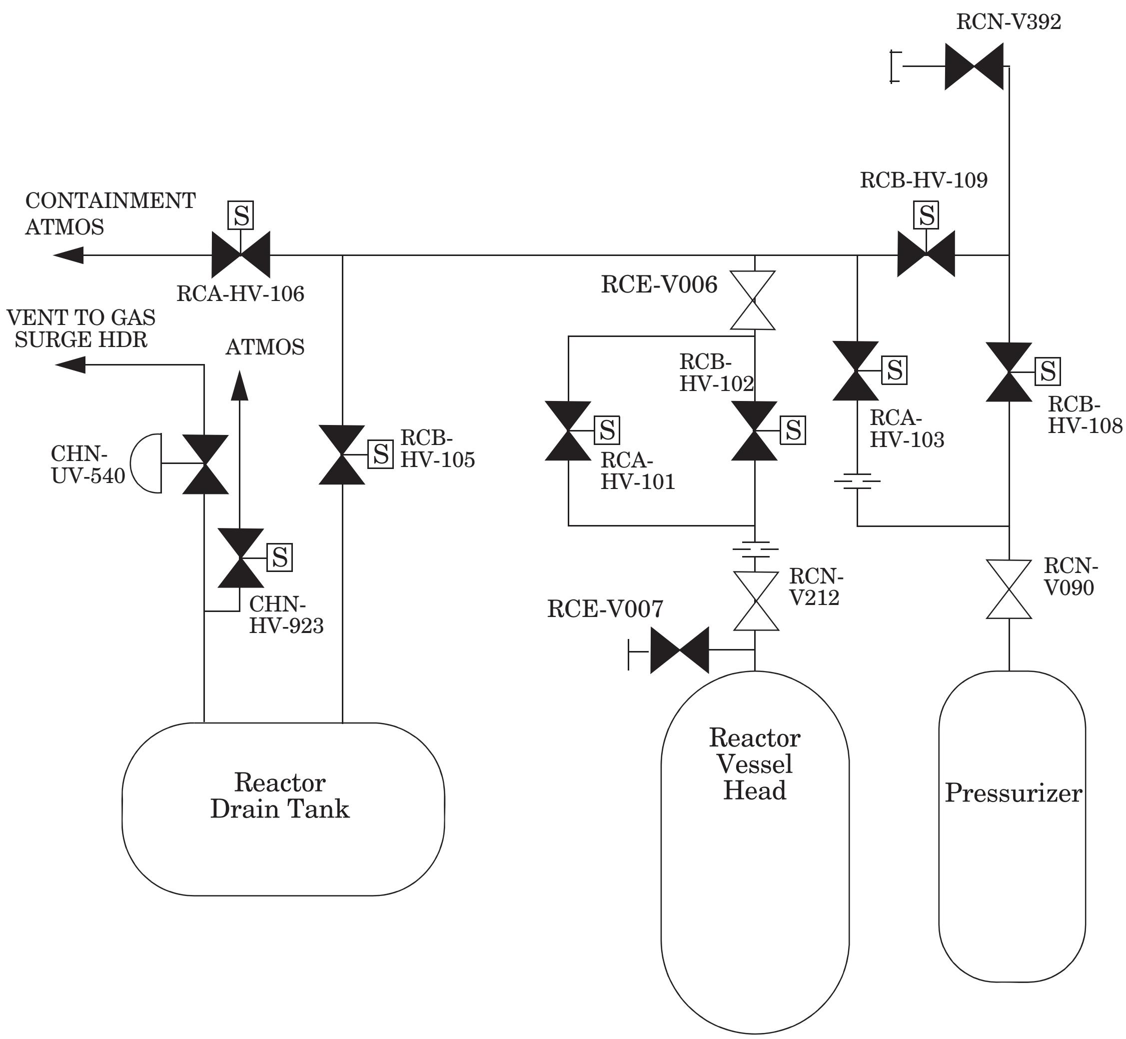
ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Two or three required pressurizer vent paths inoperable.	A.1 Restore required pressurizer vent paths to OPERABLE status.	72 hours
B. All pressurizer vent paths inoperable.	B.1 Restore one pressurizer vent path to OPERABLE status.	6 hours
C. Required Action and associated Completion Time of Condition A, or B not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4 with RCS pressure < 385 psia.	6 hours 24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	Perform a complete cycle of each Pressurizer Vent Valve.	18 months
SR 3.4.12.2	Verify flow through each pressurizer vent path.	18 months

Appendix C - RV Head and Pressurizer Vent System



This diagram is only a simplified likeness of system diagrams M-RCP-001 and M-CHP-003

End of Appendix C

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17.

This Exam Level	SRO
Appears on:	SRO EXAM 2008 SRO EXAM 2012 Tier 2 Group 2
K/A #	3.4 041 A2.02
Importance Rating:	3.9

Given the following conditions:

- Unit 1 was operating at 100% power.
- SBCV #6 failed 100% open.
- A low SG pressure reactor trip and MSIS have both automatically initiated.
- T-avg dropped to 570°F on the reactor trip.
- T-cold dropped to 546°F before the MSIS was initiated.

Which ONE of the following describes the impact to the SBCS and the appropriate response?

A. SBCS group X valves "Quick Opened" on the trip, perform SPTAs and implement 40EP-9EO02 (Rx Trip).

B. "Quick Open" was blocked to all SBCS valves on the trip, perform SPTAs and implement 40EP-9EO02 (Rx Trip).

C. SBCS group X valves "Quick Opened" on the trip, perform SPTAs and implement 40EP-9EO05 (ESD).

D. "Quick Open" was blocked to all SBCS valves on the trip, perform SPTAs and implement 40EP-9EO05 (ESD).

Answer: D

Reference Id:	Q22473	
Difficulty:	3.00	
Time to complete:	4	
10CFR Category:	CFR 55.43 (5)	55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Cognitive Level:	Comprehension / Anal	
Question Source:	PV Bank Not Modified	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40EP-9EO05 (ESD) Simplified drawings, LOIT lesson plan

K&A: Ability to (a) predict the impacts of the following malfunctions or operations on the SDS; and (b) based on those predictions or mitigate the consequences of those malfunctions or operations: Steam valve stuck open

Learning Objective: L65641 Describe the interrelationship between the Steam Bypass Control System and the Main Steam System

Justification

A. Incorrect: Quick Open is blocked on Rx trip with T-avg < 573.5°F. Should stabilize Tcold at 546 due to the MSIS actuation and enter ESD.

B. Incorrect: Quick Open is blocked but Rx Trip is not appropriate due to the "ESD" causing the low temperature.

C. Incorrect: Quick Open is blocked. due to the low Tavg, Rx Trip is not the correct procedure. ESD will stabilize Tcold and Rx Trip will not.

D. Correct: Quick Open is blocked on Rx trip with T-avg < 573.5°F, OPS expectations requires that ESD be entered if T-cold goes below 560°F due to an ESD event.

18.

This Exam Level:	SRO
Appears on:	SRO EXAM 2012 Tier 2 Group 2
K/A #:	3.7 072 A2.03
Importance Rating:	2.9

Given the following conditions:

Initial Conditions:

- Unit 2 is in an outage.
- Core Off Load is in progress.
- RU-37 (Power Access Purge Area Monitor Train A) is inoperable and in bypass on BOP-ESFAS.

Subsequently:

- RU-38 (Power Access Purge Area Monitor Train B) power supply fuses blow.

Which ONE of the following predicts the expected plant response and appropriate actions?

CPIAS actuates and provides a cross trip to ____(1)____.

IF the CPIAS did not actuate properly the CRS must suspend ____(2)____.

A. (1) FBEVAS
(2) movement of irradiated fuel assemblies in the fuel building per TRM 3.9.104 (FBEVAS).

B. (1) CREFAS
(2) movement of irradiated fuel assemblies in the fuel building per Tech Spec 3.3.9 (CREFAS).

C. (1) FBEVAS
(2) core alterations and movement of irradiated fuel assemblies in the CTMT per Tech Spec 3.3.8 (CPIAS).

D. (1) CREFAS
(2) core alterations and movement of irradiated fuel assemblies in the CTMT per Tech Spec 3.3.8 (CPIAS).

Answer: D

Reference Id:	Q43922	
Difficulty:	2.00	
Time to complete:	2	
10CFR Category:	CFR 55.43 (5)	55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Cognitive Level:	Memory	
Question Source:	New	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: Technical Specifications, Technical Requirements Manual.

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OPERATING EXPERIENCE QUESTION

K&A: Ability to (a) predict the impacts of the following malfunctions or operations on the ARM system- and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Blown power-supply fuses.

Learning Objective: 65049 Explain the operation of the CPIAS Module.

Justification:

A. Incorrect: CPIAS will cross trip to CREFAS when actuated. The TRM for FBEVAS will not apply and it directs only suspending fuel movements in the Fuel Building.

B. Incorrect: A loss of power to the ARM will result in the BOP-ESFAS module sensing a trip and actuating the CPIAS module which will result in a cross trip signal being sent to the CREFAS module. The TS for CREFAS will not apply but it does apply to irradiated fuel assembly movements..

C. Incorrect: CPIAS will cross trip to CREFAS when actuated. TS is the correct procedure.

D. Correct: A loss of power to the ARM will result in the BOP-ESFAS module sensing a trip and actuating the CPIAS module which will result in a cross trip signal being sent to the CREFAS module. TS 3.3.8 directs suspending core alterations and movement of irradiated fuel in the CTMT immediately.

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19.

This Exam Level	SRO
Appears on:	SRO EXAM 2010
	SRO EXAM 2012
	Tier 3
K/A #	2.1.14
Importance Rating:	3.1

Which ONE of the following describes when a plant-wide announcement is required to be made?

A. Changing from Mode 3 to Mode 2.

B. Energizing PNA-D25 after a permit has been cleared.

C. Starting HCN-A01C (CTMT Normal ACU Fan) from the Control Room.

D. AFB-P01 (Essential Motor Driven Aux Feed Pump) started automatically on AFAS-1.

Answer: A

Reference Id: Q43785
Difficulty: 2.00
Time to complete: 2
10CFR Category: CFR 55.43 (5) 55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Cognitive Level: Memory
Question Source: PV Bank Not Modified
Comment:
Proposed reference to be provided to applicant during examination: NONE

Technical Reference: ODP-1, Operations Department Principles and Standards

K&A:

Knowledge of criteria or conditions that require plant-wide announcements, such as pump starts, reactor trips, mode changes etc.

Justification: The announcement may not be made by the CRS but he will approve or direct it.

A. Correct – Plant-wide announcements shall be made when changing modes.
B. Incorrect – 120 Vac distribution panels are not required to be announced.
C. Incorrect – 480 Vac motor starts are not required to be announced.
D. Incorrect – Equipment that starts automatically is not required to be announced.

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20.

This Exam Level:	SRO
Appears on:	SRO EXAM 2008 SRO EXAM 2012 Tier 3
K/A #:	2.1.25
Importance Rating:	4.2

Given the following conditions:

- Unit-1 has been shutdown for five days and is currently in Mode 5
- The RCS is being maintained at 102 ft 6 inches in preparation for installing Steam Generator Nozzle Dams
- The Steam Generator primary manways are off
- RCS temperature is 135 ºF

Per the tables found in the Unit-1 Safety Analyses Operational Data (SAOD) during a sustained Loss of Shutdown Cooling the RCS ...

A. time to boil is 18.9 minutes

B. time to boil is 23.3 minutes

C. makeup flowrate to compensate for boil off is 76.9 gpm

D. makeup flowrate to compensate for boil off is 98.5 gpm

Answer: D

Reference Id:	Q5424	
Difficulty:	4.00	
Time to complete:	5	
10CFR Category:	CFR 55.43 (5)	55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Cognitive Level:	Comprehension / Anal	
Question Source:	PV Bank Not Modified	
Comment:		

Proposed reference to be provided to applicant during examination: Unit-1 Safety Analysis Operational Data (SAOD)

Technical Reference: Unit-1 Safety Analysis Operational Data (SAOD)

K&A: Ability to interpret reference materials, such as graphs, curves, tables, etc.

Learning Objective: L56598 Provided with Time to Boil curves, determine time to core boiling using the TTB curves in the back of the core data book and describe what this value is used for in accordance with 40EP-9EO11.

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Justification:

A. Incorrect: time to boil at midloop is 14.7 minutes (18.9 comes from flange level after core reload).

B. Incorrect: time to boil at midloop is 14.7 minutes (23.3 comes from flange level prior to core reload).

C. Incorrect: 76.9 gpm is the makeup requirement for midloop after core reload.

D. Correct: this is the makeup rate for midloop prior to core offload.

Manual Number
SAOD Unit 1

Manual Title: SAOD **3990** MWt - Unit 1

Description of Change: This document supersedes revision 1 of the SAOD Unit 1 manual due to the implementation of the replacement steam generators and power uprate. Clarification was provided as requested by Operations (CRAI 2819297) for RCS loop check valve work restrictions. References were also updated. Change bars are not shown (copy of Unit 2 SAOD) due to extensive changes.

		Revision **2**
	Dependent Engineer Date	N/A
	Responsible Engineer Date	Brian Blackmore
CROSS DISCIPLINE REVIEW	Operations (STAs) Date	N/A
	Outage Mgmt Date	N/A
	NED Date	N/A
	Other (Ops Standards) Date	N/A
	Mentor Date	N/A
	Independent Reviewer Date	Ness Kilic
	Responsible Section Leader Date	Craig Hasson
	Responsible Department Leader Date	N/A

Table of Contents

1.0 OBJECTIVE

The Safety Analysis Operational Data (SAOD) manual provides information to Operations and Outage Planning for certain outage shutdown activities. Specifically, this manual provides outage decay heat values, reactor coolant system expected heat up rates, outage time to boil values, outage time to core uncovery values, makeup flow for boiloff, minimum time to reduced flow shutdown cooling operations, and RCS forced flow and RWT temperature requirements to maintain sub-cooled conditions following a loss of shutdown cooling (HPSI once-through cooling).

This information is provided to clarify the time constraints needed for the refueling outage activities that are adversely affected by decay heat following reactor shutdown. It is essential that these constraints be observed in order to ensure that the unit remains in an analyzed condition for postulated events involving a loss of shutdown cooling at reduced inventories.

Questions concerning SAOD Sections 1.0 through 2.7 should be referred to the Transient Analysis section leader. Questions regarding SAOD Sections 2.8 through 2.9 should be referred to the Design Mechanical NSSS section leader.

2.0 METHOD OF SOLUTION AND RESULTS

The SAOD manual simply documents the results from various calculations and analysis packages in a convenient document readily accessible to Operations. No calculations are performed in this manual and no computer codes are used for this manual. The assumptions and input data are documented in the source calculations which are referenced with each table provided.

2.1 Decay Heat Constraints for Outages

The following constraints are required to support the assumptions in the Loss of Shutdown Cooling (LSDC) analyses. (Reference 2, 7, & 12)

- **The pressurizer manway may be removed to provide a hot leg vent path, when the reactor vessel is full (120 ft. elevation) and core decay heat rate is ≤ 20 MW.** At 48 hours post-shutdown, the decay heat rate is not expected to exceed 20.02[1] MW for a rated thermal power of 3990 MWt. A LSDC event with the pressurizer manway removed and Reactor Coolant System level at 131 ft. 5 inches (~ 50% pressurizer level) would not result in uncovering the reactor core for at least a 1-hour period.

- RCS drain operations should not commence unless indicated **RWT level is ≥ 73%** (≥ 50% for cold core post refueling conditions).

- 1-hour after SDC is lost, it is assumed that operators will have reestablished either SDC or forced flow cooling of the RCS using High Pressure Safety Injection (HPSI). The 1-hour period is based on the anticipated maximum time that it may take to place a Gas Turbine Generator (GTG) in service.

1. Even though 20.02 is slightly larger than 20 MW, the 0.1% difference is considered negligible in comparison to the analytical conservatisms described in reference 2.

- **The reactor vessel may be drained down (to the 103'8" elevation), when the pressurizer manway is removed and the core decay heat rate is < 20 MW.** This occurs at a time > 48 hours after shutdown. With the pressurizer manway removed and the RCS drained to the 103'8" elevation (with water in the steam generator tubes), a LSDC event would not result in uncovering of the reactor core for at least 1 hour, even if no credit was given for gravity feed or HPSI during that hour.

- **The reactor vessel may be drained down to midloop (101'6" to 103'1") when the pressurizer manway is removed and the core decay heat is < 16 MW. A decay heat of 16 MW occurs 87 hours after plant shutdown. With the Unit at midloop no credit can be given for liquid in the SG tubes. A LSDC event would not result in uncovering of the reactor core for at least 1 hour period.**

- **The steam generator manways may be removed and nozzle dams installed when the core decay heat is < 16 MW.** This occurs at t > 87 hours after shutdown. If a LSDC event were to occur while the plant was in this configuration, then operator intervention would be critical to achieve a 1-hour coping period. For this level of decay heat the analysis assumes the following:

  1 For hot core mid-loop the hot leg steam generator (SG) manways should be removed close to the same time without delay (minimizing the time reduces steam entrainment concerns) with the manway for the hot leg used for gravity feed removed last.

  2 The hot leg nozzle dams should be installed sequentially without delay (minimizing the time reduces potential steam entrainment inventory losses) with the nozzle dam on the leg used for gravity feed installed first.

  3 Both cold leg nozzle dams should be installed on a SG before installation of the hot leg dam. Additionally, the last nozzle dam installed should be on the hot leg opposite that to which RWT gravity feed is aligned. Installing cold leg dams prior to hot leg dams ensures a flow path for HPSI once through cooling.

- **After nozzle dam installation, the RCS may be reflooded as high as elevation 120 ft. (approximately 20% pressurizer level).** At a decay heat rate of 16 MW and nozzle dams installed, a LSDC event will not result in an RCS peak pressure that exceeds the nozzle dams' rating of 50 psig (i.e., ASME B&PV Code Level D Service Limit), if RCS level is at or below the 120 ft. elevation. This conclusion is valid both for early initiation of cold leg HPSI flow (at 10 minutes), or for hot leg gravity feed for 1 hour followed by cold leg HPSI.

- **If plant operators initiate gravity feed from the RWT to the RCS in response to a LSDC event, the DC-powered valve (J-SIC-UV-0653 or J-SID-UV0654, depending upon the train selected) should initially be throttled approximately 30% open and then throttled as needed so that RWT level decreases approximately 3% every 15 minutes.**

- **No cold leg openings (e.g. RCP impeller work or shaft replacement, RCS cold leg check valves SIEV2x7) before decay heat rate is < 14 MW. This occurs around 118 hours after shutdown.**

- RCP seal work when RCS level is above the bottom of the hot leg is allowed only if maintenance uses a shaft blocking device to prevent lifting the shaft and creating a large cold leg breach.

2.2 Midloop Operation

NOTES 2.2.0

Key Reactor Core Parameters Following a Loss of SDC

During Midloop Operation with A Large or Small

Cold Leg Opening

The key reactor core parameters following a loss of shutdown cooling (SDC) with a large (RCP impeller) or small (RCP seal) cold leg opening are based in part on a computer analysis using the RETRAN code.

Decay heat power is based on Branch Technical Position 9-2 utilizing a 550 EFPD cycle length at 100% power. (3990 MWt)(Reference TA-13-C00-1999-009)

Time to boil is based on the time for the water in the vessel to reach 210 ^{0}F. This is a function of decay heat, mass of water in vessel and initial RCS bulk temperature (inlet Shutdown Cooling Heat Exchanger temperature). It is determined by subtracting the RCS bulk temperature from 210^0, and then dividing by the heatup rate.

Heatup rate is determined by dividing the decay heat by the specific heat capacity for water and by the mass of water available in the vessel at the start of the event.

Time to core uncovery is conservatively assumed to be the amount of time it takes to boil off the water volume below the bottom of the hot leg and above the top of the core. The time associated with the pressurization effects, which result in water lost out the cold leg opening, is neglected. The water lost out the cold leg opening is conservatively assumed to result in a water level at the bottom of the hot leg. Thus, time to core uncovery is a function of decay heat, this volume of water and the pressure over the water. Time to core uncovery does not include the time associated with reaching the boiling temperature.

Makeup rate is the amount of flow required to reach the core to compensate for water loss through boil off. Note that this is the volume that must be delivered to the core in order to maintain constant inventory. Flow diverted out any cold leg opening must be accounted for (i.e. flowrate indications may not be actual flowrates reaching the core).

NOTE: VALUES NOT EXPLICITLY FOUND IN THE TABLES

SHALL BE DETERMINED VIA LINEAR INTERPOLATION

PERFORMED BY THE SHIFT TECHNICAL ADVISOR

TABLE 2.2.1

Key Reactor Core Parameters Following a Loss of SDC

During Midloop Operation with A Large or Small Cold Leg Opening

Prior to Core Reload (3990 MW Core)

Time after Reactor Shutdown (days)	Decay Heat Load (MWth)	Heatup Rate (F/Min.)	Makeup Flowrate (gpm)**	Time after Reactor Shutdown (days)	Decay Heat Load (MWth)	Heatup Rate (F/Min.)	Makeup Flowrate (gpm)
1.0	24.44	8.97	173.5	10	10.42	3.82	74.0
2.0	20.02	7.35	142.1	11	10.05	3.69	71.4
3.0	17.25	6.33	122.5	12	9.72	3.57	69.0
3.5	16.19	5.94	114.9	13	9.43	3.46	67.0
4.0	15.30	5.62	108.6	14	9.16	3.36	65.0
4.5	14.54	5.34	103.2	15	8.92	3.27	63.3
5.0	13.88	5.09	98.5	16	8.70	3.19	61.8
5.5	13.31	4.88	94.5	17	8.48	3.11	60.2
6.0	12.83	4.71	91.1	18	8.29	3.04	58.9
6.5	12.39	4.55	88.0	19	8.10	2.97	57.5
7.0	12.01	4.41	85.3	20	7.93	2.91	56.3
7.5	11.67	4.28	82.9	25	7.15	2.62	50.8
8.0	11.37	4.17	80.7	30	6.53	2.40	46.4
8.5	11.10	4.07	78.8	40	5.59	2.05	39.7
9.0	10.85	3.98	77.0	50	4.92	1.81	34.9
9.5	10.62	3.90	75.4	80	3.76	1.38	26.7

Source of Data: SA-13-C00-1996-004

** The makeup flowrate listed is to compensate for boil off (not required flow to prevent boiling)

TABLE 2.2.2

Key Reactor Core Parameters Following a Loss of SDC During Midloop Operation with A Large or Small Cold Leg Opening

After Core Reload (3990 MW Core)

Time after Reactor Shutdown (days)	Decay Heat Load (MWth)	Heatup Rate (F/Min.)	Makeup Flowrate (gpm)**	Time after Reactor Shutdown (days)	Decay Heat Load (MWth)	Heatup Rate (F/Min.)	Makeup Flowrate (gpm)
1.0	19.06	7.00	135.3	10	8.13	2.98	57.7
2.0	15.62	5.73	110.9	11	7.84	2.88	55.7
3.0	13.46	4.94	95.5	12	7.58	2.78	53.8
3.5	12.63	4.63	89.7	13	7.36	2.70	52.2
4.0	11.93	4.38	84.7	14	7.14	2.62	50.7
4.5	11.34	4.16	80.5	15	6.96	2.55	49.4
5.0	10.83	3.97	76.9	16	6.79	2.49	48.2
5.5	10.38	3.81	73.7	17	6.61	2.43	47.0
6.0	10.01	3.67	71.1	18	6.47	2.37	45.9
6.5	9.66	3.55	68.6	19	6.32	2.32	44.9
7.0	9.37	3.44	66.5	20	6.19	2.27	43.9
7.5	9.10	3.34	64.6	25	5.58	2.05	39.6
8.0	8.87	3.25	63.0	30	5.09	1.87	36.2
8.5	8.66	3.18	61.5	40	4.36	1.60	31.0
9.0	8.46	3.11	60.1	50	3.84	1.41	27.2
9.5	8.28	3.04	58.8	80	2.93	1.08	20.8

Current outage schedules do not support reloads in less than 10 days.

Source of Data: SA-13-C00-1996-004

** The makeup flowrate listed is to compensate for boil off (not required flow to prevent boiling)

TABLE 2.2.3

Time to Boil Following a Loss of SDC During Midloop Operation with A Large or Small Cold Leg Opening

Prior to Core Reload (3990 MW Core)

Time after Reactor Shutdown (days)	Time to Boil (minutes)						Time after Reactor Shutdown (days)	Time to Boil (minutes)					
	Shutdown Cooling Heat Exchanger Inlet Temperature (F)							Shutdown Cooling Heat Exchanger Inlet Temperature (F)					
	100	110	120	130	135	140		100	110	120	130	135	140
1.0	12.3	11.1	10.0	8.9	8.4	7.8	10	28.8	26.1	23.5	20.9	19.6	18.3
2.0	15.0	13.6	12.2	10.9	10.2	9.5	11	29.8	27.1	24.4	21.7	20.3	19.0
3.0	17.4	15.8	14.2	12.6	11.8	11.1	12	30.8	28.0	25.2	22.4	21.0	19.6
3.5	18.5	16.8	15.1	13.5	12.6	11.8	13	31.8	28.9	26.0	23.1	21.7	20.2
4.0	19.6	17.8	16.0	14.2	13.4	12.5	14	32.7	29.7	26.8	23.8	22.3	20.8
4.5	20.6	18.7	16.9	15.0	14.1	13.1	15	33.6	30.5	27.5	24.4	22.9	21.4
5.0	21.6	19.6	17.7	15.7	14.7	13.7	16	34.5	31.3	28.2	25.1	23.5	21.9
5.5	22.5	20.5	18.4	16.4	15.4	14.3	17	35.3	32.1	28.9	25.7	24.1	22.5
6.0	23.4	21.2	19.1	17.0	15.9	14.9	18	36.2	32.9	29.6	26.3	24.7	23.0
6.5	24.2	22.0	19.8	17.6	16.5	15.4	19	37.0	33.6	30.3	26.9	25.2	23.5
7.0	25.0	22.7	20.4	18.2	17.0	15.9	20	37.8	34.4	30.9	27.5	25.8	24.1
7.5	25.7	23.3	21.0	18.7	17.5	16.3	25	41.9	38.1	34.3	30.5	28.6	26.7
8.0	26.4	24.0	21.6	19.2	18.0	16.8	30	45.9	41.7	37.6	33.4	31.3	29.2
8.5	27.0	24.5	22.1	19.6	18.4	17.2	40	53.6	48.7	43.9	39.0	36.6	34.1
9.0	27.6	25.1	22.6	20.1	18.8	17.6	50	60.9	55.4	49.8	44.3	41.5	38.8
9.5	28.2	25.7	23.1	20.5	19.2	18.0	80	79.7	72.5	65.2	58.0	54.4	50.7

Source of Data: SA-13-C00-1996-004

TABLE 2.2.4

Time to Boil Following a Loss of SDC During Midloop Operation with A Large or Small Cold Leg Opening

After Core Reload (3990 MW Core)

Time after Reactor Shutdown (days)	Time to Boil (minutes) Shutdown Cooling Heat Exchanger Inlet Temperature (F)						Time after Reactor Shutdown (days)	Time to Boil (minutes) Shutdown Cooling Heat Exchanger Inlet Temperature (F)					
	100	110	120	130	135	140		100	110	120	130	135	140
1.0	15.7	14.3	12.9	11.4	10.7	10.0	10	36.9	33.5	30.2	26.8	25.1	23.5
2.0	19.2	17.4	15.7	14.0	13.1	12.2	11	38.2	34.8	31.3	27.8	26.1	24.3
3.0	22.3	20.3	18.2	16.2	15.2	14.2	12	39.5	35.9	32.3	28.8	27.0	25.2
3.5	23.7	21.6	19.4	17.3	16.2	15.1	13	40.7	37.0	33.3	29.6	27.8	25.9
4.0	25.1	22.8	20.5	18.3	17.1	16.0	14	42.0	38.1	34.3	30.5	28.6	26.7
4.5	26.4	24.0	21.6	19.2	18.0	16.8	15	43.1	39.2	35.2	31.3	29.4	27.4
5.0	27.7	25.2	22.7	20.1	18.9	17.6	16	44.2	40.2	36.1	32.1	30.1	28.1
5.5	28.9	26.2	23.6	21.0	19.7	18.4	17	45.3	41.2	37.1	33.0	30.9	28.8
6.0	30.0	27.2	24.5	21.8	20.4	19.1	18	46.4	42.1	37.9	33.7	31.6	29.5
6.5	31.0	28.2	25.4	22.6	21.1	19.7	19	47.4	43.1	38.8	34.5	32.3	30.2
7.0	32.0	29.1	26.2	23.3	21.8	20.4	20	48.5	44.1	39.6	35.2	33.0	30.8
7.5	32.9	29.9	26.9	23.9	22.5	21.0	25	53.7	48.9	44.0	39.1	36.6	34.2
8.0	33.8	30.7	27.7	24.6	23.0	21.5	30	58.8	53.5	48.1	42.8	40.1	37.4
8.5	34.6	31.5	28.3	25.2	23.6	22.0	40	68.7	62.5	56.2	50.0	46.9	43.7
9.0	35.4	32.2	29.0	25.8	24.1	22.5	50	78.1	71.0	63.9	56.8	53.3	49.7
9.5	36.2	32.9	29.6	26.3	24.7	23.0	80	102.2	92.9	83.6	74.3	69.7	65.0

Current outage schedules do not support reloads in less than 10 days.

Source of Data: SA-13-C00-1996-004

TABLE 2.2.5

Time to Core Uncovery Following a Loss of SDC

During Midloop Operation with A Large or Small Cold Leg Opening

(3990 MW Core)

Notes: (1) ***Caution; No cold leg openings (i.e. RCP impeller work or shaft replacement) before 14 MW (Source TA-03-C09-2001-004)***

(2) Table values can be used to estimate time to core uncovery after boiling begins assuming no leg openings. RCP seal replacement with the RCP shaft on the stop seal is allowed as long as a blocking device is installed on the shaft in the event LSDC occurs. This condition prevents a large cold leg breach.

(3) Times do not include time to boil.

Decay Heat (MW)	Time to Core Uncovery After Boiling Starts (minutes)		Decay Heat (MW)	Time to Core Uncovery After Boiling Starts (minutes)
24.44	18		10.42	42
20.02	22		10.05	44
17.25	25		9.72	45
16.19	27		9.43	46
15.30	28		9.16	48
14.54	30		8.92	49
13.88	31		8.70	50
13.31	33		8.48	52
12.83	34		8.29	53
12.39	35		8.10	54
12.01	36		7.93	55
11.67	37		7.15	61
11.37	38		6.53	67
11.10	39		5.59	79
10.85	40		4.92	89
10.62	41		3.76	117

Source of Data: TA-13-C00-2001-006

2.3 RCS Drained to Reactor Vessel Flange with Reactor Head Removed

NOTES 2.3.0

Key Reactor Core Parameters Following a Loss of SDC

With The RCS Drained to the Reactor Vessel Flange

Reactor Head and Upper Guide Structure Removed

The key reactor core parameters following a loss of shutdown cooling (SDC) with the RCS drained to the 114' elevation and head or UGS off are for 550 EFPD at 100% RTP. These results are based in part on a computer analysis using the RETRAN code.

Decay heat power is based on Branch Technical Position 9-2 utilizing a 550 EFPD cycle length at 100% power. (3990 MWt)(Reference TA-13-C00-1999-009)

Time to boil is based on the time for the water in the vessel to reach 210 ^{0}F. This is a function of decay heat, mass of water in vessel and initial RCS bulk temperature (inlet Shutdown Cooling Heat Exchanger temperature). It is determined by subtracting the RCS bulk temperature from 210^{0}, and then dividing by the heatup rate.

Heatup rate is determined by dividing the decay heat by the specific heat capacity for water and by the mass of water available in the vessel at the start of the event.

Time to core uncovery is based on the time it takes the water above the top of the core to drop to the top of the core. Time to core uncovery is the time for the mass above the core to boil off and does not include the time associated with reaching the boiling temperature. Time to core uncovery is a function of decay heat, this volume of water, and the pressure over the water.

Makeup rate is the amount of flow required to reach the core to compensate for water loss through boil off. Note that this is the volume that must be delivered to the core in order to maintain constant inventory. Flow diverted out any cold leg opening must be accounted for (i.e. flowrate indications may not be actual flowrates reaching the core).

NOTE: VALUES NOT EXPLICITLY FOUND IN THE TABLES

SHALL BE DETERMINED VIA LINEAR INTERPOLATION

PERFORMED BY THE SHIFT TECHNICAL ADVISOR

TABLE 2.3.1

Key Reactor Core Parameters Following a Loss of SDC

With the RCS Drained to the Reactor Vessel Flange

Reactor Head and Upper Guide Structure Removed

Prior to Core Reload (3990 MW Core)

Time after Reactor Shutdown (days)	Decay Heat Load (MWth)	Heatup Rate (F/Min.)	Makeup Flowrate (gpm)**	Time after Reactor Shutdown (days)	Decay Heat Load (MWth)	Heatup Rate (F/Min.)	Makeup Flowrate (gpm)
1.0	24.44	5.67	173.5	10	10.42	2.42	74.0
2.0	20.02	4.64	142.1	11	10.05	2.33	71.4
3.0	17.25	4.00	122.5	12	9.72	2.26	69.0
3.5	16.19	3.76	114.9	13	9.43	2.19	67.0
4.0	15.30	3.55	108.6	14	9.16	2.13	65.0
4.5	14.54	3.37	103.2	15	8.92	2.07	63.3
5.0	13.88	3.22	98.5	16	8.70	2.02	61.8
5.5	13.31	3.09	94.5	17	8.48	1.97	60.2
6.0	12.83	2.98	91.1	18	8.29	1.92	58.9
6.5	12.39	2.87	88.0	19	8.10	1.88	57.5
7.0	12.01	2.79	85.3	20	7.93	1.84	56.3
7.5	11.67	2.71	82.9	25	7.15	1.66	50.8
8.0	11.37	2.64	80.7	30	6.53	1.51	46.4
8.5	11.10	2.58	78.8	40	5.59	1.30	39.7
9.0	10.85	2.52	77.0	50	4.92	1.14	34.9
9.5	10.62	2.46	75.4	80	3.76	0.87	26.7

Source of Data: SA-13-C00-1996-004

** The makeup flowrate listed is to compensate for boil off (not required flow to prevent boiling)

TABLE 2.3.2

Key Reactor Core Parameters Following a Loss of SDC With the RCS Drained to the Reactor Vessel Flange Reactor Head and Upper Guide Structure Removed

After Core Reload (3990 MW Core)

Time after Reactor Shutdown (days)	Decay Heat Load (MWth)	Heatup Rate (F/Min.)	Makeup Flowrate (gpm)**	Time after Reactor Shutdown (days)	Decay Heat Load (MWth)	Heatup Rate (F/Min.)	Makeup Flowrate (gpm)
1.0	19.06	4.42	135.3	10	8.13	1.89	57.7
2.0	15.62	3.62	110.9	11	7.84	1.82	55.7
3.0	13.46	3.12	95.5	12	7.58	1.76	53.8
3.5	12.63	2.93	89.7	13	7.36	1.71	52.2
4.0	11.93	2.77	84.7	14	7.14	1.66	50.7
4.5	11.34	2.63	80.5	15	6.96	1.61	49.4
5.0	10.83	2.51	76.9	16	6.79	1.57	48.2
5.5	10.38	2.41	73.7	17	6.61	1.53	47.0
6.0	10.01	2.32	71.1	18	6.47	1.50	45.9
6.5	9.66	2.24	68.6	19	6.32	1.47	44.9
7.0	9.37	2.17	66.5	20	6.19	1.44	43.9
7.5	9.10	2.11	64.6	25	5.58	1.29	39.6
8.0	8.87	2.06	63.0	30	5.09	1.18	36.2
8.5	8.66	2.01	61.5	40	4.36	1.01	31.0
9.0	8.46	1.96	60.1	50	3.84	0.89	27.2
9.5	8.28	1.92	58.8	80	2.93	0.68	20.8

Current outage schedules do not support reloads in less than 10 days.

** The makeup flowrate listed is to compensate for boil off (not required flow to prevent boiling)

Source of Data: SA-13-C00-1996-004

TABLE 2.3.3

Time to Boil Following a Loss of SDC with the RCS Drained to the Reactor Vessel Flange

Reactor Head and Upper Guide Structure Removed

Prior to Core Reload (3990 MW Core)

Time after Reactor Shutdown (days)	Time to Boil (minutes)						Time after Reactor Shutdown (days)	Time to Boil (minutes)					
	Shutdown Cooling Heat Exchanger Inlet Temperature (F)							Shutdown Cooling Heat Exchanger Inlet Temperature (F)					
	100	110	120	130	135	140		100	110	120	130	135	140
1.0	19.4	17.6	15.9	14.1	13.2	12.3	10	45.5	41.4	37.2	33.1	31.0	29.0
2.0	23.7	21.5	19.4	17.2	16.1	15.1	11	47.2	42.9	38.6	34.3	32.2	30.0
3.0	27.5	25.0	22.5	20.0	18.7	17.5	12	48.8	44.3	39.9	35.5	33.3	31.0
3.5	29.3	26.6	24.0	21.3	20.0	18.6	13	50.3	45.7	41.1	36.6	34.3	32.0
4.0	31.0	28.2	25.4	22.5	21.1	19.7	14	51.8	47.1	42.4	37.6	35.3	32.9
4.5	32.6	29.6	26.7	23.7	22.2	20.8	15	53.2	48.3	43.5	38.7	36.2	33.8
5.0	34.2	31.1	27.9	24.8	23.3	21.7	16	54.5	49.5	44.6	39.6	37.2	34.7
5.5	35.6	32.4	29.1	25.9	24.3	22.7	17	55.9	50.8	45.7	40.7	38.1	35.6
6.0	37.0	33.6	30.2	26.9	25.2	23.5	18	57.2	52.0	46.8	41.6	39.0	36.4
6.5	38.3	34.8	31.3	27.8	26.1	24.4	19	58.5	53.2	47.9	42.6	39.9	37.2
7.0	39.5	35.9	32.3	28.7	26.9	25.1	20	59.8	54.4	48.9	43.5	40.8	38.0
7.5	40.6	36.9	33.2	29.5	27.7	25.9	25	66.3	60.3	54.3	48.2	45.2	42.2
8.0	41.7	37.9	34.1	30.3	28.4	26.5	30	72.6	66.0	59.4	52.8	49.5	46.2
8.5	42.7	38.8	34.9	31.1	29.1	27.2	40	84.8	77.1	69.4	61.7	57.8	54.0
9.0	43.7	39.7	35.8	31.8	29.8	27.8	50	96.4	87.6	78.8	70.1	65.7	61.3
9.5	44.6	40.6	36.5	32.5	30.4	28.4	80	126.1	114.6	103.2	91.7	86.0	80.2

Source of Data: SA-13-C00-1996-004

TABLE 2.3.4

Time to Boil Following a Loss of SDC with the RCS Drained to the Reactor Vessel Flange

Reactor Head and Upper Guide Structure Removed

After Core Reload (3990 MW Core)

Time after Reactor Shutdown (days)	Time to Boil (minutes) — Shutdown Cooling Heat Exchanger Inlet Temperature (F)					
	100	110	120	130	135	140
1.0	24.9	22.6	20.3	18.1	17.0	15.8
2.0	30.4	27.6	24.8	22.1	20.7	19.3
3.0	35.2	32.0	28.8	25.6	24.0	22.4
3.5	37.5	34.1	30.7	27.3	25.6	23.9
4.0	39.7	36.1	32.5	28.9	27.1	25.3
4.5	41.8	38.0	34.2	30.4	28.5	26.6
5.0	43.8	39.8	35.8	31.9	29.9	27.9
5.5	45.7	41.5	37.4	33.2	31.1	29.1
6.0	47.4	43.1	38.8	34.5	32.3	30.2
6.5	49.1	44.6	40.1	35.7	33.5	31.2
7.0	50.6	46.0	41.4	36.8	34.5	32.2
7.5	52.1	47.4	42.6	37.9	35.5	33.1
8.0	53.5	48.6	43.7	38.9	36.5	34.0
8.5	54.8	49.8	44.8	39.8	37.3	34.8
9.0	56.0	50.9	45.8	40.7	38.2	35.7
9.5	57.2	52.0	46.8	41.6	39.0	36.4

Time after Reactor Shutdown (days)	Time to Boil (minutes) — Shutdown Cooling Heat Exchanger Inlet Temperature (F)					
	100	110	120	130	135	140
10	58.3	53.0	47.7	42.4	39.8	37.1
11	60.5	55.0	49.5	44.0	41.2	38.5
12	62.5	56.9	51.2	45.5	42.6	39.8
13	64.5	58.6	52.7	46.9	44.0	41.0
14	66.4	60.3	54.3	48.3	45.2	42.2
15	68.1	62.0	55.8	49.6	46.5	43.4
16	69.9	63.5	57.2	50.8	47.6	44.5
17	71.7	65.2	58.6	52.1	48.9	45.6
18	73.3	66.7	60.0	53.3	50.0	46.7
19	75.0	68.2	61.4	54.6	51.2	47.8
20	76.7	69.7	62.7	55.7	52.3	48.8
25	85.0	77.3	69.6	61.8	58.0	54.1
30	93.1	84.6	76.2	67.7	63.5	59.2
40	108.7	98.9	89.0	79.1	74.1	69.2
50	123.6	112.3	101.1	89.9	84.2	78.6
80	161.7	147.0	132.3	117.6	110.2	102.9

Current outage schedules do not support reloads in less than 10 days.

Source of Data: SA-13-C00-1996-004

TABLE 2.3.5

Time to Core Uncovery Following a Loss of SDC with the RCS Drained to the Reactor Vessel Flange, Reactor Head and Upper Guide Structure Removed (3990 MW Core)

Note: Times do not include time to boil.

Decay Heat (MW)	Time to Core Uncovery After Boiling Starts (minutes)		Decay Heat (MW)	Time to Core Uncovery After Boiling Starts (minutes)
24.44	146		10.42	343
20.02	178		10.05	356
17.25	207		9.72	368
16.19	221		9.43	380
15.30	234		9.16	391
14.54	246		8.92	401
13.88	258		8.70	411
13.31	269		8.48	422
12.83	279		8.29	432
12.39	289		8.10	442
12.01	298		7.93	451
11.67	307		7.15	501
11.37	315		6.53	548
11.10	322		5.59	641
10.85	330		4.92	728
10.62	337		3.76	953

Source of Data: TA-13-C00-2001-006

2.4 RCS Drained to Reactor Vessel Flange with Reactor Head Installed

NOTES 2.4.0

Key Reactor Core Parameters Following a Loss of SDC

With The RCS Drained to the Reactor Vessel Flange

Reactor Vessel Head On

The key reactor core parameters following a loss of shutdown cooling (SDC) with the RCS drained to the 114' elevation and head or UGS in place are for 550 EFPD at 100% RTP. These results are based in part on a computer analysis using the RETRAN code.

Decay heat power is based on Branch Technical Position 9-2 utilizing a 550 EFPD cycle length at 100% power. (3990 MWt)(Reference TA-13-C00-1999-009)

Time to boil is based on the time for the water in the vessel to reach 210 ^{0}F. This is a function of decay heat, mass of water in vessel and initial RCS bulk temperature (inlet Shutdown Cooling Heat Exchanger temperature). It is determined by subtracting the RCS bulk temperature from 210^{0}, and then dividing by the heatup rate.

Heatup rate is determined by dividing the decay heat by the specific heat capacity for water and by the mass of water available in the vessel at the start of the event.

Time to core uncovery is conservatively assumed to be the amount of time it takes to boil off the water volume below the bottom of the hot leg and above the top of the core. The time associated with the pressurization effects, steaming, entrainment, and surge line flooding are neglected. Thus, time to core uncovery is a function of decay heat, this volume of water and the pressure over the water. Time to core uncovery does not include the time associated with reaching the boiling temperature.

Makeup rate is the amount of flow required to reach the core to compensate for water loss through boil off. Note that this is the volume that must be delivered to the core in order to maintain constant inventory. Flow diverted out any cold leg opening must be accounted for (i.e. flowrate indications may not be actual flowrates reaching the core).

NOTE: VALUES NOT EXPLICITLY FOUND IN THE TABLES

SHALL BE DETERMINED VIA LINEAR INTERPOLATION

PERFORMED BY THE SHIFT TECHNICAL ADVISOR

TABLE 2.4.1

Key Reactor Core Parameters Following a Loss of SDC

With the RCS Drained to the Reactor Vessel Flange

Reactor Vessel Head On

Prior to Core Reload (3990 MW Core)

Time after Reactor Shutdown (days)	Decay Heat Load (MWth)	Heatup Rate (F/Min.)	Makeup Flowrate (gpm)**	Time after Reactor Shutdown (days)	Decay Heat Load (MWth)	Heatup Rate (F/Min.)	Makeup Flowrate (gpm)
1.0	24.44	5.67	173.5	10	10.42	2.42	74.0
2.0	20.02	4.64	142.1	11	10.05	2.33	71.4
3.0	17.25	4.00	122.5	12	9.72	2.26	69.0
3.5	16.19	3.76	114.9	13	9.43	2.19	67.0
4.0	15.30	3.55	108.6	14	9.16	2.13	65.0
4.5	14.54	3.37	103.2	15	8.92	2.07	63.3
5.0	13.88	3.22	98.5	16	8.70	2.02	61.8
5.5	13.31	3.09	94.5	17	8.48	1.97	60.2
6.0	12.83	2.98	91.1	18	8.29	1.92	58.9
6.5	12.39	2.87	88.0	19	8.10	1.88	57.5
7.0	12.01	2.79	85.3	20	7.93	1.84	56.3
7.5	11.67	2.71	82.9	25	7.15	1.66	50.8
8.0	11.37	2.64	80.7	30	6.53	1.51	46.4
8.5	11.10	2.58	78.8	40	5.59	1.30	39.7
9.0	10.85	2.52	77.0	50	4.92	1.14	34.9
9.5	10.62	2.46	75.4	80	3.76	0.87	26.7

Source of Data: SA-13-C00-1996-004

** The makeup flowrate listed is to compensate for boil off (not required flow to prevent boiling)

TABLE 2.4.2

Key Reactor Core Parameters Following a Loss of SDC

With the RCS Drained to the Reactor Vessel Flange

Reactor Vessel Head On

After Core Reload (3990 MW Core)

Time after Reactor Shutdown (days)	Decay Heat Load (MWth)	Heatup Rate (F/Min.)	Makeup Flowrate (gpm)**	Time after Reactor Shutdown (days)	Decay Heat Load (MWth)	Heatup Rate (F/Min.)	Makeup Flowrate (gpm)
1.0	19.06	4.42	135.3	10	8.13	1.89	57.7
2.0	15.62	3.62	110.9	11	7.84	1.82	55.7
3.0	13.46	3.12	95.5	12	7.58	1.76	53.8
3.5	12.63	2.93	89.7	13	7.36	1.71	52.2
4.0	11.93	2.77	84.7	14	7.14	1.66	50.7
4.5	11.34	2.63	80.5	15	6.96	1.61	49.4
5.0	10.83	2.51	76.9	16	6.79	1.57	48.2
5.5	10.38	2.41	73.7	17	6.61	1.53	47.0
6.0	10.01	2.32	71.1	18	6.47	1.50	45.9
6.5	9.66	2.24	68.6	19	6.32	1.47	44.9
7.0	9.37	2.17	66.5	20	6.19	1.44	43.9
7.5	9.10	2.11	64.6	25	5.58	1.29	39.6
8.0	8.87	2.06	63.0	30	5.09	1.18	36.2
8.5	8.66	2.01	61.5	40	4.36	1.01	31.0
9.0	8.46	1.96	60.1	50	3.84	0.89	27.2
9.5	8.28	1.92	58.8	80	2.93	0.68	20.8

Current outage schedules do not support reloads in less than 10 days.

Source of Data: SA-13-C00-1996-004

** The makeup flowrate listed is to compensate for boil off (not required flow to prevent boiling)

TABLE 2.4.3

Time to Boil Following a Loss of SDC with the RCS Drained to the Reactor Vessel Flange

Reactor Vessel Head On

Prior to Core Reload (3990 MW Core)

Time after Reactor Shutdown (days)	Time to Boil (minutes) Shutdown Cooling Heat Exchanger Inlet Temperature (F)						Time after Reactor Shutdown (days)	Time to Boil (minutes) Shutdown Cooling Heat Exchanger Inlet Temperature (F)					
	100	110	120	130	135	140		100	110	120	130	135	140
1.0	19.4	17.6	15.9	14.1	13.2	12.3	10	45.5	41.4	37.2	33.1	31.0	29.0
2.0	23.7	21.5	19.4	17.2	16.1	15.1	11	47.2	42.9	38.6	34.3	32.2	30.0
3.0	27.5	25.0	22.5	20.0	18.7	17.5	12	48.8	44.3	39.9	35.5	33.3	31.0
3.5	29.3	26.6	24.0	21.3	20.0	18.6	13	50.3	45.7	41.1	36.6	34.3	32.0
4.0	31.0	28.2	25.4	22.5	21.1	19.7	14	51.8	47.1	42.4	37.6	35.3	32.9
4.5	32.6	29.6	26.7	23.7	22.2	20.8	15	53.2	48.3	43.5	38.7	36.2	33.8
5.0	34.2	31.1	27.9	24.8	23.3	21.7	16	54.5	49.5	44.6	39.6	37.2	34.7
5.5	35.6	32.4	29.1	25.9	24.3	22.7	17	55.9	50.8	45.7	40.7	38.1	35.6
6.0	37.0	33.6	30.2	26.9	25.2	23.5	18	57.2	52.0	46.8	41.6	39.0	36.4
6.5	38.3	34.8	31.3	27.8	26.1	24.4	19	58.5	53.2	47.9	42.6	39.9	37.2
7.0	39.5	35.9	32.3	28.7	26.9	25.1	20	59.8	54.4	48.9	43.5	40.8	38.0
7.5	40.6	36.9	33.2	29.5	27.7	25.9	25	66.3	60.3	54.3	48.2	45.2	42.2
8.0	41.7	37.9	34.1	30.3	28.4	26.5	30	72.6	66.0	59.4	52.8	49.5	46.2
8.5	42.7	38.8	34.9	31.1	29.1	27.2	40	84.8	77.1	69.4	61.7	57.8	54.0
9.0	43.7	39.7	35.8	31.8	29.8	27.8	50	96.4	87.6	78.8	70.1	65.7	61.3
9.5	44.6	40.6	36.5	32.5	30.4	28.4	80	126.1	114.6	103.2	91.7	86.0	80.2

Source of Data: SA-13-C00-1996-004

TABLE 2.4.4

Time to Boil Following a Loss of SDC with the RCS Drained to the Reactor Vessel Flange

Reactor Vessel Head On

After Core Reload (3990 MW Core)

Time after Reactor Shutdown (days)	Time to Boil (minutes)						Time after Reactor Shutdown (days)	Time to Boil (minutes)					
	Shutdown Cooling Heat Exchanger Inlet Temperature (F)							Shutdown Cooling Heat Exchanger Inlet Temperature (F)					
	100	110	120	130	135	140		100	110	120	130	135	140
1.0	24.9	22.6	20.3	18.1	17.0	15.8	10	58.3	53.0	47.7	42.4	39.8	37.1
2.0	30.4	27.6	24.8	22.1	20.7	19.3	11	60.5	55.0	49.5	44.0	41.2	38.5
3.0	35.2	32.0	28.8	25.6	24.0	22.4	12	62.5	56.9	51.2	45.5	42.6	39.8
3.5	37.5	34.1	30.7	27.3	25.6	23.9	13	64.5	58.6	52.7	46.9	44.0	41.0
4.0	39.7	36.1	32.5	28.9	27.1	25.3	14	66.4	60.3	54.3	48.3	45.2	42.2
4.5	41.8	38.0	34.2	30.4	28.5	26.6	15	68.1	62.0	55.8	49.6	46.5	43.4
5.0	43.8	39.8	35.8	31.9	29.9	27.9	16	69.9	63.5	57.2	50.8	47.6	44.5
5.5	45.7	41.5	37.4	33.2	31.1	29.1	17	71.7	65.2	58.6	52.1	48.9	45.6
6.0	47.4	43.1	38.8	34.5	32.3	30.2	18	73.3	66.7	60.0	53.3	50.0	46.7
6.5	49.1	44.6	40.1	35.7	33.5	31.2	19	75.0	68.2	61.4	54.6	51.2	47.8
7.0	50.6	46.0	41.4	36.8	34.5	32.2	20	76.7	69.7	62.7	55.7	52.3	48.8
7.5	52.1	47.4	42.6	37.9	35.5	33.1	25	85.0	77.3	69.6	61.8	58.0	54.1
8.0	53.5	48.6	43.7	38.9	36.5	34.0	30	93.1	84.6	76.2	67.7	63.5	59.2
8.5	54.8	49.8	44.8	39.8	37.3	34.8	40	108.7	98.9	89.0	79.1	74.1	69.2
9.0	56.0	50.9	45.8	40.7	38.2	35.7	50	123.6	112.3	101.1	89.9	84.2	78.6
9.5	57.2	52.0	46.8	41.6	39.0	36.4	80	161.7	147.0	132.3	117.6	110.2	102.9

Current outage schedules do not support reloads in less than 10 days.

Source of Data: SA-13-C00-1996-004

TABLE 2.4.5

Time to Core Uncovery Following a Loss of SDC with the RCS Drained to the Reactor Vessel Flange, Reactor Vessel Head On (3990 MW Core)

Note: Times do not include time to boil and assume no cold leg openings.

Decay Heat (MW)	Time to Core Uncovery After Boiling Starts (minutes)		Decay Heat (MW)	Time to Core Uncovery After Boiling Starts (minutes)
24.44	18		10.42	42
20.02	22		10.05	44
17.25	25		9.72	45
16.19	27		9.43	46
15.30	28		9.16	48
14.54	30		8.92	49
13.88	31		8.70	50
13.31	33		8.48	52
12.83	34		8.29	53
12.39	35		8.10	54
12.01	36		7.93	55
11.67	37		7.15	61
11.37	38		6.53	67
11.10	39		5.59	79
10.85	40		4.92	89
10.62	41		3.76	117

Source of Data: TA-13-C00-2001-006

2.5 RCS Filled to 120’ (Vessel Full) and Reactor Vessel Head Installed

<u>NOTES 2.5.0</u>

Key Reactor Core Parameters Following a Loss of SDC With The RCS Filled to 120’ (Vessel Full) and Reactor Vessel Head On

The key reactor core parameters following a loss of shutdown cooling (SDC) with the RCS drained to the 120’ elevation (vessel full) and head on are for 550 EFPD at 100% RTP. These results are based in part on a computer analysis using the RETRAN code.

Decay heat power is based on Branch Technical Position 9-2 utilizing a 550 EFPD cycle length at 100% power. (3990 MWt)(Reference TA-13-C00-1999-009)

Time to boil is based on the time for the water in the vessel to reach 210 ^{0}F. This is a function of decay heat, mass of water in vessel and initial RCS bulk temperature (inlet Shutdown Cooling Heat Exchanger temperature). It is determined by subtracting the RCS bulk temperature from 210^0, and then dividing by the heatup rate.

Heatup rate is determined by dividing the decay heat by the specific heat capacity for water and by the mass of water available in the vessel at the start of the event.

Time to core uncovery is based on the time it takes the water above the top of the core to drop to the top of the core. Total time to uncover is the summation of Time to Boil plus the time for the mass above the core to boil off. This also considers the pressurization effects, steaming, entrainment and surge line flooding.

Time to core uncovery is conservatively assumed to be the amount of time it takes to boil off the water volume below the bottom of the hot leg and above the top of the core. The time associated with the pressurization effects, steaming, entrainment, and surge line flooding are neglected. Thus, time to core uncovery is a function of decay heat, this volume of water and the pressure over the water. Time to core uncovery does not include the time associated with reaching the boiling temperature.

Makeup rate is the amount of flow required to reach the core to compensate for water loss through boil off. Note that this is the volume that must be delivered to the core in order to maintain constant inventory. Flow diverted out any cold leg opening must be accounted for (i.e. flowrate indications may not be actual flowrates reaching the core).

NOTE: VALUES NOT EXPLICITLY FOUND IN THE TABLES SHALL BE DETERMINED VIA LINEAR INTERPOLATION PERFORMED BY THE SHIFT TECHNICAL ADVISOR

TABLE 2.5.1

Key Reactor Core Parameters Following a Loss of SDC

With the RCS Filled to 120' (Vessel Full) and Reactor Vessel Head On

Prior to Core Reload (3990 MW Core)

Time after Reactor Shutdown (days)	Decay Heat Load (MWth)	Heatup Rate (F/Min.)	Makeup Flowrate (gpm)**	Time after Reactor Shutdown (days)	Decay Heat Load (MWth)	Heatup Rate (F/Min.)	Makeup Flowrate (gpm)
1.0	24.44	4.99	173.5	10	10.42	2.13	74.0
2.0	20.02	4.08	142.1	11	10.05	2.05	71.4
3.0	17.25	3.52	122.5	12	9.72	1.98	69.0
3.5	16.19	3.30	114.9	13	9.43	1.92	67.0
4.0	15.30	3.12	108.6	14	9.16	1.87	65.0
4.5	14.54	2.97	103.2	15	8.92	1.82	63.3
5.0	13.88	2.83	98.5	16	8.70	1.77	61.8
5.5	13.31	2.72	94.5	17	8.48	1.73	60.2
6.0	12.83	2.62	91.1	18	8.29	1.69	58.9
6.5	12.39	2.53	88.0	19	8.10	1.65	57.5
7.0	12.01	2.45	85.3	20	7.93	1.62	56.3
7.5	11.67	2.38	82.9	25	7.15	1.46	50.8
8.0	11.37	2.32	80.7	30	6.53	1.33	46.4
8.5	11.10	2.26	78.8	40	5.59	1.14	39.7
9.0	10.85	2.21	77.0	50	4.92	1.00	34.9
9.5	10.62	2.17	75.4	80	3.76	0.77	26.7

Source of Data: SA-13-C00-1996-004

** The makeup flowrate listed is to compensate for boil off (not required flow to prevent boiling)

TABLE 2.5.2

Key Reactor Core Parameters Following a Loss of SDC

With the RCS Filled to 120' (Vessel Full) and Reactor Vessel Head On

After Core Reload (3990 MW Core)

Time after Reactor Shutdown (days)	Decay Heat Load (MWth)	Heatup Rate (F/Min.)	Makeup Flowrate (gpm)**	Time after Reactor Shutdown (days)	Decay Heat Load (MWth)	Heatup Rate (F/Min.)	Makeup Flowrate (gpm)
1.0	19.06	3.89	135.3	10	8.13	1.66	57.7
2.0	15.62	3.19	110.9	11	7.84	1.60	55.7
3.0	13.46	2.74	95.5	12	7.58	1.55	53.8
3.5	12.63	2.58	89.7	13	7.36	1.50	52.2
4.0	11.93	2.43	84.7	14	7.14	1.46	50.7
4.5	11.34	2.31	80.5	15	6.96	1.42	49.4
5.0	10.83	2.21	76.9	16	6.79	1.38	48.2
5.5	10.38	2.12	73.7	17	6.61	1.35	47.0
6.0	10.01	2.04	71.1	18	6.47	1.32	45.9
6.5	9.66	1.97	68.6	19	6.32	1.29	44.9
7.0	9.37	1.91	66.5	20	6.19	1.26	43.9
7.5	9.10	1.86	64.6	25	5.58	1.14	39.6
8.0	8.87	1.81	63.0	30	5.09	1.04	36.2
8.5	8.66	1.77	61.5	40	4.36	0.89	31.0
9.0	8.46	1.73	60.1	50	3.84	0.78	27.2
9.5	8.28	1.69	58.8	80	2.93	0.60	20.8

Current outage schedules do not support reloads in less than 10 days.

Source of Data: SA-13-C00-1996-004

** The makeup flowrate listed is to compensate for boil off (not required flow to prevent boiling)

TABLE 2.5.3

Time to Boil Following a Loss of SDC with the RCS Filled to 120' (Vessel Full) and Reactor Vessel Head On

Prior to Core Reload (3990 MW Core)

Time after Reactor Shutdown (days)	Time to Boil (minutes)						Time after Reactor Shutdown (days)	Time to Boil (minutes)					
	Shutdown Cooling Heat Exchanger Inlet Temperature (F)							Shutdown Cooling Heat Exchanger Inlet Temperature (F)					
	100	110	120	130	135	140		100	110	120	130	135	140
1.0	22.1	20.1	18.1	16.0	15.0	14.0	10	51.7	47.0	42.3	37.6	35.3	32.9
2.0	26.9	24.5	22.0	19.6	18.4	17.1	11	53.7	48.8	43.9	39.0	36.6	34.1
3.0	31.3	28.4	25.6	22.7	21.3	19.9	12	55.5	50.4	45.4	40.3	37.8	35.3
3.5	33.3	30.3	27.2	24.2	22.7	21.2	13	57.2	52.0	46.8	41.6	39.0	36.4
4.0	35.2	32.0	28.8	25.6	24.0	22.4	14	58.9	53.5	48.2	42.8	40.1	37.5
4.5	37.1	33.7	30.3	27.0	25.3	23.6	15	60.5	55.0	49.5	44.0	41.2	38.5
5.0	38.8	35.3	31.8	28.3	26.5	24.7	16	62.0	56.3	50.7	45.1	42.3	39.4
5.5	40.5	36.8	33.1	29.5	27.6	25.8	17	63.6	57.8	52.0	46.2	43.4	40.5
6.0	42.0	38.2	34.4	30.6	28.7	26.7	18	65.0	59.1	53.2	47.3	44.3	41.4
6.5	43.5	39.6	35.6	31.7	29.7	27.7	19	66.6	60.5	54.5	48.4	45.4	42.4
7.0	44.9	40.8	36.7	32.7	30.6	28.6	20	68.0	61.8	55.6	49.5	46.4	43.3
7.5	46.2	42.0	37.8	33.6	31.5	29.4	25	75.4	68.6	61.7	54.8	51.4	48.0
8.0	47.4	43.1	38.8	34.5	32.3	30.2	30	82.6	75.1	67.6	60.1	56.3	52.5
8.5	48.6	44.2	39.7	35.3	33.1	30.9	40	96.5	87.7	78.9	70.2	65.8	61.4
9.0	49.7	45.2	40.7	36.1	33.9	31.6	50	109.6	99.6	89.7	79.7	74.7	69.7
9.5	50.8	46.2	41.5	36.9	34.6	32.3	80	143.4	130.4	117.3	104.3	97.8	91.3

Source of Data: SA-13-C00-1996-004

TABLE 2.5.4

Time to Boil Following a Loss of SDC with the RCS Filled to 120’ (Vessel Full) and Reactor Vessel Head On

After Core Reload (3990 MW Core)

Time after Reactor Shutdown (days)	Time to Boil (minutes) — Shutdown Cooling Heat Exchanger Inlet Temperature (F)						Time after Reactor Shutdown (days)	Time to Boil (minutes) — Shutdown Cooling Heat Exchanger Inlet Temperature (F)					
	100	110	120	130	135	140		100	110	120	130	135	140
1.0	28.3	25.7	23.1	20.6	19.3	18.0	10	66.3	60.3	54.3	48.3	45.2	42.2
2.0	34.5	31.4	28.3	25.1	23.5	22.0	11	68.8	62.5	56.3	50.0	46.9	43.8
3.0	40.1	36.4	32.8	29.1	27.3	25.5	12	71.1	64.7	58.2	51.7	48.5	45.3
3.5	42.7	38.8	34.9	31.1	29.1	27.2	13	73.3	66.6	60.0	53.3	50.0	46.7
4.0	45.2	41.1	37.0	32.9	30.8	28.8	14	75.5	68.6	61.7	54.9	51.5	48.0
4.5	47.5	43.2	38.9	34.6	32.4	30.3	15	77.5	70.5	63.4	56.4	52.8	49.3
5.0	49.8	45.3	40.8	36.2	34.0	31.7	16	79.5	72.2	65.0	57.8	54.2	50.6
5.5	51.9	47.2	42.5	37.8	35.4	33.1	17	81.5	74.1	66.7	59.3	55.6	51.9
6.0	53.9	49.0	44.1	39.2	36.7	34.3	18	83.4	75.8	68.2	60.6	56.9	53.1
6.5	55.8	50.7	45.7	40.6	38.0	35.5	19	85.3	77.6	69.8	62.1	58.2	54.3
7.0	57.6	52.3	47.1	41.9	39.2	36.6	20	87.2	79.3	71.3	63.4	59.4	55.5
7.5	59.2	53.9	48.5	43.1	40.4	37.7	25	96.7	87.9	79.1	70.3	65.9	61.5
8.0	60.8	55.3	49.7	44.2	41.5	38.7	30	105.9	96.2	86.6	77.0	72.2	67.4
8.5	62.3	56.6	51.0	45.3	42.5	39.6	40	123.7	112.4	101.2	89.9	84.3	78.7
9.0	63.7	57.9	52.1	46.3	43.4	40.5	50	140.5	127.7	115.0	102.2	95.8	89.4
9.5	65.1	59.2	53.3	47.3	44.4	41.4	80	183.9	167.1	150.4	133.7	125.4	117.0

Current outage schedules do not support reloads in less than 10 days.

Source of Data: SA-13-C00-1996-004

TABLE 2.5.5

Time to Core Uncovery Following a Loss of SDC with the RCS Filled to 120' (Vessel Full) and Reactor Vessel Head On

(3990 MW Core)

Note: Times do not include time to boil and assume no cold leg openings.

Decay Heat (MW)	Time to Core Uncovery After Boiling Starts (minutes)		Decay Heat (MW)	Time to Core Uncovery After Boiling Starts (minutes)
24.44	18		10.42	42
20.02	22		10.05	44
17.25	25		9.72	45
16.19	27		9.43	46
15.30	28		9.16	48
14.54	30		8.92	49
13.88	31		8.70	50
13.31	33		8.48	52
12.83	34		8.29	53
12.39	35		8.10	54
12.01	36		7.93	55
11.67	37		7.15	61
11.37	38		6.53	67
11.10	39		5.59	79
10.85	40		4.92	89
10.62	41		3.76	117

Source of Data: TA-13-C00-2001-006

2.6 RCS Filled to Refueling Level Operation

NOTES 2.6.0

Key Reactor Core Parameters Following a Loss of SDC

With The RCS Filled to Refueling Level

The key reactor core parameters following a loss of shutdown cooling (SDC) with the RCS filled to the refueling level are for 550 EFPD at 100% RTP. These results are based in part on a computer analysis using the RETRAN code.

Decay heat power is based on Branch Technical Position 9-2 utilizing a 550 EFPD cycle length at 100% power. (3990 MWt)(Reference TA-13-C00-1999-009)

Time to boil is based on the time for the water in the vessel to reach 210 ^{0}F. This is a function of decay heat, mass of water in vessel and initial RCS bulk temperature (inlet Shutdown Cooling Heat Exchanger temperature). It is determined by subtracting the RCS bulk temperature from 210^0, and then dividing by the heatup rate. The only value of RCS bulk temperature provided is 135 degrees F, the maximum allowed temperature for Mode 6.

Heatup rate is determined by dividing the decay heat by the specific heat capacity for water and by the mass of water available in the vessel at the start of the event.

Time to core uncovery is based on the time it takes the water above the top of the core to drop to the top of the core. Time to core uncovery is the time for the mass above the core to boil off and does not include the time associated with reaching boiling temperature. Time to core uncovery is a function of decay heat, this volume of water and the pressure over the water.

Makeup rate is the amount of flow required to reach the core to compensate for water loss through boil off. Note that this is the volume that must be delivered to the core in order to maintain constant inventory. Flow diverted out any cold leg opening must be accounted for (i.e. flowrate indications may not be actual flowrates reaching the core).

NOTE: VALUES NOT EXPLICITLY FOUND IN THE TABLES

SHALL BE DETERMINED VIA LINEAR INTERPOLATION

PERFORMED BY THE SHIFT TECHNICAL ADVISOR

TABLE 2.6.1

Time to Boil and Heatup Rates Following a Loss of SDC

With the RCS Filled to the Refueling Level

Prior to Core Reload (3990 MW Core)

Time after Reactor Shutdown (days)	Decay Heat Load (MWth)	Heatup Rate (F/Min.)	Time to Boil (hours)	Time after Reactor Shutdown (days)	Decay Heat Load (MWth)	Heatup Rate (F/Min.)	Time to Boil (hours)
1.0	24.44	0.42	3.0	10	10.42	0.18	7.0
2.0	20.02	0.34	3.6	11	10.05	0.17	7.2
3.0	17.25	0.30	4.2	12	9.72	0.17	7.5
3.5	16.19	0.28	4.5	13	9.43	0.16	7.7
4.0	15.30	0.26	4.7	14	9.16	0.16	7.9
4.5	14.54	0.25	5.0	15	8.92	0.15	8.1
5.0	13.88	0.24	5.2	16	8.70	0.15	8.4
5.5	13.31	0.23	5.5	17	8.48	0.15	8.6
6.0	12.83	0.22	5.7	18	8.29	0.14	8.8
6.5	12.39	0.21	5.9	19	8.10	0.14	9.0
7.0	12.01	0.21	6.1	20	7.93	0.14	9.2
7.5	11.67	0.20	6.2	25	7.15	0.12	10.2
8.0	11.37	0.20	6.4	30	6.53	0.11	11.1
8.5	11.10	0.19	6.5	40	5.59	0.10	13.0
9.0	10.85	0.19	6.7	50	4.92	0.08	14.8
9.5	10.62	0.18	6.8	80	3.76	0.06	19.3

Source of Data: SA-13-C00-1996-004

TABLE 2.6.2

Time to Boil and Heatup Rates Following a Loss of SDC With the RCS Filled to the Refueling Level

After Core Reload (3990 MW Core)

Time after Reactor Shutdown (days)	Decay Heat Load (MWth)	Heatup Rate (F/Min.)	Time to Boil (hours)	Time after Reactor Shutdown (days)	Decay Heat Load (MWth)	Heatup Rate (F/Min.)	Time to Boil (hours)
1.0	19.06	0.33	3.8	10	8.13	0.14	8.9
2.0	15.62	0.27	4.7	11	7.84	0.13	9.3
3.0	13.46	0.23	5.4	12	7.58	0.13	9.6
3.5	12.63	0.22	5.8	13	7.36	0.13	9.9
4.0	11.93	0.21	6.1	14	7.14	0.12	10.2
4.5	11.34	0.20	6.4	15	6.96	0.12	10.4
5.0	10.83	0.19	6.7	16	6.79	0.12	10.7
5.5	10.38	0.18	7.0	17	6.61	0.11	11.0
6.0	10.01	0.17	7.3	18	6.47	0.11	11.2
6.5	9.66	0.17	7.5	19	6.32	0.11	11.5
7.0	9.37	0.16	7.8	20	6.19	0.11	11.7
7.5	9.10	0.16	8.0	25	5.58	0.10	13.0
8.0	8.87	0.15	8.2	30	5.09	0.09	14.3
8.5	8.66	0.15	8.4	40	4.36	0.07	16.7
9.0	8.46	0.15	8.6	50	3.84	0.07	18.9
9.5	8.28	0.14	8.8	80	2.93	0.05	24.8

Current outage schedules do not support reloads in less than 10 days.

Source of Data: SA-13-C00-1996-004

TABLE 2.6.3

Time to Core Uncovery Following a Loss of SDC

RCS at Refueling Level *(3990 MW Core)*

Note: Times do not include time to boil and assume no cold leg openings.

Decay Heat (MW)	Time to Core Uncovery After Boiling Starts (hours)		Decay Heat (MW)	Time to Core Uncovery After Boiling Starts (hours)
24.44	34.9		10.42	81.8
20.02	42.6		10.05	84.8
17.25	49.4		9.72	87.7
16.19	52.6		9.43	90.4
15.30	55.7		9.16	93.0
14.54	58.6		8.92	95.5
13.88	61.4		8.70	98.0
13.31	64.0		8.48	100.5
12.83	66.4		8.29	102.8
12.39	68.8		8.10	105.2
12.01	71.0		7.93	107.5
11.67	73.0		7.15	119.2
11.37	75.0		6.53	130.5
11.10	76.8		5.59	152.5
10.85	78.5		4.92	173.2
10.62	80.2		3.76	226.7

Source of Data: TA-13-C00-2001-006

2.7 Spent Fuel Pool Operation

NOTES 2.7.0

Spent Fuel Pool Parameters for a Loss of SDC

Spent Fuel Pool (SFP) parameters will be determined on a cycle to cycle basis due to changing inventories in the SFP.

Information concerning the existing decay heat load currently in the SFP can be found in the unit specific "As-Left Decay Heat Projection" analysis maintained by Reactor Engineering. The decay heat loads for the reactor core (for periods of time during or after core offload) are included in SAOD sections 2.2 through 2.6.

Once the decay heat has been determined, the Heatup Rate (HUR) in ^{0}F/hr can be calculated from:

$$HUR = \frac{DH \cdot 1000\frac{kW}{MW_t} \cdot 3412\frac{(BTU)/(hr)}{kW}}{2,114,850 lbm \cdot C_p}$$

Time to Boil (TTB) in hours can be calculated from:

$$TTB = \frac{210 - T_{initial}}{HUR}$$

where:

DH: total SFP decay heat, current decay heat load + additional off-loaded fuel, in MWt

2,114,850 lbm is the assumed mass of water in the SFP, this is less than value based on these references (References: (1) NRC letter from M.B. Fields to G.R. Overbeck, March 2, 2000, addresses minimum SFP capacity of 320,000 gallons; (2) Updated Final Safety Analysis Report, Table 9.1-2, Maximum SFP temperature of 167 ^{0}F)

Cp: 1.0 BTU/lbm ^{0}F

210 ^{0}F assumed boiling point of SFP water

$T_{initial}$: SFP temperature at time of loss of cooling in ^{0}F

2.8 Minimum Time to Reduced Flow SDC Operations

NOTES 2.8.0

Minimum Time to Reduced Flow SDC Operations

Following Reactor Shutdown to Maintain RCS Temperature

at or Below 135 Deg. F

Table 2.8.1 presents the EW cooling water temperature and post-shutdown time criteria prior to reducing SDC system total flow rate to 3780 gpm (indicated) to support reduced inventory conditions while providing sufficient cooling capacity to maintain the RCS temperature at or below 135 ^{0}F. Specifically, this table correlates the time after shutdown where the core decay heat is equal to or less than the cooling capacity of the SDC system at the corresponding EW cooling water temperature and with the RCS temperature at 135 ^{0}F.

The data presented in Table 2.8.1 is predicated on stable RCS temperature prior to reducing SDC flow rate. Upon reduction of SDC flow rate and prior to reducing RCS water level, Operations shall monitor RCS temperature to verify that sufficient cooling capacity exists to maintain RCS temperature at or below 135 ^{0}F. Engineering recommends that RCS temperature be monitored for a period of not less than one hour (reference letter 448-00525).

Data presented in Table 2.8.1 is based on the SDC performance analyses contained in PVNGS calculation 13-MC-SI-231 for a rated thermal power of 3990 MW. EW temperature instrument uncertainty included in Table 2.8.1 is based on total loop uncertainty for instruments 1,2,3JEWNTI083/84 as indicated on the ERFDADS display. Verification of actual EW temperature by alternative means using appropriate M&TE is considered acceptable.

Notes 2.8.0 and Table 2.8.1 are maintained by Design Mechanical Engineering, NSSS. Questions concerning this information should be directed to the Design Mechanical NSSS section leader.

TABLE 2.8.1

Minimum Time to Reduced Flow SDC Operations Following Reactor Shutdown To Maintain RCS Temperature at or Below 135 Deg. F

Note: Refer to Note 2.8.0 for information regarding the use of this table

EW Inlet Temperature (Actual - ^{0}F)	EW Inlet Temperature (Indicated - ^{0}F) (1,2,3JEWN083/84)	Decay Heat / SDC capacity (MW)	Time After Shutdown (Hrs)
92	87	15.5	91
91	86	15.9	87
90	85	16.3	82
89	84	16.6	79
88	83	17.0	75
87	82	17.4	71

Source of Data: 13-MC-SI-231, Revision 4

2.9 RCS Forced Flow and RWT Temperature Requirements

NOTES 2.9.0

RCS Forced Flow and RWT Temperature Requirements

To Maintain Subcooled Conditions

Following a Loss of Shutdown Cooling

Table 2.9.1 presents the Refueling Water Tank fluid temperature and time requirements following reactor shutdown to be satisfied prior to reduced inventory conditions. These criteria must be satisfied prior to reducing RCS water level to ensure that sufficient cooling is available to maintain the core covered and subcooled by HPSI forced cooling in the event of a loss of shutdown cooling. The RWT temperature requirements in Table 2.9.1 ensure that the total heat dissipated by a single train of HPSI injection is equal to or greater than the corresponding core decay heat at the specified time after shutdown.

The data presented in Table 2.9.1 is based on the forced flow cooling analysis contained in PVNGS calculation 13-MC-SI-231 for a rated thermal power of 3990 MW. RWT temperature instrument uncertainty included in Table 2.9.1 is based on total loop uncertainty for instruments 1,2,3CHNTI200 as indicated on the ERFDADS display. Operations will establish the need to circulate the RWT volume to ensure that the tank is thermally well mixed.

Notes 2.9.0 and Table 2.9.1 are maintained by Design Mechanical Engineering, NSSS. Questions concerning this information should be directed to the Design Mechanical NSSS section leader.

TABLE 2.9.1

RCS Forced Flow and RWT Temperature Requirements

To Maintain Subcooled Conditions

Following a Loss of Shutdown Cooling

Note: Refer to Note 2.9.0 for information regarding the use of this table

RWT Temperature (Actual - ^{0}F)	RWT Temperature (Indicated - ^{0}F)	Decay Heat (MW)	Time After Shutdown (Hrs)
74	72	18.0	65
76	74	17.7	68
78	76	17.4	71
80	78	17.1	75
82	80	16.8	77
84	82	16.5	80
86	84	16.2	83
88	86	15.9	86
90	88	15.6	90

Source of Data: 13-MC-SI-231, Revision 4

3.0 IMPACT REVIEW

The SAOD provides information to the operations staff and shift technical advisors should a loss of shutdown cooling occur, but does not affect any procedures they use.

This document satisfies the design control requirements but is not a QR analysis package. This document is a collection of data from QR analysis. Changes to these referenced analysis require the performance of the impact review and other analysis package requirements (i.e. AD documentation, etc.).

The SAOD revision will not effect any outside organization or procedure. The potential impact of revising this SWMS manual was determined by talking to personnel from: Operations, Outage Management, OPS standards, Design Engineering, System Engineering, and NFM. No SABD exists for the SAOD. No other design or configuration documents were identified as impacted.

4.0 REFERENCES

1) TA-13-C00-1999-009, “Outage Decay Heats”, Revision 3, 2/25/2003.

2 TA-13-C00-2001-005, “Loss of Shutdown Cooling Analyses for RCS Drain Operations and Nozzle Dam Installation”, Revision 1, 3/28/2003

3 TA-13-C00-2001-006, “SAOD Input Data of Estimated Times to Vaporize RCS Inventory Above the Reactor Core”, Revision 1, 5/7/2003.

4 SA-13-C00-1996-004, “Stretch Power SAOD”, Revision 4, 8/5/2003.

5 13-MC-SI-231, “Calculation of Minimum Time to Reduced Flow Shutdown Cooling Operation”, Revision 4, 4/06/2005

6 Deleted.

7 Letter 162-09970-KCP dated March 5, 2002. *(Note; this letter was supplemented by the letter in reference 12 below).*

8 TA-03-C09-2001-004, “Loss of Shutdown Cooling Analysis for RCP Removal During U3C9 SNOW Outage”, Revision 1, 2/20/2001.

9 NRC letter from M.B. Fields to G.R. Overbeck, March 2, 2000.

10 UFSAR revision 13 dated June 2005.

11 Letter 448-00525-MAB/JAB dated August 17, 2001.

12 Letter 162-10794-CAH/DAM dated February 25, 2004 and CRDR 2686238

Design Review Checklist

Design Review Area *(Refer to the instructions for each specific area that needs to be considered by the review)*	**Results of Review**	
	YES	**N/A[1]**
1. Were the inputs correctly selected and properly documented?	**X**	
2. Are assumptions necessary to perform the analysis adequately described and reasonable? Where necessary, are they identified for subsequent re-verification (contingencies)?		**X**
3. Are the appropriate quality and quality assurance requirements specified?	**X**	
4. Are the applicable standards, acceptance criteria, and regulatory requirements properly identified? Are their requirements met?		**X**
5. Have applicable operating experience, conditions, issues, and plant configuration been considered?	**X**	
6. Are the systems, structures, and components credited/used/considered/specified in the analysis allowed/suitable for the required application?	**X**	
7. Was an appropriate design method used? Were the input and assumptions correctly incorporated in the design process?	**X**	
8. Is the output reasonable compared to design inputs? Are the conclusions appropriately drawn?	**X**	
9. Have the design interface requirements been satisfied? Are the impacts on other design documents properly identified? Have change mechanisms been initiated?	**X**	
10. If the analysis results need verification by further testing, are the criteria for verification that the design requirements have been satisfactorily accomplished identified?		**X**
11. Are the requirements for record preparation, review, approval, retention etc. adequately specified?	**X**	

Comments/Explanations:

For Questions 2, 4, and 10: This document is a collection of data from several QR analyses. No new assumptions, acceptance criteria, or verification requirements are established by this document.

Review Performed By: A. N. Kilic	**Date:11/30/05**

1.N/A (Not Applicable) - If marked, an explanation shall be provided in the "Comments/Explanation" Box.

Design Verification Comment Sheet

Comment Number	Reviewer's Comment	Response Required?	Preparer's Response	Response Accepted?
1	For consistency, state the time when 14 MW is reached on page 4.	No	Extrapolating from TA-13-C00-1999-009, 14 MW is reached at 117.8 hours after shutdown. 118 hours listed in SAOD.	Yes
2	Page 33, date for referenced NRC letter is wrong (typo, number should be 2 not 3)	No	Corrected typo	Yes
3	Other misc editorial comments as discussed.	No	Incorporated	Yes
4				
5				
6				

REVIEWER: A. N. Kilic **DATE: 11/30/05**

SCOPE OF REVIEW AND VERIFICATION: This document is a collection of data from several QR analyses. Thus, the review consisted of verification that the information is correctly and completely extracted from the source documents and translated into SAOD for applicable unit.

TYPE OF VERIFICATION (check all that apply):

Design Review (attach Design Review Checklist)	X
Alternate Calculations (attach Design Review Checklist and alternate calculations)	
Qualification Testing (attach Design Review Checklist and test results)	
Other (specify and attach supporting documentation and Design Review Checklist)	

VERIFICATION NOTES: (Attach relevant information)

(Larry's Copy of) PVNGS 2012 Senior Reactor Operator NRC Exam

21.

This Exam Level	SRO
Appears on:	SRO EXAM 2012 Tier 3
K/A #	2.2.11
Importance Rating:	3.3

In accordance with 81DP-0DC17 (Temporary Modification Control), which ONE of the following installations require a Temporary Modification?

A. Alternate power supplied to NHN-M04 during a refueling outage.

B. Domestic service flush line aligned to NCN-P01A while it is under clearance.

C. Discharge pressure gauge on a LPSI pump while performing a surveillance test.

D. Jumpers installed in an PPS channel while performing a troubleshooting work order.

Answer: A

Associated KA:
L57327 Identify those plant changes that are NOT considered Temporary Modifcations.

100866 Active Question Bank 2004

Reference Id: Q1363
Difficulty: 3.00
Time to complete: 2
10CFR Category: 55.43 (3) Facility licensee procedures required to obtain authority for design and operating changes in the facility
Cognitive Level: Memory
Question Source: PV Bank Not Modified
Comment:
Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 81DP-0DC17 (Temporary Modification Control)

K&A: Equipment Control Knowledge of the process for controlling temporary changes.

Learning Objective: Identify those plant changes that are NOT considered Temporary Modification.

Justification:

A. Correct: Per Appendix D of 81DP-0DC17, Temporary power installations connecting permanent plant equipment either bus, motor or valve, if the temporary power comes from one in-plant bus to another in-plant bus.
B. Incorrect: Flushing a system while under clearance is similar to air assisted draining and does not require a Tmod..
C. Incorrect: LPSI ST pressure gauge has a permanently installed plant adapter for the ST and does not require a Tmod.
D. Incorrect: This is controlled by the work control process and a Tmod is not required.

(Larry's Copy of) PVNGS 2012 Senior Reactor Operator NRC Exam

22.

This Exam Level	SRO
Appears on:	SRO EXAM 2007SRO EXAM 2012 Tier 3
K/A #	2.2.18
Importance Rating:	3.9

Given the following conditions:

- Unit 1 is in a Midloop condition
- Maintenance requests permission to re-lug ESFAS jumper leads

Prior to this Work Order being released to the field, who (by title) is responsible to verify the proper RCS perturbation code?

A. Releasing Organization and Outage Coordinator

B. Releasing Organization and Operations Shift Manager

C. Outage Coordinator and Midloop Operations Coordinator

D. Midloop Operations Coordinator and Operations Shift Manager

Answer: D

Associated KA:
30222 process for managing maintenance activities while shutdown

Reference Id: Q10380
Difficulty: 4.00
Time to complete: 3
10CFR Category: 55.43 (3) Facility licensee procedures required to obtain authority for design and operating changes in the facility
Cognitive Level: Memory
Question Source: PV Bank Not Modified
Comment:
Proposed reference to be provided to applicant during examination: NONE

Technical Reference: 40OP-9ZZ16 (RCS Drain Ops) & 40OP-9ZZ20 (Reduced Inventory Ops)

K&A: Knowledge of the process for managing maintenance activities during shutdown operations.

Learning Objective: 30222 process of managing maintenance activities while shutdown

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Justification:

A. Incorrect: The releasing organization and outage coordinator control clearances and other activities (making them seem correct), but not work orders.
B. Incorrect:The releasing organization and outage coordinator control clearances and other activities (making them seem correct), but not work orders.
C. Incorrect:The releasing organization and outage coordinator control clearances and other activities (making them seem correct), but not work orders.
D. Correct: By procedure 40DP-9ZZ30 Appendix A, only these 2 control this activity.

(Larry's Copy of) PVNGS 2012 Senior Reactor Operator NRC Exam

23.

This Exam Level	SRO
Appears on:	SRO EXAM 2008 SRO EXAM 2012 Tier 3
K/A #	2.3.11
Importance Rating:	4.3

Given the following plant conditions:

- A large break LOCA has occurred.
- Due to emergency conditions a gaseous radioactive release from Containment must be performed to relieve pressure in the containment and bring the plant to a safer condition.

Who may authorize this release without a release permit?

A. Shift Manager

B. Operations Department Leader

C. Radiation Protection Supervisor

D. Radiological Services Department Leader

Answer: A

Reference Id:	Q832	
Difficulty:	3.00	
Time to complete:	4	
10CFR Category:	CFR 55.43 (4)	55.43 (4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.
Cognitive Level:	Memory	
Question Source:	PV Bank Not Modified	
Comment:		

Proposed reference to be provided to applicant during examination: None

Technical Reference:

1. LOIT lesson plan L57256 (describe whose authority is needed to exceed requirements and what reporting is necessary)
2. 74RM-9EF20, GR release permits and offsite dose assessments

K&A: Ability to control radiation releases

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Justification:

A is Correct - Only the Shift Manager or CRS is permitted to approve emergency release's to stabilize the plant during EOP performance

B is Incorrect - review and Approval of releases that exceed >/= to 80% of limits.

C is Incorrect - Approves releases that are < 40% of the limit

D is Incorrect - Review and Approval of releases that are > 40% but < 80% of limit

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24.

This Exam Level:	SRO
Appears on:	SRO EXAM 2012 Tier 3
K/A #:	2.4.38
Importance Rating:	4.4

Which ONE of the following is the lowest (least severe) Emergency Action Level that **REQUIRES** the EC to direct accountability, per the Emergency Plan?

A. Unusual Event.

B. Alert.

C. Site Area Emergency.

D. General Emergency.

Answer: C

Reference Id:	Q8347	
Difficulty:	2.00	
Time to complete:	2	
10CFR Category:	CFR 55.43 (5)	55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Cognitive Level:	Memory	
Question Source:	PV Bank Not Modified	
Comment:		

Proposed reference to be provided to applicant during examination: NONE

Technical Reference: EP-0901 (ERO Position Checklists)

K&A: Ability to take actions called for in the facility emergency plan, including supporting or acting as emergency coordinator if required.

Learning Objective: L59732 Given an emergency event in progress, Determine if assembly and/or accountability are required.

Justification:

A. Incorrect: NUE requires the use of the ERO position checklist and may be chosen since it is the lowest of the EAL Classifications.
B. Incorrect: Per step 12 App L of EP-0900,can be performed at Alert if DESIRED.
C. Correct: Per step 6 of App L of EP-0900, Assembly/Accountability is only REQUIRED at SAE or higher.
D. Incorrect: Assembly/Accountability is REQUIRED at GE, but it is not the lowest EAL Classification.

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25.

This Exam Level:	SRO
Appears on:	SRO EXAM 2012 Tier 3
K/A #:	2.4.40
Importance Rating:	4.5

Given the following conditions:

- Unit 2 has declared a SITE AREA EMERGENCY.
- The Unit 2 Shift Manager has been relieved as Emergency Coordinator (EC).

Which ONE of the following positions must approve a PVNGS worker receiving Potassium Iodide (KI)?

A. Unit 2 Shift Manager.

B. Emergency Coordinator.

C. Radiological Protection Monitor.

D. Emergency Operations Director.

Answer: B

Reference Id: Q43919
Difficulty: 3.00
Time to complete: 2
10CFR Category: CFR 55.43 (4) 55.43 (4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.
Cognitive Level: Memory
Question Source: PV Bank Not Modified
Comment:
Proposed reference to be provided to applicant during examination: NONE

Technical Reference: EP-0905 (Protective Actions)

K&A: Knowledge of SRO responsibilities in emergency plan implementation.

Learning Objective: L92080 Identify the Emergency Coordinator's responsibilities associated with Emergency Exposure.

Justification:
A. Incorrect – If the Unit 2 SM was the EC this would be correct. SM also will direct plant operations during the event. EC controls AO movements.
B. Correct – Per step 2.5 of EP-0905, the EC-STSC and EC-TSC are responsible for approving KI use by onsite emergency workers.
C. Incorrect – the RPM is used to consult on such matters, but does not approve the dose.
D. Incorrect – EOD will make many decisions during the event. Candidate may confuse EC with the EOD.