

**U.S. Nuclear Regulatory Commission**  
**Site-Specific RO Written Examination**

**Applicant Information**

Name:

Date: 8 July 2011

Facility/Unit:

Region:    I  II  III  IV

Reactor Type: W  CE  BW  GE

Start Time:

Finish Time:

**Instructions**

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination, you must achieve a final grade of at least 80.00 percent. Examination papers will be collected 6 hours after the examination begins.

**Applicant Certification**

All work done on this examination is my own. I have neither given nor received aid.

\_\_\_\_\_  
Applicant's Signature

**Results**

Examination Value \_\_\_\_\_ Points

Applicant's Score \_\_\_\_\_ Points

Applicant's Grade \_\_\_\_\_ Percent



L091 NRC Written Exam

Multiple Choice (Circle or X your choice)

NAME: **Answer Key**

If you change your answer, write your selection in the blank and initial.

- |     |                                  |                                  |                                  |                                  |       |                           |                                  |   |                                  |                                  |       |
|-----|----------------------------------|----------------------------------|----------------------------------|----------------------------------|-------|---------------------------|----------------------------------|---|----------------------------------|----------------------------------|-------|
| 001 | A                                | B                                | <input checked="" type="radio"/> | D                                | _____ | 026                       | A                                | B   | <input checked="" type="radio"/> | D                                | _____ |
| 002 | A                                | <input checked="" type="radio"/> | C                                | D                                | _____ | 027                       | <input checked="" type="radio"/> | B   | C                                | D                                | _____ |
| 003 | <input checked="" type="radio"/> | B                                | C                                | D                                | _____ | 028                       | <input checked="" type="radio"/> | B   | C                                | D                                | _____ |
| 004 | A                                | <input checked="" type="radio"/> | C                                | D                                | _____ | 029                       | A                                | B   | <input checked="" type="radio"/> | D                                | _____ |
| 005 | A                                | B                                | C                                | <input checked="" type="radio"/> | _____ | 030                       | A                                | B   | C                                | <input checked="" type="radio"/> | _____ |
| 006 | A                                | <input checked="" type="radio"/> | C                                | D                                | _____ | 031                       | A                                | B   | C                                | <input checked="" type="radio"/> | _____ |
| 007 | A                                | <input checked="" type="radio"/> | C                                | D                                | _____ | Deleted<br><del>032</del> | <del>A</del>                     | <del><input checked="" type="radio"/></del> | <del>C</del>                     | <del>D</del>                     | _____ |
| 008 | A                                | B                                | C                                | <input checked="" type="radio"/> | _____ | 033                       | A                                | B   | <input checked="" type="radio"/> | D                                | _____ |
| 009 | A                                | B                                | <input checked="" type="radio"/> | D                                | _____ | 034                       | A                                | B   | <input checked="" type="radio"/> | D                                | _____ |
| 010 | <input checked="" type="radio"/> | B                                | C                                | D                                | _____ | 035                       | A                                | B   | C                                | <input checked="" type="radio"/> | _____ |
| 011 | A                                | B                                | C                                | <input checked="" type="radio"/> | _____ | 036                       | A                                | <input checked="" type="radio"/>            | C                                | D                                | _____ |
| 012 | <input checked="" type="radio"/> | B                                | C                                | D                                | _____ | 037                       | A                                | <input checked="" type="radio"/>            | C                                | D                                | _____ |
| 013 | A                                | B                                | C                                | <input checked="" type="radio"/> | _____ | 038                       | A                                | <input checked="" type="radio"/>            | C                                | D                                | _____ |
| 014 | <input checked="" type="radio"/> | B                                | C                                | D                                | _____ | 039                       | <input checked="" type="radio"/> | B   | C                                | D                                | _____ |
| 015 | <input checked="" type="radio"/> | B                                | C                                | D                                | _____ | 040                       | <input checked="" type="radio"/> | B   | C                                | D                                | _____ |
| 016 | A                                | B                                | C                                | <input checked="" type="radio"/> | _____ | 041                       | A                                | B   | <input checked="" type="radio"/> | D                                | _____ |
| 017 | <input checked="" type="radio"/> | B                                | C                                | D                                | _____ | 042                       | A                                | B   | <input checked="" type="radio"/> | D                                | _____ |
| 018 | A                                | B                                | <input checked="" type="radio"/> | D                                | _____ | 043                       | A                                | B   | C                                | <input checked="" type="radio"/> | _____ |
| 019 | A                                | B                                | <input checked="" type="radio"/> | D                                | _____ | 044                       | A                                | <input checked="" type="radio"/>            | C                                | D                                | _____ |
| 020 | A                                | <input checked="" type="radio"/> | C                                | D                                | _____ | 045                       | A                                | B   | <input checked="" type="radio"/> | D                                | _____ |
| 021 | A                                | B                                | C                                | <input checked="" type="radio"/> | _____ | 046                       | A                                | <input checked="" type="radio"/>            | C                                | D                                | _____ |
| 022 | A                                | B                                | <input checked="" type="radio"/> | D                                | _____ | 047                       | A                                | B   | <input checked="" type="radio"/> | D                                | _____ |
| 023 | A                                | B                                | <input checked="" type="radio"/> | D                                | _____ | 048                       | A                                | B   | C                                | <input checked="" type="radio"/> | _____ |
| 024 | A                                | B                                | <input checked="" type="radio"/> | D                                | _____ | 049                       | A                                | B   | C                                | <input checked="" type="radio"/> | _____ |
| 025 | A                                | B                                | <input checked="" type="radio"/> | D                                | _____ | 050                       | A                                | B   | C                                | <input checked="" type="radio"/> | _____ |

L091 NRC Written Exam

Multiple Choice (Circle or X your choice)

NAME: **Answer Key**

If you change your answer, write your selection in the blank and initial.

- |     |                                  |                                  |                                  |                                  |       |     |                                  |                                  |                                  |                                  |       |
|-----|----------------------------------|----------------------------------|----------------------------------|----------------------------------|-------|-----|----------------------------------|----------------------------------|----------------------------------|----------------------------------|-------|
| 051 | A                                | <input checked="" type="radio"/> | C                                | D                                | _____ | 076 | A                                | <input checked="" type="radio"/> | C                                | D                                | _____ |
| 052 | A                                | B                                | C                                | <input checked="" type="radio"/> | _____ | 077 | A                                | <input checked="" type="radio"/> | C                                | D                                | _____ |
| 053 | A                                | B                                | C                                | <input checked="" type="radio"/> | _____ | 078 | <input checked="" type="radio"/> | B                                | C                                | D                                | _____ |
| 054 | A                                | B                                | <input checked="" type="radio"/> | D                                | _____ | 079 | A                                | B                                | C                                | <input checked="" type="radio"/> | _____ |
| 055 | A                                | B                                | C                                | <input checked="" type="radio"/> | _____ | 080 | A                                | B                                | <input checked="" type="radio"/> | D                                | _____ |
| 056 | <input checked="" type="radio"/> | B                                | C                                | D                                | _____ | 081 | A                                | <input checked="" type="radio"/> | C                                | D                                | _____ |
| 057 | A                                | B                                | C                                | <input checked="" type="radio"/> | _____ | 082 | A                                | B                                | C                                | <input checked="" type="radio"/> | _____ |
| 058 | A                                | <input checked="" type="radio"/> | C                                | D                                | _____ | 083 | A                                | B                                | C                                | <input checked="" type="radio"/> | _____ |
| 059 | A                                | B                                | C                                | <input checked="" type="radio"/> | _____ | 084 | A                                | B                                | C                                | <input checked="" type="radio"/> | _____ |
| 060 | A                                | <input checked="" type="radio"/> | C                                | D                                | _____ | 085 | A                                | B                                | <input checked="" type="radio"/> | D                                | _____ |
| 061 | A                                | B                                | C                                | <input checked="" type="radio"/> | _____ | 086 | A                                | B                                | <input checked="" type="radio"/> | D                                | _____ |
| 062 | <input checked="" type="radio"/> | B                                | C                                | D                                | _____ | 087 | <input checked="" type="radio"/> | B                                | C                                | D                                | _____ |
| 063 | A                                | <input checked="" type="radio"/> | C                                | D                                | _____ | 088 | <input checked="" type="radio"/> | B                                | C                                | D                                | _____ |
| 064 | <input checked="" type="radio"/> | B                                | C                                | D                                | _____ | 089 | A                                | <input checked="" type="radio"/> | C                                | D                                | _____ |
| 065 | <input checked="" type="radio"/> | B                                | C                                | D                                | _____ | 090 | <input checked="" type="radio"/> | B                                | C                                | D                                | _____ |
| 066 | A                                | B                                | <input checked="" type="radio"/> | D                                | _____ | 091 | A                                | B                                | C                                | <input checked="" type="radio"/> | _____ |
| 067 | <input checked="" type="radio"/> | B                                | C                                | D                                | _____ | 092 | A                                | B                                | <input checked="" type="radio"/> | D                                | _____ |
| 068 | A                                | B                                | <input checked="" type="radio"/> | D                                | _____ | 093 | A                                | <input checked="" type="radio"/> | C                                | D                                | _____ |
| 069 | <input checked="" type="radio"/> | B                                | C                                | D                                | _____ | 094 | A                                | B                                | C                                | <input checked="" type="radio"/> | _____ |
| 070 | <input checked="" type="radio"/> | B                                | C                                | D                                | _____ | 095 | A                                | <input checked="" type="radio"/> | C                                | D                                | _____ |
| 071 | <input checked="" type="radio"/> | B                                | C                                | D                                | _____ | 096 | <input checked="" type="radio"/> | B                                | C                                | D                                | _____ |
| 072 | A                                | <input checked="" type="radio"/> | C                                | D                                | _____ | 097 | <input checked="" type="radio"/> | B                                | C                                | D                                | _____ |
| 073 | A                                | <input checked="" type="radio"/> | C                                | D                                | _____ | 098 | <input checked="" type="radio"/> | B                                | C                                | D                                | _____ |
| 074 | A                                | <input checked="" type="radio"/> | C                                | D                                | _____ | 099 | A                                | <input checked="" type="radio"/> | C                                | D                                | _____ |
| 075 | A                                | <input checked="" type="radio"/> | C                                | D                                | _____ | 100 | A                                | <input checked="" type="radio"/> | C                                | D                                | _____ |

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	<b>RO</b>
<b>Knowledge of the interrelations between a reactor trip and the following: Breakers, relays and disconnects</b>	<b>Tier #</b>	1
	<b>Group #</b>	1
	<b>K/A #</b>	EPE 007 EK2.02
	<b>Rating</b>	2.6

### Question 1

Which of the following occurs when an automatic trip signal is sent to the reactor trip breakers?

- A. The undervoltage (UV) coil energizes and the shunt trip coil energizes.
- B. The undervoltage (UV) coil de-energizes and the shunt trip coil de-energizes.
- C. The undervoltage (UV) coil de-energizes and the shunt trip coil energizes.
- D. The undervoltage (UV) coil energizes and the shunt trip coil de-energizes.

Proposed Answer: C. The undervoltage (UV) coil de-energizes and the shunt trip coil energizes.

#### Explanation:

- A. Incorrect.
- B. Incorrect.
- C. Correct. The UV coil is normally energized and de-energizes to open the trip breaker. The shunt trip coil is normally de-energized and energizes to send a signal to open the trip breaker.
- D. Incorrect.

**Technical References:** OIM B-6-4c

**References to be provided to applicants during exam:** None

**Learning Objective:** 3480 - Describe Eagle-21/SSPS components

**Question Source:** Bank # S-54023 X  
 (note changes; attach parent) Modified Bank #  
 New

**Question History:** Last NRC Exam No

**Question Cognitive Level:** Memory/Fundamental X  
 Comprehensive/Analysis

**10CFR Part 55 Content:** 55.41.6

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	RO
<b>Ability to determine or interpret the following as they apply to a small break LOCA: Total flow meter</b>	<b>Tier #</b>	1
	<b>Group #</b>	1
	<b>K/A #</b>	EPE 009 EA2.17
	<b>Rating</b>	3.3

## Question 2

GIVEN:

- A small break LOCA has occurred
- RCS pressure is 1300 psig

What would be the effect of closing SI pump recirculation valve 8974A?

Note:

- FI-918 – Flow indicator for SI pump 1-1
  - FI-922 – Flow indicator for SI pump 1-2
- A. Indicated flow on FI-918 would rise; indicated flow on FI-922 would remain the same.
- B. Indicated flow on both SI pump flow meters FI-918 and FI-922 would rise.
- C. Indicated flow on both SI pump flow meters FI-918 and FI-922 would remain at zero, however, SI pump 1-1 would begin to overheat.
- D. Indicated flow on both SI pump flow meters FI-918 and FI-922 would remain at zero, however, both SI pumps would begin to overheat.

Proposed Answer: B. Indicated flow on both SI pump flow meters FI-918 and FI-922 would rise.

### Explanation:

- A. Incorrect. The recirc line is common, both flows will rise.
- B. Correct. Despite the nomenclature of 8974A, the valve is in a common recirc line, closing the valve will force more flow into the injection line for both pumps.
- C. Incorrect. 1300 psig is less than the shutoff head of the pumps (procedure checks for flow if less than 1650 psig), therefore, there will be flow indicated.
- D. Incorrect. Below the shutoff head, if pressure was above approximately 1500 psig, both pumps would lose their recirc flow and overheat.

**Technical References:** E-0 appendix E step 10, OVID 106709 sheet 4, OIM B-3-1c

**References to be provided to applicants during exam:** None

**Learning Objective:** 8051 - Describe the operation of the ECCS

<b>Question Source:</b>	Bank # A-1053	X
(note changes; attach parent)	Modified Bank #	
	New	
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.7	

Examination Outline Cross-Reference	Level	RO
<b>Knowledge of the operational implications of the following concepts as they apply to the Large Break LOCA: Natural circulation and cooling, including reflux boiling</b>	<b>Tier #</b>	1
	<b>Group #</b>	1
	<b>K/A #</b>	EPE 011 EK1.01
	<b>Rating</b>	4.1

### Question 3

#### PLANT CONDITIONS:

- A LOCA has occurred on Unit 1
- RCS pressure is 700 psig and decreasing slowly
- RCS level has dropped below the top of the hot legs (68% on RVLIS) and lowering slowly
- Steam generator pressures are approximately 1000 psig and decreasing slowly
- Steam generator narrow range levels are all greater than 25%

Which of the following is currently removing the majority of heat from the core?

- A. Break flow only
- B. Break flow and reflux cooling
- C. Break flow and natural circulation
- D. Break flow and radiative heat transfer from the uncovered fuel

Proposed Answer: A. Break flow only

#### Explanation:

- A. Correct. For a large break, the RCS pressure is below steam generator pressure. For natural circulation or reflux cooling to occur, RCS pressure must be greater than steam generator pressure.
- B. Incorrect. This would be the appropriate for a small break and RCS pressure remained above steam generator pressure to remove decay heat.
- C. Incorrect. Once level is below the top of the hot legs, natural circulation will be replaced with reflux cooling, for small breaks.
- D. Incorrect. The core is still covered until RVLIS level is approximately 57%.

**Technical References:** OIM A-2-2, E-1 background for small breaks

**References to be provided to applicants during exam:** None

**Learning Objective:** 41698 - Explain how core cooling is provided during a loss of reactor coolant including the role of the following:

- Steam Generators as a Heat Sink
- Break Flow versus ECCS Flow
- Natural Circulation
- Reflux Cooling
- Venting Steam on a Cold Leg Break
- Voids

<b>Question Source:</b> (note changes; attach parent)	Bank #3 L031 NRC 2/2005 Modified Bank # New	X
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.14	



Examination Outline Cross-Reference	Level	RO
<b>Reactor Coolant Pump (RCP) Malfunctions: Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.</b>	<b>Tier #</b>	1
	<b>Group #</b>	1
	<b>K/A #</b>	APE 015/017 G2.4.49
	<b>Rating</b>	4.6

#### Question 4

Unit 1 is at 27% power.

The following events occur:

- PK05-05, RCP Vibration, alarms
- The operator reports RCP 13 Motor Shaft X and Y are at the Danger level as indicated on the Vibration Monitor Operations Console
- One minute later, PK05-03, RCP13, alarms due to motor bearing temperatures at 180°F and rising approximately 2°F/minute

Which of the following actions will be taken by the crew?

- Immediately trip RCP 13, and commence an orderly plant shutdown to MODE 3.
- Immediately trip the reactor, trip RCP 13 and go to E-0, Reactor Trip or Safety Injection.
- Continue to monitor RCP 13. If motor bearing temperatures reach 200°F, trip the reactor, trip RCP 13 and go to E-0, Reactor Trip or Safety Injection.
- Commence an orderly plant shutdown to MODE 3 to secure RCP 13. If motor bearing temperatures reach 200°F, trip the reactor, trip RCP 13 and go to E-0, Reactor Trip or Safety Injection.

Proposed Answer: B. Immediately trip the reactor, trip RCP 13 and go to E-0, Reactor Trip or Safety Injection.

#### Explanation:

- Incorrect. A reactor trip and RCP trip are required.
- Correct. Correct. For multiple diverse alarms, a trip of the reactor and RCP is required, regardless of power level.
- Incorrect. Per the foldout page and the AR PKs, don't wait for temperatures to exceed 200F.
- Incorrect. Do not wait for temperatures to escalate and a trip is necessary, not a plant shutdown.

**Technical References:** AR PK0503, AR PK0505, OP AP-28 Foldout page

**References to be provided to applicants during exam:** None

**Learning Objective:** 7927 - Given initial conditions and assumptions, determine if a reactor trip or safety injection actuation is required

#### Question Source:

(note changes; attach parent) Bank # Modified Bank # P-35983 X

<b>Question History:</b>	New	
<b>Question Cognitive Level:</b>	Last NRC Exam	No
	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.2	

Rev 1 – added “immediately to A and B based on NRC feedback.

Examination Outline Cross-Reference	Level	RO
<b>Knowledge of the operational implications of the following concepts as they apply to Loss of Reactor Coolant Makeup: Relationship of charging flow to pressure differential between charging and RCS</b>	<b>Tier #</b>	1
	<b>Group #</b>	1
	<b>K/A #</b>	APE 022 AK1.02
	<b>Rating</b>	2.7

### Question 5

Which of the following describes the basis for opening a PORV in FR-S.1, Response to Nuclear Power Generation/ATWS, if pressure is greater than 2335 psig?

- A. To prevent passing two phase flow through the safety valves.
- B. To ensure PTS limits will not be exceeded when the reactor is tripped and cools down.
- C. To minimize primary-to-secondary leakage in case of a SGTR, until other recovery actions can be taken.
- D. To allow sufficient borated injection flow into the RCS to ensure the addition of negative reactivity to the core.

Proposed Answer: D. To allow sufficient borated injection flow into the RCS to ensure the addition of negative reactivity to the core.

#### Explanation:

- A. Incorrect. 2 phase flow through safeties is a concern for accidents such as steam generator safeties and overflow, but not the bases for this check of pressure in FR-S.1
- B. Incorrect. PTS is a concern for overcooling events, such as a steam break.
- C. Incorrect. SGTR is not a concern in FR-S.1 at this time. The concern is inserting negative reactivity to shutdown power generation.
- D. D correct. From FR-S.1 Background: *The check on RCS pressure is intended to alert the operator to a condition which would reduce charging or SI pump injection into the RCS and, therefore, boration.* The PRZR PORV pressure setpoint is chosen as that pressure at which flow into the RCS is insufficient. The contingent action is a rapid depressurization to a pressure which would allow increased injection flow. When primary pressure drops 200 psi below the PORV pressure setpoint, the PORVs should be closed. The operator must verify successful closure of the PORVs, closing the isolation valves, if necessary.

**Technical References:** FR-S.1 background for step 4

**References to be provided to applicants during exam:** None

**Learning Objective:** 7920 - Explain basis of emergency procedure step

**Question Source:** Bank # 43 DCPD Exam 4/2007 X  
(note changes; attach parent) Modified Bank #

**Question History:** Last NRC Exam No

**Question Cognitive Level:** Memory/Fundamental X

Comprehensive/Analysis

**10CFR Part 55 Content:** 55.41.10

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	<b>RO</b>
<b>Knowledge of the interrelations between the Loss of Residual Heat Removal System and the following: RHR heat exchangers</b>	<b>Tier #</b>	1
	<b>Group #</b>	1
	<b>K/A #</b>	APE 025 AK2.01
	<b>Rating</b>	2.9

### Question 6

GIVEN:

- The crew is performing the actions of E-1, Loss of Primary or Secondary Coolant
- RCS pressure is 800 psig
- The operator opens the CCW valves to initiate cooling the RHR heat exchangers

Prior to stopping the RHR pumps, the crew will run the pumps for:

- 5 minutes to prevent an RHR heatup causing a pressure rise which could cause a system break and an inter-system LOCA.
- 5 minutes to prevent an RHR heatup causing thermal binding and inhibiting the operation of motor operated valves.
- 30 minutes to prevent an RHR heatup causing a pressure rise which could cause a system break and an inter-system LOCA.
- 30 minutes to prevent an RHR heatup causing thermal binding and inhibiting the operation of motor operated valves.

Proposed Answer: B. 5 minutes to prevent an RHR heatup causing thermal binding and inhibiting the operation of motor operated valves.

#### Explanation:

A incorrect. Pumps are run for 5 minutes, but the problem is not overpressure. Overpressure would be prevented by relief valve lifting.

B correct. The concern is overpressure in the RHR system, which could inhibit opening of MOVs, such as, the containment sump recirc valves, 8982A and B. this would prevent aligning the suction of the RHR pumps to a water source, effectively a loss of RHR.

C incorrect. 30 minutes is the amount of time the crew has to align CCW to the heat exchangers or the pumps could overheat.

D incorrect. The pumps are run for 5 minutes, not 30.

**Technical References:** E-1, LPE1A

**References to be provided to applicants during exam:** None

**Learning Objective:** 7920B - Explain basis of emergency procedure steps (E-1, E-1.1)

**Question Source:** Bank #  
(note changes; attach parent) Modified Bank #  
New

X

<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.7	

Rev 2 – modified question, although the intent is the same to more closely align to the training material and question balance.

Examination Outline Cross-Reference	Level	RO
Knowledge of the reasons for the following responses as the apply to a loss of Component Cooling Water: Effect on the CCW flow header of a loss of CCW	Tier #	1
	Group #	1
	K/A #	APE 026 AK3.04
	Rating	3.5

### Question 7

Unit 1 is at full power.

CCW surge tank begins to lower rapidly and the crew enters OP AP-11, Malfunction of Component Cooling Water System.

Which of the following actions will be required if CCW surge tank level goes off-scale low?

- A. Trip the reactor and any running ECCS CCP because the CCW pumps may cavitate, resulting in a loss of cooling to the ECCS CCPs.
- B. Trip the reactor and RCPs because the CCW pumps may cavitate, resulting in a loss of cooling to the RCP motors.
- C. Trip the CCW pumps and isolate RCP thermal barriers to prevent the introduction of nitrogen from the surge tank into the thermal barriers.
- D. Trip the CCW pumps and isolate RCP thermal barriers to prevent the potential introduction of this steam into the main portion of the CCW system upon CCW pump restart.

Proposed Answer: B. Trip the reactor and RCPs because the CCW pumps may cavitate, resulting in a loss of cooling to the RCP motors.

### Explanation:

- A. Incorrect. The CCW pumps may cavitate but the concern is loss of cooling to RCPs. The procedure checks that CCP 1-3 is in service (air cooled) or aligns backup cooling to the ECCS CCP, no trip is required.
- B. Correct. CCW pumps could cavitate if surge tank level is offscale low and they would have to be tripped. This would result in a loss of cooling to the RCP motors. Therefore, a trip and stopping RCPs is required. Loss of cooling to the motors requires the RCPs to be tripped in less than five minutes (OP AP-28, section E).
- C. Incorrect. Nitrogen provides over pressure on CCW surge tank. Throughout the procedures, the introduction of gas (specifically into the RCS) is avoided due to its adverse affect, ie interruption of natural circulation.
- D. Incorrect. This is the reason for isolating CCW in ECA-0.0

**Technical References:** OP AP-11, section C, ECA-0.0 background step 8

**References to be provided to applicants during exam:** None

**Learning Objective:** 35490 - Discuss abnormal conditions associated with the CCW system

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #8 DCPD 2/2009	
	New	
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.10	

Rev 1 – added trip of ECCS CCPs to A to make it clearly incorrect.

Examination Outline Cross-Reference	Level	RO
<b>Knowledge of the reasons for the following responses as the apply to the SGTR: Automatic actions provided by each PRM</b>	<b>Tier #</b>	1
	<b>Group #</b>	1
	<b>K/A #</b>	EPE 038 EK3.04
	<b>Rating</b>	3.9

### Question 8

A Steam Generator Tube Rupture has occurred on Steam Generator 1-1.

Which of the following identifies an automatic action for the radiation monitors designed to detect primary to secondary leakage and the reason for the automatic response?

	<b>Automatic Action</b>	<b>Reason</b>
A.	Isolates Steam Jet Air Ejector Discharge.	Prevent the spread of contamination to the environment.
B.	Isolates the effected Steam Generator 10% dump valve.	Prevent the spread of contamination to the environment.
C.	Closes Steam Generator Blowdown Valves -Inside Containment.	Prevents the spread of contamination to the secondary plant.
D.	Closes Steam Generator Blowdown Sample Valves -Outside Containment.	Prevents the spread of contamination to the secondary plant.

Proposed Answer: D. Close outside blowdown valves to prevent the spread of contamination to the secondary plant.

#### Explanation:

Answer A is incorrect. RE-15 Steam Jet Air Ejector Exhaust has no automatic actions.

Answer B is incorrect. RE-72 Main Steam Line Radiation Monitor has no automatic actions.

Answer C is incorrect. RE-19 or RE-23 will not isolate SG Blowdown inside containment.

Answer D is correct. RE-19 or RE-23 will isolate SG Blowdown outside containment to limit the contamination to the secondary plant caused by a SG Tube Rupture.

**Technical References:** OIM G-3-1, LD-2 page 12

**References to be provided to applicants during exam:** None

**Learning Objective:** 8469 - Analyze automatic features and interlocks associated with the Radiation Monitoring System.

<b>Question Source:</b>	Bank #10 DCPD 6/2008	X
(note changes; attach parent)	Modified Bank #	
	New	
<b>Question History:</b>	Last NRC Exam	No



**Question Cognitive Level:**

Memory/Fundamental  
Comprehensive/Analysis

X

**10CFR Part 55 Content:**

55.41.11

Examination Outline Cross-Reference	Level	RO
Knowledge of the operational implications of the following concepts as they apply to Loss of Main Feedwater (MFW): Effects of feedwater introduction on dry S/G	Tier #	1
	Group #	1
	K/A #	APE 054 AK1.02
	Rating	3.6

### Question 9

GIVEN:

- A total loss of Main Feedwater and Auxiliary Feedwater occurs
- The crew is performing the actions of FR-H.1, Response to Loss of Secondary Heat Sink
- Bleed and Feed has been initiated
- All steam generators are “dry”
- Core Exit Thermocouples are 555°F, rising slowly

The capability to feed all steam generators using the TDAFW pump has been restored.

Which of the following guidelines will be used to recover a secondary heat sink?

- Minimum flow (approximately 100 gpm) to one steam generator to ensure heat removal capability will be greater than decay heat.
- Minimum flow (approximately 400 gpm total) to all steam generators to ensure heat removal capability will be greater than decay heat.
- Maximum flow to one steam generator due to the urgent need to restore a heat sink.
- Maximum flow to all steam generators due to the urgent need to restore a heat sink.

Proposed Answer: C. Maximum flow to one steam generator due to the urgent need to restore a heat sink.

#### Explanation:

- Incorrect. Minimum (top end of allowable band) flow to one steam generator is the criteria if thermocouples are lowering.
- Incorrect. Heat removal capability with one steam generator is sufficient to remove decay heat, if core exits were decreasing. Minimum feed to all steam generators is not sufficient with rising core exit temperatures.
- Correct. With temperatures increasing, maximum flow to restore a heat sink as quickly as possible is necessary.
- Incorrect. Only one steam generator is used to limit potential faults to one steam generator.

**Technical References:** FR-H.1 foldout page, FR-H.1 background section 2.4

**References to be provided to applicants during exam:** None

**Learning Objective:** 7920N - Explain basis of emergency procedure steps (FR-Hs)

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #40 Wolf Creek 2009	X
	New	
<b>Question History:</b>	Last NRC Exam	No

**Question Cognitive Level:**

Memory/Fundamental  
Comprehensive/Analysis

X

**10CFR Part 55 Content:**

55.41.10

Rev. 1 – modified for question balance.

Examination Outline Cross-Reference	Level	RO
Ability to determine or interpret the following as they apply to a Station Blackout: Actions necessary to restore power	Tier #	1
	Group #	1
	K/A #	EPE 055 EA2.03
	Rating	3.9

### Question 10

GIVEN:

- The crew is performing the actions of ECA-0.0, Loss of All Vital AC Power
- PK16-15, DSL GEN 11 SHUTDOWN RELAY TRIP, is lit due to Diesel Generator 1-1 trip on overspeed when off site power was lost
- Corrective actions on the diesel have been completed
- An operator is performing ECA-0.3, Restore 4 kV Buses
- The diesel control switch is in AUTO
- The output breaker for Diesel Generator (D/G) 1-1 is open

Which of the following is the expected response of D/G and its output breaker when the operator presses the “D/G Shutdown Rly and Alarm Reset” pushbutton on Vertical Board 4?

- The diesel will start automatically and then the output breaker will automatically close.
- The diesel will start automatically; the output breaker will remain open but can be closed by the operator after the diesel starts and is up to rated speed and voltage.
- The diesel will not start but can now be started in MANUAL; the output breaker will then automatically close after the diesel starts.
- The diesel will not start but can now be started in MANUAL; the output breaker will remain open but can be closed by the operator after the diesel starts and is up to rated speed and voltage.

Proposed Answer: A. The diesel will start and then the output breaker will automatically close.

#### Explanation:

- Correct. Once the relay is reset, the diesel will start due to the automatic transfer to diesel signal still present. When the diesel starts, the breaker will automatically close.
- Incorrect. The breaker will close automatically once the diesel is up to speed.
- Incorrect. If the candidate believes that simply resetting the relay only removes the block and allows the diesel to be started.
- Incorrect. If it is believed the diesel must be manually started, then the need to manually close the breaker is also a possibility.

**Technical References:** ECA-0.3, AR PK16-15

**References to be provided to applicants during exam:** None

**Learning Objective:** 37725 - Analyze automatic features and interlocks associated with the Diesel Generator System

**Question Source:**

Bank #

(note changes; attach parent)	Modified Bank #	
<b>Question History:</b>	New	X
<b>Question Cognitive Level:</b>	Last NRC Exam	No
	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.7	

Examination Outline Cross-Reference	Level	RO
Ability to determine and interpret the following as they apply to the Loss of Offsite Power: ED/G indicators for the following: voltage, frequency, load, load-status, and closure of bus tie breakers	Tier #	1
	Group #	1
	K/A #	APE 056 AA2.37
	Rating	3.7

### Question 11

Diesel Emergency Generator 13 is in AUTO and carrying its vital 4 kV bus following a loss of offsite power.

The Operator takes the diesel's speed control switch to RAISE for three (3) seconds.

Which of the following describes the effect of this action on indicated bus voltage, frequency, and MWe?

	VOLTS	FREQ	MWe
A.	UP	No change	No change
B.	No change	UP	No change
C.	UP	UP	UP
D.	No change	No change	No change

Proposed Answer: D. No change – No change – No change

#### Explanation:

- A. Incorrect, this is true if volts is raised
- B. Incorrect, load is set by sequencer
- C. Incorrect, this is if in DROOP mode and volts are raised as well.
- D. Correct, Volts set by voltage control switch, freq set at 60 hz, load determined by bus loading

**Technical References:** J6B – Diesel Generator System

**References to be provided to applicants during exam:** None

**Learning Objective:** 4158 - Explain the Isochronous/Droop modes of operation

**Question Source:** Bank #49 DCPD 2/2005 X  
(note changes; attach parent) Modified Bank #

**Question History:** Last NRC Exam No

**Question Cognitive Level:** Memory/Fundamental  
Comprehensive/Analysis X

**10CFR Part 55 Content:** 55.41.7

Rev 1 – changed SAME to no change (reads better) removed KVAR, little discriminatory value.  
Modified A & B based on NRC feedback.

Examination Outline Cross-Reference	Level	RO
Loss of Vital AC Electrical Instrument Bus: Knowledge of the purpose and function of major system components and controls.	Tier #	1
	Group #	1
	K/A #	APE 057 G2.1.28
	Rating	4.1

### Question 12

Unit 1 is at 20% power.

PY-12 is de-energized.

Which of the following describes the major control that is lost and the effect on the plant?

- A. C-2 actuation will inhibit Auto and Manual rod motion.
- B. Intermediate Range High Flux actuation will trip the plant.
- C. Loss of P-13 input to P-7 will result in blocking several automatic reactor trip signals.
- D. Loss of coincidence for C-20 will cause AMSAC to no longer being armed in 240 seconds.

Proposed Answer: A. Auto and Manual rod motion is inhibited by C-2.

#### Explanation:

- A. Correct. N42 loses power, which trips the C-2 bistable. The coincidence is 1 of 4, rod motion, auto and manual, is blocked.
- B. Incorrect. N36 does lose power and the plant would trip on high flux, but the trip is blocked above 10% power.
- C. Incorrect. P-13 is 1 of 2 and therefore is not lost, additionally, P-7 inputs if from both P-13 or P-10, both are still valid inputs.
- D. Incorrect. above 40% (32% turbine load), this would be correct.

**Technical References** OIM pages B-6-2a, B-6-3a, B-6-11

**References to be provided to applicants during exam:** None

**Learning Objective:** 3332 - Discuss abnormal conditions associated with the Instrument AC System

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.7	

Rev 2 – changed question to focus on plant controls lost due to the loss of instrument bus and remove closeness of previous question to another question dealing with loss of an inverter. Changed cognitive level to higher.



<b>Examination Outline Cross-Reference</b>	<b>Level</b>	<b>RO</b>
<b>Ability to operate and/or monitor the following as they apply to the Loss of DC Power: Static inverter dc input breaker, frequency meter, ac output breaker, and ground fault detector</b>	<b>Tier #</b>	1
	<b>Group #</b>	1
	<b>K/A #</b>	APE 058 AA1.02
	<b>Rating</b>	3.1

### Question 13

The plant is at full power, in a normal electrical lineup.

PK 19-19, UPS Failure alarms. Inverter input voltages have not changed and there are no other alarms.

Which of the following has occurred?

- A. Loss of a bypass regulating transformer
- B. Loss of UPS AC output to a PY panel
- C. The AC input breaker for an inverter has opened
- D. The DC input breaker for an inverter has opened

Proposed Answer: D. The DC input breaker for an inverter has opened

#### Explanation:

- A. Incorrect. No change in voltage, however, it is not a failure of the UPS. PK19-18 alarms.
- B. Incorrect. While it alarm PK19-19, output voltage would be zero and the PY de-energized, with multiple alarms.
- C. Incorrect. PK19-18 alarms, not a failure input and voltage would be slightly lower.
- D. Correct. Loss of DC alarms PK19-19 but voltage does not change due to the inverter still supplied by the AC source, which is at a slightly higher voltage.

**Technical References:** AR PK19-18, AR PK19-19, STG J10 section 3.0

**References to be provided to applicants during exam:** None

**Learning Objective:** 37807- Describe controls, indications, and alarms associated with the Instrument AC System

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.7	

Rev 1 – changed answer and distractor order to make D (not C) the correct answer.

Examination Outline Cross-Reference	Level	RO
Loss of Nuclear Service Water: Knowledge of system purpose and/or function.	Tier #	1
	Group #	1
	K/A #	APE 062 G2.1.27
	Rating	3.9

#### Question 14

The crew is aligning Unit 1 for Cold Leg Recirculation in accordance with E-1.3, Transfer to Cold Leg Recirculation.

Only one train of ASW is available.

When the crew completes the alignment, only one RHR heat exchanger and 3 CFCUs will be in operation to prevent which of the following?

- A. exceeding CCW system temperature design limit
- B. flashing and water hammer in the ASW system
- C. runout of the CCW pump
- D. runout of the ASW pump

Proposed Answer: A. exceeding CCW system temperature design limit

#### Explanation:

Design of the ASW (Diablo equivalent to Nuclear Service Water) system is to adequately remove CCW heat. EOP E-1.3 includes operator actions to limit the heat loads during post-LOCA cold-leg recirculation if less than two ASW pumps and two CCW heat exchangers are in service.

- A. Correct. The heat load of all the loads on the CCW with only one ASW train could cause the system to not be able to meet its purpose to remove heat from the CCW system and the CCW system could exceed its design temperature.
- B. Incorrect. Temperature remains less than saturation.
- C. Incorrect. With a single train of CCW in service, its plausible the student could focus on the single CCW train and believe runout is a possibility. The problem is due to the lack of cooling (from ASW), CCW heat removal is reduced to the point that if loads are not restricted, design temperature could be exceeded.
- D. Incorrect. The system is designed for one pump to supply both CCW trains during normal operation without runout.

**Technical References:** E-1.3, STG E5 page 1-4 and 1-5

**References to be provided to applicants during exam:** None

**Learning Objective:** 8105 - Explain significant CCW system design features and the importance to nuclear safety

**Question Source:** Bank #  
(note changes; attach parent) Modified Bank #

<b>Question History:</b>	New	X
	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.8	
Rev 2 – changed order of answers to make A correct (previously C) and changed C to align with D (runout of pumps)		

Examination Outline Cross-Reference	Level	RO
<b>Knowledge of the reasons for the following responses as they apply to the Loss of Instrument Air: Knowing effects on plant operation of isolating certain equipment from instrument air</b>	<b>Tier #</b>	1
	<b>Group #</b>	1
	<b>K/A #</b>	APE 065 AK3.03
	<b>Rating</b>	2.9

### Question 15

Unit 2 is operating at 100% power.

Instrument air is lost to the Containment.

Which of the following describes the initial plant response without operator action?

- A. Pressurizer level will rise due to a loss of letdown flow.
- B. Pressurizer level will lower due to a loss of charging flow.
- C. Pressurizer pressure will lower due to maximum spray flow.
- D. Pressurizer pressure will rise due to maximum charging flow.

Proposed Answer: A. Pressurizer level will rise due to a loss of letdown flow.

#### Explanation:

- A. Correct. Letdown will isolate, charging flowpath is unaffected (containment isolation valves are MOVs and HCV-142 and FCV-128 are outside containment. Charging will lower to attempt to maintain level on program, but level will increase for a period of time. Long term, charging will lower and FCV-128 will close down and stop all charging flow to seals and the RCS and eventually pressurizer level will decrease (to zero uncorrected).
- B. Incorrect. Charging is not isolated and will cause an initial rise in pressurizer level.
- C. Incorrect. Spray valves fail closed.
- D. Incorrect. Charging control valves are outside containment and are not affected.

**Technical References:** OP AP-9

**References to be provided to applicants during exam:** None

**Learning Objective:** 3541 - List the effects that a loss of Instrument Air would have on the plant

<b>Question Source:</b>	Bank #53 STP 3/2010	X
(note changes; attach parent)	Modified Bank #	
	New	
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.5	

Examination Outline Cross-Reference	Level	RO
<b>Knowledge of the operational implications of the following concepts as they apply to Generator Voltage and Electric Grid Disturbances: Turbine / generator controls</b>	<b>Tier #</b>	1
	<b>Group #</b>	1
	<b>K/A #</b>	APE 077 AA1.02
	<b>Rating</b>	3.8

### Question 16

Unit 1 is at full power in a normal electric plant alignment.

A sudden power reduction on the 500 kV grid causes a small frequency oscillation to occur.

Which of the following will occur in response to the oscillation?

- A. The Voltage Regulator will switch to MANUAL and operator action will be needed to adjust generator frequency to 60 Hz.
- B. Frequency will oscillate until the operator takes the Voltage Regulator to MANUAL to maintain generator frequency at 60 Hz.
- C. The DEHC Load Loop will vary turbine load to dampen the oscillation.
- D. The Power System Stabilizer will vary generator terminal voltage to dampen the oscillation.

Proposed Answer: D. The Power System Stabilizer will vary generator terminal voltage to dampen the oscillation.

#### Explanation:

- A. Incorrect. There are circumstances that will cause the VR to shift to MANUAL. This is not one of them.
- B. Incorrect. Operator will not have to take any action to maintain frequency.
- C. Incorrect. If either of the generator output breakers is closed and the MOD is closed, the system uses the load loop for control of turbine load. • The load loop compares the turbine rated speed to the actual speed for droop compensation and modifies the signal as necessary, but only in the downward direction.
- D. Correct. Power System Stabilizer (PSS) - used to dampen low frequency power oscillations on the 500kV System due to a sudden reduction in load. This is accomplished by changing the generator terminal voltage such that the voltage varies in phase with the disturbance detected by transient changes in the generator speed. This stabilizer is also known as the Supplementary Excitation Control (SEC) and is used by all utilities in the Western U.S. to reduce small frequency oscillations during normal and system upset conditions.

**Technical References:** OP AP-30, J-4A, LC-3C

**References to be provided to applicants during exam:** None

**Learning Objective:** 5280 – Analyze automatic features and interlocks associated with the Main Generator

**Question Source:**

(note changes; attach parent)

Bank #

Modified Bank #

New

X

**Question History:**

Last NRC Exam

No

<b>Question Cognitive Level:</b>	Memory/Fundamental	X
	Comprehensive/Analysis	
<b>10CFR Part 55 Content:</b>	55.41.7	

Rev 1 – added “to dampen the oscillation” to D for clarity and C for question balance. Changed answer from C to D lower the number of correct answers that are “C”. added “generator” to A for balance

Examination Outline Cross-Reference	Level	RO
Ability to operate and or monitor the following as they apply to the (Loss of Secondary Heat Sink): Operating behavior characteristics of the facility.	Tier #	1
	Group #	1
	K/A #	E05 EA1.2
	Rating	3.7

### Question 17

GIVEN:

- The crew has entered FR-H.1, Response to Loss of Secondary Heat Sink
- Containment pressure is 1.5 psig and stable
- RCPs have been stopped
- RVLIS full range level is 100%
- Steam generator narrow range levels are all offscale low
- Steam generator wide range levels are all approximately 20% and lowering

The operator notes the following:

- RCS delta T has lowered from greater than 40°F to less than 10°F and is slowly lowering
- RCS pressure has risen to approximately 2350 psig and all PORVs are open

Which of the following actions, if any, will be taken by the crew?

- Initiate bleed and feed; steam generators are no longer acting as a heat sink.
- Increase dumping steam from the intact steam generators to enhance natural circulation.
- No action required; natural circulation is now fully developed and removing RCS decay heat.
- No action required; reflux cooling has replaced natural circulation as the heat removal mechanism.

Proposed Answer: A. Initiate bleed and feed; steam generators are no longer acting as a heat sink.

#### Explanation:

- Correct. This should not be confused with the onset of natural circulation in which the RCS pressure continues to increase after the RCPs are stopped and may reach the PRZR PORV setpoint. The key to determining if the RCS pressure rise is due to loss of heat sink or natural circulation is the loop delta-T. The loop delta-T is expected to be large for natural circulation and small for a loss of heat sink since there is no heat transfer to the secondary. Therefore, verifying a slowly increasing RCS pressure and temperature trend plus a large loop delta-T prior to the PORV opening confirms natural circulation whereas a relatively stable temperature and pressure and a small loop delta-T combined with SG wide range low level prior to the PORV opening confirms a loss of heat sink.
- Incorrect. Steam generators have been lost as a heat sink. In the natural circulation procedures, when heat sink is available, as steam flow drops, operators will open steam dumps to maintain the steaming rate.
- Incorrect. Natural circulation would have a larger delta T

D. Incorrect. RVLIS level is 100%, no reflux cooling is occurring.

**Technical References:** FR-H.1 background, step 3 caution 1

**References to be provided to applicants during exam:** None

**Learning Objective:** 5798 - Explain the importance and time limits involved with establishing bleed and feed on a loss of secondary heat sink

**Question Source:**

(note changes; attach parent)

Bank #

Modified Bank #

New

X

**Question History:**

Last NRC Exam

No

**Question Cognitive Level:**

Memory/Fundamental

Comprehensive/Analysis

X

**10CFR Part 55 Content:**

55.41.5



<b>Examination Outline Cross-Reference</b>	<b>Level</b>	<b>RO</b>
<b>Loss of Emergency Coolant Recirculation, Knowledge of the interrelations between the Loss of Emergency Coolant Recirculation and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.</b>	<b>Tier #</b>	1
	<b>Group #</b>	1
	<b>K/A #</b>	E11 EK2.2
	<b>Rating</b>	3.9

### Question 18

GIVEN:

- At 0900 the Reactor Trips due to an earthquake
- At 0920 a LOCA occurs
- At 0950 the crew transitioned to ECA-1.1, Loss of Emergency Recirculation, due to the failure of both RHR pumps
- Crew has reduced ECCS flow to one SI pump and one ECCS CCP per ECA-1.1
- SI flow cannot be terminated due to lack of subcooling

At 1030 the crew is performing ECA-1.1 Step 15 RNO which is to establish the minimum required ECCS flow to remove decay heat.

Which of the following is the minimum flow rate that would satisfy the ECA-1.1 Step 15 RNO?

- A. 420 gpm
- B. 350 gpm
- C. 330 gpm
- D. 300 gpm

Proposed Answer: C. 330 gpm

#### Explanation:

- A. Incorrect. Time of 40 minutes (time from entry into ECA-1.1)
- B. Incorrect. Time of 70 minutes (time from small LOCA)
- C. Correct. Time of trip, 90 minutes
- D. Incorrect. Scale begins at 10 minutes, if it is not noticed, time used would be at the 100 minute mark

**Technical References:** ECA-1.1 appendix G

**References to be provided to applicants during exam:** ECA-1.1 appendix G

**Learning Objective:** 42460 - Explain basis of emergency steps of ECA-1.1

<b>Question Source:</b>	Bank #74 Sequoyah 2008	X
(note changes; attach parent)	Modified Bank #	
	New	
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.7	

Examination Outline Cross-Reference	Level	RO
<b>Knowledge of the interrelations between Emergency Boration and the following: Pumps</b>	<b>Tier #</b>	1
	<b>Group #</b>	2
	<b>K/A #</b>	APE 024 AK2.04
	<b>Rating</b>	2.6

### Question 19

GIVEN:

- Following a refueling, the reactor reaches criticality below the RIL during the startup
- In accordance with OP L-2, Hot Standby to Startup Mode, the crew is to emergency borate approximately 900 gallons of 4% boric acid per OP AP-6, Emergency Boration
- Both Boric acid transfer pumps trip when attempting to initiate emergency boration
- Emergency Borate valve, CVCS-8104, fails to open

The operator is to emergency borate using the RWST.

What is the approximate length of time the operator will need to borate using the RWST as a suction source for the Charging pumps at the minimum flow required by OP AP-6?

- 4% Boric Acid Storage Tank boron concentration = 7200 ppm
- RWST boron concentration = 2400 ppm

- A. 10 minutes
- B. 18 minutes
- C. 30 minutes
- D. 90 minutes

Proposed Answer: C. 30 minutes

#### Explanation:

- A. Incorrect. 900 gallons of 4% (BAST) at 90 gpm.
- B. Incorrect. Procedure states FI pegs at 50 gpm. If 50 gpm is assumed, 900 gallons is 18 minutes.
- C. Correct. Need 3 times the flow, 2700 gallons, but at 3 times the rate, 90 gpm, or 30 minutes.
- D. Incorrect. Conversion incorrectly assumes flow is 30 gpm for 2700 gallons.

**Technical References:** OP L-2 and OP AP-6

**References to be provided to applicants during exam:** None

**Learning Objective:** 4149 - Explain the Emergency Boration process

**Question Source:** Bank # P-34018 X  
(note changes; attach parent) Modified Bank #

**Question History:** Last NRC Exam No

**Question Cognitive Level:** Memory/Fundamental  
Comprehensive/Analysis X

**10CFR Part 55 Content:** 55.41.5

Examination Outline Cross-Reference	Level	RO
Ability to determine and interpret the following as they apply to the Fuel Handling Incidents: Magnitude of potential radioactive release	Tier #	1
	Group #	2
	K/A #	APE 036 AA2.03
	Rating	3.1

### Question 20

Which of the following requirements for the spent fuel pool is designed to remove 99% of all the iodine activity that could be released from the rupture of an irradiated fuel assembly?

- A. Eight feet of water over the top of a fuel assembly being moved
- B. 23 feet of water above the top of stored fuel
- C. Boron concentration of at least 2000 ppm
- D. Spent fuel stored in correct locations

Proposed Answer: B. 23 feet of water above the top of stored fuel

#### Explanation:

- A. Incorrect. Eight feet is maintained to provide shielding during the fuel movement. In fact, with eight feet of water above the top of a moving assembly, there is greater than 23 feet above the top of the fuel
- B. Correct. 23 feet is to ensure a removal of approximately 99% of the iodine released from a rupture of all fuel rods in an assembly which is lying on top of the spent fuel
- C. Incorrect. Mitigate the effects of a misplaced fuel assembly
- D. Incorrect. Maintains an acceptable Keff.

**Technical References:** System Training Guide B7, Spent Fuel Pool System

**References to be provided to applicants during exam:** None

**Learning Objective:** 6612 - Explain the consequences of dropping a fuel assembly

**Question Source:**

(note changes; attach parent)

Bank #

Modified Bank #

New

X

**Question History:**

Last NRC Exam

No

**Question Cognitive Level:**

Memory/Fundamental

X

Comprehensive/Analysis

**10CFR Part 55 Content:**

55.41.9

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	RO
<b>Knowledge of the operational implications of the following concepts as they apply to Steam Generator Tube Leak: Use of steam tables</b>	<b>Tier #</b>	1
	<b>Group #</b>	2
	<b>K/A #</b>	APE 037 AK1.01
	<b>Rating</b>	2.9

**Question 21**

GIVEN:

- The crew is performing the actions of OP AP-3, Steam Generator Tube Leak
- The crew has just completed an RCS cooldown to the target temperature of 510°F
- RCS pressure is 1200 psig
- Ruptured steam generator pressure is 1015 psig

Approximately how much RCS subcooling currently exists?

- A. 35°F
- B. 38°F
- C. 55°F
- D. 58°F

Proposed Answer: D. 58°F

**Explanation:**

- A. Incorrect. Based on steam generator saturation temperature of 1000 psi (incorrectly converts to psia)
- B. Incorrect. Based on steam generator pressure plus 15 psi.
- C. Incorrect. RCS pressure minus 15 psi (incorrectly converts to psia)
- D. Correct. RCS pressure of 1215 psia is approximately 568.9°F or 58.9°F of subcooling

**Technical References:** Steam Tables, OP AP-3

**References to be provided to applicants during exam:** Steam Tables

**Learning Objective:** 65686 - Explain how steam tables are used in the Control Room.

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #47 DCPD 2/2009	X
	New	
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.5	

Examination Outline Cross-Reference	Level	RO
<b>Knowledge of the reasons for the following responses as they apply to the Accidental Liquid Radwaste Release: Actions contained in EOP for accidental liquid radioactive-waste release</b>	<b>Tier #</b>	1
	<b>Group #</b>	2
	<b>K/A #</b>	APE 059 AK3.04
	<b>Rating</b>	3.8

### Question 22

An LHUT has ruptured. The crew has entered OP AP-14, Tank Ruptures.

Which of the following describes how the Aux Building Ventilation system will be aligned?

- A. Ventilation will be secured to minimize the spread of contamination.
- B. “Safeguards Only with S” to route all exhaust through a monitored release path.
- C. “Safeguards Only with S” to route all exhaust flow through the charcoal filter and aid in cleanup of the release.
- D. “Safeguards Only” to isolate all air flow to Non-ESF areas of the Auxiliary Building, minimizing the spread of contamination.

Proposed Answer: C. “Safeguards Only with S” to route all exhaust flow through the charcoal filter and aid in cleanup of the release.

#### Explanation:

- A. Incorrect. Shutting down ventilation would stop the spread of airborne but switching to Safeguards with S will stop the airflow to the affected area of the aux building and place a charcoal filter in service.
- B. Incorrect. All exhaust is routed to the plant vent is continuously monitored.
- C. Correct. Air to the majority of the aux building is secured and the exhaust from the ESF areas is routed through a charcoal filter, to adsorb gaseous iodine (elemental iodine and organic iodides) from the Auxiliary Building exhaust, before discharging it to the environment, in order to reduce offsite radiation exposure following a RHR seal rupture accident.
- D. Incorrect. Procedure has the operator select “S” and place the system in “SAFEGUARDS.”

**Technical References:** System Training Guide H.1

**References to be provided to applicants during exam:** None

**Learning Objective:** 3477 - Given an abnormal condition, summarize the major actions of OP AP-14 to mitigate an event in progress.

#### Question Source:

(note changes; attach parent)

Bank #

Modified Bank #

New

X

#### Question History:

Last NRC Exam

No

#### Question Cognitive Level:

Memory/Fundamental

Comprehensive/Analysis

X

#### 10CFR Part 55 Content:

55.41.13

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	RO
<b>Knowledge of the interrelations between the Loss of Containment Integrity and the following: Personnel access hatch and emergency access hatch</b>	<b>Tier #</b>	2
	<b>Group #</b>	1
	<b>K/A #</b>	APE 069 AK2.03
	<b>Rating</b>	2.8

**Question 23**

Which of the following is done to ensure Containment integrity is satisfied in MODES 1 through 4?

- A. The time both the inner and outer doors of the Personnel Hatch can be opened is administratively controlled (constantly manned).
- B. An alarm informs the Control Room that one of the doors has remained open for an extended period of time (greater than 5 minutes).
- C. The air leakage past the seals for the airlocks is automatically tested each time the doors are operated and alarms if the measured flow rate is too high (greater than 1765 SCCM).
- D. If the integrity of airlock seals is potentially compromised as indicated by an inability to be pressurized to a sufficient pressure, (greater than 12 psig), the inner airlock door cannot be opened.

Proposed Answer: C. The air leakage past the seals for the airlocks is automatically tested each time the doors are operated and alarms if the measured flow rate is too high (greater than 1765 SCCM).

**Explanation:**

- A. Incorrect. Both doors are not allowed to be opened and mechanically interlock to prevent this.
- B. Incorrect. An alarm informs the control room when the door is open, but does not alarm after a set period of time.
- C. Correct. Excessive flow past the seals requires evaluation of containment leakage and is a violation of TS 3.6.2.
- D. Incorrect. The only interlock is the mechanical interlock preventing opening both doors at the same time.

**Technical References:** AR PK11-12, Tech Spec 3.6.1 and 3.9.2

**References to be provided to applicants during exam:** None

**Learning Objective:** 9697F - Apply TS 3.6 Technical Specification LCOs

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.12	

Examination Outline Cross-Reference	Level	RO
Ability to determine and interpret the following as they apply to SI Termination: Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.	Tier #	2
	Group #	1
	K/A #	E02 EA2.2
	Rating	3.5

### Question 24

Which of the following describes the sequence for stopping the ECCS CCPs and SI pumps in E-1.1, SI Termination?

- A. Stop One ECCS CCP, realign charging, restore letdown; then stop both SI pumps
- B. Stop both SI pumps, realign charging, restore letdown; then stop one ECCS CCP
- C. Stop one ECCS CCP; realign charging; then stop both SI pumps
- D. Stop both SI pumps and one ECCS CCP; then realign charging and restore letdown

Proposed Answer: C. Stop one ECCS CCP; realign charging; then stop both SI pumps

#### Explanation:

- A. Incorrect. letdown not restored until after the ECCS pumps are stopped (charging and SI)
- B. Incorrect. The SI pumps are not injecting, there is a need to secure them in E-1.1, however, the more pressing need is to stop the charging pumps and secure injection
- C. Correct. ECCS CCP is secured first to reduce SI injection flow, charging is realigned, then both SI pumps are stopped
- D. Incorrect. Order is backwards.

**Technical References:** E-1.1

**References to be provided to applicants during exam:** None

**Learning Objective:** 6745 - State the general sequence of ECCS reduction sequence

**Question Source:** Bank # P-33693 X  
 (note changes; attach parent) Modified Bank #  
 New

**Question History:** Last NRC Exam No

**Question Cognitive Level:** Memory/Fundamental X  
 Comprehensive/Analysis

**10CFR Part 55 Content:** 55.41.10

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	RO
<b>LOCA Cooldown and Depressurization: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.</b>	<b>Tier #</b>	1
	<b>Group #</b>	2
	<b>K/A #</b>	E03 G2.4.50
	<b>Rating</b>	4.2

**Question 25**

The crew is performing a cooldown in accordance with E-1.2, Post-LOCA Cooldown and Depressurization.

PK08-07, Lo-Lo Tavg Permissive P-12 alarms.

Which of the following states the setpoint for the alarm and the operator action required?

- A. 543°F; Take the Steam Dump Control switches to OFF/RESET
- B. 547°F; Take the Steam Dump Control switches to OFF/RESET
- C. 543°F; Take the Steam Dump Control switches to BYPASS INTLK
- D. 547°F; Take the Steam Dump Control switches to BYPASS INTLK

Proposed Answer: C. 543°F; Take the Steam Dump Control switches to BYPASS INTLK

**Explanation:**

- A. Incorrect. Setpoint is correct, however, going to OFF/RESET defeats arming signals and resets bypass.
- B. Incorrect. 547 is no load Tave, action is incorrect.
- C. Correct. Setpoint is correct. Switches taken (momentarily) to Bypass to allow opening one set of steam dumps.
- D. Incorrect. 547 is no-load Tave, not setpoint.

**Technical References:** E-1.2, AR PK08-07

**References to be provided to applicants during exam:** None

**Learning Objective:** 37810 - Describe controls, indications, and alarms associated with the Steam Dump System

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.7	



Examination Outline Cross-Reference	Level	RO
<b>Knowledge of the reasons for the following responses as they apply to the (Natural Circulation with Steam Void in Vessel with/without RVLIS): Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics.</b>	<b>Tier #</b>	1
	<b>Group #</b>	2
	<b>K/A #</b>	E10 EK3.1
	<b>Rating</b>	3.4

### Question 26

A natural circulation cooldown is being performed in accordance with EOP E-0.3, Natural Circulation Cooldown with Steam Void in the Reactor Vessel (with RVLIS).

Which of the following describes why the RVLIS Upper Range indication must be maintained greater than 76% during the cooldown?

- A. To maintain conditions for an RCP start should power become available.
- B. To ensure the core remains covered in the event T-hot reaches saturated conditions.
- C. To prevent steam from entering the hot leg and disrupting the natural circulation flow.
- D. To ensure that when the void in the upper head is collapsed, Pressurizer level will remain on scale.

Proposed Answer: C. To prevent steam from entering the hot leg and disrupting the natural circulation flow.

#### Explanation:

- A. Incorrect. A continuous action in the natural circulation cooldown procedures is to start an RCP.
- B. Incorrect. This is a non-accident procedure, core cooling is not a concern.
- C. Correct. If steam enters the hot legs, it would most likely be condensed by the subcooled hot leg water well before the relatively slow natural circulation flow can carry it to the SG U-tubes, therefore some voiding into the RCS hot legs should not impede the natural circulation cooling process. Even if steam were to reach the SG U-tubes, the condensation rate of steam in the U-tubes is more rapid than in the subcooled loop so significant degradation of the natural circulation process should not occur. By monitoring RVLIS and limiting the void growth to the top of the hot legs (repressurizing the RCS if necessary), the potential for introducing voids into the SG U-tubes is minimized. No uncertainty is applied to the nominal value to preclude a bias toward either preventing void growth to the top of the hot legs, or allowing excessive steam to enter the hot legs. This is considered a reasonable balance between the benefit of enabling effective upper head drain and fill cooling and the potential consequence of allowing steam to enter the hot legs.
- D. Incorrect. Pressurizer level is controlled to accommodate void growth.

**Technical References:** E-0.3, E-0.3 background

**References to be provided to applicants during exam:** None

**Learning Objective:** 5855 - Explain the effect of voids on RCS operation

**Question Source:** Bank # P-49484 X  
 (note changes; attach parent) Modified Bank #  
 New

<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	X
<b>10CFR Part 55 Content:</b>	Comprehensive/Analysis	
	55.41.10	

Examination Outline Cross-Reference	Level	RO
Ability to operate and / or monitor the following as they apply to the (High Containment Radiation): Operating behavior characteristics of the facility.	Tier #	1
	Group #	2
	K/A #	E16 EA1.2
	Rating	2.9

### Question 27

Which of the following explains why the steam line radiation monitors, RM-71, 72, 73 and 74 may increase during a large break LOCA?

- A. The monitors are responding to containment “shine”.
- B. Particulate from the RCS enters the secondary through the U-tubes.
- C. Radiation from containment causes the radiation monitors to fail high.
- D. Elevated temperatures in the area cause the radiation monitors to fail high.

Proposed Answer: A. The monitors are responding to containment “shine”.

#### Explanation:

- A. Correct. High radiation levels in Containment due to clad or fuel failure can increase process monitor readings outside Containment due to increased background levels. Normally the Containment wall will provide 105 reduction in radiation levels, however, RM-71 to 74 can be exposed to Containment radiation by streaming through the main steam lines causing an increase in ALL steam line radiation monitors. This should not be interpreted as a S/G tube rupture if there are other indications of LOCA or fuel damage that can explain the high level reading in all steam lines.
- B. Incorrect. A tube rupture has not occurred.
- C. Incorrect. Monitors have not failed
- D. Incorrect. Monitors have not failed.

**Technical References:** G4A – Radiation Monitoring page 3-15

**References to be provided to applicants during exam:** None

**Learning Objective:** 8485 Explain the conditions that effect Radiation Monitoring system radiation monitor indications

<b>Question Source:</b>	Bank # 24 DCPD 2/2009	X
(note changes; attach parent)	Modified Bank #	
	New	
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.9	

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	RO
<b>Knowledge of bus power supplies to the following: RCPs</b>	<b>Tier #</b>	2
	<b>Group #</b>	1
	<b>K/A #</b>	003 K2.01
	<b>Rating</b>	3.1

**Question 28**

12kV Bus "E" de-energizes on Unit 2.

Which RCPs have lost power?

- A. RCPs 2-1 and 2-3
- B. RCPs 2-1 and 2-4
- C. RCPs 2-2 and 2-4
- D. RCPs 2-3 and 2-4

Proposed Answer: A. Loss of RCPs 2-1 and 2-3.

**Explanation:**

- A. Correct. Bus E supplies power to RCPs 1 and 3.
- B. Incorrect. RCPs 1 and 3
- C. Incorrect. RCPs supplied by bus D.
- D. Incorrect. RCPs 1 and 3

**Technical References:** OIM J-1-1

**References to be provided to applicants during exam:** None

**Learning Objective:** 6080 - State the power supplies to RCP components

**Question Source:**

(note changes; attach parent)

Bank #

Modified Bank #

New

X

**Question History:**

Last NRC Exam

No

**Question Cognitive Level:**

Memory/Fundamental

X

Comprehensive/Analysis

**10CFR Part 55 Content:**

55.41.7

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	<b>RO</b>
<b>Knowledge of the effect of a loss or malfunction on the following CVCS components: Methods of pressure control of solid plant (PZR relief and water inventory)</b>	<b>Tier #</b>	2
	<b>Group #</b>	1
	<b>K/A #</b>	004 K6.26
	<b>Rating</b>	3.8

**Question 29**

GIVEN:

- The plant is in MODE 5
- The RCS is solid
- One train of RHR in service
- Letdown Pressure Control Valve, PCV-135, is in AUTO
- RHR Letdown Flow is being maintained via CVCS HCV-133
- Charging pump 1-3 is in service

The running RHR pump trips.

Which of the following describes the response of PCV-135 and RCS pressure?

- A. PCV-135 opens more and RCS pressure LOWERS.
- B. PCV-135 opens more and maintains RCS pressure stable.
- C. PCV-135 closes down and RCS pressure RISES.
- D. PCV-135 closes down and maintains RCS pressure stable.

Proposed Answer: C. PCV-135 closes down and RCS pressure RISES.

**Explanation:**

- A. Incorrect. Letdown pressure will lower due to the loss of the RHR pump. PCV-135 will close down to maintain pressure, reducing letdown flow, with charging in service, RCS pressure will increase.
- B. Incorrect. With charging in service, pressure will increase.
- C. Correct. Charging will continue to add inventory. With the RCS solid and charging greater than letdown, pressure will increase.
- D. Incorrect. Pressure will increase.

**Technical References:** A-2:I

**References to be provided to applicants during exam:** None

**Learning Objective:** 9668 - Explain precautions and limitations within operating procedures

<b>Question Source:</b>	Bank # 3 DCPD 4/2007	X
(note changes; attach parent)	Modified Bank #	
	New	
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.5	

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	RO
<b>Knowledge of the physical connections and/or cause-effect relationships between the RHRS and the following systems: RWST</b>	<b>Tier #</b>	2
	<b>Group #</b>	1
	<b>K/A #</b>	005 K1.11
	<b>Rating</b>	3.5

**Question 30**

A large break LOCA has occurred.

Both RHR pumps have tripped due to low RWST level.

Which of the following actions must be done in order to restart the RHR pumps for Cold Leg Recirculation?

- A. Reset SI.
- B. Close 8980, RWST to RHR pump suction.
- C. Place one “RWST LOW LEVEL TRIP TEST SWITCH” in CUTOUT.
- D. Open 8982 A or B, Containment Recirc Sump suction to RHR, for the applicable pump.

Proposed Answer: D. Open 8982 A or B, Containment Recirc Sump suction to RHR, for the applicable pump.

**Explanation:**

- A. Incorrect. Resetting SI is necessary to regain control of many ECCS controls, but the RHR pump interlock with Low RWST is not one of them.
- B. Incorrect. Closing the RWST suction source does not remove/disable the trip on Low Level.
- C. Incorrect. 2 of the 3 switches must be in CUT OUT to disable/remove the trip.
- D. Correct. The trip is defeated if:
  - 2 of 3 trip switches are in CUTOUT (and 8982A/B are closed) or
  - **8982 A/B are open (breaks low level trip logic), or**
  - The control switch is in LOCAL

**Technical References:** AR PK03-04, System lesson LB-2

**References to be provided to applicants during exam:** None

**Learning Objective:** 35317 - Analyze automatic features and interlocks associated with the RHR System

<b>Question Source:</b>	Bank # A-0740	X
(note changes; attach parent)	Modified Bank #	
	New	
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.10	

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	<b>RO</b>
<b>Knowledge of the effect of a loss or malfunction of the following will have on the ECCS: Valves</b>	<b>Tier #</b>	2
	<b>Group #</b>	1
	<b>K/A #</b>	006 K6.10
	<b>Rating</b>	2.6

**Question 31**

A safety injection occurs.

VCT outlet valve, LCV-112B fails to close due to thermal overload.

Which of the following describes the suction source, if any, currently aligned to the charging pumps?

- A. There is no suction source for the charging pumps.
- B. Both the RWST and the VCT are aligned to the suction of the charging pumps.
- C. The VCT to the charging pumps is isolated; only one of the RWST to the charging pump suction valves (8805A/B) is open.
- D. The VCT to the charging pumps is isolated; both of the RWST to the charging pump suction valves (8805A/B) are open.

Proposed Answer: D. The VCT to the charging pumps is isolated; both of the RWST to the charging pump suction valves (8805A/B) are open.

**Explanation:**

- A. Incorrect. The RWST valves open on SI (and VCT valves close).
- B. Incorrect. The VCT valves, 112B and C are in series, only one needs to close to isolate the VCT.
- C. Incorrect. The RWST and VCT valves are not interlocked. SI opens the RWST valves.
- D. Correct. Both RWST valves will open on SI (and are in parallel), the VCT is isolated by 112C.

**Technical References:** System Training Guide B3, Drawing 441318

**References to be provided to applicants during exam:** None

**Learning Objective:** 8045 - Analyze automatic features and interlocks associated with the ECCS

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.7	

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	RO
<b>Knowledge of the effect that a loss or malfunction of the PRTS will have on the following: Containment</b>	<b>Tier #</b>	2
	<b>Group #</b>	1
	<b>K/A #</b>	007 K3.01
	<b>Rating</b>	3.3

**Question 32**

A Pressurizer PORV has failed open and cannot be isolated, causing a reactor trip and safety injection.

As the crew transitions from E-0, Reactor Trip or Safety Injection, the following conditions exist:

- PRT pressure after an initial rapid rise is now 5 psig and lowering
- Tailpipe temperature after an initial rise is now 240°F and lowering
- RCS pressure is 1800 psig and lowering

Which of the following describes the expected current Containment conditions?

- A. Normal Containment parameters
- B. Rising radiation levels and temperatures
- C. Rising radiation levels; lowering temperatures after an initial rapid rise
- D. After an initial rapid rise, lowering radiation levels and temperatures

DELETED FROM EXAM

Proposed Answer: B. Rising radiation levels and temperatures

**Explanation:**

A. Incorrect. The PRT rupture disk has ruptured, and containment parameters are now being affected.

There are 3 possible answers due to improper focus in the stem. The question has been deleted.

The rupture disk has ruptured causing PRT pressure to lower from its peak of pressure lowers, tailpipe temperature will lower as well (to saturation for e. Containment temperature and radiation levels will rise. Containment pressure will be rising as the contents of the PRT are relieved to containment parameters will not follow PRT parameters. They will still be rising.

**Technical References:** steam tables,

**References to be provided to applicants during exam:** None

**Learning Objective:** 4950 - Explain the operation of PRT system

**Question Source:**

(note changes; attach parent)

Bank #

Modified Bank #

New

X

**Question History:**

Last NRC Exam

No

**Question Cognitive Level:**

Memory/Fundamental

Comprehensive/Analysis

X

**10CFR Part 55 Content:**

55.41.7

Changed to Press. PORV has failed open to clearly make D incorrect (and not argue that if the PORV was closed and RCS press. lowering due to SI cooldown.



Examination Outline Cross-Reference	Level	RO
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCWS controls including: CCW temperature	Tier #	2
	Group #	1
	K/A #	008 A1.02
	Rating	2.9

### Question 33

GIVEN:

- Unit 1 is at 100% power.
- One CCW heat exchanger is in service.
- Containment temperature is 80°F
- Spent Fuel Pool temperature is 85°F

CCW heat exchanger outlet temperature is 95°F and rising. The crew is going to reduce CCW loads in accordance with OP AP-11, Malfunction of Component Cooling Water system, Appendix B, CCW Heat Load Isolation.

Which of the following CCW heat loads could the operators isolate to lower CCW temperature while the unit is at power?

- Spent Fuel Pool Heat Exchanger
- Containment Fan Cooler Units
- Seal Water Heat Exchanger
- RCP oil coolers

Proposed Answer: C. Seal Water Heat Exchanger

#### Explanation:

- Incorrect. SFP heat exchanger may act as a heat sink and should not be isolated.
- Incorrect. CFCUs may act as a heat sink and should not be isolated.
- Correct. CCW isolated and excess letdown placed in service.
- Incorrect. Not isolated.

**Technical References:** OP AP-11 Appendix B

**References to be provided to applicants during exam:** None

**Learning Objective:** 3466 - Discuss the effects and actions associated with a loss of CCW

<b>Question Source:</b>	Bank #7 DCPD 4/2007	X
(note changes; attach parent)	Modified Bank	
	New	
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.5	

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	RO
<b>Knowledge of the physical connections and/or cause-effect relationships between the PZR PCS and the following systems: RCS</b>	<b>Tier #</b>	2
	<b>Group #</b>	1
	<b>K/A #</b>	010 K1.03
	<b>Rating</b>	3.6

**Question 34**

Unit 1 is at full power.

Which of the following will cause the RCS to operate closer to DNB?

- A. The controlling Pressurizer pressure channel fails low.
- B. Output of the Master Pressurizer Pressure controller, HC-455K, fails low.
- C. The operator lowers the setpoint of the Master Pressurizer Pressure controller, HC-455K, to 6.5 turns.
- D. The operator places the Master Pressure Controller, HC-455K, in MANUAL and presses the DECREASE pushbutton.

Proposed Answer: C. The operator lowers the setting of the Pressurizer Pressure controller to 6.5 turns.

**Explanation:**

- A. Incorrect. Controlling channel failing low will cause heaters to energize and raise pressure to PORV setpoint, further from DNB.
- B. Incorrect. Output low will cause heaters to energize and raise pressure.
- C. Correct. Pref will lower and the controller will act to lower pressure to setpoint and the RCS will be closer to DNB.
- D. Incorrect. This will lower the OUTPUT and cause heaters to energize.

**Technical References:** System Training Guide A4A, OIM A-4-4a, A-4-4b

**References to be provided to applicants during exam:** None

**Learning Objective:** 36924 - Describe controls, indications, and alarms associated with the Pzr, Pzr Pressure and Level Control System

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.7	
Changed C from “setting” to “setpoint”		

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	RO
<b>Ability to manually operate and/or monitor in the control room: Channel blocks and bypasses</b>	<b>Tier #</b>	2
	<b>Group #</b>	1
	<b>K/A #</b>	012 A4.03
	<b>Rating</b>	3.6

### Question 35

Unit 1 is at 90% power. Control Bank D is at 210 steps.

Power Range channel N44 fails high. The crew places rod control in Manual to stop the auto rod motion.

What further action, if any, is required to restore the crew's ability to manually withdraw the control rods?

- A. None, the rods may be withdrawn in MANUAL to ARO.
- B. Bypass C-2, Power Range Rod Stop, and reset the Positive Rate trip for N44.
- C. Bypass C-2, Power Range Rod Stop and Bypass C-3, OTΔT Rod Stop.
- D. Bypass C-2, Power Range Rod Stop only.

Proposed Answer: D. Bypass C-2, Power Range Rod Stop only.

#### Explanation:

- A. Incorrect. C-2 actuates on 1 of 4 greater than 102% and blocks auto and manual rod withdrawal.
- B. Incorrect. The rate trip does not need to be reset to withdraw rods.
- C. Incorrect. The C-3 rod stop is 2 of 4 and is not actuated.
- D. Correct. The Power Range Overpower Rod Stop must be bypassed at the Misc. Control and Indication drawer on the NI Rack.

**Technical References:** OIM B-6-3, B-6-3a

**References to be provided to applicants during exam:** None

**Learning Objective:** 37048 - Analyze automatic features and interlocks associated with the RPS

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank # S-47156	X
	New	
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.7	

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	RO
<b>Knowledge of the operational implications of the following concepts as the apply to the RPS: Power Density</b>	<b>Tier #</b>	2
	<b>Group #</b>	1
	<b>K/A #</b>	012 K5.02
	<b>Rating</b>	3.3

**Question 36**

Which of the following reactor trips is designed to prevent operation of the plant with a power density of greater than 21.1 kW/foot?

- A. OTΔT
- B. OPΔT
- C. Power Range Rate trip (Positive)
- D. Power Range High Flux (Low)

Proposed Answer: B. OPΔT

**Explanation:**

- A. Incorrect. OTΔT is DNB protection.
- B. Correct. OPΔT are for excessive kw/foot (power density).
- C. Incorrect. PR rate (positive) is for ejected rod (flux peaking)
- D. Incorrect. High flux low is protection from low power startup accidents.

**Technical References:** System Training Guide B-6A

**References to be provided to applicants during exam:** None

**Learning Objective:** 9941 - State the purpose of RPS components

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	X
	Comprehensive/Analysis	
<b>10CFR Part 55 Content:</b>	55.41.2	

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	<b>RO</b>
<b>Ability to manually operate and/or monitor in the control room: ESFAS-initiated equipment which fails to actuate</b>	<b>Tier #</b>	2
	<b>Group #</b>	1
	<b>K/A #</b>	013 A4.01
	<b>Rating</b>	4.5

### Question 37

Unit 1 is at 25% power.

A small break LOCA causes a reactor trip and Safety Injection actuation.

While performing actions in E-0, Reactor Trip or Safety Injection, the operator notes the following AFW pump status:

- AFW pump 1-1 – Not running
- AFW pump 1-2 – Running
- AFW pump 1-3 – Not running
- Steam Generator Narrow Range levels have remained greater than 50%

Which of the following describes what the status of the AFW pump should be for the current plant conditions?

- A. AFW pump 1-1 should have also started
- B. AFW pump 1-3 should also have started
- C. AFW pumps 1-1 and 1-3 should also have started
- D. None of the AFW pumps should have started

Proposed Answer: B. AFW pump 1-3 should have also started

#### **Explanation:**

- A. Incorrect. If the 1-1 and 1-3 are reversed, the candidate could believe the 1-1 is the motor driven pump and should be running. AFW 1-1 is a turbine driven AFW pump which not receive a start signal on the SI.
- B. Correct. Both MDAFW pumps receive start signals on SI. The TDAFW does not.
- C. Incorrect. Only the motor driven pumps will be started. The TDAFW starts on low-low on 2 of 4 steam generators
- D. Incorrect. SI starts the motor driven pumps. If the candidate focuses on the steam generator levels, this would be correct.

**Technical References:** OIM D-1-2

**References to be provided to applicants during exam:** None

**Learning Objective:** 37637 - Analyze automatic features and interlocks associated with the AFW system

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X



<b>Examination Outline Cross-Reference</b>	<b>Level</b>	RO
<b>Knowledge of ESFAS design feature(s) and/or interlock(s) which provide for the following: Vital power load control</b>	<b>Tier #</b>	2
	<b>Group #</b>	1
	<b>K/A #</b>	013 K4.11
	<b>Rating</b>	3.2

**Question 38**

Under what condition(s) are Residual Heat Removal pumps sequentially loaded onto their respective Vital 4 kV buses?

- A. Only for Transfer to Diesel with SI
- B. For either Transfer to Startup with SI or Transfer to Diesel with SI
- C. Only for Transfer to Diesel if the pumps had been previously running and SI has not been reset
- D. Only for Transfer to Diesel if the pumps had been previously running, regardless of the status of the SI signal

Proposed Answer: B. For either Transfer to Startup with SI or Transfer to Diesel with SI

**Explanation:**

- A. Incorrect. Also sequentially loads for transfer to startup. The loads are not “block loaded”.
- B. Correct. A transfer to startup or diesel, with SI will sequentially load the RHR pumps after the charging pumps.
- C. Incorrect. The pumps sequence on for transfer to startup or diesel, if SI is present. Running status is not a factor.
- D. Incorrect. This is the condition operators must be aware of in the EOPs. If SI is reset and there is a loss of power, the pumps must be restarted.

**Technical References:** OIM J-6-1

**References to be provided to applicants during exam:** None

**Learning Objective:** 9305 - State the purpose of the Electric Power Transfer System components

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	X
	Comprehensive/Analysis	
<b>10CFR Part 55 Content:</b>	55.41.7	

Examination Outline Cross-Reference	Level	RO
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: Fan motor thermal overload/high-speed operation	Tier #	2
	Group #	1
	K/A #	022 A2.03
	Rating	2.6

### Question 39

GIVEN:

- A large break LOCA occurs
- The operator is performing Appendix E of E-0, Reactor Trip or Safety Injection
- The operator notes the following indication for the CFCUs:
  - CFCUs 14 and 15 are running in HIGH
  - CFCUs 11, 12 and 13 are running in LOW

Which of the following actions will be taken by the operator?

- A. Shift CFCUs 14 and 15 to LOW speed to prevent the fans tripping on overcurrent.
- B. Shift CFCUs 14 and 15 to LOW speed to prevent CCW temperature exceeding 140°F.
- C. Shift CFCUs 11, 12 and 13 to HIGH speed to prevent exceeding Containment design pressure.
- D. Shift CFCUs 11, 12 and 13 to HIGH speed to prevent the fans tripping on overcurrent.

Proposed Answer: A. Shift CFCUs 14 and 15 to LOW speed to prevent the fans tripping on overcurrent.

#### Explanation:

- A. Correct. CFCUs are run in slow speed to prevent the fans tripping on overcurrent due to the adverse containment temperature.
- B. Incorrect. CCW is used to cool the CFCUs and design temperature is 140°F. a design feature of CCW is to provide cooling to ESF equipment, including all CFCUs during a LOCA.
- C. Incorrect. Appendix E verifies CFCU lights OFF. If lit, the action is to make the light go out by taking the CFCU to low speed. CFCUs run during a LOCA to maintain temperature less than 120°F which keeps containment from overpressurizing, but in low to keep the fans from tripping on overcurrent.
- D. Incorrect. Fans will trip on OC if running in HIGH, not LOW.

**Technical References:** E-0 Appendix E, System Training Guides F-2 and H-2

**References to be provided to applicants during exam:** None

**Learning Objective:** 48812 - Discuss abnormal conditions associated with the CFCUs

<b>Question Source:</b>	Bank # 15 DCP 2/2009	X
(note changes; attach parent)	Modified Bank #	
	New	
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.8	





<b>Examination Outline Cross-Reference</b>	<b>Level</b>	<b>RO</b>
<b>Knowledge of bus power supplies to the following: Containment Spray Pumps</b>	<b>Tier #</b>	2
	<b>Group #</b>	1
	<b>K/A #</b>	026 K2.01
	<b>Rating</b>	3.4

**Question 40**

GIVEN:

- A loss of offsite power and LOCA has occurred on Unit 2
- Containment Spray has actuated

Which Unit 2 Diesel Emergency Generators (D/G) will be powering the Containment Spray pumps?

- A. D/G 2-1 for Containment Spray pump 2-1 and D/G 2-2 for Containment Spray pump 2-2.
- B. D/G 2-1 for Containment Spray pump 2-2 and D/G 2-2 for Containment Spray pump 2-1.
- C. D/G 2-1 for Containment Spray pump 2-1 and D/G 2-3 for Containment Spray pump 2-2.
- D. D/G 2-1 for Containment Spray pump 2-2 and D/G 2-3 for Containment Spray pump 2-1.

Proposed Answer: A. D/G 2-1 for Containment Spray pump 2-1 and DEG 2-2 for Containment Spray pump 2-2.

**Explanation:**

- A. Correct. Unit 2 diesels 2-1 and 2-2 power containment spray pumps 2-1 and 2-2
- B. Incorrect. Unit 1 diesels 1-1 and 1-2 power containment spray pumps 1-2 and 1-1 (reversed)
- C. Incorrect. Diesel 2-3 does not power a spray pump.
- D. Incorrect. Diesel 2-3 does not power a spray pump.

**Technical References:** OIM J-1-1

**References to be provided to applicants during exam:** None

**Learning Objective:** 6022 - State the power supplies to CSS components

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	X
	Comprehensive/Analysis	
<b>10CFR Part 55 Content:</b>	55.41.8	
<u>Note</u> – Unit difference		

Examination Outline Cross-Reference	Level	RO
<b>Knowledge of the operational implications of the following concepts as the apply to the MRSS: Effect of steam removal on reactivity</b>	<b>Tier #</b>	2
	<b>Group #</b>	1
	<b>K/A #</b>	039 K5.08
	<b>Rating</b>	3.6

### Question 41

What reactor power and core life conditions would provide the GREATEST amount of positive reactivity for a faulted steam generator event?

- A. Beginning of core life, hot zero power
- B. Beginning of core life, 100% power
- C. End of core life, hot zero power
- D. End of core life, 100% power

Proposed Answer: C. End of core life, hot zero power

#### Explanation:

- A. Incorrect. EOL is worse case.
- B. Incorrect. EOL and no load worse case
- C. Correct. EOL, no load power (and power available) is worse case
- D. Incorrect. No load is worse case

**Technical References:** FSAR chapter 15.4, LTAA6

**References to be provided to applicants during exam:** None

**Learning Objective:** 5461 - Explain the plant's response to a faulted S/G.

<b>Question Source:</b>	Bank # R-34296	X
(note changes; attach parent)	Modified Bank #	
	New	
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	X
	Comprehensive/Analysis	
<b>10CFR Part 55 Content:</b>	55.41.1	

Examination Outline Cross-Reference	Level	RO
Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Overfeeding event	Tier #	2
	Group #	1
	K/A #	059 A2.03
	Rating	2.7

### Question 42

Unit 1 is raising power and currently at 25% power.

A malfunction results in overfeeding Steam Generator 1-2 and narrow range level rises to 95%.

Which of the following describes the initial plant and operator response?

- A. The reactor trips automatically due to a P-14 signal; the crew will perform the immediate actions of E-0, Reactor Trip or Safety Injection.
- B. The reactor trips automatically due to the automatic turbine trip; the crew will perform the immediate actions of E-0, Reactor Trip or Safety Injection.
- C. No automatic reactor trip will occur, however, feedwater is isolated and there is no running Main Feedwater pump; a manual reactor trip should be performed by the crew.
- D. No automatic reactor trip will occur, however, the main turbine has tripped; the crew will go to OP AP-29, Main Turbine Malfunction, to stabilize the plant.

Proposed Answer: C. No automatic reactor trip will occur, however, feedwater is isolated and there is no running Main Feedwater pump; a manual reactor trip should be performed by the crew.

### Explanation:

- A. Incorrect. P-14 does not directly trip the reactor. It does trip the feed pumps, the main turbine and isolate feedwater. If above P-9, the reactor would trip due to the turbine trip, or at high power due to low steam generator levels in the steam generators that lower to due to a loss of feedwater.
- B. Incorrect. If above P-9, this would be correct. At 25%, no automatic trip will initially occur.
- C. Correct. Feedwater will be isolated and the feed pump that had been running will be tripped. Steam generator levels will lower, the reactor should be tripped.
- D. Incorrect. The main turbine will be tripped, however, feedwater is isolated. If the candidate fails to make the FWI connection, this would be an appropriate response.

**Technical References:** OIM B-6-12

**References to be provided to applicants during exam:** None

**Learning Objective:** 37051 - Discuss abnormal conditions associated with the RPS

37048 - Analyze automatic features and interlocks associated with the RPS

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X



<b>Examination Outline Cross-Reference</b>	<b>Level</b>	RO
<b>Main Feedwater: Ability to recognize system parameters that are entry-level conditions for Technical Specifications.</b>	<b>Tier #</b>	2
	<b>Group #</b>	1
	<b>K/A #</b>	059 G2.2.42
	<b>Rating</b>	3.9

**Question 43**

Unit 1 is holding at 28% power, performing required testing following a refueling. The main Feedwater reg and bypass valves are open to control steam generator level.

Instrument air is lost to Main Feedwater Reg Bypass valve to Steam Generator 1-1, FCV-1510.

Which of the following describes the impact on the valve and on Technical Specifications, if any?

- A. The valve fails open; no impact on Technical Specifications, Main Feedwater valves (Main Feed Reg, Bypass and Isolation) are not covered by Technical Specifications.
- B. The valve fails closed; no impact on Technical Specifications, Main Feedwater valves (Main Feed Reg, Bypass and Isolation) are not covered by Technical Specifications.
- C. The valve fails open; evaluate the applicable LCO for OPERABILITY.
- D. The valve fails closed; evaluate the applicable LCO for OPERABILITY.

Proposed Answer: D. The valve fails closed; evaluate the applicable LCO for OPERABILITY.

**Explanation:**

- A. Incorrect. The valve fails closed and LCO 3.7.3 needs to be evaluated.
- B. Incorrect. Correct failure position, however, LCO 3.7.3 must be evaluated.
- C. Incorrect. The valve does not fail open, it fails closed.
- D. Correct. The valve fails closed and LCO 3.7.3 must be evaluated.

**Technical References:** Tech Spec 3.7.3, Lesson LC-8A

**References to be provided to applicants during exam:** None

**Learning Objective:** 9697G - Apply TS 3.7 Technical Specification LCOs

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.10	

Rev 1 – replaced question and changed correct answer to D vice C.

Examination Outline Cross-Reference	Level	RO
Knowledge of the operational implications of the following concepts as they apply to the AFW: Relationship between AFW flow and RCS heat transfer	Tier #	2
	Group #	1
	K/A #	061 K5.01
	Rating	3.6

#### Question 44

GIVEN:

- The crew is performing the actions of E-0.2, Natural Circulation Cooldown
- Cooldown rate is 10°F/hour
- RCS temperature is 520°F
- RCS pressure is 1200 psig
- Pressurizer level is 40%
- All MSIV's are open
- Narrow Range Steam Generator level is 10% in all generators and stable
- Steam dumps are in MANUAL
- AFW flow is being controlled in MANUAL and is 450 gpm total

What action can the operators take to enhance natural circulation?

- A. Close all MSIVs
- B. Raise AFW flow
- C. Lower Steam Dump controller setpoint
- D. De-energize Pressurizer heaters

Proposed Answer: B. Raise AFW flow

**Explanation:**

- A. Incorrect. Closing MSIVs will not affect natural circulation (and would isolate the currently working steam dumps). The 10% steam dumps would respond once pressure reached the no-load pressure, but this would lower the heat removal rate.
- B. Correct. Increasing AFW will increase steam generator level, cover the tubes and increase heat removal.
- C. Incorrect. While this would work if in Auto, no effect while in Manual
- D. Incorrect. This will lower pressure and decrease subcooling.

**Technical References:** Background E-0.2, E-0.2

**References to be provided to applicants during exam:** None

**Learning Objective:** 88905 - Explain the operator actions that can initiate or enhance natural circulation

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank # P-0909	X
	New	
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.4	

Examination Outline Cross-Reference	Level	RO
Knowledge of the physical connections and/or causeeffect relationships between the AFW and the following systems: S/G system	Tier #	2
	Group #	1
	K/A #	061 K1.01
	Rating	4.1

### Question 45

The plant trips from full power due to a steam break.

Plant conditions 5 minutes later:

- SI actuated
- MSI actuated
- FWI actuated
- All equipment operated as designed
- Steam Generator Pressures:
  - 11 – 900 psig, stable
  - 12 – 300 psig, lowering
  - 13 – 300 psig, lowering
  - 14 – 870 psig, stable

Which of the following is a possible location for the steam break that caused the plant to trip?

- A. Upstream of Steam Generator 12 MSIV, FCV-42
- B. Downstream of Steam Generator 13 MSIV, FCV-43
- C. Upstream of steam inlet valve to the TDAFW pump, FCV-95
- D. Downstream of steam inlet valve to the TDAFW pump, FCV-95

Proposed Answer: C. Upstream of steam inlet valve to the TDAFW pump, FCV-95

#### Explanation:

- A. incorrect, both 12 and 13 Steam Generator pressures are low. Check valves between the generators (off the AFW pump line) prevent both from depressurizing.
- B. incorrect, MSIVs are shut which would have isolated the break if this was the location.
- C. correct. Both steam generators feed the AFW pump. A break on the line upstream of the FCV would cause both to depressurize and cause the reactor trip.
- D. incorrect. The plant tripped due to the steam break. FCV-95 is normally closed. A break here would not have affected plant operation at power.

**Technical References:** OVID 106704 sheet 4

**References to be provided to applicants during exam:** None

**Learning Objective:** 7240 - Identify the location of Main Steam system valves and piping

**Question Source:** Bank # 48 L051 NRC exam 4/2007 X  
(note changes; attach parent) Modified Bank #

New  
**Question History:** Last NRC Exam No

**Question Cognitive Level:** Memory/Fundamental  
Comprehensive/Analysis X

**10CFR Part 55 Content:** 55.41.7



<b>Examination Outline Cross-Reference</b>	<b>Level</b>	RO
<b>AC Distribution: Operation of inverter (e.g., precharging synchronizing light, static transfer).</b>	<b>Tier #</b>	2
	<b>Group #</b>	1
	<b>K/A #</b>	062 A3.04
	<b>Rating</b>	2.7

**Question 46**

Unit 1 is at full power.

A problem with inverter IY-11 causes low inverter output voltage being sensed at output of the IY.

Which of the following will occur?

- A. The DC supply will become the source of power for the instrument bus.
- B. The static switch will automatically switch to the Bypass Regulating Transformer to power the instrument bus.
- C. The AC output breaker opens and the instrument bus will be de-energized until the PY is transferred to the Manual Bypass.
- D. The AC output breaker remains closed, however the instrument bus is de-energized until the PY is transferred to the Manual Bypass.

Proposed Answer: B. The static switch will automatically switch to the Bypass Regulating Transformer to power the instrument bus.

**Explanation:**

- A. Incorrect. Low output at the output means low AC and DC voltage.
- B. Correct. The function of the static switch is that if a fault, such as low inverter output, is detected, the inverter switches to the Backup Transformer. Once the inverter is restored, the static switch will switch back to normal. The Constant Voltage Transformer (CVT) alters the 140 VAC, 60 Hz square-wave input to a 120 VAC, 60 Hz sine-wave output.
- C. Incorrect. The inverter will switch automatically.
- D. Incorrect. The inverter will switch automatically.

**Technical References** AR PK19-19, STG J10 pages 2.1-8, 9 and 10

**References to be provided to applicants during exam:** None

**Learning Objective:** 3332 - Discuss abnormal conditions associated with the Instrument AC System

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	X
	Comprehensive/Analysis	
<b>10CFR Part 55 Content:</b>	55.41.7	

Examination Outline Cross-Reference	Level	RO
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ac distribution system controls including: Effect on instrumentation and controls of switching power supplies	Tier #	2
	Group #	1
	K/A #	062 A1.03
	Rating	2.5

### Question 47

Planned maintenance on an auxiliary feeder breaker results in its associated 4kV bus being placed on Startup Power and the Load Tap Changer (LTC) placed in MANUAL.

Which of the following can result if the LTC is returned to AUTO without having at least one 12kV bus supplied by Startup Power?

- A. Undervoltage of the associated 4kV or 480V equipment.
- B. Overcurrent of the associated 4kV or 480V equipment.
- C. Overvoltage of the associated 4kV or 480V equipment.
- D. Auto-start of the associated Diesel Emergency Generator.

Proposed Answer: C. Overvoltage of the associated 4 kV or 480 V equipment.

#### Explanation:

- A. Incorrect. If the LTC were to cause low side voltage to decrease, UV may occur.
- B. Incorrect. OC is a common trip for many breakers
- C. Correct. The LTC will act to raise voltage if there is no 12 kV bus to load the transformer. As a result, overvoltage could occur.
- D. Incorrect. If the condition caused the feeder breaker to trip the diesel would start.

**Technical References:** J-6A:II precautions and limitations

**References to be provided to applicants during exam:** None

**Learning Objective:** 37779 - Describe significant precautions and limitations associated with the 4KV System

<b>Question Source:</b>	Bank # R-60804	X
(note changes; attach parent)	Modified Bank #	
	New	
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.7	

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	RO
<b>Knowledge of the effect that a loss or malfunction of the DC electrical system will have on the following: ED/G</b>	<b>Tier #</b>	2
	<b>Group #</b>	1
	<b>K/A #</b>	063 K3.01
	<b>Rating</b>	3.7

**Question 48**

All diesels are running due to a safety injection.

A loss of vital 125 VDC bus 1-2 occurs. The crew is implementing OP AP-23, Loss of Vital DC Bus.

Which of the following actions will be taken by the crew for the affected diesel emergency generator?

- A. Dispatch a nuclear operator to locally secure the diesel using the Emergency Stop pushbutton.
- B. Place the diesel selector switch in MANUAL and shutdown the diesel.
- C. Dispatch a nuclear operator to perform shutdown checks on the diesel.
- D. Have control power transferred to backup and then place then shutdown the diesel.

Proposed Answer: D. Have control power transferred to backup and then place the diesel selector switch in MANUAL and shutdown the diesel.

**Explanation:**

- A. Incorrect. Loss of control power results in loss of control of the diesel until control power is transferred to backup. AP-23 directs the transfer to backup and then shutdown of the diesel. The emergency stop is used during an emergency condition, such as cardox discharge when it is dangerous to enter the room.
- B. Incorrect.. There is a loss of control, the diesel will not respond until control power is transferred.
- C. Incorrect. The diesel will not shutdown.
- D. Correct. Per AP-23, the crew transfers control power to backup (locally) and then shutdown the diesel and leaves it in MANUAL.

**Technical References:** OP AP-23 appendix B, system training guide LJ-6B

**References to be provided to applicants during exam:** None

**Learning Objective:** 5193 - Discuss abnormal conditions associated with the DC Power System

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.7	

Modified to more closely align to KA. Original focused on SSPS impact, now focuses on diesel

Examination Outline Cross-Reference	Level	RO
Ability to monitor automatic operation of the DC electrical system, including: Meters, annunciators, dials, recorders, and indicating lights	Tier #	2
	Group #	1
	K/A #	063 A3.01
	Rating	2.7

### Question 49

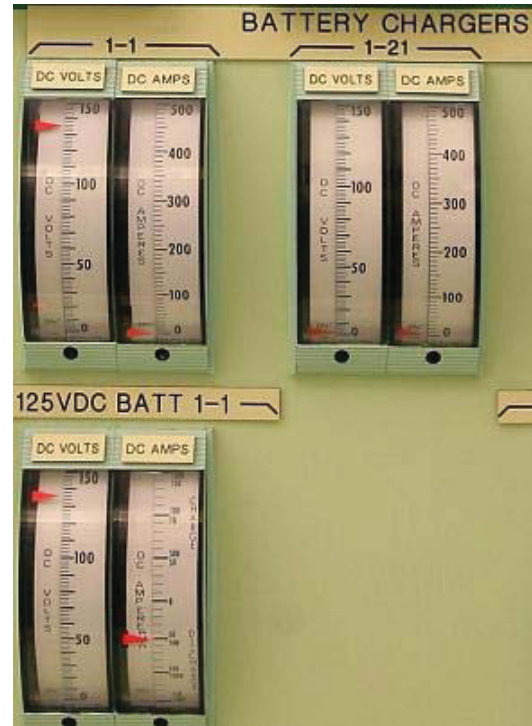
Unit 1 is in a normal full power lineup when a system transient occurs.

The following indications are available for DC Bus 11 on VB-5:

- BTC 1-1 - 135 VDC, 0 amps
- BTC 1-21, 0 VDC, 0 amps
- Battery 1-1, 135 VDC, 50 amps discharge

Which of the following could have caused the above indications?

- Loss of 480V bus 1G
- Loss of 480V bus 1H
- Equalizing charge on Battery 1-1
- Trip of the battery charger output breaker



Proposed Answer: D. Trip of the battery charger output breaker

#### Explanation:

- Incorrect. Bus G does not supply a backup charger and is the normal supply to charger 12
- Incorrect. Supply to BTC 1-21, this is the backup supply and not normally aligned.
- Incorrect, amps on BTC 1-1 would not be zero.
- Correct. Opening the output breaker of the normal BTC would cause the battery to begin supplying the DC bus.

**Technical References:** OIM J-1-2

**References to be provided to applicants during exam:** None

**Learning Objective:** 37793 - Describe controls, indications, and alarms associated with the DC Power System

<b>Question Source:</b>	Bank # R-73161	X
(note changes; attach parent)	Modified Bank #	
	New	
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.7	

Modified A due to indications are the same for either a loss of the 480 VAC or trip of the BC output breaker.

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	RO
<b>Knowledge of ED/G system design feature(s) and/or interlock(s) which provide for the following: ED/G fuel isolation valves</b>	<b>Tier #</b>	2
	<b>Group #</b>	1
	<b>K/A #</b>	064 K4.08
	<b>Rating</b>	2.9

**Question 50**

During a loss of offsite power, with all Diesel Generators (D/G) running, a fire breaks out in D/G 1-1 room.

What effect might this have on the Diesel Fuel Oil System?

- A. Fire protection relays will secure the fuel oil transfer pumps.
- B. Fire protection relays will close the fuel oil fill valves and dump the fuel racks.
- C. The D/G air receiver melt links will vent the air and dump the fuel racks.
- D. The Fuel Oil Day Tank Air Supply Valve melt links will close the fuel oil fill valves.

Proposed Answer: D. The Fuel Oil Day Tank Air Supply Valve melt links will close the fuel oil fill valves.

**Explanation:**

- A. Incorrect. There are no fire protection relays. Plausible because the goal is to stop fuel oil to the diesel.
- B. Incorrect. There are no fire protection relays. Plausible because the goal is to stop fuel oil to the diesel.
- C. Incorrect. Melt links installed on the fill valves, not the fuel racks.
- D. Correct. Melt link will cause the fill valves to close.

**Technical References:** STG J6B

**References to be provided to applicants during exam:** None

**Learning Objective:** 37725 - Analyze automatic features and interlocks associated with the Diesel Generator System

**Question Source:** Bank # A-0842 X  
 (note changes; attach parent) Modified Bank #  
 New

**Question History:** Last NRC Exam No

**Question Cognitive Level:** Memory/Fundamental X  
 Comprehensive/Analysis

**10CFR Part 55 Content:** 55.41.7

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	RO
<b>Knowledge of PRM system design feature(s) and/or interlock(s) which provide for the following: Release termination when radiation exceeds setpoint</b>	<b>Tier #</b>	2
	<b>Group #</b>	1
	<b>K/A #</b>	073 K4.01
	<b>Rating</b>	4.0

**Question 51**

A high radiation signal is detected by Steam Generator Blowdown radiation monitor, RE-23.

Steam generator overboard discharge will be diverted to which of the following locations?

- A. Floor Drain Receivers (FDRs)
- B. Equipment Drain Receivers (EDRs)
- C. Processed Waste Receivers (PWRs)
- D. Laundry and Distillate Tanks (L&DTs)

Proposed Answer: B. Equipment Drain Receivers (EDRs)

**Explanation:**

B. Correct. RE-23 isolates the overboard flowpath (RCV-17) and aligns to the EDR (opens FCV-477).

A, C and D incorrect. All are tanks in the liquid waste system but do not receive waste on a radiation alarm.

**Technical References:** OIM G-3-1

**References to be provided to applicants during exam:** None

**Learning Objective:** 8436 - Analyze automatic features and interlocks associated with the Liquid Radwaste System

**Question Source:** Bank # C-52318 X  
(note changes; attach parent) Modified Bank #

New  
**Question History:** Last NRC Exam No

**Question Cognitive Level:** Memory/Fundamental X  
Comprehensive/Analysis

**10CFR Part 55 Content:** 55.41.7

Examination Outline Cross-Reference	Level	RO
Ability to (a) predict the impacts of the following malfunctions or operations on the SWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of SWS	Tier #	2
	Group #	1
	K/A #	076 A2.01
	Rating	3.5

### Question 52

GIVEN:

- Unit 1 and Unit 2 are at full power
- ASW pump 1-1 is running
- ASW pump 1-2 is in AUTO
- ASW pump 2-1 is in AUTO
- ASW pump 2-2 is running

The following events occur on VB1:

- PK01-01, ASW SYS HX DELTA P/HDR PRESS (inputs 85 and 168 – Aux Salt Wtr to CCW Ht Exch 1-1 and 1-2 less than 40.5 psig)
- PK01-03, AUX SALT WATER PUMPS, (input 427 – Aux Salt Wtr Pps OC Trip)
- Blue and Green light lit for ASW pump 1-1
- Green light lit for ASW pump 1-2

Which of the following actions will be taken by the operator?

- Investigate the OC trip of ASW pump 1-1, if its an instantaneous or inverse time OC, attempt one start of the 1-1 ASW pump.
- Cross-tie Unit 1 and Unit 2 ASW and do not start any Unit 1 ASW pumps until the cause of the pump overcurrent trip is known.
- Start ASW pump 1-2, it should have automatically started on pump undervoltage when the breaker opened for the 1-1 ASW pump.
- Start ASW pump 1-2, it should have automatically started when PK01-01 alarmed due to low pressure.

Proposed Answer: D. Start ASW pump 1-2, it should have automatically started when PK01-01 alarmed due to low pressure.

#### Explanation:

- Incorrect. Ops Policy B-2 would have the trip investigated if the 1-1 pump was the only pump. The 1-2 pump should have auto started.
- Incorrect. Unit 2 pump is available but cross-tied ASW systems places both units in Tech Spec LCOs, the standby pump should have auto started and per OP1.DC10, is started by the operator.
- Incorrect. The pump auto start is low pressure. Under voltage on the BUS is an autostart.
- Correct. Per OP1.DC10, the operator is expected to take action if auto actions do not occur.

**Technical References:** System lesson LE-5, OP1.DC10, Ops Policy B-2, AR PK01-01, AR PK01-03

**References to be provided to applicants during exam:** None

**Learning Objective:** 5365 - Analyze automatic features and interlocks associated with the ASW System: ASW pumps

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.10	

Rev 1 – modified answers A, B and C. C is now clearly tied to the breaker opening starting the pump and A and B deal with not starting a pump on the affected unit (balance)



Examination Outline Cross-Reference	Level	RO
Knowledge of the effect that a loss or malfunction of the IAS will have on the following: Cross-tied units	Tier #	2
	Group #	1
	K/A #	078 K3.03
	Rating	3.0

### Question 53

GIVEN:

- Unit 1 and Unit 2 are at full power
- Instrument Air pressure for both units is 106 psig
- Rotary Air Compressor 0-7 is selected to LEAD
- Rotary Air Compressor 0-5 Auto Start switch is in ENABLE
- Reciprocating Air Compressors 0-1 through 0-4 are in AUTO
- Rotary Air compressor 0-6 is removed from service

A loss of offsite power occurs on Unit 1 causing a loss of power to Reciprocating Air Compressors 0-1, 0-2 and Rotary Air Compressor 0-7.

Which of the following describes the impact of the loss of the power supply on the Unit 1 Instrument Air system?

- Unit 1 Instrument Air pressure will begin to decrease until an Air Compressor powered from 25E can be aligned as the supply.
- Unit 1 Instrument Air pressure will begin to decrease until it can be cross-tied with Service Air.
- Unit 1 Instrument Air will be maintained by Reciprocating compressors 0-3 and 0-4.
- Unit 1 Instrument Air will be maintained by Rotary Air compressor 0-5.

Proposed Answer: D. Unit 1 Instrument Air will be maintained by Rotary Air compressor 0-5.

#### Explanation:

- Incorrect. Instrument air provides both units, they are not split.
- Incorrect. Air compressors powered from Unit 2 will provide air to Unit 1, systems are cross tied.
- Incorrect. Reciprocating air compressors cycle between 93 and 100 psig. Lower than the rotary air compressor.
- Correct. Rotary air compressor 0-5 will supply instrument air by cycling between 103 and 106 psig.

**Technical References:** System Training Guide K-1

**References to be provided to applicants during exam:** None

**Learning Objective:** 37565 - Analyze automatic features and interlocks associated with the Compressed Air System

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.7	

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	RO
<b>Instrument Air: Knowledge of the purpose and function of major system components and controls.</b>	<b>Tier #</b>	2
	<b>Group #</b>	1
	<b>K/A #</b>	078 G2.1.28
	<b>Rating</b>	4.1

**Question 54**

Which of the following describes the function of the nitrogen supply to the 10% Steam Dump valves and how it is placed in service?

- A. It is the first backup to instrument air and begins to supply the steam dumps when the operator places the switch on VB3 to CUT IN.
- B. It is the final backup, (after backup air bottles), to instrument air and begins to supply the steam dumps when the operator places the switch on VB3 to CUT IN.
- C. It is the first backup to instrument air and begins to supply the steam dumps immediately if air pressure decreases to less than 85 psig.
- D. It is the final backup, (after backup air bottles), to instrument air and begins to supply the steam dumps immediately if air pressure decreases to less than 80 psig.

Proposed Answer: C. It is the first backup to instrument air and begins to supply the steam dumps immediately if air pressure decreases to less than 85 psig.

**Explanation:**

- A. Incorrect. The CUT IN/CUT OUT switch on VB3 is used to place the backup air bottles (the final supply) in service.
- B. Incorrect. Backup bottles are the final supply and placed in service with the CUT IN/CUT OUT switch.
- C. Correct. Nitrogen functions as the backup (preferred) to instrument air for the 10% steam dump valves. The final backup are air bottles that must be placed in service with the cut out switches on VB3. Check valves will open and supply the steam dumps if instrument air pressure supplied to the valves lowers to less than 85 psig.
- D. Incorrect. It is the preferred backup source.

**Technical References:** System training guide LK-1 pages 31 and 32

**References to be provided to applicants during exam:** None

**Learning Objective:** 7199 - Describe Compressed Air System components.

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	X
	Comprehensive/Analysis	
<b>10CFR Part 55 Content:</b>	55.41.7	

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	RO
<b>Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the containment system controls including: Containment pressure, temperature, and humidity</b>	<b>Tier #</b>	2
	<b>Group #</b>	1
	<b>K/A #</b>	103 A1.01
	<b>Rating</b>	3.7

**Question 55**

The operator is monitoring rising Containment temperatures on YR-26 on VB1.

What is being displayed on the recorder?

- A. The highest reading Containment temperature
- B. The lowest reading Containment temperature
- C. The median select Containment temperature
- D. The average Containment temperature

Proposed Answer: D. The average Containment temperature

**Explanation:**

- A. Incorrect. it is not auctioneered high, like other instruments, such as Tave
- B. Incorrect. it is not auctioneered low, like C-16
- C. Incorrect. it is not a digital system like feedwater
- D. Correct. The recorder shows the calculated average containment temperature.

**Technical References:** STG H.1 page 3-1

**References to be provided to applicants during exam:** None

**Learning Objective:** 70345 - Describe the operation of the Area Temperature Monitoring System.

**Question Source:**

(note changes; attach parent)

Bank #

Modified Bank #

New

X

**Question History:**

Last NRC Exam

No

**Question Cognitive Level:**

Memory/Fundamental

X

Comprehensive/Analysis

**10CFR Part 55 Content:**

55.41.7

Rev 1 – corrected explanation (A)

Examination Outline Cross-Reference	Level	RO
<b>Knowledge of the operational implications of the following concepts as they apply to the PZR LCS: Criteria and purpose of PZR level program</b>	<b>Tier #</b>	2
	<b>Group #</b>	2
	<b>K/A #</b>	011 K5.12
	<b>Rating</b>	2.7

### Question 56

Unit 1 is at full power. All Pressurizer backup heaters are in AUTO.

What provides the input signal that is compared to the level program and what is the purpose of the Pressurizer +5% level deviation bistable(s)?

- A. Input is from the controlling Pressurizer level channel; purpose is to energize backup heaters to heat incoming subcooled water to saturation.
- B. Input is from the backup Pressurizer level channel; purpose is to energize backup heaters to heat incoming subcooled water to saturation.
- C. Input is from the controlling Pressurizer level channel; purpose is to flash the incoming water to steam, returning level to program.
- D. Input is from the the backup Pressurizer level channel; purpose is to flash the incoming water to steam, returning level to program.

Proposed Answer: A. Input is from the controlling Pressurizer level channel; purpose is to energize backup heaters to heat incoming subcooled water to saturation.

#### Explanation:

- A. Correct. The signal comes from the controlling channel. It energizes the heaters to heat the incoming subcooled water, which would not flash to steam for pressure control if an outsurge occurred.
- B. Incorrect. signal is from the controlling channel. The output of the controller is used for charging control based on level error.
- C. Incorrect. Flashing the water to steam does not lower the level.
- D. Incorrect. Wrong input and purpose, backup channel inputs to the high and low level bistables.

**Technical References:** System Training Guide LA-4A

**References to be provided to applicants during exam:** None

**Learning Objective:** 40724 - Explain significant Pressurizer, Pressure & Level Control System design features and the importance to nuclear safety.

**Question Source:** Bank #  
(note changes; attach parent) Modified Bank #

**Question History:** New X

**Question Cognitive Level:** Last NRC Exam No

Memory/Fundamental X

Comprehensive/Analysis

**10CFR Part 55 Content:** 55.41.3

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	<b>RO</b>
<b>Nuclear Instrumentation: Knowledge of the operational implications of EOP warnings, cautions, and notes.</b>	<b>Tier #</b>	2
	<b>Group #</b>	2
	<b>K/A #</b>	015 G2.4.20
	<b>Rating</b>	3.8

**Question 57**

GIVEN:

- The crew has entered FR-S.1, Response to Nuclear Power Generation/ATWS
- Reactor Power is 10%

Which of the following describes when, if at all, the RCPs should be stopped while the unit is still at power?

- A. The RCPs should be stopped only if there is no AFW flow, to remove their heat input.
- B. The RCPs should be stopped only if Phase B actuates, to prevent damaging the pumps.
- C. The RCPs should remain running to provide Pressurizer spray and pressure control.
- D. The RCPs should remain running, to continue to provide core cooling.

Proposed Answer: D. The RCPs should remain running, to continue to provide core cooling.

**Explanation:**

- A. Incorrect. They are tripped in H.1 to remove the heat input and delay the onset of steam generator dryout, however, not while in S.1
- B. Incorrect. RCP operation continues per the caution prior to step 1.
- C. Incorrect. While desirable to have spray control, it is not why the RCPs should remain running.
- D. Correct. Caution prior to step 1 states: RCPs should not be tripped with reactor power greater than 5%

During an ATWS, RCP operation could be beneficial by temporarily cooling the core under voided RCS conditions. If reactor power is greater than 5%, the RCPs should not be tripped even if all normal running conditions are not satisfied. Manually tripping the RCPs during some ATWS events could result in reduced heat removal and a challenge to fuel integrity.

**Technical References:** FR-S.1

**References to be provided to applicants during exam:** None

**Learning Objective:** 5216 - Describe significant precautions and limitations associated with the NIS

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	X
	Comprehensive/Analysis	
<b>10CFR Part 55 Content:</b>	55.41.2	

Rev 1 – changed C to balance distractors/answer. 2 stop RCPs, 2 do not stop RCPs

Examination Outline Cross-Reference	Level	RO
Knowledge of ITM system design feature(s) and/or interlock(s) which provide for the following: Range of temperature indication	Tier #	2
	Group #	2
	K/A #	017 K4.03
	Rating	3.1

### Question 58

Which of the following is the maximum design core temperature that can be indicated by an Incore Thermocouple?

- A. 1250°F
- B. 2300°F
- C. 4700°F
- D. 5000°F

Proposed Answer: B. 2300°F

#### Explanation:

- A. Incorrect. this is just above the red path of inadequate core cooling.
- B. Correct. This is the top of the range of the incore thermocouples (just above ECCS acceptance criteria)
- C. Incorrect. this is fuel centerline temperature
- D. Incorrect. point at which fuel melts

**Technical References:** LB-5

**References to be provided to applicants during exam:** None

**Learning Objective:** 8619 - Describe Incore Instrument System components

<b>Question Source:</b>	Bank #59 Sequoyah 1/2008	X
(note changes; attach parent)	Modified Bank #	
	New	
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	X
	Comprehensive/Analysis	
<b>10CFR Part 55 Content:</b>	55.41.2	

Examination Outline Cross-Reference	Level	RO
Ability to predict and/or monitor changes in parameter (to prevent exceeding design limits) associated with operating the HRPS controls including: Hydrogen concentration	Tier #	2
	Group #	2
	K/A #	028 A1.01
	Rating	3.4

### Question 59

GIVEN:

- Large Break LOCA is in progress
- Core damage is occurring
- Containment Hydrogen concentration is 3.2%

Which of the following describes whether or not the Inside Containment H2 Recombination System, (IHRS), should be placed in service and why?

- Do not place the IHRS in service; the system is only placed in service when Hydrogen concentration is greater than 4.0%.
- Do not place the IHRS in service; the system is not used when Hydrogen concentration is greater than 2.0%.
- Place the IHRS in service; the system is effective at any Hydrogen concentration.
- Place the IHRS in service; the system is only used when Hydrogen concentration is below 3.5%.

Proposed Answer: D. Place the IHRS in service; the system is only used when Hydrogen concentration is below 3.5%.

#### Explanation:

- Incorrect. The system is not placed in service if hydrogen concentration is greater than 3.5% (4% is the potential ignition threshold).
- Incorrect. Not used if above 3.5%, therefore, still would be used.
- Incorrect. not used above 3.5% to prevent possible ignition of hydrogen at higher concentrations (containment hydrogen purge is used).
- Correct. Used at concentrations between 0.5 and 3.5%

**Technical References:** System Training Guides H-9, E-1

**References to be provided to applicants during exam:** None

**Learning Objective:** 40834 - Explain significant CHPS design features and the importance to nuclear safety

#### Question Source:

(note changes; attach parent)

Bank

Modified Bank #

New

X

#### Question History:

Last NRC Exam

Yes

#### Question Cognitive Level:

Memory/Fundamental

X

Comprehensive/Analysis

#### 10CFR Part 55 Content:

55.41.7

Rev 1 – replacement to replace repeat question

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	RO
<b>Knowledge of the effect that a loss or malfunction of the Containment Purge System will have on the following: Containment parameters</b>	<b>Tier #</b>	2
	<b>Group #</b>	2
	<b>K/A #</b>	029 K3.01
	<b>Rating</b>	2.7

**Question 60**

GIVEN:

- Unit 1 is at full power
- A Containment purge is in progress to improve Containment air quality prior to personnel entering Containment, per OP H-4:II, Alternate Method for Purging Containment
- Service Air valve AIR-S-1-200 is open supplying additional air to Containment
- Containment Purge Air Supply Fan, S-3 is running
- Containment Purge Air Exhaust Fan, E-3 is running
- Containment pressure is 0.7 psig and lowering slowly

RM-44A, Containment Purge Exhaust Duct, radiation monitor spikes high causing one train of Containment Ventilation Isolation (CVI) to actuate.

Which of the following describes the expected Containment pressure response?

- A. Containment pressure will be stable at its current value due to CVI actuation.
- B. Containment pressure will begin to slowly rise due to AIR-S-1-200 still being open.
- C. Containment pressure rate of decrease will rise due to the exhaust continuing to run.
- D. Containment pressure will continue to slowly lower due to one train of the purge lineup unisolated and the supply and exhaust fans still running.

Proposed Answer: B. Containment pressure will begin to slowly increase due to AIR-S-1-200 still being open.

**Explanation:**

- A. Incorrect. The service air valve will still be open, supplying air to containment.
- B. Correct. Air is still supplied to containment and the purge supply and exhaust lines are isolated. Pressure would begin to slowly rise.
- C. The supply and exhaust (greater flow than the supply) fans are still running but the lines are isolated.
- D. Incorrect. one train of valves will isolate the lines, so no air will be supplied or exhausted despite the fans still running.

**Technical References:** OP H-4:I, OP H-4:II, System Training Guide H-4

**References to be provided to applicants during exam:** None

**Learning Objective:** 4706 - Discuss abnormal conditions associated with the Containment Purge System

**Question Source:**

(note changes; attach parent)

Bank #

Modified Bank #

New

X

**Question History:**

Last NRC Exam

No

**Question Cognitive Level:**

Memory/Fundamental



**10CFR Part 55 Content:**

Comprehensive/Analysis  
55.41.7

X

Examination Outline Cross-Reference	Level	RO
<b>Knowledge of the physical connections and/or cause-effect relationships between the Spent Fuel Pool Cooling System and the following systems: RWST</b>	<b>Tier #</b>	2
	<b>Group #</b>	2
	<b>K/A #</b>	033 K1.05
	<b>Rating</b>	2.7

### Question 61

Unit 1 is at full power.

Spent Fuel Pool level has decreased due to leakage and the crew has entered OP AP-22, Spent Fuel Pool Abnormalities.

Which of the following is the preferred source and can be lined up directly to the Spent Fuel Pool (SFP) to make up for the leakage?

- A. An LHUT using the Liquid Holdup Tank Recirc pump
- B. The CST
- C. Ionics using Demin Water pump 0-2
- D. The RWST

Proposed Answer: D. The RWST

#### Explanation:

- A. Incorrect. RWST is preferred flowpath, LHUT discharges to the transfer side (availability of power necessary to lineup an LHUT is another concern)
- B. Incorrect. CST is the only all class 1E flowpath, however, piping qualification not a concern
- C. Incorrect. preferred source for evaporation.
- D. Correct. Can be used without power and goes directly into the SFP (the door can be closed).

**Technical References:** OP AP-22, appendix A

**References to be provided to applicants during exam:** None

**Learning Objective:** 35694 - Describe the operation of the SFP cooling system and BSFPCS

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.10	

Rev 1 – modified C to provide another “preferred” source (although for evaporation not leakage).

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	RO
<b>Knowledge of the effect of a loss or malfunction on the following will have on the Fuel Handling System : Radiation monitoring systems</b>	<b>Tier #</b>	2
	<b>Group #</b>	2
	<b>K/A #</b>	034 K6.02
	<b>Rating</b>	2.6

**Question 62**

GIVEN:

- Spent fuel assemblies are being moved in preparation for an upcoming refueling
- Exhaust fan E-5 is running.

Fuel Handling Building (FHB) radiation monitor, RM-58, loses power.

Which of the following automatic actions, if any, occurs?

- A. No automatic actions occur.
- B. FHB exhaust fan E-5 remains running and E-6 starts.
- C. FHB exhaust fan E-5 stops and EITHER E-4 or E-6 starts.
- D. FHB exhaust fan E-5 stops and BOTH E-5 and E-6 start.

Proposed Answer: A. No automatic actions occur.

**Explanation:**

- A. Correct. Because E-5 is running, the system is in Iodine Removal mode and will not change when RM-58 alarms.
- B. Incorrect. only the selected fan, E-5 in this case, will start.
- C. Incorrect. E-4 is not started for Iodine removal
- D. Incorrect. E-5 remains running, E-6 will not start (unless E-5 trips)

**Technical References:** System Training Guide H7

**References to be provided to applicants during exam:** None

**Learning Objective:** 8469 - Analyze automatic features and interlocks associated with the RMS

<b>Question Source:</b>	Bank	
(note changes; attach parent)	Modified Bank #34 DCPN NRC 2/2009	X
	New	
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	X
	Comprehensive/Analysis	
<b>10CFR Part 55 Content:</b>	55.41.11	

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	RO
<b>Ability to monitor automatic operation of the ARM system, including: Changes in ventilation alignment</b>	<b>Tier #</b>	2
	<b>Group #</b>	2
	<b>K/A #</b>	072 A3.01
	<b>Rating</b>	2.9

**Question 63**

Area radiation levels at Unit 2 Control Room radiation monitor RE-25 have increased above the setpoint.

Which of the following changes occur to the Control Room ventilation?

- A. Control Room assumes a negative pressure
- B. Control Room assumes a positive pressure
- C. Control Room recirculation dampers close
- D. A Unit 2 Control room pressurization fan starts

Proposed Answer: B. Control Room assumes a positive pressure

**Explanation:**

- A. Incorrect. Assumes a positive pressure.
- B. Correct. A pressurization fan starts on the OPPOSITE unit to maintain a positive pressure.
- C. Incorrect. recirc is maintained.
- D. Incorrect. a unit 1 fan starts.

**Technical References:** System Lesson LH-5

**References to be provided to applicants during exam:** None

**Learning Objective:** 37289 - Analyze automatic features and interlocks associated with the Control Room Ventilation System

<b>Question Source:</b>	Bank # S-0013	X
(note changes; attach parent)	Modified Bank #	
	New	
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.7	

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	<b>RO</b>
<b>Knowledge of bus power supplies to the following: Emergency/essential SWS pumps</b>	<b>Tier #</b>	2
	<b>Group #</b>	2
	<b>K/A #</b>	075 K2.03
	<b>Rating</b>	2.6

**Question 64**

What are the power supplies for the Unit 2 Aux Saltwater Pumps?

- |    |                 |                 |
|----|-----------------|-----------------|
|    | <u>Pump 2-1</u> | <u>Pump 2-2</u> |
| A. | Bus F           | Bus G           |
| B. | Bus G           | Bus H           |
| C. | Bus H           | Bus F           |
| D. | Bus F           | Bus H           |

Proposed Answer: A. Bus F Bus G

**Explanation:**

- A. Correct. No unit difference, bus F powers pump 2-1 and bus G powers pump 2-2
- B. Incorrect. this alignment does power loads such as the RHR pumps
- C. Incorrect. this alignment does power loads such as the AFW pumps
- D. Incorrect. this alignment does power loads such as the SI pumps

**Technical References:** OIM J-1-1

**References to be provided to applicants during exam:** None

**Learning Objective:** 5339 - State the power supplies to ASW system components

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	X
	Comprehensive/Analysis	
<b>10CFR Part 55 Content:</b>	55.41.7	

Examination Outline Cross-Reference	Level	RO
<b>Ability to manually operate and/or monitor in the control room:  Halon system  NOTE: substituted CO2 for Halon. The halon system has been removed.</b>	<b>Tier #</b>	2
	<b>Group #</b>	2
	<b>K/A #</b>	086 A4.06
	<b>Rating</b>	3.2

**Question 65**

Which of the following describes what control, if any, the operator has in the Control Room for CARDOX to Turbine Generator Bearing #10?

- A. Actuation only
- B. Reset only
- C. Actuation and abort
- D. Cannot be actuated or reset from the Control Room

Proposed Answer: A. Actuation only.

**Explanation:**

- A. Correct. Reset is performed at the local panels but the system can be actuated from the control room.
- B. Incorrect. the system is actuated from the control room, no reset is performed.
- C. Incorrect. Abort is done locally.
- D. Incorrect. actuation may be performed in the control room

**Technical References:** OP K-2B:I, K-2B:II, System Training Guide K-2B

**References to be provided to applicants during exam:** None

**Learning Objective:** 3707 - Describe the operation of the Cardox System

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	X
	Comprehensive/Analysis	
<b>10CFR Part 55 Content:</b>	55.41.7	

Rev 1 – abort is a plausible distractor, because the operator may want to be able to prevent actuation if personnel are present and there would be a safety concern.

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	<b>RO</b>
<b>Knowledge of criteria or conditions that require plant-wide announcements, such as pump starts, reactor trips, mode changes, etc.</b>	<b>Tier #</b>	3
	<b>Group #</b>	1
	<b>K/A #</b>	G2.1.14
	<b>Rating</b>	3.1

### Question 66

GIVEN:

- PK11-25, Plant Vent Radiation, is in alarm.
- PK11-21, High Radiation, is in alarm.
- GDT 1-1 pressure is 30 psig
- GDT 1-2 pressure is 45 psig
- GDT 1-3 pressure is 0 psig

Upon entering OP AP-14, Tank Ruptures, the first action the Control Room crew will take is to:

- A. call Access Control to confirm radiation levels.
- B. dispatch the Aux Board watch to secure any running waste gas compressor.
- C. make a plant wide announcement on the PA system that a GDT rupture has occurred and to evacuate the area.
- D. make an announcement on the PA system for personnel in the Auxiliary Building to contact the Control Room.

Proposed Answer: C. make a plant wide announcement on the PA system that a GDT rupture has occurred question and to evacuate the area.

#### Explanation:

A incorrect. the Waste Gas compressors discharge to the GDTs from the vent header.

Only C correct. Step 1 of OP AP-14 states:

Announce the source AND location of the problem on the Plant Public Address System (PPAS)

**Technical References:** OP AP-14 step 1

**References to be provided to applicants during exam:** None

**Learning Objective:** 3477P - Given an abnormal condition, summarize the major actions of OP AP-14 to mitigate an event in progress.

**Question Source:** Bank # P-36845 X  
 (note changes; attach parent) Modified Bank #  
 New

**Question History:** Last NRC Exam No

**Question Cognitive Level:** Memory/Fundamental X  
 Comprehensive/Analysis

**10CFR Part 55 Content:** 55.41.10

Rev 1 – modified A to be clearly incorrect.

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	RO
<b>Knowledge of the station's requirements for verbal communications when implementing procedures.</b>	<b>Tier #</b>	3
	<b>Group #</b>	1
	<b>K/A #</b>	G2.1.38
	<b>Rating</b>	3.7

**Question 67**

A LOCA has occurred on Unit 2.

The crew is reviewing the foldout page as part of a procedure transition brief.

In accordance with OP1.DC10, Conduct of Operations, as a minimum, what amount of communication is required by an operator who is assigned a foldout page item to monitor?

- A. Repeat back of the high level action
- B. Simple acknowledgement of the assignment
- C. A brief summary of the action and the parameters to monitor
- D. Repeat back of the high level action and the specific parameters and values to monitor

Proposed Answer: A. Repeat back of the high level action

**Explanation:**

A Correct. Per OP1.DC10 page 45 (rev 27)... foldout page should be reviewed with the crew as a part of the procedure transition tailboard. Specific assignments should be made to appropriate control room operators by assigning the foldout page number and the operator repeating back the high level action. Specific parameters and values are not required to be repeated back. A copy of the foldout page should be given to any operator with an assignment.

B incorrect. Repeat high level action.

C incorrect. Repeat back the high level action.

D incorrect. Specific parameters etc not required.

**Technical References:** OP1.DC10

**References to be provided to applicants during exam:** None

**Learning Objective:** 7922 - Discuss the characteristics of a tailboard

**Question Source:** Bank #74 DCPD 2/2005 X  
 (note changes; attach parent) Modified Bank #  
 New

**Question History:** Last NRC Exam No

**Question Cognitive Level:** Memory/Fundamental X  
 Comprehensive/Analysis

**10CFR Part 55 Content:** 55.41.10

Rev 1 - Changed "repeat back" in stem to "communications" and modified answers A and B



<b>Examination Outline Cross-Reference</b>	<b>Level</b>	<b>RO</b>
<b>Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc.</b>	<b>Tier #</b>	3
	<b>Group #</b>	1
	<b>K/A #</b>	G2.1.43
	<b>Rating</b>	4.1

**Question 68**

GIVEN:

- Unit 1 is at full power
- Boron concentration is 150 ppm

Rod H-8 is discovered to be untrippable.

Which of the following describes the effect, if any, on Shutdown Margin (SDM)?

- A. No effect because one rod is already assumed to be stuck when calculating SDM.
- B. SDM is less by 665 pcm.
- C. SDM is less by 1236 pcm.
- D. SDM is less by 1330 pcm.

Proposed Answer: C. SDM is less by 1236 pcm.

**Explanation:**

- A. Incorrect. While one rod is assumed to be stuck, SDM is reduced because it is assumed another (unknown) rod will not trip. Therefore, SDM is conservatively reduced by the value of the most reactive rod.
- B. Incorrect. this is the BOL value, at 150 ppm, the EOL value would be used.
- C. Correct. This is the EOL value. R-19 reduces SDM by the number of known stuck rods times the value for the “most reactive” rod (1236 pcm at EOL).
- D. Incorrect. this is the value of the D-12, BOL pcm, taken twice, once for the rod and once for the “assumed rod” that will not trip, this assumed amount is already built into the SDM calculation.

**Technical References:** Table R19-1T-1, R-19 attachment 9.1

**References to be provided to applicants during exam:** Table R19-1T-1, R-19 attachment 9.1

**Learning Objective:** 10382 - Given the appropriate data from Vol 9B, perform a shutdown margin (SDM) calculation using STP R-19 and its various attachments.

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.41.5	

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	<b>RO</b>
<b>(multi-unit license) Knowledge of the design, procedural, and operational differences between units.</b>	<b>Tier #</b>	3
	<b>Group #</b>	2
	<b>K/A #</b>	G2.2.3
	<b>Rating</b>	3.8

**Question 69**

Why is the allowable cooldown rate without CRDM fans running, per E-0.2, Natural Circulation Cooldown, different for Unit 1, (25°F/hour) and Unit 2 (50°F/hour)?

- A. Upper head temperature is close to  $T_{hot}$  for Unit 1 and close to  $T_{cold}$  for Unit 2
- B. Unit 2's natural circulation flow is significantly higher than Unit 1's
- C. Unit 1's thermocouples are in a different orientation
- D. Unit 1  $RT_{NDT}$  is higher than Unit 2's

Proposed Answer: A. Upper head temperature is close to  $T_{hot}$  for Unit 1 and close to  $T_{cold}$  for Unit 2

**Explanation:**

- A. Correct. Unit 1 is a “Thot” plant and Unit 2 is a “Tcold” plant. This corresponds to a difference in the assumed upper head temperature and it is higher for Unit 1. As a result, the allowable cooldown rate is half of the allowable Unit 2 cooldown rate.
- B. Incorrect. layouts are the same, the flowrates are the same.
- C. Incorrect. the location of the thermocouples will not affect the allowable cooldown rate.
- D. Incorrect. Unit 1 is older than unit 2 but the  $RT_{ndt}$  is approximately the same and does not restrict the cooldown rate.

**Technical References:** LPE0.2

**References to be provided to applicants during exam:** None

**Learning Objective:** 7920C - Explain basis of emergency procedure steps (E-0.2 series)

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	X
	Comprehensive/Analysis	
<b>10CFR Part 55 Content:</b>	55.41.2	
Rev 1 – modified A and B		

Examination Outline Cross-Reference	Level	RO
Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.	Tier #	3
	Group #	2
	K/A #	G2.2.36
	Rating	3.1

### Question 70

Unit 1 is at full power.

One of the Diesel Generators (D/G) is to be removed from service for a planned maintenance outage window.

What Technical Specification(s) will be entered when the D/G is removed from service and inoperable?

- A. Only the Technical Specification for the D/G.
- B. Only the Technical Specifications for the D/G and the associated 4 kV bus.
- C. The Technical Specifications for the D/G, the associated 4 kV bus and 120 VAC bus.
- D. The Technical Specifications for the D/G, the associated 4 kV bus and all loads fed from the bus.

Proposed Answer: A. Only the Technical Specification for the D/G.

#### Explanation:

A. Correct. LCO 3.0.6 states: When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system.

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

With no equipment out of service, removing one diesel from service will not require entry into the LCO for the diesel (3.8.1) but no others (do not cascade tech specs). LCO 3.8.1 does not direct entry into any other LCOs.

- B. Incorrect. do not cascade LCOs
- C. Incorrect. Do not cascade LCOs
- D. Incorrect. Do not cascade LCOs

**Technical References:** LCO 3.0.6

**References to be provided to applicants during exam:** None

**Learning Objective:** 9697L - Apply TS 3.0 Technical Specification LCOs

**Question Source:** Bank #  
(note changes; attach parent) Modified Bank #

**Question History:** New X  
Last NRC Exam No

**Question Cognitive Level:** Memory/Fundamental

**10CFR Part 55 Content:**

Comprehensive/Analysis  
55.41.10

X

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	<b>RO</b>
<b>Ability to comply with radiation work permit requirements during normal or abnormal conditions.</b>	<b>Tier #</b>	3
	<b>Group #</b>	3
	<b>K/A #</b>	G2.3.7
	<b>Rating</b>	3.8

### Question 71

In accordance with RCP D-201, Writing Radiation Work Permits, a Routine Radiation Work Permit (RWP) is valid for a maximum of:

- A. One year
- B. The operating cycle for the unit the RWP covers
- C. One calendar quarter
- D. 30 days

Proposed Answer: A. One year

#### Explanation:

- A. Correct. D-201 states a Routine RWP is valid for one year.
- B. Incorrect. Valid for one year.
- C. Incorrect. Valid for one year.
- D. Incorrect. Its plausible that given it requires stable radiation levels, that this occurs during a fuel cycle.

**Technical References:** RCP D-201

**References to be provided to applicants during exam:** None

**Learning Objective:**

**Question Source:**

(note changes; attach parent)

Bank #

Modified Bank

New

X

**Question History:**

Last NRC Exam

Yes

**Question Cognitive Level:**

Memory/Fundamental

X

Comprehensive/Analysis

**10CFR Part 55 Content:**

55.41.12

Rev 1 – replacement question

Examination Outline Cross-Reference	Level	RO
Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.	Tier #	3
	Group #	3
	K/A #	G2.3.13
	Rating	3.4

### Question 72

Unit 1 is at full power.

Power is lost to CCW radiation monitor, RM-17A and PK11-22, Rad Mon Sys Failure/CVI Bypass, alarms.

What action, if any, must be taken by the crew to maintain adequate sample flow through RM-17B?

- A. Verify CCW heat exchanger 1-1 is in service.
- B. Verify CCW heat exchanger 1-2 is in service.
- C. Place a second CCW heat exchanger in service.
- D. No action required, there is adequate flow through RM-17B with either heat exchanger in service.

Proposed Answer: B. Verify CCW heat exchanger 1-2 is in service.

#### Explanation:

- A. Incorrect. RM-17A is at the outlet of heat exchanger of 1-1, the crew needs to ensure 1-2 is in service.
- B. Incorrect. RM-17A is at the outlet of heat exchanger of 1-1, the crew needs to ensure 1-2 is in service. With only RM-17B available, if heat exchanger 1-1 was in service, only a small amount of flow would pass thru the radiation monitor and the possibility of an unsafe radiological condition, such as RCS leakage into the system, would occur and the isolation the system (by closing RCV-16) would be delayed.
- C. Correct. Completed by the RP Foreman
- D. Incorrect. each individual responsible for their exposure but the RP Foreman signs the checklist.

**Technical References:** AR PK11-22, System Training Guide G4A and OVID 106714, sheet 2

**References to be provided to applicants during exam:** None

#### Learning Objective:

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	X
	Comprehensive/Analysis	
<b>10CFR Part 55 Content:</b>	55.41.12	

Rewrote question to align with Ops responsibilities for RP protection.

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	<b>RO</b>
<b>Knowledge of EOP entry conditions and immediate action steps.</b>	<b>Tier #</b>	3
	<b>Group #</b>	4
	<b>K/A #</b>	G2.4.1
	<b>Rating</b>	4.6

**Question 73**

Which of the following Emergency Operating Procedures or Function Restoration Procedures, if any, may be entered directly rather than first entering E-0, Reactor Trip or Safety Injection?

- A. None, all entries into the emergency procedure network begin with entry into E-0.
- B. ECA-0.0, Loss of All Vital AC Power only
- C. FR-S.1, Response to Nuclear Power Generation/ATWS only
- D. ECA-0.0, Loss of All Vital AC Power and FR-S.1, Response to Nuclear Power Generation/ATWS

Proposed Answer: B. ECA-0.0, Loss of All Vital AC Power only

**Explanation:**

- A. Incorrect. ECA-0.0 may be directly entered.
- B. Correct. It is permissible to directly enter ECA-0.0
- C. Incorrect. common misconception that FR-S.1 (red) is directly entered rather than attempting a reactor trip first.
- D. Incorrect. FR-S.1 is not entered directly.

**Technical References:** OP1.DC10, ECA-0.0, FR-S.1, LPEORG page 13 of 22

**References to be provided to applicants during exam:** None

**Learning Objective:** 6827 - State the entry conditions for Emergency Operating Procedures (EOPs).

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	X
	Comprehensive/Analysis	
<b>10CFR Part 55 Content:</b>	55.41.10	

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	<b>RO</b>
<b>Knowledge of general guidelines for EOP usage.</b>	<b>Tier #</b>	3
	<b>Group #</b>	4
	<b>K/A #</b>	G2.4.14
	<b>Rating</b>	3.8

### Question 74

In the Emergency Response Guidelines (ERGs) network, which of the following ERGs would take priority over all the other procedures?

- A. E-0, Reactor Trip or Safety Injection
- B. ECA-0.0, Loss of All Vital AC Power
- C. F-0, Critical Safety Function Status Trees
- D. FR-S.1, Response to Nuclear Power Generation/ATWS

Proposed Answer: B. ECA-0.0, Loss of All Vital AC Power

#### Explanation:

- A. Incorrect. E-0 is the entry point for most conditions, however, all the emergency response guidelines are based on having at least one train of vital power, as such, ECA-0.0 is the procedure with the highest importance.
- B. Correct. Entry into the Emergency Response Guideline set is limited to two specific conditions:
  - o If at any time a reactor trip or safety injection occurs or is required, the operator will enter guideline E 0, REACTOR TRIP OR SAFETY INJECTION.
  - o If at any time a complete loss of power on the ac emergency busses takes place, the operator will enter guideline ECA 0.0, LOSS OF ALL AC POWER. This includes any time during the performance of any other ERG. Direct entry into ECA 0.0, due to loss of ac power on all ac emergency busses, is expected to be a rare occurrence. However, once in ECA 0.0, special considerations come into effect. Because none of the electrically powered safeguards equipment used to restore Critical Safety Functions is operable, none of the FRGs can be implemented. A NOTE at the beginning of guideline ECA 0.0 states that "CSF Status Trees should be monitored for information only. FRGs should not be implemented." Once in ECA 0.0, the operator performs the required actions, and is not allowed to transition to any other guideline until some form of power is restored to the ac emergency busses. Even then, permission is not granted to implement FRGs until some initial status checks and actions are performed by the operator.
- C. Incorrect. while the CSF status trees are monitored for degrading plant conditions, all the FRGs require power to be effective.
- D. Incorrect. While FR-S.1 is the highest CSF, ECA-0.0 would be performed and the CSFs are monitored for information only.

**Technical References:** Westinghouse Executive Volume, EOP Users Guide

**References to be provided to applicants during exam:** None

**Learning Objective:** 6764 - Describe what procedure or procedure set would be used in an emergency event, based on plant mode/conditions.

**Question Source:** Bank # P-37064

X



<b>Examination Outline Cross-Reference</b>	<b>Level</b>	RO
<b>Knowledge of facility protection requirements, including fire brigade and portable fire fighting equipment usage.</b>	<b>Tier #</b>	3
	<b>Group #</b>	4
	<b>K/A #</b>	G2.4.26
	<b>Rating</b>	3.1

**Question 75**

The Fire Phone rings and the caller tells you it is a fire emergency.

Prior to activating the fire alarm, what should the caller be told?

- A. Leave the area.
- B. Stay on the phone.
- C. Assist in attempting to put out the fire until the fire brigade arrives.
- D. Remain in the area to provide information to the fire brigade when it arrives.

Proposed Answer: B. Stay on the phone.

**Explanation:**

- A. Incorrect. The caller remains on the bridge line with the fire captain
- B. Correct. The caller remains to speak on the bridge line
- C. Incorrect. Not a prudent action
- D. Incorrect. The caller should not be in the area of the fire when the report is made

**Technical References:** CP M-6

**References to be provided to applicants during exam:** None

**Learning Objective:** Demonstrate the ability to activate fire alarm from control room

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	X
	Comprehensive/Analysis	
<b>10CFR Part 55 Content:</b>	55.41.10	

## Question 76

Examination Outline Cross-Reference	Level	SRO
<b>Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: High-temperature computer alarm and alarm type</b>	<b>Tier #</b>	1
	<b>Group #</b>	1
	<b>K/A #</b>	APE 008 AA2.04
	<b>Rating</b>	3.4

### SRO Question 1

GIVEN:

- Unit 1 is at 100% power
- RCS pressure is 1990 psig and lowering slowly
- Pressurizer heaters are energized
- Pressurizer Safety Sonic flow is indicated on VB2
- All tailpipe temperatures are elevated
- PK05-20, PZR RELIEF/SAFETY VLVS OPEN, is in alarm
- PK05-23, PZR SAFETY OR RELIEF LINE TEMP, is in alarm

Which of the following actions will be taken by the Shift Foreman?

- A. Go to OP AP-13, Malfunction of Reactor Pressure Control System and direct the isolation of the leaking PORV.
- B. Go to OP AP-13, Malfunction of Reactor Pressure Control System and direct the action required (safety injection) for a lifting Pressurizer code safety valve.
- C. Implement OP O-24, Evaluation of Pressurizer Safety Valve High Tailpipe Temperature or Leakage, due to seat leakage from a Pressurizer code safety valve.
- D. Direct the operator to trip the reactor and go to E-0, Reactor Trip or Safety Injection due to the reactor failing to automatically trip.

Proposed Answer: B. Go to OP AP-13, Malfunction of Reactor Pressure Control System and direct the action required (safety injection) for a lifting Pressurizer code safety valve.

### Explanation:

- A. Incorrect. Tailpipe temperatures would be elevated and RCS pressure would decrease for a leaking PORV. This is the proper action for an open PORV.
- B. Correct. Sonic flow is an indication of a leaking code safety. Because pressure is lowering, this is more than seat leakage and AP-13 will direct a reactor trip.
- C. Incorrect. Sonic flow is an indication of leakage past a code safety, however, at the current RCS pressure, all heaters are on and without operator action, a reactor trip will occur. This is the proper action (per the PK procedure) if the safety has seat leakage
- D. Incorrect. Reactor trip setpoint is 1950 psig. No automatic trip have been reached at this time. (There is a rate component to the trip circuit. The pressurizer low pressure reactor trip is effectively set at seven seconds. What that means is at the existing rate of change of pressure, psi/sec, the circuit anticipates what the pressure will be in 10 seconds and sends that to the comparator circuit. There is a three second filter or lag in the signal to prevent noise from tripping the bistable. The result is that effectively seven seconds of anticipation

is provided in the reactor trip protection circuit. A rate of 1 psi/sec, the trip would occur at 1957 psig)

**Technical References:** AR PK05-20, AR PK05-23, AP-13

**References to be provided to applicants during exam:** None

**Learning Objective:** 36926 - Discuss abnormal conditions associated with the Pzr, Pzr Pressure and Level Control System

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	<b>55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.</b>	

Examination Outline Cross-Reference	Level	SRO
Loss of Component Cooling Water: Knowledge of conditions and limitations in the facility license.	Tier #	1
	Group #	1
	K/A #	APE 026 G2.2.38
	Rating	4.5

**SRO Question 2**

GIVEN:

- Unit 1 is performing a heatup
- RCS temperature is 300°F
- RCP 1-2 is in service
- All steam generator narrow range levels are approximately 35%
- All steam generator pressures are approximately 400 psig
- RHR pump 1-2 is in service
- CCW pumps 1-1 and 1-3 are running

Running CCW pump 1-1 trips and is declared inoperable. The operator verifies that CCW pump 1-2 is in service.

Which of the following describes the crew's ability to continue the heatup and proceed to MODE 3?

- No action is required, all LCOs continue to be met.
- The MODE change cannot be made until the CCW loop is restored to OPERABLE status.
- The MODE change cannot be made, a shutdown to MODE 5 within 24 hours is required.
- The MODE may be made, but the CCW pump must be restored to OPERABLE status within 72 hours.

Proposed Answer: B. The MODE change cannot be made until the CCW loop is restored to OPERABLE status

**Explanation:**

- Incorrect. Per the bases for Technical Specification 3.7.7, "To meet the LCO on CCW loops, vital headers A and B, both CCW heat exchangers, the surge tank, and all three CCW pumps must be operable."
- Correct. RCS loops are met (requires 2 operable and one in operation) with 1 RCP in operation and steam generator levels greater than 15%. Only the CCW loop is inoperable, Condition A has a 72 hour completion time.  
Note: the RHR loop is still
- Incorrect. While one RHR may be inoperable, there are an adequate number of RCS loops in operation and LCO 3.4.6 is met (otherwise the action would be to be in MODE 5 in 24 hours).
- Incorrect. Mode change with inoperable not allowed unless the action allows for continued operation (LCO 3.0.4)

**Technical References:** Tech Spec bases 3.7.7, LCO 3.7.7, 3.4.6 and 3.0.4

**References to be provided to applicants during exam:** LCO 3.7.7, 3.4.6

<b>Learning Objective:</b>	9694G - Apply TS 3.7 Technical Specification bases	
<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.43.2 – Facility operating limitations in the technical specifications and their bases.	

Rev 1 – replaced question

## Question 78

Examination Outline Cross-Reference	Level	SRO
Ability to determine and interpret the following as they apply to a reactor trip: Reactor trip first out indication	Tier #	1
	Group #	1
	K/A #	EPE 007 EA2.05
	Rating	3.9

### SRO Question 3

GIVEN:

- Unit 1 is at 70% power
- Control Bank D is at 189 steps
- Pressurizer pressure channel III, PT-457, has been removed from service and bistables tripped

The following events occur:

- Channel II bistable lights are ON and channel III bistable lights are OUT
- Rods begin to step in and PK03-13, Rod Lo Insertion Limit alarms on VB2
- PK19-19, Vital UPS Failure, alarms on VB5

What action will the Shift Foreman take?

- A. Direct the operator to trip the reactor and go to E-0, Reactor Trip or Safety Injection because the reactor should have tripped.
- B. Direct the operator to place rods in MANUAL, and go to OP AP-6, Emergency Boration to address the loss of shutdown margin.
- C. Direct the operator to place rods in MANUAL, and go to OP AP-4, Loss of Vital or Nonvital Instrument AC for a loss of PY-12.
- D. Direct the operator to place rods in MANUAL, and go to OP AP-4, Loss of Vital or Nonvital Instrument AC for a loss of PY-13.

Proposed Answer: A. Direct the operator to trip the reactor and go to E-0, Reactor Trip or Safety Injection because the reactor should have tripped.

### Explanation:

- A. Correct. Although the bistable lights for the failed pressure channel are out (loss of PY-12) the bistables are still tripped and the coincidence for a reactor trip is met with 2/4 pressure channels (low, high, SI) met. The reactor should have automatically tripped.
- B. Incorrect. The reactor should have tripped, additionally, emergency boration would not be required, normal boration is initiated if below the low and the low-low alarms.
- C. Incorrect. This would be correct if only PY-12 was lost.
- D. Incorrect. Although the channel III lights are out, this is because PY-12 provides power to the indication, but provides power to bistables for channel II.

**Technical References:** Unit 1 COLR Figure 1, AR PK19-19, OP AP-4, AR PK03-13

**References to be provided to applicants during exam:** None

**Learning Objective:** 4274 - Explain the consequences of loss of vital instrument bus

**Question Source:** Bank #  
(note changes; attach Modified Bank #  
parent)

<b>Question History:</b>	New	X
<b>Question Cognitive Level:</b>	Last NRC Exam	
	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.	

## Question 79

Examination Outline Cross-Reference	Level	SRO
Ability to determine and interpret the following as they apply to the Steam Line Rupture: Difference between a steam line rupture and a LOCA	Tier #	1
	Group #	1
	K/A #	APE 040 AA2.03
	Rating	4.7

### SRO Question 4

GIVEN:

- RCS pressure is 1600 psig and lowering slowly
- RCS temperature is 520°F and lowering slowly
- Radiation levels on containment radiation monitors are elevated, but not in alarm
- Counts on steamline radiation monitor RM-73 are elevated but below alarm setpoints
- Containment sump level is rising
- Pressure in steam generators
  - 1-1 and 1-2 - 800 psig and lowering slowly
  - 1-3 and 1-4 - 300 psig and lowering rapidly
- Steam generator levels:
  - 1-1 and 1-2 approximately 30% on the Narrow Range and stable
  - 1-3 is approximately 9% Wide Range and lowering
  - 1-4 is approximately 5% Wide Range and lowering
- Containment pressure is 24 psig and rising slowly
- Only AFW pump 1-2 is running, total AFW flow is 400 gpm

Which of the following describes the procedure transition the Shift Foreman will make from E-0, Reactor Trip or Safety Injection?

NOTE:

- FR-H.1, Response to Loss of Secondary Heat Sink
  - E-1, Loss of Reactor or Secondary Coolant
  - E-2, Faulted Steam Generator Isolation
- A. To FR-H.1 due to a RED Secondary Heat Sink Status tree, rather than to E-1 to address the loss of reactor coolant.
- B. To FR-H.1 due to a RED Secondary Heat Sink Status tree, rather than to E-2 to address the faulted steam generators.
- C. Directly to E-1 to address the loss of reactor coolant; there is no extreme challenge to secondary heat sink at this time.
- D. Directly to E-2 to address the faulted steam generators; there is no extreme challenge to secondary heat sink at this time.

Proposed Answer: D. Directly to E-2 to address the faulted steam generators; there is no severe challenge to secondary heat sink at this time.

**Explanation:**



- A. Incorrect. There is no LOCA, only faulted steam generators. Radiation levels are a result of the tube leak. Heat sink is not red due to at least one steam generator greater than 26%
- B. Incorrect, Heat sink is not red due to at least one steam generator greater than 15%.
- C. C incorrect. No LOCA is in progress (additionally, because there are faulted steam generators, E-0 will address those first).
- D. Correct. No LOCA in progress and heat sink status tree is yellow.

**Technical References:** F-0, E-0 steps 9 - 11

**References to be provided to applicants during exam:** None

**Learning Objective:** 3552 - Given initial conditions, assumptions, and symptoms, determine the correct Emergency Operating Procedure to be used to mitigate an operational event

**Question Source:** Bank #  
 (note changes; attach Modified Bank #  
 parent)

New X

**Question History:** Last NRC Exam

**Question Cognitive Level:** Memory/Fundamental  
 Comprehensive/Analysis X

**10CFR Part 55 Content:** 55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Rev 1 – raised by 2% (30%) the narrow range levels to clearly delineate that there is adequate heat sink. The previous value was very close (28 vs 26) to the less common adverse containment number.

## Question 80

Examination Outline Cross-Reference	Level	SRO
<b>Station Blackout: Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.</b>	<b>Tier #</b>	1
	<b>Group #</b>	1
	<b>K/A #</b>	EPE 055 G2.4.30
	<b>Rating</b>	4.1

### SRO Question 5

Which of the following events would require notification of the NRC no later than one hour after the occurrence?

- A. An automatic reactor trip from full power.
- B. Spurious Safety Injection actuation at full power.
- C. Loss of power to all vital AC buses for 20 minutes.
- D. A plant shutdown due to entry in Technical Specification 3.0.3.

Proposed Answer: C. Loss of power to all vital AC buses for 20 minutes.

### Explanation:

- A. Incorrect. 4 hour report
- B. Incorrect. 4 hour report.
- C. Correct. Results in E-Plan classification and is a one hour report.
- D. Incorrect. 4 hour report.

**Technical References:** XI1.ID2 attachment 5, EP G-1 attachment 2

**References to be provided to applicants during exam:** None

**Learning Objective:** 42159 - Determine reporting requirements of 24 hours or less.

**Question Source:** Bank #  
(note changes; attach Modified Bank #  
parent)

**Question History:** New X  
Last NRC Exam

**Question Cognitive Level:** Memory/Fundamental  
Comprehensive/Analysis X

**10CFR Part 55 Content:** 55.43.5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

## Question 81

Examination Outline Cross-Reference	Level	SRO
<b>LOCA Outside Containment: Knowledge of the emergency action level thresholds and classifications.</b>	<b>Tier #</b>	1
	<b>Group #</b>	1
	<b>K/A #</b>	E04 G2.4.41
	<b>Rating</b>	4.6

### SRO Question 6

GIVEN:

- Containment pressure rose 0.1 psig and is stable
- Containment radiation is 4.5 R/hr on RM-30 and RM-31 and rising slowly
- Containment sump level is offscale low
- RWST level is 40% and lowering slowly
- RCS pressure is 300 psig and lowering slowly
- RCS Subcooling is 0°F
- RVLIS full range level is 40% and lowering slowly
- Core Exit Thermocouples are 660°F and rising slowly

What is the current Emergency Action Level?

- A. Alert due to the loss of the RCS barrier.
  - B. Site Area Emergency due to loss of the RCS and Containment barriers.
  - C. Site Area Emergency due to loss of the RCS barrier and potential loss of the Fuel Clad barrier.
  - D. General Emergency due to the loss of the RCS and Containment Barriers and the potential loss of the Fuel Clad barrier.
- Proposed Answer: B. Site Area Emergency due to loss of the RCS and Containment barriers.

**Explanation:**

Containment Barrier – Lost (2), Following LOCA, Containment pressure or sump level response not consistent with LOCA conditions

RCS Barrier – Lost, RCS leak rate greater than makeup capacity, as indicated by a loss of subcooling (also potential loss due to leakrate greater than 150 gpm)

Fuel Clad Barrier – not challenged

- A. Incorrect. Two barriers have been lost, as indicated by the abnormal containment pressure response.
- B. Correct. Loss of RCS and containment barriers as indicated by loss of subcooling and containment pressure response not consistent with LOCA conditions.
- C. Incorrect. Fuel barrier not challenged. Core exits are less than 700°F and RVLIS level is above the required 32% (but less than the 46% for dynamic range, if the table is incorrectly read).
- D. Incorrect. Fuel barrier currently intact.

**Technical References:** EP G-1, attachment 2

**References to be provided to applicants during exam:** EP G-1, attachment 2, pages 14 and 15

**Learning Objective:** 42285 - Given indications of an event, use EP G-1 to classify the event with 100% accuracy within 15 minutes

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.43.4	

Rev.1 – modified containment conditions to clearly delineate the Containment barrier is breached. Discussion in the industry as to whether a break outside containment, that is isolated on a system (as ECA-1.2 may do), such as RHR, is a LOCA makes the original question open to debate. By keeping the LOCA in containment but having abnormal pressure response, clearly there has been a breach of 2 barriers and B is clearly the correct answer.

## Question 82

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	<b>SRO</b>
<b>Continuous Rod Withdrawal: knowledge of low power/shutdown implications in accident mitigation strategies.</b>	<b>Tier #</b>	1
	<b>Group #</b>	2
	<b>K/A #</b>	APE 001 G2.4.9
	<b>Rating</b>	4.2

**SRO Question 7**

GIVEN:

- Reactor power is 9%
- The turbine is rolling at 1800 rpm
- The crew is raising power from 8 to 15% power in accordance with OP L-3, Secondary Plant Startup

The following events occur:

- Turbine First Stage Pressure channel, PT-505 fails high
- All Control Bank D Rods begin to step OUT
- One of the rods indicates it is 6 steps out of alignment as the bank withdraws

Which of the following actions will be taken by the Shift Foreman?

- A. Direct the operator to place the Rod Bank/Mode Select Switch in CB D and go to OP AP-12A, Continuous Withdrawal or Insertion of a Control Rod Bank, to address the unwarranted rod motion.
- B. Direct the operator to place rods in MANUAL and go to OP AP-12B, Control Rod Misalignment, to address the Control Bank D misalignment .
- C. Direct the operator to place rods in MANUAL and go to OP AP-5, Malfunction of Eagle 21 Protection or Control Channel to address the failed PT-505 channel.
- D. Direct the operator to trip the reactor and go to E-0, Reactor Trip or Safety Injection because Rods are in MANUAL and moving.

Proposed Answer: D. Direct the operator to trip the reactor and go to E-0, Reactor Trip or Safety Injection because Rods are in MANUAL and moving.

**Explanation:**

- A. Incorrect. Rods should not be moving, they are in MANUAL below 15% and auto withdrawal is blocked below 15%. If at power and rods are in AUTO and begin to move unexpectedly, the action is to take rods to manual and go to OP AP-12A. The individual bank select position is not selected.
- B. Incorrect. Rods should not be moving and misalignment entry condition is 12 steps.
- C. Incorrect. this is the normal response to an instrument failure with expected plant response.
- D. Correct. The rods not be moving (in MANUAL). Reactor should be tripped and E-0 performed.

**Technical References:** OP AP-12A, AP-12B, OP AP-5, OP L-3

**References to be provided to applicants during exam:** None

**Learning Objective:** 7927 - Given initial conditions and assumptions, determine if a

reactor trip or safety injection actuation is required

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.43.6	

Rev 1 – bullet info in setup for clarity. Modified A to correctly identify the name of the switch.

## Question 83

Examination Outline Cross-Reference	Level	SRO
<b>Ability to determine and interpret the following as they apply to the (Degraded Core Cooling): Adherence to appropriate procedures and operation within the limitations in the facility*s license and amendments.</b>	<b>Tier #</b>	1
	<b>Group #</b>	2
	<b>K/A #</b>	E06 EA2.2
	<b>Rating</b>	4.1

### SRO Question 8

GIVEN:

- The crew is performing FR-C.2, Degraded Core Cooling step 12, Depressurize All Intact Steam Generators to Inject Accumulators
- RCS pressure is 600 psig
- Steam generator pressures are 500 psig
- Total AFW flow is 150 gpm
- Steam generator narrow range levels are off scale low

Which of the following actions will be taken by the Shift Foreman?

- A. Continue in FR-C.2 until the procedure is complete; a secondary heat sink is not required at this time.
- B. Continue in FR-C.2 until the depressurization is complete to avoid potentially creating a RED path challenge to core cooling.
- C. Continue in FR-C.2 until the procedure is complete to avoid potentially creating a RED path challenge to core cooling.
- D. Immediately go to FR-H.1, Response to a Loss of Secondary Heat Sink to address the higher status tree priority.

Proposed Answer: D. Immediately go to FR-H.1, Response to a Loss of Secondary Heat Sink to address the higher status tree priority.

**Explanation:**

- A. Incorrect. While a heat sink would not be required if RCS pressure was LESS THAN steam generator pressure, the proper procedure usage would be to go to H.1 and then transition back to C.2.
- B. Incorrect. A caution prior to the step is to not address a valid red path on P.1 but proper procedure usage would be to finish C.2 not just the depressurization.
- C. Incorrect. This would be correct if the valid RED path was on P.1. However, if a valid red path on one of the other status trees is encountered, the proper usage is to address the red path.
- D. Correct. A valid red path, anything other than RCS integrity, must be addressed.

**Technical References:** F-0, H.1 step 2, C.2 caution and step 12

**References to be provided to applicants during exam:** None

**Learning Objective:** 9703 - Identify exit conditions for the FRPs

**Question Source:** Bank #  
 (note changes; attach parent) Modified Bank #

<b>Question History:</b>	New	X
<b>Question Cognitive Level:</b>	Last NRC Exam	
	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.43.5	



## Question 84

Examination Outline Cross-Reference	Level	SRO
Plant Fire on Site: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.	<b>Tier #</b>	1
	<b>Group #</b>	2
	<b>K/A #</b>	APE 067 G2.4.4
	<b>Rating</b>	4.7

### SRO Question 9

Unit 2 is at 100%.

A large fire has broken out in the Unit 2 Cable spreading room. The operating crew determines the reactor must be tripped and the Control Room evacuated.

Which of the following describes the appropriate procedure usage?

Note:

- E-0, Reactor Trip or Safety Injection
  - OP AP-8A, Control Room Inaccessibility - Establishing Hot Standby
- A. Enter OP AP-8A and implement E-0 to perform the immediate actions.
- B. Enter OP AP-8A, and refer to E-0, to perform the immediate actions.
- C. Enter E-0, perform the immediate actions, then go to OP AP-8A.
- D. Enter OP AP-8A only.

Proposed Answer: D. Enter OP AP-8A only.

**Explanation:**

- A. Incorrect. E-0 is not performed. Implement - Used to direct an operator to perform a procedure in parallel with the procedure he is presently performing, i.e., to do both procedures simultaneously.
- B. Incorrect. E-0 is not performed. Refer to - Used to direct an operator to consult a procedure other than the one he is using. Normally for information.
- C. Incorrect. E-0 is not performed.
- D. Correct. The immediate actions are covered by AP-8A. E-0 is not used.

**Technical References:** OP AP-8A, AD1.DC12

**References to be provided to applicants during exam:** None

**Learning Objective:**

<b>Question Source:</b> (note changes; attach parent)	Bank # P-36377 Modified Bank # New	X
<b>Question History:</b>	Last NRC Exam	
<b>Question Cognitive Level:</b>	Memory/Fundamental Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.43.5	

## Question 85

Examination Outline Cross-Reference	Level	SRO
<b>Ability to determine and interpret the following as they apply to the (Containment Flooding): Adherence to appropriate procedures and operation within the limitations in the facility*s license and amendments.</b>	<b>Tier #</b>	1
	<b>Group #</b>	2
	<b>K/A #</b>	E15 EA2.2
	<b>Rating</b>	3.3

### SRO Question 10

GIVEN:

- A LOCA occurs on Unit 2
- RWST level is 36%
- Containment Integrity status tree is MAGENTA due to Containment sump level of 98 feet

Which of the following actions will be taken by the Shift Foreman?

- A. Go to E-1.3, Transfer to Cold Leg Recirculation and address the Containment flooding once the procedure is complete.
- B. Go to E-1.3, Transfer to Cold Leg Recirculation and address the Containment flooding once one leg of recirculation is in service.
- C. Go to FR-Z.2, Response to Containment Flooding, there is another source of water entering Containment that needs to be addressed.
- D. Go to FR-Z.2, Response to Containment Flooding, however, the procedure will be exited after verifying there is no leakage and the Containment level is due to the LOCA.

Proposed Answer: C. Go to FR-Z.2, Response to Containment Flooding, there is another source of water entering Containment that needs to be addressed.

#### Explanation:

- A. Incorrect. Cold leg recirc setpoint is 33%. When performing E-1.3, CSF monitoring is done after one train of recirc is in service.
- B. Incorrect. Cold leg recirc is 33%. With a valid magenta path, the procedure needs to be addressed.
- C. Correct. Sump level greater than approximately 96 feet is higher than the level of all ECCS assumed to be injected for a LOCA. Containment Flooding as indicated by sump levels greater than the maximum expected level of 96 feet 1 inch elevation. This limit comes from the following expected to be injected into Containment: RWST, RCS, 4 accumulators
- D. Incorrect. The procedure is entered if more water than is injected is indicated in the sump. Another source of water is entering the sump. No assumption is made that it is all LOCA injection and sump level should not cause a magenta status tree during a LOCA. There are FRPs that exit early, such as H.1, or C.3.

**Technical References:** F-0, FR-C.2 background, FR-C.2, STG-II

**References to be provided to applicants during exam:** None

**Learning Objective:** 7920Q - Explain basis of emergency procedure steps (FR-Zs)

**Question Source:**

(note changes; attach parent)

Bank #

Modified Bank #

New

X

**Question History:**  
**Question Cognitive Level:**  
**10CFR Part 55 Content:**

Last NRC Exam  
Memory/Fundamental  
Comprehensive/Analysis  
55.43.5

X

## Question 86

Examination Outline Cross-Reference	Level	SRO
Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Problems with RCP seals, especially rates of seal leak-off	<b>Tier #</b>	2
	<b>Group #</b>	1
	<b>K/A #</b>	003 A2.01
	<b>Rating</b>	3.9

### SRO Question 11

The plant is at full power. All control systems are in Auto.

The following events occur:

- PK 05-01 alarms, input 1259, RCP 1-1 No. 2 Seal Leakoff Flow High
- RCP 1-1 Number 1 seal leakoff is approximately 1.0 gpm
- RCP 1-2, 1-3, 1-4 Number 1 seal leakoff is approximately 3.0 gpm per RCP
- Aux watch reports an increase in RCDT fill rate
- Charging flow is approximately 87 gpm

What action will be taken by the Shift Foreman?

- A. Per AR PK05-01, direct the operator to increase seal injection.
- B. Enter OP AP-28, RCP Malfunction, section B, "RCP Number 1 Seal Failure."
- C. Enter OP AP-28, RCP Malfunction, section C, "RCP Number 2 or 3 Seal Failure."
- D. Per AR PK05-01, direct the operator to trip the reactor, stop RCP 1-1 and enter E-0, Reactor Trip or Safety Injection.

Proposed Answer: C. Enter OP AP-28, RCP Malfunction, section C, "RCP Number 2 or 3 Seal Failure."

#### Explanation:

- A. Incorrect – leakoff flow is low, increasing injection would not address the current situation.
- B. Incorrect – the symptoms for number 1 seal failure would be high or low seal leakoff.
- C. Correct – number 2 seal leakoff flow high, coupled with a corresponding decrease in number 1 seal leakoff is an indication of a number 2 seal failure.
- D. Incorrect – a trip at this time is not warranted for a number 2 seal failure.

**Technical References:** AR PK05-01, step 2.12, AP-28, section B and C.

**References to be provided to applicants during exam:** None

**Learning Objective:** 3478 - Given initial conditions, assumptions, and symptoms, determine the correct abnormal operating procedure to be used to mitigate an operational event

**Question Source:** Bank # X  
(note changes; attach parent) Modified Bank #

**Question History:** Last NRC Exam **DCPP L061, #86, 2/2009**

**Question Cognitive Level:** Memory/Fundamental  
Comprehensive/Analysis X

**10CFR Part 55 Content:** 55.43.5

## Question 87

Examination Outline Cross-Reference	Level	SRO
Ability to (a) predict the impacts of the following malfunctions or operations on the SWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of SWS	<b>Tier #</b>	2
	<b>Group #</b>	1
	<b>K/A #</b>	076 A2.01
	<b>Rating</b>	3.7

### SRO Question 12

GIVEN:

- Unit 2 is performing a plant heatup
- RCS temperature is 400°F

The Control Room operators report the following:

- PK01-01, ASW SYS HX DELTA P/HDR PRESS (input 85, Aux Salt Wtr to CCW Hx 2-2 Press Lo) in alarm
- Running ASW pump amps have risen from 52 to 65 amps

Which action will be taken by the Unit 2 Shift Foreman?

NOTE:

- OP AP-10 – Loss of Auxiliary Saltwater
- OP AP SD-3 - Loss of Auxiliary Saltwater

- A. Go to OP AP-10 for a rupture of the ASW system.
- B. Go to OP AP-10 for fouling of the CCW heat exchanger.
- C. Go to OP AP SD-3 for a rupture of the ASW system.
- D. Go to OP AP SD-3 for fouling of the CCW heat exchanger.

Proposed Answer: A. Go to OP AP-10 for a rupture of the ASW system.

**Explanation:**

- A. Correct. AP-10 is the correct procedure and a break in ASW indications exist (low header pressure and high amps). Normal amps are 40 to 60.
- B. Incorrect. High DP and low pump amps would be the indications if there was a fouled CCW heat exchanger.
- C. Incorrect. Correct indication, however, SD-3 is used in MODE 5 and 6. Currently in MODE 3.
- D. Incorrect. SD-3 is used in MODE 5 and 6. Currently in MODE 3 and the heat exchanger is not fouled.

**Technical References:** AR PK01-10, AR PK01-03, OP AP-10, OP SD-3

**References to be provided to applicants during exam:** None

**Learning Objective:** 3478 - Given initial conditions, assumptions, and symptoms, determine the correct abnormal operating procedure to be used to mitigate an operational event

**Question Source:**

Bank #

DCPP L091 Exam

Rev 0

(note changes; attach parent)	Modified Bank #	
<b>Question History:</b>	New	X
<b>Question Cognitive Level:</b>	Last NRC Exam	
	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.43.5	

## Question 88

Examination Outline Cross-Reference	Level	SRO
Residual Heat Removal: Knowledge of EOP entry conditions and immediate action steps.	Tier #	2
	Group #	1
	K/A #	005 G2.4.1
	Rating	4.8

### SRO Question 13

GIVEN:

- The crew has just completed draining the RCS to 107 feet 8 inches, in accordance with OP A-2:III, Reactor Vessel - Draining to Half Loop/Half Loop Operations with Fuel in Vessel
- RCS temperature is stable

As the operator matches charging and letdown, RHR pump amps begin to rapidly swing approximately 10 amps, peak to peak.

Which of the following actions will be taken by the Shift Foreman?

- Direct the operator to immediately secure the running RHR pump and go to OP AP SD-2, Loss of RCS Inventory.
- Direct the operator to immediately secure the running RHR pump and go to OP AP SD-5, Loss of Residual Heat Removal.
- Direct the operator to increase charging flow and raise level to greater than 108 feet, if vortexing continues, secure the running RHR pump and go to OP AP SD-2, Loss of RCS Inventory
- Direct the operator to increase charging flow and raise level to greater than 108 feet, if vortexing continues, secure the running RHR pump and go to OP AP SD-5, Loss of Residual Heat Removal.

Proposed Answer: A. Direct the operator to immediately secure the running RHR pump and go to OP AP SD-2, Loss of RCS Inventory.

#### Explanation:

- Correct. OP A-2:III states that if at any time during the procedure, RHR cavitation or vortexing is suspected, the operator is to secure the drain, reduce RHR flow to minimum and if cavitation is excessive, secure the running RHR pump and go to AP SD-2. This is not a loss of RHR issue, it is an inventory issue (causing cavitation).
- Incorrect. The pump is cavitating and must be secured, however, the procedure to deal with it is SD-2. SD-5 attempts to correct a loss of RHR, and rising RCS temperature. The need is to raise level, done in SD-2.
- Incorrect. Vortexing is smaller oscillations.
- Incorrect. Pump is cavitating, not experiencing vortexing. This would appropriate if the pump was vortexing.

**Technical References:** OP SD-2, OP SD-5, OP A-2:III

**References to be provided to applicants during exam:** None

**Learning Objective:** 3478 - Given initial conditions, assumptions, and symptoms, determine the correct abnormal operating procedure to be used to mitigate an operational event

<b>Question Source:</b> (note changes; attach parent)	Bank # Modified Bank # New	X
<b>Question History:</b>	Last NRC Exam	
<b>Question Cognitive Level:</b>	Memory/Fundamental Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.43.5	



## Question 89

Examination Outline Cross-Reference	Level	SRO
<b>Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based Ability on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations; Loss of Instrument Bus</b>	<b>Tier #</b>	2
	<b>Group #</b>	1
	<b>K/A #</b>	013 A2.04
	<b>Rating</b>	4.2

### SRO Question 14

Unit 1 is at full power

An earthquake causes the following sequence of events to occur:

- Loss of PY-11
- Bus differential fault on 4 kV Vital Bus G
- The reactor is tripped and safety injection automatically actuates due to an RCS leak
- The RCS leak progresses to a large LOCA over the next few minutes
- Containment pressure reaches 24 psig
- E-0, Appendix E, ESF Auto Actions, Secondary And Auxiliaries Status, is complete

Which of the following actions will be taken by the Shift Foreman as a transition from E-0, Reactor Trip or Safety Injection is made?

- NOTE:
    - E-1 - Loss of Reactor or Secondary Coolant
  - FR-Z.1 - Response to High Containment Pressure
- A. Go to FR-Z.1; the Containment Integrity status tree is Red because there are no Containment Spray pumps running.
- B. Go to E-1; the one available Containment Spray pump did not start automatically but was started by the operator while performing E-0, Appendix E.
- C. Go to FR-Z.1; the Containment Integrity status tree is Magenta because there are no Containment Spray pumps running.
- D. Go to E-1; one Containment Spray pump automatically started when Containment pressure exceeded the High-High setpoint and the second Containment Spray pump started by the operator while performing E-0, Appendix E.

Proposed Answer:     B. Go to E-1; the one available Containment Spray pump did not start automatically but was started by the operator while performing E-0, Appendix E.

#### Explanation:

- A. Incorrect. The only Red path for Containment Integrity is Containment pressure greater than 47 psig. Loss of Bus G will render Containment Spray pump 1-1 unavailable. Loss of PY-11 results in the failure of the 1-2 Containment Spray pump to automatically start, due to loss of power to Train A of SSPS. However, it will be manually started while the operator is performing Appendix E.

- B. Correct. The 1-2 Containment Spray pump will not start automatically but will be started by the operator while performing Appendix E. As a result, at the transition from E-0, one spray train will be operating and the status tree will be Yellow.
- C. Incorrect. With one pump started by the operator, the status tree will be Yellow and not require attention.
- D. Incorrect. The loss of Bus G renders the 1-1 Containment Spray (Train B) pump unavailable. If the candidate does not realize the bus differential prevents the auto transfer, then they will not recognize the loss of the Containment Spray pump.

**Technical References:** OIM J-1, F-0, E-0 appendix E

**References to be provided to applicants during exam:** None

**Learning Objective:** 6764 - Describe what procedure or procedure set would be used in an emergency event, based on plant mode/conditions.

<b>Question Source:</b>	Bank #	X
(note changes; attach parent)	Modified Bank #	
	New	
<b>Question History:</b>	Last NRC Exam <b>Braidwood 2009 #79</b>	
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.43.5	

Rev 1 – removed AFW pump trip in setup, not needed.

## Question 90

Examination Outline Cross-Reference	Level	SRO
Containment Spray: Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.	Tier #	22
	Group #	1
	K/A #	026 G2.2.36
	Rating	4.2

### SRO Question 15

Unit 1 is at full power.

The following sequence of events occur:

- On the 1<sup>st</sup> at 0800, Containment Spray Pump 1-1 is declared inoperable when the pump trips during a surveillance
- On the 3<sup>rd</sup> at 1000, Containment Spray Pump 1-2 is declared inoperable when a large oil leak is found on the pump
- On the 3<sup>rd</sup> at 1200, Containment Spray Pump 1-1 is restored to OPERABLE status

Technical Specifications direct that the Containment Spray Pump 1-2 be restored to OPERABLE status no later than the:

- A. 15<sup>th</sup> at 0800
- B. 16<sup>th</sup> at 0800
- C. 17<sup>th</sup> at 1000
- D. 18<sup>th</sup> at 1000

Proposed Answer: A. 15<sup>th</sup> at 0800

#### Explanation:

- A. Correct. When 1-1 is declared inoperable, LCO 3.6.6 Condition A is entered, because its unplanned, the Completion Time is 14 days (15<sup>th</sup> at 0800) and 14 days from time of discovery. When 1-2 is declared inoperable, per Condition F directs entry into 3.0.3, for 2 hours. 3.0.3 is exited when pump 1-1 is returned to service and the plant is now back to only tracking condition A, which is still running from the start of the 1-1 pump inoperability. Of the original 14 days, only 11 days remain and may not be extended.
- B. Incorrect. No time extensions are allowed for LCOs with a modified time zero. For LCOs that do not have a modified time zero, an extension to the Completion Time is allowed.
- C. Incorrect. 14 days would again be allowed if the LCO had been met when the 1-2 was declared inoperable.
- D. Incorrect. No extension is allowed for either a modified time zero or when there is only a single failure (not overlapping failures).

**Technical References:** Tech Spec 3.6.6, Tech Spec 1.3

**References to be provided to applicants during exam:** Tech Spec 3.6.6

**Learning Objective:** 9697F - Apply TS 3.6 Technical Specification LCOs

**Question Source:** Bank #  
(note changes; attach parent) Modified Bank #

<b>Question History:</b>	New	X
<b>Question Cognitive Level:</b>	Last NRC Exam	
	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.43.2	

## Question 91

Examination Outline Cross-Reference	Level	SRO
<b>Ability to (a) predict the impacts of the following malfunctions or operations on the PZR LCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of PZR heaters</b>	<b>Tier #</b>	2
	<b>Group #</b>	2
	<b>K/A #</b>	011 A2.05
	<b>Rating</b>	3.7

### SRO Question 16

GIVEN:

- A loss of offsite power and steam generator tube rupture has occurred on Unit 2
- Only Emergency Diesel Generator 2-3 is supplying its 4 kV Vital Bus
- The crew is performing the actions of E-3, Steam Generator Tube Rupture

At step 35, Control RCS Pressure and Charging Flow to Minimize RCS-To-Secondary Leakage, the Shift Foreman determines that the action to take is to “Verify PZR heaters - ON”.

Which of the following actions will the Shift Foreman take to restore power to a group of Pressurizer backup heaters?

- A. Direct the operator to transfer a group of Pressurizer backup heaters on its vital supply, only.
- B. Implement OP AP-27, Loss of Vital 4 kV and/or 480 Bus and then place a group of Pressurizer backup heaters on its vital supply.
- C. Invoke 50.54x and 50.54y, direct ECA-0.3, Appendix X, Crosstie of Vital Bus, be performed, and then place a group of Pressurizer backup heaters on its vital supply.
- D. Obtain approval of the Site Emergency Coordinator to perform ECA-0.3, Appendix X, Crosstie of Vital Bus, and then place a group of Pressurizer backup heaters on its vital supply.

Proposed Answer: D. Obtain approval of the Site Emergency Coordinator to perform ECA-0.3, Appendix X, Crosstie of Vital Bus, and then place a group of Pressurizer backup heaters on its vital supply.

#### Explanation:

- A. Incorrect. For Unit 2, bus F is energized from EDG 2-3. Pressurizer heaters can be powered from vital buses G and H. Therefore, no heaters have vital power available, unless G or H is energized.
- B. Incorrect. AP-27 does not have an appendix for crosstie of vital buses.
- C. Incorrect. 50.54x is not applicable to the situation. Pressurizer heaters have a backup vital supply and the ability to crosstie buses is a requirement in the tech spec bases.
- D. Correct. Per ECA-0.3, performance of the appendix requires SEC approval. This will allow a vital bus to be energized, then the heaters can be transferred to backup and energized.

**Technical References:** AP-27, ECA-0.3, OIM J-1-1, Tech Spec Bases 3.4.9, E-3 step 35

**References to be provided to applicants during exam:** None

**Learning Objective:** 6764 - Describe what procedure or procedure set would be used in an emergency event, based on plant mode/conditions.

**Question Source:** Bank #  
 (note changes; attach parent) Modified Bank #  
 New

X

<b>Question History:</b>	Last NRC Exam	
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.43.5	

## Question 92

Examination Outline Cross-Reference	Level	SRO
<b>Ability to (a) predict the impacts of the following malfunctions or operation on the MT/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Malfunction of electrohydraulic control</b>	<b>Tier #</b>	2
	<b>Group #</b>	2
	<b>K/A #</b>	045 A2.17
	<b>Rating</b>	2.9

### SRO Question 17

GIVEN:

- Unit 1 ramping down at 3 MW/min
- NI power is 52%
- Turbine load is 520 MWe

The following events occur:

- PK12-12, DEH System, alarms due to the following inputs:
  - Input 706 – EH Reservoir Lvl Lo
  - Input 707 - EH Reservoir Lvl Lo-Lo
  - Input 708 – EH Fluid Lvl Lo Lockout
- The NO reports:
  - EH system pressure is 1800 psig and lowering slowly
  - EH reservoir level is 7.0 inches and lowering slowly

Which of the following actions will be taken by the Shift Foreman?

- A. Go to E-0, Reactor Trip or Safety Injection due to an automatic turbine trip above P-9.
- B. Direct a turbine trip due to a loss of EH pumps and go to OP AP-2, Full Load Rejection.
- C. Direct a trip of the turbine, and go to E-0, Reactor Trip or Safety Injection due to power above P-9.
- D. Declare all the Turbine Stop and Control Valves inoperable and go to AP-25, Rapid Load Reduction or Shutdown, to increase the ramp rate to allow isolating the steam to the turbine within 6 hours.

Proposed Answer: C. Direct a trip of the turbine, and go to E-0, Reactor Trip or Safety Injection due to power above P-9.

#### Explanation:

- A. Incorrect. Although a turbine trip is required, power is above P-9 based on NI power (which causes the PK to the OUT). Turbine power is less than 50% (due to plant inefficiency) but unlike P-13, which uses turbine power, P-9 uses NI's.
- B. Incorrect. No automatic turbine trip on EH pressure.
- C. Correct. P-9 is 2/4 POWER RANGE channels above 50% power. The PK is lit below P-9. Due to the loss of the EH pumps (locked out on low-low level), a turbine trip is required, the reactor should then auto trip.
- D. Incorrect. The ECG addresses Turbine Stop and Control valves, which are held open by EH pressure, however, the ECG does not address EH system failures. Additionally, for the current conditions, a turbine trip is required.

**Technical References:** ECG 4.4, OP AP-2, PK12-12 step 2.3, PK08-04

**References to be provided to applicants during exam:** ECG 4.4

**Learning Objective:** 41077 - Discuss abnormal conditions associated with the Turbine Control Oil System

<b>Question Source:</b>	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.43.5	



## Question 93

Examination Outline Cross-Reference	Level	SRO
<b>Condensate: Ability to recognize system parameters that are entry-level conditions for Technical Specifications</b>	<b>Tier #</b>	2
	<b>Group #</b>	2
	<b>K/A #</b>	056 G2.2.42
	<b>Rating</b>	4.6

### SRO Question 18

Unit 1 is at full power.

A leak develops at a CST 1-1 weld and subsequently stops when CST level reaches 175,000 gallons.

What is the impact on the OPERABILITY of the CST and AFW systems?

- A. The CST is inoperable and the AFW system will be inoperable if the CST is not returned to OPERABLE status within 7 days.
- B. The CST is inoperable; the AFW system is OPERABLE.
- C. The CST and the AFW system are inoperable.
- D. No impact, the CST and AFW are OPERABLE.

Proposed Answer: B. The CST is inoperable; the AFW system is OPERABLE.

#### Explanation:

- A. Incorrect. From OP1.DC38, 5.10 Common Support Systems, 5.10.2 - CST - The AFW system will not be able to perform its design function without a supply of water for RCS decay heat removal via the SGs. The Required Actions for inoperability of the CST is more restrictive than for the case if all three AFW trains are inoperable. The appropriate action is to follow the TS Required Actions for an inoperable CST and not to enter the Required Actions for an inoperable AFW system. The LOSF evaluation will conclude that although there is a degradation for maintaining an AFW heat sink there is not a loss of safety function as long as there is useable inventory. The CST Required Actions are bounding for this case.
- B. Correct. For Unit 1, required level is 200,000 gallons, therefore, the CST is inoperable. As stated above, the AFW system will remain OPERABLE and the Tech Specs are not "cascaded"
- C. Incorrect. Only the CST is inoperable.
- D. Incorrect. Unit 2 only requires 166,000 gallons, however, for Unit 1, the CST is inoperable.

**Technical References:** OP1.DC38, Tech Specs 3.7.5, 3.7.6 and 5.5.15

**References to be provided to applicants during exam:** Tech Spec 3.7.5 and 3.7.6

**Learning Objective:** 5853 - Explain the Safety Function Determination Program (SFDP) 9697G- Apply TS 3.7 Technical Specification LCOs

**Question Source:** Bank #  
(note changes; attach parent) Modified Bank #  
New

X

**Question History:** Last NRC Exam

<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.43.1	

NOTE: Unit Difference

Rev 1 – clarified explanation to show that the AFW system TS is not impacted and not entered for an inoperable CST.

## Question 94

Examination Outline Cross-Reference	Level	SRO
<b>Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.</b>	<b>Tier #</b>	3
	<b>Group #</b>	1
	<b>K/A #</b>	G2.1.5
	<b>Rating</b>	3.9

### SRO Question 19

Both units are at full power.

In accordance with OP1.DC37, Plant Logs, if qualified, who would be the preferred SRO to assign the STA function if the Shift Manager is not a qualified STA?

- A. Unit 1 Shift Foreman
- B. Unit 2 Shift Foreman
- C. An extra Shift Foreman
- D. Work Control Shift Foreman

Proposed Answer: D. Work Control Shift Foreman

#### Explanation:

- A. Incorrect. The WCSFM is the preferred watchstander after the SM. Unlike some E-plan assignments, (IEOC or communicator) there is no unit preference, only that if the SFM is assigned, then both SFM must be qualified STA.
- B. Incorrect. See A.
- C. Incorrect. An extra or screen SFM is the last preference.
- D. Correct. WCSFM is the preferred watchstander if the SM is not qualified.

**Technical References:** OP1.DC37

**References to be provided to applicants during exam:** None

**Learning Objective:** 7845 - Demonstrate the ability to apply administrative guide lines relative to shift staffing

<b>Question Source:</b> (note changes; attach parent)	Bank # Modified Bank # New	X
<b>Question History:</b>	Last NRC Exam	
<b>Question Cognitive Level:</b>	Memory/Fundamental Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.43.1	

Rev 1 – replacement question

## Question 95

<b>Examination Outline Cross-Reference</b>	<b>Level</b>	SRO
<b>Knowledge of the fuel-handling responsibilities of SROs</b>	<b>Tier #</b>	3
	<b>Group #</b>	1
	<b>K/A #</b>	G2.1.35
	<b>Rating</b>	3.9

### SRO Question 20

Which of the following identifies the first CORE ALTERATION activity requiring the presence of the Refueling SRO in Containment?

- A. Lifting the reactor vessel head
- B. Unlatching RCCAs
- C. Lifting the upper internals
- D. Moving the first fuel assembly

**Proposed Answer:** B. Unlatching RCCAs

#### **Explanation:**

**CORE ALTERATION - shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel.** Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

- A. Incorrect. Prior to unlatching RCCAs, but not a core alteration
- B. Correct. Per OP B-8DS2, the Refueling SRO is responsible for supervising Core Alterations. RCCA unlatching and moving of fuel assemblies are core alterations. Unlatching is performed prior to removal of the fuel assembly.
- C. Incorrect. Performed after unlatching RCCAs
- D. Incorrect. Core alteration performed after unlatching RCCAs

**Technical References:** OP B-8D, Tech Spec 1.0, definition

**References to be provided to applicants during exam:** None

**Learning Objective:** 6497 - State the responsibilities and duties of Refueling SRO

**Question Source:** Bank # 96 DCPD L031 NRC Exam 2/2005 X  
 (note changes; attach parent) Modified Bank #  
 New

**Question History:** Last NRC Exam

**Question Cognitive Level:** Memory/Fundamental X  
 Comprehensive/Analysis

**10CFR Part 55 Content:** 55.43.7

## Question 96

Examination Outline Cross-Reference	Level	SRO
Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.	Tier #	3
	Group #	2
	K/A #	G2.2.25
	Rating	4.2

### SRO Question 21

Under what condition may LCO 3.0.5 be used to restore inoperable equipment to service and what is the reason the LCO is used?

- A. Demonstrating the OPERABILITY of the equipment, because the applicable LCO Required Action is not being met while the equipment is in service and not OPERABLE.
- B. Performing corrective maintenance or troubleshooting, because the applicable LCO Required Action is not being met while the equipment is in service and not OPERABLE.
- C. Demonstrating the OPERABILITY of the equipment, because without the exception, the Required Action is to enter LCO 3.0.3.
- D. Performing corrective maintenance or troubleshooting, because without the exception, the Required Action is to enter LCO 3.0.3.

Proposed Answer: A. Demonstrating the OPERABILITY of the equipment, because the applicable LCO Required Action is not being met while the equipment is in service and not OPERABLE.

#### Explanation:

- A. Correct. Bases for 3.0.5 states: LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s) to allow the performance of required testing to demonstrate:
  - a. The OPERABILITY of the equipment being returned to service;
  - OR
  - b. The OPERABILITY of other equipment.The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the required testing to demonstrate OPERABILITY. This specification does not provide time to perform any other preventive or corrective maintenancex
- B. Incorrect. Not allowed for testing or corrective maintenance.
- C. Incorrect. It is an exception for LCO 3.0.2 and does not require 3.0.3 entry.
- D. Incorrect. Not allowed for testing and does not preclude entry into 3.0.3

**Technical References:** B3.0.5

**References to be provided to applicants during exam:** None

**Learning Objective:** 9700 - Explain LCO 3.0 & SR 3.0 general requirements

**Question Source:** Bank #

(note changes; attach parent)	Modified Bank #	
<b>Question History:</b>	New	X
<b>Question Cognitive Level:</b>	Last NRC Exam	
	Memory/Fundamental	X
<b>10CFR Part 55 Content:</b>	Comprehensive/Analysis	
	55.43.2	

## Question 97

Examination Outline Cross-Reference	Level	SRO
<b>Ability to determine operability and/or availability of safety related equipment.</b>	<b>Tier #</b>	3
	<b>Group #</b>	2
	<b>K/A #</b>	G2.2.37
	<b>Rating</b>	4.6

### SRO Question 22

GIVEN:

- Unit 1 is at full power
- Unit 2 is in MODE 3
- Control Room Ventilation Supply Fans S-38 and S-40 are out of service

Control Room Ventilation Supply Fan S-36 trips.

Which of the following Required Actions, if any, must be taken?

- A. No action required for either unit, LCO 3.7.10 is satisfied.
- B. Unit 1 must enter LCO 3.7.10 Condition A. No Action required for Unit 2.
- C. Both Unit 1 and Unit 2 must enter LCO 3.7.10 Condition A.
- D. Both Unit 1 and Unit 2 must enter LCO 3.7.10 Condition B.

Proposed Answer: A. No action required for either unit, LCO 3.7.10 is satisfied.

#### **Explanation:**

- A. Correct. TS 3.7.10 requires two CRVS trains OPERABLE in MODES 1-6 and during movement of irradiated fuel assemblies. Each train must consist of one main supply fan, one filter booster fan, one pressurization fan, and one HEPA and Charcoal Absorber System. The trains are divided by unit; Train 1 includes Unit 1 equipment, and Train 2 includes Unit 2 equipment. For Unit 1, Supply Fans S-35 and S39 or S-36 and S-40 must be OPERABLE. With S-40 inoperable, the combination must be S-35 and S-39. Both are still available when S-36 trips. The Unit 2 fan inoperability, S-38, does not impact the LCO. Therefore, LCO 3.7.10 is met.
- B. Incorrect. LCO 3.7.10 applies to both units, also, both trains for CRVS are OPERABLE.
- C. Incorrect. Both trains remain OPERABLE. This would be the condition if fan S-35 was the fan to trip.
- D. Incorrect. Not stated in the LCO, but the background, but Condition B does not apply to fans, but to the physical control room envelope.

**Technical References:** OP H-5:III, LCO 3.7.10, Bases 3.7.10

**References to be provided to applicants during exam:** OP H-5:III attachment 1 and LCO 3.7.10

**Learning Objective:** 9694G - Apply TS 3.7 Technical Specification bases

**Question Source:** Bank #  
 (note changes; attach parent) Modified Bank #  
 New

X

**Question History:**  
**Question Cognitive Level:**  
**10CFR Part 55 Content:**

Last NRC Exam  
Memory/Fundamental  
Comprehensive/Analysis  
55.43.2

X



## Question 98

Examination Outline Cross-Reference	Level	SRO
Ability to approve release permits	<b>Tier #</b>	3
	<b>Group #</b>	3
	<b>K/A #</b>	G2.3.6
	<b>Rating</b>	3.8

### SRO Question 23

GIVEN

- The Shift Foreman (SFM) is reviewing a purge of Containment in accordance with Attachment 11.3, Authorization for Discharge of Containment Atmosphere for MODES 1-4 of CAP A-6, Gaseous Discharge Management
- The SFM notes that the time of the purge is subtracted from 200 hours for the year

Why does the SFM ensure the valves will not be open for more than 200 hours for the year prior to approving the permit?

- A. To minimize the probability of a LOCA occurring while the valves are open, limiting the offsite boundary doses.
- B. To prevent exceeding the NPDES permit.
- C. To minimize the probability the valves will not fully close when the purge is secured, resulting in a breach of Containment integrity.
- D. To prevent valve erosion and subsequent excessive leakage past the valves when they are closed.

Proposed Answer: A. To minimize the probability of a LOCA occurring while the valves are open, limiting the offsite boundary doses.

#### Explanation:

- A. Correct. Per ECG 23.3, The purging time restriction is meant to minimize the probability of a LOCA while conducting purging operations and thereby limit offsite boundary doses. The 200 hour per calendar year restriction reflects a regulatory compromise between an NRC proposed limit of 90 hours per year and a PG&E counter-proposal of 1000 hours per year.
- B. Incorrect. The NPDES covers releases to the environment, but releases to the ocean, not from Containment.
- C. Incorrect. The valves are limited in travel because these valves have not been qualified to close under accident conditions. Knowing that there is a concern with closing makes this distractor plausible.
- D. Incorrect. these butterfly valves are not fully open (limited to 50 degrees of travel), so its plausible that there is concern of wear.

**Technical References:** ECG 23.3 bases, CAP A-6,

**References to be provided to applicants during exam:** none

**Learning Objective:** 66068 - Apply the requirements of System 39 ECGs.

7428 - State gaseous radwaste system administrative controls

**Question Source:** Bank #

(note changes; attach parent)	Modified Bank	
<b>Question History:</b>	New	X
<b>Question Cognitive Level:</b>	Last NRC Exam	
<b>10CFR Part 55 Content:</b>	Memory/Fundamental	X
	Comprehensive/Analysis	
	55.43.4	
Rev 1 – replacement question		

## Question 99

Examination Outline Cross-Reference	Level	SRO
<b>Knowledge of how abnormal operating procedures are used in conjunction with EOPs</b>	<b>Tier #</b>	3
	<b>Group #</b>	4
	<b>K/A #</b>	G2.4.8
	<b>Rating</b>	4.5

### SRO Question 24

GIVEN:

- The crew is performing E-0.1, Reactor Trip Response
- A 10% steam dump valve is leaking by
- RCS temperature is 540°F and decreasing at a rate of approximately 2 to 3°F/minute
- RCS pressure is stable at approximately 2225 psig
- Pressurizer level is 25% and stable
- AFW flow is throttled to approximately 440 gpm
- MSIVs are closed

Which of the following actions will be taken by the Shift Foreman?

- A. Go to OP AP-6, Emergency Boration and then return to E-0.1.
- B. Continue in E-0.1 and have another watchstander initiate emergency boration by implementing OP AP-6, Emergency Boration.
- C. Continue in E-0.1 and have another watchstander initiate emergency boration by referring to OP AP-6, Emergency Boration.
- D. Direct the operator to initiate Safety Injection and return to step 1 of E-0.

Proposed Answer: B. Continue in E-0.1 and have another watchstander initiate emergency boration by implementing OP AP-6, Emergency Boration.

**Explanation:**

- A. Incorrect. The progression thru E-0.1 is to continue. The instructions are to implement OP AP-6.
- B. Correct. Per E-0.1, AP-6 is implemented. The procedure will be handled by another operator while the SFM stays in the higher priority.
- C. Incorrect. The procedure is implemented. It is not to be used at the operators discretion.
- D. Incorrect. Subcooling and pressurizer level are sufficient, SI is not required (per the foldout page).

**Technical References:** OP1.DC10, E-0.1 step 1 & foldout page

**References to be provided to applicants during exam:** None

**Learning Objective:** 3478 - Given initial conditions, assumptions, and symptoms, determine the correct abnormal operating procedure to be used to mitigate an operational event

**Question Source:** Bank #83 DCPP 2/2006 NRC Exam X  
 (note changes; attach parent) Modified Bank #  
 New

**Question History:** Last NRC Exam No

**Question Cognitive Level:** Memory/Fundamental

**10CFR Part 55 Content:**

Comprehensive/Analysis  
55.43.5

X

## Question 100

Examination Outline Cross-Reference	Level	SRO
Knowledge of the specific bases for EOPs	Tier #	3
	Group #	4
	K/A #	G2.4.18
	Rating	4.4

### SRO Question 25

EOP ECA-1.2, LOCA Outside Containment, has been entered from EOP E-0, Reactor Trip or Safety Injection.

A few minutes after isolating the second train of RHR, the following conditions exist:

- ECCS flow is stable
- RCS pressure is 400 psig and stable

Which of the following describes the status of the LOCA and required action to be taken by the Shift Foreman?

- A. The LOCA is isolated; go to E-1.1 to perform SI termination.
- B. The LOCA is NOT isolated, go to EOP ECA-1.1, Loss of Emergency Coolant Recirculation to delay depletion of the RWST by adding makeup and reducing outflow.
- C. The LOCA is isolated; go to E-1 to check plant conditions and check if SI termination can be performed.
- D. The LOCA is NOT isolated, go to EOP ECA-1.1, Loss of Emergency Coolant Recirculation to attempt to establish conditions to restore recirculation.

Proposed Answer: B. The LOCA is NOT isolated, go to EOP ECA-1.1, Loss of Emergency Coolant Recirculation to delay depletion of the RWST by adding makeup and reducing outflow.

### Explanation:

To answer this question, the SRO must know the bases which explains what the expected response is if the LOCA is isolated, the procedures that can be entered from ECA-1.2 and know that to get to ECA-1.2 from E-0 means that no kick out was made to another EOP (specifically E-1) and therefore, there must not be any inventory in the containment sump.

A. Incorrect. The bases for ECA-1.2 states: This step instructs the operator to check RCS pressure to determine if the break has been isolated by previous actions. If the break is isolated in Step 2, a significant RCS pressure increase will occur due to the SI flow filling up the RCS with break flow stopped.

The operator transfers to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, if the break has been isolated, for further recovery actions. If the break has not been isolated, the operator is sent to ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, for further recovery actions since there will be no inventory in the sump.

B. Correct. The break is not isolated, ECA-1.1 is the appropriate procedure. The objective of the loss of ECR guideline is threefold: 1) to continue attempts to restore emergency coolant recirculation capability, 2) to delay depletion of the RWST by adding makeup fluid

and reducing outflow, and 3) to depressurize the RCS to minimize break flow and cause SI accumulator injection

C. Incorrect. The break is not isolated.

D. Incorrect. There is no inventory in the sump, recirculation cannot be established. ECA-1.2 is entered from E-0 from step 26 after the diagnostic steps fail to provide a transition. Step 26 checks for LOCA outside containment.

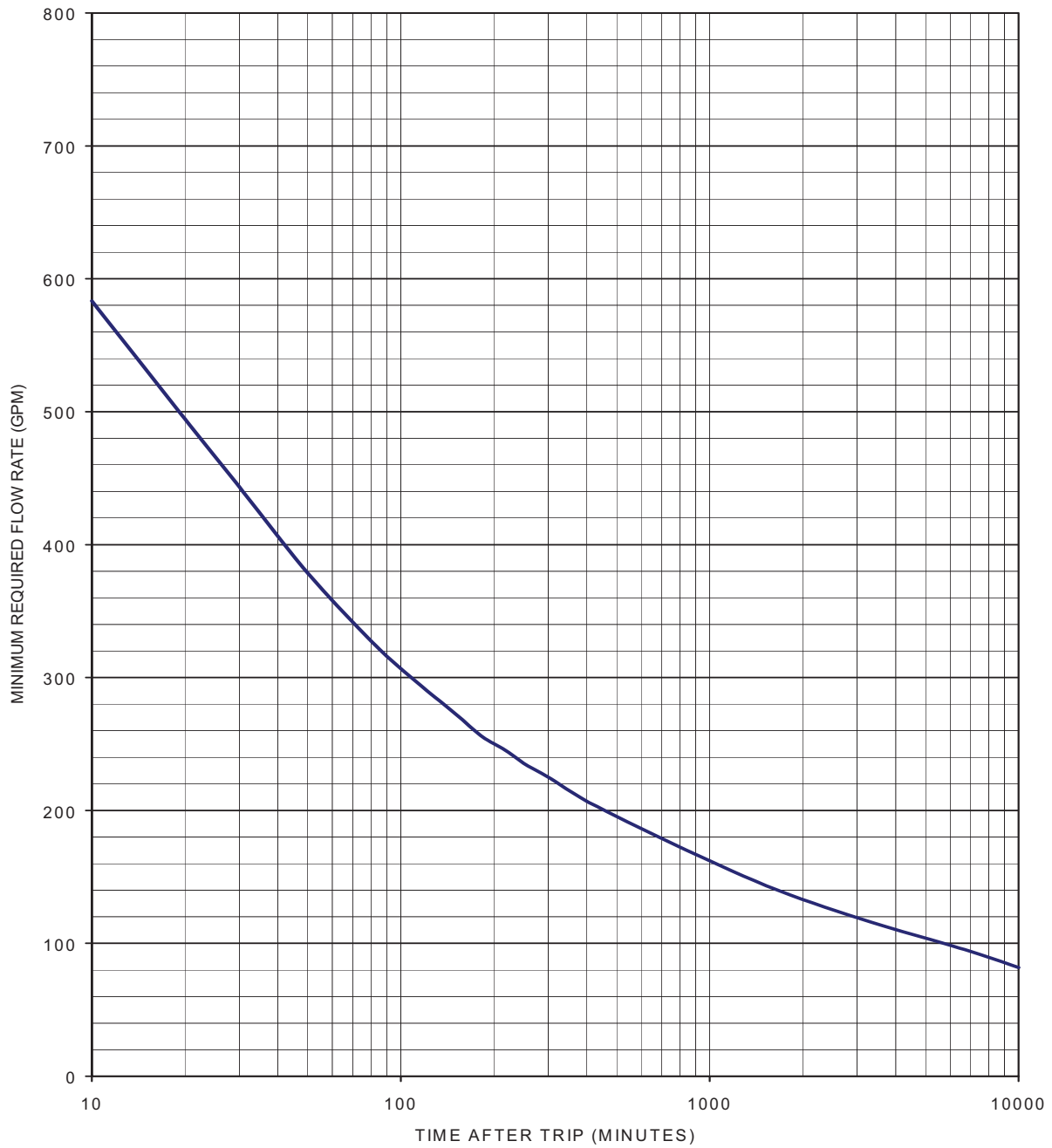
**Technical References:** E-0, ECA-1.1 and 1.2 & background

**References to be provided to applicants during exam:** None

**Learning Objective:** 3552 - Given initial conditions, assumptions, and symptoms, determine the correct Emergency Operating Procedure to be used to mitigate an operational event

<b>Question Source:</b>	Bank	
(note changes; attach parent)	Modified Bank #	
	New	X
<b>Question History:</b>	Last NRC Exam	No
<b>Question Cognitive Level:</b>	Memory/Fundamental	
	Comprehensive/Analysis	X
<b>10CFR Part 55 Content:</b>	55.43.5	

APPENDIX G  
Minimum ECCS Flow Rate After Trip



DIABLO CANYON POWER PLANT OPERATION DATA  
 TABLE R19-1T-1  
 Current Values of Reactor Operating Factors  
 Cycle 17

# 1 UNIT

**DELAYED NEUTRON FRACTION**

	BOL	MOL	EOL
$\bar{\beta}$	0.006317	0.005584	0.005221
$\bar{I}$	0.97	0.97	0.97
$\bar{\beta}_{eff}$	0.006127	0.005416	0.005064

**CONTROL ROD WORTH (BOL, HZP, NO XENON)**

BANK	NO. OF RODS	REACTIVITY WORTH (pcm)
CONTROL A (D, C, B PRESENT)	8	1028
CONTROL B (D, C PRESENT)	4	582
CONTROL C (D PRESENT)	8	638
CONTROL D	9	700
<hr/>		
TOTAL FOR CONTROL	29	2948
TOTAL FOR SHUTDOWN	24	3596
<hr/>		
TOTAL	53	6544

Most reactive single stuck rod at BOL is D-12; Worth = -665 pcm  
 Most reactive single stuck rod at EOL is E-13; Worth = -1236 pcm

**ENRICHMENT**

Zone 13A = 4.398 w/o  
 Zone 16A = 4.198 w/o  
 Zone 17A = 4.693 w/o    Zone 17B = 4.948 w/o  
 Zone 18A = 4.454 w/o    Zone 18B = 4.899 w/o  
 Zone 19A = 4.400 w/o    Zone 19B = 4.800 w/o

**CYCLE ARO POSITION DATA**

ARO Rod Height = 231 Steps for Shutdown and Control Banks

Rod Control Thumbwheel Settings:

S1=128 S2=231 S3=256 S4=359 S5=384 S6=487

MINIMUM REQUIRED BORON CONCENTRATION FOR MODE 6: 2100 PPM

SOURCE: WCAP-17273-P, Rev 0, Pages 2-9, 6-7, A-4, and A-7;

Rod Wear Management Plan, 9/90



TS 3.7.10 requires two CRVS trains OPERABLE in MODES 1-6 and during movement of irradiated fuel assemblies. Each train must consist of one main supply fan, one filter booster fan, one pressurization fan, and one HEPA and Charcoal Absorber System. The trains are divided by unit; Train 1 includes Unit 1 equipment, and Train 2 includes Unit 2 equipment. The minimum equipment required in each train is shown in the tables below.

TRAIN 1 - UNIT 1—For Train 1 to be OPERABLE, ONE of the following conditions must be met:

- Table 1, "U1 Bus H Selected Equipment", must ALL be as specified
- OR
- Table 2, "U1 Bus F Selected Equipment", must ALL be as specified

TRAIN 2 - UNIT 2—For Train 2 to be OPERABLE, ONE of the following conditions must be met:

- Table 3, "U2 Bus H Selected Equipment", must ALL be as specified
- OR
- Table 4, "U2 Bus F Selected Equipment", must ALL be as specified

<b>Table 1—U1 Bus H Selected Equipment</b>	
<u>Equipment</u>	<u>Specified State</u>
S-35 Supply Fan	OPERABLE
S-39 Supply Fan	OPERABLE
S-99 Pressurization Fan AND Dampers 1-1 and 1-1A OR S-98 Pressurization Fan AND Dampers 1-1B and 1-1C	OPERABLE
U1 HEPA and Charcoal Filter Absorber System*	OPERABLE
Dampers 2*, 2A*, 3*, 3A*, 7*, 7A*, 8*, 8A*, 11, 11A, 12, 12A, 14	ALL OPERABLE OR Closed
Dampers 4, 6, 9, 9A, 10, 10A, 13	ALL OPERABLE OR Open
Damper 5	OPERABLE OR Blocked in the 5700 CFM throttled position
Transfer Switch A-2**	Selected to a separate operable vital bus
<p>* If any of the noted dampers (or damper pairs) are not OPERABLE, closed, or have an isolation blank installed and bubble tested to 10" w.g. minimum pressure, enter TS 3.7.10.B action statement.</p> <p>** If Transfer Switches A-2 and D-2 are selected to the same vital bus, U1 Bus H selected equipment and U2 Bus H selected equipment will be powered from the same vital bus. If this configuration exists, declare either U1 Bus H selected equipment OR U2 Bus H selected equipment INOPERABLE. <sup>T26090</sup></p>	

Control Room Ventilation Trains

**U1&2** Attachment 1: Page 2 of 3

<b>Table 2—U1 Bus F Selected Equipment</b>	
<u>Equipment</u>	<u>Specified State</u>
S-36 Supply Fan	OPERABLE
S-40 Supply Fan	OPERABLE
S-99 Pressurization Fan AND Dampers 1-1 and 1-1A OR S-98 Pressurization Fan AND Dampers 1-1B and 1-1C	OPERABLE
U1 HEPA and Charcoal Filter Absorber System*	OPERABLE
Dampers 2*, 2A*, 3*, 3A*, 7*, 7A*, 8*, 8A*, 9, 9A, 10, 10A, 13	ALL OPERABLE OR Closed
Dampers 4, 6, 11, 11A, 12, 12A, 14	ALL OPERABLE OR Open
Damper 5	OPERABLE OR Blocked in the 5700 CFM throttled position
Transfer Switch B-2	Selected to a separate operable vital bus
* If any of the noted dampers (or damper pairs) are not OPERABLE, closed, or have an isolation blank installed and bubble tested to 10" w.g. minimum pressure, enter TS 3.7.10.B action statement.	

<b>Table 3—U2 Bus H Selected Equipment</b>	
<u>Equipment</u>	<u>Specified State</u>
S-37 Supply Fan	OPERABLE
S-41 Supply Fan	OPERABLE
S-97 Pressurization Fan AND Dampers 2-1 and 2-1A OR S-96 Pressurization Fan AND Dampers 2-1B and 2-1C	OPERABLE
U2 HEPA and Charcoal Filter Absorber System*	OPERABLE
Dampers 2*, 2A*, 3*, 3A*, 7*, 7A*, 8*, 8A*, 11, 11A, 12, 12A, 14	ALL OPERABLE OR Closed
Dampers 4, 6, 9, 9A, 10, 10A, 13	ALL OPERABLE OR Open
Damper 5	OPERABLE OR Blocked in the 5700 CFM throttled position
Transfer Switch D-2**	Selected to a separate operable vital bus
* If any of the noted dampers (or damper pairs) are not OPERABLE, closed, or have an isolation blank installed and bubble tested to 10" w.g. minimum pressure, enter TS 3.7.10.B action statement.	
** If Transfer Switches D-2 and A-2 are selected to the same vital bus, U2 Bus H selected equipment and U1 Bus H selected equipment will be powered from the same vital bus. If this configuration exists, declare either U2 Bus H selected equipment OR U1 Bus H selected equipment INOPERABLE. <sup>T28090</sup>	

<b>Table 4—U2 Bus F Selected Equipment</b>	
<u>Equipment</u>	<u>Specified State</u>
S-38 Supply Fan	OPERABLE
S-42 Supply Fan	OPERABLE
S-97 Pressurization Fan AND Dampers 2-1 and 2-1A OR S-96 Pressurization Fan AND Dampers 2-1B and 2-1C	OPERABLE
U2 HEPA and Charcoal Filter Absorber System*	OPERABLE
Dampers 2*, 2A*, 3*, 3A*, 7*, 7A*, 8*, 8A*, 9, 9A, 10, 10A, 13	ALL OPERABLE OR Closed
Dampers 4, 6, 11, 11A, 12, 12A, 14	ALL OPERABLE OR Open
Damper 5	OPERABLE OR Blocked in the 5700 CFM throttled position
Transfer Switch E-2	Selected to a separate operable vital bus
* If any of the noted dampers (or damper pairs) are not OPERABLE, closed, or have an isolation blank installed and bubble tested to 10" w.g. minimum pressure, enter TS 3.7.10.B action statement.	

# U1&2 Attachment 2: Page 14 of 15

## Emergency Action Level Matrix

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
<b>F</b> Fission Product Barriers	FG1.1 1 2 3 4 Loss of ANY two barriers AND Loss or potential loss of third barrier (Table F-1) (Note 5)	FS1.1 1 2 3 4 Loss or potential loss of ANY two barriers (Table F-1) (Note 5)	FA1.1 1 2 3 4 ANY loss or ANY potential loss of either Fuel Clad or RCS (Table F-1) (Note 5)	FU1.1 1 2 3 4 ANY loss or ANY potential loss of Containment (Table F-1) (Note 5)
	Note 5: The logic used for these initiating conditions reflects the following considerations: <ul style="list-style-type: none"> <li>The Fuel Clad barrier and the RCS barrier are weighted more heavily than the Containment barrier. UE EALs associated with RCS and Fuel Clad barriers are addressed under System Malfunction EALs.</li> <li>At the Site Area Emergency level, there must be some ability to dynamically assess how far present conditions are from the threshold for a General Emergency. For example, if Fuel Clad and RCS barrier "loss" EALs existed, that, in addition to offsite dose assessments, would require continual assessments of radioactive inventory and containment integrity. Alternatively, if both Fuel Clad and RCS barrier "Potential Loss" EALs existed, the SM/SEC/IED would have more assurance that there was no immediate need to escalate to a General Emergency.</li> <li>The ability to escalate to higher emergency classes as an event deteriorates must be maintained. For example, RCS leakage steadily increasing would represent an increasing risk to public health and safety.</li> </ul>			

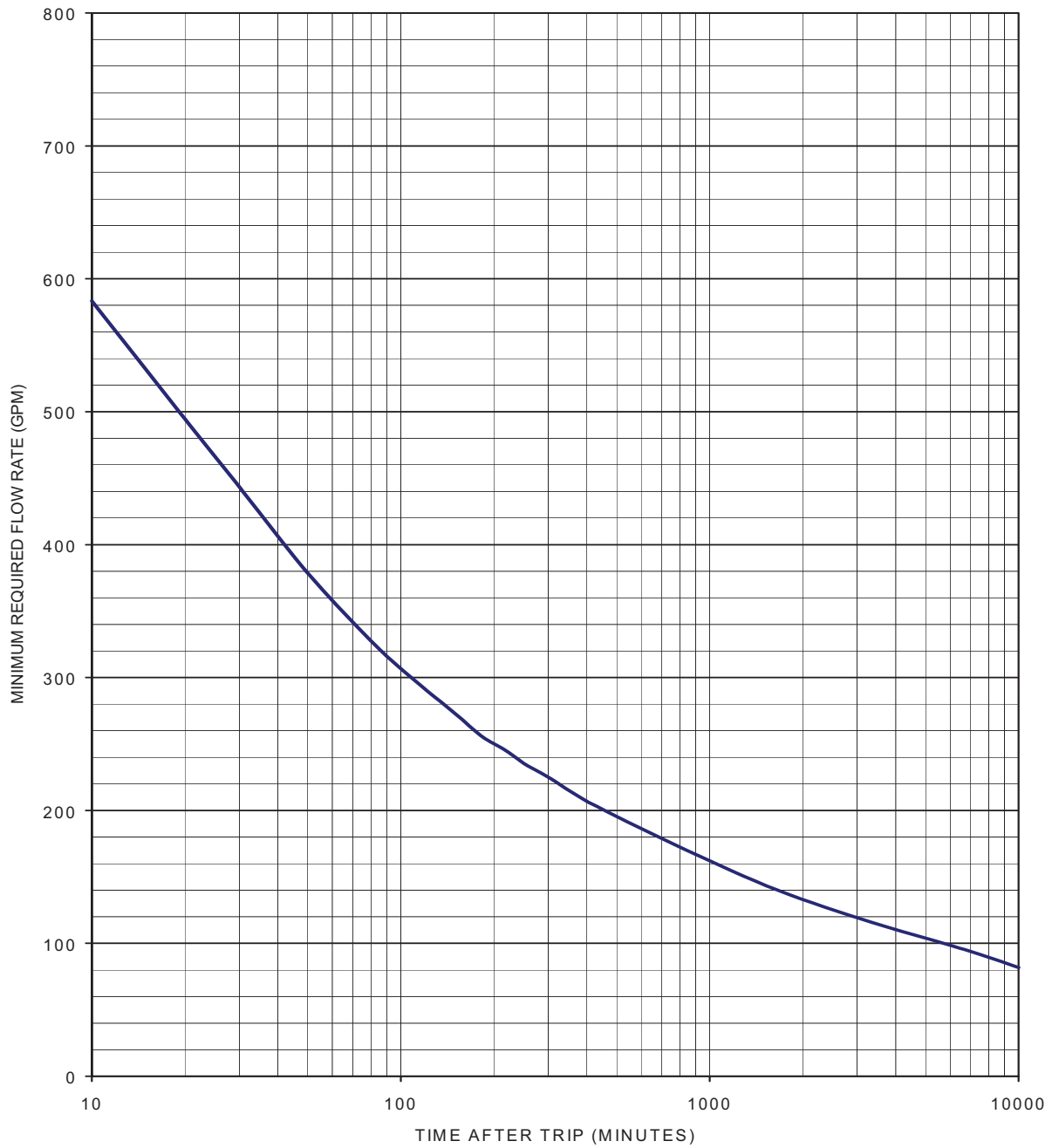
Emergency Action Level Matrix

Table F-1 Fission Product Barrier Matrix						
	Fuel Cladding Barrier		Reactor Coolant System Barrier		Containment Barrier	
	Loss	Potential Loss	Loss	Potential Loss	Loss	Potential Loss
<b>A. CSFST</b>	<input type="checkbox"/> 1. CSFST Core Cooling-RED <input type="checkbox"/> 2. Core exit TCs > 1,200°F	<input type="checkbox"/> 1. CSFST Core Cooling-MAGENTA OR CSFST Heat Sink-RED and heat sink required	None	<input type="checkbox"/> 1. CSFST RCS Integrity-RED OR CSFST Heat Sink-RED and heat sink required	None	<input type="checkbox"/> 1. CSFST Containment-RED
<b>B. Core Exit TCs</b>	<input type="checkbox"/> 1. CSFST Core Cooling-RED <input type="checkbox"/> 2. Core exit TCs > 1,200°F	<input type="checkbox"/> 1. CSFST Core Cooling-RED OR CSFST Heat Sink-RED and heat sink required <input type="checkbox"/> 2. Core exit TCs > 700°F	None	None	None	<input type="checkbox"/> 2. Core exit TCs > 1,200°F AND Restoration procedures not effective within 15 min. <input type="checkbox"/> 3. ALL of the following: - Core exit TCs > 700°F - Reactor Vessel water level < Table F-2 thresholds - Restoration procedures not effective within 15 min.
<b>C. Radiation</b>	<input type="checkbox"/> 3. Containment radiation (RM-30 or RM-31) > 20 R/hr <input type="checkbox"/> 4. With shutdown in service, EP-RB-14A Dose Point radiation > 15 R/hr <input type="checkbox"/> 5. SGTR in progress AND MSR radiation (RM-71, 72, 73 or 74) > 5.0E4 opm (> 5 min. after reactor shutdown)	None	<input type="checkbox"/> 1. Containment radiation (RM-30 or RM-31) > 6 R/hr	None	None	<input type="checkbox"/> 4. Containment radiation (RM-30 or RM-31) > 80 R/hr
<b>D. Inventory</b>	<input type="checkbox"/> 3. Reactor Vessel water level < Table F-2 thresholds <input type="checkbox"/> 6. Coolant activity > 300 µCi/gm Dose Equivalent I-131	<input type="checkbox"/> 1. RCS leak rate > available makeup capacity as indicated by a loss of RCS subcooling <input type="checkbox"/> 2. Unsoluble PCS leak exceeding the capacity of one charging pump in the normal charging mode (150 gpm) <input type="checkbox"/> 3. SGTR that results in an ECCS (SI) actuation	<input type="checkbox"/> 1. Rapid unexplained Containment pressure drop following initial increase <input type="checkbox"/> 2. Following LOCA, Containment pressure or sump level response not consistent with LOCA conditions <input type="checkbox"/> 3. Ruptured S/G is also faulted outside of Containment <input type="checkbox"/> 4. Primary-to-secondary leakage > 10 gpm with non-insoluble steam release from affected S/G to the environment <input type="checkbox"/> 5. Valves(s) not closed AND Direct pathway to the environment exists after Containment isolation signal	<input type="checkbox"/> 1. RCS leak rate > available makeup capacity as indicated by a loss of RCS subcooling <input type="checkbox"/> 2. Unsoluble PCS leak exceeding the capacity of one charging pump in the normal charging mode (150 gpm) <input type="checkbox"/> 3. SGTR that results in an ECCS (SI) actuation	<input type="checkbox"/> 1. Rapid unexplained Containment pressure drop following initial increase <input type="checkbox"/> 2. Following LOCA, Containment pressure or sump level response not consistent with LOCA conditions <input type="checkbox"/> 3. Ruptured S/G is also faulted outside of Containment <input type="checkbox"/> 4. Primary-to-secondary leakage > 10 gpm with non-insoluble steam release from affected S/G to the environment <input type="checkbox"/> 5. Valves(s) not closed AND Direct pathway to the environment exists after Containment isolation signal	<input type="checkbox"/> 5. Containment pressure 47 psig and increasing <input type="checkbox"/> 6. Containment hydrogen concentration > 4% <input type="checkbox"/> 7. Containment pressure > 22 psig with < one full train of depressurization equipment operating Note: One Containment Spray pump and two CFCUs comprise one full train of depressurization equipment
<b>E. Other</b>	<input type="checkbox"/> 7. ANY condition in the opinion of the SM/SEC/ED that indicates potential loss of the Fuel Clad barrier	<input type="checkbox"/> 4. ANY condition in the opinion of the SM/SEC/ED that indicates potential loss of the Fuel Clad barrier	<input type="checkbox"/> 4. ANY condition in the opinion of the SM/SEC/ED that indicates loss of the RCS barrier	<input type="checkbox"/> 3. ANY condition in the opinion of the SM/SEC/ED that indicates potential loss of the RCS barrier	<input type="checkbox"/> 6. ANY condition in the opinion of the SM/SEC/ED that indicates loss of the Containment barrier	<input type="checkbox"/> 8. ANY condition in the opinion of the SM/SEC/ED that indicates potential loss of the Containment barrier

Table F-2 Reactor Vessel Water Level Thresholds

RVLIS	No. RCPs	Level
Full Range	None	32%
Dynamic Head	4	46%
	3	35%
	2	26%
	1	22%

APPENDIX G  
Minimum ECCS Flow Rate After Trip



4.0 STEAM

4.4 Instrumentation - Turbine Overspeed Protection and Turbine Trip

ECG 4.4            At least one Turbine Overspeed Protection System and the Turbine Trip ESFAS instrumentation listed in Table 4.4-1 shall be OPERABLE.

APPLICABILITY: According to Table 4.4-1.

**ACTIONS**

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One stop valve or one control valve per high pressure turbine steam line inoperable.</p> <p style="text-align: center;"><u>OR</u></p> <p>One reheat stop valve or one reheat intercept valve per low pressure turbine stem line inoperable.</p>	<p>A.1 Restore the inoperable valve(s) to OPERABLE status.</p>	72 hours
<p>B. Required Action and associated Completion Time of Condition A not met.</p> <p style="text-align: center;"><u>OR</u></p> <p>Required Turbine Overspeed Protection System otherwise inoperable.</p>	<p>B.1 Isolate the turbine from the steam supply.</p>	6 hours
<p>C. One train inoperable.</p>	<p>-----NOTE----- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. -----</p> <p>C.1 Restore train to OPERABLE status.</p> <p style="text-align: center;"><u>OR</u></p> <p>C.2 Be in MODE 3.</p>	24 hours  30 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One channel inoperable.	<p>-----NOTE-----                      The inoperable channel and/or one additional channels may be surveillance tested with one channel in bypass and one channel in trip for up to 12 hours</p> <p>This note is not intended to allow simultaneous testing of coincident channels on a routine basis.                      -----</p>	
	<p>D.1 Place channel in trip.  <u>OR</u>                      D.2. Be in MODE 3.</p>	<p>72 hours  78 hours</p>

SURVEILLANCE REQUIREMENTS

-----NOTE-----  
 Refer to Table 4.4-1 to determine which SRs apply for each Function.  
 -----

SURVEILLANCE	FREQUENCY
SR 4.4.1    Cycle and directly observe the movement of each of the following valves through at least one complete cycle from the running position: 1) Four high pressure turbine stop valves, 2) Four high pressure turbine control valves, 3) Six low pressure turbine reheat stop valves, 4) Six low pressure turbine reheat intercept valves.	6 months
SR 4.4.2    Perform a CHANNEL OPERATIONAL TEST on the Turbine Overspeed Protection Systems.	24 months
SR 4.4.3    Disassemble at least one each of the above valves and perform a visual and surface inspection of valve seats, disks and stems and verify no unacceptable flaws or corrosion.	40 months
SR 4.4.4    Perform CHANNEL CHECK.	12 hours
SR 4.4.5    Perform Channel Operational Test.	184 days
SR 4.4.6    Perform CHANNEL CALIBRATION.	24 months

(continued)



SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 4.4.7	Perform ACTUATION LOGIC TEST.	92 days on a STAGGERED TEST BASIS
SR 4.4.8	Perform MASTER RELAY TEST.	92 days on a STAGGERED TEST BASIS
SR 4.4.9	Perform SLAVE RELAY TEST.	24 months

Table 4.4-1  
Turbine Overspeed Protection and Turbine Trip

Function	Applicable Modes or Other Specified Conditions	Required Channels	Conditions	Surveillance Requirements	Allowable Value	Nominal <sup>(d)</sup> Trip Setpoint
1. Turbine Overspeed Protection System	1,2,3 <sup>(a)(b)</sup>	1 <sup>(f)</sup>	A, B	SR 4.4.1 <sup>(e)</sup> SR 4.4.2 <sup>(e)</sup> SR 4.4.3 <sup>(e)</sup>	NA	NA
2. Turbine Trip						
a. Automatic Actuation Logic and Actuation Relays	1,2 <sup>(c)</sup>	2 trains	C	SR 4.4.7 SR 4.4.8 SR 4.4.9	NA	NA
b. SG Water Level-High High (P-14)	1,2 <sup>(c)</sup>	3 per SG	D	SR 4.4.4 SR 4.4.5 SR 4.4.6	≤ 90.2%	90%
c. Safety Injection	Refer to TS 3.3.2 Function 1 (Safety Injection) for all initiation functions and requirements.					

- (a) Entry into MODES 1, 2, or 3 with the Turbine Overspeed Protection System inoperable is permissible, provided appropriate measures are taken to assure steam is not admitted to the turbine.
- (b) Following removal from service, the turbine may be returned to service under administrative control solely to perform testing required to demonstrate Turbine Overspeed Protection System OPERABILITY. SR 4.4.1 does not need to be performed immediately following a return to service of the turbine. SR 4.4.1 may be performed at an appropriate time during power ascension.
- (c) Except when all MFIVs, MFRVs, and associated bypass valves are closed and de-activated or isolated by a closed manual valve.
- (d) A channel is OPERABLE with an actual Trip Setpoint value outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is re-adjusted to within the established calibration tolerance band of the Nominal Trip Setpoint. A Trip Setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions.
- (e) Entry into an OPERATIONAL MODE or other specified condition may be made without the Surveillance Requirement(s) associated with the Limiting Condition for Operation having been performed within the stated surveillance interval. Also passage through or to OPERATIONAL MODES as required to comply with ACTION requirements is permitted.
- (f) One complete turbine overspeed protection system is required to be operable, which shall consist of the mechanical overspeed trip device or the electrical overspeed trip device and the associated turbine steam inlet control and stop valves.

BASES

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BACKGROUND

This ECG is provided to ensure that the turbine overspeed protection instrumentation, the turbine speed control valves, and the turbine trip function are OPERABLE and will protect the turbine from excessive overspeed and prevent damage to the turbine due to water in the steam lines. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles, which could impact and damage safety related components, equipment or structures (Reference 1).

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APPLICABLE  
SAFETY  
ANALYSES

1. Turbine Overspeed Protection

FSAR Section 3.5, "Missile Protection," (Reference 2) evaluates missiles generated by the main turbine. FSAR Section 3.5.2.2.1 states that "Factory test procedures, redundancy in the control system, and routine testing of the main steam valves and the mechanical emergency overspeed protective system while the unit is carrying load, make generation of missiles by a turbine runaway that might penetrate the turbine casing highly improbable."

FSAR 3.5.2.2.1.2 states that "the probability of generation of an HP turbine missile by speed in excess of the design overspeed, or of an LP turbine missile of any kind, is extremely remote. These are not considered credible events."

Reference 3 states: "Although large steam turbines and their auxiliaries are not safety related systems as defined by NRC regulations, failures that occur in these turbines can produce large, high energy missiles. If such missiles were to strike and damage plant safety related structures, systems, and components, they could relegate them unavailable to perform their safety function."

In revision 1 to this ECG, the frequency for Surveillance Requirement SR 4.4.2 was extended from 18 months to 24 months to be consistent with extended fuel cycles. The basis for this extension is discussed in Reference 4.

Although extending the surveillance for the turbine overspeed protection system from 18 months to 24 months results in a slight increase in the probability of turbine failure, it is still well below the NRC acceptance criterion of 1.0 E-05 per year (Reference 3). In addition, a review of operating, maintenance, and surveillance histories for this system show no time or cycle dependent effects that would prevent the system from performing its design function if surveillance testing were extended from 18 months to 24 months.

In revision 5 to this ECG, the Turbine Trip Function was relocated to this ECG from TS 3.3.2 Table 3.3.2-1, Function 5. The format for this ECG was also revised to be consistent with the standard TS format.

(continued)

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BASES

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APPLICABLE  
SAFETY  
ANALYSES  
(continued)

2. Turbine Trip

The primary function of the Turbine Trip signal is to prevent damage to the turbine due to water in the steam lines. This function is necessary to mitigate the effects of a high water level in the SGs, which could result in carryover of water into the steam lines. The SG high water level is due to excessive feedwater flows.

The turbine trip function is actuated when the level in any SG exceeds the high high setpoint or by an SI signal.

The RTS also initiates a turbine trip signal whenever a reactor trip (P-4) is generated. In the event of SI, the unit is taken off line and the turbine generator must be tripped.

a. Turbine Trip Automatic Actuation Logic and Actuation Relays

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for TS 3.3.2 ESFAS Function 1.b.

b. Turbine Trip - Steam Generator Water Level-High High (P-14)

This signal provides protection against excessive feedwater flow. The ESFAS SG water level instruments provide input to the SG Water Level Control System. Therefore, the actuation logic must be able to withstand both an input failure to the control system (which may then require the protection function actuation) and a single failure in the other channels providing the protection function actuation. Thus, three OPERABLE channels (narrow range instrument span for each generator) are required to satisfy the requirements with a two-out-of-three logic and a median signal selector is provided to prevent control and protection function interactions.

The transmitters (d/p cells) are located inside containment. However, the events that this Function protects against cannot cause a severe environment in containment. Therefore, the Trip Setpoint reflects only steady state instrument uncertainties.

c. Turbine Trip - Safety Injection

Turbine Trip is also initiated by all Functions that initiate SI.

The Turbine Trip Function must be OPERABLE in MODES 1 and 2 except when all MFIVs, MFRVs, and associated bypass valves are closed and de-activated or isolated by a closed manual valve when the MFW System is in operation and the turbine generator may be in operation. In MODES 3, 4, 5, and 6, the turbine generator is not in service and this Function is not required to be OPERABLE.

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(continued)

BASES (continued)

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LCO

When the main turbine is in operation, at least one Turbine Overspeed Protection System and the Turbine Trip ESFAS instrumentation listed in Table 4.4-1 shall be OPERABLE to ensure an overspeed event is stopped before failure occurs that could produce large, high energy missiles and to prevent damage to the turbine due to water in the steam lines.

Table 4.4-1 requires one complete channel (system) OPERABLE for Function 1, Turbine Overspeed Protection System, using either the mechanical overspeed trip or the electrical overspeed trip, and the associated turbine steam inlet control and stop valves.

The mechanical overspeed trip consists of a spring-loaded plunger located in the turbine shaft which extends radially outward when the trip setpoint is reached. When extended, the plunger contacts a lever, which in turn dumps control hydraulic fluid (autostop oil), causing all turbine steam inlet control and stop valves to close.

The electrical overspeed trip is generated by the Digital Electro-Hydraulic control system as redundant overspeed protection. The trip signal is used to energize a solenoid valve, which dumps control hydraulic fluid (autostop oil), causing all turbine steam inlet control and stop valves to close.

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APPLICABILITY

This ECG is applicable whenever the main turbine is in operation in MODES 1, 2, and 3 for the turbine overspeed protection system, and MODES 1 and 2 for the turbine trip system.

Footnote (a) in Table 4.4-1 explains that it is permissible to enter MODES 1, 2, or 3 with the Turbine Overspeed Protection System inoperable provided appropriate measures are taken to assure steam is not admitted to the turbine. This may be accomplished by clearing the main steam isolation valves or turbine stop valves in the closed position (Reference 5). Clearing the main steam isolation valves or turbine stop valves in the closed position is acceptable because this isolates the main turbine from the main steam system. Based on qualitative judgment, the main steam system is the only steam system of sufficient capacity that can overspeed the main turbine. Other steam sources, such as gland sealing steam supplied by the other unit or by auxiliary steam do not provide enough capacity to overspeed the turbine. Electrical power applied to the generator, along with generator anti-motoring protective relay failure, could cause extended motoring of the generator. This may cause damage such as overheating, but would not cause the shaft to overspeed. There are no other sources of motive power capable of overspeeding the main turbine.

(continued)

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BASES

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APPLICABILITY  
(continued)

Footnote (b) in Table 4.4-1 explains that it is permissible to return the turbine to service under administrative control solely to perform testing necessary to demonstrate Turbine Overspeed Protection System OPERABILITY. This provision is consistent with TS LCO 3.0.5. SR 4.4.1 does not need to be performed immediately following a return to service of the turbine. SR 4.4.1 may be performed at an appropriate time during power ascension.

Footnote (c) in Table 4.4-1 explains that the Turbine Trip Function is not required when all MFIVs, MFRVs, and associated bypass valves are closed and de-activated or isolated by a closed manual valve.

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ACTIONS

A.1

Action must be taken to restore an inoperable stop, control, reheat, or intercept valve to OPERABLE status within 72 hours to preclude turbine failure from an overspeed event. The 72 hour Completion Time is reasonable, based on the availability of a second valve in the affected steam supply lead to reduce steam flow to the high or low pressure turbine and the low probability of an overspeed event that would cause turbine failure.

B.1

If the inoperable valve cannot be restored to OPERABLE status within 72 hours as required by Condition A, or the required turbine overspeed protective system is otherwise inoperable, the main turbine must be isolated from the steam supply within 6 hours. With no overspeed protection, an overspeed event could cause turbine failure resulting in production of large, high energy missiles.

If the required turbine overspeed protective system is otherwise inoperable, the turbine must be isolated from the steam supply within 6 hours. With no overspeed protection, an overspeed event could cause turbine failure resulting in production of large, high energy missiles.

The 6 hour allowed Completion time is reasonable, based on operating experience, to isolate steam to the turbine in an orderly manner without challenging unit systems.

(continued)

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BASES

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ACTIONS  
(continued)

C.1 and C.2

Condition C applies to the Automatic Actuation Logic and Actuation Relays for the Turbine Trip Function.

This action addresses the train orientation of the SSPS and the master and slave relays for this Function. If one train is inoperable, 24 hours are allowed to restore the train to OPERABLE status or the unit must be placed in MODE 3 within the following 6 hours. The Completion Time for restoring a train to OPERABLE status is justified in Reference 13. This Function is not required in MODE 3. Placing the unit in MODE 3 removes all requirements for OPERABILITY of the protection channels and actuation functions. In this MODE, the unit does not have analyzed transients or conditions that require the explicit use of the protection functions noted above.

The Required Actions are modified by a Note that allows one train to be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. This allowance is based on the reliability analysis (Reference 11) assumption that 4 hours is the average time required to perform train surveillance.

D.1 and D.2

Condition D applies to the Turbine Trip Actuation signal resulting from Steam Generator Level – High-High (P-14). If one channel is inoperable, 72 hours are allowed to restore one channel to OPERABLE status or to place it in the tripped condition. If placed in the tripped condition, the Function is then in a partial trip condition where one-out-of-two logic will result in actuation. Failure to restore the inoperable channel to OPERABLE status or place in the tripped condition within 72 hours requires the unit to be placed in MODE 3 within the following 6 hours. The allowed Completion time is justified in Reference 13. In MODE 3, this Function is not required to be OPERABLE.

The Required Actions are modified by a Note that allows the inoperable channel and/or one additional channel to be tested with one channel in bypass and one channel in trip for up to 12 hours for surveillance testing. This Function is a two-out-of-three actuation logic and the allowed testing configurations provide flexibility for testing, while assuring that during testing no configuration will cause an inadvertent actuation of the function or keep a valid signal from actuating the function as it was designed (Reference 10). The 12 hours allowed for bypass are justified in Reference 13.

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(continued)

BASES (continued)

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SURVEILLANCE  
REQUIREMENTS

SR 4.4.1

The 6 months valve test frequency required by SR 4.4.1 is based on Diablo Canyon operating experience and the results of an evaluation documented in References 6 and 9. It shows that for Diablo Canyon the probability of turbine missile generation is within the NRC acceptance criteria of Reference 3 for the required turbine valve test interval (Reference 1).

SR 4.4.2

SR 4.4.2 requires the performance of a CHANNEL OPERATIONAL TEST every 24 months. The CHANNEL OPERATIONAL TEST verifies that the turbine overspeed protection trips are OPERABLE and will protect the turbine from excessive overspeed. The frequency of 24 months is acceptable based on a review of system operating, maintenance, and surveillance histories which show no time or refueling cycle length dependent problems associated with the system (Reference 4). SR 4.4.2 performs initial verification of the Turbine Overspeed Protection System OPERABILITY, while SR 4.4.1 performs subsequent verification of the valves' ability to close on loss of EH fluid pressure, as a result of an overspeed trip signal.

SR 4.4.3

At a frequency of 40 months, at least one of each type of the valves listed under SR 4.4.1 is required to be disassembled for inspection to verify that the valves have no unacceptable internal flaws or corrosion that would prevent them from performing their intended safety function.

SR 4.4.4

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

(continued)

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 4.4.4 (continued)

Agreement criteria are established in STP I-1A, based on a combination of the channel instrument uncertainties, including indication and reliability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 4.4.5

SR 4.4.5 is the performance of a COT.

A COT is performed on each required channel to ensure the entire channel will perform the intended Function. Setpoints must be found within the Allowable Values specified in Table 3.3.1-1.

The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology. The setpoint shall be left set consistent with the assumptions of the current unit specific setpoint methodology.

The "as found" and "as left" values must also be recorded and reviewed for consistency with the assumptions of the surveillance interval extension analysis (Ref. 11) when applicable.

The Frequency of 184 days is justified in Reference 14.

SR 4.4.6

SR 4.4.6 is the performance of a CHANNEL CALIBRATION.

A CHANNEL CALIBRATION is performed every 24 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the unit specific setpoint methodology. The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology.

The Frequency of 24 months is based on the assumption of an 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.

(continued)

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BASES

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SURVEILLANCE  
REQUIREMENTS  
(continued)

SR 4.4.7

SR 4.4.7 is the performance of an ACTUATION LOGIC TEST. The SSPS is tested every 92 days on a STAGGERED TEST BASIS, using the semiautomatic tester. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE and that there is an intact voltage signal path to the master relay coils. The Frequency of every 92 days on a STAGGERED TEST BASIS is justified in Reference 14.

SR 4.4.8

SR 4.4.8 is the performance of a MASTER RELAY TEST. The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low voltage continuity check of the slave relay coil. Upon master relay contact operation, a low voltage is injected to the slave relay coil. This voltage is insufficient to pick up the slave relay, but large enough to demonstrate signal path continuity. This test is performed every 92 days on a STAGGERED TEST BASIS. This frequency is justified in Reference 14.

SR 4.4.9

SR 4.4.9 is the performance of a SLAVE RELAY TEST. The SLAVE RELAY TEST is the energizing of the slave relays. Contact operation is verified in one of two ways. Actuation equipment that may be operated in the design mitigation MODE is either allowed to function, or is placed in a condition where the relay contact operation can be verified without operation of the equipment. Actuation equipment that may not be operated in the design mitigation MODE is prevented from operation by the SLAVE RELAY TEST circuit. For this latter case, contact operation is verified by a continuity check of the circuit containing the slave relay. This test is performed every 24 months.

The Frequency is adequate, based on operating experience, considering relay reliability and operating history data (Ref. 12)

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(continued)

BASES (continued)

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REFERENCES

1. CTS Bases 3/4.3.4
  2. FSAR Section 3.5, "Missile Protection"
  3. Letter from C. E. Rossi, USNRC, to J. A. Martin, Westinghouse, dated February 2, 1987
  4. PG&E Letter DCL-98-138, dated October 1, 1998
  5. PSRC Interpretation 86-07 dated December 10, 1986
  6. WCAP-11525, "Probabilistic Evaluation of Reduction in Turbine Valve Test Frequency," dated June 1987
  7. License Amendment Request 95-07, "Relocation of Selected Technical Specifications in Accordance with NRC Final Policy Statement and NUREG-1431, Rev. 1," PG&E letter DCL-95-222 dated October 4, 1995
  8. License Amendment 120 (Unit 1) and 118 (Unit 2), dated February 3, 1998
  9. WCAP – 16054, "Probabilistic Analysis of Reduction in Turbine Valve Test Frequency for Nuclear Plants with Siemens-Westinghouse BB-95/96 Turbines", April 2003
  10. License Amendment 173 (Unit 1) and 175 (Unit 2), dated September 24, 2004
  11. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990
  12. WCAP-13900, "Extension of Slave Relay Surveillance Test intervals," April 1994
  13. WCAP-14333-P-A, Revision 1, "Probablistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times," October 1998
  14. WCAP-15376-P-A, Revision 1, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times," March 2003
  15. License Amendment 198 (Unit 1) and 199 (Unit 2), dated January 8, 2008
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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Loops-MODE 4

LCO 3.4.6 Two loops consisting of any combination of RCS loops and residual heat removal (RHR) loops shall be OPERABLE, and one loop shall be in operation.

-----NOTES-----

1. All reactor coolant pumps (RCPs) and RHR pumps may be removed from operation for  $\leq 1$  hour per 8 hour period provided:
  - a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1; and
  - b. Core outlet temperature is maintained at least 10°F below saturation temperature.
2. No RCP shall be started with any RCS cold leg temperature  $\leq$  Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR unless the pressurizer water level is less than 50%, OR the secondary side water temperature of each steam generator (SG) is  $< 50$  °F above each of the RCS cold leg temperatures.

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required loop inoperable.	A.1 Initiate action to restore a second loop to OPERABLE status.	Immediately
	<u>AND</u> A.2 -----NOTE----- Only required if one RHR loop is OPERABLE. ----- Be in MODE 5.	
		24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Two required loops inoperable.  <u>OR</u>	B.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.	Immediately
No RCS or RHR loop in operation.	<u>AND</u> B.2 Initiate action to restore one loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify one RHR or RCS loop is in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.6.2	Verify SG secondary side water levels are $\geq$ 15% for required RCS loops.	In accordance with the Surveillance Frequency Control Program
SR 3.4.6.3	Verify correct breaker alignment and indicated power are available to the required pump that is not in operation.	In accordance with the Surveillance Frequency Control Program

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Spray and Cooling Systems

LCO 3.6.6 The containment fan cooling unit (CFCU) system and two containment spray trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One containment spray train inoperable.	A.1 Restore containment spray train to OPERABLE status.	72 hours <u>AND</u> 10 days from discovery of failure to meet the LCO  -----NOTE----- For planned maintenance or inspections, the Completion Time is 72 hours. The Completion Times of Required Action A.2 are for unplanned corrective maintenance or inspections.  -----
	<u>OR</u>	
	A.2 Restore containment spray train to OPERABLE status	14 days <u>AND</u> 14 days from discovery of failure to meet the LCO.
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	B.2 Be in MODE 5.	84 hours
C. One required CFCU system inoperable such that a minimum of two CFCUs remain OPERABLE.	C.1 Restore required CFCU system to OPERABLE status.	7 days <u>AND</u> 10 days from discovery of failure to meet the LCO

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One required containment spray train inoperable and one required CFCU system inoperable such that a minimum of two CFCUs remain OPERABLE.	D.1 Restore one required containment spray system to OPERABLE status,	72 hours
	<u>OR</u> D.2 Restore one CFCU system to OPERABLE status such that four CFCUs or three CFCUs, each supplied by a different vital bus, are OPERABLE.	72 hours
E. Required Action and associated Completion Time of Condition C or D not met.	E.1 Be in MODE 3.	6 hours
	<u>AND</u> E.2 Be in MODE 5.	36 hours
F. Two containment spray trains inoperable. <u>OR</u> One containment spray train inoperable and two CFCU systems inoperable such that one or less CFCUs remain OPERABLE. <u>OR</u> One or less CFCUs OPERABLE.	F.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.2	Operate each CFCU for $\geq 15$ minutes.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.3	Verify component cooling water flow rate to each required CFCU is $\geq 1650$ gpm.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.4	Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.6.6.5	Verify each automatic containment spray valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.6	Verify each containment spray pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.7	Verify each CFCU starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.8	Verify each spray nozzle is unobstructed.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.9	Verify each CFCU starts on low speed.	In accordance with the Surveillance Frequency Control Program



3.7 PLANT SYSTEMS

3.7.5 Auxiliary Feedwater (AFW) System

LCO 3.7.5 Three AFW trains shall be OPERABLE.

-----NOTE-----  
Only one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4.  
-----

APPLICABILITY: MODES 1, 2, and 3,  
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

-----NOTE-----  
LCO 3.0.4b is not applicable.  
-----

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One steam supply to turbine driven AFW pump inoperable.	A.1 Restore steam supply to OPERABLE status.	7 days <u>AND</u> 10 days from discovery of failure to meet the LCO
B. One AFW train inoperable in MODE 1, 2 or 3 for reasons other than Condition A.	B.1 Restore AFW train to OPERABLE status.	72 hours <u>AND</u> 10 days from discovery of failure to meet the LCO
C. Required Action and associated Completion Time for Condition A or B not met. <u>OR</u> Two AFW trains inoperable in MODE 1, 2 or 3.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4.	6 hours  18 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>D. Three AFW trains inoperable in MODE 1, 2, or 3.</p>	<p>D.1 -----NOTE----- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status. ----- Initiate action to restore one AFW train to OPERABLE status</p>	<p>Immediately</p>
<p>E. Required AFW train inoperable in MODE 4.</p>	<p>E.1 Initiate action to restore AFW train to OPERABLE status.</p>	<p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify each AFW manual, power operated, and automatic valve in each water flow path, and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.2	<p>-----NOTE-----</p> <p>Not required to be performed for the turbine driven AFW pump until 24 hours after <math>\geq</math> 650 psig in the steam generator.</p> <p>-----</p> <p>Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.</p>	In accordance with the Inservice Test Program.
SR 3.7.5.3	<p>-----NOTE-----</p> <p>Not applicable in MODE 4 when steam generator is relied upon for heat removal.</p> <p>-----</p> <p>Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.</p>	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.4	<p>-----NOTES-----</p> <ol style="list-style-type: none"> <li>1. Not required to be performed for the turbine driven AFW pump until 24 hours after <math>\geq</math> 650 psig in the steam generator.</li> <li>2. Not applicable in MODE 4 when generator is relied upon for heat removal.</li> </ol> <p>-----</p> <p>Verify each AFW pump starts automatically on an actual or simulated actuation signal.</p>	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.5	Not used.	



3.7 PLANT SYSTEMS

3.7.6 Condensate Storage Tank (CST)

LCO 3.7.6 The CST shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,  
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CST inoperable.	A.1 Verify by administrative means OPERABILITY of backup water supply.	4 hours <u>AND</u> Once per 12 hours thereafter
	<u>AND</u> A.2 Restore CST to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 4, without reliance on steam generator for heat removal.	18 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.6.1 Verify the CST water volume is $\geq 200,000$ gallons for Unit 1 and $\geq 166,000$ gallons for Unit 2.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.7 Vital Component Cooling Water (CCW) System

LCO 3.7.7 Two vital CCW loops shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One vital CCW loop inoperable.	A.1 -----NOTE----- Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by CCW. ----- Restore vital CCW loop to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours  36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.7.1 -----NOTE----- Isolation of CCW flow to individual components does not render the CCW System inoperable ----- Verify each CCW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.7.2	Verify each CCW automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.7.3	Verify each CCW pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.10 Control Room Ventilation System (CRVS)

LCO 3.7.10 Two CRVS trains shall be OPERABLE.

-----NOTE-----

The control room envelope (CRE) boundary may be opened intermittently under administrative controls.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6.  
During movement of recently irradiated fuel assemblies.

ACTIONS

-----NOTE-----

ACTIONS apply simultaneously to both units.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CRVS train inoperable for reasons other than Condition B.	A.1 Restore CRVS train to OPERABLE status.	7 days
B. One or more CRVS trains inoperable due to inoperable CRE boundary in MODE 1, 2, 3, or 4.	B.1 Initiate action to implement mitigating actions.	Immediately
	<u>AND</u>	
	B.2 Verify mitigating actions ensure CRE occupant exposures to radiological hazards will not exceed limits, and CRE occupants are protected from smoke and chemical hazards.	24 hours
	<u>AND</u>	
	B.3 Restore CRE boundary to OPERABLE status.	90 days
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	C.2 Be in MODE 5.	36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6, or during movement of recently irradiated fuel assemblies.	D.1.1 Place OPERABLE CRVS train in pressurization mode.  <u>AND</u>	Immediately
	D.1.2 Verify that the OPERABLE CRVS train is capable of being powered by an OPERABLE emergency power source.	Immediately
	<u>OR</u>  D.2 Suspend movement of recently irradiated fuel assemblies.	Immediately

(continued)



ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>E. Two CRVS trains inoperable in MODE 5 OR 6, or during movement of recently irradiated fuel assemblies.</p> <p><u>OR</u></p> <p>One or more CRVS trains inoperable due to an inoperable CRE boundary in MODE 5 or 6, or during movement of recently irradiated fuel assemblies.</p>	E.1 Suspend movement of recently irradiated fuel assemblies.	Immediately
<p>F. Two CRVS trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.</p>	F.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each CRVS train for $\geq 15$ minutes.	In accordance with the Surveillance Frequency Control Program
SR 3.7.10.2	Verify that each CRVS redundant fan is aligned to receive electrical power from a separate OPERABLE vital bus.	In accordance with the Surveillance Frequency Control Program
SR 3.7.10.3	Perform required CRVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with VFTP
SR 3.7.10.4	Verify each CRVS train automatically switches into the pressurization mode of operation on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.10.5	Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.	In accordance with the Control Room Envelope Habitability Program

DIABLO CANYON POWER PLANT OPERATION DATA  
 TABLE R19-1T-1  
 Current Values of Reactor Operating Factors  
 Cycle 17

# 1 UNIT

**DELAYED NEUTRON FRACTION**

	BOL	MOL	EOL
$\bar{\beta}$	0.006317	0.005584	0.005221
$\bar{I}$	0.97	0.97	0.97
$\bar{\beta}_{eff}$	0.006127	0.005416	0.005064

**CONTROL ROD WORTH (BOL, HZP, NO XENON)**

BANK	NO. OF RODS	REACTIVITY WORTH (pcm)
CONTROL A (D, C, B PRESENT)	8	1028
CONTROL B (D, C PRESENT)	4	582
CONTROL C (D PRESENT)	8	638
CONTROL D	9	700
<hr/>		
TOTAL FOR CONTROL	29	2948
TOTAL FOR SHUTDOWN	24	3596
<hr/>		
TOTAL	53	6544

Most reactive single stuck rod at BOL is D-12; Worth = -665 pcm  
 Most reactive single stuck rod at EOL is E-13; Worth = -1236 pcm

**ENRICHMENT**

Zone 13A = 4.398 w/o  
 Zone 16A = 4.198 w/o  
 Zone 17A = 4.693 w/o    Zone 17B = 4.948 w/o  
 Zone 18A = 4.454 w/o    Zone 18B = 4.899 w/o  
 Zone 19A = 4.400 w/o    Zone 19B = 4.800 w/o

**CYCLE ARO POSITION DATA**

ARO Rod Height = 231 Steps for Shutdown and Control Banks

Rod Control Thumbwheel Settings:

S1=128 S2=231 S3=256 S4=359 S5=384 S6=487

MINIMUM REQUIRED BORON CONCENTRATION FOR MODE 6: 2100 PPM

SOURCE: WCAP-17273-P, Rev 0, Pages 2-9, 6-7, A-4, and A-7;

Rod Wear Management Plan, 9/90