Final Submittal
(Blue Paper)

COMBINED RO/SRO WRITTEN EXAM

WITH KAS, ANSWERS,

AND ANALYSIS

SURRY JAN./FEB. 2006 EXAM

0500380/2006301 AND 05000281/2006301

JANUARY 23 - FEBRUARY 3, 2006
FEBRUARY 8, 2006 (WRITTEN)

2/9/06

Final LXRTest RO/SRO Written

1. 001AA2.04 002/1/2/ROD WITHDRAWAL/C/A4.3/N/SR06301/S/FJE

Unit 1 INITIAL conditions are as follows:

- Reactor power 95%
- Rod control in AUTOMATIC
- NO MCR alarms
- D Bank Control Rods are at 220 steps

During a continuous rod withdrawal, D bank rods will physically step to __(1)__ and reactor power will increase to a value __(2)__.

A.✓ (1) 230 steps

(2) Less than 100%

B. (1) 230 steps

(2) Greater than 100%

C. (1) 227 steps

(2) Less than 100%

D. (1) 227 steps

(2) Greater than 100%

K/A
001 Continuous Rod Withdrawal
AA2.04 Ability to determine and interpret the following as they apply to the Continuous Rod Withdrawal: Reactor power and its trend (CFR 43.5)

Notes

Developed a new question based on K/A.

Answer Option Analysis

A. Correct. During a continuous rod withdrawal rods will step out to 230 steps (maximum) and reactor power will increase. It will not increase to greater than 100% due to the amount of reactivity added by the rods given their initial position.

B. Incorrect. During a continuous rod withdrawal rods will step out to 230 steps (maximum) and reactor power will increase. It will not increase to greater than 100% due to the amount of reactivity added by the rods given their initial position.

C. Incorrect. (1) Plausible, as current upper rod limit is 227 steps. (2) is correct.

D. Incorrect. Both (1) and (2) are incorrect. See B, C.

References

ND-95-1-LP-5, Rev. 9, Rod Control Cluster Assembly (RCCA) Malfunctions

RFA accept 12/21/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	A B D C B C A A D D	Scramble Range: A - D

Tier:	1	Group:	2
Key Word(s):	ROD WITHDRAWAL	Cog Level:	C/A4.3
Source:	N	Exam:	SR06301
Test:	S	Author / Reviewer:	FJE

2. 002G2.1.32 001/2/2/RCS LIMITS/C/A3.8/N/SR06301/S/FJE

Unit 1 status is as follows:

- Unit 1 is in HOT SHUTDOWN and is shutting down to COLD SHUTDOWN due to an inoperable Number 1 Emergency Diesel Generator.
- All plant systems are operating normally.
- There is a bubble in the PRZR.
- All RCPs are running.
- Charging and letdown are in service.
- Rapid RCS Cooldown in progress per 0-AP-23.01, RAPID RCS COOLDOWN
- RCS cooldown is via condenser steam dump valves.
- 0-OSP-RC-001, RCS and PRZR Heatup/Cooldown Verification has been initiated.

You are reviewing Unit 1 cooldown data (data sheet attached) before you relieve the Unit 1 SRO. Which ONE of the following describes BOTH the condition that was exceeded AND the appropriate action to take?

Reference Provided.

A. 0-AP-23.01 RCS cooldown rate administrative limit.
REDUCE RCS cooldown rate by adjusting 1-MS-PC-1464B, STEAM HDR PRESS CNTRL.

B. 0-AP-23.01 RCS cooldown rate administrative limit.
Verify that all available PRZR heaters are energized.

C✓ Unexpected drop in PRZR Surge Line or Liquid temperature due to PRZR insurge.
Verify that all available PRZR heaters are energized.

D. Unexpected drop in PRZR Surge Line or Liquid temperature due to PRZR insurge.
REDUCE RCS cooldown rate by adjusting 1-MS-PC-1464B, STEAM HDR PRESS CNTRL.

K/A
002 Reactor Coolant
G2.1.32 Ability to explain and apply all system limits and precautions (CFR 43.2)

Notes

Attachments to Exam

Cooldown data (Attachments 2 and 3 data, or facsimile that includes RCS loop Tcold, PRZR water temp, and PRZR surge line temp.) inserted after question page.
0-AP-23.01 (all) included in reference package.

Answer Option Analysis

A. Incorrect. Cooldown rate not exceeded. Data provided (T0406A) show a steady 60 °F RCS cooldown rate. 0-AP-23.01 (Attachment 3 item 10) administrative RCS cooldown rate limit above 350 °F is 75 °F/hr. Plausible because admin limit below 350 °F and admin limit per 1-GOP-2.5, Unit Cooldown, HSD to 351 °F, is 50 °F/hr. Action is correct per 0-AP-23.01, but cooldown rate not exceeded.

B. Incorrect. C/D rate not exceeded (see A.). Incorrect action IF C/D rate were exceeded.

C. Correct. 0-AP-23.01 Attachment 4 provides direction for monitoring PRZR liquid and surge line temperatures and RCS temperature. Att. 4, Step 3 states that if PRZR surge line or liquid temperatures unexpectedly drop by 50 °F due to a PRZR insurge, then immediately perform the following steps... Step 3.c is to verify that all available PRZR heaters are energized. 1-GOP-2.5, Precaution and Limitation 4.6 states that "When PRZR surge line temperature decreases faster than PRZR liquid temperature, a PRZR insurge is occurring." Data provided show that, after 16:15, PRZR surge line temperature has gone down faster than przr water temp and that surge line temperature has gone down by more than 50 °F.

D. Incorrect. Limit has been violated, but action is incorrect. 0-AP-23.01 Attachment 3 Steps a directs operators to "Immediately stop the insurge to the PRZR." Step b directs operators to "Stabilize PRZR level." Reducing the cooldown rate will make the insurge that is occuring worse.

References

0-AP-23.01, Rev. 5, Rapid RCS Cooldown
1-GOP-2.4, Rev. 26, UNIT COOLDOWN, HSD TO 351°F, Precaution and Limitation 4.9; Step 5.3.11.
1-GOP-2.5, Rev. 22, UNIT COOLDOWN, 351 °F TO LESS THAN 205°F, Precaution and Limitation 4.9

Allow use of reference. RFA accept 12/21/05

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C D D D D A A D A C Scramble Range: A - D

Tier:	2	Group:	2
Key Word(s):	RCS LIMITS	Cog Level:	C/A3.8
Source:	N	Exam:	SR06301
Test:	S	Author / Reviewer:	FJE

3. 003AG2.4.6 001/1/2/DROPPED ROD/C/A4.0/N/SR06301/S/FJE

Unit 1 conditions are as follows:

- Unit 1 was operating at 40% power.
- Rod K-6 dropped to the bottom of the core at 01:45
- At 14:45 the crew completed all required actions of 0-AP-1.01, Control Rod Misalignment, and is ready to realign rod K-6.
- Current (at 14:45) reactor power is 40%.

During realignment of rod K-6, reactor power should be held constant at __ % and rod K-6 should not exceed a MAXIMUM of ____ steps per hour.

A. 40% 4

B.✓ 40% 5

C. 50% 4

D. 50% 5

K/A
003 Dropped Control Rod
AG2.4.6 Knowledge symptom based EOP mitigation strategies (CFR 43.5)

Notes

Answer Option Analysis

0-AP-1.01 Note before step 14 states "Realignment SHALL be performed with Reactor power held constant at less than or equal to 75%."

0-AP-1.01 Caution before step 18 states "The affected withdrawal rate during realignment is limited to 2/P (P= fraction of Core Power where 100% power is equal to 1.0) steps per hour (if not a whole number, round down to the whole number) if affected rod remains misaligned for more than 12 hours or the duration of misalignment can NOT be determined."

Tech Spec 3.12.B.5 states "The allowable QUADRANT POWER TILT is 2.0% and is only applicable while operating at THERMAL POWER > 50%".

A. Incorrect. Correct power, incorrect maximum rod withdrawal rate. 2/.4 = 5 steps per hour. Plausible if applicant recognizes constant power limitation, but believes allowable rate is based on a 50% limit (2/.5 = 4 steps/hr).

B. Correct. During realignment, reactor power should be held constant. Current power is 40%. Increasing power by 10% would not be considered holding power constant. Rod withdrawal rate is 2/.4 = 5 steps per hour.

C. Incorrect. Incorrect power, incorrect rod withdrawal rate. Plausible if applicant believes that power is limited by Tech Spec 3.12, Control Rod Assemblies and Power Distribution Limits (QPTR less than 2% unless below 50% power).

D. Incorrect. Incorrect power, correct rod withdrawal rate. Plausible if applicant believes that power is limited by Tech Spec 3.12, Control Rod Assemblies and Power Distribution Limits (QPTR less than 2% unless below 50% power).

References

0-AP-1.01, Rev. 16, Control Rod Misalignment
Licensee examination bank question no. 208
ND-93.3-LP-3, Rev. 15, Rod Control System

No changes.

RFA accept 12/21/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	B C D B D D D D D B	Scramble Range: A - D
Tier:	1		Group:	2	
Key Word(s):	DROPPED ROD		Cog Level:	C/A4.0	
Source:	N		Exam:	SR06301	
Test:	S		Author / Reviewer:	FJE	

4. 004A2.12 003/2/1/CVCS SI SIGNAL/C/A4.3/N/SR06301/S/FJE

Unit 1 is at full power with all systems in their normal alignments when an inadvertent B-Train Safety Injection signal occurs.

Which ONE of the following correctly describes the DIRECT response of the Charging and Letdown system to the B-Train SI signal AND the procedure the crew will use to restore the Letdown system?

Normal letdown flow isolates when ________ close(s) after receiving an SI signal and charging isolates when _______ closes after receiving an SI signal. The crew will restore the Letdown system per ______.

A. 1-CH-HCV-1200 A, B, and C
1-CH-HCV-1310A
1-ES-1.1, SI TERMINIATION

B. 1-CH-HCV-1200 A, B, and C
1-CH-MOV-1289B
1-GOP-1.5, UNIT STARTUP, 2% REACTOR POWER TO MAX ALLOWABLE POWER

C. 1-CH-TV-1204B
1-CH-HCV-1310A
1-GOP-1.5, UNIT STARTUP, 2% REACTOR POWER TO MAX ALLOWABLE POWER

D.✓ 1-CH-TV-1204B
1-CH-MOV-1289B
1-ES-1.1, SI TERMINIATION

K/A
004 Chemical and Volume Control
A2.12 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: CIAA, SIAS
(CFR 43.5)

Notes

Need facility to verify correct answer and distractors (i.e. 1-CH-MOV-1289B and 1-CH-TV-1204B receive B-train SI signals).

Answer Option Analysis

A. Incorrect. HCV-1200A/B/C close on an "A" train SI signal only. HCV-1310A is a fail open valve that does not receive any automatic signals. Plausible because HCV-1200A/B/C close on a full SI. Correct procedure (see D).

B. Incorrect. See A. regarding HCV-1200A/B/C. Correct regarding charging valve 1-CH-MOV-1289B. Incorrect procedure (see D) 1-GOP-1.5 does not direct restoration of charging or letdown.

C. Incorrect. Correct regarding letdown valve 1-CH-TV-1204B. See A. regarding HCV-1310A. Incorrect procedure (see D) 1-GOP-1.5 does not direct restoration of charging or letdown.

D. Correct. Correct system response and correct procedure. A B-train SI signal will cause a reactor trip. The crew will transition from 1-E-0 to 1-ES-0.1 and restore charging per Step 3 and letdown per Step 5.b).

References

ND-88.3-LP-2, Rev. 12, Charging and Letdown
1-ES-1.1, SI Termination, Rev. 32
1-GOP-1.5, Rev. 42, UNIT STARTUP, 2% REACTOR POWER TO MAX ALLOWABLE POWER
Following a spurious/inadvertent SI the team will not transition to ES-0.1, they will restore charging in E-0 and letdown in ES-1.1. Deleted charging from question to allow single procedure answer (consistent with other distractors) for letdown restoration. See ES-1.1 step 14 (letdown) and E-0 step 35 (charging).

Added numbers to blanks for clarification.

Updated references (procedure revision and deleted ES-0.1 and added ES-1.1)

Minor format change. RFA accept 12/21/05.

Clarification of normal letdown flow to make the question technically accurate. As letdown flow continues through a relief valve when the 1204s are closed.

RFA approved 1/12/06

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D D D A B A D B B A Scramble Range: A - D

Tier:	2	Group:	1
Key Word(s):	CVCS SI SIGNAL	Cog Level:	C/A4.3
Source:	N	Exam:	SR06301
Test:	S	Author / Reviewer:	FJE

5. 011A2.10 002/2/2/PRESSURIZER LEVEL/C/A2.8/M/SR06301/S/FJE

Unit 1 conditions are as follows:
- Unit 1 is at 100% power.
- All control systems are in AUTO.
- Pressurizer level transmitter LT-459 is selected for control (as the upper channel).
- The reference leg of LT-459 has just developed a slow leak.

Which ONE of the following, in accordance with procedure hierarchy, describes the INITIAL instrument/plant response AND the procedure that contains the operator actions necessary to deselect the failed channel and restore PRZR level?

	LI-**459** PZR LVL INDICATION	LI-**461** PZR LVL INDICATION
A.	INCREASES	INCREASES
	1C-E8, PRZR LO LVL HTRS OFF & LETDOWN ISOL	
B.	DECREASES	DECREASES
	1C-E8, PRZR LO LVL HTRS OFF & LETDOWN ISOL	
C✓	INCREASES	DECREASES
	0-AP-53.00, LOSS OF VITAL INSTRUMENTATION/CONTROLS	
D.	DECREASES	INCREASES
	0-AP-53.00, LOSS OF VITAL INSTRUMENTATION/CONTROLS	

K/A
011 Pressurizer Level Control System
A2.10 Ability to (a) predict the impacts of the following malfunctions or operations on the PZR LCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure of PZR level instrument - high (CFR 43.5)

Notes

Modified from facility examination bank question PLC0009

Answer Option Analysis

A leak in the reference leg of LT-459 will reduce the d/P (between actual pzr level and the reference leg) seen by the controlling channel and LI-459 level indication will go UP. Actual pzr level (as indicated on the non-affected channels) will go DOWN as charging flow is automatically reduced by the pressurizer level control system.

A. Incorrect. Although correct regarding LI-459, actual pzr level and LI-461 will go DOWN as charging flow is reduced. Incorrect ARP. The actions in 1C-C8 address charging or letdown valve failures and a secondary load rejection.

B. Incorrect. Although correct regarding LI-461, the controlling channel indication will increase as a result of the leak on the reference leg due to a reduced d/P between actual level and the reference leg. Incorrect ARP (see A).

C. Correct. AP-53.00 step 10 directs operators to deselect the failed PRZR level channel and step 10 also directs operators to restore PRZR level.

D. Incorrect. See A., B. Correct Procedure (see C.)

References

Facility examination bank question PLC0009
ND-93.3-LP-7, Rev. 7, Pressurizer Level Control System
0-AP-53.00, LOSS OF VITAL INSTRUMENTATION/CONTROLS, Rev. 4
1C-C8, Rev. 0., PRZR HI LVL HTRS ON

Revised response 'C' and 'D' to list AP-53.00 as the correct procedure to use to restore pressurizer level and deselect the failed pressurizer level channel. This was done since AP-53.00 is the controlling procedure for instrumentation failures. As a result, we also removed the words "Annunciator Response" from the stem of the question.

Updated procedure references to include AP-53.00 and remove ARP 1C-E8.

Editorial Change.

Changed distractors A and B (second half). Added " in accordance with procedure hierarchy" to stem. RFA accept 12/21/05

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C B A C D A B A B C Scramble Range: A - D

Tier:	2	Group:	2
Key Word(s):	PRESSURIZER LEVEL	Cog Level:	C/A2.8
Source:	M	Exam:	SR06301
Test:	S	Author / Reviewer:	FJE

6. 011EA2.01 002/1/1/LOCA/C/A4.7/N/SR06301/S/FJE

Unit 1 conditions are as follows:

- A reactor trip occurred 14 minutes ago due to a LOCA.
- ALL RCPs are SECURED.
- AC Emergency Bus 1H is DEENERGIZED.
- Containment pressure is 20.5 psia and slowly going DOWN.
- RCS pressure is 350 psig and slowly going DOWN.
- 1-SI-MOV-1860B (LHSI Pump B Suction from Sump) is CLOSED and DEENERGIZED and can NOT be opened.
- 1B HHSI pump (1-CH-P-1B) and 1B LHSI pump (1-SI-P-1B) are RUNNING.
- Full range RVLIS indicates 100%.
- RWST level is 62% and going DOWN.
- Reactor Coolant System Subcooling Margin Monitors (both trains) indicate - 30 °F (negative 30 °F).
- 1-ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, is in progress <u>up to</u> Step 19.

Which ONE of the following is the NEXT correct action per 1-ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION?

References provided.

A. Check RVLIS indication per Step 24.a)

B. Stop 1B LHSI pump 1-SI-P-1B and put pump in auto per Step 21.

C✓ Consult with TSC to determine if SI valves should be throttled per Step 19.b) RNO b)1.

D. Check CTMT pressure less than 14 psia per Step 20.a).

K/A
011 Large Break LOCA
EA2.01 Ability to determine or interpret the following as they apply to a Large Break LOCA: Actions to be taken, based on RCS temperature and pressure – saturated and superheated (CFR 43.5)

Notes

Need to verify how RCS subcooling is determined per Step 16.b) (using SMM or other?) and how SMM is displayed on ICCS.

Attachments

1-ECA-1.1, Rev. 23, LOSS OF EMERGENCY COOLANT RECIRCULATION, pages 12-17 (steps 16-24) and Attachment 2.

Answer Option Analysis

A. Incorrect. Plausible if appliant fails to recognize full range RVLIS is greater than 63%.

B. Incorrect. Plausible if appliant fails to recognize minimum SI flow is greater than 150 gpm (e.g. misreads 14 minutes as 104 minutes).

C. Correct. With full range RVLIS greater than 63%, RCS subcooling NOT greater than 135 F, and minimum SI flow required greater than 150 gpm, the next step to perform is to consult with TSC.

D. Incorrect. Plausible if applicant fails to recognize that negative 174 F subcooling is not greater than 135 F.

References

1-ECA-1.1, Rev. 23, Loss of Emergency Coolant Recirculation
Provided noun name for 1-SI-MOV-1860B.

Adjusted RWST level in the stem of the question for added realism based on LOCA time and components running.

Adjusted ICCM reading for sub-cooling, per NRC request.

Added words "Reference provided", per NRC request.

Updated procedure revision for ECA-1.1 and corrected attachment page number so that steps 16-24 are included.

RFA accept 12/21/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	C B B A C A B D D D	Scramble Range: A - D	
Tier:	1			Group:	1	
Key Word(s):	LOCA			Cog Level:	C/A4.7	
Source:	N			Exam:	SR06301	
Test:	S			Author / Reviewer:	FJE	

7. 012G2.4.31 002/2/1/TEMP ALARM/C/A3.4/N/SR06301/S/FJE

Unit 1 is at 100% reactor power with NO MCR annunciators LIT.

Which ONE of the following correctly describes which unit 1 alarm(s) will remain LIT, assuming NO operator action, if the "A" Loop Narrow Range Tcold fails HIGH?

A.✓ 1H-A3, HI-LO T AVG LOOP 1A
1H-B4, TAVG LOOP DEVIATION

B. 1H-A4, T AVG > <T REF DEVIATION
1H-A2, DELTA T DEVIATION LOOP 1A >< LOOP 1B

C. 1H-A3, HI-LO T AVG LOOP 1A
1H-A4, T AVG > <T REF DEVIATION

D. 1H-B4, TAVG LOOP DEVIATION
1E-C7, RX TRIP CH-1 OVPWR DELTA T LOOP 1A

K/A
012 Reactor Protection
G2.4.31 Knowledge of annunciators alarms and indications, and use of the response instructions (CFR 41.10 / 45.3)

Notes

Modified SR04301 question from Tcold fails LOW to Tcold fails HIGH.

Answer Option Analysis

A. Correct. The "A" loop Tcold RTD failing high will cause loop "A" Tavg to increase to the alarm setpoint and a Tave loop deviation per ND-93.3-LP-2.

B. Incorrect. Per ND-93.3-LP-2, the input to 1-H-A4 comes from the output of the Tave Median Signal Selector (MSS). The failed loop Tave signal is filtered out by the Tave MSS. Second part (1H-A2) is correct.

C. Incorrect. Part 1 correct, part 2 incorrect. See A, B.

D. Incorrect. Part 1 correct, part 2 incorrect. The "A" loop Tcold RTD failing high will cause loop "A" delta T to decrease, increasing the margin to the loop "A" setpoint even though loop "A" Tavg goes up. Plausible because these alarms will be lit for the "A" loop cold leg RTD failing LOW.

References

ND-93.3-LP-2, Rev. 10, Delta T/Tavg Instrumentation Systems

Modified stem to read "No MCR annunciators lit" vice "All MCR annunciators dark" for clarification and SPS terminology.

Deleted word "only" from answer 'A' and 'B' as these would not be the only annunciators lit. The following annunciators would also be lit: 1H-A2, 1H-C2, and 1H-B4.

Modified all distractors (and answer). RFA accept 12/21/05

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A C A A B B B C B A Scramble Range: A - D

Tier:	2	Group:	1
Key Word(s):	TEMP ALARM	Cog Level:	C/A3.4
Source:	N	Exam:	SR06301
Test:	S	Author / Reviewer:	FJE

8. 015AG2.1.33 003/1/1/RCP/C/A4.0/N/SR06301/S/FJE

Unit 1 plant conditions are as follows:

- Reactor power is 20% and the unit is ON-LINE.
- 1-GOP-1.5, UNIT STARTUP, 2% REACTOR POWER TO MAX ALLOWABLE POWER, is in progress following a refueling outage.
- PDTT Level is going UP.

RCP parameters are as follows:

	RCP 1A	RCP 1B	RCP 1C
Pump Started:	2 days ago	20 hours ago	2 days ago
VAPOR SEAL TK HI LVL alarm:	Not Lit	Lit	Not Lit
Seal Leakoff Flow:	6.5 gpm and STABLE	1.0 gpm and SLOWLY increasing	3.0 gpm and STABLE

- ALL other RCP parameters are below Action Level values and STABLE.

Which ONE of the following statements describes the MINIMUM actions necessary in order to comply with plant Technical Specifications AND 1-AP-9.00, RCP ABNORMAL OPERATION?

Reference provided.

A. Trip unit 1 reactor, perform 1-E-0, REACTOR TRIP OR SAFETY INJECTION, and trip ONLY RCP 1**A** within 5 minutes.

B. Trip unit 1 reactor, perform 1-E-0, REACTOR TRIP OR SAFETY INJECTION, and trip RCP 1**A** AND 1**B** within 5 minutes.

C.✓ Take unit 1 off line per 1-GOP-2.2, UNIT SHUTDOWN, LESS THAN 30% TO HSD, insert all control banks, then trip ONLY RCP 1**A** within 8 hours.

D. Take unit 1 off line per 1-GOP-2.2, UNIT SHUTDOWN, LESS THAN 30% TO HSD, insert all control banks, then trip RCPs 1**A** AND 1**B** within 8 hours.

K/A
015 RCP Malfunction
AG2.1.33 Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications (CFR 43.2 / 43.3)

Attachments

1-AP-9.00, RCP Abnormal Conditions, Rev. 21

Notes

Question originally written as open reference, RCP B seal leakoff @ 0.65 gpm. Changed to closed reference due to 026AG2.4.4

Answer Option Analysis

A. Incorrect. Although RCP A seal leakoff is high, caution before 1-AP-9.00 step 34 allows for RCP operation up to 8 hours as long as 1) seal leakoff is less than 8 gpm, and 2) Attachment 2 parameters are less than Action Level and NOT continuously going up. Additionally, Tech Spec does not require a reactor trip. 1-AP-9.00 allows removing unit from service per GOP-2 series. Plausible if applicant fails to recognize pump operation is allowed for 8 hours with high seal leakoff flow and/or fails to recognize that Tech Specs do not require a reactor trip for loss of an RCP below P-7.

B. Incorrect. See A. Additionally, although seal leakoff on RCP B is low, it is caused by high #2 seal leakage (PDTT level increasing, Standpipe Level Hi alarm) AND it is within the normal operating range of Attachment 1 of 1-AP-9.00. Plausible if applicant fails to recognize that RCP B seal leakoff is acceptable (greater than 0.8 gpm).

C. Correct. Caution before step 34 directs stopping RCP within 8 hours if total seal leakoff flow is greater than 6 gpm and Attachment 2 parameters are stable. Tech Spec 3.1.A.1.b requires that the plant be shutdown, with all control banks inserted, if a loss of one or more RCPs occurs below 10% power (P-7).

D. See B.

References
1-AP-9.00, Rev. 21, RCP Abnormal Condition
Surry Technical Specification 3.1.A.1.b
ND-88.1-LP-6, Rev. 17, Reactor Coolant Pumps

Added data concerning 1-RC-P-1C, as the plant does not operate with the reactor critical and only two RCPs running.

Changed "both" to "all" for the addition of 1-RC-P-1C data.

Changed verbage for 1-RC-P-1B from "going UP" to "SLOWLY increasing" to provide rate information.

Changed "dark" to "not lit".

Adjusted procedure revision number.

Would like a reference due to difficulty of question without reference (AP-9.00)

Allow use of reference. RFA accept 12/21/05.

Changed reactor power to 20% and added that the unit is ON-LINE. This allows for proper flow path through the procedure.

RFA accept 1/12/06.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C B C C A A B D B A Scramble Range: A - D

Tier:	1	Group:	1
Key Word(s):	RCP	Cog Level:	C/A4.0
Source:	N	Exam:	SR06301
Test:	S	Author / Reviewer:	FJE

9. 015G2.1.33 002/2/2/INTERMEDIATE RANGE/MEM4.0/N/SR06301/S/FJE

Unit 1 conditions are as follows:

- Reactor startup in progress per 1-OP-RX-006, WITHDRAWAL OF THE CONTROL BANKS TO CRITICAL CONDITIONS
- 1G-E3, NIS INT RNG CH2 LOSS OF COMP VOLT alarm was just received
- PRIOR to the failure N35 AND N36 BOTH indicated 3E-10 amps and stable

Which ONE of the following actions will result in compliance with Technical Specifications?

A.✓ RAISE power ABOVE 10% within 24 hours

B. Restore the inoperable channel to OPERABLE status before increasing power above P-6

C. Restore the inoperable channel to OPERABLE status within 48 hours

D. LOWER power BELOW P-6 within 48 hours

K/A
015 Nuclear Instrumentation
G2.1.33 Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications (CFR 43.2 / 43.3)

Notes

Need to verify current revision of references

Answer Option Analysis

A. Correct. Unit 1 is above P-6 (1.0E-10 amps). With one failed IRNI, TS Table 3.7-1, Action 3.b. provides two options: 1) decrease power below P-6 within 24 hours, OR, 2) increase power above 10% within 24 hours.

B. Incorrect. This is the action for one failed IRNI below P-6. Table 3.7-1, Action 3.a.

C. Incorrect. This is one possible action for a failed SRNI below P-6. Table 3.7-1, Action 4.a.

D. Incorrect. Power must be reduced below P-6 within 24 hours to meet the TS requirement.

References

Surrey Tech Spec 3.7, Instrumentation Systems, Table 3.7-1
1G-E3, Rev. 0, NIS INT RNG CH2 LOSS OF COMP VOLT
1-AP-4.00, Rev. 21, NUCLEAR INSTRUMENTATION MALFUNCTION, Att. 2, INTERMEDIATE RANGE FAILURE
Added "prior to the failure" to minimize confusion as after the failure the channels should not indicate the same value.

Updated reference for AP-4.00

RFA accept 12/21/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	A C B A D D A C C C	Scramble Range: A - D
Tier:	2		Group:	2	
Key Word(s):	INTERMEDIATE RANGE		Cog Level:	MEM4.0	
Source:	N		Exam:	SR06301	
Test:	S		Author / Reviewer:	FJE	

10. 036AA2.01 002/1/2/FUEL HANDLING/C/A3.9/N/SR06301/S/FJE

Unit 1 conditions are as follows:

- Unit 1 is in Refueling Shutdown
- The Refueling SRO stopped core alterations at 11:15 after a fuel rod separated from a fuel assembly in transit in containment.
- Containment closure is NOT established
- Fission products were detected in a containment air sample taken at 11:30.
- 1-RM-RMS-162, Manipulator Crane, is in HIGH Alarm
- 1-RM-RMS-160, Containment Gas, is in HIGH Alarm
- The crew is implementing 0-AP-22.00, Fuel Handling Abnormal Conditions
- 1-VG-RMS-131/132, Kaman Vent/Vent readings are as follows:

11:15	1.2×10^4 uCi/sec
11:30	3.1×10^4 uCi/sec
11:45	5.3×10^4 uCi/sec
12:00	7.2×10^4 uCi/sec
12:15	8.3×10^4 uCi/sec
12:30	7.9×10^4 uCi/sec
12:45	6.4×10^4 uCi/sec

The Emergency Action Level classification for this event is a/an ____________ and, when acceptable radiological conditions exist, containment closure should be established within ___________.

References provided.

A. Notification of Unusual Event 45 minutes

B. Notification of Unusual Event 4 hours

C✓ Alert 45 minutes

D. Alert 4 hours

K/A
036 Fuel Handling Accident
AA2.01 Ability to determine and interpret the following as they apply to the Fuel Handling Incidents: ARM system indications (CFR 43.5)

Notes

Could not find basis for 45 minutes in Tech Specs, FSAR, or Lesson Plan
See if facility has any data on what Kaman would read with time for failed fuel over a 2 hour period

Attachments

EPIP-1.01, Rev. 46, Attachment 1 (Tab C), Fuel Failure or Fuel Handling Accident (pgs 11-16).
EPIP-1.01, Rev. 46, Attachment 1 (Tab E), Radioactivity Event (pgs 19-22).

Answer Option Analysis

A. Incorrect. Incorrect EAL classification, correct time. Plausible classification if applicant focuses on Kaman readings (greater than 2.8E4 for greater than 1 hour per Tab E).

B. Incorrect. Incorrect EAL, incorrect time. Plausible time beause 4 hours is time requirement for containment isolation valves with RCS above 400F.

C. Correct. Correct EAL per EPIP-1.01, Tab C, Condition 10. Correct time per 0-AP-22.00 Step 21 Note.

D. Incorrect. Correct EAL, incorrect time. See B.

References

0-AP-22.00, Rev. 19, Fuel Handling Abnormal Conditions
EPIP-1.01, Rev. 46, Emergency Manager Controlling Procedure
ND-92.5-LP-7, Refueling Abnormal Procedures

Added "in containment" for clarification.

Added word "HIGH" for alarms as they could be in ALERT or HIGH.

Corrected mark number for Containment Gas RM.

Added words "References provided", per the NRC.

Updated procedure revision for AP-22.00.

45 minutes is based on Alternate Source Term Implementation Plan.

RFA accept 12/21/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	C D A A D D D B D C	Scramble Range: A - D	
Tier:	1			Group:	2	
Key Word(s):	FUEL HANDLING			Cog Level:	C/A3.9	
Source:	N			Exam:	SR06301	
Test:	S			Author / Reviewer:	FJE	

11. 038EK3.06 005/1/1/SGTR/C/A4.2/N/SR06301/S/FJE

Unit 1 conditions are as follows:

- A SGTR occurred on Unit1
- 1-ES-3.1, POST-SGTR COOLDOWN USING BACKFILL, is in progress at Step 1
- The 1C RCP is RUNNING
- Ruptured S/G pressure is 1035 psig.
- Intact S/G pressures are 730 psig.

1-ES-3.1, step 1 states "Turn on PRZR heaters to saturate PRZR water at ruptured SG pressure."

Per 1-ES-3.1 step 1, the crew should maintain PRZR water temperature at approximately _____ and the basis for maintaining PRZR water temperature is to minimize _____________ associated with starting the RCPs.

A. 550 °F an RCS dilution

B. 510 °F an RCS dilution

C✓ 550 °F the RCS pressure transient

D. 510 °F the RCS pressure transient

K/A
038 Steam Generator Tube Rupture
EK3.06 Knowledge of the reasons for the following responses as they apply to the SGTR: Actions contained in EOP for RCS water inventory balance, S/G tube rupture, and plant shutdown procedures (CFR 41.5)

Notes

Attachments

1-ES-3.1, Rev. 17, POST-SGTR COOLDOWN USING BACKFILL, page 2 (Step 1).
Steam Tables

Answer Option Analysis

A. Incorrect. Althought this is the correct saturation temperature for 1050 psia, the reason is incorrect. The reason is plausible because the purpose of other steps during a SGTR is to minimize leakage through the ruptured tube.

B. Incorrect. This temperature is the saturation temperature for the INTACT SG's (745 psia). Plausible for the reason above and if the applicant calculates saturation temperature of the intact SG's.

C. Correct. This is the saturation temperature of the ruptured S/G and is the correct reason.

D. Incorrect. Although the reason is correct, the saturation temperature incorrect. Plausible if the applicant calculates saturation temperature for the INTACT SG's.

References

1-ES-3.1, Rev. 17, POST-SGTR COOLDOWN USING BACKFILL
ND-95.3-LP-14, Rev. 11, ES-3.1, POST-SGTR COOLDOWN USING BACKFILL
WOG ES-3.1, SI Termination, HP-Rev. 1
Changed Ruptured SG pressure to 1035 psig (SG PORV lift setpoint). This caused the saturation temperatures in answer 'A' and 'C' to be adjusted to 550 (answer analysis was also updated).

Updated procedure revision.

RFA accept 12/20/05

Basis is for minimizing the pressure transient for starting the RCPs.
Some steps may result in increased tube leakage.
"leakage through the ruptured tubes" was removed and changed to "minimize the possibility of a dilution". This is plausible because because if RCS pressure is not raised, SG backflow will result in an RCS dilution.

RFA Accepted 1/10/06

Added description of 1-ES-3.1, step 1 to stem, removed "references provided" and swapped with SRO TIER 1 GROUP 1 question 007EA2.02 due to required knowledge.

RFA accepted 1/19/06

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C D C C A B A C D A Scramble Range: A - D

Tier:	1	Group:	1
Key Word(s):	SGTR	Cog Level:	C/A4.2
Source:	N	Exam:	SR06301
Test:	S	Author / Reviewer:	FJE

12. 055EG2.1.32 003/1/1/BLACKOUT/C/A3.8/M/SR06301/S/FJE

Unit 1 plant conditions are as follows:

- Unit 1 reactor core is at End of Life.
- Unit 1 experienced a Loss of All AC Power. The crew is implementing 1-ECA-0.0, Loss of All AC Power. All efforts to restore an emergency bus have been unsuccessful.
- Containment parameters are NORMAL.
- The crew has just stopped depressurization of ALL SGs.
- Plant parameters <u>immediately</u> after stopping the depressurization are listed below:

"A" SG:	Narrow Range Level	21%	Pressure	185 psig
"B" SG:	Narrow Range Level	24%	Pressure	190 psig
"C" SG:	Narrow Range Level	24%	Pressure	190 psig

PRZR Level	off scale LOW
RVLIS Full range	90%
"A" RCS Tcold	316°F
"B" RCS Tcold	314°F
"C" RCS Tcold	315°F
N31 startup rate	0 dpm
N32 startup rate	0 dpm
N35 startup rate	0 dpm
N36 startup rate	0 dpm

Which ONE of the following is the basis for the limit that was EXCEEDED?

A. Prevent loss of heat sink.

B. Prevent inadequate core cooling.

C✓ Prevent challenging RCS integrity.

D. Prevent return to criticality.

K/A
055 Station Blackout
EG2.1.32 Ability to explain and apply all system limits and precautions (CFR 43.2)

Notes

Modified from facility ILT exam bank questions158 and 198.

Answer Option Analysis

A. Incorrect. Containment conditions are NORMAL and 2/3 SG NR levels are greater than 12%. Plausible if applicant believes that depressurization must be stopped if ANY SG NR level is less than 12%. Caution before step 21 states that SG NR levels should be maintained greater than 12% (normal containment) in at least one intact SG.

B. Incorrect. All SG pressures are greater than 175 psig, which, per caution before step 21, will preclude significant nitrogen injection to the RCS from SI accumulators (which could impede natural circulation). Plausible if applicant believes that indicated PRZR level or RVLIS level is sufficient justification for stopping depressurization. Note before step 21 states that depressurization should NOT be stopped under these conditions.

C. Correct. 2/3 RCS cold leg temperatures are less than or equal to 315°F. Step 21.e) states "Check RCS cold leg temperatures - GREATER THAN 315°F." RNO e)1) states "Stop SG depressurization." Per ND-95.3-LP-17, reason is to ensure that depressurization does not impose a challenge to the INTEGRITY CSF.

D. Incorrect. All IRNI and SRNI startup rates are zero. Plausible if applicant believes that EOL conditions justify termination of depressurization before reaching 175 psig. Step 22 explicitly addresses the criticality concern by checking IR channels and SR channels - zero or negative startup rate.

References

1-ECA-0.0, Rev. 23, Loss of All AC Power
ND-95.3-LP-17, Rev. 11, ECA-0.0, Loss of All AC Power
Added words "Narrow Range" for clarification on SG level.

Added "when the applicable limit was exceeded" for clarification and because teams may stop depressurization with values so close to their limits.

RFA accept 12/21/05
Changed Upper range RVLIS to Full Range RVLIS and adjusted percentages. The majority of the procedures refer to Full Range indication, Upper Range values are not commonly used, therefore this add additional challenges to this question for the applicant.

Changed question stem for applicant clarity.

RFA accept 1/12/06.

Raised SG levels based on expected SG shrink following the securing of the steam flow.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9

Answer: C D A C D A C D B B Scramble Range: A - D

Tier:	1	Group:	1
Key Word(s):	BLACKOUT	Cog Level:	C/A3.8
Source:	M	Exam:	SR06301
Test:	S	Author / Reviewer:	FJE

13. 057AG2.2.22 002/1/1/VITAL AC BUS/C/A4.1/N/SR06301/S/FJE

Unit 1 conditions are as follows:

- Unit 1 is operating at 100% power.
- At 01:12 on 1/4/06, Unit 1 lost Vital Bus 1-IIA.
- The crew has completed the applicable steps of 1-AP-10.02, Loss of Vital Bus II.

Vital Bus 1-IIA must be re-energized no later than ______ OR Unit 1 must be placed in Hot Shutdown no later than ______.

A. 03:12 on 1/4/06 07:12 on 1/4/06

B.✓ 03:12 on 1/4/06 09:12 on 1/4/06

C. 09:12 on 1/4/06 15:12 on 1/4/06

D. 09:12 on 1/4/06 15:12 on 1/5/06

K/A

057 Loss of Vital AC Instrument Bus

AG2.2.22 Knowledge of limiting conditions for operations and safety limits (CFR 43.2)

Notes

Could not find reference to 2 hour time requirement in Tech Specs. Did not have Surry TRM at time question was written. Need to find source of 2 hour time requirement (e.g. TRM or other commitment) and re-evaluate distractors. NRC GL 91.11 references the 2-hour requirement.

Answer Option Analysis

A. Incorrect. Correct time for reenergizing 1-IIA, but incorrect regarding time to HSD. Plausible if applicant applies 6 hours to HSD to time that bus was deenergized instead of expiration of 2 hour clock.

B. Correct per 1-AP-10.02 step 18 note. 2 hours to reenergize (01:12 + 2) is 03:12. 6 hours to HSD from expiration of 2 hour clock is 03:12 + 6 = 09:12

C. Incorrect. Incorrect time for reenergizing 1-IIA (even though 6 hours to HSD is correctly applied from this incorrect time). Plausible since 8 hours is time to make dependable alternate source (4kV) available.

D. Incorrect. See C. Time to HSD is plausible if applicant believes 30 hours is allowed (this is typical time to CSD).

References

1-AP-10.02, Rev. 14, Loss of Vital Bus II, Note before step 18 (pg. 5 of 11)

Updated procedure reference.
Responded to NRC question.
Deleted bolding on "/" for answer 'B'.

RFA accept 12/21/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	B C C D A C D D A A	Scramble Range: A - D
Tier:	1		Group:	1	
Key Word(s):	VITAL AC BUS		Cog Level:	C/A4.1	
Source:	N		Exam:	SR06301	
Test:	S		Author / Reviewer:	FJE	

14. 061A2.05 002/2/1/AFW MALFUNCTION/MEM3.4/N/SR06301/S/FJE

Unit 1 status is as follows:

- Unit 1 was operating at 100% power.
- The "C" SG is faulted inside containment.
- The TDAFW pump tripped immediately after starting.
- BOTH MD AFW pumps are running.
- SG C AFW Isolation MOV 1-FW-MOV-151A will NOT close from the MCR.
- SG C AFW Isolation MOV 1-FW-MOV-151B is CLOSED.

The AFW cavitating venturi will limit AFW flow to the C SG to a MAXIMUM of____ and the crew must locally isolate the ____ per 1-E-2 (FAULTED STEAM GENERATOR ISOLATION) to isolate AFW flow to the C SG.

A.✓ 350 gpm   A AFW header (H Bus)

B. 350 gpm   B AFW header (J Bus)

C. 382 gpm   A AFW header (H Bus)

D. 382 gpm   B AFW header (J Bus)

K/A
061 Auxiliary/Emergency Feedwater
A2.05 Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Automatic control malfunction (CFR 43.5)

Notes

Answer Option Analysis

A. Correct. Per ND-89.3-LP-4, the AFW cavitating ventiris are designed to limit flow to a SF affected by a MSLB or MFLB inside containment to 350 gpm based on the loss of the TDAFW pump and the availability of both MDAFW pumps. 1-FW-MOV-151A is the A AFW Header supply to the C SG. Per 1-E-2 Step 4, RNO, the crew must "locally isolate the AFW header corresponding to the failed MOV," i.e. the A header.

B. Incorrect (second part). 1-FW-MOV-151B is the B AFW Header supply to the C SG. This valve is already closed and isolating the B AFW header will not isolate AFW flow to the C SG. Plausible if applicant does not understand AFW system configuration and nomenclature.

C. Incorrect (first part). Plausible because AFW venturis are also designed to permit 382 gpm flow to the INTACT SGs.

D. Incorrect (both parts). See B, C.

References

ND-89.3-LP-4, Rev. 21, Auxiliary Feed System
1-E-0, Rev. 52, Reactor Trip or Safety Injection, Att. 9, Faulted SG(s) Isolation and AFW Flow Control
1-E-2, Rev. 10, Faulted Steam Generator Isolation, Step 4, RNO.
E-0 Attachment 9 does not procedurally address locally isolating AFW headers, this is covered in E-2. Updated question stem with E-2 information.

Included bus power supply descriptions to all distractors, as this is how E-2 describes the AFW headers.

Updated E-0 procedure revision.

RFA accept 12/21/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	A D D B A D B A D D	Scramble Range: A - D	
Tier:	2			Group:	1	
Key Word(s):	AFW MALFUNCTION			Cog Level:	MEM3.4	
Source:	N			Exam:	SR06301	
Test:	S			Author / Reviewer:	FJE	

15. 062AA2.02 002/1/1/SERVICE WATER/MEM3.6/N/SR06301/S/FJE

Plant conditions are as follows:
- 1D and **2A** water box inlets are supplying Charging Pump Service Water.
- **2A** water box inlet is supplying #5 MER Service Water.
- 1-VS-E-4E is out of service electrically for maintenance.

Workers placed a stoplog on LOW level screenwell **2A** for suction pipe cleaning on **2-CW-P-1A** (Unit 2 A Circulating Water Pump).

Which ONE of the following describes the effect on the plant?

A.✓ Emergency Service Water Pump **1B** is isolated.

B. Emergency Service Water Pump **1C** is isolated.

C. Emergency Service Water Pump **1B** is isolated AND one SW supply to Unit 2 RSHX **B** is lost.

D. Emergency Service Water Pump **1C** is isolated AND one SW supply to Unit 2 RSHX **C** is lost.

K/A
062 Loss of Nuclear Svc Water
AA2.02 Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: The cause of possible SWS loss (CFR 43.5)

Notes

Answer Option Analysis

A. Correct.

B. Incorrect. Plausible because 1-SW-P-1-C suction source is low level screenwell 2B.

C. Incorrect. Correct regarding ESW pump isolation, incorrect regarding SW supplyu to U2 RSHX B. Plausible because SW supply to Unit 2 RSHX B&C is 2A screenwell HIGH level.

D. Incorrect. See B and D.

References

ND-89.5-LP-2, Rev. 23, Service Water System.
ND-89.5-LP-1, Rev. 14, Circulating Water System.
ND-91-LP-6, Rev. 10, Recirculation Spray System.
Licensee examination bank questions 80, 82, 89.

Changed terminology from "Service Building" to "Charging Pump"

Added "for suction pipe cleaning on 2-CW-P-2A (Unit 2 A Circulating Water Pump) for clarification and to due to the low frequency of stop logging the low levels compared to high levels.

Corrected spelling error in analysis answer 'C'

RFA accept 12/21/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	A A A B C A C C B D	Scramble Range: A - D
Tier:	1		Group:	1	
Key Word(s):	SERVICE WATER		Cog Level:	MEM3.6	
Source:	N		Exam:	SR06301	
Test:	S		Author / Reviewer:	FJE	

16. 063A2.01 004/2/1/DC GROUND/C/A3.2/B/SR06301/S/STAFF/

Unit 1 conditions are as follows:

- Unit 1 is at 100% power, steady state conditions.
- The DC Positive Ground Detection light is off.
- The white light for 1-FW-P-3A, "A" AFW pump, is out.

Which ONE of the following could cause the white light for the "A" AFW pump to be out?

A. The "1H" bus is de-energized.

B✓ A hard ground exists on the "A" DC bus.

C. The "B" DC bus indicates < 75 volts.

D. Placing the control switch for 1-FW-P-3A in Pull to Lock.

K/A

063 DC Electrical Distribution

A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the DC electrical systems; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Grounds (CFR 43.5)

Notes

Need facility to determine procedure reference - could not find reference to light in references sent. Need to verify D. is not partially correct.
From exam bank - SR02301 written by Surry staff.
Made minor editorial modifications to original question.

Answer Option Analysis

A. Incorrect. Plausible because the "1H" bus is the power supply for the "A" AFW pump motor.
B. Correct.
C. Incorrect. Plausible if the applicant believes that the "B" DC bus supplies control power to the "A" AFW pump.
D. Incorrect. Plausible if the applicant believes that the indications are not symptoms of a ground. However, it is a possible answer since placing the switch in PTL will turn off all other lights for this pump.

References

ND-90.3-LP-6, Rev. 12, 125 VDC Distribution, Obj. A, C
ND-89.3-LP-4, Rev. 21, Auxiliary Feed System

Changed "dim" to "off" for proper indication of a grounded DC bus that would result in a loss of DC control power to 1-FW-P-3A.

Added the word "hard" to answer 'B' as a DC ground large enough to cause a loss of control power to the AFW pump would have to be a hard ground.

Changed answer 'D' to "Placing the control switch for 1-FW-P-3A in Pull to Lock" as "Operation from the ASDP " will cause the white light to extinguish (this has been verified on the simulator and ESK 5K, 5K1).

The new 'D' is a plausible distractor if the applicant believes that all lights are extinguished when the switch is placed in PTL (the red, green, and amber lights will be off). This also indicates a mis-understanding of the "white" light. Other pumps red, amber, and green lights extinguish when their switch is in PTL.

Applicable References:
AP-10.06, Loss of DC Power, Rev. 9.
11448-ESK-5K
11448-ESK-5K1

RFA accept 12/21/05

We have two ground detection lights (negative and positive). This was added for clarification.

RFA accept 1/12/06.

MCS	Time:	1	Points:	1.00	Version:	0 1 2 3 4 5 6 7 8 9	
					Answer:	B C D B C D B A A C	Scramble Range: A - D

Tier:	2	Group:	1
Key Word(s):	DC GROUND	Cog Level:	C/A3.2
Source:	B	Exam:	SR06301
Test:	S	Author / Reviewer:	STAFF/

17. 076G2.4.31 002/2/1/SERVICE WATER ALARM/MEM3.4/N/SR06301/S/FJE

Plant conditions are as follows:

- Unit 1 and Unit 2 are both at full power with all systems in their normal alignments.
- Outside ambient air temperature is 10 °F.
- 1-SW-P-1C was started 30 minutes ago for an engineering surveillance.
- ALL 1-SW-P-1C pump and engine parameters are NORMAL.
- 0-VSP-M6, ESW PP HSE LO TEMP, alarmed 5 minutes ago.
- A local operator at the ESW Pump House reports that building temperature is 53 °F.

Which ONE of the following is the correct action to take per 0-VSP-M6 for the given plant conditions?

A. Secure 1-SW-P-1C

B. Initiate 0-AP-12.00, Service Water System Abnormal Conditions

C. Throttle 1-SW-5, ESW Pump 1C Engine Clg Outlet, to raise engine water temperature.

D✓ Declare the diesel drivers for 1-SW-P-1A and 1-SW-P-1B inoperable.

K/A
076 Service Water
G2.4.31 Knowledge of annunciators alarms and indications and use of the response instructions (CFR 41.10 / 45.11)

Notes

Answer Option Analysis

A. Incorrect. Not directed by 0-VSP-M6. Additionally, this action would allow the 1C ESW pump diesel lube oil to cool off, contrary to the intent of 0-VSP-M6. Plausible because alarm indicates problem potentially associated with running ESW pump.

B. Incorrect. Not directed by 0-VSP-M6 and not an entry condition per 0-AP-12.00. 0-AP-12.00 actions do not address given plant conditions (ESW Pump House low temp). Plausible because the alarm indicates that SW system conditions are "abnormal."

C. Incorrect. Not directed by 0-VSP-M6. Not directed by 0-OP-SW-002 when operating parameters are normal. Plausible since low ambient air temperature could lead to lower engine water temperatures.

D. Correct per 0-VSP-M6. Per Caution before step 7, if pump house temperature drops below 55 ^{o}F, the starting batteries for the ESW Pumps, and any non-running ESW pump diesel engines are considered inoperable. Additionally, Caution states that if the diesels or their starting batteries become inoperable, an LCO clock will be entered IAW Tech Spec 3.14.B.

References

0-VSP-M6, Rev. 3, ESW PP HSE LO TEMP
ND-89.5-LP-2, Rev. 23, Service Water System
0-OP-SW-002, Rev. 20, Emergency Service Water Pump Operations
0-AP-12.00, Rev. 10, Service Water System Abnormal Conditions
Surry Tech Spec 3.14, Circulating and Service Water Systems
Corrected mark number for 1-SW-P-1A, B, and C.

Based on the KA, provide reference 0-VSP-M6

Reference not allowed. Will reevaluate use of question based on validation results.
RFA accept 12/21/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	D D B C C C A D D B	Scramble Range: A - D
Tier:	2		Group:	1	
Key Word(s):	SERVICE WATER ALARM		Cog Level:	MEM3.4	
Source:	N		Exam:	SR06301	
Test:	S		Author / Reviewer:	FJE	

18. G2.1.4 002/3//SHIFT MANNING/MEM3.4/M/SR06301/S/FJE

The following plant conditions exist:

- Unit 1 is in COLD SHUTDOWN

- Unit 2 is in REFUELING

Which ONE of the following is the MINIMUM shift manning requirement for the Station under the conditions shown above per Tech Spec 6.1, "Organization, Safety, and Operation Review."

	SS	SRO	RO	Health Physics Technician
A.	1	1	3	1
B.✓	1	0	2	1
C.	0	0	2	2
D.	0	1	3	2

K/A
G2.1.4 Knowledge of shift staffing requirements (CFR 43.2)

Answer Option Analysis

A. Incorrect for SRO and RO. Correct for HP Tech.
B. Correct answer. SRO and RO are from Table 6.1-1. HP Tech value is from 6.1.B.5
C. Incorrect for RO and HP Tech. Correct for SRO.
D. Incorrrect for SRO, RO, and HP Tech.

References

Surrey Tech Specs Section 6 (from Surrey 2003 reference material - verify revision)

Notes

Modified from Surrey ILT exam bank ID: TS00126
Added SS column for consistency with TS. Also SS are SROs, stating that 0 SROs are required is not correct/misleading.

Deleted 'e' from Surry.

RFA accept 12/21/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	B B C C B B D A B C	Scramble Range: A - D
Tier:	3		Group:		
Key Word(s):	SHIFT MANNING		Cog Level:	MEM3.4	
Source:	M		Exam:	SR06301	
Test:	S		Author / Reviewer:	FJE	

19. G2.1.7 002/3//ROD MISALIGNMENT/C/A4.4/N/SR06301/S/FJE

Unit 1 is operating at approximately 80% power, increasing to 100% at 155 MW/hr.

GP D1 Step Counter: 172 steps
GP D2 Step Counter: 172 steps

IRPI Rod B8 : 172 steps
IRPI Rod P8 : 172 steps
IRPI Rod K 6 : 172 steps
IRPI Rod K10 : 170 steps
IRPI Rod H14 : 172 steps
IRPI Rod H2 : 172 steps
IRPI Rod F10 : 172 steps
IRPI Rod F6 : 156 steps

Power Range Nuclear Instruments read as follows:

- N-44: 81.2%
- N-43: 74.3%
- N-42: 80.1%
- N-41: 80.1%

The following annunciators are LIT:

- 1E-E3, (delta) FLUX DEVIATION
- 1G-H1, NIS DROPPED ROD FLUX DECREASE (greater than) 5% PER 2 SEC

Which ONE of the following describes the correct Abnormal Operating Procedure to enter and the action the crew should take for the plant conditions listed above?

A. Enter 0-AP-1.02, INDIVIDUAL ROD POSITION INDICATORS (IRPI).
Match and stabilize reactor and turbine power at approximately 80%.

B. Enter 0-AP-1.02, INDIVIDUAL ROD POSITION INDICATORS (IRPI).
Reduce reactor power to less than or equal to 70% within 1 hour.

C. Enter 0-AP-1.01, CONTROL ROD MISALIGNMENT
Reduce reactor power to less than or equal to 50% within 8 hours.

D✓ Enter 0-AP-1.01, CONTROL ROD MISALIGNMENT
Reduce reactor power to less than or equal to 70% within 1 hour.

K/A
G2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation (CFR 43.5)

Notes

If necessaray, can include PRNI values:

Power Range Nuclear Instruments read as follows:
- N-44: 81.2%
- N-43: 74.3%
- N-42: 80.1%
- N-41: 80.1%

Answer Option Analysis

Deviation between IRPI for rod F6 and associated Bank D step counter, together with annunciators listed in stem, are indications of a misaligned (inoperable) control rod. TS 3.12.C requires reducing power to less than 75% of rated power within 1 hour if one control rod is inoperable during power operation. Procedure 0-AP-1.01 requires reducing power to less than or equal to 70%.

A: Incorrect, Incorrect procedure/diagnosis. Incorrect action. Match and stabilize reactor power is plausible per 0-AP-1.01 Step 4 AFTER reactor power is reduced to 70% or less.

B: Incorrect procedure/diagnosis. Correct action (per 0-AP-1.01). 0-AP-1.02 directs a reduction in power level to less than 50% within 8 hours per TS 3.12.E, Rod Position Indication System if MIDS is inoperable.

C: Correct procedure/diagnosis, incorrect action (see B and D).

D: Correct answer. Correct procedure, correct action (step 3).

References

0-AP-1.02, Individual Rod Position Indicators (IRPI), Rev. 10
0-AP-1.01, Control Rod Misalignment, Rev. 16
0-AP-1.00, Rod Control System Malfunction, Rev. 14
Surrey TS 3.12.C

Adjusted power increase rate to correspond to normal power increase rate.

Changed rod N8 to P8 to correspond to D Bank rods (N8 is not in D bank)

Added Power Range data per Notes section of NRC data.

RFA accept 12/21/05

MCS	Time: 1	Points:	1.00	Version:	0 1 2 3 4 5 6 7 8 9		
				Answer:	D B A A B A A C B B	Scramble Range: A - D	
Tier:	3				Group:		
Key Word(s):	ROD MISALIGNMENT				Cog Level:	C/A4.4	
Source:	N				Exam:	SR06301	
Test:	S				Author / Reviewer:	FJE	

20. G2.2.27 002/3//REFUELING/MEM3.5/N/SR06301/S/FJE

REFUELING OPERATIONS are in progress on Unit 1 following a complete core offload. Cavity level is 26 ft. 6 in.

The condition requiring IMMEDIATE suspension of REFUELING OPERATIONS per 1-OP-FH-001, CONTROLLING PROCEDURE FOR REFUELING, is...

A. Seven hours later, the Unit 1 RO reports that cavity level is 26 ft. 4 in.

B.✓ A change from 4 cps to 7 cps on N-31, and a change from 3 cps to 7 cps on N-32, following insertion of the 15th fuel assembly into the core.

C. RCS temperature changed from 110 °F to 102°F in 30 minutes due to an RH system problem which has since been corrected.

D. The latest two Unit 1 RCS boron samples reveal 2422 ppm and 2412 ppm respectively.

K/A
G 2.2.27 Knowledge of the refueling process (CFR 43.6)

Answer Option Analysis

A: Incorrect. TS 3.10-3 minimum level for movement of fuel assemblies is 23 feet above the top of the reactor vessel pressure vessel flange. Plausible because 1-OP-FH-001 step 5.4.25 states that a cavity level of 26 ft 6 in is preferred and because cavity level has slowly decreased.

B: Corrrect answer. 1-OP-FH-001 Precaution and Limitation 4.49 states "If the Source Range count rate on either detector doubles from the reference value, ..., all core alterations must be stopped immediately..."

C. Incorrect answer. No reference to RCS temperature in TS 3.10-4 or 1-OP-FH-001. Plausible because of temperature effect on reactivity.

D. Incorrect answer. Plausible because RCS boron concentration is decreasing (but has not decreased below the administrative requirement of 1-OP-FH-001).

Notes

References

Surrey TS 3.10 Section A
1-OP-FH-001, Rev. 14, Controlling Procedure for Refueling

Step 5.1.33.a of 1-OP-FH-001 requires RCS boron concentration greater than or equal to 2350 ppm to allow for refueling operations. Without raising this value, two correct answers could exist.

Adjusted answer 'C' to indicate that the temperature decrease has ceased.

A reference should be provided for this question, as the correct answer is item #49 of the precautions and limitations. Is this level of detail required knowledge?

RFA accept 12/21/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	B A C C A B A A C B	Scramble Range: A - D	
Tier:	3			Group:		
Key Word(s):	REFUELING			Cog Level:	MEM3.5	
Source:	N			Exam:	SR06301	
Test:	S			Author / Reviewer:	FJE	

21. G2.2.34 003/3//STARTUP REACTIVITY/C/A3.2/N/SR06301/S/FJE

Given the following plant conditions on Unit 1:

- You are the Unit SRO responsible for maintaining supervisory oversight of a reactor startup 12 hours after a reactor trip from full power at core MOL.
- Calculation of estimated critical rod position (ECP) per 1-OP-RX-004, THE CALCULATION OF ESTIMATED CRITICAL CONDITIONS, is complete.
- Withdrawal of control rods to the actual critical rod position (ACP) is in progress per 1-OP-RX-006, WITHDRAWAL OF THE CONTROL BANKS TO CRITICAL CONDITIONS.
- RCS temperature is being maintained at 547 °F using SG PORVs.
- Normal plant conditions for a reactor startup exist.

Condition #1

The SG PORV setpoints are inadvertently changed to 985 psig five minutes before the reactor becomes critical.

Condition #2

The reactor startup is delayed 2 hours before the reactor becomes critical.

Considering each condition separately, the actual critical rod position (ACP) compared to the estimated critical rod postion (ECP) will be__________ in Condition #1, and the actual critical rod position (ACP) compared to the estimated critical rod position (ECP) will be __________ in Condition #2.

A. higher, higher

B.✓ lower, lower

C. higher, lower

D. lower, higher

K/A
G2.2.34 Knowledge of the process for determining the internal and external effects on core reactivity (CFR 43.6)

Answer Option Analysis

Lowering SG PORV setpoint from 1005 psig to 985 psig will lower RCS temperature below that assumed in the ECC (547 ^{o}F), adding positive reactivity and resulting in a lower critical rod position.

Waiting 2 additional hours, after the reactivity worth of Xe has peaked (at about 8 hours), will result in less reactivity due to XE in the core than assumed in the ECP.

A. Incorrect.

B. Correct. Actual critical rod position goes down relative to ECC when Tave and Xe reactivity decrease.

C. Incorrect.

D. Incorrect.

Notes

References

1-GOP-1.4, Rev. 39, Unit Startup, HSD to 2% Reactor Power
1-DRP-003, Rev. 71, Curve Book, Attachment 35, SURRY UNIT 1 - CYCLE 20 XENON REACTIVITY WORTH FOLLOWING REACTOR TRIP (25 HOURS)

Updated procedure revisions.

RFA accept 12/21/05

Changed BOL to MOL as MTC could be positive at BOL.

RFA accept 1/12/06.

MCS Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
		Answer:	B A A C C D A C B C	Scramble Range: A - D	
Tier:	3		Group:		
Key Word(s):	STARTUP REACTIVITY		Cog Level:	C/A3.2	
Source:	N		Exam:	SR06301	
Test:	S		Author / Reviewer:	FJE	

22. G2.3.4 001/3//RADIATION CONTROL/MEM3.1/N/SR06301/S/FJE

A worker who currently has no (zero) dose for the quarter and for the year estimates that he will need to spend 2 hours in a room in order to perform a system inspection and lineup.

A survey map for the room provides the following information:

-Loose Surface Contamination Level: 500 dpm / 100 cm^2 beta-gamma
: 15 dpm / 100 cm^2 alpha

-Airborne Radioactivity Level: 2.0 DAC

Which ONE of the following correctly describe the posting and administrative requirements that must be met per VPAP-2101S, Radiation Protection Program (Surry)?

A. The room must be posted "CAUTION CONTAMINATED AREA" and "CAUTION AIRBORNE RADIOACTIVITY AREA."
An Internal Dose Extension request is required to perform the work.

B✓ The room must ONLY be posted "CAUTION AIRBORNE RADIOACTIVITY AREA."
An Internal Dose Extension request is NOT required to perform the work.

C. The room must be posted "CAUTION CONTAMINATED AREA" and "CAUTION AIRBORNE RADIOACTIVITY AREA."
An Internal Dose Extension request is NOT required to perform the work.

D. The room must ONLY be posted "CAUTION AIRBORNE RADIOACTIVITY AREA."
An Internal Dose Extension request is required to perform the work.

K/A
G2.3.4 Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized (CFR 43.4)

Answer Option Analysis

A.Incorrect. Neither loose beta/gamma or alpha exceed threshold for contaminated area (1000 dpm / cm2 and 20 dpm / cm2 respectively). Additionally, the workers expected exposure does not exceed the threshold requiring an Internal Dose Extension request per VPAP-2101 Section 6.4.1.

B. Correct. Airborne radioactivity level exceeds 0.3 DAC. The workers expected exposure (2 DAC x 2 hr = 4 DAC-Hours) does not exceed the threshold requiring an Internal Dose Extension request per VPAP-2101 Section 6.4.1.

C. Incorrect. Neither loose beta/gamma or alpha exceed threshold for contaminated area (1000 dpm / cm2 and 20 dpm / cm2 respectively)

D. Incorrect. Althought airborne radioactivity level exceeds 0.3 DAC, the workers expected exposure does not exceed the threshold requiring an Internal Dose Extension request per VPAP-2101 Section 6.4.1.

Note

Reference

VPAP-2101S, Rev. 0, Radiation Protection Program(Surry)

A reference is required for this question. As the correct answer is a note on Page 51 of 103. Is this level of knowledge required to be memorized or should the applicant be able to locate the information and apply it?

Updated procedure reference/revision.

Add DAC limit to equation sheet provided to applicant.

RFA accept 12/21/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	B C A B B D C C D B	Scramble Range: A - D	
Tier:	3			Group:		
Key Word(s):	RADIATION CONTROL			Cog Level:	MEM3.1	
Source:	N			Exam:	SR06301	
Test:	S			Author / Reviewer:	FJE	

23. G2.4.36 001/3//CHEMISTRY/MEM2.8/N/SR06301/S/FJE

The following plant conditions exist:

- An incident has occurred and a Site Area Emergency has been delcared.
- The Emergency Response Organization has been activated.

Which ONE of the following statements describes the primary responsibility of the Chemistry Team?

A.✓ Responsible for conducting liquid and gaseous sampling and analysis.

B. Responsible for monitoring station personnel at the Remote Assembly Area following a site evacuation.

C. Responsible for approving changes in the emergency classification based on chemistry conditions.

D. Responsible for monitoring and sample collection within the owner controlled area but outside the protected area.

K/A
G2.4.36 Knowledge of chemistry / health physics tasks during emergency operations (CFR 43.5)

Answer Option Analysis

A. Correct answer. See SEP 5.2.1.14
B. Incorrect. Describes responsibility of Evacuation Monitoring Team.
C. Incorrect. Describes responsibility of Station Emergency Director.
D. Incorrect. Describes responsibility of Onsite (Out of Plant) Monitoring Team.

Note

Reference

Surrey Emergency Plan, Rev. 49, Section 5.2
No Changes

RFA accept 12/21/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	A A B A D A B D B D	Scramble Range: A - D
Tier:	3		Group:		
Key Word(s):	CHEMISTRY		Cog Level:	MEM2.8	
Source:	N		Exam:	SR06301	
Test:	S		Author / Reviewer:	FJE	

24. G2.4.38 002/3//EMERGENCY DOSE/MEM4.0/N/SR06301/S/FJE

The following conditions exist:

- A Site Area Emergency was declared due to a LOCA outside of containment.
- An offsite release is in progress due to the leak.
- A worker has volunteered to isolate the leak.
- The volunteer has a current year to date exposure of 1 Rem TEDE.
- The volunteer has a current lifetime exposure of 3 Rem TEDE.

Which ONE of the following describes the MAXIMUM dose the worker could be allowed to receive in order to isolate the leak?

A. 5 Rem TEDE

B. 7 Rem TEDE

C. 9 Rem TEDE

D.✓ 10 Rem TEDE

K/A

G2.4.38 Ability to take actions called for in the facility emergency plan, including (if required) supporting or acting as emergency coordinator (CFR 43.5)

Answer Option Analysis

A. Incorrect. 5 Rem TEDE is EPIP-4.04 General Emergency Exposure Activities Limit.

B. Incorrect. This is the dose limit for protecting valuable property minus the worker's current year to date exposure (which is not used for this activity).

C. Incorrect. This is the dose limit for protecting valuable property minus the worker's current lifetime exposure (which is not used for this activity).

D. Corrrect Answer. See EPIP-4.04 Attachment 1

Notes

Need to verify current revision of references.

References

EPIP-4.04, Rev. 6, Emergency Personnel Radiation Exposure

Minor typographical change.

RFA accept 12/21/05

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9

Answer: D D A D D A B A C D Scramble Range: A - D

Tier:	3	Group:	
Key Word(s):	EMERGENCY DOSE	Cog Level:	MEM4.0
Source:	N	Exam:	SR06301
Test:	S	Author / Reviewer:	FJE

25. WE07EG2.4.4 002/1/2/INADEQUATE COOLING/MEM4.3/N/SR06301/S/FJE

Unit 1 conditions are as follows:

- A small break LOCA has occurred.
- Operators are performing 1-FR-C.2 in response to an ORANGE path on core cooling.
- CURRENT plant parameters are as follows:
  - Containment conditions are ADVERSE.
  - NO RCPs are running.
  - Narrow Range levels in ALL SGs are 16%.
  - Total FW flow to SGs is 550 gpm.
  - RWST level is 28%.
  - Core Exit TCs are 915 °F.
  - RCS subcooling is -30 °F.
  - RVLIS Full Range is 40%.

Which ONE of the following is the correct action to take for the given plant conditions?

A. GO TO 1-FR-H.1, Response to Loss of Heat Sink.

B. GO TO 1-ES-1.3, Transfer to Cold Leg Recirculation.

C✓ GO TO 1-FR-C.1, Response to Inadequate Core Cooling.

D. REMAIN in 1-FR-C.2, Response to Degraded Core Cooling.

K/A
WE07 Saturated Core Cooling
EG2.4.4 Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures (CFR 43.2)

Notes

Answer Option Analysis

A. Incorrect. Although NR SG levels are all less than adverse criteria of 18%, total FW flow is greater than adverse criteria of 450 gpm. Heat sink CSF must be yellow or lower. Plausible if applicant cannot remember adverse containment values for Heat Sink CSF.

B. Incorrect. RWST level is greater than transition criteria given in Caution before step 1 of 1-FR-C.2. Plausible if applicant cannot remember transition criteria for RWST level (which is independent of containment conditions).

C. Correct per F-2, Core Cooling CSF Status Tree.

D. Incorrect. Plausible if applicant cannot remember RVLIS Full Range Threshold Value of 46% or Core Exit Tc threshold value of 700 oF.

References

F-2, Rev. 1A, Core Cooling
F-3, Rev. 5, Heat Sink
1-FR-C.2, Rev. 16, Response to Degraded Core Cooling
ND-95.3, Rev. 6, Critical Safety Function Status Trees
RFA accept 12/21/05

Subcooling must be negative based high CETC. See steam tables.

RFA accept 1/12/06.

MCS Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
		Answer:	C D C A D C C C B D	Scramble Range: A - D	
Tier:	1		Group:	2	
Key Word(s):	INADEQUATE COOLING		Cog Level:	MEM4.3	
Source:	N		Exam:	SR06301	
Test:	S		Author / Reviewer:	FJE	

1. 002K5.07 002/2/2/RCS REACTIVITY/C/A3.3/N/SR06301/R/FJE

The following conditions exist on Unit 1:

- 100% power, steady state.
- Control rods in MANUAL.
- Letdown line temperature is going DOWN due to an equipment malfunction.

The resulting change in RCS boron concentration causes a ______reactivity addition.

The resulting change in RCS temperature causes a ______ reactivity addition.

A. NEGATIVE
NEGATIVE

B. NEGATIVE
POSITIVE

C✓ POSITIVE
NEGATIVE

D. POSITIVE
POSITIVE

K/A
002 Reactor Coolant

K5.07 Knowledge of the operational implications of the following concepts as they apply to the RCS: Reactivity effects of RCS boron, pressure, and temperature (CFR 41.5)

Notes

Could not find a direct reference, in lesson plan or operating procedure, stating that a decrease in letdown temp has the potential to cause an RCS deboration.

Answer Option Analysis

Decreasing letdown line temperature results in removal of boron from the RCS, adding positive reactivity. With control rods in manual, RCS temperature increases adds negative reactivity.

A. Incorrect for boron, correct for temperature.
B. Incorrect for boron, incorrect for temerature.
C. Correct Answer. Correct for both boron and temperature.
D. Correct for boron, incorrect for temperature.

References

ND-86.2-LP-3, Rev. 5
LORT Exam Bank ID: RX00064

Changed part two of question from RCS pressure to RCS temperature, as the slow nature of this transient would be easily compensated for by the master pressure controller, thus resulting in no RCS pressure transient.

Added the word "resulting" for clarification.

ND-81.1-LP-3, Slide #12.

RFA accept 12/20/05

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C A B D A B A B A C Scramble Range: A - D

Tier:	2	Group:	2
Key Word(s):	RCS REACTIVITY	Cog Level:	C/A3.3
Source:	N	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

2. 003A1.04 002/2/1/RCP OIL RESERVOIR/C/A2.6/N/SR06301/R/FJE

Unit 1 plant conditions are as follows:

- Unit 1 is in INTERMEDIATE SHUTDOWN.
- 1-RH-P-1A is operating with 1-RH-E-1A inservice.
- 1C RCP is running.
- The 1A RCP was started 15 minutes ago.

- 1B-D7, RCP 1A 1A OIL RSVR HI-LO LVL, alarmed 5 minutes ago.
- ALL 1A RCP bearing temperatures are NORMAL.
- Electricians report that 1B-D7 was caused by LOW level in the LOWER oil pot.
- An operator in containment reports that 1A lower oil pot is 2 inches LOW AND that there is NO sign of oil leakage at the 1A RCP.

Which ONE of the following describes the correct actions to perform for the current plant conditions?

A. Trip the 1A RCP AND direct Station Safety and Loss Prevention to stage foam near the containment personnel hatch.

B. Trip the 1A RCP AND monitor 1A RCP bearing temperatures at least once per hour.

C. Monitor Unit 1 CC surge tank level AND 1A RCP bearing temperatures at least once per hour.

D.✓ Increase surveillance of RCP Oil Collection System AND direct electricians to fill the 1A RCP lower oil pot.

K/A

003 Reactor Coolant Pump

A1.04 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RCPS controls including: RCP oil reservoir levels (CFR 41.5)

Notes

Did not find separate system description for RCP oil collection system. Used information in ND-88.1-LP-6.

Can change distractor B. to "Have Station Safety and Loss Prevention stage foam near the containment personnel hatch" per TRM 3.7.10 A.2 if desired.

Answer Option Analysis

A. Incorrect. Plausible because tripping the RCP is required per 1B-D7 step 2 RNO IF the RCP Oil Collection System is NOT containing oil and staging foam is required per TRM 3.7.10 Action A.2 IF the RCP Oil Collection System is NOT Operable.

B. Incorrect. Plausible because tripping the RCP is required per 1B-D7 step 2 RNO IF the RCP Oil Collection System is NOT containing oil and monitoring temperatures is required per TRM 3.7.10 Action A.2 IF the RCP Oil Collection System is NOT Operable.

C. Incorrect. First part is plausible per 1B-D7 if lower pot oil level was HIGH (e.g. CC to RCP oil cooler leakage). Second part not required (see B).

D. Correct per 1B-D7 (step 3). 1A RCP lube oil piping is intact (normal bearing temperatures, no oil leakage) and the oil collection system is operable (no oil leakage).

References

ND-88.1-LP-6, Rev. 17, Reactor Coolant Pumps
1B-D7, Rev. 2, RCP 1A OIL RSVR HI-LO LVL
SPS TRM 3.7.10, Rev. 1, Reactor Coolant Pump Oil Collection System
0-AP-35.00, Rev. 12, OIL AND HAZARDOUS SUBSTANCES RELEASE CONTROL

Changed 1A RH Loop to 1-RH-P-1A and 1-RH-E-1A for clarification (plant terminology).

Updated procedure revision.

No other changes.

RFA accept 12/20/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	D C A C D D B D A C	Scramble Range: A - D	
Tier:	2			Group:	1	
Key Word(s):	RCP OIL RESERVOIR			Cog Level:	C/A2.6	
Source:	N			Exam:	SR06301	
Test:	R			Author / Reviewer:	FJE	

3. 003K2.02 001/2/1/CCW POWER SUPPLY/MEM2.5/N/SR06301/R/FJE

Which ONE of the following is the power supply for component cooling water pump 1-CC-P-1A?

A. 4160 VAC Emergency Bus breaker 15H6

B.✓ 4160 VAC Emergency Bus (stub bus) breaker 15H10

C. 4160 VAC Emergency Bus (stub bus) breaker 15J10

D. 4160 VAC Emergency Bus breaker 15J11

K/A
003 Reactor Coolant Pump
K2.02 Knowledge of bus power supplies to the following: CCW pumps (CFR 41.7)

Notes

Answer Option Analysis

A. Incorrect. CC pumps are powered from emergency bus stub bus. 15H6 is power to 1C charging pump. Plausible because 15H6 and 15H10 are both powered from same division of emergency power.

B. Correct.

C. Incorrect. 15J10 is power supply to 1B CC pump. Plausible if applicant does not remember that "A" emergency loads are powered from 15H supplies.

D. Incorrect. 15J11 is power supply to 1B RH pump (RH pumps are the only other load on the "stub" busses). See C.

References
ND-90.3-LP-7, Rev. 18, Station Service and Emergency Distribution Protection and Control
ND-88.5-LP-1, Rev. 21, Component Cooling System

No Changes.
RFA accept 12/20/05

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B D C D D C A A B D Scramble Range: A - D

Tier:	2	Group:	1
Key Word(s):	CCW POWER SUPPLY	Cog Level:	MEM2.5
Source:	N	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

4. 004K6.02 003/2/1/DEMINERALIZER/C/A2.5/N/SR06301/R/FJE

Unit 1 plant conditions are as follows:

- The reactor is stable at full power at MOL.
- Charging and normal letdown are in service.
- 1-CC-TCV-103, Non Regenerative Heat Exchanger Temperature Control Valve, is in manual.

The auxiliary building operator notifies the control room that he believes the letdown filter (1-CH-FL-5) is becoming clogged.

This will cause temperature through the mixed bed demineralizers to __(1)__ and flow through the letdown radiation monitors to __(2)__.

A. (1) INCREASE

(2) INCREASE

B. (1) INCREASE

(2) DECREASE

C✓ (1) DECREASE

(2) INCREASE

D. (1) DECREASE

(2) DECREASE

K/A
004 Chemical and Volume Control
K6.02 Knowledge of the effect of a loss or malfunction on the following CVCS components: Demineralizers and ion exchangers (CFR 41.7)

Notes

Alternate question.

Answer Option Analysis

A. Incorrect. Plausible if applicant mis-understands location of letdown filter. Part 2 is correct.

B. Incorrect. Plausible if applicant mis-understands location of letdown filter. Part 2 is plausible if the applicant does not understand the driving head for flow through the letdown radiation monitors.

C. Correct. When the letdown filter clogs, this will result in decreased flow (and therefore temperature) through the mixed bed demineralizers. The higher differential pressure across the mixed bed IXs will result in higher letdown RM flow.

D. Incorrect. Correct for part 1, incorrect for part 2.

References

ND-88.3-LP-2, Rev. 12, Charging and Letdown
11448-FM-88A, Sheet 4, Rev. 27

RFA accept 12/20/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	C C B D B A A C B B	Scramble Range: A - D	
Tier:	2		Group:	1		
Key Word(s):	DEMINERALIZER		Cog Level:	C/A2.5		
Source:	N		Exam:	SR06301		
Test:	R		Author / Reviewer:	FJE		

5. 005A4.03 003/2/1/RHR TEMP FLOW/C/A2.8/N/SR06301/R/FJE

Unit 1 plant conditions are as follows:

- Unit 1 is in INTERMEDIATE SHUTDOWN.
- RCS is being maintained at 320 °F / 350 psig.
- 1-RC-LI-1462, PRZR LEVEL START UP is 30%.
- The RHR system is in service per 1-OP-RH-001, RHR Operations.
- 1-RH-FCV-1605, RHR HXS BYP FLOW is in AUTO.
- 1-RH-HCV-1142, RHR LETDOWN FLOW, is CLOSED.
- 1-CH-FCV-1122 is in MANUAL.

If the operator throttles fully OPEN on 1-RH-HCV-1758, RHR HXS FLOW, then RHR system flow (1-RH-FI-1605, RHR SYS FLOW) will ________ and pressurizer level will __________.

A. go DOWN — go DOWN

B. go DOWN — remain CONSTANT

C.✓ remain CONSTANT — go DOWN

D. remain CONSTANT — remain CONSTANT

K/A
005 Residual Heat Removal
A4.03 Ability to manually operate and/or monitor in the control room: RHR temperature, PZR heaters and flow, and nitrogen (CFR 41.7)

Notes

Check with facility regarding type of controller (and typical demand values, if applicable) for 1-HCV-1758.

Answer Option Analysis

A. Incorrect (first part). Total RH system flow will remain constant, even though flow through the in-service RHR HX is increased, because 1-RH-FCV-1605 is in AUTO and will throttle closed, reducing bypass flow. Plausible if applicant does not understand system configuration or control relationships of normal RH operations (e.g. confuses bypass and system flows).

B. Incorrect (both parts). See a for first part of question. PRZR level will go down, even though RHR letdown is isolated, because RCS temperature drops when 1-HCV-1758 is throttled open. RCS temperature drops because flow through the in-service RHR HX goes up when 1-HCV-1758 is throttled open (and total RHR system flow remains constant). Plausible if applicant does not recognize that PRZR level will go down with RCS temperature, even though RH letdown is isolated.

C. Correct.

D. Incorrect (second part). See B.

References

Licensee examination bank question numbers RHR0002, RHR0006, RHR0015
ND-88.2-LP-1, Rev. 8, Residual Heat Removal System
1-OP-RH-001, Rev. 14, RHR Operations
Corrected 1758 mark number.
RFA accept 12/20/05

Added the word "fully" to clarify system response.

RFA accept 1/19/06

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	C C C A D B D A B A	Scramble Range: A - D
Tier:	2		Group:	1	
Key Word(s):	RHR TEMP FLOW		Cog Level:	C/A2.8	
Source:	N		Exam:	SR06301	
Test:	R		Author / Reviewer:	FJE	

6. 005K5.05 003/2/1/SOLID PLANT/C/A2.7/N/SR06301/R/FJE

Unit 1 plant conditions are as follows:

- Unit 1 is in COLD SHUTDOWN.
- RHR is in service.
- The RCS is SOLID.
- RCS pressure is 350 psig.
- 1-CH-PCV-1145, Low Pressure Letdown Line Pressure Control Valve, is in AUTO, controlling RCS pressure.
- Maintenance workers have broken the air line on 1-CH-PCV-1145.

Which ONE of the following describes the correct initial response on the CVCS and RCS to this event with NO operator action?

A. Charging flow INCREASES, RCS pressure INCREASES.

B. Letdown flow DECREASES, RCS pressure INCREASES.

C. Charging flow DECREASES, RCS pressure DECREASES.

D✓ Letdown flow INCREASES, RCS pressure DECREASES.

K/A
005 Residual Heat Removal
K5.05 Knowledge of the operational implications of the following concepts as they apply to the RHRS: Plant response during "solid plant" – pressure change due to the relative incompressibility of water (CFR 41.5)

Notes

Have facility verify component names.

Answer Option Analysis

A. Incorrect. Initially charging flow will not change. RCS pressure will go down (see D). Plausible if applicant doesn't understand solid plant RH and CV system configuration and interaction.

B. Incorrect. Letdown flow will rise as PCV-1145 fails open and RCS pressure will fall. Applicant must know that PCV-1145 fails open on a loss of air. Plausible if applicant doesn't understand solid plant RH and CV system configuration and interaction.

C. Incorrect. Charging flow will remain constant even though RCS pressure will fall as letdown flow increases due to PCV-1145 failing open. Plausible if applicant doesn't understand solid plant RH and CV system configuration and interaction.

D. Correct. Letdown flow will rise since PCV-1145 fails open on the loss of air and RCS pressure will fall since letdown has increased and charging flow is constant.

References

ND-88.3-LP-2, Rev. 12, Charging and Letdown
Facility exam bank questions 75, 282, 302, 339

Changed falls to decreases and rises to increases.
RFA accept 12/20/05

MCS	Time: 1	Points: 1.00	Version: 0 1 2 3 4 5 6 7 8 9		
			Answer: D A D B C C B B C C		Scramble Range: A - D
			Group:	1	
Tier:	2		Cog Level:	C/A2.7	
Key Word(s):	SOLID PLANT		Exam:	SR06301	
Source:	N		Author / Reviewer:	FJE	
Test:	R				

7. 006K1.11 003/2/1/ECCS CV CC/MEM2.8/N/SR06301/R/FJE

A complete loss of Unit 1 Charging Pump Component Cooling Water System has occurred. Which ONE of the following adverse conditions will arise, assuming no operator action?

Charging pump:

A. lube oil temperature will increase.

B. motor temperatures increase.

C.✓ mechanical seal temperature increases.

D. speed increaser gear temperature increases.

K/A
006 Emergency Core Cooling
K1.11 Knowledge of the physical connections and/or cause-effect relationships between the ECCS and the following systems: CCWS (CFR 41.2 to 41.9)

Notes

Answer Option Analysis

A. Incorrect. Plausible if the applicant believes lube oil is cooled by Charging Pump CC vice Charging Pump SW.

B. Incorrect. Plausible if the applicant believes that the motor is cooled by changing pump CC.

C. Correct per ND-88.3-LP-5.

D. Incorrect. Plausible if the applicant believes the speed increaser gear is cooled by charging pump CC vice lube oil.

References

ND-88.5-LP-1, Rev. 21, Component Cooling
ND-88.3-LP-5, Rev. 16, Charing Pumps
SPS RHR system is not an ECCS. This is an alternate question. CC and ECCS interactions are limited.
RFA accept 12/20/05

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C D B D B A B D A D Scramble Range: A - D

Tier:	2	Group:	1
Key Word(s):	ECCS CV CC	Cog Level:	MEM2.8
Source:	N	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

8. 007A1.01 004/2/1/PRT LEVEL/C/A2.9/M/SR06301/R/FJE

Unit 1 conditions are as follows:

- RCS is solid.
- The team is performing 1-GOP-2.6, Unit Cooldown, Less than 205 °F to Ambient.
- RCS temperature is being maintained at 160 °F.
- The "A" RHR train is in service with letdown established via 1-RH-HCV-1142.
- The "B" RCP is in operation.
- Letdown flow is 90 gpm.
- Charging flow is 85 gpm and seal injection flow is an additional 8 gpm/pump (established to ALL three RCPs).

Which ONE of the following describes the result of placing the letdown pressure control valve, 1-CH-PCV-1145, in MANUAL with no further operator action?

A. VCT level would go UP.

B.✓ PRT level would go UP.

C. RCS pressure would remain STABLE.

D. "B" RCP #1 seal differential pressure would go DOWN.

K/A
007 Pressurizer Relief/Quench Tank

A1.01 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRTS controls including: Maintaining quench tank water level within limits (CFR 41.5)

Notes

Modifed from bank question SM04301

Answer Option Analysis

A. Incorrect. VCT level will go down. Plausible because charging plus seal injection is greater than letdown.

B. Correct. RCS pressure will go up until Pressurizer PORV lifts (OPMS), causing PRT level to go up.

C. Incorrect. RCS pressure will go up because charging flow plus seal injection flow (85 + 3*8) = 109 gpm is greater than letdown flow plus seal leakoff flow (90 + 3*3) = 99 gpm. Plausible if applicant does not understand mass balance during solid plant ops.

D. Incorrect. Seal differential pressure will increase because RCS pressure increases. Plausible if applicant does not understand mass balance during solid plant ops or RCP seal configuration.

References

1C-G7, Rev. 1, PRZR RELIEF TK HI LVL
ND-88.1-LP-3, Rev. 13, Pressurizer and Pressure Relief
ND-88.2-LP-1, Rev. 8, Residual Heat Removal System
ND-88.3-LP-2, Rev. 12, Charging and Letdown
1-OP-RH-001, Rev. 14, RHR Operations
1-GOP-2.6, Rev. 20, Unit Cooldown, Less Than 205 ^{o}F to Ambient

Separated bullet #1 into two bullets for clarification and capitalized the word ALL to avoid confusion with preceeding bullet (1-RC-P-1B running).

In answer section changed RHR relief valve to PRZR relief valve, as the PRZR relief valve would lift before the RHR relief valve.
RFA accept 12/20/05

Re-arranged bullets so that RHR information (cooling) is with the RCS temperature.
RFA Accepted 1/10/06

Clarified charging and seal injection flows in last bullet of modified stem.

RFA Accepted 1/19/06

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	B D B C B D B B B B	Scramble Range: A - D	
Tier:	2		Group:		1	
Key Word(s):	PRT LEVEL		Cog Level:		C/A2.9	
Source:	M		Exam:		SR06301	
Test:	R		Author / Reviewer:		FJE	

9. 007A2.02 002/2/1/PRT PRESSURE/C/A2.6/N/SR06301/R/FJE

Unit 1 conditions are as follows:

- Unit 1 is at 20% power and shutting down to repair leaking pressurizer safety valve 1-RC-SV-1551A.
- PRT level is 68%.
- PRT pressure is 9 psig.
- Letdown is in service. Regen HX Relief Valve, 1-CH-RV-1203, leaks by at 4 gallons / day.
- **Unit 2** stripper degas evolutions are in progress.

Which ONE of the following is the correct method to reduce PRT (Pressurizer Relief Tank) pressure in accordance with 1-OP-RC-011, Pressurizer Relief Tank Operations?

Reference provided.

A. Drain the PRT to the PDTT (Primary Drains Transfer Tank).

B. Vent the PRT to the Vent Vent System.

C. Vent the PRT to the Process Vent System.

D✓ Vent the PRT to the Overhead Gas System.

K/A
007 Pressurizer Relief/Quench Tank
A2.02 Ability to a) predict the impacts of the following malfunctions or operations on the PS; and b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Abnormal pressure in the PRT
(CFR 41.5)

Notes

Need facility to provide the right words to the effect that stripper degas evolution(s) are in progress on U1 or U2.
Need facility to verify that leakage past 1-RV-1203 (and not 1-RC-SV-1551A) is considered unisolated leakage into the PRT.

Answer Option Analysis

A. Incorrect. Per caution before step 5.3.3, draining is prohibited during stripper degas evolutions in order to prevent an unplanned RCS dilution. Plausible because draining the PRT would reduce pressure.

B. Incorrect. With letdown in service, the leak past 1-RV-1203 is an unisolated leakage path into the PRT. Step 5.5.1 directs performance of section 5.5.7 (vent to Overhead Gas System) if an unisolated leakage path into the PRT exists. Plausible because venting to the Vent Vent System would reduce PRT pressure.

C. Incorrect. Same reasons as B.

D. Correct.

References

1-OP-RC-011, Rev. 16, Pressurizer Relief Tank Operations
ND-88.1-LP-3, Rev. 13, Pressurizer and Pressure Relief
A reference is required for this question. This corresponds to the K/A which states "...use procedures to...". Without the procedure this question is too difficult, with the procedure it is still diffilcult, but within the abilities of a licensed operator.

Corrected mark number for 1-CH-RV-1203.

1-RC-SV-1551A and 1-CH-RV-1203 are unisolated leakage to the PRT. Does not appear to impact question.
RFA accept 12/20/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	D D D C C C D B B C	Scramble Range: A - D	
Tier:	2			Group:	1	
Key Word(s):	PRT PRESSURE			Cog Level:	C/A2.6	
Source:	N			Exam:	SR06301	
Test:	R			Author / Reviewer:	FJE	

10. 007EA2.02 002/1/1/TRIP ATWS/MEM4.6/N/SR06301/R/FJE

Unit 1 conditions are as follows:

- The unit was operating at 100% power.
- NO Main Feed Pumps are running.
- The reactor would not trip.
- Rod control is in automatic.
- The turbine would not trip using the trip pushbuttons.
- ALL other systems and controls are functioning normally.
- NO other operator action have been taken.

Which ONE of the following is the next correct immediate action for the crew to perform as directed by 1-FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS?

A. Verify MD AFW pumps -- RUNNING

B. Reduce turbine load using Turbine Manual.

C. Close MSTVs.

D✓ Reduce turbine load using the Valve Position Limiter.

K/A
007 Reactor Trip
EA2.02 Ability to determine or interpret the following as they apply to a reactor trip: Proper actions to be taken if the automatic safety functions have not taken place (CFR 45.5 / 45.6)

Notes

Need to verify terminology for manual trip pushbutton.

Answer Option Analysis

A. Incorrect. This action is step 4 of 1-FR-S.1 and is not an immediate action. Reducing turbine load takes priority over verifying AFW actuation (30 sec vs. 60 sec for an ATWS event where a loss of normal FW has occurred.)

B. Incorrect. This action would only be taken for a loss of main feedwater flow with reactor power between 65% and 85%.

C. Incorrect. 1-FR-S.1 step 2 RNO directs the operator to attempt to reduce turbine load using the limiter BEFORE closing the MSTVs. This action would be correct if the turbine did not trip and the crew were following the actions of 1-E-0. However, because the reactor would not trip, the immediate actions of 1-FR-S.1 are applicable.

D. Correct. Because the reactor would not trip, the immediate actions of 1-FR-S.1 are applicable. If the turbine did not trip, step 2 RNO first directs the operator to reduce turbine load using the limiter.

References

1-FR-S.1, Rev. 19, RESPONSE TO NUCLEAR POWER GENERATION/ATWS
1-E-0, Rev. 52, REACTOR TRIP OR SAFETY INJECTION
ND-95.3-LP-36, Rev. 10 1-FR-S.1, RESPONSES TO NUCLEAR POWER GENERATION/ATWS
1-AP-21.00, Rev. 6, LOSS OF MAIN FEEDWATER FLOW

Deleted "the RO is manually inserting control rods." and replaced with "rod control is in automatic." This was done to ensure proper procedure flow path. No change of intent.

Updated procedure revision numbers and corrected FR-S.1 step number.

Editorial Change.

RFA accept 12/21/05

Swapped with SRO TIER 1 GROUP 1 question 038EAK3.06 due to required knowledge.

RFA accepted 1/19/06

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	D A B C C A C D A D	Scramble Range: A - D

Tier:	1	Group:	1
Key Word(s):	TRIP ATWS	Cog Level:	MEM4.6
Source:	N	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

11. 007EG2.4.4 002/1/1/TRIP CRITERIA/C/A4.0/N/SR06301/R/FJE

At 10:00, Unit 1 completed a load reduction from 50% to 28% power at 1.0% / minute.

At 10:07, Unit 1 plant conditions are as follows:

- Reactor power is 28 %.
- Turbine load is 190 MW.
- Condenser vacuum is 25.5 IN-HG.
- Intake canal level is 25.5 FEET.
- ROD CONT MODE SEL switch is in MANUAL.
- Control Bank D is at 160 steps and NOT moving.
- IRPI for CBD rod K10 indicates 182 steps.

ALL parameters listed above have been STABLE since completing the load reduction at 10:00.

- The following Unit 1 alarms are lit:
  - 1H-A4, T AVG >< T REF DEVIATION
  - 1B-E1, INTAKE CANAL HI-LO LVL
  - 1F-H8, INTK CANAL LO LVL CH-2
- Additionally, ONLY the following **Unit 2** alarm is lit:
  - 2B-E1, INTAKE CANAL HI-LO LVL

The Unit 1 SRO has directed you to trip the Unit 1 reactor.

Which ONE of the following describes the correct procedural basis for the Unit 1 SRO to direct a Unit 1 reactor trip?

A. 0-AP-12.01, LOSS OF INTAKE CANAL LEVEL

B✓ 1-AP-14.00, LOSS OF MAIN CONDENSER VACUUM

C. 0-AP-1.00, ROD CONTROL SYSTEM MALFUNCTION

D. 1-H-A4, T AVG >< T REF DEVIATION

QUESTIONS REPORT
for SURRY 2006-301 RO FINAL 02_03_06

K/A -

007 Reactor Trip

EG2.4.4 Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures (CFR 41.10)

Notes

Need to verify Unit 2 alarm location/title.
Need to verify rod positions against RIL.

Answer Option Analysis

A. Incorrect. Trip criteria in referenced procedures is either intake level less than or equal to 23.5 ft or 3/4 intake canal low level alarms lit. With intake canal level stable 2 feet above 23.5 ft and 1/4 low level alarms lit (spurious), a trip is not directed by procedure.

B. Correct. Caution before step 1 of 1-AP-14.00, as well as Attachment 3, Trip Criteria During Rapid Load Reduction, state that the turbine must be taken off line if condenser vacuum is less than 26.5 in-Hg (for Turbine load less than 30%) and cannot be recovered within 5 minutes. Vacuum has been less than 26.5 in-Hg and stable for seven minutes. Therefore, trip criteria of 1-AP-14.00 are met.

C. Incorrect. One control rod is misaligned with it's group demand position. With only one rod affected above 25% reactor power, and both reactor and turbine power stable, trip criteria in 0-AP-1.00 are not met.

D. Incorrect. 1H-A4 does not contain trip criteria. For deviations due to abnormal transients, ARP directs a return to procedure in effect.

References
1H-A4, Rev. 3, T AVG T REF DEVIATION
1B-E1, Rev. 2, INTK CANAL HI-LO LVL
1F-H8, Rev. 3, INTK CANAL LO LVL CH-2
1F-G1, Rev. 3, INTK CANAL LOW LVL TRIP
0-AP-12.01, Rev. 22, LOSS OF INTAKE CANAL LEVEL
0-AP-1.00, Rev. 14, ROD CONTROL SYSTEM MALFUNCTION
1-AP-14.00, Rev. 3, LOSS OF MAIN CONDENSER VACUUM
RIL is about 70 steps for 28% power, Unit 2 alarm location/title is correct.

Changed ramp rate from 2.5% / minute to 1.0% / minute to be consistent with normal plant operations.

Added "at 10:00" for clarity and to break it out from the bullets.

Added >< to annunciator 1H-A4 for consistency with alarm title.

Added the word ONLY to the unit 2 alarms to prevent the assumption that other unit 2 alarms may exist. Corrected grammer (changed are to is).

Updated procedure revision for AP-12.01.

RFA accept 12/20/05

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B B B A B A D D A D Scramble Range: A - D

Tier:	1	Group:	1
Key Word(s):	TRIP CRITERIA	Cog Level:	C/A4.0
Source:	N	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

12. 008A4.08 002/2/1/CCW PUMP CONTROL/C/A2.5/M/SR06301/R/FJE

Unit 1 conditions are as follows:

- At 10:00:00 a Hi Hi CLS actuation occurred.
- At 10:01:15 the "A" Component Cooling pump tripped.
- At 10:02:00 component cooling system pressure was less than 55 psig.

Which ONE of the following is correct?

The "B" Component Cooling pump will auto start at ___________.

A. 10:01:15 with NO operator action.

B. 10:05:15 with NO operator action.

C. 10:02:00 IF the "A" CC pump control switch is placed to STOP.

D✓ 10:05:15 IF the local reset switch on the GDC-17 panel is turned to RESET.

K/A
008 Component Cooling Water
A4.08 Ability to manually operate and/or monitor in the control room: CCW pump control switch (CFR 41.7)

Notes

Modified from SR04301
Need facility to verify GDC-17 test switch name and position terminology and verify "C" incorrect.

Answer Option Analysis

A. Incorrect. Auto start inhibit is blocked for 5 min 15 sec following the Hi-Hi CLS.

B. Incorrect. Pump will not auto start without taking manual action to either start pump or reset auto start relay.

C. Incorrect. 5 min 15 sec has NOT elapsed since Hi-Hi CLS and manipulating "A" CC pump C/S will not clear auto start inhibit.

D. Correct per ND-88.5-LP-1 pg. 50, last paragraph. At 10:05:15, CC header pressure is less than the auto start setpoint of 55 psig (no pumps running for greater than 5 minutes) and the action specified will clear the auto start inhibit relay since the auto start inhibit timer (set for 5:15 on a Hi Hi CLS) has timed out.

References

ND-88.5-LP-1, Rev. 21, Component Cooling
Facility examination bank question numbers CC000001, CC00014, CC00039

Modified stem to capitalize ONE for consistency. No impact.

Corrected noun name of local reset switch on the GDC-17 panel, per NRC request.

Corrected typo in answer selection 'D'.

RFA accept 12/20/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	D B B A B C A A D A	Scramble Range: A - D
Tier:	2		Group:	1	
Key Word(s):	CCW PUMP CONTROL		Cog Level:	C/A2.5	
Source:	M		Exam:	SR06301	
Test:	R		Author / Reviewer:	FJE	

13. 009EK1.01 001/1/1/REFLUX/MEM4.2/N/SR06301/R/FJE

Unit 2 plant conditions are as follows:

- The reactor has tripped and a prolonged Loss of All AC Power condition is occurring.
- Efforts to restore AC power have NOT been successful.
- RCP seal leakage has caused steam void formation in the reactor vessel head and in the S/G U-tubes.

If AC power is NOT restored, natural circulation will.......

A. stop, and all means of decay heat removal will be lost when a void forms in the reactor vessel upper head. Then, inadequate core cooling may occur.

B. decrease, but continue to provide adequate decay heat removal for as long as the secondary heat sink is maintained.

C✓ stop, but reflux boiling will provide some decay heat removal until enough RCS inventory is lost to uncover the reactor fuel. Then, inadequate core cooling may occur.

D. decrease, then reflux boiling will provide adequate decay heat removal for as long as the secondary heat sink is maintained.

K/A
009 Small Break LOCA
EK 1.01 Knowledge of the operational implications of the following concepts as they apply to the small break LOCA: Natural circulation and cooling, including reflux boiling (CFR 41.8 / 41.10)

Notes

Answer Option Analysis

An extended loss of all AC power will result in loss of coolant through the RCP seals (SBLOCA). Until AC power is restored, the RCS will continue to depressurize and lose inventory, resulting in progressive saturation of the RV upper head and the hot leg piping, causing steam voids to form in the SG U-tubes. Significant voiding in the U-tubes will stop natural circulation through the RCS loops and heat will be removed through reflux boiling between the core and SG's, until there is insufficient mass in the RCS to support reflux boiling and the reactor fuel is uncovered. Once the fuel is uncovered, indadequate core cooling may occur.

A. Incorrect. Although natural circulation eventually stops, reflux boiling will continue to remove decay heat. Additionally, neither natural circulation or reflux boiling cease when the RV upper head voids. As long as water level in at least one SG is above the top of the U-tubes, both forms of cooling will continue until the RCS inventory is further depleted.

B. Incorrect. Natural circulation stops. Additionally, without restoration of power, RCS inventory continues to deplete to the point that natural circulation is not possible, regardless of the water level in the SG(s).

C. Correct. RCS inventory will decrease to the point where natural circulation is no longer supported, but reflux boiling is still effective at removing decay heat. As RCS inventory continues to be lost, reflux boiling will cease and the core will uncover.

D. Incorrect. Natural circulation will eventually stop, as will reflux boiling, regardless of SG water level(s).

References

WOG ECA-0.0, HP-Rev. 1C
ND-95.2-LP-7, Rev. 8, Loss of Reactor Coolant Accident
No Changes.

RFA accept 12/20/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	C B A A D A C C C B	Scramble Range: A - D	
Tier:	1			Group:	1	
Key Word(s):	REFLUX			Cog Level:	MEM4.2	
Source:	N			Exam:	SR06301	
Test:	R			Author / Reviewer:	FJE	

14. 010A2.01 004/2/1/PRESSURIZER HEATERS/C/A3.4/M/SR06301/R/FJE

Unit 1 conditions are as follows:

- The reactor is at 100% power.
- A malfunction in the Pressurizer Heater Control Circuit resulted in the Proportional Heaters being deenergized.
- The crew has entered 1-AP-31.00, Increasing or Decreasing RCS Pressure.
- The Pressurizer Heater Control Circuit has been repaired (Proportional Heaters are operating properly and the control switch has been reset).
- A small amount of leakage past the Pressurizer Auxiliary Spray Valve is occurring.
- Pressurizer Pressure is 2225 psig and SLOWLY going UP.

Which ONE of the following lists the correct positions of the Proportional Heaters and Backup Heaters?

A.✓ Proportional Heaters - ON
Backup Heaters - ON

B. Proportional Heaters - OFF
Backup Heaters - OFF

C. Proportional Heaters - ON
Backup Heaters - OFF

D. Proportional Heaters - OFF
Backup Heaters - ON

K/A

010 Pressurizer Pressure Control System
A2.01 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 operation: Heater failures
(CFR: 41.5 / 43.5)

Notes

Modified from SR04301 027AK3.01. Facility may want to validate on simulator.

Answer Option Analysis

A. Correct. Due to low pressure, proportional heaters are on and back-up heaters are maintained 'on'.

B. Incorrect. (both parts) Proportional heaters are ON between 2220psig and 2250 psig and back-up heaters are in 'on'. Plausible if the applicants do not recall pressurizer pressure control system alignment.

C. Incorrect. Proportional heaters will be ON. See B

D. Incorrect Part 2 is correct, Part 1 is incorrect at this pressure.

References

1-AP-31.00, Rev. 7, Increasing or Decreasing RCS Pressure
ND-93.3-LP-5, Rev. 11, Pressurizer Pressure Control

Added "and the control switch has been reset" to ensure the applicants understand the proportional heaters are operating normally (all recovery actions completed).

Changed correct answer to 'A' as normal plant operations maintains the back-up heater control switches in the "on" position. Reference ND-93.3-LP-5 (page 8).

RFA accept 12/20/05

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A A B C A B A C D B Scramble Range: A - D

Tier:	2	Group:	1
Key Word(s):	PRESSURIZER HEATERS	Cog Level:	C/A3.4
Source:	M	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

15. 010K6.01 001/2/1/PRESSURE CONTROL/MEM2.7/N/SR06301/R/FJE

Plant conditions are as follows:

- Unit 1 is operating at 50% reactor power with all pressurizer pressure controls in AUTO.
- Pressurizer pressure transmitter PT-444 just failed LOW.

With no operator action, unit 1 will ______ and RCS pressure will stabilize at ______.

A. TRIP 2000 psig

B. TRIP 2335 psig

C. remain AT POWER 2000 psig

D✓ remain AT POWER 2335 psig

K/A
010 Pressurizer Pressure Control
K6.01 Knowledge of the effect of a loss or malfunction of the following will have on the PZR PCS: Pressure detection systems (CFR 41.7)

Notes

Answer Option Analysis

A. Incorrect (both parts). Plausible because plant will trip and stabilize at 2000 psig for PT-444 failing HIGH.

B. Incorrect (first part). Plant will not trip. Plausible because plant will trip for PT-444 failing HIGH.

C. Incorrect (second part). Plausible because plant stablizes at 2000 psig for HIGH failures of PT-444 and PT-445.

D. Correct. PT-444 provides input to the Pressure Control System Master Controller Module. The Master Controller provides control signals to the spray valves, pressurizer heaters, and PORV-1455C. With all pressurizer controls in auto and PT-444 failed low, both spray valves will close and all heaters will energize. RCS pressure will increase to 2335 psig and PORV-1456 (controlled by PT-445) will cycle to maintain pressure at approximately 2335 psig.

References

ND-93.3-LP-5, Rev. 11, Pressurizer Pressure Control
No Changes. Could change PT-444 to PT-1444.

RFA accept 12/20/05

MCS Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
		Answer:	D C B B D B D A C A	Scramble Range: A - D	
Tier:	2	Group:	1		
Key Word(s):	PRESSURE CONTROL	Cog Level:	MEM2.7		
Source:	N	Exam:	SR06301		
Test:	R	Author / Reviewer:	FJE		

16. 011EG2.1.20 003/1/1/LOCA/C/A4.3/N/SR06301/R/FJE

Unit 1 conditions are as follows:

- A large break LOCA has occurred.
- HI HI CLS has been actuated.
- ALL CS, ISRS, and OSRS pumps started automatically.
- 1-SW-MOV-105C ('C' RSHX SW Outlet Valve) failed to open upon receipt of the HI HI CLS signal and cannot be opened from the MCR.
- 1-RS-E-1A (1A RS HX) and 1-RS-P-1A were removed from service per 1-RM-B7 due to 1-SW-RI-114 alarming HIGH.
- Containment pressure is currently 13 psia and STABLE.
- The 1G/2G transformers have failed, therefore, ALL three Emergency SW Pumps are RUNNING.
- 1-E-1, LOSS OF REACTOR OR SECONDARY COOLANT, is in progress and has been completed through Step 8 RNO, UP TO step 9.

Which RS pump(s) will be RUNNING after the crew performs step 12, CHECK IF CTMT DEPRESSURIZATION EQUIPMENT CAN BE STOPPED, of 1-E-1?

Reference provided.

A.✓ 1-RS-P-1B, 1-RS-P-2A, AND 1-RS-P-2B

B. 1-RS-P-1B AND 1-RS-P-2A ONLY

C. 1-RS-P-1B ONLY

D. 1-RS-P-2A ONLY

K/A
011 Large Break LOCA
EG2.1.20 Ability to execute procedure steps (CFR 41.10)

Notes

Attachments

Attach 1-E-1, Rev. 24, LOSS OF REACTOR OR SECONDARY COOLANT, pages 6 - 13 (steps 8 - 13)

Answer Option Analysis

A. Correct. 1-RS-P1A is secured per 1-RM-B7. Although SW is isolated to RS HX 1C, the associated OSRS pump, 1-RS-2A, is left running IAW 1-E-1.

B. Incorrect. Plausible if applicant fails to recognize 1-SW-MOV-105B AND C must be open and incorrectly performs step 9.d) RNO 3).

C. Incorrect. Plausible if applicant fails to recognize that containment pressure is NOT less than 12 psia in step 12.a)

D. Incorrect. Plausible if applicant fails to skip step 10 and performs step 10.b)4)

References

1-E-1, Rev. 24, LOSS OF REACTOR OR SECONDARY COOLANT
Corrected mark number in stem of question for 1-SW-MOV-105C. Provided a noun name for 1-SW-MOV-105C for clarification.

Added "and cannot be opened from the MCR", as attachment 1 in E-0 will direct manually opening the valve.

Added 1-RS-P-1A removed from service per the ARP, to allow for proper flow path through E-1, no impact on question difficulty. K/A supported.

Added the loss of the 1G/2G transformers to give a reason why three ESW pumps are running.

RFA accept 12/20/05

Corrected mark numbers in choice A and B.

Replaced "Slowly trending down" with "STABLE" and replaced "completed" with "performs" to allow for proper completion of the procedure step.

RFA Accepted 1/10/06

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A B D B B A B B D D Scramble Range: A - D

Tier:	1	Group:	1
Key Word(s):	LOCA	Cog Level:	C/A4.3
Source:	N	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

17. 012K2.01 002/2/1/RPS POWER SUPPLY/MEM3.3/B/SR06301/R/FJE

Unit 1 plant conditions are as follows:
- Reactor power is 5%.
- The "A" reactor trip breaker and the "B" reactor trip **bypass** breaker are CLOSED.

Assuming no operator action, which ONE of the following correctly describes the response of the Reactor Protection System to the loss of the "**A**" DC Bus?

A. "B" reactor trip bypass breaker remains CLOSED, "A" reactor trip breaker OPENS.

B. "A" reactor trip breaker remains CLOSED, "B" reactor trip bypass breaker OPENS.

C. BOTH the "A" reactor trip breaker AND "B" reactor trip bypass breaker remain CLOSED.

D.✓ BOTH the "A" reactor trip breaker AND "B" reactor trip bypass breaker OPEN.

K/A
012 Reactor Protection

K2.01 Knowledge of bus power supplies to the following: RPS channels, components, and interconnections (CFR 41.7)

Notes

Modified from facility examination bank question number DC00012.

Answer Option Analysis

A. Incorrect. Plausible if applicant believes "B" train reactor trip bypass breaker UV coils are powered from "B" 125 VDC.

B. Incorrect. Plausible if applicant believes "A" train reactor trip breaker UV coils are powered from "B" 125 VDC.

C. Incorrect. Plausible if applicant does not recognize that 125 VDC powers reactor trip and bypass breakers.

D. Correct. The "A" 125V DC train supplies power to both the "A" reactor trip UV trip coils and the "B" reactor trip bypass breaker UV trip coils.

References

ND-93.3-LP-10, Rev. 5, Reactor Protection - General
ND-90.3-LP-6, Rev. 12, 125 VDC Distribution
Facility examination bank question number DC00012
No Changes.

RFA accept 12/20/05

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D D B B D A B B C B Scramble Range: A - D

Tier:	2	Group:	1
Key Word(s):	RPS POWER SUPPLY	Cog Level:	MEM3.3
Source:	B	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

18. 013K2.01 003/2/1/ESF POWER SUPPLY/C/A3.6/N/SR06301/R/FJE

Unit 1 plant conditions are as follows:

- 1D-B8, SFGDS DC CONT PWR TRBL, is LIT
- The white DC Control Power light on Safeguards Panel "A" door is NOT LIT
- The white DC Control Power light on Safeguards Panel "B" door is LIT
- Unit 1 has remained at 100% power with both reactor trip breakers CLOSED

Which ONE of the following is correct for the given plant conditions?

A. DC Bus 1-1 has lost power.
If an SI signal is generated, 1-SI-P-1A will start automatically.

B.✓ DC Cabinet 1-1, BKR 17 has tripped open.
If an SI signal is generated, 1-SI-P-1A will NOT start automatically.

C. Vital Bus I has lost power.
If an SI signal is generated, "A" train SI can NOT be reset from the MCR or ESGR.

D. Vital Bus I, BKR 28 has tripped open.
If an SI signal is generated, "A" train SI can ONLY be reset from the ESGR.

K/A
013 Engineered Safety Features Actuation
K2.01 Knowledge of bus power supplies to the following: ESFAS/safeguards equipment control (CFR 41.7)

Notes

Need facility to verify nomenclature/location(s) of Safeguards Panel(s) indication(s).
Can change AC Vital Bus distractors to DC Cabinet 1-2, breaker 17, and "B" train DC if desired.

Answer Option Analysis

A. Incorrect. Diagnosis is incorrect. The unit would not remain at power following a loss of DC Bus 1-1 because train A reactor trip relays and the "A" reactor trip breaker UV coils would lose power. Consequence is correct IF DC Bus 1-1 had lost power. Plausible because stem implies a DC bus problem and applicant may believe that only a loss of DC Bus 1A causes a reactor trip.

B. Correct per 1D-B8.

C. Incorrect. Incorrect diagnosis. Vital Bus I is an AC bus. Incorrect consequence. IF Vital Bus I had deenergized, "A" train SI could be reset from the ESGR per 1-AP-10.01, Attachment 2. Plausible if applicant does not understand power supplies and cabinet indications for SFGDS.

D. Incorrect. Incorrect diagnosis. Vital Bus I is an AC bus. Consequence is correct IF Vital Bus I, bkr 28 were tripped. Plausible if applicant does not understand power supplies and cabinet indications for SFGDS.

References

1D-B8, Rev. 0, SFGDS DC CONT PWR TRBL
1-AP-10.06, Rev. 9, Loss of DC Power
ND-90.3-LP-6, 125 VDC Distribution
1-AP-10.01, Rev. 15, Loss of Vital Bus I

Modified stem to indicate NOT LIT vice DARK for clarity.

Modified answer 'A' and 'B' to remove double negatives in each item.

RFA Accepted 1/10/06

RFA accept 12/20/05

Changed light indication to correspond to actual plant equipment line-up. These lights are normally lit. Corrected name of light per plant.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B D C A C C A B A D Scramble Range: A - D

Tier:	2	Group:	1
Key Word(s):	ESF POWER SUPPLY	Cog Level:	C/A3.6
Source:	N	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

19. 016A2.04 002/2/2/SEMI-VITAL BUS/C/A2.5/N/SR06301/R/FJE

Unit 1 status is as follows:

- Stable at 30% power for a Chemistry Hold Point per 1-GOP-1.5, UNIT STARTUP, 2% REACTOR POWER TO MAX ALLOWABLE POWER.
- ALL Condensate Pumps RUNNING.

The Unit 1 RO reports the following:

- SG PORV controllers have shifted from LOCAL to REMOTE
- BOTH Reactor Trip Breakers are CLOSED and ENERGIZED
- Loss of Unit No. 1 Load Megawatts chart recorder
- Loss of all RCP CC flow and temperature indications

You observe the following:

- ALL Emergency AND Station Service Busses remain ENERGIZED.
- ALL SG levels are trending DOWN
- Feed flow is LESS THAN steam flow

Which ONE of the following is the correct action to take for the given plant conditions?

A.✓ Reduce turbine load

B. Trip the Unit 1 reactor

C. Verify MD AFW pumps - RUNNING

D. Verify Turbine Trip

K/A
016 Non-nuclear Instrumentation
A2.04 Ability to (a) predict the impacts of the following malfunctions or operations on the NNIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Voltage to instruments, both too high and too low (CFR 41.5)

Notes

Answer Option Analysis

A. Correct. RO reports entry conditions for 1-AP-10.05. SRO observes inadequate feed flow, per step 2. Step 2 RNO is to reduce turbine load (due to condensate and feedwater pump flow control valves failing open and flow control valves to condenser air ejectors failing closed).
B. Incorrect. This action would be taken for loss of a Vital Bus per 1-AP-10.02-.04, if RCP parameters were abnormal - OR - for inadequate feedwater flow above 85% power with fewer than 3 condensate pumps running per 1-AP-10.05.
C. Incorrect. This action would be taken for loss of either 4160V Emergency Bus per 1-AP-10.07.
D. Incorrect. This action would be taken for loss of DC Power per 1-AP-10.06 (following verification of one rx trip bkr open).

References

1-AP-10.05, Rev. 19, Loss of Semi-Vital Bus
ND-90.3-LP-5, Rev. 13, Objectives E. and F.
Removed first bullet due to redundant power supplies for CERPI and added an additional item that would occur during a loss of SV power.

Corrected answer description 'B'. AP-10.01 does not require a Rx Trip.

RFA accept 12/20/05

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A D D C A B B A C C Scramble Range: A - D

Tier:	2	Group:	2
Key Word(s):	SEMI-VITAL BUS	Cog Level:	C/A2.5
Source:	N	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

20. 017A3.01 003/2/2/CETC SUBCOOLING/C/A3.6/N/SR06301/R/FJE

Unit 1 conditions are as follows:

- A Small Break LOCA is in progress
- HHSI flow is ONLY 200 gpm due to a failure of 1-CH-P-1A and 1-CH-P-1B
- Containment pressure is 18 psia
- Loop C (PT-402) wide range RCS pressure: 1600 psig
- ICCM Train A CETCs are reading as per the attached

ASME Steam Tables are provided for reference.

ICCM Train A Subcooled Margin should indicate __(1)__ and 1G-B1, APPROACH TO SATURATION TEMP ALARM, indicacting light should be __(2)__.

	(1)	(2)
A.	34 °F	LIT
B.✓	34 °F	NOT LIT
C.	25 °F	LIT
D.	25 °F	NOT LIT

017A301.xls

Quad I		Quad II		QUAD III		QUAD IV	
LOC	°F	LOC	°F	LOC	°F	LOC	°F
B07	568	B10	567	H01	40	J12	568
				H04	568	J15	568
D03	40	C08	570				
D05	568			J03	571	L08	571
		D12	567	J07	573	L11	570
F04	569						
F06	2300	E11	569	L05	570	N10	570
G07	573	F09	572	N07	570	R08	569
		F13	570				
		H11	571				

K/A
017 In-core Temperature Monitor
A3.01 Ability to monitor automatic operation of the ITM system including: Indications of normal, natural, and interrupted circulation of RCS (CFR 41.7)

Notes

Need to verify P/T curve used by ICCM is saturation curve (w/ no margin).
Need to see if there is an Ops surveillance on ICCM that can be used to provide necessary data.
Can choose lower pressures if wider spread between correct/incorrect subcooling values is desired.

Attachments

ICCS Train A CETC Quadrant I/II/III/IV thermocouple page.

Answer Option Analysis

Five hottest CETCs are (G07, J07) = 573, F09 = 572 , (H11, L08, J03) = 571. Average of 5 highest valid CETCs = 572 F. Loop C WR pressure is input to Train A ICCS. Tsat for 1600 psig = 1615 psia = 606 F. Difference between Tsat and ave of 5 highest CETC = subcooling margin = 34 F. This value is greater than alarm setpoint of 30 F, so alarm is NOT lit.

CETC values of 2300 F and 40 F should be disregarded as invalid.

Tsat for 1500 psig = 1515 psia = 597 F, resulting in a subcooling margin of 25 F if Loop B WR is incorrectly used as input to Train A ICCS.

RCP trip criteria is if subcooling is less than 30 degrees and CH flow indicated to the core. It is plausible that due to the degraded HHSI system the applicants may determine that SI flow is not adequate to allow for tripping the RCPs.

A. Incorrect. Correct value of subcooling margin, incorrect alarm status, and incorrect RCP trip.
B. Correct Answer. Correct value of subcooling margin. Correct alarm status and correct RCP trip.
C. Incorrect. Incorrect value of subcooling margin. Incorrect alarm status and RCP trip.
D. Incorrect. Incorrect value of subcooling margin. Correct alarm status and correct RCP trip.

References

ND-93.4-LP-3, Rev. 10
1G-B1, Rev. 0, Approach to Saturation Alarm
1967 ASME Steam Tables

Changed Dark to NOT LIT for clarity.

Corrected subcooling information for psia vice psig.

Corrected answer B reason.

Added SB LOCA vice LOCA to stem and HHSI flow vavle and HHSI pump status to stem. This was done to increase the difficulty level of the RCP trip criteria determination.

Added third column to test the applicants knowledge concerning required actions during a loss of subcooling during a SBLOCA. This is an important knowledge item since if the RCPs are not tripped in a timely manner a greater amount of core uncovery could occur, thus degrading the subcooling margin (potentially superheating the core).

Deleted third column regarding RCP trip criteria. Deleted Loop B pressure from stem. RFA accept 12/20/05

Removed repeated stem.

RFA Accepted 1/10/06

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B C A C D A A C D D Scramble Range: A - D

Tier:	2	Group:	2
Key Word(s):	CETC SUBCOOLING	Cog Level:	C/A3.6
Source:	N	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

21. 022K1.02 003/2/1/REMOTE MONITORING/MEM3.7/N/SR06301/R/FJE

The following conditions exist on Unit 1.

- A large break LOCA has ocurred in Unit 1 containment.
- Containment pressure is 30 psia and slowly increasing.

Which ONE of the following responses is correct for the Containment Ventilation/Cooling System?

A. ONLY Containment Air Recirculation Fans A & C will trip off

B.✓ ONLY Containment Air Recirculation Fans A & B will trip off

C. ONLY Containment Air Recirculation Fans B & C will trip off

D. ALL Containment Air Recirculation Fans will trip off

K/A
022 Containment Cooling
K1.02 Knowledge of the physical connections and/or cause-effect relationships between the CCS and the following systems: SEC/remote monitoring systems (CFR 41.2 to 41.9)

Notes

Modified ILO exam question 230 (CONT002)

Answer Option Analysis

A. Incorrect. 'C' Containment Air Recirculation Fan will not trip, as it is powered from station service, vice the emergency buses. Plausible if the applicants do not recall power supplies for the fans and if they recall seeing a HI HI CLS in the simulator switch location leads to this selection.

B. Correct.

C. Incorrect. 'C' Containment Air Recirculation Fan will not trip, as it is powered from station service, vice the emergency buses. Plausible if the applicants do not recall power supplies for the fans.

D. Incorrect. C' Containment Air Recirculation Fan will not trip, as it is powered from station service, vice the emergency buses. Plausible if the applicants do not recall power supplies for the fans or believe that all CARFs trip on a HI HI CLS to protect the fans.

References

ND-88.4-LP-6, Rev. 7, Containment Ventillation (pages 4 and 5)

K/A match.

RFA accept 12/20/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	B C D B B D A C C D	Scramble Range: A - D	
Tier:	2		Group:	1		
Key Word(s):	REMOTE MONITORING		Cog Level:	MEM3.7		
Source:	N		Exam:	SR06301		
Test:	R		Author / Reviewer:	FJE		

22. 024AK2.01 003/1/2/EMERGENCY BORATE/C/A2.7/N/SR06301/R/FJE

Unit 1 conditions are as follows:
- Unit 1 was manually tripped due a rupture of the VCT.
- RCPs were manually secured due to a complete loss of VCT pressure.
- The crew is performing 1-ES-0.1, Reactor Trip Response.
- Two control rods did not fully insert.
- The crew is attempting to perform an Emergency Boration of the Unit 1 RCS.
- The "A" CHG pump is running and providing 110 gpm charging flow.
- The in-service BATP is in FAST.
- 1-CH-MOV-1350 will not open from the MCR or locally.
- 1-CH-FCV-1113A failed closed and CANNOT be opened from the MCR.

Which ONE of the following valve manipulations, alone, will provide borated water to the Unit 1 RCS?

A. Locally opening 1-CH-FCV-1113A, Boric Acid to Blender Valve

B. Locally opening 1-CH-228, Manual Emergency Borate Valve

C✓ Manually opening 1-CH-MOV-1115B, Chg Pump Suct from RWST

D. Manually opening 1-SI-MOV-1862A, LHSI Pump 'A' Suction from RWST

K/A
024 Emergency Boration
AK2.01 Knowledge of the interrelations between the Emergency Boration and the following: Valves (CFR 41.7)

Notes

Answer Option Analysis

A. Incorrect. Opening 1-CH-FCV-1113A by itself, for the given conditions, will not provide a boration flowpath. 1-CH-228 would also have to be opened. Plausible because 1-CH-FCV-1113A is a valve in an alternate emergency boration flowpath.

B. Incorrect. Opening 1-CH-228 by itself, for the given conditions, will not provide a boration flowpath. 1-CH-FCV-1113A would also have to be opened. Plausible because 1-CH-228 is a valve in an alternate emergency boration flowpath.

C. Correct. Opening 1-CH-MOV-1115B (or 1-CH-MOV-1115D) aligns the RWST to the suction of the running charging pump.

D. Incorrect. Since no SI has occurred, the LHSI pumps are not running and will not supply RWST water to the operating charging pump. Plausible because these valves are in a boration flowpath to the RCS (during an SI).

References

Licensee examination bank question number CH00001
1-AP-3.00, Rev. 2, Emergency Boration
1-ES-0.1, Reactor Trip Response, Attachment 1
ND-88.3-LP-9, Rev. 14, Blender Control Subsystem

Added bullet for 1-CH-FCV-1113A to be failed closed. This is a normally open valve, so if it is not failed closed, then both 'B' and 'C' are correct answers. Adding the bullet, does not detract from the difficulty of the question.

Added noun names to valves for clarity.

Added the words "by itself" for clarity.

In answer 'A' changed manually to locally due to addition of bullet in question stem.

RFA accept 12/20/05

Answer 'D' was a correct answer. Changed 1-SI-MOV-1863A/B to SI-MOV-1862A/B. Opening these valves alone will not provide a boration path; however, the valve name may cause some applicants to believe that a suction source from the RWST will cause a boration. Similar nomenclature to correct answer.

Added VCT information to allow 'C' to be the correct answer. At normal VCT pressure, the RWST does not have enough head to provide flow.

RFA Accepted 1/10/06 - changed "by itself" to "alone", removed 'B' LHSI information from choice 'D'

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C A B D D A B A A D Scramble Range: A - D

Tier:	1	Group:	2
Key Word(s):	EMERGENCY BORATE	Cog Level:	C/A2.7
Source:	N	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

23. 025AK2.03 002/1/1/RHR/C/A2.7/N/SR06301/R/FJE

Plant conditions are as follows:

- Unit 1 is in Intermediate Shutdown
- 1B RHR Pump is running with the 1B RHR Heat Exchanger in service
- 1-CC-P-1B ('B' CC Pump) is running.
- The CC system is NOT crosstied between units.

- **Unit 2:** 1-CC-P-1C ('C' CC Pump) is out of service for post maintenance testing.

IF the 1B CC Pump trips and operators are unable to start the 1A CC pump, which ONE of the following procedures should be implemented in order to restore unit 1 decay heat removal capability?

A. 1-AP-15.00, Loss of Component Cooling.

B.✓ 1-AP-27.00, Loss of Decay Heat Removal Capability.

C. 1-ES-0.2, Natural Circulation Cooldown

D. 1-FR-C.3, Response to Saturated Core Cooling.

K/A
025 Loss of RHR System
AK2.03 Knowledge of the interrelations between the Loss of Residual Heat Removal System and the following: Service water or closed cooling water pumps (CFR 41.7).

Notes

Answer Option Analysis

With no running CC pumps, decay heat removal on Unit 1 is lost even though RH pumps remain in operation. CC systems cannot be crosstied per 1K-E7, CC PPS DISCH HDR LO PRESS, due to only one U2 CC pump available.

A. Incorrect. Plausible because equipment failure is a CC malfunction. 1-AP-15.00 would not be effective at restoring decay heat removal; the cross connection in Attachment 2 merely transfers some non-RHR heat loads from U1 to U2, delaying the U1 RCS heatup.

B. Correct. Attachments to 1-AP-27.00 contain steps for cooling the RCS that do not rely on using the CC system as a heat sink.

C. Incorrect. Plausible because Attachment 4 of 1-AP-27.00 is titled Natural Circulation Cooling. Neither loss of CC or RH are entry conditions for 1-ES-0.2, which was designed to be performed with no accident in progress.

D. Incorrect. Plausible because RCS is above 200 F and subcooling will decrease as RCS temperature increases due to loss of decay heat removal. Step 1 of 1-FR-C.3 checks if the RHR system has been placed in service, and if so, directs a transition to 1-AP-27.00.

References

1K-E7, Rev. 0, CC PPS DISCH HDR LO PRESS
1-AP-15.00, Rev. 3, LOSS OF COMPONENT COOLING
1-AP-27.00, Rev. 12, LOSS OF DECAY HEAT REMOVAL CAPABILITY
1-ES-0.2, Rev. 16, NATURAL CIRCULATION COOLDOWN
1-FR-C.3, Rev. 9, RESPONSE TO SATURATED CORE COOLING

Mark number adjustment for 1-CC-P-1B.

Added a statement that CC is NOT crosstied between units. This was added since CC is normally crosstied and one would assume that the B and D pumps are running, so with the loss of the B pump (depending on heat load) a loss of decay heat removal may not occur.

Corrected mark number for 1-CC-P-1C. The 2A pump in CC pumps does not exist.

RFA accept 12/20/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	B A C C A C A B D D	Scramble Range: A - D	
Tier:	1		Group:	1		
Key Word(s):	RHR		Cog Level:	C/A2.7		
Source:	N		Exam:	SR06301		
Test:	R		Author / Reviewer:	FJE		

24. 026A2.08 002/2/1/SECURE CS/MEM3.2/N/SR06301/R/FJE

Unit 1 conditions are as follows:

- Unit 1 experienced a large break LOCA.
- ALL ESF systems responded as designed.
- Containment pressure peaked at 27 psia and is trending DOWN.
- The crew has completed 1-E-0, Reactor Trip or Safety Injection, and has transitioned to 1-E-1, Loss of Reactor or Secondary Coolant

Which ONE of the following is the correct MAXIMUM pressure at which the Containment Spray pumps can be secured in accordance with Emergency Operating Procedures?

A. 22.9 psia

B. 17.7 psia

C. 13 psia

D✓ 11.9 psia

K/A
026 Containment Spray
A2.08 Ability to (a) predict the impacts of the following malfunctions or operations on the CCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Safe securing of containment spray (when it can be done) (CFR 41.5)

Notes

Answer Option Analysis

A. Incorrect. Plausible because this is less than the pressure at which CS is required (23 psia).

B. Incorrect. Plausible because 1-E-0 step 12 requires verification of valve alignments if CTMT pressure exceeds 17.7 psia.

C. Incorrect. Plausible because 1-E-1 step 12.g) directs operation of ISRS pumps to maintain CTMT pressure between 10 psia and 13 psia after stopping CS pumps.

D. Correct. 1-E-1 step 12 directs stopping CS pumps when CTMT pressure is "less than 12 psia".

References

1-E-0, Rev. 52, Reactor Trip or Safety Injection
1-E-1, Rev. 24, Loss of Reactor or Secondary Coolant
Updated procedure revisions.

Actual transistion would be E-0, to FR-Z.1 to E-1.

RFA accept 12/20/05

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D A C D C B A B A D Scramble Range: A - D

Tier:	2	Group:	1
Key Word(s):	SECURE CS	Cog Level:	MEM3.2
Source:	N	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

25. 026AG2.4.4 002/1/1/COMPONENT COOLING/C/A4.0/N/SR06301/R/FJE

Unit 1 plant conditions are as follows:

- Unit 1 has been at 100% power for 28 days.
- 1C-A2, RCP 1A THERMAL BARRIER CC HI FLOW is LIT
- 1C-A3, RCP 1A THERMAL BARRIER CC HI TEMP is LIT
- 1C-A4, RCP 1A SEAL LEAKOFF HI FLOW is LIT
- 1-CC-TV-120A, Thermal Barrier Isolation Valve, CLOSED automatically.
- ALL other plant AND RCP parameters are NORMAL AND STABLE.

Which ONE of the following procedures should the crew implement NEXT to address the current plant conditions?

A.✓ 1-AP-9.00, RCP ABNORMAL CONDITIONS

B. 1-AP-9.02, LOSS OF RCP SEAL COOLING

C. 1-AP-15.00, LOSS OF COMPONENT COOLING

D. 1-AP-8.00, LOSS OF NORMAL CHARGING FLOW

K/A
026 Loss of Component Cooling Water
AG2.4.4 Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures (CFR 41.10).

Notes

Answer Option Analysis

A. Correct. The three alarms together indicate failure of the controlled leakage seal (floating ring seal). High RCP seal leakoff, as indicated by 1C-A4, is an entry condition for 1-AP-9.00.

B. Incorrect. Plausible if applicant interprets plant conditions as indicating a loss of BOTH seal injection AND thermal barrier cooling. Although 1-CC-TV-120 A is shut, seal injection has been maintained(all other plant and RCP parameters are normal and stable).

C. Incorrect. Plausible if applicant interprets plant conditions as indicating a loss of CC flow (all other plant and RCP parameters are normal and stable).

D. Incorrect. AP-8.00 entry conditions are not met and would not aid in this event.

References

1C-A2, Rev. 5, RCP 1A THERMAL BARRIER CC HI FLOW
1C-A3, Rev. 0, RCP 1A THERMAL BARRIER CC HI TEMP
1C-A4, Rev. 4, RCP 1A SEAL LEAKOFF HI FLOW
1-AP-9.00, Rev. 21, RCP ABNORMAL CONDITIONS
ND-88.5-LP-1, Rev. 21, Component Cooling Water System

Updated procedure revision and Answer Option Analysis for item 'D'.

RFA accept 12/20/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	A D A D B A D D B D	Scramble Range: A - D

Tier:	1	Group:	1
Key Word(s):	COMPONENT COOLING	Cog Level:	C/A4.0
Source:	N	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

26. 027AK2.03 003/1/1/PRESSURE CONTROL/MEM2.6/M/SR06301/R/FJE

Given the following plant conditions on Unit 1:

- The reactor is at full power.
- The RCS is at normal operating pressure.
- The Pressurizer Pressure Master Controller potentiometer is set at 6.68.

What is the FIRST automatic response that will occur if the SETPOINT on the Pressurizer Pressure Master Controller is VERY SLOWLY raised from its current value to 10.0?

A. PRZR spray valves open.

B. PORV-1455C opens.

C.✓ PORV-1456 opens.

D. A PRZR high pressure reactor trip occurs.

K/A
027 Pressurizer Pressure Control Malfunction
AK2.03 Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following: Controllers and positioners (CFR 41.7).

Notes

Modified from ILT bank question regarding first effects of a step change of PPMC setpoint.

Answer Option Analysis

The PPMC compares the input signal from PT-444 (actual pressurizer pressure) against the reference setpoint (pot setting). Raising the PPMC pot increases the reference pressure setpoint for the pressure control system, lowering controller output, which causes the spray valves to shut and the proportional heaters to energize to raise pressure to the setpoint. As the pot is continually raised, pressure will continue to increase and controller output will remain at or below the normal output of 30% (steady state, 2235 psig). Pressure will increase until PT-445 senses pressure at the PCV-1456 (PORV) setpoint. Since the other PORV, PCV-1455C, is driven by the output of the PPMC, it will not open.

A. Incorrect. Plausible because this is the normal system response to an increase in pressure. Incorrect because sprays and heaters are driven by the output of the PPMC, which remains below 30%. Sprays close and heaters energize in order to raise pressure to the (increasing) setpoint.

B. Incorrect. Plausible if the applicant does not know which PORV is driven from the PPMC.

C. Correct. Only PORV-1456 opens because it is independent of PPMC output and opens based on actual pressure sensed by PT-445.

D. Incorrect. Plausible if applicant does not think either PORV will open before reacing the trip setpoint.

References

ILT # PPC0017
ND-93.3-LP-5, Rev. 11, Pressurizer Pressure Control

Adjusted MPC normal output to 30% in answer option analysis.

RFA accept 12/20/05

Added the word setpoint to the stem for clarity.

RFA Accepted 1/10/06

MCS Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
		Answer:	C B C D D D D A A D	Scramble Range: A - D	
Tier:	1	Group:	1		
Key Word(s):	PRESSURE CONTROL	Cog Level:	MEM2.6		
Source:	M	Exam:	SR06301		
Test:	R	Author / Reviewer:	FJE		

27. 028AA1.04 003/1/2/LEVEL CONTROL/C/A2.7/N/SR06301/R/FJE

Unit 1 conditions are as follows:

- Unit 1 is operating at 100 % steady state power.
- All control systems are in AUTOMATIC.
- The pressurizer level upper control channel has failed to 22% pressurizer level.
- Assume NO operator action has occurred.

Immediately following failure of the pressurizer level upper control channel, actual charging flow will be ______ than before the failure and Regenerative Heat Exchanger Letdown Line Temperature will go ____.

A. LOWER DOWN

B. LOWER UP

C✓ HIGHER DOWN

D. HIGHER UP

K/A
028 Pressurizer Level Malfunction
AA1.04 Ability to operate and / or monitor the following as they apply to the Pressurizer Level Control Malfunction: Regenerative heat exchanger and temperature limits (CFR 41.7)

Notes

Verify response with facility.

Answer Option Analysis

The output of the pzr level program is an input to the pzr level controller. The pzr level controller compares programmed level with actual level and generates an error signal proportional to the level difference. The error signal is directed through the flow limit summator (limits max/min flow to 115gpm/25gpm with charging flow controller in auto) to the charging flow controller. The charging flow controller compares actual charging flow to the control signal and develops an error signal for positioning the normal charging flow control valve.

A. Incorrect. Incorrect charging flow trend even though letdown temperature trend is correct. Plausible if applicant does not understand operation (input/output) of pressurizer level controller.

B. Incorrect. Incorrect charging flow trend, incorrect letdown temperature trend.

C. Correct. Program level (failed high) is above actual level and charging flow will increase (up to the high flow limit of 115 gpm). Letdown is cooled by charging flow in the regenerative heat exchanger. Increased charging flow causes increased cooling of letdown flow, resulting in a decrease in letdown line temperature.

D. Incorrect. Correct charging flow trend, but incorrect letdown line temperature trend. Plausible if applicant does not understand physical configuration of charging / letdown or cause/effect in a regenerative heat exchanger.

References

ND-93.3-LP-7, Rev. 7, Pressurizer Level Control System
ND-88.3-LP-2, Rev. 12, Charging and Letdown
Removed information about the Pressurizer Level Program. When at 100% power the program will send a maximum value of 53.7%, so a failure high will result in no change. If the upper channel fails low, the response is as expected by the answers.

RFA accept 12/20/05

Original failure would cause letdown to isolate. New value will prevent that event, while still testing the knowledge of the impact on letdown line temperature.

RFA Accepted 1/10/06

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	C C A A B B A A D D	Scramble Range: A - D	
Tier:	1			Group:	2	
Key Word(s):	LEVEL CONTROL			Cog Level:	C/A2.7	
Source:	N			Exam:	SR06301	
Test:	R			Author / Reviewer:	FJE	

28. 029K1.04 002/2/2/CONTAINMENT PURGE/C/A3.0/N/SR06301/R/FJE

Plant status is as follows:

Unit 1 is in REFUELING SHUTDOWN
Unit 1 CTMT purge is on filtered exhaust IAW 1-OP-VS-001

Unit 2 just experienced a spurious reactor trip and safety injection from 100% power.

1-VS-F-58A and 1-VS-F-58B should be _______ and the **Unit 1** containment purge supply MOVs (1-MOV-100A/B/C/D) should be _______.

Which ONE of the following is correct?

A.✓	RUNNING	OPEN
B.	RUNNING	CLOSED
C.	TRIPPED	OPEN
D.	TRIPPED	CLOSED

K/A
029 Containment Purge
K1.04 Knowledge of the physical connections and/or cause-effect relationships between the Containment Purge System and the following systems: Purge system (CFR 41.2 to 41.9)

Notes

Need to verify system operation using operating procedures (1-OP-VS-001?)
Can change question to test other fans, e.g. F58A/B, F59, but need to check fan interlocks.

Answer Option Analysis

A. Correct. SI on either unit will start the 58 fans. Additionally 1-MOV-100A/B/C/D shut on high radiation on 1-RM-159/160 (Containment Particulate / Gaseous RM), not on SI, so they will remain open.

B. Incorrect. See A. Additionally, 1-MOV-100A/B/C/D shut on high radiation on 1-RM-159/160 (Containment Particulate / Gaseous Radiation Monitors).

C. Incorrect. A SI on either unit will not trip the 58 Fans. 1-MOV-100A/B/C/D remain open since Unit 1 has NOT experienced a high alarm on either 1-RM-159 or 1-RM-160.

D. Incorrect.

References

ND-88.4-LP-6, Rev. 7, Containment Ventilation

Changed question to 58A/B fan, based on note section from NRC and the 4A/B fans are not run.

RFA accept 12/20/05

Changed correct answer to 'A'

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A B A D D B B B D B Scramble Range: A - D

Tier:	2	Group:	2
Key Word(s):	CONTAINMENT PURGE	Cog Level:	C/A3.0
Source:	N	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

29. 033G2.4.31 004/2/2/FUEL POOL COOLING/C/A3.3/N/SR06301/R/FJE

Plant conditions are as follows:

- Current outside temperature is 85 °F.
- Unit 1 has just completed core off-load.
- 1-FC-P-1A is running with a normal discharge pressure.
- Spent Fuel Pit HX A (1-FC-E-1A) is in service with normal CC outlet flow.
- 0-VSP-A4, Spent Fuel Pit High Temp, alarmed 5 minutes ago.
- TI-FC-103, SPENT FUEL PIT TEMP, currently reads 116 °F and slowly going UP.

Which ONE of the following is the correct action(s) to take to address plant conditions in accordance with 0-VSP-A4, SPENT FUEL PIT TEMP?

A. Throttle open on 2-CW-MOV-200B and 2-CW-MOV-200D.

B. Align the Chilled CC system for service on the spent fuel pit.

C✓ Place FC HX B in service.

D. Check Panel 1ABD1A, Breaker 15 - CLOSED.

K/A
033 Spent Fuel Pool Cooling
G2.4.31 Knowledge of annunciators alarms and indications, and use of the response instructions (CFR 41.10)

Notes

Need facility to provide normal value for pump discharge pressure.

Answer Option Analysis

A. Incorrect. Applicants may believe that opening the CW MOVs will provide additional cooling flow to the CCHXs. Incorrect, as SW to the CCHXs comes from the B/D CW lines on Unit 1 (not Unit 2).

B. Incorrect. Applicants may believe that chilled CC system can be aligned to the SFP. As this system is normally used during hotter months to maintain various CC cooled components at a lower temperature (e.g., Neutron Shield Tank, PDT vent condenser cooler, SG Recirc and Transfer Cooler).

C. Correct. Per 0-VSP-A4, Step 2.e), if SFP temp is NOT stable or decreasing, RNO directs placing second HX in service.

D. Incorrect. This action is only correct for coincident SFP HI and LO level alarms (caused by loss of power).

References

0-VSP-A4, Rev. 4, SPENT FUEL PIT HI TEMP
0-VSP-C4, Rev. 3, SPENT FUEL PIT LO LVL
0-VSP-D4, Rev. 1, SPENT FUEL PIT HI LVL
0-OP-FC-002, Rev. 1, SWAPPING SFP COOLERS AND UNIT CC SFP COOLING

A reference should be provided to correspond with K/A. No per RFA.

Minor changes to stem.

RFA accept 12/20/05

Added a bullet to the stem to provide the applicants with a reason for a rise in SFP temperature. Without a reason for temperature increase, SFP temperature would not rise in a short time frame and the applicants will determine that a system fault must have occurred.

RFA Accepted 1/10/06

Modified/re-arranged stem bullets. Changed choice A/B.

RFA accepted 2/1/06.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C B A D D A A B B B Scramble Range: A - D

Tier:	2	Group:	2
Key Word(s):	FUEL POOL COOLING	Cog Level:	C/A3.3
Source:	N	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

30. 034K4.01 002/2/2/FUEL HANDLING/C/A2.6/N/SR06301/R/FJE

Which ONE of the following design features/interlocks or operating practices prevents dropping a fuel assembly?

A. During fuel movement the safety knob on the new fuel handling tool is pad locked into place to PREVENT gripper finger disengagement.

B. The auxiliary hoist on the manipulator crane AUTOMATICALLY stops when using the RCCA unlatching tool if excess weight is sensed.

C. The hoist-gripper INTERLOCK can be bypassed to allow for disengagement of a fully seated fuel assembly with authorization from the Shift Manager.

D.✓ The manipulator crane gripper will NOT disengage unless the gripper is on top of the fuel assembly and a downward force of approximately 450 pounds exists.

K/A
034 Fuel Handling Equipment
K4.01 Knowledge of design feature(s) and/or interlock(s) which provide the following: Fuel protection from binding and dropping (CFR 41.7)

Notes

Answer Option Analysis

A. Incorrect. Although the safety knob is used to prevent fuel dropping, it is not pad locked into place, it is screwed into place.

B. Incorrect. The auxiallary hoist is routinely used during fuel movement to seperate the RCCA and the fuel assembly. Plausible since excess weight of fuel assembly and CRDM could cause the fuel assembly to drop.

C. Incorrect. Plausible but the refueling SRO (not SM) has this authority.

D. Correct. Manipulator crane interlock to prevent dropping a fuel assembly. Fuel assembly can only be released when ~450 lbs downward force exist.

References
ND-92.5-LP-3, Rev. 7, Fuel Handling Tools
0-OP-4.6, Rev. 5, Rod Cluster Control Assembly (RCCA) Change Tool
1-OP-FH-015, Rev. 6, Manipulator Crane
New question due to high difficulty level. This question is still difficult.

RFA accept 12/20/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	D D A C B D A A A A	Scramble Range: A - D
Tier:	2		Group:	2	
Key Word(s):	FUEL HANDLING		Cog Level:	C/A2.6	
Source:	N		Exam:	SR06301	
Test:	R		Author / Reviewer:	FJE	

31. 035K3.01 002/2/2/MAIN STEAM VALVE RCS/MEM4.4/M/SR06301/R/FJE

Which ONE of the following describes the plant response if one Main Steam Trip Valve is inadvertently closed with the plant at 50% load? (Assume all controls are in AUTOMATIC and that NO reactor trip occurs.)

AFFECTED loop **Delta T** will go ____.

UNAFFECTED loop **Tave** will go ____.

A.✓ DOWN

DOWN

B. UP

DOWN

C. DOWN

UP

D. UP

UP

K/A
035 Steam Generator
K3.01 Knowledge of the effect that a loss or malfunction of the S/GS will have on the following: RCS (CFR 41.7)

Notes

Answer Option Analysis

Closure of one MSTV with the reactor at a constant power will result in less heat removal from the affected loop and more heat removal from the unaffected loops. With a constant steam generator pressure (Tsat / loopThot), affected loop Tc will go UP, decreasing affected loop Delta T. Unaffected loop Tc will go down, decreasing unaffected loop Tave.

A. Correct for both affected and unaffected loops.
B. Incorrrect. Incorrect for affected loop, although correct for unaffected loop.
C. Incorrect. Although correct for affected loop, incorrect for unaffected loop.
D. Incorrect. Incorrect for both affected and unaffected loops.

References

ILT Question Bank ID: TAA0018, TAA0044

No Changes.

RFA accept 12/20/05

Adjusted stem for clarification.

RFA Accepted 1/10/06

Bolded and underlined items being tested.

RFA accepted 1/19/06

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	A B B A C C A B D D	Scramble Range: A - D
Tier:	2		Group:	2	
Key Word(s):	MAIN STEAM VALVE RCS		Cog Level:	MEM4.4	
Source:	M		Exam:	SR06301	
Test:	R		Author / Reviewer:	FJE	

32. 037AK1.01 003/1/2/STEAM TABLE/C/A2.9/N/SR06301/R/FJE

Unit 1 conditions are as follows:

- 1-ECA-3.1, SGTR With Loss of Reactor Coolant - Subcooled Recovery, is in progress at Step 28.c).
- Containment pressure is 14 psia.
- Containment radiation is 1.0E2 R/hr.
- Both trains of ICCM subcooled margin are inoperable.
- The average of the five highest CETCs is 540°F.

Assuming PRZR level remains less than 69%, which ONE of the following is the MAXIMUM pressure at which the crew can stop the RCS depressurization?

Reference provided.

A. 400 psig

B. 665 psig

C✓ 1311 psig

D. 1985 psig

K/A
037 Steam Generator Tube Leak
AK1.01 Knowledge of the operational implications of the following concepts as they apply to Steam Generator Tube Leak: Use of steam tables (CFR 41.8 / 41.10)

Notes

Modified from bank FA03301

Attachments

1-ECA-3.1, Rev. 28, SGTR With Loss of Reactor Coolant, Subcooled Recovery, Step 28 (pg. 21).
Steam Tables

Answer Option Analysis

540 CETC + 40 max. subcooling margin (normal containment parameters) = 580 ^{o}F. Psat (580 ^{o}F) = 1326.17 psia + 14.7 = 1340 psig

A. Incorrect. This is approximately Psat (540 - 95). Plausible if applicant subtracts adverse subcooling margin to determine P.

B. Incorrect. This is approximately Psat (540 - 40). Plausible if applicant subtracts subcooling margin to determine P.

C. Correct.

D. Incorrect. This is approximately Psat (540 + 95). Plausible if applicant uses adverse subcooling margin.

References

1-ECA-3.1, Rev. 28, SGTR With Loss of Reactor Coolant - Subcooled Recovery
Updated procedure revision.

Added "Reference provided."

RFA accept 12/20/05

Corrected psia/psig error.

RFA Accepted 1/10/06

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	C A D B D A B B C A	Scramble Range: A - D

Tier:	1	Group:	2
Key Word(s):	STEAM TABLE	Cog Level:	C/A2.9
Source:	N	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

33. 039K5.01 004/2/1/WATER HAMMER/MEM2.9/N/SR06301/R/FJE

Which two (2) of the following four (4) conditions are MOST LIKELY to result in water hammer?

1. Overfilling a steam generator (YELLOW path condition for secondary inventory).
2. Rapidly heating up secondary piping using the MS trip valve bypass valves.
3. Reinitiating feedwater to a steam generator shortly after the feedring is uncovered.
4. Throttling the Condensate Pump discharge valve before securing the Condensate Pump.

A.✓ Conditions 1 and 2

B. Conditions 1 and 4

C. Conditions 2 and 3

D. Conditions 3 and 4

K/A
039 Main and Reheat Steam
K5.01 Knowledge of the operational implications of the following concepts as the apply to the MRSS: Definition and causes of steam/water hammer (CFR 41.5)

Notes

Need facility to verify condition 4 (HP Drain Pump) is incorrect.

Answer Option Analysis
1. Correct per ND-95.3-LP-43 and ND-83-LP-10, Rev. 11, Applications of Fluid Phenomena.
2. Correct per Caution statements in 1-OP-FW-001 and -002.
3. Incorrect. Per ND-88.1-LP-4, the main feed ring bottom discharge holes are plugged and the ring is outfitted with J-tubes in order to prevent quick drainage of the feedring and minimize the chance of water hammer.
4. Incorrect.

A. Correct.
B. Incorrect. One correct condition (1), one incorrect condition (4).
C. Incorrect. One correct condition (2), one incorrect condition (3).
D. Incorrect. Two incorrect conditions (3 and 4).

References

1-OP-FW-001, Rev. 9, Motor Driven AFW Pumps Startup and Shutdown
1-OP-FW-002, Rev. 14, Turbine Driven AFW Pump Startup and Shutdown
Facility examination bank questions MS00010, MS00011, SD00001
ND-88.1-LP-4, Rev. 4, Steam Generators
ND-95.3-LP-43, Rev. 6, FR-H.3, Response to Steam Generator High Level

Added Turbine Driven to item #2, as the caution is based on flow instabilities with 1-FW-P-2, as per CTS 51108. Analysis was based on 1-FW-P-2 and carried over to the motor driven for conservatism. See PI and CTS information.

RFA accept 12/20/05

Adjusted item #2 to make it correspond to the procedural requirements.
Changed item number 4 from HP DRN pump to CN pump. There is a possibility of a water hammer when the HP DRN pump is secured (due to level control system diverting to condenser (at a vac)).

RFA Accepted 1/10/06 Item #2, will validate #4 on the simulator. Validated.

RFA Accepted 1/12/06

Changed item #2, based on plant issue database. Original item was technically incorrect.

RFA accepted 1/19/06

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A B C C C A B D D D Scramble Range: A - D

Tier:	2	Group:	1
Key Word(s):	WATER HAMMER	Cog Level:	MEM2.9
Source:	N	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

34. 045A4.02 002/2/2/GENERATOR BREAKERS/MEM2.7/M/SR06301/R/FJE

Unit 1 conditions are as follows:

- Unit 1 was manually tripped from 100% power due to lowering pressurizer pressure.
- The crew entered 1-E-0, Reactor Trip or Safety Injection.
- While performing 1-E-0, step 2, Verify Turbine Trip, an operator noticed that the Main Generator failed to trip.

Which ONE of the following correctly describes ALL of the IMMEDIATE action(s) to take per 1-E-0?

A. Manually OPEN the generator output breakers AND check that the voltage regulator has TRIPPED.

B. Manually OPEN the generator output breakers ONLY.

C. Place the excitation control switch in OFF AND check that the voltage regulator has TRIPPED.

D✓ Manually OPEN the generator output breakers AND place the excitation control switch in OFF.

K/A
045 Main Turbine Generator System
A4.02 Ability to manually operate and/or monitor in the control room: T/G controls, including breakers (CFR 41.7)

Notes

Modified from SR02301 045A3.11

Answer Option Analysis

A. Incorrect (second part). 1-E-0 step [2]d) RNO directs operator to "... AND place the EXCITATION control switch in OFF. Checking that the voltage regulator has tripped is not an immediate action.

B. Incorrect. (incomplete). 1-E-0 step [2]d) RNO directs operator to "... AND place the EXCITATION control switch in OFF.

C. Incorrect (incomplete). 1-E-0 step [2]d) RNO directs operator to "Manually open output breakers AND place the EXCITATION control switch in OFF." Checking that the voltage regulator has tripped is not an immediate action.

D. Correct.

References

1-E-0, Rev. 52, Reactor Trip or Safety Injection
SR02301 045A3.11
Procedure revision update.

RFA accept 12/20/05

MCS Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
		Answer:	D A D B D C D D C B	Scramble Range: A - D	
Tier:	2	Group:	2		
Key Word(s):	GENERATOR BREAKERS	Cog Level:	MEM2.7		
Source:	M	Exam:	SR06301		
Test:	R	Author / Reviewer:	FJE		

35. 054AK3.04 003/1/1/MAIN FEEDWATER/C/A4.4/N/SR06301/R/FJE

Unit 1 conditions are as follows:

- Unit 1 tripped on low-low steam generator level from 100% power due to a main feedwater line break in the Main Steam Valve House.
- 1-FR-H.1 has just been initiated. The crew is currently performing Step 2 (TRY TO ESTABLISH AFW FLOW TO AT LEAST ONE SG).
- NO Unit 1 MFW or AFW pumps are able to supply feedwater to Unit 1
- RCS pressure is 2155 psig .
- CETCs are 612 °F and stable.
- Containment pressure is 10 psia.
- WIDE RANGE SG levels are as follows:
  1A: 6%
  1B: 19%
  1C: 19%

Which ONE of the following is correct for the given plant conditions?

The crew must stop all RCPs ________________.

A. and immediately initiate RCS bleed and feed to prevent or minimize core uncovery.

B.✓ and establish greater than 350 gpm AFW flow from Unit 2 to prevent steam generator dryout.

C. and immediately depressurize the steam generators in preparation for establishing condenstate flow to prevent steam generator dryout.

D. and manually initiate safety injection to prevent saturated conditions in the core and subsequent core uncovery.

K/A
054 Loss of Main Feedwater
AK3.04 Knowledge of the reasons for the following response as they apply to the Loss of main Feedwater (MFW): Actions contained in EOPs for loss of MFW (CFR 41.5, 41.10)

Notes

Time requirement for restoration of AFW from U2 is not specified in 1-FR-H.1. Can eliminate time requirements from answers, potentially making question more difficult.

Answer Option Analysis

Question describes the limiting main feedwater line break in UFSAR 14B.6. Per UFSAR and ND-95.2-LP-3, operator action must be taken within 10 minutes of the start of the event to secure RCPs and align AFW cross-tie from the opposite unit in order to prevent steam generator dryout (one of two limiting cases - the other being RCS and MS overpressurization). These operator actions are decribed in 1-FR-H.1, Step 2.d) RNO

A. Incorrect. Incorrect action/correct reason. B&F criteria in 1-FR-H.1 are WR level in any 2 SGs less than 7% (22%) OR PRZR pressure greater than or equal to 2335 due to loss of heat sink. Plausible because reason is correct and applicant may fail to recognize that B&F criteria are NOT met.

B. Correct. Question stem states that NO U1 AFW pumps are able to supply U1 (AFW is not available per 1-FR-H.1 step 2.d).) RNO directs stopping all RCPs and maintain U1 AFW flow greater than 350 gpm using U2 AFW pump and cross-tie. See discussion above for reason.

C. Incorret. Action is incorrect per A. Time requirement is incorrect (per 1-FR-H.1 requirement is "IMMEDIATELY"). Reason for stopping RCP is correct. Plausible because reason for stopping RCP is correct and applicant may fail to recognize B&F critteria are NOT met AND that time requirement for B&F is Immediate.

D. Incorrect. Action is correct, but time requirement is incorrect AND reason is incorrect. Time requirement is within 10 minutes. Reason provided is for limiting restoration of flow to a hot, dry (WR less than 7%) SG with CETCs stable or decreasing.

References

1-FR-H.1, Rev. 20, RESPONSE TO LOSS OF SECONDARY HEAT SINK
ND-95.3-LP-41, Rev. 9, RESPONSE TO LOSS OF SECONDARY HEAT SINK
ND-95.2-LP-3, Rev. 7, OVER COOLING TRANSIENTS
Surry UFSAR, Rev. 32, 14B.6, TRANSIENT ANALYSIS OF A HIGH ENERGY LINE BREAK IN THE MAIN STEAM VALVE HOUSE

Modified question stem. Changed "FR-H.1 is in progress" to "1-FR-H.1 has just been initiated. The crew is currently performing Step 2 (TRY TO ESTABLISH AFW FLOW TO AT LEAST ONE SG). This was added since step 2 is not a continuous action step and the applicant may believe that step is completed already. This does not make any distractors more or less plausible, as the CAP is applicable and the applicant may still believe bleed and feed is required per the CAP.

Adusted 'B' and 'C' SG levels to 19% to minimize confusion that if at 9% that bleed and feed would occur within 10 minutes (due to low SG levels).

Adjusted 'C' distractor to add word "BOTH" to so that both actions are assigned the 10 minute criteria (corresponds to answer analysis).

Updated procedure revision.

Relocated "both" in distractor C. Minor editorial changes to all distractors.
RFA accept 12/20/05

Eliminated time items (10 minutes)as allowed by NRC notes from choice B and C. The time aspect was left in choice 'A' for plausibility (words are from the CAP).

Provided new choice 'D', as the reason could be argued as correct. Since CETC were at 612 and SGs are almost empty (heading to a hot/dry generator). New choice 'D' was developed. This new 'D' is plausible, as manually initiating Safety Injection will prevent core uncovery and is part of the Bleed and Feed criteria (actions are contained in FR-H.1 steps 11-18).

Provided a new choice 'C', as when time requirements were removed, this answer became techinically correct. Item 'C' is plausible as this is an action taken later in the procedure.

RFA Accepted 1/12/06

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	B D B C A C C A B B	Scramble Range: A - D	

Tier:	1	Group:	1
Key Word(s):	MAIN FEEDWATER	Cog Level:	C/A4.4
Source:	N	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

36. 055EA1.06 002/1/1/ECA-0.0/C/A4.1/N/SR06301/R/FJE

Plant conditions are as follows:

- A Loss of Off-site Power occurred 2 minutes ago.
- #1 EDG failed to start and no actions have been taken to start it.
- #2 EDG started and then tripped.
- #3 EDG loaded on **Unit 2**.
- The **Unit 1** crew is performing 1-ECA-0.0. Loss of All AC Power, Step 5, Try to Restore Power to Any AC Emergency Bus.

Which ONE of the following actions is the NEXT correct action for the **Unit 1** crew to perform, per 1-ECA-0.0, in order to restore power to Unit 1?

A. Place 25J3 in PTL, place Unit 2 Pnl 3-2 Switch 43-15J3 in BYP, turn Synch ACB-15J3 to ON, close ACB-15J3.

B. Initiate 0-AP-17.06, AAC Diesel Generator Emergency Operations, and energize bus 1J from the AAC Diesel Generator.

C.✓ Place #1 EDG in EXERCISE, start #1 EDG, depress #1 EDG Field Flash pushbutton, place #1 EDG in AUTO.

D. Place #1 EDG in EXERCISE, turn Snc key switch for 15H3 to ON, depress #1 EDG Field Flash pushbutton, CLOSE 15H3.

K/A

055 Station Blackout

EA1.06 Ability to operate and monitor the following as they apply to a Station Blackout: Restoration of power with one ED/G (CFR 41.7)

Notes

Modified from facility ILT question bank #156

Answer Option Analysis

A. Incorrect. While this would restore power to Unit 1, it would cause a loss of all AC on Unit 2. Plausible because it is Step 5. c)3) RNO steps for transfering EDG 3 to bus 1J (and is similar to the answer to ILT #156, re-written for a loss of all AC on Unit 1).

B. Incorrect. While this would restore power to Unit 1, the crew should, per 1-ECA-0.0, first attempt to start any EDG that did not start. Starting an EDG that did not automatically start can be done from the MCR and would result in more rapid power restoration than performing ~~local~~ actions of 0-AP-17.06. Plausible because this is the correct option if NO EDGs are running and no AC emergency bus is energized.

C. Correct per 1-ECA-0.0, Step 5.a) RNO. The #1 EDG failed to auto start. Step 5 checks EDG RUNNING and, if not, provides direction to start it from the MCR.

D. Incorrect. Step does not start the #1 EDG. Plausible because these are the correct RNO steps for flashing the field and closing the output breaker of an EDG running unloaded.

References

1-ECA-0.0, Rev. 23, Loss of All AC Power

Added the word "Automatically" for clarity.

Answer analysis 'B'. Loading the AAC on the bus is done from the MCR, local actions are not required (deleted word local).

Removed "automatic" from second bullet in stem. RFA accept 12/20/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	C C B A A D A D B A	Scramble Range: A - D
Tier:	1		Group:	1	
Key Word(s):	ECA-0.0		Cog Level:	C/A4.1	
Source:	N		Exam:	SR06301	
Test:	R		Author / Reviewer:	FJE	

37. 056AK1.01 001/1/1/NATURAL CIRCULATION/MEM3.7/M/SR06301/R/FJE

Unit 1 was operating at 10% reactor power when a loss of off site power caused the RCPs to trip.

Which ONE of the following identifies ALL of the correct indications that would verify natural circulation is in progress.

1 - RCS hot leg temperature -- stable or decreasing
2 - RCS hot leg temperature -- increasing
3 - S/G pressure -- stable or decreasing
4 - S/G pressure -- increasing
5 - RCS hot leg temperature -- at saturation for S/G pressure
6 - RCS cold leg temperature -- at saturation for S/G pressure

A. 1, 3, and 5

B.✓ 1, 3, and 6

C. 2, 4, and 5

D. 2, 4, and 6

K/A
056 Loss of Off-site Power
AK1.01 Knowledge of the operational implications of the following concepts as they apply to the Loss of Offsite Power: Principles of cooling by natural convection (CFR 41.8 / 41.10)

Notes

Modified from bank question FA01301

Answer Option Analysis

A. Incorrect. Although conditions 1 and 3 are indications of natural circulation, condition 5 is incorrect. Plausible if applicant does not understand RCS to SG heat transfer.

B. Correct per 1-AP-39.00, Attachment 1.

C. Incorrect. Increasing temperatures indicate an impending LOSS of natural circulation per 1-AP-39.00.

D. Incorrect. Althought condition 6 is an indication of natural circulation, conditions 2 and 4 are not (see C.). Plausible if applicant focuses only on RCS cold leg temperature.

References

1-AP-39.00, Rev. 6, natural Circulation of the RCS
ND-86.3-LP-4, Rev. 7, Natural Circulation
No changes

RFA accept 12/20/05

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B D D B D C B D D B Scramble Range: A - D

Tier:	1	Group:	1
Key Word(s):	NATURAL CIRCULATION	Cog Level:	MEM3.7
Source:	M	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

38. 057AK3.01 002/1/1/VITAL AC BUS/MEM4.1/N/SR06301/R/FJE

Unit 1 conditions are as follows:

- Unit 1 is at 100% power
- A complete loss of Vital Bus I-1/1A has occurred.
- 1-AP-10.01, LOSS OF VITAL BUS I, is in progress at Step 2.

The reason for closing 1-CH-LCV-1460A and 1-CH-LCV-1460B to isolate letdown in Step 2 is because ____________________ due to loss of power to Vital Bus I-1/1A.

A.✓ Letdown line flow was lost when 1-CH-TV-1204A, LETDOWN LINE I/S TV, failed closed

B. Letdown line pressure control was lost when 1-CH-PC-1145, LTND LINE PRESS CNTRL, shifted to AUTO-HOLD

C. The ability to control VCT level was lost when 1-CH-LC-1112C, 1-CH-LCV-1115A VCT LVL CNTRL, shifted to AUTO-HOLD

D. Letdown temperature control was lost when 1-CC-TCV-103, NRHX OUTLET TEMP CNTRL, valve failed open

K/A
057 Loss of Vital AC Inst. Bus
AK3.01 Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus: Actions contained in EOP for loss of vital ac electrical instrument bus (CFR 41.5, 41.10)

Notes

Need to verify correct failure positon of valves 1TV-1204A.
Need to verify name of 1-CH-LC-1112C.
Need to find power supply of 1-CC-TCV-103 and verify failure position.

Answer Option Analysis

A. Correct. 1-AP-10.01 Step 2.a) directs operator to check 1-CH-TV-1204A closed or deenergized. Att. 1, pg. 2 lists 1-CH-TV-1204 as a load off of Bkr 14. 11448-FM-088C, Rev. 22., Sh. 1 shows 1-CH-TV-1204A, but no failure position. 11488-FM-088A, Rev. 27, Sh. 4 shows 1-CH-TV-1204B fails closed.

B. Incorrect. Plausible because 1-CH-PC-1145 is a load off of Vital Bus 1-III, Bkr 16.

C. Incorrect. Plausible because 1-CH-LC-1112Cis a load off of Vital Bus 1-II, Bkr 17.

D. Incorrect. Plausible because 1-CC-TCV-103 is also a component that will affect letdown operation.

References

ND-90.3-LP-5, Rev. 11, Vital and Semi-Vital Bus Distribution
1-AP-10.01, Rev. 15, Loss of Vital Bus I
1-AP-10.02, Rev. 14, Loss of Vital Bus 1-II
1-AP-10.03, Rev. 11, Loss of Vital Bus 1-III
ND-88.3-LP-2, Rev. 12, Charging and Letdown
1-CH-TV-1204A fails closed (containment isolation valve).
1-CH-LC-1112C noun name is correct.
1-CC-TCV-103 fails open and is powered from Vital Bus 2.

Added "/1A" to the stem to clarify the complete loss of vital bus 1.

Corrected mark number for 1-CH-TV-1204A and 1-CH-LCV-1115A.

Corrected typographical errors in answer 'D'.

Updated procedure revision.

RFA accept 12/20/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	A D A B D B B A C B	Scramble Range: A - D	
Tier:	1			Group:	1	
Key Word(s):	VITAL AC BUS			Cog Level:	MEM4.1	
Source:	N			Exam:	SR06301	
Test:	R			Author / Reviewer:	FJE	

39. 058AA2.02 002/1/1/LOSS OF DC BUS/C/A3.3/M/SR06301/R/FJE

Plant conditions are as follows:

Unit 1 has tripped following steady state operation at 100% power.
1K-B7, BATT SYSTEM 1B TROUBLE, is LIT.
1B Battery Voltmeter indication reads 0 VOLTS.

Which ONE of the following describes the consequences of a loss of the Unit 1 "B" DC bus and subsequent RCP response after the E-0 immediate actions have been completed?

A.✓ The 1J Emergency Bus does NOT have DC control power AND the 'B' and 'C' RCPs are NOT running.

B. The 1H Emergency Bus does NOT have DC control power AND only the 'B' RCP is running.

C. The 1J Emergency Bus does NOT have DC control power AND only the 'C' RCP is running.

D. The 1H Emergency Bus does NOT have DC control power AND the 'B' and 'C' RCPs are NOT running.

K/A
058 Loss of DC Power
AA2.02 Ability to determine and interpret the following as they apply to the Loss of DC Power: 125V dc bus voltage, low/critical low, alarm (CFR 43.5, 45.13)

Notes

Modified from TP00301 exam.

Answer Option Analysis

A. Correct per 1-AP-10.06 and 'B' DC control power supplies both the 'B' and 'C' SS bus.

B. Incorrect (both parts). This describes the consequences of a loss of the 1A DC bus. Plausible if the applicant believes that RCPs will trip on a loss of DC following swap-over to reserve station power.

C. Part 1 correct. Part 2 incorrect, see B.

D. Part 1 incorrect, part 2 correct.

References

ND-90.3-LP-6, Rev. 12, 125 VDC Distribution
1-AP-10.06, Rev. 9, Loss of DC Power
1K-B7, Rev. 5, BATT SYSTEM 1B TROUBLE

Removed information concerning letdown and steam dump operation because memorization of individual loads (so deep in the procedure) is not required. Adding the RCP information maintains the difficulty of this question, but is integrated information that the applicants should know (SS swap over to RSS during E-0 and then which bus will have power and which will not following the swap over). The applicants could become confused and believe that all RCP will trip due to the loss of DC power. SS seperated from the grid during E-0 immediate actions.

Changed distractors B and C from "no RCPs..." to "only the B..", "only the C...".
RFA accept 12/20/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	A D C B A B B C C D	Scramble Range: A - D	

Tier:	1	Group:	1
Key Word(s):	LOSS OF DC BUS	Cog Level:	C/A3.3
Source:	M	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

40. 059G2.4.49 002/2/1/LOSS OF FEEDWATER/MEM4.0/N/SR06301/R/FJE

Unit 2 plant conditions are as follows:

- Reactor power is 82%
- 2H-G8, FW PP DISCH HDR LO PRESS, is LIT.
- 2H-H8, FW PP OVERCURRENT TRIP, is LIT.
- 2H-E5, 6, and 7, STM GEN 2A, 2B, and 2C FW > < STM FLOW, are LIT.
- The 2A MFW pump is TRIPPED.
- The 2B MFW pump and the 2A and 2B Condensate Pumps are RUNNING.

Which one of the following lists ALL of the <u>immediate actions</u>, in the correct sequence, that the crew should perform for the given plant conditions?

A. 1. CLOSE 2-FW-FCV-150A, FW PUMP 1A RECIRC FLOW CONTROL VALVE.
   2. Start the 2C Condensate Pump.

B. 1. Start the 2C Condensate Pump.
   2. OPEN 2-CP-MOV-200 to bypass the Condensate Polishing Building.

C.✓ 1. Start the 2C Condensate Pump.
   2. Reduce turbine load to match steam flow with feed flow.

D. 1. Reduce turbine load to match steam flow with feed flow.
   2. Start the 2C Condensate Pump.

K/A
059 Main Feedwater
G2.4.49 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls (CFR 41.10)

Notes

Question based on Unit 1 procedure. Need facility to verify correct for Unit 2 procedure.
Need facility to provide noun name for MOV-CP-100 (unit specific?).

Answer Option Analysis

A. Incorrect. Manually closing MFW pump recirc valve is not an immediate action. Plausible because this action is contained in 2H-H8.

B. Incorrect. Opening 2-CP-MOV-200 is not an immediate action. Reducing turbine load is an immediate action.

C. Correct per 2-AP-21.00.

D. Incorrect. Plausible because reducing steam demand first will immediately address the steam flow / feed flow mismatch.

References

2-AP-21.00, Rev. 6, Loss of Main Feedwater Flow
Corrected mark number for 2-CP-MOV-200. Noun name not required (CN Polisher Bypass Valve).

RFA accept 12/20/05

Corrected typographical errors in answer analysis option section.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	C C C D B A A B C B	Scramble Range: A - D
Tier:	2		Group:	1	
Key Word(s):	LOSS OF FEEDWATER		Cog Level:	MEM4.0	
Source:	N		Exam:	SR06301	
Test:	R		Author / Reviewer:	FJE	

41. 061A3.03 002/2/1/AFW AUTO START/MEM3.9/N/SR06301/R/FJE

Unit 1 status is as follows:

- Unit 1 was operating at 30% power.
- An automatic reactor/turbine trip was generated 20 seconds ago due to a trip of the running MFP.
- Narrow range SG levels are all going DOWN and currently read as follows:
  A SG: 23%
  B SG: 24%
  C SG: 22%

Which ONE of the following correctly describes the response of the AFW MOVs for the current plant conditions?

The AFW MOVs:

A. immediately receive an open signal and can then be throttled closed immediately.

B.✓ immediately receive an open signal and can NOT be throttled closed immediately.

C. will receive an open signal in about 30 seconds and can then be throttled closed immediately.

D. will receive an open signal in about 30 seconds and can NOT be throttled closed until after they have gone full open.

K/A
061 Auxiliary/Emergency Feedwater
A3.03 Ability to monitor automatic operation of the AFW, including: AFW S/G level control on automatic start (CFR 41.7)

Notes

Modified from facility examination bank question AFW0024

Answer Option Analysis

A. Incorrect (second part). AFW MOVs will receive and open signal following auto-initiation. This open signal is maintained for 45 seconds by a time delay relay. The MOVs cannot be closed during this 45 second period. Plausible because many MOVs do not have a time delay to maintain open power following receipt of an actuation signal.

B. Correct. The AFW MOVs receive an open signal due to the MFP trip and cannot be throttled closed until the time delay relay that maintains opening power to them times out (see A.).

C. Incorrect (first part). Plausible because a SI signal will cause AFW MOVs to open or reopen after a 50 second time delay. No SI signal is present.

D. Incorrect (both parts). See C. for first part of distractor. Second part of distractor is plausible because some MOVs cannot be stroked closed after receiving an open signal until after an open limit is first made up.

References

Facility examination bank question AFW0024
ND-89.3-LP-4, Rev. 21, Auxiliary Feed System
Changed begin to open to receive an open signal, since the AFW MOVs are normally open when at power (adjusted answer analysis section).

Moved "The AFW MOVs.." to the stem. RFA accept 12/20/05.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	B D C D A B D B C C	Scramble Range: A - D
Tier:	2		Group:	1	
Key Word(s):	AFW AUTO START		Cog Level:	MEM3.9	
Source:	N		Exam:	SR06301	
Test:	R		Author / Reviewer:	FJE	

42. 062AA1.01 003/1/1/SW TEMPERATURE/C/A3.1/N/SR06301/R/FJE

With the unit operating at 100% power, the running charging pump's bearing temperature has just exceeded 180 degrees due to a charging pump service water strainer becoming clogged.

Which ONE of the following identifies the action, if any, to be taken in accordance with AP-12.00, Service Water System Abnormal Conditions?

A. Place all charging pumps in pull to lock, swap SW supply strainers, and then re-establish charging flow.

B. No action is required until bearing temperatures exceed 200 degrees.

C✓ Start a stand-by charging pump and then secure the running charging pump.

D. Align the alternate dilution flow path to the running charging pump suction.

K/A

062 Loss of Nuclear Svc Water

AA1.01 Ability to operate and / or monitor the following as they apply to the Loss of Nuclear Service Water (SWS): Nuclear service water temperature indications (CFR 41.7)

Notes

Need facility to verify Tech Specs met in stem and answer options. 0-AP-12.00 discusses 5 chillers (A-E), Tech Spec 3.23 only mentions A-C and states there is a pending mod to add more MER#5 chillers.
Need facility to verify MER 3 chiller high temp trip setpoint is above parameters in stem. 0-VSP-D5 does not provide setpoints for high temp trip.
Could not find lesson plan for Control Room Chillers.

Answer Option Analysis

A. Incorrect. This answer is plausible as these are actions that are taken in AP-8.00 for a gas bound CH pump. The swapping of the SW strainer is also a correct answer. Incorrect per AP-12.00

B. Incorrect. This is plausible as the applicant may confuse RCP bearing criteria with CH bearing temperatures.

C. Correct.

D. Incorrect. This is plausible as placing the charging pumps on the alternate dilution flow path will cool overall pump temperatures, but is incorrect per AP-12.00.

References

0-AP-12.00, Rev. 10, Service Water System Abnormal Conditions

RFA Accepted 1/12/06

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C A A B A B B D A A Scramble Range: A - D

Tier:	1	Group:	1
Key Word(s):	SW TEMPERATURE	Cog Level:	C/A3.1
Source:	N	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

43. 062G2.4.6 002/2/1/ECA0.0 INVENTORY/MEM3.1/N/SR06301/R/FJE

Which ONE of the following correctly lists ALL of the operator actions that prevent OR minimize RCS inventory loss following a loss of ALL AC power?

1. Defeating automatic start of CHG pumps.
2. Defeating automatic start of LHSI pumps.
3. Locally isolating the RCP seal injection lines.
4. Locally isolating the RCP thermal barrier CC return.
5. Depressurize all intact SGs to 175 psig.

A. 1 and 4 ONLY

B. 2 and 5 ONLY

C.✓ 1, 3, and 5

D. 2, 3, and 4

K/A
062 AC Electrical Distribution
G2.4.6 Knowledge of symptom based EOP mitigation strategies (CFR 41.10)

Notes

Answer Option Analysis

Preventing or minimizing RCP seal leakage is designed to minimze RCS inventory loss, a major strategy (along with restoration of power and maintenance of secondary heat sink) of ECA-0.0.
Per ND-95.3-LP-17, the bases for the listed actions are as follows:

1. Avoid potential overload of energized emergency bus AND **protect the RCP** from damage by preventing automatic delivery of cold seal injection water to RCP #1 seal chamber and shaft area.
2. Avoid potential overload of energized emergency bus.
3. **Protects RCPs** from seal and shaft damage that may occur when a CHG pump is started as part of the recovery.
4. Protects the CC system from steam formation due to RCP thermal barrier heating.
5. Reduces RCS temperature and pressure in order to **minimize RCP seal degradation and reduce RCP seal leakage.**

A. Incorrect. Item 1 is correct, 4 is not. Item 4 is plausible because it relates to cooling of an RCP component (the thermal barrier) and a potential failure point for RCS leakage (RCS to CC).

B. Incorrect. Item 2 is incorrect, item 5 is correct. Item 2 is plausible because it relates to the normal RCP seal cooling return flow path.

C. Correct.

D. Incorrect. Item 3 is correct, items 2 and 4 are not. Plausible because all items relate to components interfacing with the RCPs.

References

ND-95.3-LP-17, Rev. 11, ECA-0.0, Loss of All AC Power
1-ECA-0.0, Rev. 23, Loss of All AC Power
Licensee examination bank questions EOP0099, EOP 0213, EOP0314
#2 was replaced, as #2 is an actual loss of inventory that is addressed by ECA-0.0. This would allow potentially two correct answers. Replaced with Defeating the autostart of the LHSI pumps as the applicant may confuse the reason for protecting the bus with preventing inventory loss (updated the answer analysis section) due to the loss of the bus once it is recovered.

RFA accept 12/20/05

MCS Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9
		Answer:	C B A A B A B A C C	Scramble Range: A - D

Tier:	2	Group:	1
Key Word(s):	ECA0.0 INVENTORY	Cog Level:	MEM3.1
Source:	N	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

44. 063A3.01 002/2/1/DC GROUND/MEM2.7/N/SR06301/R/FJE

Which ONE of the following is an indication of a "10 K positive ground" on a 125 VDC bus?

A.✓ Bus current indication is NORMAL and control board positive ground indicating light is OFF.

B. Bus current indication is NORMAL and control board positive ground indicating light is ON.

C. Bus current indication is HIGH and control board positive ground indicating light is OFF.

D. Bus current indication is HIGH and control board positive ground indicating light is ON.

K/A
063 DC Electrical Distribution

A3.01 Ability to monitor automatic operation of the DC electrical system, including: Meters, annunciators, dials, recorders, and indicating lights (CFR 41.7)

Notes

Modified from facility examination bank question DC00007

Answer Option Analysis

A. Correct per DC0007 and ND-90.3-LP-6

B. Correct current indication, incorrect light indication. Plausible because some ground detectors energize a light when a ground is present.

C. Correct light indication, incorrect current indication. Plausible because a ground can result in a high current condition.

D. Incorrect current and light indications.

References

Facility examination bank question DC00007
ND-90.3-LP-6, Rev. 12, 125 VDC Distribution
No Changes

RFA accept 12/20/05

Added the word "positive" to all choices for accuracy.

RFA Accepted 1/12/06

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	A A B B A D B A D B	Scramble Range: A - D

Tier:	2	Group:	1
Key Word(s):	DC GROUND	Cog Level:	MEM2.7
Source:	N	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

45. 064K4.10 003/2/1/LOAD SEQUENCE/MEM2.6/N/SR06301/R/FJE

Unit 1 plant conditions are as follows:

- A spurious safety injection occurred 5 minutes ago.
- A grid problem has JUST resulted in a Station Blackout.

With NO operator action, which ONE of the following describes the SEQUENCE that ALL equipment will automatically load onto the **"H"** bus after #1 EDG re-energizes the bus?

A. 1-VS-F-58A (Filtered Exhaust Fan) THEN "E" Group Pzr Heaters ONLY.

B.✓ 1-FW-P-3A (MDAFW) THEN 1-VS-F-58A (Filtered Exhaust Fan) THEN "E" Group Pzr Heaters.

C. 1-VS-F-58A (Filtered Exhaust Fan) THEN 1-FW-P-3A (MDAFW) THEN "E" Group Pzr Heaters.

D. 1-FW-P-3A (MDAFW) THEN "E" Group Pzr Heaters ONLY.

K/A
064 Emergency Diesel Generator
K4.10 Knowledge of ED/G system design feature(s) and/or interlock(s) which provide for the following: Automatic load sequencer – blackout (CFR 41.7)

Notes

Facility verify answers and distractors

Answer Option Analysis

A. Incorrect. 1-FW-3A also starts 10 seconds after a loss of 2/2 RSS busses for affected unit with an SI in-service. Plausible if applicant only focuses on loss of voltage to the "H" bus.

B. Correct. 1-FW-3A also starts 10 seconds after a loss of 2/2 RSS busses for affected unit with an SI in-service. 1-VS-F-58A sequences onto bus "H" at 30 seconds, and the "E" Group of Pzr Heaters sequences onto bus "H" at 180 seconds.

C. Incorrect. Plausible because MDAFW pumps will sequence onto the respective bus at 140 sec (between 1-VS-F-58A and "E" Pzr Heaters) if the LOOP occurs with a Hi-Hi CLS.

D. Incorrect. Plausible if applicant believes Filtered Exhaust Fans are unaffected (e.g. load sequencing is generated on the fan's alternate power source only) and would not sequence.

References

Licensee exam bank question # 101, 106
ND-89.3-LP-4, Rev. 21, Auxiliary Feed System
ND-90.3-LP-7, Rev. 18, Station Service and Emergency Distribution Protection and Control

Modified stem to have a safety injection in service, thus allowing the 58 fans to be running. Otherwise the 58 fans will not load on to a bus (not normally running).

RFA accept 12/20/05

Corrected EDG/Emergency Bus item and corrected a spelling error.

RFA Accepted 1/12/06

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B B B C A A D A D D Scramble Range: A - D

Tier:	2	Group:	1
Key Word(s):	LOAD SEQUENCE	Cog Level:	MEM2.6
Source:	N	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

46. 065AA1.04 005/1/1/EMERGENCY COMPRESSOR/C/A3.5/N/SR06301/R/FJE

Initial plant conditions are as follows:

- Unit 1 Service Air is cross-connected to **Unit 2** SA.
- Unit 1 IA is isolated from **Unit 2** IA for system engineering leak testing on **Unit 2**
- IA compressors 1-IA-C-1 and **2**-IA-C-1 are RUNNING
- SA compressor **2**-SA-C-1 is out of service due to a motor electrical fault.

The following has occurred:

1-SA-C-1 has just tripped and the following indications are avaiable in the Main Control Room:

- 1B-E5, SA COMPR TRBL, is LIT
- Service Air Pressure is 90 psig and going DOWN
- Unit 1 IA pressure is 95 psig and STEADY

Which ONE of the following is the correct set of actions to RESTORE Service Air pressure for the given plant conditions in accordance with 1B-E5, SA COMPR TRBL?

Locally OPEN:

A.✓ 1-SA-223 (SULLAIR compressor discharge valve) and start the Sullair Diesel Driven Air Compressor.

B. 1-IA-446 (CTMT IA Cross-tie Valve) and 1-IA-447 (CTMT IA Cross-tie Valve) to allow CTMT IA to supply station instrument air.

C. 1-CP-632 (IA Cross-tie Valve) and 1-CP-1041 (IA Cross-tie Valve) to cross-tie IA and CP air systems.

D. 1-IA-1476 (Manual bypass valve for dryer) and locally close 1-IA-1477 (Manual isolation valve for dryer) to bypass IA Dryer 1-IA-D-1.

K/A
065 Loss of Instrument Air
AA1.04 Ability to operate and / or monitor the following as they apply to the Loss of Instrument Air: Emergency air compressor
(CFR 41.7)

Notes

Did not receive operating procedures on Sullair compressor. Ask facility how to start compressor and revise answer as necessary.
Need facility to verify nomenclature for power supply 2A2 to outage ("blue and grey") compressors.
Verify facility expectation for starting or not starting U2 IA compressor.

Answer Option Analysis

A. Correct per 1B-E5. Starting emergency diesel air compressor after opening outlet valve may restore SA pressure.

B. Incorrect. These valves are commonly used to align station instrument air and containment instrument air.

C. Incorrect. IA pressure is normal and steady. Plausible because this is an action in 1B-E6 for low IA pressure (less than 90 psig).

D. Incorrect. See C.

References

1B-E5, Rev. 4, SA COMPR TRBL
ND-92.1-LP-1, Rev. 14, Station Air Systems
1B-E6, Rev. 9, IA LO HDR PRESS/IA COMPR 1 TRBL

Expanded SA to Service Air for clarity.

Deleted bullet for 480V..., the blue and gray compressors exist, but are abandoned in place.

Added the words FIRST and "ATTEMPT to" to the stem, due to the leak location. The leak is unisolable and sullair will not restore pressure, but will attempt to maintain pressure.

Corrected answer analysis section. Changed 1B-E6 to E5 and removed words "regardless of leak location."

RFA accept 12/20/05

Modified stem because for a leak pressure could not be restored. Applicants may not start a compressor and feed the fault. With a tripped compressor, the same knowledge is being tested. Capitalized restore to emphasize that this question is testing how to recover SA vice conserve SA.

RFA Accepted 1/12/06

Added noun names and functional descriptions in parenthasis folllowing valve numbers in the distractors.

RFA accepted 1/19/06

Modified choice B, added procedure reference to stem, moved 'Locally OPEN' from the choice to the stem.

RFA accepted 2/1/06.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A C D B A A C D B C Scramble Range: A - D

Tier:	1	Group:	1
Key Word(s):	EMERGENCY COMPRESSOR	Cog Level:	C/A3.5
Source:	N	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

47. 067AG2.1.23 002/1/2/FIRE COOLDOWN/C/A3.9/N/SR06301/R/FJE

Unit 1 conditions are as follows:

- A fire occurred in the Cable Vault and Cable Tunnel
- The crew is performing 0-FCA-17.00, Limiting Fire Cooldown
- NO RCPs are operating
- Two (2) CRDM fans are operating
- RCS pressure is 2200 psig
- ALL RCS COLD leg temperatures are 450°F.

The MAXIMUM RCS cooldown rate is ____ and the MINIMUM RCS COLD leg temperature is ____.

References provided.

A.✓ 10°F/hour   395°F

B. 10°F/hour   585°F

C. 25°F/hour   390°F

D. 25°F/hour   560°F

K/A
067 Plant Fire On-site
AG2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation (CFR 45.2 / 45.6)

Notes

Attachments

0-FCA-17.00, Rev. 21, Limiting Fire Cooldown, pg. 12 (steps 18.g), 19.a)-c)), Attachment 1, Saturation Curve, Attachment 2, Cooldown Curves (pages 1-3).

Answer Option Analysis

A. Correct per 0-FCA-17.00 Attachment 2, page 3 (less than 3 CRDM fans).

B. Incorrect. Correct cooldown rate, incorrect Tcold. Plausible if applicant determines MAX Tcold.

C. Incorrect. Incorrect cooldown rate. Plausible if applicant uses Attachment 2 page 2 (3 CRDM fans).

D. Incorrect. Incorrect cooldown rate. Plausible if applicant uses Attachment 2, page 2.

References

0-FCA-17.00, Rev. 21, Limiting Fire Cooldown
ND-95.6-LP-3, Rev. 6, Safe Shutdown Fire Area FCAs

Updated procedure revisions.

RFA accept 12/20/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	A D A A A B A A A A	Scramble Range: A - D	
Tier:	1			Group:	2	
Key Word(s):	FIRE COOLDOWN			Cog Level:	C/A3.9	
Source:	N			Exam:	SR06301	
Test:	R			Author / Reviewer:	FJE	

48. 071A1.06 002/2/2/WASTE GAS/C/A2.5/N/SR06301/R/FJE

Initial plant conditions are as follows:

- The Gaseous Waste / Process Vents system is operating normally.
- Process Vent Blower 1-GW-F-1A is RUNNING.
- A WGDT release is in progress.
- Decay Tank Bleed Isolation Valve (FCV-GW-101) flow controller is in AUTO and controlling gas bleed flow at 3.0 cfm.
- Process Vent Effluent Flow Controller (FIC-100) is maintaining PV system flow at 300 scfm in automatic.

The FCV-GW-101 flow controller subsequently fails to 100 cfm, causing 1-GW-RI-102, Process Vent Gas Monitor, to alarm above the HIGH setpoint.

Which ONE of the following statements correctly describes PV system operation 20 minutes after the controller failure and RM alarm assuming NO operator action?

1-GW-F-1A is ________ and PV system flow is ____ scfm.

A.✓ RUNNING 300

B. RUNNING 400

C. TRIPPED 300

D. TRIPPED 200

K/A
071 Waste Gas Disposal

A1.06 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with Waste Gas Disposal System operating the controls including: Ventilation system (CFR 41.5)

Notes

Need to verify maximum value that FCV-GW-101 can pass when full open (100 cfm?).

Need to check possible modes of operation of FIC-100 (e.g. manual/auto, demand/setpoint).

Answer Option Analysis

When 1-GW-RI-102 detects activity greater than the high setpoint, the Decay Tank Bleed Isolation valve (1-GW-FCV-101) and the Containment Vacuum Pump Discharge Isolation valves (1-GW-FCV-160 and 1GW-FCV-260) automatically close. The Process Vent Blower(s) remain running.

PV system flow is controlled by FCV-GW-100. This flow control valve is controlled from FIC-100, which senses actual PV flow. PV flow initially increases when FCV-GW-101 fails to 100 cfm and then returns to within 3 cfm of the pre-transient value after 1-GW-FCV-101 closes.

A. Correct. The Process Vent Blower remains running and PV system flow returns to approximately the pre-transient value.

B. Incorrrect. Although the PV Blower remains running, and PV system flow initially increases, flow will return to the pre-transient value of 300 scfm.

C. Incorrect. Althouth PV system flow returns to approximately the pre-transient value, the PV blower remains running.

D. Incorrect. The PV blower remains running and PV system flow does not decrease.

References

ND-92.4-LP-1, Rev. 10, Gaseous and Liquid Waste Processing Systems
0-RM-K4, Rev. 2, 1-GW-RI-102 HIGH

Unsure of maximum value but the concept is still tested by using 100, however the meter pegs out at 7.5 scfm.

Moved "in automatic" to last part of sentence in last bullet of stem. RFA accept 12/20/05

FIC normally controlly in automatic, this was added to the stem.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	A D D D B A C D B B	Scramble Range: A - D
Tier:	2		Group:	2	
Key Word(s):	WASTE GAS		Cog Level:	C/A2.5	
Source:	N		Exam:	SR06301	
Test:	R		Author / Reviewer:	FJE	

49. 073K3.01 003/2/1/RADIOACTIVE EFFLUENT/C/A3.6/N/SR06301/R/FJE

Plant conditions are as follows:

- Unit 1 is operating at 100% power.
- 1-SW-RI-120 FAIL light is LIT due to calibration of the Discharge Tunnel radiation monitor (1-SW-RM-120) which started 4 hours ago.
- 1-RM-A7, RS/SW HX A ALERT / FAILURE alarmed 2 minutes ago.
- 1-SW-RI-114 FAIL light LIT two minutes ago.
- There is NO work activity in progress on RS/SW HX A radiation monitor 1-SW-RM-114 or the 1A Recirc Spray train.
- Readings on 1-SW-RI-114 were at background levels up until the FAIL light lit.

Which ONE of the following specifies the correct actions to take for the given plant conditions?

A. Enter Tech Spec 3.7, Instrumentation Systems, for 1-SW-RM-114. Increase surveillance on 1-SW-RI-115/116/117, RS/HX B/C/D radiation monitors.

B. Verify all automatic actions have occurred. Enter Tech Spec 3.4, Spray Systems, for the 1A Recirc Spray train.

C. Place the control switches for 1-SW-MOV-104A and 1-SW-MOV-105A to close to ensure SW remains isolated to the 1A RX HX. Enter Tech Spec 3.4, Spray Systems, for the 1A Recirc Spray train.

D✓ Enter Tech Spec 3.7, Instrumentation Systems, for 1-SW-RM-114. Direct HP to initiate sampling IAW VPAP-2103, Offsite Dose Calculation Manual.

K/A
073 Process Radiation Monitoring
K3.01 Knowledge of the effect that a loss or malfunction of the PRM system will have on the following: Radioactive effluent releases (CFR 41.7)

Notes

Facility verify distractor C definitely incorrect.

Answer Option Analysis

A. Incorrect. Plausible because this action is a subset of the actions required IF the 1A RSHX were in service and then removed, per per 1-RM-A7.

B. Incorrect. There are no automatic actions associated with the RS/SW HX radiation monitors. Additionally, per 1-RM-A7 Caution before step 1, "Operation of the RSHX SW radiation monitors is not required to directly support RS functional requirements." Plausible because there are automatic actions associated with numerous other process radiation monitors.

C. Incorrect. Plausible because this action is a subset of the actions required IF the 1A RSHX were in service and then removed, per per 1-RM-A7.

D. Correct per 1-RM-A7 step RNO 1.b)1). Per Tech Spec Table 3.7-6, one rad monitor is required per RSHX and only one is available (before the instrument failure of 1-SW-RM-114). With 1-SW-RI-120 inoperable for IMD cal, HP sampling is the only available alternative to meet the preplanned alternate monitoring method for the 1A RSHX.

References

1-RM-A7, Rev. 5, RS/SW HX A ALERT/FAILURE
1-RM-G7, Rev. 4, DISCH TNL ALERT/FAILURE
ND-91-LP-6, Rev. 10, Recirculation Spray System
ND-93.5-LP-1, Rev. 9, Pre-TMI Radiation Monitoring System
Surry Tech Spec 3.7, Table 3.7-6
Surry examination bank question # 354

Deleted "IMD", unfamiliar with terminology and added word "which" for clarity.

Added the words "to close" to grammatically complete answer 'C'.

RFA accept 12/20/05

Corrected typo in choice 'D' (offside to offsite)

RFA Accepted 1/12/06

MCS Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
		Answer:	D A B D D A D B C D	Scramble Range: A - D	
Tier:	2	Group:	1		
Key Word(s):	RADIOACTIVE EFFLUENT	Cog Level:	C/A3.6		
Source:	N	Exam:	SR06301		
Test:	R	Author / Reviewer:	FJE		

50. 073K5.01 001/2/1/RADIATION DETECTION/MEM2.5/M/SR06301/R/FJE

Which ONE of the following describes the type of detector used in the Steam Generator Blowdown radiation monitors (SS-RM-112/113)?

A. Beta scintillation

B.✓ Gamma scintillation

C. Low range G-M tube

D. High range ion chamber

K/A

073 Process Radiation Monitoring

K5.01 Knowledge of the operational implications as they apply to concepts as they apply to the PRM system: Radiation theory, including sources, types, units, and effects (CFR 41.5)

Notes

Answer Option Analysis

A. Incorrect. Plausible because this type of detector is used on multiple particulate and gas monitors.

B. Correct per ND-93.5-LP-1, B.3.b.(3)

C. Incorrect. Plausible because this type of detector is used in "NRC" monitors.

D. Incorrect. Plausible because this type of detector is used in "NRC" and "CHRRMS" monitors.

References

Licensee examination question bank item 359

ND-93.5-LP-1, Rev. 9, Pre-TMI Radiation Monitoring System

ND-93.5-LP-3, Rev. 7, Post-TMI Radiation Monitoring System

No Change.

RFA accept 12/20/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	B B D D D A A A B B	Scramble Range: A - D

Tier:	2	Group:	1
Key Word(s):	RADIATION DETECTION	Cog Level:	MEM2.5
Source:	M	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

51. 076A1.02 002/2/1/SERVICE WATER/MEM2.6/N/SR06301/R/FJE

Unit 1 is at 100% power with all systems in their normal lineups.

Which ONE of the following correctly describes the effect of inadvertently closing 1-SW-MOV-101A, BC HX SW Supply?

A.✓ Main steam line secondary sample temperatures will increase.

B. Charging pump lube oil temperatures will increase.

C. Steam generator blowdown sample temperatures will increase.

D. Service Air Compressor inner cooler air temperatures increase.

K/A
076 Service Water
A1.02 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SWS controls including: Reactor and turbine building closed cooling water temperatures (CFR 41.5)

Notes

Facility provide noun name for MOV-SW-101A.

Answer Option Analysis

MOV-101A provides SW from the 1C CW inlet to the BC HXs.
MOV-101B provides SW from the 1A CW inlet to the BC HXs.
Normally, only one of these valves is open.

A. Closing MOV-SW-101A isolates SW to the bearing cooling heat exchangers. The main steam line sample coolers are cooled by bearing cooling water.

B. Incorrect. Charging pump lube oil coolers are cooled by Charging Pump Service Water, which is supplied via the 1D, 2A, or 2C CW inlets. Plausible because bearing cooling cools numerous other pump lube oil coolers.

C. Incorrect. SG blowdown sample chillers are cooled by Component Cooling water. The CC HXs receive SW via the 1B and 1D CW inlets. Plausible because SG B/D sample chillers may be associated with other secondary sample systems cooled by bearing cooling water.

D. Incorrect. Plausible as the applicant may believe SA compressors are cooled by BC, similar to the IA compressors.

References

ND-89.5-LP-2, Rev. 23, Service Water System
ND-89.5-LP-3, Rev. 12, Bearing Cooling Water System
ND-88.5-LP-1, Rev. 21, Component Cooling System

Replaced distractor D, as when BC was lost MFP bearing temperatures would increase therefore the motor has to work harder thus increasing winding temperatures. New D is plausible because the instrument air compressors are cooled by bearing cooling.

RFA accept 12/20/05

MCS	Time: 1	Points: 1.00	Version: 0 1 2 3 4 5 6 7 8 9		
			Answer: A B B A D C B B A D		Scramble Range: A - D

Tier:	2	Group:	1
Key Word(s):	SERVICE WATER	Cog Level:	MEM2.6
Source:	N	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

52. 076AK3.05 002/1/2/RCS ACTIVITY/C/A2.9/N/SR06301/R/FJE

Unit 1 conditions are as follows:

- Unit 1 is in Intermediate Shutdown and currently cooling down/depressurizing.
- Letdown purification is in service at 40 gpm using one (1) 60 gpm orifice.
- Hydrogen peroxide has just been added to the RCS to cause a crud burst.
- 1-RM-E7, RC LDN HIGH ALERT / FAILURE is LIT
- The radiation level on 1-CH-RI-118, letdown radiation monitor, is going UP

Which ONE of the following corrective actions should the crew perform for the given plant conditions?

A.✓ Increase letdown flow by placing another letdown orifice in service

B. Place 1-CH-TCV-1143, LETDOWN LINE DIVERT, in the DIVERT position

C. Vent the VCT and reduce VCT hydrogen pressure to less than 12 psig

D. Place 1-CH-LCV-1115A, VCT LEVEL DIVERT, in the DIVERT position

K/A

076 High Reactor Coolant Activity

AK3.05 Knowledge of the reasons for the following responses as they apply to the High Reactor Coolant Activity: Corrective actions as a result of high fission-product radioactivity level in the RCS (CFR 41.5, 41.10)

Notes

Need facility to verify correct answer by procedure. Exam submittal did not include 1-OP-CH-011, CVCS Mixed Bed Demin Operations.

Answer Option Analysis

A. Correct per 1-RM-E7/1-OP-CH011 (?). Current letdown flow is less than available flow due to lower (than NOP) RCS pressure. Increasing flow through the letdown purification demineralizers will reduce the increased RCS activity resulting from the crud burst.

B. Incorrect. This action will bypass the letdown purification demineralizers. Plausible if the applicant believes that the concern is protecting the IX vs. reducing high RCS activity.

C. Incorrect. Per ND-88.3-LP-2, VCT pressure must be maintained above 15 psig to ensure proper operation of RCP pump seals. This action will NOT reduce the RCS activity from the crud burst and may result in damage to RCP seals. Plausible if applicant believes crud burst may result in an increase in VCT pressure or the VCT must be degasses.

D. Incorrect. This action will NOT reduce RCS activity. Plausible if the applicant believes that the IX will not sufficiently process the RCS and the water must not be returned to the RCS (e.g. to avoid clogging RCS seal injection filters).

Referernces

1-RM-E7, Rev. 4, RC LDN HIGH ALERT / FAILURE
1-OP-CH-020, Rev. 2, Placing Letdown In Service Following Auto or Manual Isolation
No Changes.

RFA accept 12/20/05

Corrected typo in choice 'A', remove the word "the"

RFA Accepted 1/12/06

MCS	Time: 1	Points:	1.00	Version:	0 1 2 3 4 5 6 7 8 9		
				Answer:	A B A B B B C D A C	Scramble Range: A - D	
Tier:	1			Group:	2		
Key Word(s):	RCS ACTIVITY			Cog Level:	C/A2.9		
Source:	N			Exam:	SR06301		
Test:	R			Author / Reviewer:	FJE		

53. 076K3.01 002/2/1/SERVICE WATER/C/A3.4/N/SR06301/R/FJE

Unit 1 conditions are as follows:

- Unit 1 in REFUELING SHUTDOWN
- Reactor vessel level is 16.00 feet
- "A" RHR pump is RUNNING
- 1-CC-E-1A and -1B are in service
- 1-CC-P-1A and 1B are RUNNING

The following alarms actuated five (5) minutes ago.
- 1B-H6, RHR HX OUT HI TEMP
- 1C-G3, CTMT CC OUT HDR 1A HI TEMP
- 1C-H3, CTMT CC OUT HDR 1B HI TEMP

Additionally:

- A local operator reports that CC Surge Tank level is slowly going UP with no makeup to the tank .
- All CC pump parameters (discharge pressure, flow, amps) have remained constant.
- 1C-A1/B1/C1, RCP 1A/1B/1C CC RETURN LO FLOW annunciators are all NOT LIT.

Which ONE of the following is the probable cause for ALL of the above plant conditions?

A. Closing Containment Trip valve 1-CC-TV-109**B**

B.✓ Actuation of an Intake Canal Low level isolation signal (3/4 channels)

C. A SW tube leak inside 1-CC-E-1**A**

D. Loss of Vital Bus 1-IV

K/A
076 Service Water

K3.01 Knowledge of the effect that a loss or malfunction of the SWS will have on the following: Closed cooling water (CFR 41.7)

Notes

Facility verify affect of SW tube leak in the CC HX on CC surge tank level.

Answer Option Analysis

A. Incorrect. While this could cause 1C-H3, it would not cause 1B-H6, 1C-G3, or the observed increase in CC Surge Tank level. Plausible because this action could cause one of the plant conditions listed.

B. Correct. An intake canal low level isolation signal causes 1-SW-MOV-102A and B to close, isolating SW to the CC system. This would cause temperatures on both trains of CC to rise, also causing CC water volume to expand, increasing surge tank level.

C. Incorrect. A SW tube leak inside the "A" CC HX would not cause CC surge tank level to go up or 1C-H3. Plausible because a SW tube leak on the "A" CC HX could cause increasing temperatures on "A" train components cooled by CC.

D. Incorrect. Loss of vital bus 1-IV would be indicated by 1C-C1, RCP 1C CC RETURN LO FLOW (CC-TV-105C, RCP C CLR CC RTN TV loses power and closes). 1C-A1/B1/C1 DARK indicate that vital busses 1-II, 1-III, and 1-IV are all energized. Plausible because loss of a vital bus can impact CC (flow to RCP oil coolers and from thermal barrier HXs) and can affect multiple trains (i.e. "A" and "B").

References

Licensee examination bank question SW00010
1B-H6, Rev. 0, RHR HX OUT HI TEMP
1C-G3, Rev. 1, CTMT CC OUT HDR 1A HI TEMP
1C-H3, Rev. 1, CTMT CC OUT HDR 1B HI TEMP
1-AP-10.04, Rev. 10, Loss of Vital Bus IV
ND-89.5-LP-2, Rev. 23, Service Water System
ND-88.5-LP-1, Component Cooling

Changed DARK to NOT LIT for consistency. Corrected answer option analysis for 'A'.

RFA accept 12/20/05

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B A B A D A D A C B Scramble Range: A - D

Tier:	2	Group:	1
Key Word(s):	SERVICE WATER	Cog Level:	C/A3.4
Source:	N	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

54. 078K3.02 004/2/1/INSTRUMENT AIR/C/A3.4/N/SR06301/R/FJE

With Unit 2 at 100% power, a loss of instrument air occurs. In accordance with 2B-E6, IA LO HDR PRESS/IA COMPR 1 TRBL, and 0-AP-40.00, NON-RECOVERABLE LOSS OF INSTRUMENT AIR, the reactor should be manually tripped at _____ instrument air pressure due to ______

A.✓ 50 psig — the anticipated closure of the MSTVs at 35 psig.

B. 50 psig — the anticipated closure of 2-IA-TV-200, CTMT IA COMP DISCHARGE TRIP VALVE, at 30 psig.

C. 70 psig — the anticipated closure of the Main Feed Reg Valves at 65 psig

D. 70 psig — the anticipated closure of 2-CH-TV-2204B, LETDOWN LINE O/S TV, at 65 psig.

K/A
078 Instrument Air
K3.02 Knowledge of the effect that a loss or malfunction of the IAS will have on the following: Systems having pneumatic valves and controls (CFR 41.7)

Notes

Answer Option Analysis

A. Correct. When instrument air pressure decreases to less than 50 psig, the ARP directs implementation of AP-40.00 which will direct a reactor trip. The reason for the reactor trip will be due to the anticipated closure of the MFRVs or MSTVs.

B. Incorrect. Part 1 is incorrect. The setpoint for the reactor trip is 50 psig. Part 2 is plausible as 2-IA-TV-200 is supplied from station instrument air and will fail closed at 30 psig. This will result in multiple valve failures in containment, but not necessarily require a reactor trip.

C. Incorrect. 70 psig is plausible, as all instrument, station, and containment air compressor automatically start above this setpoint. Part 2 is incorrect, as the MSTVs will close at approximately 35 psig.

D. Incorrect. 70 psig is plausible, as all instrument, station, and containment air compressor automatically start above this setpoint. Plausible as a loss of letdown will require actions from the MCR team, but not require a reactor trip. 2204B will close at 65 psig.

References

ND 92.1-LP-1, Station Air Sysytem
ARP 2B-E6, IA LO HDR PRESS/IA COMPR 1 TRBL
0-AP-40.00, Rev. 19, Non-Recoverable Loss of Instrument Air

RFA accept 12/20/05.

Switched answer A and C due to recent plant events. Air dome pressure on a MFRV was reduced to approximately 35 psig and the valve did not close.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A C C D D A D D B A Scramble Range: A - D

Tier:	2	Group:	1
Key Word(s):	INSTRUMENT AIR	Cog Level:	C/A3.4
Source:	N	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

55. 086K6.04 002/2/2/FIRE DETECTION/MEM2.6/N/SR06301/R/FJE

With Unit 1 at 100% power, a failure of the fire detection system in containment has occurred. This resulted in annunciator 0-VSP-F3, FIRE DET SYS TRBL, to alarm.

Which ONE of the following actions is required to monitor containment for a fire following failure of the normal detection system?

A.✓ Activate the containment temperature monitoring program on the plant computer system.

B. Commence hourly walkdowns of containment.

C. Observe containment conditions through the personnel hatch view port hourly.

D. Direct the shift technical advisor to determine the expected containment temperature based on containment heat load every 8 hours.

K/A
086 Fire Protection System (FPS)
K6.04 Knowledge of the effect of a loss or malfunction on the following will have on the FPS: Fire, smoke, and heat detectors (CFR 41.7)

Notes

Answer Option Analysis

A. Correct. Per 0-VSP-F3.

B. Incorrect. Plausible since VSP-F3 does direct waldowns of containment, but is does not require placement of fire suppression gear at the containment hatch. Other failures related to fire detection/suppression do require strategic placement of fire suppression gear.

C. Incorrect. See B. Plausible as in the past containment conditions would be monitored through the view ports.

D. Incorrect. Plausible as the STA has information available that would and should be able to predict containment tempertures, however, this would not aid in fire detection.

References

0-VSP-F3, Rev. 12, FIRE DET SYS TRBL
ND-92.2-LP-1, Rev. 10, Fire Protection Systems

New question.

Removed non-monitoring actions from distractors B and C. RFA accept 12/20/05

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A D A C A B B C D C Scramble Range: A - D

Tier:	2	Group:	2
Key Word(s):	FIRE DETECTION	Cog Level:	MEM2.6
Source:	N	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

56. 103K1.07 003/2/1/CONTAINMENT VACUUM/MEM3.5/N/SR06301/R/FJE

The "A" containment vacuum pump is running in AUTO.
Hand station HIC-CV-100 is set to 9.3 psia.

Which ONE of the following conditions will DIRECTLY cause the "A" containment vacuum pump to STOP?

A. Containment partial pressure increases to 9.4 psia.

B.✓ Containment partial pressure drops to 9.2 psia.

C. An operator CLOSES the "A" containment vacuum pump suction valves using the pushbuttons.

D. A HIGH alarm is received on process vent gas monitor 1-GW-RM-102.

K/A
103 Containment
K1.07 Knowledge of the physical connections and/or cause-effect relationships between the containment system and the following systems: Containment vacuum system (CFR 41.2 to 41.9)

Notes

Need facility to verify that distractor C. is possible (i.e. no electrical interlocks).

Answer Option Analysis

A. Incorrect. The CV pump will not stop until partial pressure is 0.2 psi above the setpoint. Plausible because the pump stops at 0.1 psi BELOW the setpoint.

B. Correct. The CV pump will stop when containment partial pressure is -0.1 psi below the setpoiont. A containment partial pressure of 8.9 psia is -0.1 psi below the setpoint of 9.0 psia.

C. Incorrect. Plausible because a SI will both automatically trip the pump and close the vacuum pump suction valves, but manually closing the suction valves using the pushbuttons will not directly trip the CV pump.

D. Incorrect. Plausible because a HIGH alarm on RM-102 will automatically close the CV pump discharge valve, but the operator must manually secure the CV pump.

References

ND-88.4-LP-5, Rev. 10, Containment Vacuum System
Facility examination bank questions CV00001, CV00004
0-RM-K4, Rev. 2, 1-GW-RI-102 HIGH
Added words "currently running" as this pump does not frequently operate.

Changed the word TRIP to STOP for clarity.

RFA accept 12/20/05

Adjusted containment vacuum setpoint to correspond to plant operating practices.

RFA Accepted 1/12/06

MCS Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
		Answer:	B D C A A C D D B D	Scramble Range: A - D	
Tier:	2		Group:	1	
Key Word(s):	CONTAINMENT VACUUM		Cog Level:	MEM3.5	
Source:	N		Exam:	SR06301	
Test:	R		Author / Reviewer:	FJE	

57. 103K4.06 002/2/1/ISOLATION SIGNAL/C/A3.1/N/SR06301/R/FJE

Unit 1 conditions are as follows:

- Unit 1 is at 100% power.
- Containment pressure transmitter 1-PT-LM-100B failed a calibration surveillance 4 days ago.
- All Tech Spec 3.7 (Instrumentation Systems) required actions for 1-PT-LM-100B have been completed.

Which ONE of the following correctly describes the MINIMUM containment pressure and the MINIMUM number of OPERABLE containment pressure channels that must actuate in order to close 1-TV-RM-100A/B/C (Containment Particulate & Gas Radiation Monitor 1-RM-RI-159/160 trip valves).

A.✓ 17.7 psia two (2)

B. 17.7 psia three (3)

C. 23 psia two (2)

D. 23 psia three (3)

K/A
103 Containment
K4.06 Knowledge of containment system design feature(s) and/or interlock(s) which provide for the following: Containment isolation system (CFR 41.7)

Notes

Answer Option Analysis

A Hi-CLS signal closes 1-TV-RM-100A/B/C.

The setpoint for the Hi-CLS signal is 17.7 psia.

The logic for a Hi-CLS signal (with NO inoperable containment pressure channels) is 3/4.

TS 3.7 requires tripping an inoperable containment pressure channel within 72 hours (3 days), resulting in a 2/3 actuation logic for the remaining (operable) channels.

A. Correct.

B. Incorrect. Correct pressure setpoint, but only 2/3 channels are required to actuate. Plausible if applicant does not understand CLS logic or fails to recognize TS 3.7 requires tripping (e.g. vs. bypassing) the inoperable channel.

C. Incorrect. Correct logic, but incorrect setpoint. Plausible if applicant believes a Hi-Hi CLS is required to close the RM trip valves.

D. Incorrect. Incorrect logic and setpoint. See B. and C.

References

ND-91-LP-5, Rev. 15, Containment Spray System
Tech Spec 3.7, Table 3.7-2, Action 17

Minor change on terminology

RFA accept 12/20/05

MCS	Time:	1	Points:	1.00	Version:	0 1 2 3 4 5 6 7 8 9	
					Answer:	A A D D D B D B D C	Scramble Range: A - D
Tier:		2			Group:	1	
Key Word(s):		ISOLATION SIGNAL			Cog Level:	C/A3.1	
Source:		N			Exam:	SR06301	
Test:		R			Author / Reviewer:	FJE	

58. G2.1.14 002/3//NOTIFICATION/MEM2.5/N/SR06301/R/FJE

Which of the following listed plant event(s) procedurally require MCR Operators to perform notification of plant personnel using the Gaitronics?

1. The MCR receives a report of a fire at the low level intake structure.
2. A rapid ramp at 1% per minute is in progress due to a failure of an isophase bus duct cooling fan when the unit is on excess letdown.
3. The plant is currently experiencing a flooding event in Unit 1 Turbine Building.

A. 1 ONLY

B. 2 AND 3 ONLY

C.✓ 1 AND 3 ONLY

D. 1 AND 2 ONLY

K/A
G2.1.14 Knowledge of system status criteria which require the notification of plant personnel (CFR 43.5 / 45.12)

Notes

Answer Option Analysis

1. 0-AP-48.00, step 1.
2. 0-AP-23.00, not applicable.
3. 0-AP-37.01, step 19.

A. Incorrect. Because choice 3 also requires notification
B. Incorrect. A rapid load reduction does not require plant notification. This is plausible because of the amount of activity that would be occurring in the Turbine Building to support the load reduction. Since the plant is on excess letdown, additional operators in the MCR may be required to facilitate pressurizer level maintenance and load reduction rate.
C. Correct. AP-48 and AP-13 require plant notification.
D. Item 1 is correct. Item 2 is not correct. Plausible since the notification of personnel in AP-13.00 is not an immediate action. (see 2)

References

0-AP-48.00, Rev. 18, Fire Protection - Operations Response
0-AP-23.00, Rev.18, Rapid Load Reduction
0-AP-13.00, Rev. 17, Turbine Building or MER 3 Flooding

RFA accept 12/20/05

MCS Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
		Answer:	C A C C D B B B A A	Scramble Range: A - D	
Tier:	3		Group:		
Key Word(s):	NOTIFICATION		Cog Level:	MEM2.5	
Source:	N		Exam:	SR06301	
Test:	R		Author / Reviewer:	FJE	

59. G2.1.23 004/3//PLANT PROCEDURES/C/A3.9/N/SR06301/R/FJE

Unit 1 plant conditions are as follows:

- Unit 1 shut down from 100% power 160 hours ago.
- Unit 1 is in a Reduced Inventory condition with"B" train RHR providing shutdown cooling.
- The crew entered 1-AP-27.00, Loss of Decay Heat Removal Capability, due to a decreasing trend on 1-RC-LR-105, Cold Shutdown RCS Level - Narrow Range
- The first two (2) steps of 1-AP-27.00 are complete.
- The conditions before entering 1-AP-27.00 and the current conditions, are as follows:

	Before	Current
RCS Level:	10.3 inches and STABLE	7.9 inches and STABLE
RCS Temp:	130 °F and STABLE	150 °F and trending UP
RHR Flow:	3200 gpm and STABLE	2600 gpm and STABLE

In order to satisfy the requirements of 1-AP-27.00 WHILE reducing RCS temperature AND preventing the RHR pumps from vortexing the team must FIRST _____ and then _____

References provided.

A.✓ increase RCS level / increase RHR flow.

B. increase RHR flow / increase RCS level.

C. decrease RHR flow / increase RCS level.

D. decrease RCS level / increase RHR flow.

K/A
G2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation (CFR 45.2 / 45.6).

Notes

Attachments

1-AP-27.00, Rev. 12, Loss of Decay Heat Removal Capability, pages 1-3, 9, Attachments 1-3.

Answer Option Analysis

The current RCS level / RHR flow combination is not within the Acceptable region of Attachment 3. 1-AP-27.00, step 15, requires that the operator either raise level or reduce RH flow.

A. Correct. Both parts.

B. Incorrect. Actions are correct, but order is incorrect. Plausible because this will accomplish the task of completing AP-27.00, but the RHR pumps will vortex.

C. Incorrect. Decreasing RHR flow will meet the intent of AP-27.00 (flow/ level). Part 2 is correct to recover level (action that should be taken). Plausible because these actions will prevent caviation of the RHR pump.

D. Incorrect. Part 2 will lower RCS temperature, but the pumps will vortex.

References

ND-95.2-LP-12, Rev. 9, Loss of RHR Events
1-AP-27.00, Rev. 12, Loss of Decay Heat Removal Capability

RFA accept 1/12/06.

MCS Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
		Answer:	A D A C C C A D B A	Scramble Range: A - D	
Tier:	3		Group:		
Key Word(s):	PLANT PROCEDURES		Cog Level:	C/A3.9	
Source:	N		Exam:	SR06301	
Test:	R		Author / Reviewer:	FJE	

60. G2.1.31 002/3//SWITCH POSITION/MEM4.2/N/SR06301/R/FJE

The Unit 1 reactor makeup system is being used with the following switch/controller settings and plant indications:

- 1-CH-FC-1114A, PG FLOW CNTRL set to 5.0
- 1-CH-YIC-1114A, PRI WATER SUP BATCH INTEGRATOR set to 300 GAL
- BOTH of the following control switches are in the AUTO position.
  1-CH-FCV-1113B, BLENDER TO CHG PUMP
  1-CH-FCV-1114B, BLENDER TO VCT
- MKUP MODE SEL switch in ALT DIL

Which ONE of the following describes the correct makeup valve operation when the MAKE-UP MODE CNTRL switch is placed to START?

A.✓ 1-CH-FCV-1113B OPEN 1-CH-FCV-1114B OPEN

B. 1-CH-FCV-1113B OPEN 1-CH-FCV-1114B CLOSED

C. 1-CH-FCV-1113B CLOSED 1-CH-FCV-1114B OPEN

D. 1-CH-FCV-1113B OPEN 1-CH-FCV-1114B MODULATED

K/A
G2.1.31 Ability to locate control room switches, controls and indications and to determine that they are correctly reflecting the desired plant lineup (CFR 45.12).

Notes

Answer Option Analysis

A. Correct per ND-88.3-H/T-9.7 and 1-OP-CH-007.

B. Incorrrect. Plausible because these are the correct valve positions when the MKUP MODE SEL switch is in AUTO.

C. Incorrect. Plausible because these are the correct valve positions when the MKUP MODE SEL switch is in DILUTE.

D. Incorrect. Plausible because these are the correct valve positions if the operator directs water to the charging pump suction only (by placing the C/S for 1114B to CLOSE) per step 5.4.6 of 1-OP-CH-007.

References

ND-88.3-LP-9, Rev. 14, Blender Control Subsystem
1-OP-CH-007, Rev. 17, Blender Operations
Facility examination bank questions CH00077, CH00078, CH00084

Minor typographical error.

Updated procedure revision

RFA accept 12/20/05

MCS	Time: 1	Points:	1.00	Version:	0 1 2 3 4 5 6 7 8 9		
				Answer:	A D D D A C D A A B	Scramble Range: A - D	
Tier:	3			Group:			
Key Word(s):	SWITCH POSITION			Cog Level:	MEM4.2		
Source:	N			Exam:	SR06301		
Test:	R			Author / Reviewer:	FJE		

61. G2.2.12 002/3//SURVEILLANCE/MEM3.0/B/SR06301/R/FJE

Unit 1 is at 100% power.

The Auxiliary Feedwater system engineer has called the control room and has requested that 1-OPT-FW-003, Turbine Driven Auxiliary Feedwater Pump 1-FW-P-2, be performed due to an emergent industry issue. The TDAFW pump test is NOT on the plan of the day.

Which ONE of the following is the correct action for the control room crew?

A. Ensure that the Motor Driven Auxiliary Feedwater pumps are operable and then perform the TDAFW test.

B. Perform the TDAFW test within 1 hour and request a Probabilistic Safety Analysis (PSA) evaluation after the test is complete. The PSA evaluation must be completed before the end of the work week.

C. Obtain permission from the OMOC and then perform the TDAFW test. No other evaluation is required.

D✓ Request a Probabilistic Safety Analysis (PSA) evaluation. If the PSA evaluation is satisfactory, then notify the OMOC and perform the TDAFW test.

K/A
G2.2.12 Knowledge of surveillance procedures (CFR 41.10).

Notes

From SR02301 exam with editorial changes to distractors.
Need to verify "OMOC" title still valid.

Answer Option Analysis

A. Incorrect. A PSA evaluation is required. Plausible if applicant believes only concern is with Technical Specification LCO.
B. Incorrect. PSA is of no value if test is performed before results are obtained. Plausible if applicant does not understand purpose of PSA evaluation.
C. Incorrect. A PSA evaluation is required. Plausible because OMOC monitors work schedule.
D. Correct. A PSA evaluation is required per 1-OPT-FW-003. DNOS-0204 requires an evaluation of risk.

References

1-OPT-FW-003, Rev. 29, Turbine Driven Auxiliary Feedwater Pump 1-FW-P-2, pg. 9
DNOS-0204, Rev. 1, Work Control
Added notification of the OMOC in accordance with DNOS-0204 (page 1).

RFA accept 12/20/05

Updated standard revision.

MCS Time: 1 Points: 1.00		Version:	0 1 2 3 4 5 6 7 8 9
		Answer:	D C C B C D B A B A Scramble Range: A - D
Tier:	3	Group:	
Key Word(s):	SURVEILLANCE	Cog Level:	MEM3.0
Source:	B	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

62. G2.2.24 003/3//LCO STATUS/C/A2.6/N/SR06301/R/FJE

Unit 1 conditions are as follows:

- Unit 1 is at 100% power.
- The Instrument and Electrical Maintenance Departments are planning to perform emergent maintenance on Unit 1.
- The maintenance will temporarily (estimated duration of 10 minutes) disable the ability of the "B" CHG and "B" LHSI pumps to automatically start when a SI signal is generated.
- The maintenance technicians have told you that the "B" CHG and LHSI pumps will be able to be started manually from the MCR in an emergency.

Which ONE of the following is true regarding the effect of the maintenance activity on the operability of the "B" CHG and LHSI pumps?

A.✓ The ability of the pumps to automatically start on a SI signal is an automatic function required by Tech Specs and the pumps MUST be declared INOPERABLE.

B. Because the pumps can be manually started from the MCR, the pumps do NOT need to be declared INOPERABLE.

C. IF the NRC has reviewed the technical adequacy of the maintenance, the pumps do NOT need to be declared INOPERABLE.

D. Because the time duration that the auto start feature is disabled is very limited compared to allowable outage time limit of the Tech Spec Action Statement, the pumps do NOT need to be declared INOPERABLE.

K/A
G2.2.24 Ability to analyze the affect of maintenance activities on LCO status (CFR 43.2 / 45.13).

Notes

Answer Option Analysis

A. Correct per TS 4.11 5.b. and DNAP-1408, paragraph 3.4.2.

B. Incorrect. Plausible because compensatory measures consisting of manual operator actions may be used in some instances (10CFR50.59 evaluation, actions controlled by an approved procedure and/or approved by NRC)to maintain system operability.

C. Incorrect. Plausible because in some cases, NRC approval is required to substitute manual operator actions for an automatic function.

D. Incorrect. Plausible because in some cases an Operability Determination may be made on limited duration equipment outage times if approved by the NRC.

References

DNAP-1408, Rev. 0, Dominion Operability Determination Program
Surry Technical Specification Section 1.D., Operable
Surry Technical Specifications Section 3.3 Safety Injection System
Surry Technical Specifications Section 4.11, Safety Injection System Tests
Added LH for LHSI pumps for clarification

RFA accept 12/20/05

Corrected format on last bullet. Adjusted power level to correspond with TS requirements (CH and SI are criticality TS). This does not change the intent of the question, just prevents potential applicant confusion.

RFA Accepted 1/12/06

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	A B A C D C A B D D	Scramble Range: A - D
Tier:	3		Group:		
Key Word(s):	LCO STATUS		Cog Level:	C/A2.6	
Source:	N		Exam:	SR06301	
Test:	R		Author / Reviewer:	FJE	

63. G2.3.1 002/3//10CFR20/MEM2.6/N/SR06301/R/FJE

An area in the plant is a Restricted Area due to a dose rate of 700 mrem/hr.

Which ONE of the following access control requirements is correct for this area per Surry Technical Specifications Section 6.4?

A. The area must be posted and locked as a LOCKED HIGH RADIATION AREA and the keys maintained under Administrative Control.

B. The area must be posted as a HIGH RADIATION AREA and provided with a control device that alarms upon entry.

C✓ The area must be barricaded and posted as a HIGH RADIATION AREA and an HP tech must control activities in the area and perform periodic surveillances.

D. The area must be barricaded and posted as a LOCKED HIGH RADIATION AREA and an HP tech must provide continuous coverage for the area.

K/A
G2.3.1 Knowledge of 10 CFR: 20 and related facility radiation control requirements (CFR 441.12 / 43.4)).

Notes

Answer Option Analysis

A. Incorrect. Plausible because Restricted Areas with dose rates exceeding 1000 mrem/hr must be locked (posted as Locked High Radiation Area).

B. Incorrect. While this satisfies 10CFR20, TS 6.4 specifies access controls "in lieu of the control device or alarm signal required by paragraph 20.1601 of 10 CFR 20." Plausible if applicant does not recognize the area is a High Radiation Area and is not familiar with plant access controls for relatively high dose rate areas.

C. Correct per TS 6.4.B.1.c.

D. Incorrect. Plausible because continuous HP coverage is required for entrance into Restricted Areas exceeding 15 R/hr (posted as High Radiation Area Exceeding 15 R/hr)

References

10CFR20.1601
Surry Technical Specification Section 6.4
ND-81.2-PP-3, Rev. 0, External Exposure Control
The correct answer is one of several choices in TS. All items were modified for clarity and consistency with TS 6.4.B.1.

Rewrote question stem. RFA accept 12/20/05
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C B C A D B A C B A Scramble Range: A - D

Tier:	3	Group:	
Key Word(s):	10CFR20	Cog Level:	MEM2.6
Source:	N	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

64. G2.3.2 001/3//ALARA DOSE/C/A2.5/B/SR06301/R/FJE

Operations has a task to be performed in the Auxiliary Building near a 20 foot line source that reads 300 mrem/hr at two (2) feet. Two options exist to complete the task:

Option 1: Operator A can perform the task in 1 hour, working at a distance of four (4) feet from the line source.

Option 2: Operators B and C can perform the same task, using special extension tooling, in 90 minutes working at a distance of nine (9) feet from the source.

According to the facility ALARA plan, __________ should be selected and the total personnel exposure is ___________.

A. Option 1 75 mrem

B✓ Option 1 150 mrem

C. Option 2 45 mrem

D. Option 2 200 mrem

K/A
G2.3.2 Knowledge of facility ALARA program (CFR 41.12).

Notes

Bank question from SR02301 with minor editorial changes.

Answer Option Analysis

A. Incorrect. Option is correct but dose was determined using point source method.
B. Correct. Correct option and correct dose (line source method).
C. Incorrect. Incorrect option and incorrect dose (calculated using point source method).
D. Incorrect. Incorrect option although dose was correctly calculated using line source method.

References

ND-81.2-LP-3, Rev. 8, External Exposure Control

No Changes.

RFA accept 12/20/05

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B C C B B D A D A A Scramble Range: A - D

Tier:	3	Group:	
Key Word(s):	ALARA DOSE	Cog Level:	C/A2.5
Source:	B	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

65. G2.4.15 002/3//COMMUNICATION/MEM3.0/M/SR06301/R/FJE

Plant conditions are as follows:

- A LOCA occurred at 10:35
- An Alert was declared at 10:40
- The initial notification to State and Local Governments has NOT been completed.
- ALL plant communications systems are operating properly and are clear of traffic.

The Shift Supervisor has handed you a completed and approved EPIP-2.01, Attachment 2, Report of Emergency to State and Local Governments, and has asked you to perform the initial notification.

In order to complete this task, you must use the __________ and initiate contact with the first government agency NO LATER THAN __________.

A. Instaphone. 10:50

B.✔ Instaphone. 10:55

C. DEM ARD phone 10:50

D. DEM ARD phone 10:55

K/A
G2.4.15 Knowledge of communications procedures associated with EOP implementation (CFR 41.10).

Notes

Modified from Surry ILT exam bank questions 500, 507, 544, 546.
Questions are not designated SRO only.

Answer Option Analysis

A. Incorrect. Correct phone, incorrect time. Plausible if applicant applies time requirement to time of event vs. time of event classification.

B. Correct per EPIP-2.01

C. Incorrect phone and time. Plausible because DEM ARD is backup to Instaphone. See A.

D. Incorrect. Incorrect phone, correct time. See C.

References

EPIP-2.01, Rev. 32, Notification of State and Local Governments.

Updated procedure reference.

RFA accept 12/20/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	B D A B A A B A D A	Scramble Range: A - D
Tier:	3		Group:		
Key Word(s):	COMMUNICATION		Cog Level:	MEM3.0	
Source:	M		Exam:	SR06301	
Test:	R		Author / Reviewer:	FJE	

66. G2.4.34 002/3//AUX SHUTDOWN PANEL/MEM3.8/N/SR06301/R/FJE

Plant conditions are as follows:

- A fire has occurred in the main control room.
- You are an extra Reactor Operator.
- The MCR switches for both 2-MS-SOV-202A and B are both in the CLOSED position.
- The normal transfer via the H and J Group Transfer Switches have failed.

The Unit 2 SRO has directed you to transfer control of 2-MS-SOV-202A and B to the Auxiliary Shutdown Panel, in accordance with 0-FCA-1.00, Limiting MCR Fire.

In order to perform this task, you must go near the ___(1)___ and put the mode switches associated with 2-MS-SOV-202A and B in the ___(2)___ position and the transfer switches in the ___(3)___ postion.

	(1)	(2)	(3)
A.	Auxiliary Shutdown Panel	CLOSED	Main B
B.✓	Auxiliary Shutdown Panel	OVERRIDE	AUX_P
C.	XFER Relay Cabinet in Unit 2 Cable Vault	CLOSED	Main B
D.	XFER Relay Cabinet in Unit 2 Cable Vault	OVERRIDE	AUX_P

K/A
G2.4.34 Knowledge of RO tasks performed outside the main control room during emergency operations including system geography and system implications (CFR 43.5 / 45.13).

Notes

Need facility to verify position titles.
Need facility to verify unit 2 switch positions and locations are analogous to unit 1 positions and locations.

Answer Option Analysis

A. Incorrect. Correct location, but incorrect switch position. A step in FCA-1.00 requires all switch positions to be as the Main Control Board. This is plausible as each switch has two different positions..

B. Correct. Correct location and correct switch positions.

C. Incorrect. Incorrect location. Incorrect switch position. See A.

D. Incorrect. Incorrect location. Correct switch position (for Aux Shutdown Panel). See B.

References

0-FCA-1.00, Rev.35, Limiting MCR Fire
0-AP-20.00, Rev. 12, Main Control Room Inaccessibility
ND-89.3-LP-4, Auxiliary Feed System
Adjusted question per NRC request to verify switch position titles. This correlates to the methodology in FCA-1.00 for a failed normal transfer, otherwise switches near the ASDP would not be used.

Update answer analysis to reflect this.

Updated procedure revision.

Minor editorial change to stem. RFA accept 12/20/05

MCS Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
		Answer:	B A C C B C B D A D	Scramble Range: A - D	
Tier:	3	Group:			
Key Word(s):	AUX SHUTDOWN PANEL	Cog Level:	MEM3.8		
Source:	N	Exam:	SR06301		
Test:	R	Author / Reviewer:	FJE		

67. G2.4.35 002/3//AUX OPERATOR TASK/MEM3.3/N/SR06301/R/FJE

The MCR has just received notification of significant flooding in the **Unit 2** Turbine Building and has announced the flooding over the plant public address system.

The following conditions exist in the Turbine Buildings:

- Six (6) Turbine Building Sump Pumps are OPERABLE. The control switch for one of the operable pumps, 1-PL-P-2D, is OFF.
- The Unit 2 Turbine Building Operator has OPERATOR CONTROL of 1-PL-P-2D in accordance with 0-OSP-PL-003, Turbine Building Sump Pump Status Verification.
- There are no Temporary Sump Pumps prestaged in either Turbine Building.
- The basement roll-up door between Turbine Buildings is CLOSED.

Which ONE of the following describes the duty of the Unit 2 Turbine Building Operator in response to the announcement?

A.✓ Place 1-PL-P-2D control switch in HAND within 10 minutes.

B. Place 1-PL-P-2D control switch in AUTO within 2 hours.

C. Locally open basement roll-up doors between Turbine Buildings within 10 minutes.

D. Notify the Supervisor Facility Services that prestaging Temporary Sump Pumps is required within 2 hours.

K/A
G2.4.35 Knowledge of local auxiliary operator tasks during emergency operations including system geography and system implications (CFR 43.5 / 45.13).

Notes

Need facility to verify operator and public address terminology.
0-OSP-PL-003 contains requirements based on IPE internal flooding analysis risk assumptions and/or NRC commitments.

Answer Option Analysis

A. Correct per 0-OSP-PL-003, 7.2.b.

B. Incorrect. Wrong switch position, wrong time. Plausible because control switch is normally in Auto and 2 hours is time requirement for operating any staged temporary pumps per 0-OSP-PL-003, Attachment 1.

C. Incorrect. Action is only applicable for Unit 1 after performing 0-AP-13.00, step 5.

D. Incorrect. With 6 operable installed pumps, a temporary pump is not required per 0-OSP-PL-003 Step 7.2.2.f. Plausible because 2 hours is time requirement to be able to operate any required prestaged temporary pumps per 0-OSP-PL-003, Attachment 1.

References

0-OSP-PL-003, Rev. 0, Turbine Building Sump Pump Status Verification
0-AP-13.00, Rev. 17, Turbine Building or MER 3 Flooding
Seperated first bullet in stem into two bullets for clarification and assigned a specific operator control of 1-PL-P-2D in accordance with 0-OSP-PL-003. This was done for clarity.

RFA accept 12/20/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	A B C B B C C C B A		Scramble Range: A - D
Tier:	3			Group:		
Key Word(s):	AUX OPERATOR TASK			Cog Level:	MEM3.3	
Source:	N			Exam:	SR06301	
Test:	R			Author / Reviewer:	FJE	

68. WE03EA1.1 002/1/2/LOCA DEPRESSURIZE/C/A4.0/N/SR06301/R/FJE

Plant conditions on Unit 1 are as follows:

- 1-ES-1.2, Post LOCA Cooldown and Depressurization, is in progress.
- HHSI to cold leg flow is 450 gpm.
- RCS pressure is 1400 psig.
- CETCs are 520 °F.
- Containment pressure is 12 psia.
- PRZR level is 0 %
- ONLY the 1B RCP is in operation.

The method that should be used to depressurize the RCS to refill the PRZR is to OPEN:

A. Spray valve 1-RC-PCV-1455A

B. Spray valve 1-RC-PCV-1455B

C.✓ PRZR PORV 1-RC-PCV-1455C

D. PRZR Aux Spray 1-CH-HCV-1311

K/A

WE03 LOCA Cooldown and Depressurization

EA1.1 Ability to operate and / or monitor the following as they apply to the LOCA Cooldown and Depressurization: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features (CFR 41.7)

Notes

Need to verify valve nomenclature and names with facility.
PRZR PORV 1-RC-PCV-456 would also be a viable method

Answer Option Analysis

A. Incorrect. Without A or C RCP, normal przr spray is not available. Plausible if applicant does not understand RCP/PRZR spray configuration.

B. Incorrect. See A.

C. Correct per 1-ES-1.2, step 14.a) RNO

D. Incorrect. Plausible if applicant recognizes normal spray is not available. Not allowed per procedure and normal charging is not restored until after RCS depressurization.

References

1-ES-1.2, Rev. 25, Post LOCA Cooldown and Depressurization
ND-88.1-LP-3, Rev. 13, Pressurizer and Pressure Relief
ND-93.3-LP-5, Rev. 11, Pressurizer Pressure Control

Updated procedure revision and adjusted HHSI flow to a more realistic number for the given RCS pressure.

RFA accept 12/20/05

MCS	Time:	1	Points:	1.00	Version:	0 1 2 3 4 5 6 7 8 9	
					Answer:	C B D B B C C C D D	Scramble Range: A - D

Tier:	1	Group:	2
Key Word(s):	LOCA DEPRESSURIZE	Cog Level:	C/A4.0
Source:	N	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

69. WE04EK1.2 002/1/1/ECA-1.2/C/A3.5/N/SR06301/R/FJE

Unit 1 plant conditions are as follows:

- A manual reactor trip and SI was inserted due to rapidly lowering PRZR level and pressure.
- The crew is implementing 1-E-0, Reactor Trip or Safety Injection
- CTMT radiation levels are normal and STABLE
- CTMT pressure is 10 psia and STABLE
- CTMT RS sump level has NOT increased during this event.
- 1B-F3, SFGDS Area Sump Hi Level, is LIT
- 1-VG-RM-110, VENT VENT 2 GAS, is below the HIGH setpoint but rapidly trending UP
- RM-GW-130-1, PROCESS VENT STK PART, is below the HIGH setpoint but rapidly trending UP

After completing the applicable steps of 1-E-0, the crew should transtion to_________.

A release path to the environment __________ exist.

A. 1-E-1, Loss of Reactor or Secondary Coolant DOES

B. 1-E-1, Loss of Reactor or Secondary Coolant DOES NOT

C✓ 1-ECA-1.2, LOCA Outside Containment DOES

D. 1-ECA-1.2, LOCA Outside Containment DOES NOT

K/A
WE04 LOCA Outside Containment
EK1.2 Knowledge of the operational implications of the following concepts as they apply to the LOCA Outside Containment: Normal, abnormal and emergency operating procedures associated with LOCA Outside Containment CFR 41.8 / 41.10)

Notes

Need facility to verify Vent Vent rad monitor nomenclature and names.
Need facility to supply normal values for CTMT radiation and CTMT RS sump level.
Can rewrite second part to ask whether containment integrity has/has not been lost, if desired.

Answer Option Analysis

A. Incorrect. Containment parameters are all normal and a release path to the environment exists. Plausible if the applicant fails to recognize normal containment parameters and abnormal Aux Bldg parameters.

B. Incorrect. Containment parameters are all normal. See B.

C. Correct. Containment parameters are normal, but Aux Bldg parameters are abnormal. 1-E-0 step 26 RNO directs transition to 1-ECA-1.2 if the cause of abnormal conditions is a loss of RCS inventory (indicated by lowering PRZR level and safeguards area high sump alarm). A release path exists because the LOCA is outside of containment.

D. Incorrect. Correct procedure but incorrect regarding existence of release path. Plausible if applicant believes that radiation levels below the alarm setpoint do not count as a release path or that realignment of ventilation when radiation levels exceed the alarm setpoint will secure (vs. mitigate) the release path.

References

1-E-0, Rev. 52, Reactor Trip or Safety Injection
ND-95.3-LP-21, Rev. 7, ECA-1.2, LOCA Outside Containment

Adjusted containment pressure to normal operating pressure.
Added RS sump level is not increasing.
Updated procedure revision.

RFA accept 12/20/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	C D B B C C C C A D	Scramble Range: A - D	
Tier:	1			Group:	1	
Key Word(s):	ECA-1.2			Cog Level:	C/A3.5	
Source:	N			Exam:	SR06301	
Test:	R			Author / Reviewer:	FJE	

70. WE05EA2.2 002/1/1/HEAT SINK/C/A3.7/M/SR06301/R/FJE

Unit 1 conditions are as follows:

- Operators are performing 1-FR-H.1, Response to Loss of Secondary Heat Sink, after transitioning from 1-E-0, REACTOR TRIP OR SAFETY INJECTION
- The crew has just successfully completed initiating RCS bleed and feed per steps 11-18 of 1-FR-H.1.
- Unit 1 containment parameters are NORMAL.
- The RCS is intact.
- Both AC emergency busses are energized.
- Both PRZR PORVs are OPEN.

A Control Room Operator subsequently announces that feedwater has been restored using the Turbine Driven Auxiliary Feedwater pump, 1-FW-P-2.

Which ONE of the following describes the correct crew response?

A. When total feed flow is greater than 350 gpm, then RETURN TO 1-E-0, REACTOR TRIP OR SAFETY INJECTION.

B. When narrow range level in at least one SG is greater than 12%, then RETURN TO 1-E-0, REACTOR TRIP OR SAFETY INJECTION.

C. When wide range level in any two SGs is greater than 7%, then RETURN TO 1-E-0, REACTOR TRIP OR SAFETY INJECTION.

D✓ Continue performing 1-FR-H.1 until directed to GO TO 1-ES-1.1, SI TERMINATION.

K/A
WE05 Loss of Secondary Heat Sink
EA2.2 Ability to determine and interpret the following as they apply to the Loss of Secondary Heat Sink: Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments (CFR 43.5 / 45.13)

Notes

Bank question from TP03301.

Answer Option Analysis

A. Incorrect per 1-FR-H.1. Plausible because 350 gpm feed flow to SGs is an indicator of adequate heat sink per 1-FR-H.1 Step 2. Restoration of secondary heat sink is not sufficient reason to transition from 1-FR-H.1 after bleed and feed has been initiated.

B. Incorrect per 1-FR-H.1. Plausible because 12% (normal containment parameters) NR SG level is an indicator of adequate heat sink.

C. Incorrect per 1-FR-H.1. Plausible because WR level in any 2 SGs less than 7% is initiation criteria for RCS feed and bleed.

D. Correct. After RCS bleed and feed is intiated (steps 11-18), 1-FR-H.1 must be completed in order to ensure that SI flow reduction is completed and the pressurizer PORVs are closed. With the RCS intact and the PRZR PORVs operating properly, the only transition after step 18 (besides 1-ECA-0.0 in Attachment 4) is to 1-ES-1.1.

References

1-FR-H.1, Rev. 20, Response to Loss of Secondary Heat Sink
ND-95.3-LP-41, Rev. 9, FR-H.1, Respose to Loss of Secondary Heat Sink
Changed PRZR PORVs from operating to OPEN as they would be in the procedure.

RFA accept 12/20/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	D C C D A C B B A D	Scramble Range: A - D
Tier:	1		Group:	1	
Key Word(s):	HEAT SINK		Cog Level:	C/A3.7	
Source:	M		Exam:	SR06301	
Test:	R		Author / Reviewer:	FJE	

71. WE06EK1.2 001/1/2/DEGRADED COOLING C.2/MEM3.5/M/SR06301/R/FJE

Plant conditions are as follows:

- NO RCPs are running
- Core exit TCs are going UP

Which ONE of the following correctly describes the basis for depressurizing intact SGs to atmospheric pressure when performing 2-FR-C.2, Response to Degraded Core Cooling?

A. To depressurize the RCS to reduce loss of primary coolant via RCS vent paths.

B✓ To depressurize the RCS to recover the core via SI accumulator injection.

C. To depressurize the intact SGs to increase natural circulation flow.

D. To depressurize the intact SGs to increase AFW flow.

K/A

WE06 Degraded Core Cooling

EK1.2 Knowledge of the operational implications of the following concepts as they apply to the Degraded Core Cooling: Normal, abnormal and emergency operating procedures associated with Degraded Core Cooling (CFR 41.8 / 41.10)

Notes

Modified from licensee question bank item number 415

Answer Option Analysis

A. Incorrect. Plausible because depressurizing SGs will reduce RCS pressure, lowering any existing RCS break flow.

B. Correct per references.

C. Incorrect. Stem does not indicate whether forced or natural circulation is occuring. Plausible because depressurizing SGs will reduce SG saturation temperature and create a larger delta T between the RCS and the secondary heat sink, increasing natural circulation flow.

D. Incorrect. Plausible because depressurising SGs will reduce AFW pump delta P, resulting in an increase in AFW flow.

References

ND-95.3-LP-39, Rev. 8, FR-C.2, Response to Degraded Core Cooling
FR-C.2, HP-Rev. 1C, Background Information for WOG Emergency Response Guideline
No Changes.

RFA accept 12/20/05

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B A C A B B D C D D Scramble Range: A - D

Tier:	1	Group:	2
Key Word(s):	DEGRADED COOLING C.2	Cog Level:	MEM3.5
Source:	M	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

72. WE07EA2.1 002/1/2/SATURATED CORE/MEM3.2/N/SR06301/R/FJE

Unit 1 plant conditions are as follows:

- Containment pressure is 15 psia.
- Containment radiation is 1.0E2 R/HR.
- Core Exit TCs are 680 °F.
- RCS Subcooling is 20 °F.
- NO RCPs are running.
- RVLIS Full Range is 58%.

Which ONE of the following is the correct Core Cooling Critical Safety Function color and the applicable procedure?

Reference provided.

A. RED. Go to 1-FR-C.1.

B. ORANGE. Go to 1-FR-C.2.

C✓ YELLOW. Go to 1-FR-C.3.

D. GREEN. Remain in procedure and step in effect.

K/A

WE07 Saturated Core Cooling

WE07EA2.1 Ability to determine and interpret the following as they apply to the Saturated Core Cooling: Facility conditions and selection of appropriate procedures during abnormal and emergency operations (CFR 43.5 / 45.13)

Notes

Answer Option Analysis

A. Incorrect. Plausible if applicant cannot remember Core Exit TC and RVLIS Full Range threshold values with no RCPs running.

B. Incorrect. Plausible per A.

C. Correct per F-2.

D. Incorrect. Plausible if applicant cannot remember RCS Subcooling threshold value.

References

F-2, Rev. 1A, Core Cooling
ND-95.3-LP-40, Rev. 6, Response to Saturated Core Cooling

Need to provide the core cooling status tree to the applicants. Since yellow paths are not required to be implemented, their entry conditions are also beyond required knowledge.

RFA comment: We recognize that this question is a Level 1 question when reference is provided. However, reference must be provided in order to meet K/A at Surry. Therefore the question is accepted as is. RFA 12/20/05.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C A A D A C C A A D Scramble Range: A - D

Tier:	1	Group:	2
Key Word(s):	SATURATED CORE	Cog Level:	MEM3.2
Source:	N	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

73. WE08EA2.2 002/1/2/PTS/C/A3.5/N/SR06301/R/FJE

Unit 1 plant conditions are as follows:

- A main steam line break has occurred.
- 1-FR-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION, is in progress.
- At 10:00, the crew determined that an RCS temperature soak was required.
- Temperature has not decreased below 200 °F at any time.

See Yokogawa Trace on <u>next</u> page for cooldown data.

Additionally, a reference is provided.

What is the EARLIEST time the RCS cooldown could have resumed in accordance with 1-FR-P.1?

A. 1130

B. 1200

C.✓ 1230

D. 1300

QUESTIONS REPORT
for SURRY 2006-301 RO FINAL 02_03_06

K/A
WE08 Pressurized Thermal Shock
EA2.2 Ability to determine and interpret the following as they apply to the Pressurized Thermal Shock: Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments (CFR 43.5 / 45.13)

Notes

Answer Option Analysis

1-FR-P.1, Step 27.b)1) states "Do NOT cooldown RCS until temperatures have been stable for 1 hour"

A. Incorrect. Looking back 1 hour from 11:30, temperatures have gone down. Plausible if applicant determines that a greater than 100 F change in temp occurred from 0930 to 1030 and adds one hour to this time.

B. Incorrect. Looking back 1 hour from 1200, temperatures have gone down. Plausible if the applicant determines that the post-soak cooldown limit of 50 F/hr is applicable during the soak.

C. Correct. Looking back 1 hour from 1230, temperature has been constant at 200 F.

D. Incorrect. Temperature has been constant at 200 F for more than one hour. Plausible if applicant approaches the problem by looking back 1 hour (and no further) from 1300.

References

1-FR-P.1, Rev. 12, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION
ND-95.3-LP-46, Rev. 9, RESPONSE TO IMMINENT PTS CONDITION

This information is contained in the second to last step of FR-P.1

Per RFA, no procedure allowed. Facility to validate on 5 operators. If 2/5 answer correctly, question stands as is. Otherwise, facility to propose replacement question. RFA 12/20/05.

Added last bullet in stem regarding 200 F.
RFA accept 1/12/06.

Added trace and reference. RFA accepted 2/1/06.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C D A C D D D B A C Scramble Range: A - D

Tier:	1	Group:	2
Key Word(s):	PTS	Cog Level:	C/A3.5
Source:	N	Exam:	SR06301
Test:	R	Author / Reviewer:	FJE

74. WE11EK2.1 004/1/1/ES-1.3/C/A3.6/N/SR06301/R/FJE

Unit 1 conditions are as follows:

- The crew has implemented 1-ES-1.3, Transfer to Cold Leg Recirculation, following a LOCA.
- The first seven (7) steps of 1-ES-1.3 were completed approximately 5 minutes ago.
- RWST level is 3%.
- The "B" CHG pump is tagged out.
- The "A" and "C" CHG pumps and the "A" and "B" LHSI pumps are providing recirculation flow.

The RO reports the following:

- "A" CHG pump amps and discharge pressure are oscillating.
- "B" LHSI pump amps and flow are oscillating.
- "A" LSHI pump amps and flow are normal.

Which ONE of the following describes the correct action to take for plant conditions?

A. GO TO 1-SACRG-1, Severe Accident Control Room Guideline Initial Response.

B. GO TO 1-ES-1.4, Transfer to Hot Leg Recirculation

C. Place the "A" CHG pump in PTL. Place the "B" LHSI pump in PTL. Equalize level in CAT and RWST per 1-ES-1.3, step 8.

D✓ Place the "A" and "C" CHG pumps in PTL. Place the "B" LHSI pump in PTL. Align CHG pump suction crosstie IAW Attachment 2 of 1-ES-1.3, Establishing CHG Pump Suction Crosstie.

K/A
WE11 Loss of Emergency Coolant Recirc.
EK2.1 Knowledge of the interrelations between the Loss of Emergency Coolant Recirculation and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features (CFR 41.7)

Notes

Answer Option Analysis

A. Incorrect. Plausible if applicant believes conditions indicate no flow path from the sump to the RCS.

B. Incorrect. Plausible if applicant believes cold leg flow can NOT be established due to a problem downstream of the CHG pumps (step 5.e) RNO).

C. Incorrect. Per 1-ES-1.3, Attachment 1, ALL CHG pumps are placed in PTL and remaining LHSI pump is monitored. Plausible if applicant believes merely stopping pumps losing suction source is sufficient to address indications of containment sump screen blockage.

D. Correct per 1-ES-1.3, Attachment 1, steps 1.a.-c.

References

1-ES-1.3, Rev. 14, Transfer to Cold Leg Recirculation
ND-95.3-LP-10, Rev. 10, ES-1.3, Transfer to Cold Leg Recirculation
Added to the stem that 1-CH-P-1B is tagged out, otherwise ES-1.3 requires all CH pumps to be placed in PTL. Also, the preferred order of charging pumps running is C, B, A. This minor modification does not impact what the question is testing, just lets it flow a little better.

Seperated the condition bullets and the abnormal conditions bullets to aid in indication that this has not been on-going.

RFA accept 12/20/05

Added a bullet concerning the operability of the A LHSI pump. Due to a recent procedure change; a complete loss of recirc capability (sump blockage or not) directs performance of ECA-1.1, therefore knowledge that the other LHSI pump is function properly is required.

Removed ECA-1.1 and replaced with SACRG-1. See above.

Minor editorial changes.
RFA accept 1/12/06.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	D D A C A C B B A B	Scramble Range: A - D
Tier:	1		Group:	1	
Key Word(s):	ES-1.3		Cog Level:	C/A3.6	
Source:	N		Exam:	SR06301	
Test:	R		Author / Reviewer:	FJE	

75. WE12EA2.2 002/1/1/ECA 2.1/C/A3.4/N/SR06301/R/FJE

Unit 1 conditions are as follows:
- Pressurizer level is off-scale LOW
- Pressurizer pressure is steadily going DOWN
- Containment pressure is steadily going UP
- A manual reactor trip and SI was initiated based on pressurizer level and containment pressure
- ALL SG pressures are rapidly going DOWN
- ALL SG NR levels are off-scale LOW

Which ONE of the following lists the correct procedure transitions after the crew completes the applicable steps of 1-E-0, Reactor Trip or Safety Injection?

A. 1-E-1, Loss of Reactor or Secondary Coolant THEN
1-E-2, Faulted Steam Generator Isolation

B. 1-E-1, Loss of Reactor or Secondary Coolant THEN
1-ES-1.2, Post LOCA Cooldown and Depressurization

C✓ 1-E-2, Faulted Steam Generator Isolation THEN
1-ECA-2.1, Uncontrolled Depressurization of All Steam Generators

D. 1-E-2, Faulted Steam Generator Isolation THEN
1-E-1, Loss of Reactor or Secondary Coolant

K/A
WE12 Uncontrolled Depressurization of all Steam Generators
EA2.2 Ability to determine and interpret the following as they apply to the Uncontrolled Depressurization of all Steam Generators: Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments (CFR 43.5 / 45.13)

Notes

Modified from facility ILT question bank #404

Answer Option Analysis

A. Incorrect. Check for faulted SG occurs in 1-E-0 at step 23, check for LOCA inside containment occurs at step 25. IF 1-E-1 were entered (misdiagnosis), step 2 checks pressures in all SGs stable or increasing and RNO directs transition to 1-E-2. Plausible if applicant initially misdiagnoses event as LOCA or incorrectly interprets procedure transitions.

B.Incorrrect. See A for 1-E-1. If 1-E-1 were entered, another opportunity for transition occurs at step 19, which checks if RCS cooldown and depressurization are required. If RCS pressure is greater than 250 psig, procedure directs transition to 1-ES-1.2 (stem provides no information regarding value of RCS pressure). Plausible if applicant misdiagnoses event as LOCA.

C. Correct. 1-E-0 step 23 checks that pressure in all SGs stable or increasing. RNO directs transition to 1-E-2. 1-E-2 step 2 checks any SG pressure stable or increasing. RNO directs transition to 1-ECA-2.1

D. Incorrect. See C for 1-E-2. Next opportunity for transition in 1-E-2 is at step 8 (last step), which directs transition to 1-E-1. Plausible if applicant incorrectly interprets procedure transition.

References

1-E-0, Rev. 52, Reactor Trip or Safety Injection
1-E-1, Rev. 24, Loss of Reactor or Secondary Coolant
1-E-2, Rev. 10, Faulted Steam Generator Isolation

Changed steadily to rapidly. If a SBLOCA existed, then SI flow combined with AFW flow would cause a steady decrease in RCS temperature which would result in a steadily decreasing SG pressure. Using the word rapidly eliminates the possibility of two correct answers.

Updated procedure revisions.

RFA accept 12/20/05

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	C C C B C C B C B C	Scramble Range: A - D	
Tier:	1			Group:	1	
Key Word(s):	ECA 2.1			Cog Level:	C/A3.4	
Source:	N			Exam:	SR06301	
Test:	R			Author / Reviewer:	FJE	