Final Submittal
(Blue Paper)

COMBINED RO/SRO WRITTEN EXAM

WITH KAS, ANSWERS, REFERENCES,

1. 006 A2.10 004

Unit 1 is in Mode 1. Chemistry has provided sample results for boron concentration of 1A and 1B Accumulators with the following results:

- 1A Accumulator boron concentration is 2350 ppm.
- 1B Accumulator boron concentration is 2198 ppm.

Which ONE of the following describes the impact of this condition; and the action required in accordance with Technical Specifications and SOP-8.0, Safety Injection System-Accumulators?

A. • Ability to maintain subcriticality after an accident is reduced;

• Drain and fill the 1A Accumulator to lower boron concentration.

B. • Ability to maintain minimum boron precipitation time is reduced;

• Drain and fill the 1A Accumulator to lower boron concentration.

C.✓ • Ability to maintain subcriticality after an accident is reduced;

• Feed and bleed the 1B Accumulator to raise boron concentration.

D. • Ability to maintain minimum boron precipitation time is reduced;

• Feed and bleed the 1B Accumulator to raise boron concentration.

Meets 10 CFR 55.43 (b) 2 requirements for SRO level question.

A. is incorrect; "1A" Accumulator boron concentration is within spec.
SOP-8.0 does provide guidance that fills and/or drains the accumulators in a separate section, but not to raise or lower the Boron concentration.
Step 4.1 fills the accumulators but does not address TS requirements for level and pressure while filling. Also does not address boron concentration unless a 12% level change is made, so sampling is not required due to the fill.
Step 4.2 provides guidance to lower Accumulator level but not to lower boron C.
Appendix 2 is provided expressly to raise boron C. to >2300 PPM

B. incorrect, "1A" Accumulator boron concentration is within spec.

C. Correct.
TS 3.5.4 basis states that boron concentration of the RWST is designed to ensure subcriticality is maintained with uncontrolled cooldown coincident with most reactive rod stuck fully out.
For a large break LOCA analysis, the minimum water volume limit of 321,000 gallons and the lower boron concentration limit of 2300 ppm are used to compute the post LOCA sump boron concentration necessary to assure subcriticality. The large break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core.

Within the same bases the following is found and may cause the applicant to choose the precipitation idea which is what bounds the upper limit.
A water volume of 506,600 gallons and the upper limit on boron concentration of 2500 ppm are used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching from cold leg to hot leg injection is to avoid boron precipitation in the core following the accident.

SOP-8.0, APP 2, FEED AND BLEED OF ACCUMULATOR 1A (1B, 1C) TO RAISE BORON CONCENTRATION >2300 PPM would be used to raise boron concentration

SOP-8.0
CAUTION: Accumulator boron concentration must be maintained between 2200 and 2500 ppm; the intent of this appendix is to raise accumulator boron concentration > 2300 ppm.

D. Incorrect. Basis is incorrect

QUESTIONS REPORT
for 25 SRO Questions

006 A2.10 Emergency Core Cooling
Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Low boron concentration in SIS.

Question Number: 86

Tier 2 Group 1

Importance Rating: 3.9
Technical Reference: TS B3.5.1/3.5.2 and basis, SOP-8, SOP-2.3,
Proposed references to be provided to applicants during examination: None
Learning Objective: OPS52102B01
10 CFR Part 55 Content: 43.2

Comments:
fixed per FJE comments

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

Source :	MODIFIED	Source if Bank:	
Cognitive Level:	LOWER	Difficulty:	
Job Position:	SRO	Plant:	FARLEY
reviewed:	GTO	Previous 2 NRC exams:	NO

2. 009 EA2.01 005

Given the following:

- A small break LOCA has occurred on Unit 1.
- The crew is performing EEP-1.0, Loss of Reactor or Secondary Coolant, step 7 that checks SI Termination Criteria.
- Containment pressure is 3.6 psig.
- Subcooled Margin Monitor value is 14°F in CETC mode.
- RCS pressure is 1100 psig and stable.
- Pressurizer level is 20% and rising slowly.

Which ONE of the following correctly describes the procedure flow path when evaluating step 7, Check SI Termination Criteria, of EEP-1.0, and the reason?

The crew will...

A✓ remain in EEP-1 because RCS subcooling is too low.

B. remain in EEP-1 because RCS pressure is NOT rising.

C. remain in EEP-1 because pressurizer level is too low.

D. go to ESP-1.1, SI Termination, because all SI Termination criteria are met.

Meets 10 CFR 55.43 (b) 5 requirements for SRO level question.

A:Correct. Subcooling does not meet requirements.
Check SUB COOLED MARGIN
MONITOR indication - GREATER
THAN 16F{45F} SUBCOOLED IN
CETC MODE.

B: Incorrect. pressure may be stable or rising.

7.3 Check RCS pressure - STABLE OR RISING.

C: Incorrect. level meets requirement.

7.4 Check pressurizer level - GREATER THAN 13%{43%}.

D: Incorrect. Subcooling must be raised by cooldown or pressure increase
NOTE: For certain break sizes, SI termination criteria may be met due to injection flow exceeding mass flow out of the break. Step 7.5 is not intended to terminate SI when a known LOCA exists.

7.5 IF all SI termination criteria satisfied, THEN go to FNP-1-ESP-1.1, SI TERMINATION.

009 EA2.01
009 small break LOCA
Ability to determine or interpret the following as they apply to a small break LOCA: Actions to be taken, based on RCS temperature and pressure, saturated and superheated

Question Number: 76

Tier 1 Group 1

Importance Rating: SRO 4.8

Technical Reference: EEP-1, Step 7
Proposed references to be provided to applicants during examination: None
Learning Objective: OPS52301B09
10 CFR Part 55 Content: 43.5

Comments:
fixed per FJE comments
009 small break LOCA
Ability to determine or interpret the following as they apply to a small break LOCA: Actions to be taken, based on RCS temperature and pressure, saturated and superheated

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

Source :	NEW	Source if Bank:	
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	SRO	Plant:	FARLEY
reviewed:	GTO	Previous 2 NRC exams:	NO

3. 011 A2.12 001

Given the following:

- A reactor trip has occurred and 1C RCP is the only operating RCP.
- Auxiliary Spray has been placed in service IAW ESP-0.1, Reactor Trip Response, to control and reduce RCS pressure.
- The plant is preparing for a cooldown IAW UOP-2.2, Shutdown of Unit from Hot Standby to Cold Shutdown, with the following parameters:

  - RCS pressure is 2230 psig.
  - RCS temperature is 537°F.
  - Pressurizer level is 23%.

• The crew is at step 5.2 to begin raising pressurizer level to 55%.

Which ONE of the following correctly describes the limit associated with cooling down the pressurizer, and while raising pressurizer level, the method used to prevent thermal stratification in accordance with UOP-2.2?

A. • The temperature difference between the pressurizer steam space and charging water must not exceed 320°F;

• A pressurizer **insurge** must occur during the pressurizer cooldown.

B✓ • The temperature difference between the pressurizer steam space and charging water must not exceed 320°F;

• A pressurizer **outsurge** must occur during the pressurizer cooldown.

C. • The pressurizer cooldown rate must not exceed 100°F in any 1 hour period;

• A pressurizer **insurge** must occur during the pressurizer cooldown.

D. • The pressurizer cooldown rate must not exceed 100°F in any 1 hour period;

• A pressurizer **outsurge** must occur during the pressurizer cooldown.

Meets 10 CFR 55.43 (b) 2 and 5 requirements for SRO level question.

5.4 IF auxiliary spray is in operation, THEN on Data Sheet 1 record the time and the differential temperature between regenerative heat exchanger outlet charging TI-123 and pressurizer vapor space TI-454. Ensure that the differential temperature does not exceed 320°F.

A incorrect; outsurge is required; correct parameters used to determine differential temperature.

B correct;
step 5.2 begins raising the level to 55% and there is a caution prior to and a note after that step to limit delta T and level <63.5% and an outsurge is to be maintained.

5.35.1 Verify **Delta T between pressurizer and charging < 320°F.**
5.35.1.1 Commence recording delta T in FNP-1-STP-1.0, OPERATIONS DAILY AND SHIFT SURVEILLANCE REQUIREMENTS, misc. section every 12 hours during auxiliary spray operation.

P&L of UOP-2.2
3.3.3 Do not exceed a 200°F/hr pressurizer cooldown rate.

3.3.4 The temperature differential between the pressurizer and the RCS must not exceed 320°F. The pressurizer liquid, surge line and loop B hot leg temperatures should be monitored to ensure that a pressurizer outsurge is taking place whenever the pressurizer is being cooled or filled. **This will prevent thermal stratification from taking place.** A pressurizer outsurge is indicated by surge line temperature approximately equal to pressurizer liquid temperature and greater than "B" Hot Leg temperature.

C incorrect; PRZR cooldown rate is 200°F
CAUTION: PRZR cooldown rate must be limited to < 200°F/hr.
P&L
3.3.2 Do not exceed RCS cooldown rate specified in PTLR section 2.0, Operating Limits. The maximum cooldown rate is 100°F in any one hour period.

D incorrect; PRZR cooldown rate limit is 200°F

The following flow path could cause entry into UOP-2.2 with aux spray in service. If 1C RCP is the only running pump after a Rx trip aux spray would be put on service as long as Letdown is in service. . ESP-0.1 will send you to UOP-2.3 which could send you to UOP-2.2. In this scenario the P&Ls of UOP-2.2 would apply and the limits of 320°F and 200°F would be applicable.

011 Pressurizer Level control
A2.12 **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 **Operation of auxiliary spray**

Question Number: 91

Tier 2 Group 2

Importance Rating: 3.3

Technical Reference: UOP-2.2, TRM B 13.4, TS B 3.4.3, STP-35.0
Proposed references to be provided to applicants during examination: None
Learning Objective: OPS52510E06
10 CFR Part 55 Content: 43.5

Comments:

fixed per FJE comments and added some verbiage to stem to clarify where procedurally you are at. Otherwise the question does not make sense.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	B D C C B B D C A A	Scramble Range: A - D	
Source :	NEW		Source if Bank:			
Cognitive Level:	HIGHER		Difficulty:			
Job Position:	SRO		Plant:		FARLEY	
reviewed:	GO		Previous 2 NRC exams:		NO	

4. 016 G2.4.31 001

Unit 1 is at 95% power when the following occurred:

- LT-474, 1A SG NR LVL, was declared INOPERABLE and the channel placed in trip 2 hours ago IAW Tech Specs 3.3.1, Reactor Trip System (RTS) Instrumentation and 3.3.2, Engineered Safety Feature Actuation System (ESFAS) Instrumentation.

Subsequently, the card power supply for LT-475, 1A SG NR LVL, failed.

The following MCB annunciators are in alarm:

- EC1, PROC CAB PWR FAILURE
- JA1, 1A SG LO LVL
- JC1, 1A SG LO-LO LVL ALERT
- JD1, 1A SG HI-HI LVL Alert
- JF1, 1A SG LVL DEV

Which one of the following is the appropriate procedure(s) and actions to be taken for this condition?

A.✓ • Enter EEP-0, Reactor Trip and Safety Injection, and then go to ESP-0.1, Reactor Trip Response.

- Control AFW flow >395 gpm until at least one SG is > 31% NR level.
- Maintain SG levels 31%-65% when conditions permit.

B. • Enter EEP-0, Reactor Trip and Safety Injection, and then go to ESP-0.1, Reactor Trip Response. Implement FRP-H.3, Response to Steam Generator High Level, in conjunction with ESP-0.1.

- Verify BOTH SGFPs are tripped and Main Feedwater and AFW is isolated to ALL SGs.
- When ALL SGs are < 65% NR level, then maintain SG levels 31-65%.

C. • Enter AOP-100, Instrumentation Malfunction.

- Control 1A SG FRV as required to raise 1A SG level to 65%.
- Return SGWLC to Automatic when conditions permit.

D. • Enter AOP-100, Instrumentation Malfunction.

- Control 1A SG FRV as required to lower 1A SG level to 65%.
- Reference T.S. 3.3.1 and 3.3.2, and notify the Shift Manager.

Meets 10 CFR 55.43 (b) 5 requirements for SRO level question.

A: Correct. With 2 LTs on one SG less than 28%, a Rx trip will be generated and an autostart of MDAFWPs is generated. The appropriate path is EEP-0 to ESP-0.1.

No SI signal is generated.

B: Incorrect. With 2 LTs on one SG less than 28%, a Rx trip will be generated and an autostart of MDAFWPs is generated. If a candidate thought that a card failure would cause a high level and the card caused a high level condition which is plausible in that JD1 is in alarm, then this would be an appropriate action to take. Since the failures listed cause a low level alarm and condition, a Rx trip occurs and ESP-0.1 actions taken.

C: Incorrect. LT failure meets entry conditions for AOP-100 and subsequent required actions, however, is the incorrect procedure to enter based upon ERG entry requirement

D: Incorrect. LT failure meets entry conditions for AOP-100 and subsequent required actions, however, is the incorrect procedure to enter based upon ERG entry requirement

With 2 LTs on one SG less than 28%, a Rx trip will be generated and an autostart of MDAFWPs is generated. The appropriate path is EEP-0 to ESP-0.1.

QUESTIONS REPORT
for 25 SRO Questions

016 Non-Nuclear Instrumentation System (NNIS),
G2.4.31 Emergency Procedures / Plan Knowledge of annunciators alarms and indications, and use of the response instructions.

Question Number: 92

Tier 2 Group 2

Importance Rating: 3.4

Technical Reference: EEP-0 and AOP-100
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 43.2

Comments:
changed to a different bank question and modified it due to many technical issues.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	A C A A B A A B B C	Scramble Range: A - D	
Source :	MODIFIED		Source if Bank:		FARLEY	
Cognitive Level:	HIGHER		Difficulty:			
Job Position:	SRO		Plant:		FARLEY	
reviewed:	GO		Previous 2 NRC exams:		NO	

5. 022 G2.1.14 002

Given the following:

- Unit 2 is at 100% power
- The following alarms are received:
  - HA1, PRZR LVL HI RX TRIP ALERT
  - HA2, PRZR LVL DEV HI B/U HTRS ON
  - HB1, PRZR LVL HI
  - DE1, REGEN HX LTDN FLOW DISCH TEMP HI
  - EA2, CHG HDR FLOW HI-LO
- Actual Pressurizer level is 46% and trending DOWN.
- VCT level is 43% and trending UP.
- RCS temperature and pressure are stable.

Which ONE of the following describes the procedure entry required, and a required notification for the event in progress?

A. • Enter AOP-100, Instrument Malfunction;

• Notify the Shift Manager to initiate a 1 hour report IAW EIP-8.0, Non-Emergency Notifications.

B.✓ • Enter AOP-100, Instrumentation Malfunction;

• Initiate a CR and notify the Work Week Coordinator.

C. • Enter AOP-1.0, RCS Leakage;

• Notify the Shift Manager to initiate a 1 hour report IAW EIP-8.0, Non-Emergency Notifications.

D. • Enter AOP-1.0, RCS Leakage;

• Initiate a CR and notify the Work Week Coordinator.

Meets 10 CFR 55.43 (b) 5 requirements for SRO level question.

A. Incorrect. Credible due to correct procedure for a failed LT. This question gives the indications for a failed LT and IAW AOP-100 the WWC would be notified and the crew would initiate a CR. This is not a 1 hour report.
AOP entry is found in EIP-8.0 under 20.0 Additional Corporate Duty Manager Notifications 20.9 Events requiring entry into the EOPs or AOPs

The time required to notify the CDM is not defined in this section but would be done as soon as reasonably possible. The point of the above is that a notification is made IAW EIP-8 and the candidate would have to know that the notification is not a 1 hour notification.

B. Correct. Due to a failed level instrument, AOP-100 would be entered. The following people need to be notified, both the SM and the WWC. The reason for the SM in the other distracters is not correct since the E-plan does not need to be implemented for this condition.

C. Incorrect. Credible due to PZR level trend, and this is not a 1 hour report.
AOP entry is found in EIP-8.0 under 20.0 Additional Corporate Duty Manager Notifications 20.9 Events requiring entry into the EOPs or AOPs

D. Incorrect. Incorrect procedure and incorrect person to notify for an AOP-100 entry, but correct for an AOP-100 entry.

LT 459 has failed high, BU heaters will be on, Charging flow will go to a minimum value, and due to letdown still on service with charging at a minimum, DE1 will be in alarm.
Due to this failure, Pressurizer level is trending DOWN and VCT level is trending up due to the charging flow to a minimum.

AOP-100 actions

8 Notify the Shift Manager.

9 Submit a Condition Report for the failed level channel, and notify the Work Week Coordinator (Maintenance ATL on backshifts) of the Condition Report.

022 G2.1.14
APE 022 Loss of **Reactor coolant makeup** (this affects charging system)
Conduct of Operations: Knowledge of system status criteria which require the notification of **plant personnel.**

Question Number: 77

Tier 1 Group 1

Importance Rating: SRO 3.3

Technical Reference: AOP-100 and above ARPs
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 43.5

Comments:
changed to meet KA for notification requirements and procedural entry.

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

Source :	NEW	Source if Bank:	
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	SRO	Plant:	FARLEY
reviewed:	GTO	Previous 2 NRC exams:	NO

6. 025 AA2.04 004

Given the following:

- Unit 1 is in Mode 5 at 120°F with SG manways open to remove nozzle dams after core reload.
- 'A' Train RHR is in service with 'B' Train RHR in standby.
- The following alarms are received:
  - EC5 - RCS LVL HI-LO
  - BE5 - BOP PANELS ALARM
  - LE2 - (BOP) 1A RHR PUMP RM SUMP LVL HI-HI OR TRBL

The operator observes the following indications:

- RCS level 123'1" and falling.
- The leak is estimated to be 25 gpm.
- Both 1A RHR pump room sump pumps are running.
- 1A RHR pump flow, amps, and discharge pressure are stable.

Which ONE of the following correctly describes the procedure **required** to be entered, what the procedure will accomplish, and how to apply Technical Specification 3.4.13 for the conditions above?

A.✓ • AOP-12.0, Residual Heat Removal System Malfunction, is required to be entered and will identify the location of the leak and **WILL** isolate the leak.

• LCO 3.4.13, RCS Operational LEAKAGE, is **NOT** applicable in Mode 5.

B. • AOP-12.0, Residual Heat Removal System Malfunction, is required to be entered and will identify the location of the leak but will **NOT** isolate the leak.

• LCO 3.4.13, RCS Operational LEAKAGE, is **NOT** applicable in Mode 5.

C. • AOP-1.0, RCS Leakage, is required to be entered and will identify the location of the leak and **WILL** isolate the leak.

• Enter LCO 3.4.13, RCS Operational LEAKAGE, for IDENTIFIED leakage.

D. • AOP-1.0, RCS Leakage, is required to be entered and will identify the location of the leak but will **NOT** isolate the leak.

• Enter LCO 3.4.13, RCS Operational LEAKAGE, for IDENTIFIED leakage.

Meets 10 CFR 55.43 (b) 5 requirements for SRO level question.

A. Correct - IAW EC5, validation of the low level alarm would send the operator to AOP-12. This procedure applies to mode 4,5 6. AOP-12 will identify where the leak is and aid in isolting the leak at step 5.

EC5 setpoint Mid Loop: HI - 123'10" LO - 123'2"

PROBABLE CAUSE

1. Improper RCS level control
2. Improper valve lineup
3. RCS leakage

ACTION

2. IF low level condition exists, THEN monitor RHR pump(s) for evidence of cavitation and if necessary, THEN refer to FNP-1-AOP-12.0, RHR SYSTEM MALFUNCTION.

IAW LE2, the operator could secure the running pump and then go to SOP-7.0, but this is not an option. The leak is outside ctmt due to the running sump pumps which shows the leak to be in the 1A RHR pump room.

The TS for RCS leakage is NOT applicable in modes 5 or 6 but is applicable in modes 1-4.
B. Incorrect- AOP-12 will isolate the leak and this question says AOP-12 will not isolate the leak.

C. Incorrect-
wrong procedure Since AOP-1.0 is not applicable in this mode. AOP-1 is only applicable in Mode 1 - 3

The TS for RCS leakage is NOT applicable in modes 5 or 6 but is applicable in modes 1-4. If in mode 1-4 then this would be correct.

D. incorrect. wrong procedure. AOP-1 only applicable in Mode 1 - 3

025 Loss of the RHRS
AA2.04 Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: **Location and isolability of leaks**

Question Number: 78

Tier 1 Group 1

Importance Rating: SRO 3.6

Technical Reference: AOP-12.0 and AOP-1.0
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 43.5

Comments:
Fixed per FJE comment to include the location of the leak to meet the KA and then the procedural guidance to be entered to meet the SRO portion of the question.
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A D A B C B D B A D Scramble Range: A - D

Source :	BANK	Source if Bank:	
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	SRO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

7. 029 G2.1.33 002

Given the following:

- The plant was at 100% power.
- At 1000, Both Reactor Trip Breakers were declared INOPERABLE.
- SSPS has been determined to be operable.
- The crew immediately initiated a plant shutdown.
- At 1025, a reactor trip signal was generated.
- The Reactor Trip Breakers did NOT open.
- 1A CRDM MG set breaker did NOT open.

At 1030, ALL Reactor Trip and Bypass Breakers were verified open.

UOP-2.3, Shutdown of Unit following Reactor Trip, has been entered.

Which ONE of the following correctly describes the mode the unit is allowed to remain in or must be placed in IAW Technical Specifications and the reason?

A.✓ • The plant can remain in Mode 3 indefinitely;

• since the RTBs are now open and rod control is no longer capable of rod withdrawal.

B. • The plant must proceed to Mode 4, but can remain in Mode 4 indefinitely;

• since the RTBs are now open and rod control is no longer capable of rod withdrawal.

C. • The plant can remain in Mode 3 for up to 6 hours, but must be in Mode 4 in 13 hours and Mode 5 in 37 hours;

• since BOTH RTBs are inoperable and Technical Specification 3.0.3 is in effect.

D. • The plant can remain in Mode 3 for up to 1 hour while trying to repair one RTB, but if one RTB cannot be fixed, the plant must be in Mode 4 in 13 hours and Mode 5 in 37 hours ;

• since BOTH RTBs are inoperable and Technical Specification 3.0.3 is in effect.

Meets 10 CFR 55.43 (b) 2 requirements for SRO level question due to the application of 3.0.3 and knowledge that 3.0.3 applies in this case and how it applies, specifically.

A. Correct, 3.0.3 no longer applies since the RTBs are opened. The (a) With RTBs closed and Rod Control System capable of rod withdrawal for modes 3, 4, 5 show that when the RTBs are open, the spec no longer applies.

B. incorrect - but plausible because 3.0.3 applies until the RTBs are open and has to be evaluated.

C. incorrect. The plant can remain in Mode 3 indefinitely so the plant can remain in mode 3 for 6 hours, however the plant does not have to go to mode 4 and 3.0.3 is no longer in effect due to the RTBs being open.

D is incorrect - The plant can remain in Mode 3 indefinitely so the plant can remain in mode 3 for 2 hours, however the plant does not have to go to mode 4 and 3.0.3 is no longer in effect due to the RTBs being open.

3.3.1

18. Reactor Trip Breakers (j)	1,2	2 trains	R, V
	3 (a) , 4 (a) , 5 (a)	2 trains	C, V

(a) With RTBs closed and Rod Control System capable of rod withdrawal.

V. Two RTS trains inoperable.	V.1 Enter LCO 3.0.3.	mediately

LCO 3.0.3 When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

a. MODE 3 within 7 hours;
b. MODE 4 within 13 hours; and
c. MODE 5 within 37 hours.

EPE 029 **ATWT**
G2.1.33 Conduct of Operations: Ability to recognize indications for system operating parameters which are **entry level conditions for technical specifications**

Question Number: 79

Tier 1 Group 1

Importance Rating: SRO 4.0

Technical Reference: TS 3.3.1 and 3.0.3

Proposed references to be provided to applicants during examination: no reference
Learning Objective:
10 CFR Part 55 Content: 43.2

Comments:
This was rewritten to incorporate an ATWT into the stem and in such a way as to make TS entry a requirement to meet. Just entering Mode 3 on an ATWT event with 3.0.3 in effect would require the plant to be in mode 5, 37 hours after the RTBs were found to be inoperable as long as they can not be opened. Since they are opened in the stem, and as expected per procedure, then the plant is no longer bound to be in mode 4 or 5 and no time limit applies.

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

Source :	NEW	Source if Bank:	
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	SRO	Plant:	FARLEY
reviewed:	GTO	Previous 2 NRC exams:	NO

8. 035 A2.06 003

Given the following:

- A small break LOCA has occurred on Unit 2.
- RCS pressure is 1550 psig and lowering slowly.
- SG pressures are 1000 psig and stable.
- Total AFW flow is 500 gpm.
- SG narrow range levels are 5% and rising slowly.
- PRZR level is off scale low.
- Containment pressure is 3 psig and stable.
- The crew is in EEP-1, Loss of Reactor or Secondary Coolant.

Which ONE of the following describes the correct sequence of actions the crew must use to cool down the RCS in order to place RHR in service?

A. • Cooldown in accordance with EEP-1 until RCS pressure is less than SG pressure;

• Go to ESP-1.2, Post LOCA Cooldown and Depressurization, and cooldown to RHR entry conditions.

B.✓ • NO cooldown will be performed in EEP-1;

• Go to ESP-1.2, Post LOCA Cooldown and Depressurization, and cooldown to RHR entry conditions.

C. • Cooldown in accordance with EEP-1 until RCS pressure is less than SG pressure;

• Go to ESP-1.2, Post LOCA Cooldown and Depressurization, and cooldown to Hot Standby;

• Then go to UOP-2.2, Shutdown of Unit from Hot Standby to Cold Shutdown, and cooldown to RHR entry conditions.

D. • NO cooldown will be performed in EEP-1;

• Go to ESP-1.2, Post LOCA Cooldown and Depressurization, and cooldown to Hot Standby;

• Then go to UOP-2.2, Shutdown of Unit from Hot Standby to Cold Shutdown, and cooldown to RHR entry conditions.

Meets 10 CFR 55.43 (b) 5 requirements for SRO level question.

A. Incorrect. A cooldown is not done in EEP-1. SGWL is maintained and the procedure transitions to either ESP-1.2 or ESP-1.3 for a small break LOCA.

B. Correct. Entry to ESP-1.2 is required and cooldown to RHR entry is directed in ESP-1.2.

C. Incorrect. A cooldown is not done in EEP-1. SGWL is maintained and the procedure transitions to either ESP-1.2 or ESP-1.3 for a small break LOCA. ESP-1.2 does not send you to UOP-2.2. see below discussion.

D. Incorrect. ESP-1.2 cools the plant down to RHR entry conditions, not Hot Standby conditions.

Plausibility-

While UOP-2.2 is not required or directed by EOPs, it **could** be used in part to recover the plant from this point. It might be directed by the TSC staff to enter at a step that would consider getting the plant into a condition in which the appropriate procedure would be used. Since UOP-2.2 is cooldown from Hot Stby and the unit will be on RHR with temp < 200°F, this UOP would not be appropriate at certain steps and sections being used and others being N/A ed. However, The TSC would **Evaluate long term plant status IAW ESP-1.2** and then look for an appropriate procedure to use to clean up the plant and get back to operational status. They could decide many different and/or appropriate procedures depending on the plant conditions.

What is entirely incorrrect with this statement is that ESP-1.2 would not cooldown to Hot Standby, it actually goes all the way down to 200°F before the appropriate procedure would be addressed.

035 **SG system**
A2.06 Ability to (a) predict the impacts of the following mal-functions or operations on the SGS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: **Small break LOCA**

Question Number: 93

Tier 2 Group 2

Importance Rating: 4.6

Technical Reference: EEP-1 and ESP-1.2
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 43.5

Comments:
fixed per FJE comments

QUESTIONS REPORT

for 25 SRO Questions

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

Source :	NEW	Source if Bank:	
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	SRO	Plant:	FARLEY
reviewed:	GTO	Previous 2 NRC exams:	NO

9. 037 G2.4.11 001

The following Unit 1 conditions exist while at 10% power:

The Shift Radiochemist reports the following:
- 1A SG Primary to Secondary Leakage = 148 gpd
- 1B SG Primary to Secondary Leakage = 185 gpd
- 1C SG Primary to Secondary Leakage = 134 gpd

The OATC reports the following:
- Pressurizer PORV-445A is leaking to the PRT at 2.2 gpm.

Which ONE of the following correctly describes the procedure that must be entered and the required action and completion time IAW Technical Specification LCO 3.4.13, RCS Operational LEAKAGE?

A. • Enter AOP-2.0, Steam Generator Tube Leakage.
 • Reduce leakage to within limits within 4 hours.

B.✓ • Enter AOP-2.0, Steam Generator Tube Leakage.
 • Be in Mode 3 within 6 hours.

C. • Enter AOP-1.0, RCS Leakage.
 • Reduce leakage to within limits within 4 hours.

D. • Enter AOP-1.0, RCS Leakage.
 • Be in Mode 3 within 6 hours.

Meets 10 CFR 55.43 (b) 5 requirements for SRO level question.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. All leakage listed in stem is identified leakage. Plausible because applicant may think that PORV leakage is unidentified and at 2.2 gpm, this would exceed the limit.

AOP-2 is correct but the actions are not correct.

B. Correct. 150 gpd is the TS limit.
bases 3.4.13-3
"The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.
AOP-2 is the correct procedure to enter for the above conditions and the TS is a immediately go to mode 3 in 6 hours.

C. Incorrect. All leakage listed in stem is identified leakage. Plausible for same reason as A above and AOP-1 could be entered but not for the reasons given. The action is not correct for a tube leak.

D. Incorrect. Incorrect procedure. Plausible because it is partially correct in that the action is correct.

REFERENCES
1. Technical Specification 3.4.13, Operational Leakage.
2. Technical Specification 3.4.13 Basis.

AOP-2 lesson plan
The guidance is based on anticipating a tube rupture and is more restrictive than the required actions of Technical Specifications. If the steam generator leak rate is determined to be greater than 150 gpd in any steam generator and the unit is in MODE 1 or 2, then the unit is to be placed in MODE 3 per UOP-3.1 and UOP-2.1, within 6 hours.

037 Steam generator tube leakage
G2.4.11 Knowledge of Abnormal operating procedures.

Question Number: original question # 82

Tier 1 Group 2

Importance Rating: 3.6

Technical Reference: AOP-2 and bases 3.4.13-3
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 43.5 B 5

wrote new question for new KA approved by FJE. 10-30-2007(was KA 059G2.4.30)

K/A MATCH ANALYSIS
This question tests the AOP selection and the TS involved. Since it is an immediate be in mode 3 in 6 hours then it is required knowldge and reference is not provided.
Tech Specs can be considered a procedure that is used by the operators. The question tests the knowledge of whether or not a limit is violated. The applicant must have this knowledge in order to have the ability to execute the Tech Specs. Testing the procedural entry requirement makes it SRO-only level.

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

Source :	NEW	Source if Bank:	
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	SRO	Plant:	FARLEY
reviewed:	GTO	Previous 2 NRC exams:	NO

10. 039 A2.01 002

A Large Break LOCA has occurred on Unit 1 with the following conditions:

- B Train is the on service train.
- An LOSP has occurred and B Train emergency power is not available.
- SG pressures are 680 psig and stable.
- Containment pressure rose to 31 psig and is now 8 psig and slowly lowering.
- The crew is at the step to verify SI flow stable in ESP-1.3, Transfer to Cold Leg Recirculation, with the following conditions:
  - Containment Spray is aligned to the RWST.
  - 1A RHR pump is running with proper flow and is aligned to the containment sump.
  - 1A Charging pump has tripped on overcurrent.

Which ONE of the following describes the procedure flow path required and the action that would be taken to reduce SG pressure?

A. • Transition to ECP-1.1, Loss of Emergency Coolant Recirculation;

• Dump steam from the SGs using the steam dumps and maintain the cooldown rate less than 100°F per hour.

B. • Transition to ECP-1.1, Loss of Emergency Coolant Recirculation;

• Dump steam from the SGs using the Atmospheric Relief Valves and maintain the cooldown rate less than 100°F per hour.

C. • Continue in ESP-1.3 and align the CS system for recirculation, then transition back to EEP-1.0, Loss of Reactor or Secondary Coolant;

• Dump steam from the SGs using the steam dumps at the maximum attainable rate.

D✓ • Continue in ESP-1.3 and align the CS system for recirculation, then transition back to EEP-1.0, Loss of Reactor or Secondary Coolant;

• Dump steam from the SGs using the Atmospheric Relief Valves at the maximum attainable rate.

Meets 10 CFR 55.43 (b) 5 requirements for SRO level question.

A. incorrect - The step in ESP-1.3 (7.27) has the crew verify SI flow is stable. If it is, then continue in ESP-1.3 and place CS on recirculation, then go to procedure and step in effect. Since a LBLOCA has occurred, EEP-1 would be the procedure used to get to ESP-1.3. Since the 1A RHR pump is running with proper flow, SI flow will be stable and the transition to ECP-1.1 not required. The way the step is written, as shown below, could confuse the candidate in that they could assume all SI flow is stable when only one is required to be stable.

7.27 Verify SI flow - STABLE.

A TRN
HHSI FLOW
[] FI 943

HHSI
B TRN RECIRC
FLOW
[] FI 940
1A(1B)
RHR HDR
FLOW
[] FI 605A
[] FI 605B

7.27 IF at least one train of flow from the containment sump to the RCS can NOT be established or maintained, THEN go to FNP-1-ECP-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION.

If the applicant decided ECP-1.1 was the proper procedural flow path, then the two choices of using dumps or ARVs is available and the CDR would be correct for this procedure. Dumps are not available since CTMT pressure went to 31 psig and the MSIVs are closed.

B. incorrect - see above

C. incorrect - since the MSIVs would be closed due to the LBLOCA and the LOSP, the dumps would not and could not be used. Our design of the MSIVs do not allow them to be re-opened until a 50 PSID is reached across the valve.

D. Correct - The step in ESP-1.3 (7.27) has the crew verify SI flow is stable. Since it is, then they continue in ESP-1.3 and place CS on recirculation, then go to procedure and step in effect. Since a LBLOCA has occurred, EEP-1 would be the procedure used to get to ESP-1.3 and would be returned to. Then EEP-1 has the crew decide to release pressure from the SGs to decrease the delta P across the tubes. Since the dumps are not available, the ARVs would be used. The max attainable rate is procedurally driven by EEP-1 and makes for a great distracter as well because someone not familiar with the reason for decreasing pressure at this time would be confused and would probably rethink entering EEP-1.

039 A2.01 **Main and Reheat Steam**
Ability to (a) predict the impacts of the following mal-functions or operations on the Main Reheat Steam System; and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: **Flow paths of steam during a LOCA.**

Question Number: 87

Tier 2 Group 1

Importance Rating: 3.2

Technical Reference: E-1 and ESP-1.3
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 43.5

Comments:

This question was rewritten in entirety. One reason is the original submitted question did not meet the KA. I had to rewrite it to a LB LOCA since 035 A2.06 on this exam tests the procedural transition to ESP-1.2. This would have been ideal for this question but was double jeopardy. I did not find a procedural transition question to ECP-1.1 or back to EEP-1 on this exam.
This question also had to deal with steam flows during a LOCA. There is no steam flow in EEP-1 except at step 18 which depressurizes the SGs. Since ESP-1.2 has been taken away by a previous question, the logical step was to go to step 18. To get the procedural transition piece, I had to place enough in the stem for the applicant to analyze and to decide which procedure would be best.

If this is not satisfactory, I will need another suggestion or a replacement KA.

We ran on simulator to get proper pressures and temperatures

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	D A B B A C A D D D	Scramble Range: A - D
Source :	NEW		Source if Bank:		
Cognitive Level:	HIGHER		Difficulty:		
Job Position:	SRO		Plant:	FARLEY	
reviewed:	GTO		Previous 2 NRC exams:	NO	

11. 040 AA2.01 002

Given the following on Unit 1:

- Reactor Trip and Safety Injection have occurred.
- RCS Pressure is 2010 psig and DECREASING.
- Pressurizer level is 22% and rising.
- LOOP A - Tcold on TR0410, RCS COLD LEG, is 510°F and DECREASING.
- Containment Pressure is 16 psig and INCREASING.
- 1A SG Pressure is 520 psig and DECREASING.
- 1B and 1C SG pressures are 840 psig and STABLE.
- Sub Cooled Margin Monitor is reading 130°F.

Which ONE of the following describes the location of the break and the next procedure the crew will perform after transition from EEP-0.0, Reactor Trip or Safety Injection?

A. Downstream of 1A MSIV; ESP-1.1, SI Termination.

B. Downstream of 1A MSIV; EEP-2.0, Faulted Steam Generator Isolation.

C. Upstream of 1A MSIV; ESP-1.1, SI Termination.

D.✓ Upstream of 1A MSIV; EEP-2.0, Faulted Steam Generator Isolation.

Meets 10 CFR 55.43 (b) 5 requirements for SRO level question.

A. incorrect because the break is in the wrong place, as indicated by containment pressure.

B. incorrect because the break is in the wrong place, as indicated by containment pressure.

C. incorrect due to incorrect procedure, but credible because the break is in the correct place and the procedure would be correct for a downstream break.

D. correct. Indications of a Faulted SG upstream of MSIV due to containment pressure. EEP-2 will be addressed because the SG will eventually depressurize.

040 **Steam line rupture**

AA2.01 Ability to determine and interpret the following as they apply to the Steam Line Rupture:
Occurrence and location of a steam line rupture from pressure and flow indications

Question Number: 80

Tier 1 Group 1

Importance Rating: SRO 4.7

Technical Reference: EEP-0.0
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 43.5

Comments:
The parameters were picked based on running this event on the simulator and picking hypothetical values this could happen depending on the reaction time of the crew and a small steam break.

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

Source :	NEW	Source if Bank:	
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	SRO	Plant:	FARLEY
reviewed:	GTO	Previous 2 NRC exams:	NO

12. 061 AA2.03 001

Given the following:

- Unit 1 is at 100% power.
- A fuel shuffle is in progress in the SFP room.
- The following alarm is received:
    - FH1, RMS HI-RAD

The OATC reports the following:
- R-5, Spent Fuel Pool Area Monitor, Red HIGH alarm light is LIT.
- R-25A and R-25B, SFP VENT, radiation monitor
    - Amber ALERT light is LIT.
    - Red HIGH alarm light is **NOT** LIT.

Which ONE of the following describes the current status of Spent Fuel Pool Supply and Exhaust Fans, and the actions that will be required IAW FH1, RMS HI-RAD?

A. • Spent Fuel Pool Supply and Exhaust fans are **running**;

• Implement EIP-9, Emergency Actions, determine if Automated Rapid Dose Assessment (ARDA) has actuated, and verify both trains of PRF running.

B. • Spent Fuel Pool Supply and Exhaust fans are **tripped**;

• Implement EIP-9, Emergency Actions, determine if Automated Rapid Dose Assessment (ARDA) has actuated, and verify both trains of PRF running.

C. • Spent Fuel Pool Supply and Exhaust fans are **tripped**;

• Enter AOP-30.0, Refueling Accident, isolate the Control Room and place Control Room Emergency Filtration system (CREFs) in service.

D✓ • Spent Fuel Pool Supply and Exhaust fans are **running**;

• Enter AOP-30.0, Refueling Accident, isolate the Control Room and place the Control Room Emergency Filtration system (CREFs) in service.

Meets 10 CFR 55.43 (b) 4 and 5 requirements for SRO level question.

A. incorrect. because the actions are incorrect for the event taking place. Plausible because it is an action that would be performed for different conditions. ARDA will activate when R-25A/B go red, and then EIP-9 would be referred to. Both trains of PRF would actuate for R-25A/B high alarm
SFP exhaust goes to the AB exhaust plenum which feeds the plant vent stack. The plant vent stack is monitored by R-14, 29 and 22 which would activate ARDA.
EIP-9.1
ARDA will automatically start when any of the following monitors go into alarm for two consecutive system polls one minute apart on the applicable unit and use the latest 15 minute average monitor value to perform the calculations:

Monitor	Setpoint
Plant Vent Stack R29 (SPING)	
Noble Gas	4.44e-4 µc/ml
Iodine	1.20e-6 µc/ml
Particulate	4.00e-5 µc/ml
Steam Jet air Ejector R15C	.027 R/hr
TDAFW Exhaust R60D	.038 R/hr
Steam Generator A/B/C R60A/B/C	.038 R/hr

ARDA will also automatically start when any of the following monitors go into alarm for two consecutive system polls one minute apart on the applicable unit. The ARDA system will use the plant Vent stack SPING latest 15 minute average monitor value to perform the calculations when these monitors activate the system:

Monitor	Setpoint
Plant Vent stack Monitors	
Gas monitor R 14	13000 (U1) 11571 (U2) CPM
Particulate monitor R 21	1800 (U1) 4280 (U2) CPM
Gas monitor R 22	156 (U1) 143 (U2) CPM

B. incorrect. R-25A or B RED alarm light realigns FHB ventilation NOT the AMBER alert light.

C is incorrect. because FHB fans are running. Credible because R-25A/B RED alarm setpoint has not been reached, so applicant may think FHB ventilation has realigned. AOP-30 directs Control Room isolation and starting CREFs

D. Correct. because FHB fans are running. Credible because R-25A/B RED alarm setpoint has not been reached, AOP-30 directed Control Room isolation and starting CREFs

061 **Area Rad Monitoring alarms**:
AA2.03 Ability to determine and interpret the following as they apply to the Area Radiation Monitoring (ARM) System Alarms: **Setpoints for alert and high alarms**

Question Number: 83

Tier 1 Group 2

Importance Rating: 3.3
Technical Reference: ARP FH5 and FH1 and EIP-9.1; U258400; AOP-30; A181015; OPS-52106D
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 43.5/6 and 55.43(b) (4 and 5)

Comments:

Rewrote to FJE comments.

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

Source :	NEW	Source if Bank:	
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	SRO	Plant:	FARLEY
reviewed:	GTO	Previous 2 NRC exams:	NO

13. 062 G2.1.33 002

Given the following:

- Unit 1 is in Mode 1.
- WD2, 1B INV FAULT, comes into alarm.
- The ROVER reports the following indications on the 1B Inverter panel:
  - The BYPASS SOURCE POWERING LOAD light is LIT.
  - The INVERTER POWERING LOAD light is **NOT** LIT.
  - The battery input breaker has tripped open.
  - The BYPASS SOURCE AVAILABLE light is LIT.

Which ONE of the following statements describes the Technical Specification ACTION statement(s) that **MUST** be entered?

A.✓ • Enter the TS LCO action statement for 3.8.7, Inverters - Operating.

• LCO 3.8.9, Distribution Systems - Operating, entry is **NOT** required.

B. • Enter the TS LCO action statement for 3.8.7, Inverters - Operating.

• Enter the TS LCO action statement for 3.8.9, Distribution Systems - Operating.

C. • LCO 3.8.7, Inverters - Operating, entry is **NOT** required.

• Enter the TS LCO action statement for LCO 3.8.9, Distribution Systems - Operating.

D. • LCO 3.8.7, Inverters - Operating, entry is **NOT** required.

• LCO 3.8.9, Distribution Systems - Operating, entry is **NOT** required.

Meets 10 CFR 55.43 (b) 2 requirements for SRO level question. This is SRO only in that a candidate would have to look at bases to determine whether 3.8.7 and 3.8.9 apply. The definitions of operability are found in bases. .

A. Correct.
Since the 1B Inverter has lost the DC source, the indications above show that the inverter swapped to the bypass source. Since this is true, the 1B vital panel is still energized. TS 3.8.7 applies and Condition A has the SRO look at applicability of 3.8.9. If the inverter did not swap per design and the vital panel was de-energized, then 3.8.9 would be required to be entered also. The reason the inverter is INOPERABLE is b/c of the DC source is required to be the primary source of power to the inverter. Ac is just the backup.
The vital panel is operable since it is powered from an inverter. 3.8.9 says the vital panel can be powered from an inverter that is powered from either AC or DC source.

3.8.7
Operable inverters require the associated vital bus to be powered by the inverter with output voltage and frequency within tolerances, **and** power input to the inverter from a 125 VDC station battery.

A.1 ------------NOTE------------
Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating" with any vital bus deenergized.
----

bases of 3.8.7
With a required inverter inoperable, its associated AC vital bus becomes inoperable until it is re-energized from its Class 1E CVT. For this reason a Note has been included in Condition A requiring the entry into the Conditions and Required Actions of LCO 3.8.9, "Distribution Systems—Operating." This ensures that the vital bus is re-energized within 8 hours. The associated static transfer switch normally provides a bumpless transfer of power to the alternate AC source (Class 1E CVT).

3.8.9
OPERABLE vital bus electrical power distribution subsystems require the associated buses to be energized to their proper voltage from the associated inverter via inverted DC voltage **or** Class 1E constant voltage transformer.

B. Incorrect. 3.8.7 is entered and 3.8.9 is NOT.

C. Incorrect. 3.8.7 is entered and 3.8.9 is NOT.

D. Incorrect. 3.8.7 is entered and 3.8.9 is NOT.

062 **AC electrical distribution** -
G2.1.33 Conduct of operations: Ability to **recognize indications** for system operating parameters which are **entry conditions for technical specifications**

Question Number: 88

Tier 2 Group 1

Importance Rating: 4.0

Technical Reference: TS 3.8.7, 3.8.9 and bases
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 43.2

Comments:
This KA tests the recognition of entry conditions to TSs and is SRO in that bases knowledge has to be understood for the TS referenced and also detailed knowledge of the note in LCO 3.8.7 that sends the SRO to 3.8.9 and why.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	A D C C B D C B C D	Scramble Range: A - D	
Source :	NEW			Source if Bank:		
Cognitive Level:	HIGHER			Difficulty:		
Job Position:	SRO			Plant:	FARLEY	
reviewed:	GTO			Previous 2 NRC exams:	NO	

14. 073 A2.01 001

Given the following:

- Unit 1 is at 100% power.
- The following alarm is received:
    - FH2, RMS CH FAILURE

R-23B, SGBD TO DILUTION, radiation monitor is indicating downscale with all indicating lights extinguished.

Which ONE of the following describes the effect of the failure and the associated ODCM requirement?

A. • FCV-1152, SGBD Heat Exchanger Discharge valve, will close;
 • SGBD releases to the environment can **NOT** continue.

B. • FCV-1152, SGBD Heat Exchanger Discharge valve, will close;
 • SGBD releases can continue provided chemistry analyzes grab samples.

C.✓ • RCV-23B, SGBD Dilution Discharge valve, will close;
 • SGBD releases can continue provided chemistry analyzes grab samples.

D. • RCV-23B, SGBD Dilution Discharge valve, will close;
 • SGBD releases to the environment can **NOT** continue.

Meets 10 CFR 55.43 (b) 2 requirements for SRO level question in the realm of the ODCM requirements when a radiation monitor fails.

A. incorrect. FCV-1152, SGBD Heat Exchanger Discharge valve will not close for this rad monitor. credible due to R-23A will close 1152. The release can be continued.

<u>Purification Outlet Radiation Monitor (RE-23A)</u>
The purification outlet radiation monitor determines the activity level of the fluid entering the surge tank. When the demineralizer train is bypassed, this instrument indicates the activity of the untreated blowdown fluid. If the blowdown is being processed, this instrument will indicate a radioactive breakthrough across the demineralizers. In any case, a high activity signal from RE-23A closes FCV-1152, which stops the blowdown. Indication and a high alarm are on the radiation monitoring system (RMS) panel in the main control room.

~~B. Incorrect - FCV-1152, SGBD Heat~~ Exchanger Discharge valve will not close for this rad monitor. The action listed is correct.

C. correct. since the rad monitor gives a high signal upon a loss of power, the actions for the high alarm will occur. This will close RCV-23B.

FH2 AUTOMATIC ACTION
1. The radiation monitors fail to a "High Radiation" condition on loss of instrument and/or control power that will result in actuation of associated automatic functions. Refer to annunciator FH1 for automatic actions.

ODCM requirements

Instrument	Minimum Channels OPERABLE	ACTION
Gross Radioactivity Monitors Providing Automatic Termination of Release		
a. Liquid Radwaste Effluent Line (RE-18)	1	28
b. Steam Generator Blowdown Effluent Line (RE-23B)	1	29

ODCM page 2-4
ACTION 29 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue, provided grab samples are analyzed for gross radioactivity (beta or gamma) at a MINIMUM DETECTABLE CONCENTRATION no higher than 1 x 10 -7 micro Ci/mL.

Discharge Radiation Monitor (RE-23B)
RE-23B monitors the activity level of the fluid leaving the SGBD. A high activity signal from this instrument closes RCV-23B, which prevents the discharge of high activity fluid to the environment. Indication and a high alarm are located on the RMS panel in the main control room.

D is incorrect since the release can continue.

FH1 guidance

R-23A	SG Blowdown Surge Tank Inlet (AB 130')	Liquid	Scint. (W)	Closes FCV-1152	Perform Step 4.23
R-23B ODCM	SG Blowdown Surge Tank Discharge (AB 130')	Liquid	Scint. (W)	Closes RCV-23B	Perform Step 4.23

073 **Process radiation monitoring**
A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the PRM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Erratic or **failed power supply**

Question Number: 89

Tier 2 Group 1

Importance Rating: 2.9

Technical Reference: ARP-1.6, FH2, FH1 SGBD lesson plan and ODCM page 2-4
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 43.4

Comments:
This was rewritten to incorporate comments from FJE and made the ODCM applicable to make it SRO level

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

Source :	NEW	Source if Bank:	
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	SRO	Plant:	FARLEY
reviewed:	GTO	Previous 2 NRC exams:	NO

SRO QUESTION 076G2.1.2

Distractor “B” is also a correct answer. See examination report 05000348/2007301 and 05000364/2007301 Enclosure 2.

15. 076 G2.1.2 001

Given the following:

- Unit 2 is at 100% power with"A" Train on service.
- At 1200 on 11/7/2007, 2E Service Water pump tripped and "B" Train SW was declared INOPERABLE.

Which ONE of the following describes the Technical Specification REQUIRED ACTION IAW 3.7.8, Service Water System, and the action required to make "B" Train Service Water OPERABLE?

A.✓ • Immediately declare the DG supported by Train "B" Service Water INOPERABLE.

• Place "B" Train of SW on service and align 2C SW pump to auto start for 2E SW pump IAW SOP-24.0, Service Water System.

B. • Immediately declare the DG supported by Train "B" Service Water INOPERABLE.

• Align 2C SW pump to auto start for 2E SW pump IAW AOP-10.0, Loss of Service Water.

C. • Declare the DG supported by Train "B" Service Water INOPERABLE no later than 1600 on 11/7/2007 (4 hours later).

• Align 2C SW pump to auto start for 2E SW pump IAW AOP-10.0, Loss of Service Water.

D. • Declare the DG supported by Train "B" Service Water INOPERABLE no later than 1600 on 11/7/2007 (4 hours later).

• Place B Train of SW on service and align 2C SW pump to auto start for 2E SW pump IAW SOP-24.0, Service Water System.

Meets 10 CFR 55.43 (b) 2 and 5 requirements for SRO level question

A. Correct. This TS immediately entered from 3.7.8 and the DG is declared INOP. Then the 2C SW pump is aligned to auto start for 2E. The trains are swapped to do this. This will allow both trains to be operable.

AOP-10

CAUTION: Based on plant needs, shifting electrical trains in FNP-1-SOP-24.0, SERVICE WATER SYSTEMS, may be delayed. Subsequent shifting of electrical trains is required for train separation.
****

19 IF affected train NOT leaking, THEN evaluate aligning 1C SW pump to affected train using FNP-2-SOP-24.0, SERVICE WATER SYSTEM.

Bases 3.7.8

LCO Two SWS trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming that the worst

case single active failure occurs coincident with the loss of offsite power.
An SWS train is considered OPERABLE during MODES 1, 2, 3, and 4 when:

a. Two pumps are OPERABLE; and
b. The associated piping, valves, and instrumentation and controls required to perform the safety related function are OPERABLE.

Note from A.1

The first Note indicates that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources—Operating," should be entered if an inoperable SWS train results in an inoperable emergency diesel generator.

FSD 181001

3.1.5.1 The Service Water pumps shall be automatically started by a signal from the LOSP or ESS sequencer. The Service Water swing pump shall be automatically started by a signal from the LOSP or ESS sequencer when in service replacing one of the train oriented pumps. (References 6.7.039 and 6.1.009)

SOP-24 P&L

3.3 Service Water pump 1C may be selected for auto-start from the ESS or the LOSP sequencers, instead of an A Train or B Train pump, by using key-interlocked selector switches at the SW local control panels. Normal position of both the A Train and B Train selector switches will be the 1C position and 1C SW pump will not autostart.

B. Incorrect.
This TS is immediately entered from 3.7.8 and the DG is declared INOP.
The second part is in part correct but B Train would be however AOP-10 sends the operator to SOP-24 to select the 2C SWP to autostart and if this was done w/o swapping trains it would be in an incorrect alignment. This has to be done in 72 hours (3 days later) IAW TS 3.7.8. NOT 7 days.

C. incorrect.
4 hours would NOT be allowed to declare inop if DG was OOS.
The second part is in NOT correct. See above.

D. incorrect.
4 hours would NOT be allowed to declare inop if DG was OOS.
Second part of this is correct.

076 **Service Water System**
G2.1.2 Conduct of Operations: Knowledge of **operator responsibilities** during all modes of plant operation.

Question Number: 90

Tier 2 Group 1

Importance Rating: 4.0

Technical Reference: TS 3.7.8, 3.8.1, AOP-10, SOP-24
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 43.2

Comments:
fixed per FJE comments and added how to restore B train to operable status.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	A D D D A C C B C D	Scramble Range: A - D
Source :	NEW		Source if Bank:		
Cognitive Level:	HIGHER		Difficulty:		
Job Position:	SRO		Plant:	FARLEY	
reviewed:	GO		Previous 2 NRC exams:	NO	

16. E03 G2.4.4 006

A spurious SI has occurred on Unit 1 with the following conditions:

- All systems functioned as required.
- ALL but one charging pump has been secured IAW EEP-0, Reactor Trip or Safety Injection.

After establishing normal charging in EEP-0, RCS pressure started to decrease and PRZR level started trending down from 35% and is now 14%.

Which ONE of the following describes the actions and procedural transition the SRO must direct at this point?

A. Reinitiate a manual Safety Injection and transition back to step 1 of EEP-0.

B. Return to the diagnostic steps (13 through 15) of EEP-0, and then transition to EEP-1, Loss of Reactor or Secondary Coolant.

C. Transition to ESP-1.1, SI Termination, step 6, and apply the foldout page of ESP-1.1 to re-establish HHSI flow.

D✓ Re-establish HHSI flow per EEP-0, and then transition to ESP-1.2, Post LOCA Cooldown and Depressurization.

Meets 10 CFR 55.43 (b) 5 requirements for SRO level question

A - Incorrect; If PZR level can not be maintained, the flow path must be reestablished and a transition to ESP-1.2 is warranted. There is no need to manually SI and transition back to step 1 of EEP-0.

B - Incorrect; returning to diagnostics once they have been completed and additional actions taken may seem plausible to get to EEP-1, but this is not allowed per sop-0.8 procedural use guidelines procedure. Also the RNO column of EEP-0 directs the correct actions for this condition.

C - Incorrect; the very next step of EEP-0 sends the user to ESP-1.1. While this would eventually lead the crew to the right place, ESP-1.1 foldout page has the operator go to EEP-1 after the HHSI was re-established. This may seem to be a method to use but is not procedurally correct.

D - Correct; From EEP-0, step 19 and 21, RNO, says that if PZR level or pressure cannot be maintained, the procedural requirement is to go to ESP-1.2.

E03 **LOCA cooldown and depressurization**
G2.4.4 Emergency Procedures / Plan Ability to recognize abnormal indications for system operating parameters which are **entry-level conditions** for **emergency** and abnormal operating procedures.

Question Number: 84

Tier 1 Group 2

Importance Rating: 4.3

Technical Reference: EEP-0 and SOP-0.8
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 43.5

Comments:
This question was written to address the double jeopardy transition issue and not giving away answers by other questions with 035 A2.06. Instead of using EEP-1 to transition to ESP-1.2, EEP-0 is being used to give a similar procedural transition.

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

Source : NEW
Cognitive Level: HIGHER
Job Position: SRO
reviewed: GTO

Source if Bank:
Difficulty:
Plant: FARLEY
Previous 2 NRC exams: NO

17. E04 G2.2.22 001

Given the following:

- Reactor trip and safety injection have occurred on Unit 1.
- ECP-1.2, LOCA Outside Containment, has been completed.
- ESP-1.1, SI Termination, has been completed.
- UOP-2.1, Shutdown of Unit from Minimum Load to Hot Standby, is in progress.
- PRZR level is 75% and increasing slowly.
- 1B RCP is operating.
- 1A and 1C RCPs are secured.
- RCS Tavg is 526°F and increasing slowly.

Which ONE of the following describes the Technical Specification LCO action statement in effect for the given conditions and the basis for the LCO?

A.✓ • 3.4.9, Pressurizer;

• To maintain pressure control to minimize the consequences of potential overpressure transients.

B. • 3.4.9, Pressurizer;

• To maintain RCS subcooling during natural circulation conditions.

C. • 3.4.5 RCS Loops - Mode 3;

• To ensure adequate decay heat removal from the core in the event of an inadvertent control rod withdrawal.

D. • 3.4.5 RCS Loops - Mode 3;

• To ensure adequate decay heat removal from the core and proper boron mixing throughout the RCS.

Meets 10 CFR 55.43 (b) 2 requirements for SRO level question

A is correct.
Since the PZR must be kept below 63.5% in mode 3. Adequate volume for a steam bubble for pressure control is the reason for the 63.5% max pzr level in Mode 1-3.
The pressurizer shall be OPERABLE with:
a. Pressurizer water level < or equal to 63.5% indicated;

Bases for LCO 3.4.9 page B3.4.9-2
The LCO requirement for the pressurizer to be OPERABLE with a water volume = 868 cubic feet, which is equivalent to 63.5% indicated, ensures that a steam bubble exists. Limiting the LCO maximum operating water level preserves the steam space for pressure control.
The LCO has been established **to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients.** Requiring the presence of a steam bubble is also consistent with analytical assumptions.

B. is incorrect. Basis for PZR heaters.

C. is incorrect. 3.4.5 is not in effect since the RTBs are open and there is one RCP running.
Two RCS loops shall be OPERABLE, and either:
a. Two RCS loops shall be in operation when the Rod Control System is capable of rod withdrawal; or
b. One RCS loop shall be in operation when the Rod Control System is not capable of rod withdrawal.

but basis for that spec would be correct in Mode 2 with RTBs closed

D. is incorrect. Incorrect spec (see above), but correct basis for operability requirements with 1 RCP, RTBs open.

E04 **LOCA outside ctmt**

G2.2.22 Equipment Control: Knowledge **Limiting Conditions for Operations** and safety limits

Question Number: 81

Tier 1 Group 1

Importance Rating: SRO 4.1
Technical Reference: TS 3.4.9 & 3.4.5
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 43.2

Comments:
Changed per FJE comments note from es-401-9 below:

Examiner Note: Question meets first half of K/A (LOCA outside of containment) because the event is necessary to provide a credible context for the given plant conditions.

MCS	Time: 1	Points:	1.00	Version:	0 1 2 3 4 5 6 7 8 9		
				Answer:	A D B A A B D B C D	Scramble Range: A - D	
Source :		NEW			Source if Bank:		
Cognitive Level:		HIGHER			Difficulty:		
Job Position:		SRO			Plant:	FARLEY	
reviewed:		GTO			Previous 2 NRC exams:	NO	

18. E15 EA2.1 005

Given the following:

- A Large Break LOCA has occurred on Unit 1.
- RWST level is 14.8 feet and slowly lowering.

The crew is at step 15 in EEP-1.0, Loss of Reactor or Secondary Coolant, to check LHSI flow in progress, when the following containment indications are reported by the OATC:

- FI-958A, CS flow, reads 0 gpm.
- FI-958B, CS flow reads 1850 gpm.
- Ctmt Pressure is 29.5 psig and rising slowly.
- Ctmt Sump Level 8.0 feet and rising slowly.
- Ctmt Radiation Level is 3.6 Rem/Hr on both High Range instruments.

Which ONE of the following describes the **next** action to take for these conditions?

A. Implement FRP-Z.1, Response to High Containment Pressure.

B✓ Implement FRP-Z.2, Response to Containment Flooding.

C. Implement FRP-Z.3, Response to High Containment Radiation.

D. Transition to ESP-1.3, Transfer to Cold Leg Recirculation.

Meets 10 CFR 55.43 (b) 2 requirements for SRO level question

A. Incorrect. 27 psig is ORANGE Path on pressure. Plausible because if flow for CS dropped below 1000 gpm this would be the correct procedure on an orange path. As is it would be a yellow path IF CTMT sump level was less than 7.6 feet.

B. Correct. Since CS flow is > 1000 gpm, and Ctmt sump level > 7.6 feet, this is an orange path on Z.2.

C. Incorrect. This is a yellow path in the same CSF network and the candidate has to know it is a yellow path and it is below or less critical than Ctmt sump level. This would be a correct choice if ctmt sump level is < 7.6 feet.

D. Incorrect. Once ESP-1.3 is entered, no CSF applies. In this case, the crew is holding at step 16 of EEP-1.0 waiting for RWST level to drop below 12.5 feet and then transition to ESP-1.3. Due to the RWST level at 14.8 feet the crew would not wait on the transition but would enter FRP-z.2 until ESP-1.3 was required.

E15 **CTMT flooding**

EA2.1 Ability to determine and interpret the following as they apply to the (Containment Flooding) **Facility conditions and selection of appropriate procedures** during abnormal and emergency operations.

Question Number: 85

Tier 1 Group 2

Importance Rating: 3.2

Technical Reference: CSF-0.5, CSFSTs
Proposed references to be provided to applicants during examination: None
Learning Objective: OPS 52533M02
10 CFR Part 55 Content: 43.5

Comments:
Rewrote the question to specifically address the KA. With the conditions given the candidate has to evaluate the CSFs and determine the appropriate procedure to go to which is tied to the KA.

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

Source :	MODIFIED	Source if Bank:	FARLEY
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	SRO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

19. G2.1.25 004

Unit 1 is in a refueling outage. The following conditions exist:

- Nozzle dams are installed on ALL Steam Generators.
- Both trains of RHR are in service.
- RCS level is 123'3" and stable.
- RCS temperature is 120°F and stable.
- Secondary side of all SGs is > 85% wide range.
- A charging pump is in service; B and C are tagged out.
- The equipment hatch is closed.
- The time to saturation is 35 minutes.

The OATC reports that both RHR pumps have just tripped due to breaker problems.

Using UOP-4.0 Appendix 1, SHUTDOWN SAFETY ASSESSMENT, which one of the following is the correct procedure to go to and proper condition based on the events in progress?

References Provided

A. Go to AOP-42, SHUTDOWN CORE COOLING, under an Orange condition.

B.✓ Go to AOP-42, SHUTDOWN CORE COOLING, under a Red condition.

C. Go to AOP-45, SHUTDOWN INVENTORY, under an Orange condition.

D. Go to AOP-45, SHUTDOWN INVENTORY, under a Red condition.

References: UOP-4.0 Appendix 1, figure 1a page 6 of 16 Version 28

Meets 10 CFR 55.43 (b) requirements for SRO level question in that this is a task only performed by an SRO at FNP.

A. Incorrect- This is a red condition not orange. plausible since time to saturation is a 1 and if the SGs were evaluated improperly, then an orange condition would be selected.

B. Correct- RED
25 min to saturation based on Table B for 100°F. The SG tubes are not filled and vented which gives them a 0 and #5 is not met as well. With the loss of the RHR system, AOP-42 would be entered. Even if a 1 was entered for #5, the condition would still be an unexpected RED.

CORE COOLING		Subtotal	Condition	AOP
1. ≥ 2 SGs Avail with loops filled (Ref step 2.7)	0	0-1	RED	42
2. Cavity level = 142' 1" w/ Upper Internals Removed	0	2-3	ORANGE	42
3. RHR Subsystems Available (0, 1 or 2)	0	4	YELLOW	
4. RCS level = 126' 6"	0	≥5	GREEN	
5. Time to saturation > 30 minutes OR RCS press > 325 psig with at least one RCP available for operation and at least one SG available	1	(GREEN if Defueled)		
Core Cooling Subtotal	1			

C. Incorrect- yellow is correct evaluated condition for Inventory, not orange and then if it was yellow the AOP would not be addressed.

D- Incorrect- yellow is correct. Plausible b/c the HHSI flow path is not identified and not in use and could be considered not available, especially in a shutdown mode. The RCS is intact since the nozzle dams are installed.

INVENTORY		Subtotal	Condition	AOP
1. Refueling Cavity ≥ 23 Feet (142' 1") Above Fuel	0	0	RED	45
2. LHSI Pump/Flowpath Available	0	1	ORANGE	45
3. **HHSI Pump/Flowpath Available**	1	2	YELLOW	
4. **RCS is Intact below the Reactor Vessel Flange**	1	3-4	GREEN	
Inventory Subtotal	2	(GREEN if Defueled)		

G2.1.25
Ability to obtain and **interpret station reference materials** such as graphs, monographs, and **tables** which contain performance data.

Question Number: 94

Tier 3 Group 1

Importance Rating: 3.1

Technical Reference: UOP-4.0
Proposed references to be provided to applicants during examination:
UOP-4.0 Appendix 1, figure 1a page 6 of 16 Version 28

Learning Objective:
10 CFR Part 55 Content:

Comments:
This question was changed out to get a better match to the KA. This is an SRO task during an outage done daily.

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

Source :	MODIFIED	Source if Bank:	FARLEY
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	SRO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

FARLEY NOVEMBER/DECEMBER 2007

FINAL EXAMINATION

NOS. 05000348/2007301 & 05000364/2007301

THIS PAGE CONTAINING

SRO QUESTION G2.1.6 OMITTED

FROM DISTRIBUTION TO THE PUBLIC

21. G2.2.17 008

Given the following:

- During a power reduction to 15% power, an SO reports a flange leak of 5 drops per minute on a Main Steam Line flange upstream of the MSIVs in the MSVR that can not be isolated.
- The FIN team leader wants his team to tighten the flange and stop the leak as TOOLPOUCH MAINTENANCE.

Which ONE of the following correctly states whether the work can be performed as TOOLPOUCH MAINTENANCE per FNP-0-ACP-52.1, Guidelines for Scheduling of On-Line Maintenance, and the reason?

A. May be performed as TOOLPOUCH work because the work will not interrupt the flow of process fluid.

B. May be performed as TOOLPOUCH work because the flange is not part of a safety related system.

C✓ May **NOT** be performed as TOOLPOUCH work because the system pressure and temperature are too high.

D. May **NOT** be performed as TOOLPOUCH work because the work will require entry into a technical specification LCO.

Meets 10 CFR 55.43 (b) requirements for SRO level question due to being a supervisory knowledge of work control procedures. This is also an IR of 2.3 for an RO.

A. Incorrect- This is not acceptable because the flange is part of a high temperature/high pressure system. Even though the work will not interupt flow, the TPM can not be performed.

B. Incorrect- This is not acceptable because the flange is part of a high temperature/high pressure system and because of the high energy of the system.

C. Correct - NOT acceptable because TOOLPOUCH WORK is not allowed on systems where the pressure is greater than 1000 psig or temperature is greater than 200 degrees F.
ACP-52.1
section 2.0
o Tightening of un-isolatable fittings with process fluids <1000 psig or <200 degrees F can be done as tool pouch work. If the system pressure is >1000 psig or temperature is >200 degrees F, then a work order is required with Team Leader or above approval. (AI # 2004202241)

D. Incorrect- it is not acceptable but the reason given is incorrect. TS LCO entry would not be required to tighten the flange. The stem placed the flange upstream of the MSIVs to give TS entry credibility.

TOOLPOUCH WORK is defined as work that can be conducted without detailed written Instructions and without overall plant scheduling.
section 3.0 table
Flanges Tighten to stop leakage (within maximum torque limits)

G2.2.17
Knowledge of the process for managing **maintenance activities** during power operations.

Question Number: 97

Tier 3 Group 2

Importance Rating: 3.5

Technical Reference: ACP-52.1, Appendix 3; NMP-GM-006

Proposed references to be provided to applicants during examination: None
Learning Objective:

10 CFR Part 55 Content: 43.5

Comments:
Fixed per FJE comments and removed ctmt from stem. each distracter has a valid reason why it could or could not be correct for the conditions given.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C C C C C C C C C C Items Not Scrambled

Source :	MODIFIED	Source if Bank:	
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	SRO	Plant:	FARLEY
reviewed:	GTO	Previous 2 NRC exams:	NO

22. G2.2.25 005

Technical Specification 3.4.16, RCS Specific Activity, states:

> The specific activity of the reactor coolant shall not exceed 100/E bar microCi/gm of gross activity. If this limit is not satisfied, the reactor shall be shut down and cooled to 500°F or less within 6 hours after detection.

Which one of the following is the basis for reducing Tavg to less than 500°F if the specific activity of the reactor coolant is not within the limits of LCO 3.4.16?

A. Minimize the release of radioactivity in the event of a LOCA outside containment.

B✓ Prevent venting a ruptured steam generator to the environment.

C. Ensure that the 1-hour dose at the SITE BOUNDARY will not exceed a small fraction of the 10 CFR Part 100 dose guideline limits in the event of a SGTR.

D. Ensure that the 1-hour dose at the SITE BOUNDARY will not exceed a small fraction of the 10 CFR Part 20 dose guideline limits in the event of a LOCA.

Meets 10 CFR 55.43 (b) 2 requirements for SRO level question.

A. Incorrect, LOCA dose is not the bases.

B. Correct- **Minimize the release of radioactivity should a steam generator tube rupture occur.**

APPLICABLE : **The LCO limits on the specific activity of the reactor coolant ensures SAFETY ANALYSES that the resulting doses will not exceed an appropriate fraction of the 10 CFR 100 dose guideline limits following a SGTR accident.** The SGTR safety analysis (Ref. 2) assumes the specific activity of the reactor coolant at 0.5 ∝Ci/gm, a conservatively high letdown flow of 145 gpm, and a bounding reactor coolant steam generator (SG) tube leakage of 1 gpm total for three SGs. The MSLB analysis assumes a steam generator tube leakage of 500 gpd in the faulted loop and 470 gpd in each of the intact loops for a total leakage of 1440 gpd.

Condition B.1

The change within 6 hours to MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and prevents venting the SG to the environment in an SGTR event. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

C. Incorrect, this is a 2 hour dose and the reason for the specific activity limit, not the temperature.

D. Incorrect, LOCA is not the concern and 10 CFR part 20 is not correct.

TS 3.4.16 Basis

G2.2.25
Knowledge of **bases in technical specifications for limiting conditions for operations** and safety limits.

Question Number: 96

Tier 3 Group 2

Importance Rating: 3.7

Technical Reference: TS 3.4.16 Basis
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 43.2

Comments:
Changed out the question since the other question did not meet the KA. Per our telephone discussion this KA can test bases in technical specifications for limiting conditions for operations and bases in technical specifications for limiting conditions for safety limits (which are RO as well as SRO knowledge level questions.)

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

Source :	MODIFIED	Source if Bank:	FARLEY
Cognitive Level:	LOWER	Difficulty:	
Job Position:	SRO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

23. G2.3.8 002

Given the following:

- A Waste Gas release of Waste Gas Decay Tank #5 was started on November 8, 2007 at 1500.

Which ONE of the following describes a condition that would **require** termination of the release once initiated, and the person who is **required** to be notified IAW with SOP-51.1, Waste Gas System Gas Decay Tank Release?

A. • R-29A, PLANT VENT STACK, is declared INOPERABLE.
   • Shift Supervisor

B. • R-29A, PLANT VENT STACK, is declared INOPERABLE.
   • Health Physics Foreman

C. • Waste Gas Decay Tank #4 pressure decreases during the release.
   • Health Physics Foreman

D✓ • Waste Gas Decay Tank #4 pressure decreases during the release.
   • Shift Supervisor

A is incorrect.

R-29A becoming INOPERABLE will not cause the release to be stopped, but R-14 would (see below). R-29A is a backup to R-29B in the event that R-29B fails and would be used to comply with ODCM to take grab samples.
3.2 IF R-14 becomes inoperable while discharging gaseous waste to the vent stack, THEN discharge shall be stopped immediately and the Shift Supervisor notified.

B is incorrect. first not correct (see above). Second part not correct, see C below.

C is incorrect.
first part is correct, see below.

second part NOT correct. The HP foreman is in the approval chain for the release, but is not required to be notified of termination per SOP-51.1

D is correct. If another tank pressure drops, stop the release and Notify the Shift Supervisor.

SOP-51.1 step 4.1.15
Monitor all gas decay tank pressures during the release. Ensure that only the tank which is being released exhibits a pressure decrease and no other tank pressure increases. Stop the release and notify the Shift Supervisor if one of the above occurs.

G2.3.8
Knowledge of the **process for performing a planned gaseous radioactive release.**

Question Number: 98

Tier 3 Group 3

Importance Rating: 3.2

Technical Reference: FNP-1-SOP-51.1, FNP-0-CCP-213.0
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 43.4

Comments:
This fits the KA in that it is the SS job function to know the process for the release and what is required should a particular P&L not be met or if an instrument should fail such as R-14 or R-22.

MCS	Time: 3	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	D D C C A A A D B D	Scramble Range: A - D
Source :	MODIFIED		Source if Bank:		
Cognitive Level:	LOWER		Difficulty:		
Job Position:	SRO		Plant:	FARLEY	
reviewed:	GTO		Previous 2 NRC exams:	NO	

24. G2.4.27 002

Given the following:

- Unit 1 and 2 are operating at 100% Power.
- The Outside System Operator reports a fire in the Liquid H2 storage tank vent stack.

Which ONE of the following describes the INITIAL response the SRO should direct IAW AOP-29.0, Plant Fire; **and** contains only notifications REQUIRED, IAW EIP-8.0, Non-Emergency Notifications?

A. • Assemble the fire brigade and direct the Fire Brigade Leader to extinguish the fire by spraying water directly on the hydrogen vent stack.

• FNP Duty Manager and the Air Products company.

B. • Assemble the fire brigade and direct the Fire Brigade Leader to extinguish the fire by spraying water directly on the hydrogen vent stack.

• Corporate Duty Manager and the Nuclear Regulatory Commission Operations Center (NRCOC).

C✓ • Direct the Outside SO to use SOP-34.0 to extinguish the fire by establishing a helium purge and isolating the leak.

• FNP Duty Manager and the Air Products company.

D. • Direct the Outside SO to use SOP-34.0 to extinguish the fire by establishing a helium purge and isolating the leak.

• Corporate Duty Manager and the Nuclear Regulatory Commission Operations Center (NRCOC).

Meets 10 CFR 55.43 (b) due to EIP notifications are the responsibility of the SRO position.

A. incorrect. The first part is not correct. It is not directed since it would not secure the source of the hydrogen and may not put the fire out.

In the case of a hydrogen vent stack fire, AOP-29 has the following on the symptoms and entry page:
I. IF the fire is in the Liquid H2 storage tank vent stack, THEN go to FNP-0-SOP-34.0, section 4.10, HYDROGEN - OXYGEN SYSTEM. **IF the actions of FNP-0-SOP-34.0 are unsuccessful,THEN the Shift Supervisor should enter FNP-0-AOP- 29.0 and at his/her discretion, assemble the fire brigade to respond.**

SOP-34 P&L
3.11 In the event of a fire at exit of vent stack do not spray water on the vent stack or safety relief valves. Allow fire to continue to burn at top of vent stack until hydrogen source is located, THEN extinguish per section 4.10.

<u>The second part is correct IAW EIP-8, SOP-34 and EIP-13.</u>
Notification of the FNP Duty Manager and the Air Products company is required (if the emergency involves a liquid hydrogen tank, a liquid oxygen tank, or associated use systems.) by EIP-13.0 Fire Emergencies and EIP-8.0 Non-Emergency Notifications.

B. incorrect. first part and second part is NOT correct.
The Nuclear Regulatory Commission Operations Center (NRCOC) is NOT required to be notified as delieniated in EIP-8.0 P&Ls unless the fire is an emergency classification per step 6.2.3 EIP-8.0. Some fires are Emergency classifications, but a vent stack fire is not. Corporate Duty Manager is correct for this event. There is no requirement to file a non-emergency report per EIP-8 (which would require notifying the NRCOC per figure 1) nor is there a requirement to notify the NRCOC in SOP-34.

C. Correct. This is the correct action and notifications

IAW AOP-29, SOP-34 would be used to extinguish the fire per the below steps of SOP-34

4.10.1 Note tank pressure and attempt to quickly determine probable source of H2 leakage by frost on lines from relief valves or purge lines. The most probable source of H2 leakage is PCV-3 which is set at 130 psig and is isolated by valve NSP14V757 (V-27).
4.10.2 Open isolation valve on installed helium bottle, THEN establish helium purge of vent stack.
4.10.3 Isolate or attempt to isolate leaking valve.

EIP-8.0
<u>6.0 Notification for EIP-13, "Fire Emergencies"</u>
NOTE: Notifications are required for all plant fires including small fires and hydrogen vent stack fires. EXCEPTION: Notifications are not required for intentionally set fires at the Fire Training Facility.

6.1 The Shift Manager shall ensure the following are notified:
6.1.5 The FNP Duty Manager.
6.1.8 Air Products, if the emergency involves a liquid hydrogen tank, a liquid oxygen tank, or associated use systems.

6.2 The ED / FNP Duty Manager shall notify:
6.2.2 Corporate Duty Manager.

D. incorrect - first part is correct, notifications is incorrect.

G2.4.27
Knowledge of **fire in the plant procedure.**

Question Number: 99

Tier 3 Group 4

Importance Rating: 3.5

Technical Reference: AOP-29.0 and SOP-34 and EIP-8.0
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 43.5

Comments:

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

Source :	NEW	Source if Bank:	
Cognitive Level:	LOWER	Difficulty:	
Job Position:	SRO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

SRO QUESTION G2.4.44

The correct answer for this question is "C" and not "B."

The distractor analysis for "B" should read as follows:

> B. Incorrect: First part incorrect (see first part of A,) second part incorrect. For initial notifications the Form "Guideline 1" states: (Unaffected Unit(s) Status Not Required for Initial Notifications)

The answer analysis for "C" should read as follows:

> C: Correct: Notification of Protective Action Recommendations is required to be completed for the Initial Notification of a General Emergency. (Not required for any other classification including Site Area Emergency). Announcement with evacuation instructions required per step II. A. 2. of Guideline 2, EIP-9.0.

25. G2.4.44 045

A Site Area Emergency was declared 35 minutes ago. Subsequently, conditions have degraded and a General Emergency classification needs to be declared.

When upgrading to the General Emergency classification, which one of the following contains **ONLY** required actions IAW FNP-0-EIP-9.0, Emergency Actions?

A. • Sounding of the plant emergency alarm.

• Announce needed evacuation instructions to plant personnel.

B.✓ • Sounding of the plant emergency alarm.

• Notify Alabama and Georgia of the status of the unaffected Unit.

C. • Notify Alabama and Georgia of Protective Action Recommendations.

• Announce needed evacuation instructions to plant personnel.

D. • Notify Alabama and Georgia of Protective Action Recommendations.

• Notify Alabama and Georgia of the status of the unaffected Unit.

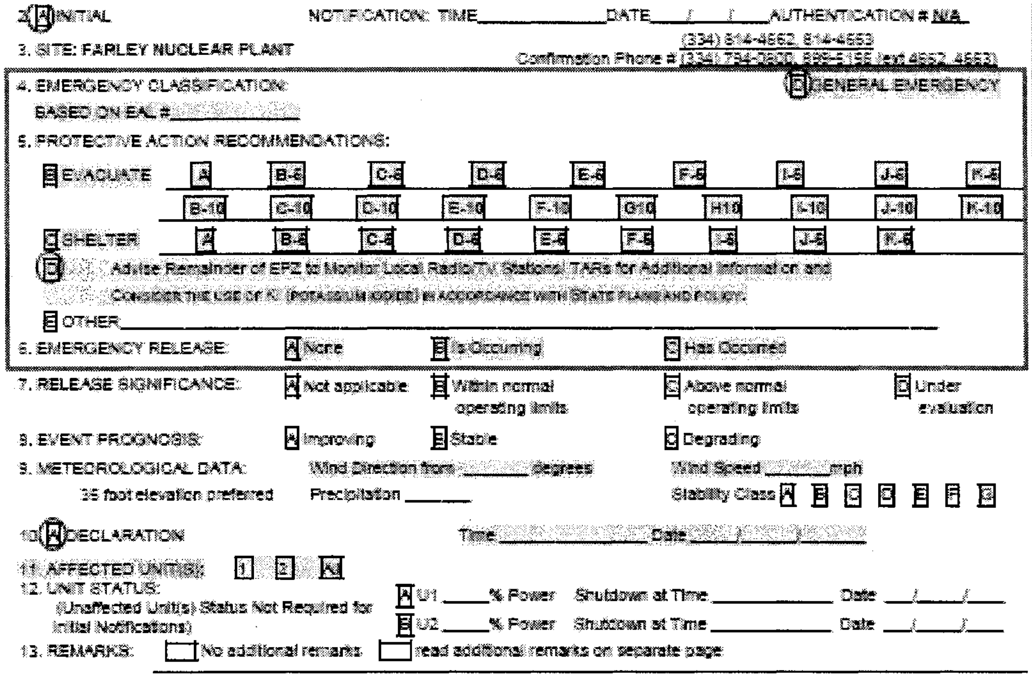
Meets 10 CFR 55.43 (b) requirements for SRO level question since the IR is a 2.1 for an RO and not required knowledge or an action for an RO.

EIP-9.0

A: Incorrect: This action is required by the General Emergency Guideline Procedure only when not already previously performed. The SRO must know that it was required, and was already sounded, for the SAE. Second part correct

II. Emergency Director Actions
NOTE: THE SHIFT MANAGER SHALL PERFORM THE DUTIES OF THE EMERGENCY DIRECTOR UNTIL HIS ARRIVAL AND ASSUMPTION OF DUTIES.
Initials
A. Notify personnel on site
1. **If the Plant Emergency alarm has not already been activated,** then announce over the public address system "All Plant Personnel Report to Designated Assembly Area," **activate the PEA [Plant Emergency alarm]** for 30 seconds and repeat the announcement.
2. **Announce** the classification, and the condition, request setup of the TSC and OSC **and give needed evacuation instructions over plant public address system.**

QUESTIONS REPORT
for 25 SRO Questions



2. A INITIAL NOTIFICATION: TIME ____________ DATE ___/___/___ AUTHENTICATION # N/A

3. SITE: FARLEY NUCLEAR PLANT (334) 814-4662, 814-4663 Confirmation Phone # (334) 794-0800, 899-5156 (ext 4662, 4663)

4. EMERGENCY CLASSIFICATION: D GENERAL EMERGENCY
BASED ON EAL # ____________

5. PROTECTIVE ACTION RECOMMENDATIONS:

B EVACUATE A B-5 C-5 D-5 E-5 F-5 I-5 J-5 K-5
B-10 C-10 D-10 E-10 F-10 G10 H10 I-10 J-10 K-10

C SHELTER A B-5 C-5 D-5 E-5 F-5 I-5 J-5 K-5

D Advise Remainder of EPZ to Monitor Local Radio/TV Stations/ TARs for Additional Information and Consider the use of KI (potassium iodide) in accordance with State plans and policy.

E OTHER ____________

6. EMERGENCY RELEASE: A None B Is Occurring C Has Occurred

7. RELEASE SIGNIFICANCE: A Not applicable B Within normal operating limits C Above normal operating limits D Under evaluation

8. EVENT PROGNOSIS: A Improving B Stable C Degrading

9. METEOROLOGICAL DATA: Wind Direction from ______ degrees Wind Speed ______ mph
35 foot elevation preferred Precipitation ______ Stability Class A B C D E F G

10. A DECLARATION Time ____________ Date ___/___/___

11. AFFECTED UNIT(S): 1 2 A

12. UNIT STATUS: (Unaffected Unit(s) Status Not Required for Initial Notifications)
A U1 _____% Power Shutdown at Time ____________ Date ___/___/___
B U2 _____% Power Shutdown at Time ____________ Date ___/___/___

13. REMARKS: [ ] No additional remarks [ ] read additional remarks on separate page

B. Correct: Notification of Protective Action Recommendations is required to be completed for the Initial Notification of a General Emergency. (Not required for any other classification including Site Area Emergency). Announcement with evacuation instructions required per step II. A. 2. of Guideline 2, EIP-9.0.

C: Incorrect: First part correct, second part incorrect. For initial notifications the Form "Guideline 1" states: (Unaffected Unit(s) Status Not Required for Initial Notifications)

D: Incorrect: First part correct, second part incorrect.
G2.4.44
Knowledge of **emergency plan protective action recommendations**.

Question Number: 100

Tier 3 Group 4

Importance Rating: 4.0

Technical Reference: EIP-9.0

Proposed references to be provided to applicants during examination: NO
Learning Objective:
10 CFR Part 55 Content: 43.5

Comments: Replaced the question with one that does not require reference material and is not a direct lookup. It is more closely related to what an SRO duty is in the emergency plan and required knowledge for an SRO.

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

QUESTIONS REPORT
for 25 SRO Questions

Source :	BANK	Source if Bank:	FARLEY
Cognitive Level:	LOWER	Difficulty:	
Job Position:	SRO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

1. 001 AK2.01 001

Initial conditions (Time = 1000) with Rod control in AUTO:

- Tavg - Tref deviation is 0°F and stable.
- Pressurizer level is 45% and stable.
- Reactor Power is approximately 75% and stable.
- Control Bank D step counters are at 144 steps.

Current conditions (Time = 1002) with no load change in progress:

- Tavg - Tref deviation is approximately +2°F and rising.
- Pressurizer level 46% and slowly rising.
- Pressurizer spray valves have throttled open.
- Reactor Power is approximately 76% and slowly rising.
- Control Bank D step counters are at 150 steps and rising at 8 steps per minute.

Which ONE of the following describes the event in progress; and then the **FIRST** action that must be performed IAW AOP-19.0, Malfunction of Rod Control System?

A. • Inadvertent RCS dilution;

• Trip the reactor and enter EEP-0, Reactor Trip or Safety Injection.

B. • Inadvertent RCS dilution;

• Place the rod control mode selector switch to MANUAL and match Tavg with Tref by inserting rods.

C. • Uncontrolled Continuous Rod Withdrawal;

• Trip the reactor and enter EEP-0, Reactor Trip or Safety Injection.

D✓ • Uncontrolled Continuous Rod Withdrawal;

• Place the rod control mode selector switch to MANUAL and verify that rod motion stops.

A is incorrect; if an inadvertent dilution were taking place, the rods would go in not OUT To trip the reactor at this point would be incorrect.

B is incorrect;
See above for the first part.
Second part is correct IAW AOP-19 for a continuous rod withdrawal.

C is incorrect; is the correct accident, however, the action stated is the RNO if rods do not cease moving once they have been placed in manual IAW AOP-19.

D. is correct for the stated situation.
A CRW is taking place due to temperature shows rods should actually be moving in due to high temperature and the action is to place rods in Manual if they are stepping while in AUTO

001 AK2.01 Continuous Rod Withdrawal
Knowledge of the interrelations between the Continuous Rod Withdrawal and the following:
Rod bank step counters

Question Number: 57

Tier 1 Group 2

Importance Rating: 2.8

Technical Reference: OPS 52201E, AOP-19.0
Proposed references to be provided to applicants during examination: None
Learning Objective: OPS52520S07
10 CFR Part 55 Content: 41.5

Comments:
Fixed per FJE comments 10/4/2007

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	D A B D A B C B A B	Scramble Range: A - D
Source :	BANK		Source if Bank:	WBN BANK	
Cognitive Level:	HIGHER		Difficulty:		
Job Position:	RO		Plant:	FARLEY	
reviewed:	GO		Previous 2 NRC exams:	NO	

2. 001 K5.36 001

During a power DECREASE, the change in power defect will add __(1)__ reactivity to the core.

Assuming the operator does **NOT** borate or dilute, control rod __(2)__ will initially be required to maintain Tavg on program.

A. (1) negative

(2) insertion

B. (1) negative

(2) withdrawal

C.✓ (1) positive

(2) insertion

D. (1) positive

(2) withdrawal

A. incorrect because power defect adds positive reactivity on a power decrease.

B. incorrect because power defect adds positive reactivity on a power decrease.

C. correct. Power defect adds positive reactivity for a negative change in load.

Curve 27 shows for 6000 MWD/MTU power defect will go from -1269 to -658. Positive reactivity will cause Tavg to rise. With no boration, rods must be inserted.

D. incorrect because rods must be inserted to maintain Tavg on program.

001 K5.36 Control Rod Drive Systems
Knowledge of the following operational implications as they apply to the CRDS:
Significance of sign (always minus) of a calculated power defect

Question Number: 29

Tier 2 Group 2

Importance Rating: 3.1

Technical Reference: T&AA, CORE PHYSICS CURVEs pcb-1-vol1-crv27, 34 & 60
Proposed references to be provided to applicants during examination: None
Learning Objective: OPS52510F04
10 CFR Part 55 Content: 41.1

Comments:
Fixed per FJE comments 10/5/2007

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	C D B A A A B D D C	Scramble Range: A - D
Source :	NEW		Source if Bank:		
Cognitive Level:	LOWER		Difficulty:		
Job Position:	RO		Plant:	FARLEY	
reviewed:	GO		Previous 2 NRC exams:	NO	

3. 002 K3.02 001

Which ONE of the following correctly describes the reason that, in the event of a design basis Large Break LOCA, the plant is realigned from Cold Leg Recirculation to Simultaneous Cold and Hot Leg Recirculation?

A.✓ To prevent fuel temperatures from increasing due to boron precipitation at the **TOP** of the core.

B. To prevent fuel temperatures from increasing due to boron precipitation at the **BOTTOM** of the core.

C. To prevent a reduction in Shutdown Margin due to boron precipitation at the **TOP** of the core.

D. To prevent a reduction in Shutdown Margin due to boron precipitation at the **BOTTOM** of the core.

A. Correct. Hot Leg Recirc is aligned to 'backflush' the core due to boron precipitation that occurs due to boil-off.

B. Incorrect. Concern is top of the core, not the bottom which will be covered with water and have continuos flow.

C. Incorrect. Shutdown Margin may be ultimately affected, but core cooling and blockage of flow channels is the concern that Hot Leg Recirc addresses

D. Incorrect. Shutdown Margin may be ultimately affected, but core cooling and blockage of flow channels is the concern that Hot Leg Recirc addresses, and the concern is at the top of the core

Executive volume Rev 2 of ERG guidelines

The operators should continue with the guideline and transfer to cold leg recirculation (ES-1.3) when the RWST level reaches the switchover setpoint. The plant engineering staff may also recommend hot leg recirculation (ES-1.4, TRANSFER TO HOT LEG RECIRCULATION), at a later time, if a boron precipitation concern is possible.

CONCERN(s)
Should the SI system be aligned for hot-leg recirculation in order to prevent boron precipitation in core? **Boron precipitate can plate out on the fuel cladding surface, thereby reducing heat transfer from the fuel to the coolant.**

This requirement is conservative in that for all cases except the design-basis LOCA, the actual rate of boron concentration within the core will be less than that assumed in the FSAR design-basis calculation of the time at which switchover is required from cold- to hot leg recirculation. This is due to the core boiling rate being less than that assumed in the calculation. Additionally, for any LOCA smaller than the design-basis LOCA, the saturation temperature will be higher than that assumed in the calculation, resulting in a boron precipitation limit that is higher than assumed. These factors substantially lengthen the time to the onset of boron precipitation within the core and the time before switchover from cold- to hot-leg recirculation is required.

Conservative analysis has shown that, following a large cold-leg break in the RCS, the boric acid concentration limit established by the NRC (the boric acid solubility limit of 27.53% minus 4% for conservatism) would be exceeded if cold leg recirculation is maintained for an extended period. The analysis considers the increase in boric acid concentration in the reactor vessel during the long-term cooling phase of a LOCA assuming a conservatively small effective vessel volume including only the free volumes of the reactor core and the upper plenum below the bottom of the hot leg nozzles. This assumption conservatively neglects the mixing of boric acid solution with directly connected volumes, such as the reactor vessel lower plenum.

Effects of Break Location
Cold Leg Break
The calculation of boric acid concentration in the reactor vessel considers a cold leg break of the reactor coolant system in which steam is generated in the core from decay heat while the boron associated with the boric acid solution is completely separated from the steam and remains in the effective vessel volume. The cold leg safety injection flow is not effective in counteracting this boiloff from the core since for larger breaks the downcomer level is low and the injection flow is primarily refilling the downcomer as opposed to the core, and no flushing of the core occurs. If the plant is transferred from cold leg to hot leg recirculation prior to the time the boric acid concentration limit is reached in the reactor vessel, the hot leg safety injection flow will dilute the vessel boron concentration by passing relatively dilute boron solution from the hot leg through the vessel to the cold leg break location and will terminate boiloff from the core. This will prevent boron precipitation in the core along with any resultant plateout on the fuel cladding which could reduce heat transfer from the fuel to the reactor coolant.

Lesson text ESP-1.3 OPS-52531G
Approximately 7.5 hours following the loss of coolant accident (LOCA), the cold leg recirculation phase will be terminated and the simultaneous cold and hot leg recirculation phase is initiated. Switching to a hot leg recirculation path will wash out the boron that may have plated out on the fuel rods at the top of the core. Maintaining a cold leg recirculation path provides a normal flow path through the core. If the boron were allowed to build up in the top of the core, it could reduce flow through the core and degrade the heat transfer capability of the fuel. This would also result in a depletion of the boron concentration in the recirculated fluid from the sump.

QUESTIONS REPORT

for 75 RO Questions

002 K3.02 Reactor Coolant System

Knowledge of the effect that a loss or malfunction of the RCS will have on the following:
Fuel

Question Number: 30

Tier 2 Group 2

Importance Rating: 4.2

Technical Reference: OPS-52102B
Lesson text ESP-1.3 OPS-52531G, Executive volume Rev 2 of ERG guidelines

Proposed references to be provided to applicants during examination: None
Learning Objective: OPS52102B03
10 CFR Part 55 Content: 41.7

Comments:
Fixed per FJE comments 10/5/2007

MCS	Time: 1	Points:	1.00	Version:	0 1 2 3 4 5 6 7 8 9		
				Answer:	A C B B A B C B C A	Scramble Range: A - D	
Source :	NEW			Source if Bank:			
Cognitive Level:	LOWER			Difficulty:			
Job Position:	RO			Plant:		FARLEY	
reviewed:	GO			Previous 2 NRC exams:		NO	

4. 003 A1.04 001

Given the following:

- Unit 1 is in Mode 4.
- 1A RCP has just been started.
- A CCW leak is occurring in the tube section of the upper bearing oil cooler of the 1A RCP.

Which ONE of the following correctly describes the effect on the 1A RCP Oil Reservoir level and the **MINIMUM** motor bearing temperature that requires tripping the RCP?

	Oil Reservoir Level	MINIMUM Temperature
A.✓	INCREASES	195° F
B.	INCREASES	302° F
C.	DECREASES	195° F
D.	DECREASES	302° F

A. correct. CCW would leak into the bearing oil reservoir because it is at a higher pressure. The correct temperature to trip the RCP is
HG1
2**. IF any 1A RCP motor bearing temperature exceeds 195°F,** THEN perform the following actions:
a) Trip the reactor, AND go to FNP-1-EEP-0.0, REACTOR TRIP OR SAFETY INJECTION.
b) Stop 1B RCP.
c) Perform the actions required by FNP-1-AOP-4.0, LOSS OF REACTOR COOLANT FLOW.
d) Manually close pressurizer spray valve, PK 444C

B. incorrect due to temperature setpoint.
KK5
PHASE 1 alarm setpoint 275°F MFG max safe operating temp. 302°F

C. incorrect because the reservoir level will be high, not low.

D. incorrect because the reservoir level will be high, not low.

195°F correct per ARP, 302°F is Plausible because per ARP-1.10, KK5, Max safe temperature for RCP Motors is 302°F.

On a complete Loss of CCW Flow to RCP Motor Bearing Oil Coolers, the bearing temperatures will exceed 195°F in approximately 2 minutes.

QUESTIONS REPORT
for 75 RO Questions

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

Question Number: 1

Tier 2 Group 1

Importance Rating: 2.6

Technical Reference: HG1 & HH1 ARP-1.8, AOP-4.1
Proposed references to be provided to applicants during examination: None
Learning Objective: OPS52520I07
10 CFR Part 55 Content: 41.5

Comments:
Fixed per FJE comments 10/5/2007

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

Source :	NEW	Source if Bank:	
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

5. 003 K5.03 001

Unit 1 is at 6% reactor power when 1B RCP trips.

Which ONE of the following describes the **INITIAL** response of Tavg in the 1B Loop and the reason for that response with no operator action?

1B Loop Tavg will ____________________________

A. **INCREASE** because Thot in the unaffected loops INCREASES.

B. **INCREASE** due to the reverse flow of primary coolant in the 1B loop.

C. **DECREASE** because Tcold in the unaffected loops DECREASES.

D✓ **DECREASE** due to the reverse flow of primary coolant in the 1B loop.

A. Incorrect; 1B loop Tave will not increase even though the unaffected loops Tavg will increase.

B. Incorrect; because B Loop Tavg will not increase.
Plausible because the applicant may misunderstand Thot and Tcold values for reverse flow in a loop.

C. Incorrect; 1B loop Tavg will decrease but the unaffected loops Tc will increase.

D. Correct; Tavg will initially decrease due to reverse flow, which occurs when the 1B Loop RCP is tripped.

OPS- 52520D
If the reactor is less than 30% power and there is a loss of coolant flow in one loop (two or more loops if below 10% power), the operator must respond in an efficient manner in order to minimize the effects on primary and secondary systems. In the loop that has lost coolant flow, temperatures will stabilize at approximately the cold leg temperature (TC) of the unaffected loop(s). This will drop the saturation temperature and pressure of the affected loop's steam generator (SG), causing SG level to drop (shrink), and will also reduce the amount of steaming and power output from the affected SG to a minimum.

The loop flow indications observed by the operators would be as follows: For the affected loop, flow would slowly decrease to 0 and then return to approximately 10%; for the unaffected loops, the flow should increase to approximately 105% (each loop). The flow indication in the idle loop occurs as flow stops and then begins again in the reverse direction. Since flow rates in the RCS loops are derived from the differential pressure felt in an elbow in each loop, any flow at all will be indicated, regardless of the direction. The indication observed in the two loops with the running pumps is due simply to the pumps in those loops picking up a small portion of the flow lost in the idle loop.

003 K5.03 Reactor Coolant Pump System (RCPS)
Knowledge of the operational implications of the following concepts as they apply to the RCPS: **Effects of RCP shutdown on T-ave., including the reason for the unreliability of T-ave. in the shutdown loop**

Question Number: 2

Tier 2 Group 1

Importance Rating: 3.1

Technical Reference: OPS 52520D
Proposed references to be provided to applicants during examination: None
Learning Objective: OPS40301A08
10 CFR Part 55 Content: 41.7

Comments:
Fixed per FJE comments 10/5/2007
meets the KA in that this addresses the operational implications of the loss of a RCP on Tavg and the reason the temperature is reading below the other 2 loops (ie., not reliable or different from the operating loops)

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	D B C D D D D B C C	Scramble Range: A - D
Source :	BANK		Source if Bank:	FARLEY	
Cognitive Level:	LOWER		Difficulty:		
Job Position:	RO		Plant:	FARLEY	
reviewed:	GO		Previous 2 NRC exams:	NO	

6. 004 A4.05 001

Given the following:

- Unit 1 is in Mode 5.
- Solid plant operations are in progress.
- HIK-142, RHR TO LTDN HX, has been adjusted to the full open position.
- FK-122, CHG FLOW, is in MANUAL.
- PK-145, LTDN PRESS, is in MANUAL.

The OATC **lowers** the demand on PK-145 . Which ONE of the following describes the effect on PCV-145, Letdown PCV, and RCS pressure?

PCV-145 throttles ___(1)___ , RCS pressure _____(2)_______.

A. (1) OPEN (2) INCREASES

B. (1) CLOSED (2) INCREASES

C✓ (1) OPEN (2) DECREASES

D. (1) CLOSED (2) DECREASES

A. incorrect- PCV-145 WILL open to decrease pressure when the controller is taken to the the lower position. Due to the location of the valve in the system and with HCV-142 fully open when PCV-145 is opened RCS pressure will drop with no change in charging flow.

B. incorrect - The valve will open, not close. Distractor is credible because changing demand does change valve position, and it is easy to associate lowering demand with valve closure

C. correct. Reducing the demand on PK-145 in manual will cause the valve to open, reducing backpressure on the letdown line, therefore reducing RCS pressure upstream.

D. incorrect - The valve will open, not close. Distracter is credible because changing demand does change valve position, and it is easy to associate lowering demand with valve closure

0% demand on the controller = lower system pressure and the valve will open
100% demand on the controller = higher system pressure and the valve will close

QUESTIONS REPORT
for 75 RO Questions

004 A4.05 Chemical and Volume Control System
Ability to manually operate and/or monitor in the control room:
Letdown pressure and temperature control valves

Question Number: 4

Tier 2 Group 1

Importance Rating: 3.6

Technical Reference: CVCS LP OPS-52101F
Proposed references to be provided to applicants during examination: None
Learning Objective: OPS40301F08
10 CFR Part 55 Content: 41.5

Comments:

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

Source :	NEW	Source if Bank:	
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

7. 004 K2.01 001

Which ONE of the following states the power supply to 1A Boric Acid Transfer Pump?

600 Volt MCC ___________ .

A.✓ 1A

B. 1B

C. 1D

D. 1E

A is correct. per the load list for unit 1, page F-92, 1A Boric Acid Transfer pump comes off FAC4, which comes from ED10 and from DF03.

B is incorrect. Plausible because it supplies power to 1B BAT pump. (FBB4 on MCC 1B which comes from EE10 and DG03)

C is incorrect. Plausible because it supplies power to CVCS components: 1A Charging pump Aux Lube Oil pump HDL5. This MCC is on the rad side aux bldg and supplies many rad side AB loads.

D is incorrect. Plausible because this MCC is on the non- rad side aux bldg but supplies some rad side AB loads such as the Boric acid batching tank cond return unit and power to CVCS components: 1B and 1C Charging pump Aux Lube Oil pumps from HEK2 and K3.

004 K2.01 Chemical and Volume Control System
Knowledge of bus power supplies to the following:
Boric acid makeup pumps

Question Number: 3

Tier 2 Group 1

Importance Rating: 2.9

Technical Reference: OPS 52101I, F, & G, FNP-Unit 1 Load List A-506250
Proposed references to be provided to applicants during examination: None
Learning Objective: OPS40301I04
10 CFR Part 55 Content: 41.5

Comments:

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

Source :	NEW	Source if Bank:	
Cognitive Level:	LOWER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GTO	Previous 2 NRC exams:	NO

8. 005 K6.03 001

Which one of the following would prevent the 1A(B) RHR heat exchangers from performing their design function?

A. A loss of air to Heat Exchanger discharge valves HCV-603A and HCV-603B.

B.✓ Closing the component cooling water outlets from the RHR heat exchangers during Mode 3 operation.

C. Closing the manual valve to HCV-142, RHR Discharge to CVCS Letdown Line, during Mode 5 solid plant operation.

D. A loss of air to Heat Exchanger bypass valves FCV-605A and FCV-605B.

Distractor analysis:

A: Incorrect - Loss of air to the HCV-603's, HXs discharge valves, will not prevent the HX from performing their design function, since these valves fail open the HXs are still available.

B: Correct - CCW system must be able to provide flow through the RHR HXs in order for them to perform their design function of removing RCS heat to facilitate cooldown from 350 to 140 within 16 hrs.

C: Incorrect - Manually closing this valve during solid plant operation will result in a loss of L/D and may cause a pressure increase in the RCS, however, this is not the design function of the RHR HX.

D: Incorrect - Loss of air to the FCV-605's, HXs bypass valves, will not prevent the HX from performing their design function, since these valves fail closed the HXs are still available.

REFERENCES:
1. FNP-1-SOP-7.0, RESIDUAL HEAT REMOVAL SYSTEM
2. OPS-52102G-40204A COMPONENT COOLING WATER

Also found in bases

bases 3.4.6

In MODE 4, the primary function of the reactor coolant is the removal of decay heat and the transfer of this heat to either the steam generator (SG) secondary side coolant or the component cooling water via the residual heat removal (RHR) heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

bases 3.4.7

In MODE 5 with the RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer this heat either to the steam generator (SG) secondary side coolant via natural circulation (Ref. 1) or the component cooling water via the residual heat removal (RHR) heat exchangers. While the principal means for decay heat removal is via the RHR System, the SGs via natural circulation (Ref. 1) are specified as a backup means for redundancy

bases 3.7.7

In MODES 1, 2, 3, and 4, the CCW System is a normally operating system, which must be prepared to perform its post accident safety functions, primarily RCS heat removal, which is achieved by cooling the RHR heat exchanger.

bases 3.9.4

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of borated coolant and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchanger(s), where the heat is transferred to the Component Cooling Water System

005 K6.03

Knowledge of the effect of a loss or malfunction on the following will have on the RHRS:
RHR heat exchanger

Question Number: 5

Tier 2 Group 1

Importance Rating: 2.5

Technical Reference:

1. FNP-1-SOP-7.0, RESIDUAL HEAT REMOVAL SYSTEM
2. OPS-52102G-40204A COMPONENT COOLING WATER

Proposed references to be provided to applicants during examination: None
Learning Objective: OPS40301K06
10 CFR Part 55 Content: 41.7

Comments:
This exact question has been used on 3 NRC exams, 2002 surry exam, 2006 summer exam and 2006 FNP exam for the same KA.
It tests the knowledge of the loss of the RHR ht exchanger (ie. cooling function) has on the design basis for the ht exchanger.

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

Source :	BANK	Source if Bank:	FARLEY
Cognitive Level:	LOWER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	YES

9. 006 A4.01 011

Given the following:

- Unit 1 was operating at 100% power.
- B Train is on service with 1B charging pump running.
- An SI/LOSP has just occurred.
- At 22 seconds after the SI/LOSP actuation annunciator EB1, CHG PUMP OVERLOAD TRIP, comes into alarm.
- The operator notices the amber light on the handswitch for the 1C Chg pump.

Which ONE of the following is correct concerning 1B Chg Pump?

1B Chg Pump ______________________

A. must be manually started.

B. will start from the LOSP sequencer.

C✓ will start due to 1C Chg Pump tripping on overload.

D. will remain running throughout the event per design.

A. incorrect; On the LOSP, the 1B chg pump will load shed. 1B charging pump does not need to be manually started since when the 1C CHG pump trips, the 1B pump should automatically start due to an overload trip.

B. Incorrect. Plausible because the sequencer will only start the 1B charging pump if the 1C charging pump breaker is racked out or has tripped on overload. After 22 seconds have passed the sequencer will be at about step 2 of returning equipment to service. Once a step is complete, the sequencer signal is no longer available to start any other component on a previous step. Charging pumps come off step 1 and this will occur about 17 seconds into the event.

C. Correct. The sequencer sequences on the 1C chg pump (unless it is racked out, then it would sequence on the 1B) after about 17 seconds (approx.12 secs for DG to start and tie on, no more than 5 secs for sequencer to start load.). Then, an overload trip of 1C will cause 1B chg pump (when aligned to same train) to auto start. In this case B Train is on service so 1B chg pump is aligned to the B train with 1C Chg pump.

P&L of SOP-2.1
3.30 If the on-service charging pump trips on overload, the off-service charging pump for the particular train which has two operable charging pumps will automatically start.
3.31 If 1A (1C) Charging Pump trips on overload or is racked out, 1B Charging Pump will automatically start upon safety injection or loss of offsite power.

D. Incorrect. because if there was an SI with no LOSP, the SI Sequencer would leave 1B chg pump running, and would not load shed 1B Chg pump and not start 1C.
006 A4.01 Chemical and Volume Control System
Ability to manually operate and/or monitor in the control room:
Pumps

Question Number: 7

Tier 2 Group 1

Importance Rating: 4.1

Technical Reference:
CVCS LP OPS-52101F
FNP-1-ARP-1.5 EB1
SOP-2.1, CHEMICAL AND VOLUME CONTROL SYSTEM PLANT STARTUP AND OPERATION version 84

Proposed references to be provided to applicants during examination: None
Learning Objective: OPS52101F06
~~10 CFR Part 55 Content: 41.7~~

Comments:

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

Source :	BANK	Source if Bank:	FARLEY
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

10. 006 K3.01 001

Given the following:

- A small break LOCA has occurred on Unit 1 and EEP-0, Reactor Trip or Safety Injection, is in progress.
- Sub Cooled Margin Monitor is reading 38°F.
- Containment pressure is 9 psig.
- ALL RCPs are running.
- FI-943, A TRN HHSI FLOW, indicates 0 GPM.

Which ONE of the following describes how the RCPs must be operated IAW EEP-0 and the reason?

A.✓ RCPs must remain operating to provide core cooling.

B. RCPs must remain operating to simplify RCS temperature and pressure control during plant recovery.

C. RCPs must be tripped to prevent damage to the RCPs seals due to the loss of seal injection flow.

D. RCPs must be tripped to prevent excessive loss of RCS water inventory and to keep the core covered.

A. Correct. RCPs may not be tripped because there is no HHSI flow per EEP-0 Fold out page.
RCP Trip Criteria
RCP trip criteria have been developed and incorporated into the ERPs to provide for RCP trip when required for Small Break LOCAs and to minimize the probability of RCP trip when not required. The RCP trip criteria consist of two fundamental parts:
· Successful operation of the SI system
AND
· Subcooling less than 16°F {45°F}

In the ERPs, the RCPs are not tripped unless this two-part criterion is satisfied.
The following summary is provided from the RCP TRIP/RESTART document:
If RCPs continue to operate during a small break LOCA, the forced circulation provides core cooling, but also results in greater loss of coolant inventory due to continued discharge of saturated liquid (rather than steam) from the break. Continuous operation of the RCPs during a LOCA cannot be guaranteed since tripping of the RCPs would occur upon a loss of offsite power or other essential support conditions which could occur at any time. **The reason for purposely tripping the RCPs during an accident (when the RCP trip criterion is met) is to prevent excessive loss of RCS water inventory through a small break which might lead to severe core uncovery if the RCPs were tripped for some reason later in the accident.**

B. incorrect. although RCPs do remain running the reason is not to make it easier to control temperature and pressure with no charging flow.

C. Incorrect. Would be correct if HHSI flow was being indicated.

D. incorrect, RCPs would be tripped if HHSI flow was being indicated. A caution in E-0 says the following: _CAUTION: RCP seal degradation may occur if seal injection flow is not maintained to all RCPs. This could be used to trip the RCPs if Seal injection were lost, however, core cooling is more important at this time due to the loss of HHSI flow.

QUESTIONS REPORT
for 75 RO Questions

006 K3.01 Emergency Core Coolant System
Knowledge of the effect that a loss or malfunction of the ECCS will have on the following:
RCS

Question Number: 6

Tier 2 Group 1

Importance Rating: 4.1

Technical Reference: EEP-0.0 foldout page
Proposed references to be provided to applicants during examination: None
Learning Objective: OPS52530A03
10 CFR Part 55 Content: 41.10

Comments:
This tests the ability to determine what to do with RCPs running and a loss of subcooling with a SB LOCA in EEP-0, and the effects on the RCS of that decision in relation to a loss of HHSI , which meets the KA above.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	A A B B B C A D D A	Scramble Range: A - D	
Source :	NEW			Source if Bank:		
Cognitive Level:	HIGHER			Difficulty:		
Job Position:	RO			Plant:	FARLEY	
reviewed:	GO			Previous 2 NRC exams:	NO	

11. 007 K5.02 001

The crew is forming a pressurizer steam space (drawing a bubble) per UOP-1.1, Startup of Unit from Cold Shutdown to Hot Standby. The vacuum refill procedure will **NOT** be performed.

- Unit 1 is in Mode 5 maintaining 325-375 psig.
- 1B RCP is running.
- A Train RHR is on service with low pressure letdown aligned.
- RCS is in solid plant pressure control with pressurizer temperature at 178°F.
- All PRZR heaters have been energized.

Which ONE of the following correctly describes the condition that will indicate when the pressurizer is at saturation conditions (ie. a bubble is ready to be formed) IAW UOP-1.1; and the effect on PRT level during this evolution?

A. • Letdown flow decreases;

• PRT level will remain constant.

B✓ • RCS Pressure will increase;

• PRT level will remain constant.

C. • RCS Pressure will increase;

• PRT level will rise.

D. • Letdown flow decreases;

• PRT level will rise.

A. Incorrect. Plausible, because RCS pressure will start to rise and letdown flow will increase as pressure starts to rise. The candidate may not know what to expect from the letdown flow as they may not know the position of PCV-145, LETDOWN PCV, FCV-122, CHG FLOW REG, and HCV-142, RHR TO LETDOWN LINE.

The PRT parameters will remain constant.

B. Correct. IAW step 5.10, letdown flow will increase as RCS pressure increases.
The PRT parameters will remain constant since the liquid from the pzr is diverted to the RHTs.

Uop-1.1 step 5.10
WHEN pressurizer temperature increases to the saturation temperature for 375 psig (approximately 442°F) as indicated by increasing RCS pressure or letdown flow, THEN establish a steam space in the pressurizer as follows:

UOP-1.1 shows that the liquid from the pressurizer will go to the RHTs. There will be no level increase or liquid that will go to the PRT.

5.10.5 WHEN VCT level increases to 81%, THEN verify VCT HI LVL /DIVERT VLV Q1E21LCV115A in the fully diverted position.

C Incorrect. First part is correct.
second part is NOT correct. see above.

D. Incorrect. both first and second part are not correct.
007 K5.02 Pressurizer Relief Tank
Knowledge of the operational implications of the following concepts as the apply to PRTS:
Method of forming a steam bubble in the PZR

Question Number: 8

Tier 2 Group 1

Importance Rating: 3.1

Technical Reference: UOP 1.1, RHR FSD A-181002
Proposed references to be provided to applicants during examination: None
Learning Objective: OPS40301F06
10 CFR Part 55 Content: 41.10

Comments: In order to meet the KA, the PRT had to be used to in some fashion. Since the PZR liquid is directed to the RHTs as the level is being decreased, the PRT level, temp and pressure will be unaffected. To meet the KA; the method of forming the bubble is addressed by the indications that will be available when the steam space just begins to be formed.

Operational implications of the PRT are none so level, pressure and temp will remain constant.

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

Source :	NEW	Source if Bank:	
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

12. 008 A2.08 001

Given the following:

- Unit 1 is at 100% power.
- The temperature input to TCV-3083 (also called TCV-144), LTDN HX CCW DISCH TCV, fails low.
- The OATC reports that TI-144, CCW LTDN HX, temperature indicator, is reading at the bottom of the scale (50°F) due to the temperature input failure.

The consequences of the failure is a small RCS ____1____ ; and the action required by the OATC would be to use MCB TK-144, LTDN HX OUTLET TEMP, in Manual Control and ________2________ CCW flow.

A. 1. dilution
   2. increase

B. 1. dilution
   2. decrease

C.✓ 1. boration
   2. increase

D. 1. boration
   2. decrease

From SOP-23.0:
CAUTION: CCW temperature should be maintained as stable as possible due to the effects on reactivity due to changes in letdown temperature. Also, changing CCW temperature could affect RCP oil levels which could cause level annunciators to come in.

From SOP-2.1 rev 84
CAUTION: Changes in letdown temperature can have a significant effect on reactor power. Care should be taken to closely coordinate changes in CCW flow between personnel at LTDN HX CCW TEMP CONT, Q1P17TV3083, and Control Room personnel at LTDN HX OUTLET TEMP TK144.

Letdown Temperature controller, TK-144, failed low. The controller senses a lower temperature and sends a signal to the CCW valve to close down to provide less cooling to raise the temperature of Letdown. When Letdown temperature goes up, the demineralizers have less affinity for boron, and some of the boron in the demineralizers is released. **This is a boration effect.**

ARP'S DF5 & DF1 both direct taking manual control of TK-144 when needed to control temperature.

A. incorrect; a dilution will not occur, increasing is correct.

B. incorrect; a dilution will not occur. Decreasing CCW flow is **not** correct even though it would be done using TK-144 IN MANUAL. It is plausible because the TCV failure does cause a temperature change, just opposite from the change that will cause a dilution.

C. Correct. a boration will occur and increasing CCW flow to the Ht exchanger using TK-144 IN MANUAL is the correct answer. As letdown water heats up, boron will be released in ion exchangers, resulting in a small boration.

D. incorrect; a boration will occur, Decreasing CCW flow is **not** correct.

008 A2.08 Component Cooling Water System
Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Effects of shutting (automatically or otherwise) the isolation valves of the letdown cooler

Question Number: 9

Tier 2 Group 1

Importance Rating: 2.5

Technical Reference: CVCS LP, SOP-2.1, Sec 4.18 & 4.19 cautions
Proposed references to be provided to applicants during examination: None
Learning Objective: OPS52101F02
10 CFR Part 55 Content: 41.7

Comments:
This question meets the KA in that it is a failure closed of the CCW isolation valve to a cooler and has operational procedures and actions to combat the event.

MCS Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
		Answer:	C B A B C A D C A D	Scramble Range: A - D	
Source :	MODIFIED		Source if Bank:	FARLEY	
Cognitive Level:	HIGHER		Difficulty:		
Job Position:	RO		Plant:	FARLEY	
reviewed:	GO		Previous 2 NRC exams:	NO	

13. 008 AK2.01 005

Given the following plant conditions:

- A reactor trip and safety injection have occurred.
- RCS pressure is 1050 psig and lowering.
- Tavg is 550°F and lowering.
- Pressurizer level is 65% and rising rapidly.
- Containment pressure is 2 psig and rising.

Which ONE of the following describes the cause of this event?

A. Letdown line break.

B. Small Break LOCA on an RCS cold leg.

C.✓ Stuck open pressurizer PORV.

D. Stuck open pressurizer spray valve.

A & B. Incorrect. PZR level would be lowering or off-scale low if either of these events occurred.
C. Correct. A vapor space LOCA is occurring, due to RCS pressure lowering and Containment pressure rising with PZR level rising.
D. Incorrect because spray valve failure would not result in containment pressure rising.

008 AK2.01 Pressurizer Vapor Space Accident -
Knowledge of the interrelations between the Pressurizer Vapor Space Accident and the following:
Valves

Question Number: 39

Tier 1 Group 1

Importance Rating: RO 2.7

Technical Reference: AOP-100, HC1 ARP-1.8 HE3 and 4 and 5
Proposed references to be provided to applicants during examination: None
Learning Objective: OPS52201H12
10 CFR Part 55 Content:

Comments:
This was originally written as a spray valve failure. This is not a vapor space accident and did not meet the KA. Rewritten to meet the KA for vapor space accident and a valve issue. Subsequent comments from FJE made question unsat, swapped for question on 2004 Robinson NRC exam and also on VC Summer 2007 NRC exam.

MCS	Time: 1	Points:	1.00	Version:	0 1 2 3 4 5 6 7 8 9		
				Answer:	C B C A D A A A B A	Scramble Range: A - D	
Source :		BANK			Source if Bank:	FARLEY	
Cognitive Level:		HIGHER			Difficulty:		
Job Position:		RO			Plant:	FARLEY	
reviewed:		GO			Previous 2 NRC exams:		

14. 009 EA1.17 001

Given the following:

- A reactor trip and safety injection have occurred.
- RCS pressure is 1450 psig.
- Containment pressure is 7.5 psig.
- SG pressures are 1000 psig.
- All equipment has operated as designed.

Which ONE of the following would have a rising level due to the RCP #1 seal return flow?

A. VCT

B.✓ PRT

C. RCDT

D. Containment Sump

A is incorrect. credible because it is the normal #1 seal flowpath.

B is correct. Containment isolation will isolate seal return flow, and the seal return relief valve will lift and direct the flow to the PRT.

C is incorrect. credible because it is the #2 seal flowpath.

D is incorrect. credible because #3 seal flow path is directed to the Ctmt sump.

009 EA1.17 Small Break LOCA -

Ability to operate and monitor the following as they apply to a small break LOCA:
PRT

Question Number: 40

Tier 1 Group 1

Importance Rating: RO 3.4

Technical Reference: OPS-52101F EEP-0 attachment 3 figure 1
drawing D-175039 sheet 1 and D-175037 sheet 2 D-7

Proposed references to be provided to applicants during examination:
Learning Objective: OPS40301F05
10 CFR Part 55 Content:

Comments:
I changed the stem from describes where the the RCP seal return flow is being directed to would have a rising level due to the RCP seal return flow to meet the ability to monitor the PRT parameters piece of the KA.

MCS Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
		Answer:	B C C C B D C B B B	Scramble Range: A - D	
Source :	BANK		Source if Bank:	SONGS 2005	
Cognitive Level:	LOWER		Difficulty:		
Job Position:	RO		Plant:	FARLEY	
reviewed:	GTO		Previous 2 NRC exams:	NO	

15. 010 K3.01 002

Given the following:

- Unit 1 is at 100% power.
- All control systems are in their normal alignments.

PT-445, Pressurizer Pressure Channel fails **HIGH**.

Which ONE of the following describes the **initial** effect on RCS pressure and the reason for that effect?

RCS pressure ____________________________

A. **rises** due to ONLY Variable heaters energizing.

B. **rises** due to ALL Backup and Variable heaters energizing.

C✓ **lowers** due to one PRZR PORV opening.

D. **lowers** due to ALL Backup and Variable heaters de-energizing, and both spray valves and one PORV opening.

A incorrect; because pressure will lower initially. Credible because it is consistent with a controller failure, which could be confused with an input failure from PT-444. Also the heaters will come on when pressure starts dropping from PK-444A. The pressure will not initially drop and will not rise until the PORV closes (cycles at 2000#)

B incorrect; see above, only all heaters are involved and could be confused between a level control failure, PT444 failure and this failure.

C Correct; When PT-445 fails high, PORV 445A opens and will drop pressure to 2000 psig. The valve will close per design at 2000 psig which comes from PT-455, 456 and 457 on 2/3 < 2000 psig. The PORV will cycle at 2000 psig if a rx trip and SI did not occur.

D incorrect. Would be correct for PT-444 failing high. The lesson plan says that all heaters will turn on when the RCS pressure drops, sprays will close and PORV 444B will remain closed.

010 K3.01 Pressurizer Pressure Control System
Knowledge of the effect that a loss or malfunction of the PZR PCS will have on the following:
RCS

Question Number: 10

Tier 2 Group 1

Importance Rating: 3.8

Technical Reference: AOP-100, OPS-52201H

Proposed references to be provided to applicants during examination: None
Learning Objective: OPS52201H17
10 CFR Part 55 Content: 41.7

Comments:

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

Source :	NEW	Source if Bank:	
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

16. 011 A3.03 002

Given the following conditions:

- The plant is stable at 90% power.
- Charging, Letdown, and Pressurizer Level Control systems are in automatic.
- The Pressurizer Level Selector Switch is in the I/II Position.
- LT-459, Pressurizer level Transmitter, has failed low.

Which one of the following describes the system response?

No operator action is taken

Charging flow will ____(1)____ and letdown flow will ____(2)____.

A. (1) increase

(2) remain the same

B.✓ (1) increase

(2) decrease

C. (1) decrease

(2) remain the same

D. (1) decrease

(2) decrease

A. Incorrect. charging flow increases through FCV-122, but letdown will isolate and flow will drop to zero.

B. correct - Charging flow will increase and letdown will isolate and flow will drop to zero.

C. Incorrect. Charging flow will increase due to indicated PRZR level low and letdown will isolate and flow will drop to zero.

D. Incorrect. Charging flow will increase due to lower indicated PRZR level and letdown will isolate and flow will drop to zero.

Reference:
CFR: 41.7 / 45.5

OPS-52201H (in part
459 (I) Low Low level alarm
LCV-459 closes
Orifice isolation valves close
All pressurizer heaters turn off
Charging flow increases to maximum
Actual pressurizer level increases because of secured letdown and maximum charging flow
High level alarm from channel 460 (III)
Reactor trip on high pressurizer level if no operator action is taken

011 A3.03 Pressurizer Level Control System
Ability to monitor automatic operation of the PZR LCS, including:
Charging and letdown

Question Number: 31

Tier 2 Group 2

Importance Rating: 3.2

Technical Reference: PZR level/Press LP OPS-52101E & H, UOP-3.1
Proposed references to be provided to applicants during examination: None
Learning Objective: OPS52201H15
10 CFR Part 55 Content: 41.7

~~Comments: changed out question b/c it was not correct.~~
This meets the KA in that the questions has the candidate monitor letdown and charging flows and the affects on one other system during this failure.
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B C D A B C D A B C Scramble Range: A - D

Source :	MODIFIED	Source if Bank:	FARLEY
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

17. 012 A2.02 001

At 10:00 plant conditions were as follows:

- Unit 1 was at 41% power, ramping down due to RCS leakage greater than Tech Spec limit.
- 120V AC vital panel 1A was been de-energized 2 hrs ago due to damage to the breaker panel.
- DF01, 1A S/U transformer to 1F 4160V bus, tripped open.

At 10:10, a Large Break LOCA occurred.

Which ONE of the following describes the (1) the status of the Reactor Trip Breakers at 10:05; **and** (2) the action(s) required IAW EEP-0, Reactor Trip or Safety Injection, concerning ESF components?

At 10:05 the Reactor Trip Breakers will be ____(1)____.

After the LBLOCA, the operator is required to manually align ____(2)____

A.✓ (1) open

(2) "A" Train ESF components ONLY.

B. (1) open

(2) BOTH trains of ESF components.

C. (1) closed

(2) "A" Train ESF components ONLY.

D. (1) closed

(2) BOTH trains of ESF components.

A. Correct, The RTBs will open due to the loss of power from the solas to the RCP Single loop loss of flow (SLLOF) to 2 RCPs on A Train. At 41% power, this will result in a Rx trip.
Then due to the loss of the vital panel, the "A" Train ESF components will not actuate since the "A" train output relay cabinet slave relays will not actuate due to the loss of power.

FSD A-181007

2. Reactor Coolant Pump Breaker Trip
Opening of one or two reactor coolant pump breakers (depending upon power level), which is indicative of an imminent loss of coolant flow in the loop or loops, will cause a reactor trip. If two of three pump breakers trip with plant power > P-7 10% RTP, a reactor trip will occur. Below P-7 (10% RTP), the trip is

automatically blocked. Also if 1/3 pump breakers are opened with plant power > P-8 (30% RTP), a reactor trip will occur.
(References 6.1.003, 6.4.007, 6.7.012)

ops-52201 RPS

Low Flow or RCP Breaker Open Trip

There are three low flow protection bistable status lights for each loop (total of nine lights) on TSLB-2. There is also a protection bistable status light for each reactor coolant pump (RCP) breaker on TSLB-2. A low flow condition in any loop as detected by the open RCP breaker or 2/3 low flow signals will energize the respective loop low flow partial reactor trip alarm, A(B, C) RCS LOOP FLOW LO OR A(B, C) RCP BKR OPEN. If reactor power is greater than the P-8 setpoint (30 percent), the low flow condition will cause a reactor trip. This is indicated by the ONE LOOP LO FLOW OR RCP BKR OPEN RX TRIP alarm. If reactor power is less than the P-8 setpoint but greater than the P-7 setpoint, a low flow condition in 2/3 loops will cause a reactor trip. This is indicated by the TWO LOOP LO FLOW OR RCP BKRS OPEN RX TRIP alarm.

B. Incorrect - first part is correct.
second part in not correct in that the master relay is not the relay that causes this issue and B Train ESF components will actuate from B Train.

C. Incorrect, RTBs will open
second part is correct.

D. Incorrect, both parts incorrect. see above for explanation.

FSD A-181007

Figure F-1 is a block diagram, illustrating the Reactor Protection System FSD boundaries. The equipment shown depicts the Reactor Protection System and its interfaces as follows:
1. Analog protection system cabinets (W 7300 System Racks) containing the bistables which input to the Reactor Protection. Although the process instruments which interface with the analog protection system are considered part of the reactor protection system as defined by IEEE 279, they are also considered as part of their respective fluid systems. For completeness, the process sensors are included in this FSD. The functional requirements associated with the interfacing process input components to the RPS are those that are applicable to these type devices on a generic basis.
2. NIS Racks (the bistables which input to the RPS and the sensors contained within).
3. Control board switches
4. Field contacts (RCP breakers, turbine stop valves, etc.)
5. Solid State Protection System (SSPS) initiates reactor trip or ESF actuation in accordance with defined logic that is based on the bistable outputs from the process racks (7300/NIS Racks). Table T-8 depicts the system interfaces associated with the SSPS Output Cabinet providing Reactor Trip and ESF actuation functions.
6. Reactor trip switchgear, normal and bypass breakers
7. Computer and control board demultiplexers (demux)

8. AMSAC (Anticipated Transient Without Trip (ATWT) Mitigation System Actuation Circuitry) is not shown nor considered part of the RPS FSD but is being mentioned here for completeness.

The W 7300 and NIS Racks provide the signal conditioning, setpoint comparison, process analog signal actuation, control board/control room/ miscellaneous indications and compatible electrical signal output to the protection devices. The bistable outputs pertaining to these systems which provide this input to the RPS have been included as part of this FSD and are listed in Section 7. The process instruments which input into the W 7300 and NIS Racks have been included as part of this FSD and are listed in Table T-7. (References 6.1.001, 6.1.002, 6.7.001, 6.7.002, 6.7.004, 6.7.010, 6.7.057)

012 Reactor Protection System

A2.02 Ability to (a) predict the impacts of the following malfunctions or operations on the RPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Loss of Instrument Power

Question Number: 25

Tier 2 Group 1

Importance Rating: 3.6

Technical Reference: **FSD A-181007** , Figure 12 of ops-52201 RPS lesson plan

Proposed references to be provided to applicants during examination: None
Learning Objective: OPS52201D12
10 CFR Part 55 Content: 41.10

Comments:
This question tests the loss of power to both sides of SSPS, the affects from the loss of power thru the vital panel and the affects from the loss of power from the solas, and the action required by the ERGs due to that loss. The second half of the KA is met by knowing from the question what the effects are to the plant to the failures listed, and then using EEP-0, know what the operator will have to do based on the failure to correct the malfunctions that are identified. A loss of power to solid state affects the ESF components as well as RPS components equally since power is lost to both protection and control as well as RPS.

Logic Cabinet (Figures 4 and 11)
The logic cabinet is to the right of the input relay cabinet. It houses the circuitry to make logic decisions. The logic circuitry receives signals from the input relay cabinet and if appropriate signals are received, **it will initiate a reactor trip or actuate the ESF systems**. In addition to logic decision making, information is collected, stored, and transmitted (via multiplexing techniques) to the computer and control board (via demultiplexing techniques).
Both signals come from the logic cabinets.
The logic circuits look for coincidence between protection channels. If the logic requirements are met for a reactor trip, the circuit sends a signal to the UV driver card. The UV driver card output drops from 48V DC to zero and de-energizes its associated reactor trip and bypass breaker UV coils. This action trips open the breakers and de-energizes the control rod drive mechanisms. This releases the control rod assemblies into the core. The train A UV driver card sends its trip signal to reactor trip breaker RTA, and to bypass breaker BYB.
If an unsafe condition calls for safeguards actuation, the logic circuits will send a signal to the safeguards driver card. The card's output will increase from zero to 48V DC and will energize the required master relays for the specific safeguards actuation. The master relays energize their slave relays using 120V AC, which supply either AC or DC control power to ESF loads as appropriate.

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

Source :	MODIFIED	Source if Bank:	FARLEY
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	

18. 012 A4.01 001

Given the following:

- Unit 1 is operating at 100% power when a PRZR PORV spuriously opens.
- The control room operators attempt to close the PORV but are unsuccessful.
- The UO closes the block valve for the open PORV.

The following conditions exist:

- Tavg is 575°F.
- PRZR level is 63%.
- PRZR pressure is 1845 psig and rising slowly.
- Reactor power is 98%.

Which ONE of the following actions are required?

A. Restore RCS pressure to >2209 psig within 1 hour.

B. Commence plant shutdown and be in hot standby per UOP-3.1, Power Operation.

C.✓ Manually trip the reactor, initiate SI, and enter EEP-0, Reactor Trip or Safety Injection.

D. Maintain the PORV block valve closed with power available.

A. Incorrect - TS 3.4.1 requires restoration w/i 2 hours or be in mode 2 in 6 hours. This would be true if a reactor trip were not required at this time.

B. Incorrect - This would be an action due to rising PRZR level IAW 3.4.9 when Pzr level is greater than 63.5%.

C. Correct - this action is necessary since PRZR pressure is below the reactor trip and SI setpoint.

D. Incorrect - This is the TS action if the plant would remain at power.

012 A4.01 Reactor Protection System
Ability to manually operate and/or monitor in the control room:
Manual trip button

Question Number: 11

Tier 2 Group 1

Importance Rating: 4.5

Technical Reference: AOP-100; TS 3.4.11
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.10

Comments:

MCS	Time: 1	Points:	1.00	Version:	0 1 2 3 4 5 6 7 8 9		
				Answer:	C B A B B B A A D B	Scramble Range: A - D	
Source :		BANK			Source if Bank:		FARLEY
Cognitive Level:		LOWER			Difficulty:		
Job Position:		RO			Plant:		FARLEY
reviewed:		GO			Previous 2 NRC exams:		NO

19. 013 A4.02 002

Given the following:

- A LOCA has occurred.
- RCS pressure is 500 psig and stable.
- Containment pressure is 29 psig and lowering slowly.
- All equipment is operating as designed.
- The crew is performing actions contained in ESP-1.2, Post LOCA Cooldown and Depressurization, preparing to reset ESF Actuation signals.

Which ONE of the following describes the conditions required to be met, if any, to reset Containment Isolation Phase A and B?

A.✓ • Phase A may be reset without additional conditions.
• Phase B may be reset without additional conditions.

B. • Phase A may be reset without additional conditions.
• Containment Spray must be reset prior to resetting Phase B.

C. • Phase A may be reset without additional conditions.
• Phase A must be reset before resetting Phase B.

D. • Safety Injection must be reset before resetting Phase A.
• Containment pressure must be less than the actuation setpoint before resetting Phase B.

A is correct. Manual resets for Phase A and Phase B may be performed even with actuating signal present.

B is incorrect. Credible because CTMT spray and Phase B have the same actuating signal. CS actuation and Phase B can be reset independently of each other and at any time.

C is incorrect. Credible because procedure directs Phase B reset after Phase A. Phase B can be reset at any time.

D is incorrect. Credible because SI is the automatic initiation signal for Phase A. Phase A can be reset prior to SI reset.

013 Engineered Safety Features Actuation System

A4.02 Ability to manually operate and/or monitor in the control room:
Reset of ESFAS channels

Question Number: 13

Tier 2 Group 1

Importance Rating: 4.3

Technical Reference: EEP-1.0, Reactor Protection System FSD, A181007, Figure F-2 sheet 8 & Table T-4
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.7 / 45.5 to 45.8

Comments:
meets the KA in that the question asks for the conditions needed to reset ESFAS channels in the CR

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

Source :	BANK	Source if Bank:	
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

20. 013 G2.4.31 008

Plant conditions at 09:00 were as follows:

- Unit 1 was at 100% power.
- SSPS train "B" surveillance testing was in progress.
- ED4, SSPS B TRN TRBL, was in alarm due to the SSPS testing.
- The "B" Reactor Trip Bypass Breaker were closed and the associated annunciator alarms have been acknowledged.

At 09:30 the following occurs:

- EC4, SSPS A TRN TRBL, has come into alarm.
- All remaining annunciators are unchanged.
- The plant operator reports that the SSPS train "A" Output Relay Mode Selector switch was **inadvertently** placed in the **TEST** position.

Which ONE of the following actions are required?

A. Apply Technical Specification 3.0.3 and initiate actions within 1 hour to shut down to Hot Standby.

B. Stop the testing, check the "B" Reactor Trip Breaker closed, then open the "B" Reactor Trip Bypass Breaker.

C✓ Initiate a manual reactor trip; if unsuccessful, enter FRP-S.1, Response to Nuclear Power Generation/ATWT.

D. Immediately place the SSPS train "A" Output Relay Mode Selector switch in the **OPERATE** position; then verify EC4, SSPS A TRN TRBL, annunciator has cleared.

For this condition while testing is on-going in the SSPS B Train cabinets, with the bypass breaker closed, the general warning light will be lit for B Train. This will cause ED4 to be in alarm. Then when the SSPS train "A" Output Relay Mode Selector switch is placed in the **TEST** position the other trains GW light will be LIT and 2 GW lights should cause a reactor trip.
some P&Ls of STP-33.0 follow:
4.2 Ensure that the GENERAL WARNING lamps on SOLID-STATE PROTECTION TRAIN-A and B LOGIC CABINETS are OFF prior to commencing this test.
4.6 IF a failure occurs during testing, THEN hold at the point the failure occurs and contact Maintenance for troubleshooting and repair.
4.1 The GENERAL WARNING lamps on SOLID-STATE PROTECTION TRAIN-B and B LOGIC CABINETS will be ON during this test.

A. incorrect. Both trains of Solid State are inoperable, and a common TS which applies when both trains of safety related equipment is inoperable is 3.0.3, but a reactor trip is required for this condition.

B. incorrect. "Backing out" of a test when the other train becomes inoperable is normally done, but due to the GW on both trains a reactor trip is required.

C. correct. Reactor should have tripped with 2 trains in test due to both trains have a general warning in which is the input to the automatic trip coincidence of 2/2 GW alarms in.
ARP EC4
AUTOMATIC ACTION
1. IF both Train A AND B Solid State Protection System Trouble alarms are actuated, THEN a reactor trip will occur.

D. incorrect. If SSPS train A had not been momentarily inoperable, no further action would be required. The coincidence for a reactor trip would no longer be met. Quickly restoring the output switch to it's original position would not change the fact that the SSPS should have initiated an automatic trip, and met coincidence for one, but it did not occur. A reactor trip is necessary.

013 G2.4.31 Engineered Safety Features Actuation System
Emergency Procedures / Plan:
Knowledge of annunciators alarms and indications, and use of the response instructions.

Question Number: 12

Tier 2 Group 1

Importance Rating: 3.3
Technical Reference: ARP EC4
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.10

Comments:

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

Source :	BANK	Source if Bank:	
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GTO	Previous 2 NRC exams:	

21. 015 AK3.01 002

Given the following:

- Unit 2 is in Mode 4 with two RCPs running.

The crew is at a step in UOP-1.1, Startup of Unit from Cold Shutdown to Hot Standby, to start a third RCP.

Which ONE of the following correctly describes a RCP failure mechanism that will still allow the remaining RCP to be started, the damage that would occur and the reason?

A.✓ • The anti-reverse rotation device pawls are not engaged in the ratchet plate.

• RCP motor winding damage due to high starting currents.

B. • The anti-reverse rotation device pawls are not engaged in the ratchet plate.

• RCP radial bearing damage due to reverse flow through the RCP.

C. • The oil lift pump does not develop 600 psig oil lift pressure.

• RCP radial bearing damage due to high starting torque.

D. • The oil lift pump does not develop 600 psig oil lift pressure.

• RCP motor winding damage due high starting torque.

A. Correct - the anti rotation device pawls not being engaged in the ratchet plate would cause the high motor winding temps and possible damage to the windings due to high starting currents.
A flywheel and an anti-reverse rotation device have been mounted at the top of the motor. Stopping one or more RCPs while other pumps are running will cause a reverse flow through the inactive loops. Reverse flow will turn the de-energized pump backwards. **Although no mechanical damage would result from reverse rotation, an attempt to start a pump in this condition would cause starting currents to exist for an excessive length of time, resulting in overheating of the motor. To prevent reverse rotation, each pump has been equipped with an anti-reverse rotation device.**

B. Incorrect - first part is correct. second part is not due to the fact that no damage will result to a RCP due to reverse flow thru the pump.

C. Incorrect - The oil lift pump does not develop 600 psig or greater oil lift pressure would not allow the RCP to be run. If it were to be started with the pressure not being >600 psig, then the second part is in fact correct for the oil lift pump but not for the radial bearing. It would actually potentially damage the windings.
3.6 DO NOT attempt to start a RCP unless its oil lift pump has been delivering oil to the upper thrust shoes for at least two minutes. Observe the oil lift pumps indicating lights to verify correct oil pump motor operation and oil pressure. The oil lift pumps should run at least 1 minute after the RCP's are started. **An interlock will prevent starting a RCP until 600 psig oil pressure is established.**

system before starting the motor. The oil "lifts" the thrust shoes away from the thrust runner.

D. Incorrect - see above for the first part. RCP would not start in this condition due to an interlock.
second part deals with the lower bearing. information below. RCP motor winding damage could result due to the upper thrust shoes.

RCP lesson plan
The lower thrust bearing takes the weight of the rotating parts when the reactor coolant loop is at low pressure. As the loop pressure increases, the unbalanced force on the number one seal causes the shaft to lift and transfer the thrust to the upper thrust bearing. By the time loop pressure is sufficient to allow pump operation (350 psig), all thrust acts on the upper bearing, which is the normal operating condition.

The lower thrust bearing functions only when the motor runs uncoupled from the pump.
In this condition, the weight of the motor rotating parts acts downward.

In order to reduce starting torque, the thrust bearing shoes receive oil from the oil lift system before starting the motor. The oil "lifts" the thrust shoes away from the thrust runner.
The lower thrust bearing takes the weight of the rotating parts when the reactor coolant loop is at low pressure. As the loop pressure increases, the unbalanced force on the number one seal causes the shaft to lift and transfer the thrust to the upper thrust bearing. By the time loop pressure is sufficient to allow pump operation (350 psig), all thrust acts on the upper bearing, which is the normal operating condition.
The lower thrust bearing functions only when the motor runs uncoupled from the pump.
In this condition, the weight of the motor rotating parts acts downward.

015 Reactor Coolant Pump Malfunction -
AK3.01 Knowledge of the reasons for the following responses as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow):
Potential damage from high winding and/or bearing temperatures

Question Number: 41

Tier 1 Group 1

Importance Rating: RO 2.5

Technical Reference: RCP LP OPS-52101D, UOP-1.1
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.7

Comments: had to replace this question since it did not meet the KA, ie the reason portion of ~~the KA, and it is an exact question from 003 A1.04~~

meets the KA in that there is a potential failure identified that would cause loss of RC flow form the RCP and the reason that malfunction would cause that problem, specifically motor windings. I did not test the bearing side of the KA due to the KA already on the test 003A1.04.

QUESTIONS REPORT
for 75 RO Questions

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

Source :	MODIFIED	Source if Bank:	FARLEY
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

22. 015 K6.02 002

Given the following:

- Unit 2 was initially at 95% power.
- The reactor has tripped.
- Compensating voltage on N-35, Intermediate Range NI, is set too **HIGH**.

Which ONE of the following correctly describes the response of Intermediate Range N-35 following the trip AND the effect of this response on the Source Range (SR) HI FLUX TRIP?

N-35 will indicate ___(1)___ than actual power.

The SR HI FLUX TRIP will reinstate __________(2)__________

A. (1) **LOWER**

(2) as soon as **N-35** reaches the P-6 setpoint.

B.✓ (1) **LOWER**

(2) only when **N-36** reaches the P-6 setpoint.

C. (1) **HIGHER**

(2) only when **N-35** reaches the P-6 setpoint.

D. (1) **HIGHER**

(2) as soon as **N-36** reaches the P-6 setpoint.

A Incorrect; If one channel indicates low, it will satisfy 1/2 of the required P-6 reset logic. Therefore, if the logic was 1 of 2, A would be correct.

B Correct; An overcompensated channel means that compensating voltage is too high for the channel, cancelling out part of the actual signal, resulting in a **lower** indication. The P-6 permissive is satisfied when **2 out of 2** IR channels are below the setpoint.

C Incorrect; the channel would indicate lower per above discussion and
Would be correct if the channel was undercompensated with the actual logic (2 out of 2 logic for going below P-6)

D. Incorrect; If the logic was 1 out of 2 versus 2 out of 2, and the IR indicated high (undercompensated)

015 Nuclear Instrumentation System
K6.02 Knowledge of the effect of a loss or malfunction on the following will have on the NIS: **Discriminator/compensation circuits**

Question Number: 32

Tier 2 Group 2

Importance Rating: 2.6

Technical Reference: UOP-2.3, ESP-0.1, step 11 and NI LP OPS-52201D
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.7

Comments: good match

MCS	Time: 4	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	B C C A D A B B B A	Scramble Range: A - D	
Source :	BANK		Source if Bank:		VCS	
Cognitive Level:	HIGHER		Difficulty:			
Job Position:	RO		Plant:		FARLEY	
reviewed:	GO		Previous 2 NRC exams:		NO	

23. 017 K1.02 001

The Subcooled Margin Monitor is in the RTD mode. Which ONE of the following correctly describes the temperature instruments used in the Subcooling calculations for this mode of operation?

A.✓ The highest reading RTD of the 3 Wide Range RCS Hot leg and 3 Wide Range RCS Cold Legs is used.

B. The highest reading of Core Exit and Upper Head Thermocouples is used.

C. The fifth hottest of all Core Exit Thermocouples (excluding the Upper Head thermocouples) is used.

D. The fifth hottest RTD of all Wide Range RCS Hot legs and Cold Legs is used.

A. Correct. This is the method used in the RTD mode, but has a greater time delay in indication of a loss of subcooling due to loop transit time and instrument RTD response time than the CETC mode.

B. Incorrect. This is correct for the Individual Value display mode.

C. Incorrect. This incorrect for the Individual Value display mode. The upper head TCs are not excluded from the calculation in the individual value mode, and also the fifth hottest is not used in the calculation even though the fifth hottest is used for diagnostic purposes throughout the ERG procedure network.

D. Incorrect. This is incorrect. The fifth hottest of all RTDs is not used

017 K1.02 In-Core Temperature Monitor System
Knowledge of the physical connections and/or cause effect relationships between the ITM system and the following systems:
RCS

Question Number: 33

Tier 2 Group 2

Importance Rating: 3.3

Technical Reference: ICCMS LP OPS-52202E, SOP-68.0
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.5

Comments:

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

Source :	MODIFIED	Source if Bank:	
Cognitive Level:	LOWER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

24. 022 AA2.04 002

Given the following:

- Unit 1 is at 60% power.
- Pressurizer level is on program.
- All charging flow has been lost and **NO** charging pump is running.
- Letdown has been secured.
- PRZR level is lowering at a rate of 1% every five (5) minutes.
- 100% Tref = 573°F

Approximately how much time will pass before all pressurizer heaters will automatically secure assuming no operator action?

A. 1 hour

B. 1.5 hours

C✓ 2 hours

D. 2.5 hours

A. Incorrect; 1/2 the value of C.

B. Incorrect; 60% X 28.8 will result in approx. 17.8%. Level lowering at 1% every 5 minutes yields approx. 89 minutes, about 1.5 hours.

C. Correct; At 60% power, program level is approximately 38.7%. This is calculated by taking 573 - 547 = 26 which = 2.6 X 10% change in power. 60% power or 6 x 2.6 = 15.6 + 547 = 562.6°F.
Level at this temperature is 38.7% based on 50.2 - 21.4 = 28.8 or 2.88 per 10

6 X 2.88 = 17.28 + 21.4 = 38.68 (level at 50% + 35.8 or 1/2 of 28.8 +21.4)

Letdown isolates at 15%.
38.7 -15 = 23.68 % level decrease and since 1% per 5 minutes is the level decrease, the time would be 5 X 23.68 = 118.4 or 2 minutes less than 2 hours

D. Incorrect; Using the 18.7% and subtracting from 50.2% level yields approx. 34.6% x 5 = 157.5 minutes or 2.5 hours

calculation:
at 100% power level is 50.2% and at 0% power level is 21.4%. This gives a level change of approx. .288% per % power. Letdown isolates at 15%.

A508617 pzr level setpoint document shows Pzr level to be 21.4% at 547°F and 54.9% at 577.2°F. The lesson plan shows level to be 21.4% at 547°F and 50.2 at 573°F. figure 6 shows program level to be 21.4 to 50.2 from 547 to 573°F Tavg

Options are plausible because they are symmetrical and not significantly different from actual time. A math error or misunderstanding of program level at this power could direct an applicant to any option.

022 Loss of Reactor Coolant Makeup -

AA2.04 Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Pump Makeup:
How long PZR level can be maintained within limits

Question Number: 42

Tier 1 Group 1

Importance Rating: RO 2.9

Technical Reference: OPS 52201H FIG 6 and AOP-16
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.7

Comments:
Tref is given due to the fact that both units are different and we teach Tref at 573°F and explain that this value changes most every outage and is a moving target.

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

Source :	BANK	Source if Bank:	WTSI
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	

25. 022 K1.01 001

Given the following initial conditions with Unit 1 at 100% power:

There is a small leak in the 1A containment cooler. The cooler has been isolated IAW SOP-12.1, Containment Cooling System. The following valves are closed for leak isolation:

- MOV-3019A, SW TO 1A CTMT CLR AND CTMT FPS
- MOV-3024A, EMERG SW FROM 1A CTMT CLR
- MOV-3441A, SW FROM 1A CTMT CLR

A Large Break LOCA occurs at this time. Which ONE of the following correctly describes the Service Water flow rate (if any) through the 1A containment cooler with no operator action?

SW flow will be ___________.

A. secured

B. approximately 600 gpm

C. approximately 800 gpm

D.✓ approximately 2000 gpm

Cooling water normally discharges through a 10-inch line including MOV-3441A, B, C, and D, which are located inside containment, then through a 6-inch line and MOV-3023A, B, C, and D. On an "S" signal, water also discharges through a 10-inch discharge line through MOV-3024A, B, C, and D, thus increasing the flow through the coolers.

MOV-3024A, B, C, and D Containment Cooler Emergency Service Water Discharge Valves (Figure 19)

Each motor-operated valve is controlled by a three-position MCB handswitch (CLOSE/AUTO/OPEN, spring return to AUTO). In the AUTO position, the valve automatically opens upon receiving an S-signal. Valve position indication lights are above each switch.

MOV-3441A, B, C, and D Containment Cooler Service Water Discharge Isolation Valves

The operation of these MCB motor-operated valves (Figure 19) is identical to the emergency service water discharge valves (3024A, B, C, and D).

MOV-3019A, B, C, and D Containment Cooler Service Water Inlet Isolation Valves

The operation of these MCB motor-operated valves (Figure 19) is identical to the emergency service water discharge valves (3024A, B, C, and D).

A. Incorrect - due to the SI signal MOV-3019A, MOV-3441A and MOV-3024A open to provide emergency SW to the coolers.
MOV-3023A is normally open and is not closed for the leak isolation IAW SOP-12.1.

This would be correct if the candidate did not know what valves rolled open then the effect on containment temperature due to the failure.

B. Incorrect - This could be confused with the TS bases requirement to have 600 GPM flow from one Ctmt cooler to meet post LOCA conditions.

C. Incorrect - This could be confused with not knowing the correct valve line up and this is the normal flow rate thru the coolers which if the emergency SW to the valve was not taken into account for this event, this would be the correct answer.

D. Correct - Since due to the SI signal MOV-3019A, MOV-3441A and MOV-3024A open to provide emergency SW to the coolers flow rate would increase to 2000 gpm thru all coolers with two SW pumps running whether the fan is running or not. Cooling at 2000 gpm will provide the cooling necessary in a LBLOCA to cool containment.

022 Containment Cooling System (CCS)

K1.01 Knowledge of the physical connections and/or cause-effect relationships between the CCS and the following systems:
SWS/ cooling system

Question Number: 14

Tier 2 Group 1

Importance Rating: 3.5

Technical Reference: FSD A181001 Service Water System
OPS-52102F, SW lesson plan and TS bases 3.6.6
Proposed references to be provided to applicants during examination: None
Learning Objective:

10 CFR Part 55 Content: 41.5

Comments: This KA was changed to 022K1.01 due to not being able to meet the selected KA. This was approved by FJE and recorded on ES-401-4 record of rejected KAs.
This meets the KA in that it asks for the cause effect of only having one sw flow path to one operating ctmt cooler.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	D C C D C C D D C A	Scramble Range: A - D
Source :	MODIFIED		Source if Bank:	FARLEY	
Cognitive Level:	HIGHER		Difficulty:		
Job Position:	RO		Plant:	FARLEY	
reviewed:	GO		Previous 2 NRC exams:	NO	

26. 025 AK2.03 002

The following conditions exist:

- The plant is in Mode 1.
- 1A CCW pump is tagged out to have the motor rebuilt.
- 1B and 1C CCW pumps are both in operation.
- B Train is the "On Service" train.

The 1C CCW pump has just tripped.

Which one of the following correctly describes **ONLY** components that have lost ALL CCW flow due to the 1C CCW pump trip?

A. 1B RHR Hx, 1B RHR Pump Seal Cooler, 1B Spent Fuel Pool Hx

B. 1B RHR Hx, 1B RHR Pump Seal Cooler, 1A Spent Fuel Pool Hx

C. 1A RHR Hx, 1A RHR Pump Seal Cooler, 1A Spent Fuel Pool Hx

D.✓ 1A RHR Hx, 1A RHR Pump Seal Cooler, 1B Spent Fuel Pool Hx

C CCW pump is the A train pump-
The B CCW pump is the onservice train and is carrying the misc. header and on B Train.

A. Incorrect - RHR components are not correct, SFP is correct.

B. Incorrect - All are B Train ESF loads supplied by B CCW pump.

C. Incorrect - RHR components are correct, SFP is not correct.

D. correct - All are A Train ESF loads supplied by C CCW pump

OPS-52102G
The ESS loads consist of the following:
1. Charging pumps
2. Spent fuel pool heat exchangers
3. RHR heat exchangers
4. RHR pumps

The secondary heat exchanger loads consist of the following:
1. RCP oil coolers and thermal barrier heat exchangers
2. Reactor coolant drain tank (RCDT) heat exchanger
3. Excess letdown heat exchanger
4. Seal water heat exchanger
5. Letdown heat exchanger
9. Waste gas compressors
10. Sample system heat exchangers
11. Gross failed fuel detector

The CCW system provides cooling for the Train A and Train B emergency core cooling system components. The C CCW pump and the C heat exchanger are designated as Train A. The A CCW pump and the A CCW heat exchanger are designated as Train B. The B CCW pump and the B CCW heat exchanger can be aligned to either train.
CCW is normally lined up so that one CCW pump and one CCW heat exchanger is in operation supplying the on-service train. The on-service train is the one that supplies the secondary heat exchangers. The swing pump and heat exchanger (1B CCW pump and 1B HX) is normally aligned in standby to the on-service train with the heat exchanger outlet valve shut. The remaining pump and heat exchanger is valved into a closed loop with the redundant safety train. This train is idle and is designated as the off-service train. The off-service train CCW pump must be running before starting the off-service train charging pump or RHR pump.

25 Loss of Residual Heat Removal System
AK2.03 Knowledge of the **interrelations** between the Loss of Residual Heat Removal System and the following:
Service water or Closed cooling water pumps

Question Number: 43

Tier 1 Group 1

Importance Rating: RO 2.7

Technical Reference: ARP-1.1 AD5 and AE4, UOP-1.1, SOP-7.0 and SOP-23

Proposed references to be provided to applicants during examination: None
Learning Objective:

10 CFR Part 55 Content: 41.7 / 45.7

Comments: meets the KA in that the interrealtionships between RHR and CCW and a CCW pump trip has to be evaluated to arrive at the correct answer.

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

Source :	MODIFIED	Source if Bank:	FARLEY
Cognitive Level:	LOWER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

27. 026 A3.01 004

Given the following:

- Unit 1 was in Mode 5; Unit 2 was at 100% power.
- A Dual-unit Loss of Offsite Power has occurred.
- All DGs have started and tied onto the 4160V ESF buses.
- Vital load sequencing has been completed.
- ESP-0.1, Reactor Trip Response, has been entered on Unit 2.
- An inadvertent 'A' Train CS actuation signal is received while the crew is responding to the reactor trip.

Which ONE of the following correctly describes the status of the Unit 2 Train 'A' Containment Spray (CS) system?

	2A CS Pump	2A CS Pump Discharge Valve
A.	Stopped	Closed
B.✓	Stopped	Open
C.	Running	Closed
D.	Running	Open

A. Incorrect - Neither CS pump will start, due to the LOSP with no SI actuation (pump controls logic diagram), but the Train A inadvertent actuation for containment spray would cause the A train pump discharge MOV to open.

B. Correct - See FSD narrative below. ESS loading sequencer requires an SI signal to be present before sequencing on ESS loads. An SI signal has not occurred for this transient so the pump will not start but the valve will open.

C. Incorrect - The inadvertent Train A containment spray actuation would not cause the 2A spray pump to start for reason given in A above. The A train MOV would open for reasons given in A above. The B Train would be unaffected.

D. Incorrect - See 2A pump will not start per the FSD narrative below.

OPS-52102C
Containment Spray Pumps (Figure 6)
A three-position (STOP/AUTO/START, spring return to AUTO) handswitch controls each pump. Placing the switch in the START position will start the pump. Placing the switch in the STOP position will stop the pump and reset the 86 relay. In the AUTO position, the pump will automatically start upon receipt of a containment spray actuation signal ("P" signal) if an LOSP has not occurred. **If an LOSP has occurred with the "P" signal present, a safety injection signal to the ESF sequencer must also be present, or the ESF sequencer must be in test, to start the pump.** T

Containment Spray System FSD A181008
3.1.5.2 With offsite power available, the "P" signal shall start both CSS pumps. Without offsite power available, the CSS pumps shall start by the diesel generator ESS loading sequencer. Starting will occur at step two of the sequence if the "P" signal is present at that time. If the "P" signal occurs between the completion of step two and step six of the ESS sequence, then starting will occur at the completion of step six of the loading sequence. If the "P" signal occurs after the completion of step six, starting will take place immediately. Automatic starting of the CSS pumps shall not occur unless the pump control switch on the main control board is in the "AUTO" position (References 6.4.001, 6.4.006, 6.4.007, 6.4.008).

3.6.1.1
These active valves shall open automatically upon receipt of a containment spray actuation signal ("P" signal) from the ESFAS and remain open for the containment spraying function (Reference 6.2.001).

026 Containment Spray System

A3.01 Ability to monitor automatic operation of the CSS, including:
Pump starts and correct MOV positioning

Question Number: 15

Tier 2 Group 1

Importance Rating: 4.3

Technical Reference: CS & Cool OPS-52102C
drawings - D207195 D207645 D207653 D207646

Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.7

Comments:

MCS Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
		Answer:	B A D C A C D A D C	Scramble Range: A - D	
Source :	BANK		Source if Bank:	FARLEY	
Cognitive Level:	HIGHER		Difficulty:		
Job Position:	RO		Plant:	FARLEY	
reviewed:	GO		Previous 2 NRC exams:	NO	

28. 026 AA1.02 005

The following conditions exist:

- Unit 2 is in Mode 3 preparing for a reactor startup.
- 1B CCW pump is tagged out.
- The on service CCW pump trips due to over-current.
- The other CCW pump will not start from the MCB.
- The crew just tripped the RCPs IAW AOP-9.0, Loss of Component Cooling Water.

Which ONE of the following correctly describes operation of the charging pumps while performing AOP-9.0, Attachment 1, Establishing Firewater Cooling to a Charging Pump?

A. • Stop all charging pumps until CCW or fire water is established to at least one charging pump.
• Maximum allowable Charging Pump lube oil temperature is **160°F**.

B. • Stop all charging pumps until CCW or fire water is established to at least one charging pump.
• Maximum allowable Charging Pump lube oil temperature is **140°F**.

C✓ • Swap operating charging pumps until fire water is established to one charging pump.
• Maximum allowable Charging Pump lube oil temperature is **160°F**.

D. • Swap operating charging pumps until fire water is established to one charging pump.
• Maximum allowable Charging Pump lube oil temperature is **140°F**.

AOP-9.0, Version 18

A. Incorrect- It is not correct to stop all charging pumps since the RCP seals could be damaged. The temperature is correct.

B. Incorrect - It is not correct to stop all charging pumps since the RCP seals could be damaged. The temperature is NOT correct.

C. Correct - This is the correct way to operate the chg pumps and the correct temperature.

D. Incorrect - This is the correct way to operate the chg pumps and NOT the correct temperature.

AOP-9 attachment 1
Note:
Until alternate cooling is established, swapping the operating CHG PUMP may lengthen the time that RCP seal injection is maintained.

EA3
Dispatch operator to determine the affected pump and the actual temperature as indicated on the local temperature indicators. IF local temperature indication is:
1.1 Between 140°F and 155°F, THEN operation may continue during subsequent troubleshooting.
1.2 Between 155°F and 160°F, THEN consider shutdown of pump.
1.3 > 160°F, THEN immediately shutdown the affected charging pump.
2. IF a loss of CCW has occurred, THEN perform the actions required by FNP-1-AOP-9.0 LOSS OF COMPONENT COOLING WATER.

026 Loss Component Cooling Water
AA1.02 Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water:
Loads on the CCWS in the control room

Question Number: 44

Tier 1 Group 1

Importance Rating: RO 3.2

Technical Reference: CCW LP OPS-52102G and AOP-9
Proposed references to be provided to applicants during examination: None
Learning Objective:

~~10 CFR Part 55 Content: 41.5~~

Comments:

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

Source :	BANK	Source if Bank:	FARLEY
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

29. 026 G2.1.2 002

Given the following conditions on Unit 2:

- The plant was at 100% power when the 2A SG Main Steam line ruptured inside containment.
- All systems actuated as per design.
- Containment pressure spiked to 33 psig and is now continuing to decrease slowly.
- The crew has entered ESP-1.1, SI Termination.

Which one of the following correctly describes the **MAXIMUM** containment pressure and **MINIMUM** recirculation time that will allow the OATC to secure the CS pumps per ESP-1.1?

	Containment Pressure	Time Aligned for Recirculation
A.	15 psig	7.5 hours
B.✓	15 psig	10 hours
C.	18 psig	7.5 hours
D.	18 psig	10 hours

A - Incorrect, The 7.5 hours of operation applies to HHSI/LHSI transferring from Cold Leg recirc to Simultaneous hot/cold leg recirc, but is not long enough to meet the 8 hour minimum for operation of CS on recirc prior to securing CS.
EEP-1
[CA] WHEN 7.5 hours have passed since the start of the event, THEN go to FNP-1-ESP-1.4, TRANSFER TO SIMULTANEOUS COLD AND HOT LEG RECIRCULATION.

B - Correct, **CS has been aligned for recirculation flow for 10 hours and containment pressure is 15 psig.**
ESP-1.3, does provide guidance; Containment pressure is <16# and the time on recirc is > 8 hours.

C - Incorrect, containment pressure has to be <16 psig and it is not. The RWST at 4.5 feet and decreasing would be a time to align the CS pumps for recirc, and if this could not be done then they would be secured.

D - Incorrect, 18 psig is too high a pressure and CS has not been aligned for recirc for at least 8 hours.
ESP-1.3. step 10.3
WHEN containment spray recirculation flow has been established for at least 8 hours, AND containment pressure is less than 16 psig, THEN stop both CS PUMPs.

ESP-1.1
step 18.3
18.3 [CA] WHEN containment spray recirculation flow has been aligned for at least 8 hours, AND containment pressure is less than 16 psig, THEN stop both CS PUMPs.

026 G2.1.2 Containment Spray System
Conduct of Operations:
Knowledge of operator responsibilities during all modes of plant operation.

Question Number: 16

Tier 2 Group 1

Importance Rating: 3.0

Technical Reference: ECP-1.1
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.10

Comments:
replaced to match the KA. original question did not have an operator responsibility. This question does have an operator responsibility b/c this is a Continuing action step and would occur a long period of time later in which they would be responsible for identifying and securing this system properly.

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

QUESTIONS REPORT
for 75 RO Questions

Source :	MODIFIED	Source if Bank:	FARLEY
Cognitive Level:	LOWER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

30. 027 AA2.16 001

Given the following:

- Unit 2 is at 100% power.
- PT-444, PRZR PRESS, fails **LOW**.

Which ONE of the following lists an action contained in AOP-100, Instrumentation Malfunction, that will terminate the pressure transient and stabilize RCS pressure?

A.✓ Place all pressurizer heaters to the OFF position and cycle the heaters as required.

B. Fully open both PRZR Spray valves and cycle the heaters as required.

C. Take manual control of PK-444A, PRZR PRESS REFERENCE, and reduce demand to 0%.

D. Take manual control of PK-444A, PRZR PRESS REFERENCE, and increase demand to 100%.

A correct. PT-444 inputs to the master controller. If it fails low, heaters will energize in an attempt to raise pressure. Eventually pressure will rise to a point where 1 PORV (PORV 445A) will open unless heaters are secured manually.

B incorrect. opening sprays partially would stop the pressure rise and stabilize pressure, but opening them FULLY will cause pressure to drop and continue dropping.

C incorrect. Lowering demand to 0% will secure heaters, but additionally open spray valves and one PORV. This would cause pressure to continue to drop.

D incorrect. PT-444 failing low does not cause the controller to fail low. The controller output demand is actually high at 100% due to the failure.

PT444 Fails Low

Backup heaters turn on
Variable heaters turn on to maximum
Spray valves close (if open)
Actual pressurizer pressure increases to PORV PCV-445A open setpoint causing PORV to open
Actual pressurizer pressure eventually decreases to the PORV
PCV-445A close setpoint, causing PORV to close
Plant pressure cycles around PORV open/close setpoints

AOP-100 step 4 actions:

IF an alarm was caused by a CONTROL instrument (PT-444/445) OR component failure, THEN perform the following as required to restore RCS pressure to desired value.

Take manual control of the following as required:

• Pressurizer Heaters

[ ] 1A PRZR HTR GROUP BACKUP
[ ] 1B PRZR HTR GROUP BACKUP
[ ] 1C PRZR HTR GROUP VARIABLE
[ ] 1D PRZR HTR GROUP BACKUP
[ ] 1E PRZR HTR GROUP BACKUP

027 Pressurizer Pressure Control Malfunction -
AA2.16 Ability to determine and interpret the following as they apply to the Pressurizer Pressure Control Malfunctions:
Actions to be taken if PZR pressure instrument fails low

Question Number: 45

Tier 1 Group 1

Importance Rating: RO 3.6
Technical Reference: AOP-100
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.7

Comments:
This is the action to be taken on a PT-444 failure low IAW AOP-100 and matches the KA.
Revised all choices per FJEs comments.

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

Source :	NEW	Source if Bank:	
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

31. 028 K2.01 002

Unit 1 has just lost power to 600V Motor Control Center 1B.

Which ONE of the following components will not have power?

A.✓ 1B Post LOCA Hydrogen Recombiner.

B. 1B Containment Cooler Fan - High speed.

C. HHSI TO RCS CL Isolation Valve, MOV-8803B.

D. 1B Accumulator Discharge Isolation Valve, MOV-8808B.

A. Correct - **1B Post LOCA Hydrogen Recombiner.**
OPS-52102D
B. The recombiners receive power from separate vital electrical power trains. Recombiners A and B are powered from 600V MCC A and B, respectively.

B. Incorrect - LCC B is the power supply.

C. Incorrect - MCC V; valve is outside CTMT.

D. Incorrect - MCC V; valve is inside CTMT.

028K2.01 Hydrogen Recombiner and Purge Control System
Knowledge of bus power supplies to the following:
Hydrogen Recombiners

Question Number: 34

Tier 2 Group 2

Importance Rating: 2.5

Technical Reference: Post LOCA Atm Control, OPS-LP 52102D
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.5

Comments:
different way to ask the power supply to a component to make it different from 004 K2.01.

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

QUESTIONS REPORT
for 75 RO Questions

Source :	BANK	Source if Bank:	FARLEY
Cognitive Level:	LOWER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GTO	Previous 2 NRC exams:	NO

32. 029 EK1.01 001

Given the following:

- An ATWT has occurred on Unit 2 during coastdown prior to entering a refueling outage.
- The crew is performing actions of FRP-S.1, Response to Nuclear Power Generation/ATWT.
- An operator has been dispatched to trip the reactor locally.
- Attempts to establish Emergency Boration have been unsuccessful.
- Reactor power indicates 6%.
- Intermediate Range Startup rate is slightly positive.
- The RCS temperature is slowly rising.

Which ONE of the following describes the actions required IAW FRP-S.1?

A.✓ Allow the RCS to heat up, and continue attempts to place the reactor in a subcritical condition.

B. Allow the RCS to heat up, and open one PORV as necessary to maintain pressurizer pressure less than 2135 psig to increase charging flow.

C. Stop the RCS heatup by increasing AFW flow to greater than 700 gpm, and verify dilution paths isolated.

D. Stop the RCS heatup by dumping steam to the main condenser, and continue attempts to place the reactor in a subcritical condition.

FRP-S.1 version 25

17 Continue emergency boration. 17 Perform the following.

17.1 Determine if moderator temperature coefficient positive or negative.

[] Core Physics Curve 5

17.2 IF moderator temperature coefficient negative, THEN allow RCS to heat up.

A. correct. During coastdown at EOL, MTC is negative under all conditions. Do not leave FRP-S.1 until power below 5%. IF power was to be >%5 or a positive SUR on the IR, then in addition to continuing the emergency boration, if the MTC is negative, then the RCS would be allowed to HU to add positive reactivity to the core and help shut it down.

B. incorrect The RCS is allowed to heatup, but the PORvs are not cycled to maintain pressure less than 2135 psig unless pressure is > 2335 psig.

~~C. incorrect because S.1 does not have the AFW flow to be >700 gpm for this reason,~~ but does have an RNO step to increase AFW flow to 700 gpm if SGWLs are not >31%

D. incorrect RCS temperature is not stabilized, it is allowed to rise.

029 Anticipated Transient Without Scram (ATWS)
EK1.01 Knowledge of the operational implications of the following concepts as they apply to the ATWS:
Reactor nucleonics and thermo-hydraulics behavior

Question Number: 46

Tier 1 Group 1

Importance Rating: RO 2.8

Technical Reference: FRP-S.1
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.10

Comments:
This meets the KA in that this tests the operational implications during an ATWT and the effects that we would take if the reactor was still critical after emergency boration and rods going in what would happen temperature wise, ie. thermo-hydralic behavior.

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

Source :	NEW	Source if Bank:	
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

33. 033 AK3.01 057

Given the following:

- Unit 1 is in Mode 2, a reactor startup is in progress.
- Intermediate Range Instrument N-35 indicates 1 X 10^{-8} Amps.
- FB3, NI 36 LOSS OF COMPENSATING VOLTAGE, has come in to alarm.
- N-36 indicates 1 X 10^{-11} Amps and has failed LOW.
- Repairs to N-36 will take 4 hours.

Which one of the following will satisfy the requirements of FB3 and Technical Specifications, and the reason for those actions?

A.✓ Shutdown the reactor since two IR NIs are required to remain at the current power level for the next 4 hours.

B. Remain at 1 X 10^{-8} Amps since two IR NIs are required to raise power to the POAH.

C. Remain at 1 X 10^{-8} Amps since positive reactivity additions must be suspended at this power level with one IR NI failed.

D. Increase power to >5% since the plant can remain at >5% power indefinitely with one IR NI failed.

A. Correct- Due to TS 3.3.1 below with IR power above P-6 and below P-10, two IR NIs are required or power has to be decreased below P-6 in 2 hours or >10% power in 2 hours. Since no answer allows to go to >10% power, this is the only option with one IR NI broke.

B. incorrect- power can not remain at 10^{-8} Amps and 2 IR range NIs are not required to go to the POAH. This is where critical data is taken and power is leveled off during a startup.

C. Incorrect, with power in the IR and loss of 1 channel, TS 3.3.1 requires power to be >P-10 where the PR instruments are operable, or <P-6 where SR will be operable to provide protection against uncontrolled rod withdrawl (TS Basis).

D. incorrect- if the plant could get to >10% power then the plant could remain in this mode indefinitely. Since the reactor power is in the IR there is not time to get 10% power due tot he hold at 85 AND due to placing a SGFP onservice and meeting all mode 1 entry requirements. Also the 8% power is not high enough to stay there and the UOP- 1.3 has the plant stabilize at 8%.

F	THERMAL POWER > P-6 and < P-10, one Intermediate Range Neutron Flux channel inoperable.	F.1	Reduce THERMAL POWER to < P-6.	2 hours
		<u>OR</u>		
		F.2	Increase THERMAL POWER to > P-10.	2 hours

With thermal power >P-6 and <P-10, one intermediate range neutron flux channel inoperable requires that thermal power be either reduced to <P-6 or raised to >P-10 within 2 hours.
A failure of both intermediate range detectors, with thermal power between P-6 and P-10, requires suspension of operations involving positive reactivity additions immediately and reduction of thermal power to <P-6 within 2 hours. If thermal power is <P-6 and either one or both of the intermediate range detectors become inoperable, actions must be taken to restore channel(s) to operable status prior to increasing thermal power to >P-6.

033 AK3.01 Loss of Intermediate Range Nuclear Instrumentation
Knowledge of the **reasons** for the following responses as they apply to the Loss of Intermediate Range Nuclear Instrumentation:
Termination of startup following loss of intermediate- range instrumentation

Question Number: 58

Tier 1 Group 2

Importance Rating: 3.2

Technical Reference: TS 3.3.1 Function 4; A-181007 T5-1`
Proposed references to be provided to applicants during examination: none
Learning Objective:
10 CFR Part 55 Content: 41.10/43.2

Comments:
replaced question as the original question required SRO knowledge.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	A C C D A A B D A B	Scramble Range: A - D	
Source :	MODIFIED		Source if Bank:	FARLEY		
Cognitive Level:	HIGHER		Difficulty:			
Job Position:	RO		Plant:	FARLEY		
reviewed:	GO		Previous 2 NRC exams:	NO		

34. 035 K4.06 001

Which ONE of the following describes the design requirement of the Steam Generator Safety Valves?

Limits SG pressure to no greater than 110% of design ________________

A. assuming a 100% loss of load with no credit taken for reactor trip on High PRZR pressure.

B✓ assuming a 100% loss of load with no credit taken for automatic steam dump or rod control operation.

C. assuming a limiting ATWT initiated by a loss of feedwater with no credit taken for operation of primary PORVs or secondary ARVs.

D. assuming a limiting ATWT initiated by a loss of feedwater with no credit taken for operation of primary PORVs or automatic steam dump operation.

A is incorrect but credible because credit is not taken for the initiating event, although the turbine trip could ultimately give a high pressure trip - It is credited.

B is correct.

C is incorrect because the initiating event is wrong. Credible because turbine trip and PORV operation are tied to safety valves and their capacity, and the events not credited in these options are similar to actual conditions for the safety analysis.

D is incorrect because the initiating event is wrong and the PORV operations is wrong.

035 Steam Generator System
K4.06 Knowledge of S/GS design feature(s) and/or interlock(s) which provide for the following:
S/G pressure

Question Number: 35

Tier 2 Group 2

Importance Rating: 3.1

Technical Reference: TS basis, 3.7.1 FSAR, 15.2
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.5/43.2

Comments:
this is written at an SRO level due to the references, but it is a base level of knowledge about the design of the SG safety valves.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	B C B B D B C A D D	Scramble Range: A - D
Source :	NEW		Source if Bank:		
Cognitive Level:	LOWER		Difficulty:		
Job Position:	RO		Plant:	FARLEY	
reviewed:	GO		Previous 2 NRC exams:	NO	

35. 038 EK1.02 001

Given the following:

- A Steam Generator Tube Rupture has occurred on Unit 1.
- RCS cooldown and depressurization are complete.
- The crew is maintaining the plant stable while preparing to transition to ESP-3.1, Post SGTR Cooldown using Backfill.
- The ruptured SG narrow range level is 73% and slowly decreasing.
- PRZR level is approximately 38% and slowly increasing.
- The OATC turns on PRZR heaters IAW the guidance in EEP-3, Steam Generator Tube Rupture.

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

To maintain pressurizer saturation temperature ____________________

A.✓ corresponding to ruptured SG pressure to minimize SG leakage into the RCS.

B. above the intact SG pressure to maintain adequate secondary heat sink with intact SGs.

C. above the corresponding ruptured SG pressure to ensure RCS Subcooling is maintained.

D. corresponding to intact SG pressure to ensure RCS Subcooling is maintained.

A is correct. Attempting to maintain an inventory balance between RCS and ruptured SG prior to ruptured SG cooldown.

B is incorrect because if intact SG pressure was higher than RCS pressure, EEP-3.0 would not be the governing procedure, ECP-3.1 would.

C. is incorrect. Cooldown is to ensure subcooling. RCS and ruptured SG will act like 2 pressurizers. Subcooling is not the issue.

D. is incorrect. Cooldown is to ensure subcooling. RCS and ruptured SG will act like 2 pressurizers. Subcooling is not the issue.

<u>Background document for EEP-3 page 83 of 119</u>

Purpose: To control RCS pressure and charging flow to maintain an indicated pressurizer level while minimizing primary-to-secondary leakage.

Basis: In order to explain the basis for the guidance provided in this step, consider again equilibrium conditions between leakage through the failed SG tube and charging flow, as shown in Figure 30. For primary system pressures greater than the ruptured steam generator pressure (PSG), primary-to-secondary leakage will occur so that excess charging flow, i.e., greater than letdown and coolant shrinkage, is necessary to maintain pressurizer inventory. Conversely, for letdown flows greater than charging flow, the equilibrium RCS pressure is less than the ruptured steam generator pressure and secondary-to-primary leakage will occur. The ideal conditions, shown by Point B, occur when charging flow exactly compensates for letdown and coolant shrinkage so that RCS pressure and the ruptured steam generator pressure equalize. For these conditions both the pressurizer and ruptured steam generator inventories will remain constant. Obviously fluctuations about these ideal conditions will occur due to variations in ruptured steam generator pressure, cooldown rates, and letdown flows. Consequently, the operator must continuously adjust RCS pressure and charging flow to control pressurizer and ruptured steam generator inventories. This step provides guidance for performing these actions in the form of a table. Figure 30 can be divided into four different regions which are characterized by pressurizer and ruptured steam generator level behavior. For primary pressures greater than the ruptured steam generator pressure, leakage into the steam generator will increase steam generator water level (LSG). Alternatively, water level will decrease for RCS pressures less than the ruptured steam generator pressure. Similarly, pressurizer level (LPRZR) will increase for RCS pressures less than equilibrium. This leads to the four regions illustrated in Figure 30. The steps one performs to stabilize the plant at the ideal, equilibrium conditions depend on the pressurizer inventory and ruptured steam generator water level behavior. For example, if pressurizer level is low, region II or region III must be entered to increase pressurizer level. This requires one to increase charging flow or decrease RCS pressure, as shown in Figure 30. The further into these regions, the more rapidly pressurizer level will increase. Of course, if pressurizer level is high, the opposite response would be necessary. However, the ruptured steam generator water level must also be considered. STEP DESCRIPTION TABLE FOR E-3Step29 If the steam generator water level is increasing, RCS pressure must be reduced to stop primary_to_secondary leakage. **If the steam generator water level is decreasing, primary pressure should be increased by energizing pressurizer heaters to minimize leakage into the RCS.** Note that in some cases, actions which address pressurizer level conflict with those which address steam generator level. For example, if steam generator level is increasing one must decrease RCS pressure. Since this will also increase pressurizer level, the pressurizer could fill with water if level is initially high. However, by reducing charging flow, pressurizer level will decrease. Since this will also decrease RCS pressure if heaters are not energized, steam generator level will also stabilize. Hence, for this situation the preferred action is to reduce charging flow.

038 Steam Generator Tube Rupture
EK1.02 Knowledge of the operational implications of the following concepts as they apply to the SGTR: **Leak rate vs. pressure drop**

Question Number: 47

Tier 1 Group 1

Importance Rating: RO 3.2

Technical Reference: EEP-3 and background documents page 83
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.5

Comments:
this meets the KA in that the question tests the concept of why PRZR pressure and temp affect the leak rate into or out of the Przr
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A D B B B B D D C C Scramble Range: A - D

QUESTIONS REPORT
for 75 RO Questions

Source :	BANK	Source if Bank:	MCGUIRE 2003
Cognitive Level:	LOWER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GTO	Previous 2 NRC exams:	NO

36. 039 A1.10 001

Given the following:

- Unit 1 is at 100% power.
- A Steam Generator Tube Leak has developed.
- R-15A, SJAE EXH, is approximately mid-scale.

Which ONE of the following describes:
(1) the R-15A indication that will be observed if SJAE Filtration is placed on service **and** (2) the R-15A indication that will be observed if a reactor trip occurs?

R-15A _______(1)_______ when SJAE Filtration is placed on service.

R-15A _______(2)_______ if the reactor trips.

A. (1) Trends down

(2) Remains stable

B. (1) Trends down

(2) Trends down

C. (1) Remains stable

(2) Remains stable

D✓ (1) Remains stable

(2) Trends down

A. Incorrect
(1) is incorrect, Plausible because it is true for R-15B & R-15C which are both downstream of the SJAE Filtration system (which is normally bypassed)
(2) is incorrect. but plausible because the reactor trip does not isolate or stop the SG Tube leak, and may not consider the d/p across the tube is proportional to leak flow.

B. Incorrect
(1) is incorrect, Plausible because it is true for R-15B & R-15C which are both downstream of the SJAE Filtration system (which is normally bypassed)
(2) is correct.

C. Incorrect
(1) is correct.
(2) is incorrect, but plausible because the reactor trip does not isolate or stop the SG Tube leak, and may not consider the d/p across the tube is proportional to leak flow.

D. Correct. When the Steam Jet air ejector Filtration system is placed on service there is no change in the reading since the SJAE is upstream of the Filtration system. When the reactor trips, steam flow is decreased, steam pressure goes up, & d/p across SG U-tubes goes down. This causes tube leakage rate to decrease which causes R-15 indication to decrease.

039 A1.10 Main and Reheat Steam System (MRSS)
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MRSS controls including:
Air ejector PRM

Question Number: 17

Tier 2 Group 1

Importance Rating: 2.9

Technical Reference: RAD MONITORING LP OPS 52106D
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.5 / 45.5

Comments:

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9			
			Answer:	D C A C C B B A B D	Scramble Range: A - D		
Source :	NEW		Source if Bank:				
Cognitive Level:	HIGHER		Difficulty:				
Job Position:	RO		Plant:		FARLEY		
~~reviewed:~~	~~GO~~		~~Previous 2 NRC exams:~~		~~NO~~		

37. 041 A1.02 016

Given the following:

- Unit 1 reactor power is steady at 14%.
- Tavg is at 551° F.
- Rod control is in Manual.
- Turbine power is steady at 7%.
- Steam Dumps are open in the STM PRESS mode.
- PK-464, STM HDR PRESS, controller is in AUTO and set to control temperature at 551°F.

Which ONE of the following is the correct response of the steam dump system if PT-464, STM HDR PRESS, fails HIGH under these conditions?

Assume no operator action is taken.

A.✓ • All steam dumps will open and then close at P-12.

• PK-464 will shift to MANUAL.

B. • All steam dumps will open and then close at P-12.

• PK-464 will remain in AUTO.

C. • All steam dumps will open and then cycle at P-12.

• PK-464 will remain in AUTO.

D. • All steam dumps will open and then cycle at P-12.

• PK-464 will shift to MANUAL.

A. Correct-
The low-low T_{AVG} (P-12) block actuates when 2/3 T_{AVG} instruments indicate below 543°F.
It should be noted that the AUTO feature of PK-464 can be selected only under certain conditions. First, if the mode selector switch is in the T_{AVG} mode, PK-464 shifts to manual control. Secondly, if the low-low T_{AVG} signal (P-12) exists and the BYP INTLK position on both A and B Train STEAM DUMP INTERLOCK SWITCHES has not been selected, PK-464 will shift to manual control. By shifting to manual control, the output of the P+I portion of the controller is set to zero and thus prevents small pressure errors from being integrated into large controller output signals.

B. Incorrect- they do go closed at 543°F, and the block does reset at 545°F, however, the controller shifts to minimum and manual and the dumps do not cycle.

C. Incorrect- the steam dumps will go closed and shift to minimum and manual. They will not cycle.

D. Incorrect- the dumps do open and PK 464 will go to manual but the dumps will remain closed and not cycle to control Tavg.

041 Main Turbine Generator System
A1.02 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SDS controls including:
Steam pressure

Question Number: 36

Tier 2 Group 2

Importance Rating: 3.1

Technical Reference: SD LP OPS-52201G; AOP-100
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.5 / 45.5

Comments:
this meets the KA in that the failure of the stm pressure transmitter affects the dumps and RCS temp and the operator has to predict and monitor the RCS for these changes.

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

Source :	BANK	Source if Bank:	FARLEY
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

38. 054 AA2.05 001

The following plant conditions exist:

- Unit 1 is operating at 28% power.
- A Feed Water control malfunction has caused 1A SG NR level to reach 84%.

Assuming the plant responds AS DESIGNED, with NO operator action, which ONE of the following describes the current valve alignment?

Assume all valves were open prior to the event

	Main Feed Reg Valves FCV-478,488, 498	SGFP Discharge Valves MOV503A, B	Feedwater Isolation Valves MOV-3232A,B,C
A.	All open	All open	All shut
B.	All shut	All open	All shut
C.	All open	All shut	All open
D✓	All shut	All shut	All shut

A. Incorrect - The SGFP trip will cause the discharge valves to go closed. The FWI signal causes the FRV and bypasses to close.

B. Incorrect - SGFP discharge valves go shut due to the SGFP trip at 82% level.

C. Incorrect - FRVs close on FWIS and FWI valves close on SGFP trip.

D. Correct. See FSD, Student text, & SOP-21.0.
The FWI signal will cause the SGFPs to trip and the main turbine to trip. The SGFP trip will cause the discharge valves to go closed and the FW isolation valves to go closed. The FWI signal causes the FRV and bypasses to close.

FNP Units 1 & 2 REACTOR PROTECTION SYSTEM A-181007 [FSD]:

2-26 Rev. 10
2.7.1
4. Main Feedwater Isolation and Turbine Trip

The Main Feed Line Isolation is initiated to prevent excessive cooldown of the reactor or to lessen the severity of the transient overall. The following signals are utilized to initiate the Main

Feed line Isolation:
a. Safety injection
b. High-high steam generator water level (P-14) set at = 82% of narrow range steam generator span on 2/3 coincidence
c. Low Tavg; = 554°F in coincidence with reactor trip P-4

manual reset to clear.
(References 6.1.022, 6.4.007, 6.4.015, 6.7.012, 6.4.21)

7. High-High Steam Generator Water Level
If water level in a steam generator increases to 82% on 2/3 narrow range level instruments, the main turbine trips, the main feedwater pumps trip and main feedwater isolation signals are initiated. Tripping the turbine is a protective measure to ensure no damage occurs from moisture carry-over. Main feedwater is isolated so that no further water is added to the steam generator with the high high level to protect the primary side from excessive cooldown when safety injection is actuated. (References 6.1.022, 6.4.007, 6.4.014, 6.7.012)

SOP-21.0, CONDENSATE AND FEEDWATER SYSTEM

4.6.3 At approximately 2450 RPM, trip the SGFP.

4.6.4 Verify that the SGFP high and low pressure stop valves are closed.

NOTE: In the following step, annunciator KC3 should clear after approximately three minutes.

4.6.5 Verify that annunciator KC3, 1A OR 1B SGFP TRIPPED comes in.

4.6.6 Verify closed the 1A(1B) SGFP DISCH VALVE, N1N21V503A(B).

4.6.7 At the 1A(1B) SGFP Oil Test Station, depress the LOW LEVEL TEST push-button until the low level alarm light is illuminated.

CONDENSATE & FEEDWATER STUDENT TEXT, OPS-52104C, OPS-40201B

Main Feedwater Stop Valves (3232A, B, and C) (Figure 14)
A three-position handswitch (CLOSE/AUTO/OPEN, spring return to AUTO) on the MCB controls each motor-operated isolation valve. In AUTO, the valve automatically closes on a SGFP trip signal from both pumps. Valve position lights indicate above each switch.

APE 054 Loss of Main Feedwater -
AA2.05 Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW):
Status of MFW pumps, regulating and stop valves

Question Number: 48

Tier 1 Group 1

Importance Rating: RO 3.5

Technical Reference: FSD, Student text, & SOP-21.0.

Proposed references to be provided to applicants during examination: None
10 CFR Part 55 Content: 41.10

Comments:
original question did not meet KA. Replaced. This question meets the KA in that when the SGWL reached 82%, the SGFPs tripped. The operator has to determine the status of the

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

Source :	BANK	Source if Bank:	FARLEY
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GTO	Previous 2 NRC exams:	NO

39. 055 EA1.02 001

Given the following:

- A LOSS OF ALL AC POWER has occurred on Unit 1.
- VA2, 1B DG GEN FAULT TRIP, annunciator has come into alarm.
- The crew is at the step in ECP-0.0, Loss of All AC Power, to verify breakers for major loads OPEN.
- A Safety Injection occurs on Unit 1 at this time.

Which ONE of the following describes how the 2C DG will be started and the events that will take place or need to take place to energize the ESF equipment?

A. • Start 2C DG from EPB in **Mode 2** using the start pushbutton.

• The LOSP sequencer will run to start all ESF loads.

B. • Start 2C DG from EPB in **Mode 2** using the start pushbutton.

• ALL ESF loads will have to be manually aligned.

C. • Start 2C DG from EPB in **Mode 1** using the start pushbutton.

• The LOSP sequencer will run to start all ESF loads.

D✓ • Start 2C DG from EPB in **Mode 1** using the start pushbutton.

• ALL ESF loads will have to be manually aligned.

A. Incorrect. not mode 2 - ECP-0.0 step 5 directs starting 2C DG in MODE 1. the LOSP sequencer will **not** run due to the SI signal present. Using MODE 2 is plausible since all other DGs are started in Mode 2 in this condition IAW ECP-0. 2C DG is the only DG that is started in Mode 1. The sequencer is plausible since it would run if the SI signal was not present.

B. Incorrect. not mode 2. second part is correct.

C. Incorrect. first part is correct. Second part is NOT correct.

D. Correct. ECP-0.0 step 5 directs starting 2C DG in MODE 1. the LOSP sequencer will **not** run due to the SI signal present. see below.

The LOSP sequencer will not run per the note below in ECP-0 with an SI signal present.

ECP-0 note at step 5.2.1.5 RNO

NOTE: The LOSP sequencer should run when output breaker closes, if no SI signal is present. If an SI signal is present, neither sequencer will run and SI loads must be started manually.

5.2 Perform the following:

5.2.1 Perform 2C DG SBO start as follows.

5.2.1.1 Verify 2C DG MODE SELECTOR switch in MODE 1.

5.2.1.3 WHEN load shed verified,THEN depress 2C DG DIESEL START pushbutton.

055 EA1.02 Station Blackout
Ability to operate and monitor the following as they apply to a Station Blackout:
Manual ED/G start

Question Number: 49

Tier 1 Group 1

Importance Rating: RO 4.3
Technical Reference: FNP-1-ECP-0.0
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.10

Comments:
replaced this question since the original did not meet the KA. This demonstrates the ability to start and monitor a manual start of the 2C DG and the subsequent actions to energize equipment in that train after the start.

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

QUESTIONS REPORT
for 75 RO Questions

Source :	NEW	Source if Bank:	
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GTO	Previous 2 NRC exams:	NO

40. 056 A2.04 010

Given the following:

- Unit 2 is at 80% power ramping to 100%.
- 2C Condensate Pump has tripped.
- Annunciator KB4, SGFP SUCT PRESS LO, came into alarm 35 seconds ago.
- PR-4039, SGFP Suction PRESS recorder, indicates SGFP pressure is 280 psig and is decreasing.

Which ONE of the following is the expected result of this condition and action required per AOP-13.0, Condensate and Feedwater Malfunction?

A. • The standby condensate pump **should** have AUTO started;

• Trip the reactor and enter EEP-0, Reactor Trip or Safety Injection.

B✓ • The standby condensate pump **should** have AUTO started;

• Verify the standby condensate pump started and if suction pressure is still falling, then reduce load rapidly IAW AOP-17, Rapid Load Reduction.

C. • The standby condensate pump **should NOT** have AUTO started;

• Trip the reactor and enter EEP-0, Reactor Trip or Safety Injection.

D. • The standby condensate pump **should NOT** have AUTO started;

• Verify the standby condensate pump started and if suction pressure is still falling, then reduce load rapidly IAW AOP-17, Rapid Load Reduction.

The condensate pumps have 2 auto starts, one on the trip of the other pump and one for suction pressure <275 psig for >10 sec. In this question the auto start for suction pressure is not met but the other one is. This has been a high miss question since most do not think of the autostart for the tripped pump.

A - Incorrect; The stby condensate pump should start immediately when the other condensate pump trips (if the stby pump is in AUTO, which is the normal alignment at 80% power). Plausible because the low suction pressure auto start of < 275 psig for > 10 seconds has not yet been met. Tripping the reactor is not required unless approaching trip criteria or if BOTH SGFPs are tripped and this has not happened.

B - Correct; The standby condensate pump SHOULD auto start immediately when the other condensate pump trips. With suction pressure dropping, AOP-13 directs verifying stby pump started prior to 275 psig decreasing, If pressure continues to drop, rapidly ramp down IAW AOP-17, Rapid Load Reduction.

C - Incorrect; first part is NOT true: condensate pump should have auto started on the trip of the other pump.
second part is not correct for this situation. It is plausible in that if the candidate thought both SGFPs tripped due to the alarm being in for >275 psig, then this would be correct.

D - Incorrect; first part is NOT true: condensate pump should have auto started on the trip of the other pump.

Second part is incorrect for this condition also, but plausible because the ramp at ≤ 5 MW/MIN is incorrect for a condensate pump but plausible since it is correct for a HDT pump trip also covered by AOP-13.

At 275 psig falling the standby condensate pump will start after 10 sec. IF suction pressure is NOT greater than 275 psig within 30 sec, THEN the SGFP's will trip. This could result in a reactor trip.

056 A2.04 Condensate System

Ability to (a) predict the impacts of the following malfunctions or operations on the Condensate System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those mal-functions or operations:

Loss of condensate pumps

Question Number: 37

Tier 2 Group 2

Importance Rating: 2.6

Technical Reference: AOP-13 & ARP 1.10 KB4

Proposed references to be provided to applicants during examination: None

Learning Objective:

10 CFR Part 55 Content: 41.10

Comments:

significantly modified the distracters to make sure this does not answer 059 A1.03 and this is also a different part of AOP-13.

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

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

Source :	MODIFIED	Source if Bank:	
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

41. 057 AK3.01 001

Given the following:

- Unit 1 is at 14% power and ramping up in preparation for rolling the main turbine.
- The bypass feed regulating valves are in service, and SG level is being maintained at 65%.
- 1G 4160 V Bus has been de-energized due to DG15, 1B S/U XFMR to 4160V Bus 1G, tripping open.
- The 1B DG has re-energized the 'B' Train emergency buses.

Which ONE of the following describes the correct operator actions IAW AOP-5.0, Loss of A or B Train Electrical power?

A. Trip the reactor and restore the 1G 4160 V bus to the grid.

B. Shut down the reactor and place the unit in Mode 3.

C.✓ Stabilize the plant and restore the 1G 4160 V bus to the grid.

D. Verify the reactor tripped and stabilize the unit in Mode 3.

AOP-5.0 Version 24

A. incorrect - AOP-5.0 no longer directs tripping the reactor.

B. Incorrect- Ramping off line is not required in AOP-5.

C. correct - IAW AOP-5, step 14 the long term status is to maintain the reactor stable and then restore the grid, then continue with whatever procedure the operator was in at the time of the problem. This was changed a few years ago to prevent an unnecessary reactor trip for this condition.

D. Incorrect - an automatic reactor trip will not occur.

APE 057 Loss of Vital AC Instrument Bus -
AK3.01 Loss of Vital AC Instrument Bus , Knowledge of the **reasons** for the following responses as they apply to the (Loss of Vital AC Instrument Bus):
Actions contained in the EOP for loss of vital AC electrical instrument bus.

Question Number: 56

Tier 1 Group 1

Importance Rating: 4.1

Technical Reference: AOP-5.0
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.10

Comments:
this KA is for a loss of a vital bus which is an emergency bus and/or panel at FNP. This meets the KA in that there is a loss of the vital bus and the procedure guidance of AOP-5 is followed. The reason for those actions is implied and agreed upon as part of the actions that would be done. . FNP does not have a loss of a vital panel procedure and that issue is vaguely discussed in an ARP in which it says if a vital panel has been lost, then recover from it when the event is over by doing the following that apply.

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

Source :	MODIFIED	Source if Bank:	FARLEY
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

42. 058 G2.1.32 002

Given the following on Unit 1:

- A loss of "A" Train Auxiliary Building Battery bus has occurred.
- The crew is performing AOP-29.1, Plant Stabilization in Hot Standby and Cooldown Without "A" Train AC or DC Power.
- The RCS temperature is 547°F and pressure is being maintained 2220 psig.
- Seal injection flow has been lost to all three Reactor Coolant Pumps (RCPs).
- LB3, RCP THRM BARR ISO HV-3184 AIR PRESS LO, annunciator has come into alarm.
- All three RCPs have been tripped.

Which ONE of the following correctly describes one of the actions required by AOP-29.1 for the conditions given above, **and** the reason for performing that action?

A.✓ Isolate seal injection flow to all RCPs to prevent potential RCP seal damage.

B. Isolate seal injection flow to all RCPs to prevent a potential radioactive release to the Auxiliary Building from occurring.

C. Isolate seal return flow to all RCPs to prevent potential RCP seal damage.

D. Isolate seal return flow to all RCPs to prevent potential thermal barrier heat exchanger damage.

A is correct. CAUTION (for step 5): CAUTION: To prevent potential seal damage, neither seal injection flow nor CCW flow to the thermal barrier shall be re-established to an RCP which has lost both seal injection and CCW cooling.
the background documents for a loss of all ac describe what happens on a loss of chg and CCW to a seal. The seal injection is isolated to prevent potential seal damage.

B is incorrect. This is a correct action but the reason is for the seal return flow.

C. is incorrect. see explanation below.

D is incorrect. seal return is isolated but not to prevent seal damage. It is isolated for several reasons:

<u>Isolating the seal return line prevents seal leakage from filling the volume control tank (VCT) (via seal return relief valve outside containment) and subsequent transfer to other auxiliary building holdup tanks (via VCT relief valve) with the potential for radioactive release within the auxiliary building.</u> Such a release, without auxiliary building ventilation available, could limit personnel access for local operations.

Isolating the seal return line also enables pressure in the number 1 seal leakoff line to increase up to the relief valve setpoint of 150 psig. Maintaining a backpressure in the seal leakoff line of at least 150 psig enables development of high pressure in the number 1 seal leakoff cavity with a steady-state seal leakage rate established due to the self-limiting leakage characteristic of

the number 1 seal. Under these conditions, with the number 1 seal functioning as expected and the number 2 seal remaining closed, the expected leakage flow rate is 21.1 gpm/pump.
This is consistent with the steady state pressure distribution and seal leakage determined in the WCAP-10541 analysis and used in the latest RCP seal leakage PRA model in WCAP-15603

3.0 IF ONE of the following conditions occur at anytime during the event, AND cannot be readily restored.

[] A total loss of RCP seal cooling as indicated by loss of seal injection and loss of CCW to the Thermal Barrier Heat Exchanger.

OR

[] A total loss of the operating train of charging without the ability to quickly restore the redundant train, and RCP seal leakoff before the loss was less than 2.5 gpm per pump.

background document for ECP-0

Purpose: To isolate the RCP seals

Basis: This step groups three actions, with different purposes, aimed at isolating the RCP seals. The actions are grouped since all require an auxiliary operator, dispatched from the control room, to locally close containment isolation valves (the reference plant utilizes motor operated valves for the RCP seal return, RCP thermal barrier CCW return lines and RCP seal injection lines). This grouping assumes that the subject valves are located in the same penetration room area and that they are accessible. Concurrent with dispatching the auxiliary operator, the control room operator should place the valve switches for the motor operated valves in the closed position so that the valves remain closed when ac power is restored.

. **Isolating the RCP seal injection lines prepares the plant for recovery while protecting the RCPs from seal and shaft damage that may occur when a charging/SI pump is started as part of the recovery. With the RCP seal STEP DESCRIPTION TABLE FOR ECA-0.0Step 8 injection lines isolated, a charging/SI pump can be started in the normal charging mode without concern for cold seal injection flow thermally shocking the RCPs. Seal injection can subsequently by established to the RCP consistent with appropriate plant specific procedures. Isolating the RCP thermal barrier CCW return outside containment isolation valve prepares the plant for recovery while protecting the CCW system from steam formation due to RCP thermal barrier heating.**
Following the loss of all ac power, hot reactor coolant will gradually replace the normally cool seal injection water in the RCP seal area.

APE 58 G2.1.32 Loss of DC Power
Conduct of Operations: Ability to explain and apply all system limits and precautions.

Question Number: 50

Tier 1 Group 1

Importance Rating: 3.4

Technical Reference: AOP-29.1
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.7/10

Comments:
This meets the KA in that under the conditions given, a loss of DC power, a system caution to isolate CCW and Seal injection to the RCP seal to prevent seal damage upon re-initiation of either Seal injection or CCW flow which could cause a SBLOCA from the RCP seal of 300 gpm. This question asks the operator to explain why these actions are taken.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A A A A A A A A A A Items Not Scrambled

Source :	NEW	Source if Bank:	
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

43. 059 A1.03 001

Given the following:

- Unit 1 is at 100% power.
- 1A SGFP has tripped.
- The crew has entered AOP-13.0, Condensate and Feedwater Malfunction.
- Emergency Boration is in progress.

Which ONE of the following describes the subsequent restrictions on operation of the unit in accordance with AOP-13.0?

(1) Load must be reduced to __________ ;
(2) The reactor **must** be tripped immediately if ________________________

A. (1) ≤ 730 MWe;

(2) SG NR levels cannot be maintained above the minimum value specified.

B. (1) ≤ 730 MWe;

(2) FE1, CONT ROD BANK POSITION LO, annunciator comes into alarm with Tavg at 577°F.

C✓ (1) ≤ 540 MWe;

(2) SG NR levels cannot be maintained above the minimum value specified.

D. (1) ≤ 540 MWe;

(2) FE1, CONT ROD BANK POSITION LO, annunciator comes into alarm with Tavg at 577°F.

A. is incorrect. Due to load setpoint, although plausible because this is the value at which the decrease load button is released when DEH manual load reduction is used per the RNO column.

B is incorrect. Due to load setpoint, although plausible because this is the value at which the decrease load button is released when DEH manual load reduction is used per the RNO column.
TAVG 541°F - 580°F 577 F is still within the band to not trip.

AOP-13 has a step to evaluate the plant per the below:
1.14 IF the Team is NOT confident that a parameter is being restored, THEN trip the reactor and go to FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION.
one of the parameters checked is FE2, not FE1. Since the team is emergency borating per the procedure, it is unlikely this alarm would come in, but if it did the reactor is not required to be immediately tripped. If FE2 came into alarm, then action here would be to place rods in manual and with the emergency boration in progress, evaluate the plant, not immediately trip the reactor.

C. Correct. AOP-13.0 directs load to be reduced to 540 MWe, and SG NR levels must remain above 28%.

AOP-13 step 1.8
IF SG narrow range levels NOT maintained greater than 28%, THEN trip the reactor and go to FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION.

D is incorrect. see B above for second part.
TAVG 541°F - 580°F 577 F is still within the band to not trip.

OPERATOR ACTION for FE1

1. Check indications and determine that actual control bank rod position is at low insertion limit.
1.1 Click on Rod Supervision button on Applications Menu.
1.2 Click on Rod Insertion Limits button.
1.3 Determine if low insertion limit exceeded.
2. IF reactor coolant system dilution is in progress, THEN stop dilution.
3. IF a plant transient is in progress, THEN place the turbine load on "HOLD".
4. Refer to FNP-1-UOP-3.1, POWER OPERATIONS.
5. Borate the Control Bank "OUT" as necessary using the Boron Addition Nomographs. {**CMT 0008900**}
6. Refer to the Technical Specifications section on Reactivity Control.

059 A1.03 Main Feedwater System
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MFW controls including:
Power level restrictions for operation of MFW pumps and valves.

Question Number: 18

Tier 2 Group 1

Importance Rating: 2.7
Technical Reference: AOP-13.0, FE2 and KB4
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.7

Comments: good match for KA since it tests power level restrictions with 1 SGFP and monitoring levels to prevent exceeding design limits and the actions required.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	C C C A C C D A C C	Scramble Range: A - D
Source :	NEW		Source if Bank:		
Cognitive Level:	HIGHER		Difficulty:		
Job Position:	RO		Plant:	FARLEY	
reviewed:	GO		Previous 2 NRC exams:	NO	

44. 061 K1.07 001

Which ONE of the following describes the Service Water Train normally aligned to the TDAFW Pump for Emergency Makeup, **and** if the TDAFW Pump is running and aligned to the CST, the **MINIMUM** time available to swap to the emergency supply when JD4 and JE4, CST LVL LO-LO A and B TRN, alarms are received?

(1) Service Water Train normally aligned to ____________ .

(2) The minimum time available to swap to the emergency supply after receiving the CST LO-LO level alarm is _________________ .

A.✓ (1) A Train
(2) 20 minutes

B. (1) A Train
(2) 2 hours

C. (1) B Train
(2) 20 minutes

D. (1) B Train
(2) 2 hours

A is correct. The CST LO-LO level alarm received means that at least 20 minutes of normal supply remains.
65,300 Gallons (5'3") is when the LO-LO comes in.

B is incorrect. 2 hours is the time that 150,000 gallons of CST will maintain Hot Standby.

C is incorrect. Wrong train, plausible because B train can be aligned and would be if A train power or SW was unavailable.

D is incorrect. Wrong train and time.

SOP-22, AFW section 4.7
4.7 Aligning Service Water to the AFW System. This shows how to swap SW to B train and how to align it to A Train. The initial valve line up per the SOP also shows it is aligned to A Train and D-175007 shows the normal line up is to A Train.

FSAR 6.55 instrumentation after recieving low level alarm setpoint

061 K1.07 Auxiliary Emergency Feedwater (AFW) System
Knowledge of the physical connections and/or cause-effect relationships between the AFW and the following systems:
Emergency water source.

Question Number: 19

Tier 2 Group 1

Importance Rating: 3.6

Technical Reference: LP 52102H AFW D-175007 and SOP-22 AFW

Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.5

Comments:

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

Source :	NEW	Source if Bank:	
Cognitive Level:	LOWER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GTO	Previous 2 NRC exams:	NO

45. 062 G2.1.23 001

Given the following:

- Unit 1 is operating at 38% power with 'A' Train on service.
- The 1C Service Water Pump is tagged out for motor replacement.

A fire is reported on the 1K 4160 volt bus and the plant operators de-energize 1K 4160 volt bus. The following conditions exist:

- CCW FROM CCW HX TEMP, TI-3042C, is elevated slightly and rising slowly.
- PI-3001A, A Train SW TO CCW HX HDR PRESS, reads 42 psig.
- PI-3001B, B Train SW TO CCW HX HDR PRESS, reads 75 psig.

Which one of the following actions are **required** IAW AOP-10.0, Loss of Service Water?

A. Trip the reactor and perform EEP-0, Reactor Trip or Safety Injection.

B. Reduce power to less than 35%, then trip the Main Turbine and refer to AOP-3.0, Turbine Trip below the P-9 Setpoint.

C✓ Start 1A CCW pump and 1C charging pump, secure 1A charging pump, and swap on service trains of CCW.

D. Start 1C CCW pump and 1A charging pump, secure 1C charging pump, and swap on service trains of CCW.

A. Incorrect - a rx trip is not required at this time. AOP-10.0 Step 4.2.2.2 RNO and step 6.3 has the crew trip the reactor if one train of SW does not have at least 60 psig. Since one train is available and operating, a reactor trip is not called for.

B. Incorrect - removing the main turbine from service would be an option if both trains of SW were less than 60 psig per the RNO step 6 of AOP-10 if power was less than 35%. Being so close to 35% in the stem and temperatures are elevated slightly makes this plausible since it would make sense to ramp to below 35% and remove the turbine from service vs trip the rx.

C. Correct - Start a CCW pump and charging pump in the nonaffected train, secure affected train charging pump, and swap on service trains of CCW.
This is the correct response because enough time is allowed before RCP temps increase to 195°F to mitigate the loss of SW and prevent the need to trip the reactor. It will take time to heat up the On service train of CCW, and AOP-10.0 takes that into account.

D. Incorrect - Swapping on-service trains is partially correct, but the unit is not ramped to 35% power and the main turbine tripped.

APE 062 Loss of Nuclear Service Water -

G2.1.23 Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Question Number: 51

Tier 1 Group 1

Importance Rating: 3.9

Technical Reference: AOP-10
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.10

Comments:
this meets the KA in that the ability to perform AOP-10 is demonstrated during mode 1 at a low power level.

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

Source :	BANK	Source if Bank:	FARLEY
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

46. 062 K2.01 001

Given the following conditions on Unit 2:

- 'A' train is the "On Service" train.
- 2A Charging Pump breaker has been racked out for maintenance.
- 2G 4160 V Bus has been de-energized due to a fault.

Which one of the following states the ECCS pumps that will have power based on current conditions?

A.✓ 2B Charging Pump, 2A RHR Pump.

B. 2B Charging Pump, 2B RHR Pump.

C. 2C Charging Pump, 2A RHR Pump.

D. 2C Charging Pump, 2B RHR Pump.

A. correct, both powered from 2F

B. Incorrect, 2B chg powered from 2F. 2B RHR powered from 2G

C. Incorrect, 2C chg powered from 2G. 2A RHR powered from 2F

D. Incorrect, both powered from 2G

062 K2.01 A.C. Electrical Distribution
Knowledge of bus power supplies to the following:
Major system loads

Question Number: 20

Tier 2 Group 1

Importance Rating: 3.3

Technical Reference OPS 52103B, Unit 1 Equipment Load List A506250
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.5

Comments:

KA match in that these are major loads on the emergency busses.

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

QUESTIONS REPORT
for 75 RO Questions

Source :	MODIFIED	Source if Bank:	FARLEY
Cognitive Level:	LOWER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GTO	Previous 2 NRC exams:	NO

47. 063 A3.01 001

Unit 1 is at 100% power with the following conditions:

- 1A Battery Charger is on service.
- EM personnel are doing preventative maintenance on the 1A battery.

The following indications are received:

- The UNIT 1 AUX BLDG DC BUS - A TRN GROUND DET white light comes on momentarily and then goes OFF.

• Then the following alarms are received:
  - WC2, 1A 125V DC BUS UV OR GND
  - WC3, 1A 125V DC BUS BATT BKR 72-LA05 TRIPPED
  - THEN WC2 clears.

Which ONE of the following describes the status of the indications on the EPB for the 1A DC BUS and the 1A and 1B Inverters?

1A DC BUS VOLTAGE reads approximately ______(1)______

1A and 1B INVERTER AMPERES are reading approximately ______(2)______

A. (1) 0 DC VOLTS.

(2) 25 amps and being powered from the bypass source.

B. (1) 0 DC VOLTS.

(2) 0 amps and being powered from the normal source.

C. (1) 125 DC VOLTS.

(2) 0 amps and being powered from the bypass source.

D✓ (1) 125 DC VOLTS.

(2) 25 amps and being powered from the normal source.

explanation

When the Battery output breaker is opened, LA-05, WC3 will come into alarm due to the b contact from breaker LA05. WC2 shows either a low voltage condition or a ground. In this case it would be a ground.
The battery output breaker has opened due to a ground on the battery and when it opens WC2 clears. The annunciators provide indication that the breaker opened and the white light provides indication of the ground. For this set of circumstances, the battery is no longer aligned to the bus and the battery charger is carrying the load. The indications will remain normal and the inverters will have normal indications. The inverters will not swap to the bypass source and will still be powered from the BC.

A. Incorrect. 0 DC volts on the 1A DC bus indicates the bus is de-energized. The bus still has power from the Batt. chger. The inverters will be powered from the BC or the normal supply and will indicate 25 amps. If it were to swap to the bypass source, it would still have amp readings, but if the manual bypass switch were to be placed in the bypass position, then the amps would be 0 amps.

B. Incorrect. 0 is not correct for both. Normal is correct.

C. Incorrect - 125 is correct. 0 is not correct and bypass is not correct.

D. Correct. 125 is correct and 25 is correct from the normal source.

063 D.C. Electrical Distribution
A3.01 Ability to monitor automatic operation of the dc electrical system, including:
Meters, annunciators, dials, recorders, and indicating lights

Question Number: 21

Tier 2 Group 1

Importance Rating: 2.7

Technical Reference: ARP WC2, WC3 and D177082
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.7

Comments: This was replaced to fully meet the KA.

It meets the KA in that it tests the ability to determine the proper readings on the EPB for an abnormal condition based on the indications and alarms received (white light and annunciators). The automatic portion of the KA is the breaker opening on an overcurrent condition.

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

QUESTIONS REPORT
for 75 RO Questions

Source :	NEW	Source if Bank:	
Cognitive Level:	LOWER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

48. 063 K4.04 001

Given the following:

Unit 2 is at 100% power. Reactor Trip Breakers A (RTA) and B (RTB) are closed, Reactor Trip Bypass Breakers are open.

- 125V DC distribution panel breaker 2B-16, "A" Reactor Trip switchgear control power to Bypass breaker and Reactor Trip breaker, has tripped open.

Which one of the following statements correctly describes how a loss of DC to the A Train reactor trip switchgear would effect the operation of Reactor Trip Breaker A?

A. RTA will immediately trip open.

B.✓ RTA will remain closed and will still open from either a manual or automatic signal.

C. RTA will remain closed and will **not** open from either a manual or automatic signal.

D. RTA will remain closed and will **not** open from a manual reactor trip signal; an automatic trip will still open the breaker.

A. Incorrect - This will not cause a direct Rx trip due to SSPS still powered up and the only loss is DC to the STC Rx Trip.

B. Correct- **RTA would still open from either a manual or automatic signal.**
125 V DC bus A allows the RTA shunt trip coil to energize on a Rx trip. With no DC available, the Rx trip Brkers will still open on a signal from SSPS A train by de-energizing the UV coil. With the loss of the DC, no RT brker will open immediately because SSPS is still energized and the breaker does not have a trip signal.

C. Incorrect - It will open on both.

D. Incorrect - manual Rx trips cause the STC to be energized and the UV coil to be de-energized. A Rx trip will still operate on a loss of DC due to the UV coil.
according to Table 6 of OPS-52103C, 125V DC distribution panel feeds to Rx trip swgr #1. According to the load list page F-51/52 LA-13 feeds 125V DC A Rx trip swgr control power to Byp brker & Rx trip bker. (2B-16)

FSD A-181007
2.2.18 The RPS shall be designed for fail safe operation. Loss of power to the protection logic or rod control system shall trip the reactor. The only exception to fail safe criteria shall be containment spray (H1-3) and shunt trip attachment for reactor trips. (References 6.1.022, 6.7.031) page 3-10

The Shunt Trip Attachment coil shall operate on 125 Vdc and function as a backup for the undervoltage trip device.

From the load list, this is the breaker designation and the nomenclature for this panel
125V DC distribution panel breaker 2B-16, "A" Reactor Trip switchgear control power to Bypass breaker and Reactor Trip breaker,

063 K4.04 D.C. Electrical Distribution

Knowledge of DC electrical system design feature(s) and/or interlock(s) which provide for the following:

Trips

Question Number: 22

Tier 2 Group 1

Importance Rating: 2.6

Technical Reference: Electrical Dist FSD, DC LP; SEQ LP OPS-52103F reactor protection lesson plan and FSD A-181007

Proposed references to be provided to applicants during examination: None

Learning Objective:

10 CFR Part 55 Content: 41.7

Comments: replaced this question to meet the KA. This is a backward way to meet the KA in that DC power is provided to the Rx trip breakers to cause a rx trip via the shunt trip coil. A loss of the DC will not allow DC to provide for a rx trip so one will not occur from the STC but will occur from SSPS via the UV coil. This requires knowledge of the DC bus supply to the RT breakers and what the loss means to the operator.

MCS	Time: 1	Points:	1.00	Version:	0 1 2 3 4 5 6 7 8 9		
				Answer:	B C D C C A A C D B	Scramble Range: A - D	
Source :		BANK			Source if Bank:	FARLEY	
Cognitive Level:		LOWER			Difficulty:		
Job Position:		RO			Plant:	FARLEY	
reviewed:		GO			Previous 2 NRC exams:	NO	

49. 064 K6.08 001

Given the following:

- Hurricane force winds have caused damage in the HVSYD and a dual unit LOSP.
- Long term Emergency Diesel Generator operation is anticipated.
- It is not known how soon the on-site Diesel supplies can be replenished
- The Fuel Oil Storage Tank (FOST) for 1-2A DG is **empty.**
- The Fuel Oil Day Tank for 1-2A DG is **full.**

Which ONE of the following describes the maximum time available for 1-2A DG to continue to run at full load?

A. 2 hours

B.✓ 4 hours

C. 8 hours

D. 24 hours

A. Incorrect. The day tank is sized to supply 4 hours at full load, 3 hours at minimum TS level.

B.Correct.

C. Incorrect. Plausible because it is close to the amount of time and symmetrical as a distractor

D. Incorrect. Plausible due to amount of time and also name of tank (day tank)
Lesson plan 52101i
Diesel Fuel Oil Storage System
Refer to Figure 14. Each diesel generator is connected to a shared fuel oil storage and transfer system, which consists of five storage tanks, two fuel oil transfer pumps per storage tank, a day tank for each diesel generator, and interconnecting piping and valves. The diesel fuel oil storage system has a total of five 40,000-gallon storage tanks, two 1000-gallon day tanks (for the little diesels), three 1325-gallon day tanks (for the big diesels), and two redundant capacity fuel oil transfer pumps per storage tank. The storage tanks are designed with sufficient fuel oil storage capacity to supply the minimum number of diesels required for seven days of operation with ten percent excess for testing using the deliverable capacity of four of the five storage tanks. The electrical distribution system supplies Class 1E, 120V AC power to each storage tank level transmitter.
Two fuel oil transfer pumps are mounted on each storage tank. The pumps are motordriven, vertical, submersible, wet pit-type pumps. The capacity of each pump is in excess of the amount required to simultaneously supply the diesel generator full load fuel requirements and fill the associated day tank. One pump automatically maintains the required day tank level, and the other is strictly manual. When full, the day tanks provide sufficient storage for four hours of full load operation. Tech Spec required minimum day tank volume ensures sufficient fuel oil is available to allow 3 hours of full load operation of the respective diesel generator.

FSD 3.6.3.2 A-181005
Functional Requirements
Each diesel engine has an individual fuel oil day tank. **When full, the fuel oil day tanks have sufficient fuel oil volume to supply their respective diesel for 4 hours at continuous rated load.** However, the Technical Specifications state that the minimum amount of fuel oil is 900 gallons each in the day tanks of the 1-2A, 1B and 2B diesel engines and 700 gallons each in the day tanks of the 1C and 2C diesel engines. **This capacity is sufficient to ensure that each day tank can supply its corresponding diesel with at least three hours of fuel at continuous rated load** (References 6.1.006, 6.1.014, 6.3.012, 6.5.014, 6.7.009 and 6.7.030).

QUESTIONS REPORT
for 75 RO Questions

064 K6.08 Emergency Diesel Generators
Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system:
Fuel oil storage tanks

Question Number: 23

Tier 2 Group 1

Importance Rating: 3.2

Technical Reference: TS 3.8.3.a, f; DG LP OPS-52102I
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.10/43.2

Comments: good question that tests length of time a DG can run with the day tank full and no other fuel available. This is the loss of portion of the KA and the affect as it relates to the DGs.

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

Source :	NEW	Source if Bank:	
Cognitive Level:	LOWER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GTO	Previous 2 NRC exams:	NO

50. 065 AK3.03 003

Given the following:

- Unit 2 is at 100% power.
- N2P19HV3825, Instrument Air to Penetration Room valve, has closed and **cannot** be opened.

Which ONE of the following will occur with no operator action taken?

A. Pressurizer pressure and level will remain stable.

B. Pressurizer pressure will increase until the PORVs lift.

C✓ Pressurizer pressure and level will increase until a reactor trip occurs.

D. Pressurizer level will decrease until letdown isolates and backup heaters turn off, then increase until a reactor trip occurs.

HV 3825 supplies air to AB loads. FCV-122 will go full open, letdown will secure sprays, PORVs will go closed and stay closed.

A. Incorrect- Due to the loss of air pressure Przr pressure will be rising and level will be rising due to FCV-122 and loss of letdown.

B. Incorrect- there is no air to the PORVs

C. Correct- due to no air, charging will be at a max rate and letdown will secure. Pressure will also be rising and a Rx trip on high pressure will occur.

D. Incorrect- Level will actually rise.

APE 065 Loss of Instrument Air

AK3.03 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

Question Number: 52

Tier 1 Group 1

Importance Rating: 2.9

Technical Reference: AOP-6.0
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.7

Comments: changed out the question completely to meet the KA.

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

Source :	MODIFIED	Source if Bank:	FARLEY
Cognitive Level:	LOWER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

51. 068 G2.1.20 002

Given the following:

- Both Units are operating at 100% power.
- Toxic gas has made the Control Room inaccessible.
- AOP-28.0, Control Room Inaccessibility, has been implemented.

Which ONE of the following are the **minimum** and **complete** actions required IAW AOP-28.0 before leaving the control room?

A. Trip the reactor and trip the main turbine ONLY.

B. Trip the reactor, trip the main turbine, trip both SGFPs, and sound the plant emergency alarm.

C.✓ Trip the reactor, trip the main turbine, verify at least one train of 4160 V ESF buses are energized, and sound the plant emergency alarm.

D. Trip the reactor, trip the main turbine, verify at least one train of 4160 V ESF buses are energized and actuate a safety injection.

A. Incorrect. These are the Immediate actions of FRP-S.1, not IAW AOP-28. These actions are **not complete** IAW AOP-28.

B. Incorrect. Both SGFPs are not required to be tripped at this point in the procedure. It is done locally at step 15. Sounding the plant emergency alarm is the only place this can be done and is in the first four steps.

C. Correct. These are the strategies addressed by the steps in the procedure.
AOP-28.0 rev 11 actions:
1.0 Verify reactor tripped.
2.0 Verify the turbine tripped.
3.0 Verify at least one train of 4160 V ESF buses energized.
4.0 Perform the following.
4.1 Direct Operation's personnel to man the Hot Shutdown Panels.
4.2 **Actuate the plant emergency alarm.**
4.3 Announce "Main control room evacuation. Report to your designated assembly areas."
4.4 Verify control room and C.A.S. evacuated.
4.5 Notify appropriate support groups to report to the Hot Shutdown ?? Panels.
4.6 Direct Security to station personnel at each control room door to prevent entry.

D. Incorrect. These are almost the first 4 steps of E-0, an SI is actuated if one is required. however, if no signal is calling for an SI, it would not be conservative to actuate an SI, and might be considered a good idea, but not IAW AOP-28. Checking the SI actuated is not required.

APE068 Control Room Evacuation

G2.1.20 Conduct of Operations:
Ability to execute procedure steps.

Question Number: 65

Tier 1 Group 2

Importance Rating: 4.3
Technical Reference: AOP-28
Proposed references to be provided to applicants during examination: None
Learning Objective: OPS 52533M02
10 CFR Part 55 Content: 41.10

Comments:
This is a new KA replaced per FJE. This question asks the operator the actions required that an RO should know to evacuate the control room. These should be committed to memory and if not properly executed would cause operational concerns. This demonstrates the ability to execute procedural steps in AOP-28.0.

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

Source :	MODIFIED	Source if Bank:	FARLEY
Cognitive Level:	LOWER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	YES

52. 069 AA2.01 007

Which ONE of the following conditions represents a loss of containment integrity and would cause entry into Technical Specification 3.6.1, Containment?

A. Mode 3 and one of the Personnel Airlock doors will not close.

B.✓ Mode 4 and Integrated Leak Rate test determines that leakage is not within limits.

C. Mode 5 and it is discovered that the Phase 'B' isolation valve for CCW to the RCPs, will not close.

D. Mode 6 and the Equipment Hatch is held in place by 4 bolts.

Containment integrity

A is incorrect. Both doors inop would be a loss of Containment Integrity, this is just an inop of one of the doors in the Personnel Airlock Plausible becasue one of two series valves makes containment integrity LCO not met.

B is correct. Surveillance requires ILRT to be within limits for Containment Integrity to be set.

C is incorrect. because Containment Integrity is not required in Mode 5, plausible because the valve is part of a containment penetration that would affect integrity in modes 1-4.

D is incorrect. 4 bolts meets the minimum requirement for Containment Closure in Mode 6, but not containment integrity in the modes that containment integrity is required.

APE 069 AA2.01 Loss of Containment Integrity -
Ability to determine and interpret the following as they apply to the Loss of Containment Integrity:
Loss of containment integrity

Question Number: 59

Tier 1 Group 2

Importance Rating: 3.7
Technical Reference: TS section 3.6
Proposed references to be provided to applicants during examination: None
Learning Objective: OPS-52102A-1
10 CFR Part 55 Content: 43.2/41.10

Comments: meets the KA in that it tests the ability to determine IF Ctmt integrity is met in different modes IAW Tech Specs.

Mode applicability (1-4) & one hour or less tech specs (one or more air locks with one door inoperable) are RO Knowledge and this question meets K/A for ROs.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B B B B B B B B B B Items Not Scrambled

Source :	BANK	Source if Bank:	HARRIS
Cognitive Level:	LOWER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GTO	Previous 2 NRC exams:	NO

53. 072 G2.1.2 001

Which ONE of the following Area Radiation Monitors requires entry to a Technical Specification Action Statement if it is declared INOPERABLE?

A. R-1A, Control Room Area Radiation (Unit 1)

B. R-2, Containment Area Radiation

C. R-4, Charging Pump Area Radiation

D✓ R-27A, Containment Area Radiation (High Range)

A. Incorrect. Plausible because this is an important radiation monitor indicating the habitability of the Control Room, but it is not in TS.

B. Incorrect. Plausible because this is an important radiation monitor indicating the abnormally high radiation level in containment. This is used in the emergency procedures for diagnosis of a LOCA, but it is not in TS.

C. Incorrect. Plausible because this is an important radiation monitor indicating radiation levels in the Charging Room area. This is used in the emergency procedures for diagnosis of a LOCA outside Containment, but it is not in TS.

D is correct. This radiation monitor is in TS in 3.3.3 table , and monitored on STP-1.0 every shift to ensure operable.

19. Containment Area Radiation (High Range)

072 G2.1.2 Area Radiation Monitor
Conduct of Operations:
Knowledge of operator responsibilities during all modes of plant operation.

Question Number: 38

Tier 2 Group 2

Importance Rating: 3.0

Technical Reference: FNP-1-ARP-1.6 FH1
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.10/43.2

Comments: This meets the KA in that it is the operator responsibility to know what area rad monitors are entry conditions to TSs.

QUESTIONS REPORT
for 75 RO Questions

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

Source :	NEW	Source if Bank:	
Cognitive Level:	LOWER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GTO	Previous 2 NRC exams:	NO

54. 073 A4.02 001

Given the following:

- R-19, SGBD SAMPLE, radiation monitor is in alarm and stable above the alarm setpoint.
- The Shift Chemist requests to sample the Steam Generators.

Which ONE of the following correctly describes the actions that will allow the shift chemist to obtain a sample of the SGs IAW SOP-45.0, Radiation Monitoring System?

A. Manually open the sample valves one at a time.

B. Pull the INSTRUMENT power fuses for R-19 to allow opening the sample valves.

C. Pull the DC power fuses to each sample valve solenoid to fail the valve open.

D✓ Place R-19 Operations Selector Switch to the RESET position, then open the sample valves.

A. Incorrect. These valves can not be manually opened. The SGBD sample valves do not have manual jacks as they have solenoids powered from DC power and fail closed.

B. Incorrect. This is the procedure directed action for a monitor in saturation, but not to clear a valid alarm.

C. Incorrect. these solenoid valves are DC powered and fail closed. Even though the designator is HV3328 and there is no manual operator on the valve.

D. Correct. SOP-45.0, Section 4.4 directs this.

Q1P15HV3328 1A Steam Generator Blowdown sample valve
Q1P15HV3329 1B Steam Generator Blowdown sample valve
Q1P15HV3330 1C Steam Generator Blowdown sample valve

073 Process Radiation Monitoring System
A4.02 Ability to manually operate and/or monitor in the control room:
Radiation monitoring system control panel

Question Number: 24

Tier 2 Group 1

Importance Rating: 3.7

Technical Reference: OPS 52106D; SOP-45.0 section 4.4
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.11

Comments:

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	D A B B D A A C A A		Scramble Range: A - D
Source :	NEW		Source if Bank:			
Cognitive Level:	LOWER		Difficulty:			
Job Position:	RO		Plant:		FARLEY	
reviewed:	GO		Previous 2 NRC exams:		NO	

55. 076 K3.01 001

Given the following:

- Unit 2 is at 39% power.
- 'A' Train is On-Service.
- 1-2A DG is running at full load IAW STP-80.1, Diesel Generator 1-2A Operability Test, and tied to Unit 2.
- A Loss of 'A' Train Service Water is occurring due to SW Pump failure.
- The crew is performing actions of AOP-10.0, Loss of Train A or B Service Water.
- 'B' Train SW header pressure is 72 psig.

Which ONE of the following describes a potential effect on the unit and the actions required in accordance with AOP-10.0?

A. • RCP Motor Air Coolers will lose cooling water flow;

• Trip the reactor and any RCP if its motor stator temperature exceeds the temperature limit.

B.✓ • RCP bearing temperatures will rise;

• Trip the reactor and any RCP if its bearing temperature exceeds the temperature limit.

C. • 1-2A DG will lose cooling water flow;

• Isolate Service Water to the Turbine Building and trip the reactor.

D. • Main Generator bearing and Hydrogen temperatures will rise;

• Isolate Service Water to the Turbine Building and trip the reactor.

A. Incorrect. plausible because Service Water does supply cooling to the motor air coolers, but Train B does, not Train A.

B. Correct. CCW temperature will rise, and as it does, RCP bearing temperatures will rise. This action is done in both AOP-10 and AOP-9 which AOP-10 sends the user to to accomplish in conjunction with AOP-10 at step 11.

C. Incorrect. plausible because Service Water does supply cooling to this DG and is in fact aligned to both units SW. Therefore a loss of Unit 2 does not cause a loss of cooling water to the DG. The candidate may believe that for a unit 2 STP, SW would be secured from unit 1 or flow is affected since STP-80.1 has the following note.
If service water from either unit is secured, a partial surveillance may be performed for the non affected unit. Full surveillance credit for this STP may be taken once service water is returned to service and verified operable by rerunning this STP.
The actions would be performed if the DG was required. It is NOT required. The RNO of step 4.2 says to isolate SW to the TB for the affected train and trip the reactor if BOTH trains were isolated. In this case the DG would be secured or left running if SW flow was sufficient from unit 1.

D. Incorrect.
First part is correct. second part incorrect since at step 6, SW pressure is checked to be > 60 psig. For this event, the SW pressure is 72 psig. (If it was less than 60 psig, then actions would be performed to isolate Service water to the Turbine Bldg. If this were done, then a reactor trip would be required.) Since it is not required to go to the RNO column, then it is not correct to do these actions.

076 K3.01 Service Water System
Knowledge of the **effect that a loss** or malfunction **of the SWS** will have on the following:
Closed cooling water

Question Number: 26

Tier 2 Group 1

Importance Rating: 3.4

Technical Reference: FNP-2-AOP-10.0
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.10

Comments:
This meets the KA in that there is a loss of SW to a train and requires knowledge of how this affects the equipment that receives CCW and is cooled by the SWS. This also tests the actions required for this event.

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

QUESTIONS REPORT
for 75 RO Questions

Source :	NEW	Source if Bank:	
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

56. 078 G2.1.32 001

Which ONE of the following describes the reason why one Air Compressor should be aligned with the air compressor panel key switch in **LOCAL** <u>OR</u> in the **MCB** position with the AUTOMATIC OPERATION LED (green light) LIT?

A.✓ To prevent a complete loss of Instrument Air pressure due to a single failure of the sequencer panel pressure transducer.

B. To allow one air compressor to be started from the MCB and operate from the selected sequencer for any complete loss of Instrument Air pressure situation.

C. To allow one air compressor to start after an LOSP and load and unload based on its Internal Mode pressure setting.

D. To prevent all three air compressors from running at the same time to prevent a complete loss of Instrument Air pressure in the event that Service Water is lost to the Turbine Building.

A. Correct. see P&L below
3.19 Failure of the sequencer panel pressure transducer could unload all air compressors selected (integrated) and result in complete loss of air pressure. **To prevent loss of air from a single failure, at least one air compressor should be aligned with the air compressor panel key switch in LOCAL OR in MCB with the AUTOMATIC OPERATION LED lit (green).**

B is incorrect. The air compressor in LOCAL will start from the MCB if OFF is selected first, then AUTO, but will not operate on the sequencer, but will start by its internal mode pressure switch.
3.8 Any air compressor with the panel key switch in the MCB position will (1) stop if the MCB handswitch is selected to OFF and (2) start and load if the MCB handswitch is taken to the START/RUN position and returned to AUTO position, based on the **Internal Mode pressure settings on the air compressor**. The AUTOMATIC OPERATION LED on the air compressor panel will be lit (green) when the MCB handswitch has been taken to the START/RUN position and returned to AUTO. The lit LED indicates the air compressor will load and unload based on its Internal Mode pressure settings. IF the MCB handswitch is taken from START/RUN to OFF, THEN the air compressor panel AUTOMATIC OPERATION LED will not be lit AND the LED will remain off if the MCB handswitch is taken from OFF back to AUTO without going to START/RUN AND the air compressor will not load and unload based on its Internal Mode pressure settings.

C is incorrect. There is a P&L applicable to 1C air compressor and the compressor will cycle on the sequencer after the load shed and LOSP is complete. This is not necessarily true for any air compressor operation after an LOSP. The Air compressor will also sequence back on and run on the sequencer, not the Internal Mode pressure setting.
3.11 During an LOSP or SI/LOSP the emergency section of Load Center 1A will automatically align to Load Center 1D, and 1C air compressor supply breaker EA-15 will automatically close. If the air compressor was operating prior to the LOSP, the compressor will resume operation after the LOSP if (1) the MCB handswitch is in AUTO (returned from START/RUN and not been taken to OFF) and the 1C panel key switch is in MCB position OR (2) the 1C panel key switch is in the SEQ position and 1C is selected (integrated) to the sequencer OR (3) the 1C panel key switch is in the LOCAL position.

D. incorrect. If air pressure drops with the switches in the above configuration, then all 3 air compressors will be running. This does not prevent 3 a/cs from running.

The normal system line-up is three air compressors in AUTO on the MCB, two air compressor selected (integrated) on the sequencer, and one air compressor de-selected (isolated) from the sequencer. This will allow the sequencer to control two air compressors based on header pressure and allow the de-selected (isolated) air compressor to auto start based on its receiver pressure.

078 G2.1.32 Instrument Air System
Conduct of Operations:
Ability to explain and apply all system limits and precautions.

Question Number: 27

Tier 2 Group 1

Importance Rating: 3.4

Technical Reference: FNP 1-SOP-31.0
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.10

Comments:
This meets the KA in that a precaution is required to be known and applied for a failure of a pressure switch for the instrument air compressors.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	A C A D A B A A C A	Scramble Range: A - D	
Source :	NEW			Source if Bank:		
Cognitive Level:	LOWER			Difficulty:		
Job Position:	RO			Plant:	FARLEY	
reviewed:	GO			Previous 2 NRC exams:	NO	

57. 103 K4.06 001

Given the following conditions:

- A LOCA has occurred.
- Containment pressure is currently 19 psig and rising.
- All automatic actions have occurred as required.
- No manual actions have been taken.

Which ONE of the following describes the ESF actuations that have taken place?

A. Safety Injection ONLY.

B. Safety Injection and Containment Isolation Phase A ONLY.

C✓ Safety Injection, Containment Isolation Phase A, and Main Steam Line Isolation ONLY.

D. Safety Injection, Containment Isolation Phase A, Main Steam Line Isolation, and Containment Isolation Phase B.

A is incorrect. because if SI actuates, Phase A will also be actuated.

B is incorrect. because Phase A is actuated, but MSLI is also actuated.

C is correct. Containment Isolation Phase A, and Main Steam Line Isolation ONLY, due to containment pressure.

D is incorrect. because containment pressure is not high enough for phase B.

QUESTIONS REPORT
for 75 RO Questions

103 K4.06 Containment System
Knowledge of containment system design feature(s) and/or interlock(s) which provide for the following:
Containment isolation system

Question Number: 28

Tier 2 Group 1

Importance Rating: 3.1

Technical Reference: E-0 Attachment 3
Proposed references to be provided to applicants during examination: None
Learning Objective:

10 CFR Part 55 Content:

Comments:
This tests the KA appropriately in that these are design features that provide for ctmt isolation at an RO level of knowledge.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	C D C D D A B C B B	Scramble Range: A - D	
Source :	BANK			Source if Bank:	WOLF CREEK	
Cognitive Level:	LOWER			Difficulty:		
Job Position:	RO			Plant:	FARLEY	
reviewed:	GTO			Previous 2 NRC exams:	NO	

58. E02 EA1.3 001

Given the following:

- Unit 1 was operating at 10% Reactor power when a Loss of Off-Site Power caused a loss of ALL RCP's and a spurious safety injection.
- Due to the LOSP, no RCP can be started.
- The crew has just adjusted the atmospheric relief valves.
- The crew is performing actions of ESP-1.1, SI Termination, and are at the step to determine if adequate natural circulation exists.

Which ONE of the following correctly lists indications that are consistent with **adequate** natural circulation IAW ESP-1.1?

1 - RCS hot leg temperature --- stable or decreasing
2 - RCS hot leg temperature --- increasing
3 - SG pressure --- stable or decreasing
4 - SG pressure --- increasing
5 - RCS hot leg temperature --- at saturation for SG pressure
6 - RCS cold leg temperature --- at saturation for SG pressure

A. 2, 3, and 5

B. 2, 4, and 6

C. 1, 4, and 5

D.✓ 1, 3, and 6

A. incorrect. RCS HL temps would not be increasing, and not RCS HL at SG saturation temperature.

B. incorrect. RCS HL temps would not be increasing, SG pressure would not be increasing

C. incorrect. SG pressure would not be increasing

D. Correct. **1, 3, and 6 -**

ESP-1.1 step 21.4 RNO lists NC flow requirements:

Verify adequate natural circulation.

a) Check SG pressure stable or falling. **#3**

b) Check SUB COOLED MARGIN MONITOR indication greater than 16°F subcooled in CETC mode.

c) Check RCS hot leg temperatures stable or falling. **#1**

d) Check core exit T/Cs stable or falling.

e) Check RCS cold leg temperatures at saturation temperature for SG pressure. **#6**
W/E02 SI Termination -

EA1.3 Ability to operate and / or monitor the following as they apply to the (SI Termination): **Desired operating results during abnormal and emergency situations.**

Question Number: 60

Tier 1 Group 2

Importance Rating: 3.8

Technical Reference: ESP-1.1 steps 2-18
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.10

Comments:
This question tests the ability to monitor for desired operating parameters IAW ESP-1.1, SI termination, during natural circ flow conditions.

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

Source :	MODIFIED	Source if Bank:	FARLEY
Cognitive Level:	LOWER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

59. E04 EK1.3 002

Given the following:

- A reactor trip and an SI have occurred.
- Containment pressure is reading 2 psig.
- RCS pressure is reading 1755 psig.
- All systems have operated as required.

At the step in EEP-1.0, Loss of Reactor or Secondary Coolant, the following indications are observed by the Unit Operator:

- The following BOP annunciator is in alarm:
  - NE2, 1B RHR PUMP RM SUMP LVL HI-HI OR TRBL
- 1A and 1B RHR pump discharge pressures are reading 750 psig.
- MK4, LIQ OR GAS PROC PNL ALARM, has just come into alarm.

Which ONE of the following describes 1) the <u>operator actions</u>; **AND** 2) the <u>operational implications</u> of those actions performed IAW ECP-1.2, LOCA Outside Containment, in an attempt to mitigate this leak?

A.✓ 1) Isolate the discharge to ONE train of RHR and check RCS pressure rising;
2) Loss of one train of LHSI for injection and recirculation.

B. 1) Isolate the RWST suction to ONE train of RHR and check RWST level stable;
2) Loss of one train of LHSI for injection ONLY.

C. 1) Isolate the discharge to BOTH trains of RHR and check RCS pressure rising;
2) Loss of BOTH trains of LHSI for injection and recirculation.

D. 1) Isolate the RWST suction to BOTH trains of RHR and check RWST level stable;
2) Loss of BOTH trains of LHSI for injection ONLY.

A. Correct. At FNP, the most credible source of an ISLOCA is from the RCS to the LHSI pump suction piping which is a low pressure system. The high level action steps of ECP-1.2 are to verify proper valve alignment, attempt to isolate the break, check if the break is isolated. The first system to be isolated is the RHR system which in this case would cause a loss of one train of LHSI due to the isolation of RHR valves. MOVs 8888A and 8887A are closed and the leak checked, then the other train. Only one train is checked at a time and isolated so BOTH trains are not affected.

At FNP, the most credible mechanism for initiation of RHR suction ISLOCA during power operation is the catastrophic rupture of the closed MOVs isolating the RHR pump suction from the RCS.
Following the RCS to RHR pressure boundary failure, the RHR system will be able to withstand RCS pressure if the hoop stress imposed on the RHR system by exposure to RCS pressures is below the yield stress. However, **<u>the RHR pump seals in both trains are expected to rupture</u>. <u>This rupture of pump seals is assumed to result in failure of both RHR pumps</u>**

(i.e., motors short due to water spray).

B. incorrect- correct location, incorrect strategy. incorrect operational implication. The RHR system is not secured to **both** trains at the same time. ECP-1.1 isolates one train first and then restores and isolates the other train if the problem is not corrected. ECP-1.2 never isolates the RWST to any component. However, ECP-1.1 does. The loss of the train would be for injection and recirc for one train.

C. Incorrect. Incorrect location, incorrect strategy. incorrect operational implication.

D. incorrect. incorrect location. incorrect strategy. incorrect operational implication. Location - Components between the RCS and the low pressure LHSI Pump hot leg injection piping include three check valves. A LOCA through the upstream hot leg injection piping is less likely than through the cold leg piping due to the addition of an additional in-series check valve and because the upstream isolation valves are normally closed. Also the Background states that the piping is able to withstand RCS pressure if the hoop stress imposed on the RHR system by exposure to RCS pressures is below the yield stress.
The RWST will not be lost due to the isolation of the system, it will be saved due to this action. ECP-1.2 never isolates the RWST to any component. However, ECP-1.1 does.

Further background-

Purpose: To ensure that normally closed valves are closed
Basis: This step instructs the operator to verify that all normally closed valves in low pressure lines and other plant specific lines that penetrate containment are closed. The valving connecting the RHR System to the RCS is of particular interest in this step since the RHR System is a low pressure system (600 psig) connected to the high pressure reactor coolant system (2500 psig). Therefore, a rupture or break outside containment is most probable to occur in the low pressure RHR System piping.

ERG StepText: *Check If Break Is Isolated*
Purpose: To determine if the LOCA outside containment has been isolated from previous actions
Basis: 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.

ERP StepText: Identify source of leak.
ERG StepText: *Try To Identify And Isolate Break*
Purpose: To attempt to identify and isolate a LOCA outside containment

Basis: This step instructs the operator to sequentially close and open all normally opened valves in paths that penetrate containment to identify and isolate the break outside containment. Again, as in Step 1, the valving connecting the low pressure (600 psig) RHR System to the high pressure (2500 psig) RCS is of primary interest, since the probability of a break occurring outside containment is most probable to occur in the low pressure RHR System piping.
Knowledge: The potential exists for RWST inventory to be lost to the auxiliary building for a LOCA that occurs outside containment in the RHR system piping. The RWST could be drained to the auxiliary building if the RCS pressure is reduced to below the static head pressure in the

<u>RWST. If this condition occurs, actions should be taken to isolate this potential leakage path and loss of inventory from the RWST.</u>

W/E04 LOCA Outside Containment

EK1.3 Knowledge of the **operational implications** of the following concepts as they apply to the (LOCA Outside Containment):
Annunciators and conditions indicating signals, and remedial actions associated with the (LOCA Outside Containment).

Question Number: 53

Tier 1 Group 1

Importance Rating: 3.5

Technical Reference: ECP-1.2, FNP-0-ECB-1.2 specific background document for ECP-1.2, lesson plan for ECP-1.2, OPS-52532E, WOG background document
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.10

Comments:
This question gives the indications of an ISLOCA and then the operational implications and remedial actions or high level actions. The procedure would have the operator isolate one train at a time so a complete loss of injection is not done. The implications of this action if the system is isolated IAW ECP-1.2 is to lose one train of LHSI. The RWST is a credible distracter in that ECP-1.1 which is where this procedure could send the operator to does isolate and turn off pumps due to low RWST level.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	A C D A C B B B B A	Scramble Range: A - D
Source :	NEW		Source if Bank:		
Cognitive Level:	HIGHER		Difficulty:		
Job Position:	RO		Plant:	FARLEY	
reviewed:	GTO		Previous 2 NRC exams:	NO	

60. E05 G2.1.27 003

A Reactor Trip and Safety Injection have occurred on Unit 1:

The crew has entered FRP-H.1, Response to Loss of Secondary Heat Sink, from EEP-1, Loss of Primary or Secondary Coolant, with the following conditions:

- RCS Pressure is 175 psig and decreasing.
- Intact SG pressures are 475 psig and trending down.

Which ONE of the following describes the status of the Steam Generators and the associated procedural requirement for the conditions given above?

The Steam Generators are ______________________

A. • Available to provide secondary heat sink.

• Remain in FRP-H.1.

B.✓ • NOT Available to provide secondary heat sink.

• Return to EEP-1.

C. • Available to provide secondary heat sink.

• Return to EEP-1.

D. • NOT Available to provide secondary heat sink.

• Remain in FRP-H.1.

A - Incorrect. Secondary heat sink is not required if SGs are at a higher pressure than the RCS. They act as a heat source. Plausible because the conditions for FRP-H.1 entry are met except for the first step of FRP-H.1. RNO sends to procedure and step in effect.

B - Correct. If SGs are NOT required for heat sink, the crew will return to EEP-1.

C - Incorrect. SGs are NOT required, because RCS pressure is below SG pressure.

D - Incorrect. LBLOCA, RCS less than SG pressure, return to EEP-1.

W/E05 Loss Secondary Heat Sink

G2.1.27 Conduct of Operations: Knowledge of system purpose and or function.

Question Number: 54

Tier 1 Group 1

Importance Rating: 2.8

Technical Reference: FRP-H.1
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.10

Comments: meets the KA in that the operator has to know the purpose of the SGs and the function/role they play in removing/adding heat on a LOCA.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	B B D A D B A C B B	Scramble Range: A - D
Source :	BANK		Source if Bank:	WTSI	
Cognitive Level:	HIGHER		Difficulty:		
Job Position:	RO		Plant:	FARLEY	
reviewed:	GO		Previous 2 NRC exams:	NO	

61. E06 EA1.2 064

Given the following:

- A LOCA has occurred.
- Due to ECCS failures, the crew is performing FRP-C.2, Response to Degraded Core Cooling.
- The crew is depressurizing ALL Steam Generators to 100 psig.
- The STA reports a RED condition on the Integrity CSF Status Tree.
- The Shift Supervisor continues in FRP-C.2.

Which ONE of the following describes the reason that the SS remains in FRP-C.2?

A. Actions cannot be taken for a RED condition on the Integrity CSF because there is inadequate ECCS equipment available to mitigate the degraded core cooling condition.

B. The RED condition on the Integrity CSF is not valid because of the dumping of steam to the condenser at a maximum rate IAW FRP-C.2.

C. FRP-C.2 has a higher priority than any lower level CSF and no other procedural actions are allowed to be implemented until a transition is directed IAW FRP-C.2.

D✓ The RED condition on the Integrity CSF is expected and is based upon the accumulators injecting.

A. Incorrect. Actions taken would actually be to reduce ECCS flow, so unavailability of SI for a PTS issue would not be a priority.

B. Incorrect. The red condition is valid. The dumps are NOT opened to dump steam at a maximum rate, the limit of 60°F/hour is the limit in C.2. In FRP-C.1, Steam is dumped at a maximum rate.

C. Incorrect. Core Cooling is high priority, but FRP-C.2 is entered on an orange condition, so a red condition on another CSF tree would take priority. However, in the specific case of the integrity CSF, the dumping of the accumulators is expected and subsequent entry into this FRP would only cause core temperatures to rise and C.1 entry could be required.

D. Correct. Due to the dumping of the steam above, accumulator injection is the goal and the dumping of steam is done at a rate of 60°F/ hr.

FRP-C.2 Caution prior to step 12 just before depressurizing all intact SGs to 100 psig.

CAUTION: Performance of step 12 will cause accumulator injection which may result in a red path on the INTEGRITY status tree. This procedure should be completed before transition to FNP-1-FRP-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITIONS.

background documentation

ERP StepText: Performance of step 12 will cause accumulator injection which may result in a red path on the INTEGRITY status tree. This procedure should be completed before transition to FRP-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITIONS.
ERG StepText: *The following step will cause accumulator injection which may cause a red path condition in F-0.4, INTEGRITY Status Tree. This guideline should be completed before transition to FRP. 1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK.*

Purpose: To alert the operator to complete entire guideline FR-C.2 even if a red path occurs in the Integrity Status Tree, F-0.4.

Basis: Once the RCS is cooled/depressurized in step 10 to the point at which the accumulators inject, the RCS cold leg temperature could be reduced such that a transition to FR-P.1, Response to Imminent Pressurized Thermal Shock Condition, is required via the red path of Status Tree F-0.4. The operator would stop the cooldown after entering FR-P.1.
While the operator is allowing the thermal shock to soak out, the core will continue to boil away the injected accumulator water and begin to uncover once again. Eventually, core exit temperatures and/or RVLIS level values could exist which would require the operator to transfer to FR-C.1, Response to Inadequate Core Cooling, via one of the red paths on Status Tree F-0.2. Thus, by going from FR-C.2 to FR-P.1 and stopping the cooldown and soaking, a degraded core cooling condition could be allowed to deteriorate to an inadequate core cooling condition. Therefore, this caution will require the operator to complete guideline FRC. 2 to ensure core cooling even if a red path condition occurs in the Integrity Status Tree, F-0.4.

W/E06 Degraded Core Cooling

EA1.2 Knowledge of the reasons for the following responses as they apply to the (Degraded Core Cooling) Operating behavior characteristics of the facility.

Question Number: 61

Tier 1 Group 2

Importance Rating: 3.5

Technical Reference: 1-FRP-C.2 and FNP-0-FRB-C.2 background document

Proposed references to be provided to applicants during examination: None

Learning Objective:
10 CFR Part 55 Content: 41.7

Comments: This question tests the operating behavior characteristic in that when this condition is entered and the SGs are depressurized, the resulting accumulator injection is expected to cause FRP-P.1 conditions due to the rapid cooldown. This is a high level action of FRP-C.2 that an RO is expected to know.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	D D D C A D B D D A	Scramble Range: A - D	
Source :	MODIFIED		Source if Bank:			
Cognitive Level:	LOWER		Difficulty:			
Job Position:	RO		Plant:		FARLEY	
reviewed:	GTO		Previous 2 NRC exams:		NO	

62. E08 EA2.1 001

Given the following:

- While operating at 100% power, Unit 1 experienced a steam break accident inside CTMT.
- After the transition from EEP-0, Reactor Trip or Safety Injection, RCS cold leg temperatures had dropped to 240°F in 20 minutes.
- Reactor power is currently < 1% and Both Intermediate Range SUR meters are reading -.03 dpm.
- AFW flow is reading 300 gpm.
- SG narrow range water levels are reading:
  - 1A SG = 49%
  - 1B SG = 29%
  - 1C SG = 30%
- Sub Cooled Margin Monitor is reading 34°F in the CETC mode.
- Containment pressure is currently 28 psig and slowly decreasing.
- CS flow is reading 950 gpm with one CS pump running.
- RCS pressure is currently 1500 psig with all ECCS equipment running.

Which ONE of the following is the **highest** level Functional Restoration Procedure (FRP) required to be entered under these conditions?

A. FRP-Z.1, Response to High Containment Pressure.

B. FRP-S.1, Response to Nuclear Power Generation- ATWT.

C✓ FRP-P.1, Response to Imminent Pressurized Thermal Shock Conditions.

D. FRP-H.1, Response to Loss of Secondary Heat Sink.

reference provided is the **RCS pressure - temperature graph CSF-0.4** rev 17 so a determination can be made to which area of the graph applies.

All distracters are plausible in that evaluation has to be made to determine which FRP is valid and what condition it is in, orange green yellow. Then a determination of which order these are referenced in.

Answer A is incorrect: Z.1 is an Orange path but it is lower than P.1 orange path.

Answer B is incorrect: FRP S.1 is a green path based on <5% power, IR SUR more negative than -.02.

Answer C is correct: P.1 entered on an orange path.

Answer D is incorrect. H.1 entry conditions are < 395 gpm or all sg nr levels < 31%. if one sgwl NR is > 31% no entry is required.

QUESTIONS REPORT
for 75 RO Questions

E08 EA2.1 Pressurized Thermal Shock - EPE

Ability to operate and / or monitor the following as they apply to the (Pressurized Thermal Shock) Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Question Number: 62

Tier 1 Group 2

Importance Rating: 3.4

Technical Reference: none

Learning Objective: OPS 52533K-8
10 CFR Part 55 Content: 41.10

Comments: matches KA in that the operator has to monitor the correct parameters and then based on those parameters select the appropriate procedure to follow during the emergency event and it deals with a PTS event.

All distracters are plausible in that evaluation has to be made to determine which FRP is valid and what condition it is in, orange green yellow. Then a determination of which order these are referenced in.

This does not overlap with G2.4.22

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

Source :	MODIFIED	Source if Bank:	FARLEY
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

63. E10 EK1.1 005

Given the following:

- An RCS cooldown is in progress IAW ESP-0.3, Natural Circulation Cooldown with Allowance for Reactor Vessel Head Steam Voiding (WITH RVLIS).
- The plant is being depressurized using auxiliary spray.
- Charging and Letdown flows are matched.
- As RCS pressure drops through 1300 psig, a rapid rise in pressurizer level is observed.
- Pressurizer level has increased to 66%.
- Reactor vessel level indication has dropped **below** the minimum required UPPER PLENUM level of 44%.

Which ONE of the following correctly describes the required response IAW ESP-0.3?

A. Increase auxiliary spray flow and verify Both CRDM cooling fans running.

B. Reduce charging flow, increase letdown flow and stop the cooldown in progress.

C. Increase the RCS cooldown rate while maintaining charging and letdown flows matched.

D✓ Reduce the auxiliary spray flow and energize additional pressurizer heaters.

A. Incorrect - Plausible because verifying both CRDM cooling fans running would cool the head and reduce void formation & is required later in procedure. Spray flow should be reduced, not increased, to establish subcooling.

B. Incorrect - Plausible because ESP-0.3 says to control or reduce charging and increase letdown or continue the cooldown if PRZR level is >90%.Level is only 66%. the method at this point in the procedure.

C. Incorrect - Plausible because cooling down the RCS will cool the head, but with a time delay. With the head still hot, this will cause the void to increase and PRZR to go solid.

D. Correct - This is the correct response IAW ESP-0.3 when Reactor vessel level indication drops to less than 44% upper plenum raise RCS pressure and this would be done by controlling pressure with sprays and heaters.

QUESTIONS REPORT

for 75 RO Questions

W/E10 EK1.1 Natural Circulation with Steam Void in Vessel with/without RVLIS -
Knowledge of the operational implications of the following concepts as they apply to the (Natural Circulation with Steam Void in Vessel with/without RVLIS) Components, capacity, and function of emergency systems.

Question Number: 63

Tier 1 Group 2

Importance Rating: 3.3

Technical Reference: ESP-0.3
Proposed references to be provided to applicants during examination: None
Learning Objective: OPS 52531C06
10 CFR Part 55 Content: 41.10

Comments:
matches KA in that it tests the knowledge of ESP-0.3 and the implications of void formation and what to do about it using the components that the operator os required to control during this evolution.

MCS Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
		Answer:	D A A A A C D B B D	Scramble Range: A - D	
Source :	BANK		Source if Bank:	FARLEY	
Cognitive Level:	HIGHER		Difficulty:		
Job Position:	RO		Plant:	FARLEY	
reviewed:	GO		Previous 2 NRC exams:	NO	

64. E11 EK2.2 004

The following conditions exist on Unit 1:

- 1B RHR pump is tagged out and the oil is drained from the motor.
- The Control room team is responding to a LOCA.
- The reactor was tripped and an SI manually actuated.
- RCS pressure is 1000 psig.
- 1A RHR pump has tripped.

The control room team has transitioned to ECP-1.1, Loss Of Emergency Coolant Recirculation. Make up has been established to the RWST.
Which one of the following describes the correct actions to take in ECP-1.1 under these conditions?

A.✓ • Initiate an RCS cooldown to Cold Shutdown at less than 100°F/hr,
• establish only one charging pump running and
• reduce RCS pressure to reduce break flow.

B. • Initiate an RCS cooldown to Cold Shutdown at the maximum rate possible,
• establish only one charging pump running and
• reduce RCS pressure to dump the accumulators.

C. • Initiate an RCS cooldown to Cold Shutdown at less than 100°F/hr,
• establish two charging pumps running and
• reduce RCS pressure to dump the accumulators.

D. • Initiate an RCS cooldown to Cold Shutdown at the maximum rate possible,
• establish two charging pumps running and
• reduce RCS pressure to reduce break flow.

ECP-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION OPS-52532D

A. Correct - **Start makeup to the RWST, initiate an RCS cooldown, minimize ECCS flow and reduce RCS pressure.**
The following criteria are the high level actions needed to be successful in ECP-1.1
Makeup to the RWST is necessary
Inventory in the RWST is a concern for recovery from a loss of ECR capability. Makeup is added to the RWST to extend the time the SI pumps and containment spray pumps (if operating) can take suction from the RWST and provide core cooling to the RCS.
Begin Cool Down to Cold Shutdown
The purpose is to begin a controlled RCS cool down to cold shutdown temperature using a preferred or alternate method with a specified maximum cool down rate. Shutdown margin should be monitored during RCS cool down using Curve 61 and/or 61A.
The objective is to reduce the overall temperature of the RCS coolant and metal to reduce the need for supporting plant systems and equipment required for heat removal. The maximum cool down rate of 100°F/hr will preclude violation of the integrity status tree, thermal shock limits.
Stop SI Pumps
To reduce flow into the RCS, the low-head injection pumps and all but one high-head pump are stopped. Satisfaction of conditions for SI termination implies that control can be maintained by the operator without all of the ECCS pumps running. In this step, all but one high-head pump are stopped and placed in standby for future use.
Reduce RCS Pressure to Reduce Subcooling
This step is performed to decrease RCS pressure to the lowest pressure possible without losing adequate subcooling. The RCS pressure reduction is done to decrease RCS break flow. The RCS should be depressurized until RCS subcooling indicates between 16°F (45°F) and 26°F (55°F) on the Subcooled Margin Monitor in CETC mode. A second criterion for stopping the pressure reduction is PRZR level greater than 73% (50%).

B. Incorrect - See above, first and third part incorrect, second part correct
C. Incorrect - See above, first part correct, second and third part incorrect
D. Incorrect - See above, first and second part incorrect, third part correct

W/E11 Loss of Emergency Coolant Recirculation -

EK2.2 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.

Question Number: 55

Tier 1 Group 1

Importance Rating: 3.9
Technical Reference: ECP-1.1 and ECP-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION OPS-52532D and FNP-0-ECB-1.1

Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.10

Comments: This meets the KA in that the operator has to know the strategy of the procedure and the proper operation of the the various heat removal systems to control the casualty in progress.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	A A A A A A A A A A	Items Not Scrambled	
Source :		BANK		Source if Bank:		FARLEY
Cognitive Level:		HIGHER		Difficulty:		
Job Position:		RO		Plant:		FARLEY
reviewed:		GO		Previous 2 NRC exams:		NO

65. E15 EK3.4 003

Which ONE of the following is a potential source of the flooding that is checked for in FRP-Z.2; and the concern if the maximum expected post-accident containment water level (design basis containment flood level) is exceeded?

A. • Condensate Storage Tank.
 • Thermal shock to the reactor vessel lower head due to quenching.

B. • Condensate Storage Tank.
 • Damage to vital system or components rendering them inoperable.

C✓ • Service Water system.
 • Damage to vital system or components rendering them inoperable.

D. • Service Water system.
 • Thermal shock to the reactor vessel lower head due to quenching.

A. Incorrect. CST is not one of the potential sources of water FRP-Z.2 addresses and the background does not mention. However, since AFW goes into ctmt and is in use, it is plausible that this large source of water would be checked for and is a concern. The maintenence sump is isolated from the bottom of the reactor vessel by a wall with an elevation higher than the vital equipment of concern. Plausible, because a high enough containment level would allow water to potentially thermally shock the hot post accident reactor vessel.

B. Incorrect. CST is not one of the potential sources of water FRP-Z.2 addresses and the background does not mention. The reason is correct.

C. Correct. Service water is the most likely source since it does not isolate to the ctmt on a phase A or B signal and is the largest source of water available to ctmt.

The purpose of the sump is to collect and divert water in areas that will not affect vital plant equipment. Flooding may jeopardize that function. RWST, CST, & RCS are expected to provide their full volumes to the CTMT sump in accident analysis.

D. Incorrect. SW is correct.
The reason is incorrect. Plausible, because a high enough containment level would allow water to potentially thermal shock the hot post accident reactor vessel.

FNP-0-FRB-Z.2 specific background document for FRP-Z.2 for step 1
Basis: This step instructs the operator to try to identify the unexpected source of the water in the containment sump. Containment flooding is a concern since critical plant components necessary for plant recovery may be damaged and rendered inoperable. A water level greater than the design basis flood level provides an indication that water volumes other than those represented by the emergency stored water sources (e.g., RWST, accumulators, etc.) have been introduced into the containment sump. Typical sources which penetrate containment are service water, component cooling water, primary makeup water and demineralized water. All possible plant specific sources which penetrate containment should be included in this step. These systems provide large water flow rates to components inside the containment and a major leak or break in one of these lines could introduce large quantities of water into the sump. Identification and isolation of any broken or leaking water line inside containment is essential to maintaining the water level below the design basis flood level.

QUESTIONS REPORT
for 75 RO Questions

E15 EK3.4 Containment Flooding - EPE

Knowledge of the reasons for the following responses as they apply to the (Containment Flooding)

RO or SRO function as a within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated.

Question Number: 64

Tier 1 Group 2

Importance Rating: 2.9

Technical Reference: FRP-Z.2 and FNP-0-FRB-Z.2 specific background document for FRP-Z.2

Proposed references to be provided to applicants during examination: None
Learning Objective: OPS 52533M01
10 CFR Part 55 Content: 41.10

Comments:
This meets the KA in that it tests the knowledge of the operator on where the most likely source of water would come from and then the limitations that the bkgrd documents speak of.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	C A A A C C D C C A	Scramble Range: A - D	
Source :	NEW		Source if Bank:			
Cognitive Level:	LOWER		Difficulty:			
Job Position:	RO		Plant:		FARLEY	
reviewed:	GO		Previous 2 NRC exams:		NO	

66. G2.1.10 001

Unit 2 is in MODE 2 and a reactor startup is in progress.

In accordance with Technical Specifications 2.1, Safety Limits (SLs), which ONE of the following describes the RCS Pressure Safety Limit, and the **MAXIMUM** time to take action if it is exceeded?

Limit	MAXIMUM time
A. 2735 psig	5 minutes
B.✓ 2735 psig	1 hour
C. 2750 psig	5 minutes
D. 2750 psig	1 hour

A. Incorrect. In Modes 3,4,5, TS allows 5 minutes to restore pressure

B. Correct. In Modes 1 or 2, TS allows 1 hour to Mode 3. In Modes 3,4,5, TS allows 5 minutes to restore pressure.

C. Incorrect. 2750 would be correct in psia, but not in psig.

D. Incorrect. 2750 would be correct in psia, but not in psig.

G2.1.10 Conduct of Operations
Knowledge of conditions and limitations in the facility license.

Question Number: 67

Tier 3 Group 1

Importance Rating: 2.7

Technical Reference: TS 2.2.2
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 43.2/41.10

Comments: This meets the KA in that it tests the knowledge of a RCS safety limit and the applicable tech spec requirements for that limit for RO knowledge.

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

Source :	MODIFIED	Source if Bank:	
Cognitive Level:	LOWER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GTO	Previous 2 NRC exams:	NO

67. G2.1.12 001

Unit 1 is at 100% power. All three Auxiliary Feedwater Pumps have just been declared INOPERABLE.

Which ONE of the following actions MUST be taken?

A. Be in Mode 3 in 6 hours and be in Mode 4 in in 12 hours.

B. Take action to restore at least one AFW pump to OPERABLE status within 1 hour, and a second AFW pump within 6 hours, or be in Mode 3 in the next 6 hours.

C✓ Immediately take action to restore at least one AFW pump to OPERABLE status.

D. Immediately enter LCO 3.0.3 and take actions to initiate a shutdown within 1 hour.

A. Incorrect. T.S. 3.7.5 with 2 trains INOP or Required Action of A or B not met, then the action is to enter mode 3 in 6 and mode 4 in 12 hours.

B. Incorrect. T.S. 3.7.5 does not have a 1 hour Action for AFW. If the AFW pump was returned in one hour a case can be made it was RTS immediately, but the second pump being INOP is not allowed to be INOP for 6 hours w/o action. Therefore, the distracter is incorrect since when one AFW pump is RTS, Required Action for C. is to place the unit in mode 3 in 6 hours, not wait an additional 6 hours to fix it. This is a sly way of using a 6 hour LCO from memory in that the unit has to be placed in mode 3 in 6 hours but the LCO C is entered immediately once one AFW pump is RTS.

C. Correct. This answer reflects the Note contained in action D as discussed above.

TS 3.7.5 in Condition D has a NOTE that states: LCO 3.0.3 and all other action statements requiring a Mode change are suspended until one AFW train is restored to operable status.
This prevents placing the plant in a much higher risk condition than required.

D.1 says: Initiate action to restore one AFW train to OPERABLE status.

D. Incorrect. note in LCO 3.7.5
LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status.

G2.1.12 Conduct of Operations
Ability to apply technical specifications for a system.

Question Number: 66

Tier 3 Group 1

Importance Rating: 2.9

Technical Reference: TS section 3.7.5
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 43.2/41.10

Comments: this meets the KA at an RO expected level of knowledge for the AFW system.
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C D C A A A D A C B Scramble Range: A - D

Source :	MODIFIED	Source if Bank:	FARLEY
Cognitive Level:	LOWER	Difficulty:	
Job Position:		Plant:	FARLEY
reviewed:	GTO	Previous 2 NRC exams:	NO

68. G2.1.8 004

Given the following:

- An oil spill has occurred from a non-PCB oil source in the Turbine Building.
- The following conditions exist:
  - The oil has reached the Turbine Building sump.
  - The sump pump is running and releasing water to the environment.

Which ONE of the following actions is **required** to be performed by the control room team IAW AP-60, Oil Spill Prevention Control and Countermeasure Plan, Hazardous Waste Contingency Plan?

A.✓ • Dispatch the SSS to the scene.
• Direct the TB System Operator stop the sump pump, then close and tag the discharge valve.

B. • Dispatch the shift chemist to the scene.
• Direct the TB System Operator place the sump on recirc until the sump contents can be analyzed.

C. • Dispatch the SSS to the scene.
• Direct the TB System Operator place the sump on recirc until the sump contents can be analyzed.

D. • Dispatch the shift chemist to the scene.
• Direct the TB System Operator stop the sump pump, then close and tag the discharge valve.

A. Correct - Dispatch the SSS to the scene, have the TB System Operator stop the sump pump, close and tag the discharge valve. AP-60 requires these actions: SSS to the scene, stop the release and close and tag sump discharge valves.

B. Incorrect - The shift radiochemist is not required to be dispatched, though this may be a good idea, however chemistry supervision is required to be notified. The shift radiochemist is not necessarily supervision. The TB System Operator should not place the sump on recirc. With a release in progress the requirement is to stop the release.

C. Incorrect- The TB System Operator is required to stop the release immediately, not evaluate how much more water can be released. The release needs to analyzed prior to starting and needs supervisor approval to start it.

D. Incorrect - The shift radiochemist is not required to be dispatched, the release would not continue with oil going to it due to the potential for release to the environment, and the discharge valve would be closed, and also tagged.

G2.1.8 Conduct of Operations
Ability to coordinate personnel activities outside the control room.

Question Number: 68

Tier 3 Group 1

Importance Rating: 3.8

Technical Reference: AP-60 Appendix 1
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content:

Comments:

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	A A A A A A A A A A	Items Not Scrambled
Source :	BANK		Source if Bank:	FARLEY	
Cognitive Level:	LOWER		Difficulty:		
Job Position:	RO		Plant:	FARLEY	
reviewed:	GO		Previous 2 NRC exams:	YES	

69. G2.2.22 002

Unit 1 is in a Refueling Outage with fuel being loaded into the core.

Which one of the following describes the MINIMUM temperature and the MINIMUM borated water volume that must be met to maintain an operable Boric Acid Storage Tank (BAT Tank)?

	Solution Temperature	Borated Water Volume
A.	35°F	2,000 gal.
B.	35°F	11,336 gal.
C.✓	65°F	2,000 gal.
D.	65°F	11,336 gal.

Reference:
Technical Requirements Manual, TRM 13.1.6.4 and 13.1.6.6

A. Incorrect, Mode 5 and 6. TRS 13.1.6.6 Verify the contained borated water volume in the boric acid storage tank is ≥ 2,000 gal., TRS 13.1.6.1 Verify RWST solution temperature is > or equal to 35°F

B. Incorrect, Mode 5 and 6, TRS 13.1.6.1 Verify RWST solution temperature is > or equal to 35°F. Mode 1,2,3&4, TRS 13.1.7.4 Verify the contained borated water volume in the boric acid storage tank is ≥ 11,336 gal

C. Correct, Plant is in Mode 6. The following TRSs apply. TRS 13.1.6.4 Verify boric acid storage tank solution temperature is > or equal to 65°F, TRS 13.1.6.6 Verify the contained borated water volume in the boric acid storage tank is ≥ 2,000 gal

D. Incorrect, Mode 5 and 6, TRS 13.1.6.4 Verify boric acid storage tank solution temperature is > or equal to 65°F. Mode 1,2,3&4, TRS 13.1.7.4 Verify the contained borated water volume in the boric acid storage tank is ≥ 11,336 gal

G2.2.22 Equipment Control
Knowledge of limiting conditions for operations and safety limits.

Question Number: 69

Tier 3 Group 2

Importance Rating: 3.4
Technical Reference: TRM 13.1.6.4 and 13.1.6.6

Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.10/43.2

Comments:
replaced this question since the concept is already tested on the SRO portion of the exam, ie., IR instrument failed at 10-8 amps and what to do and why.

This is more of an RO question with no overlap on the exam and meets the LCO requirements above.

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

Source :	BANK	Source if Bank:	FARLEY
Cognitive Level:	LOWER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GTO	Previous 2 NRC exams:	NO

70. G2.2.34 002

Given the following:

- Unit 1 is in Mode 3.
- Reactor tripped from 100% RTP.
- ECC has been calculated for a startup 12 hours after the trip.
- Estimated critical rod position is Control Bank D at 100 steps.
- Startup is delayed for TWO (2) hours.

Which ONE of the following describes the effect on 1/M plot data taken during the approach to critical?

The 1/M plot will predict criticality at a.....

A. **LOWER** rod height due to Xenon concentration **greater** than that assumed in ECC calculation.

B✓ **LOWER** rod height due to Xenon concentration **less** than that assumed in ECC calculation.

C. **HIGHER** rod height due to Xenon concentration **less** than that assumed in ECC calculation.

D. **HIGHER** rod height due to Xenon concentration **greater** than that assumed in ECC calculation.

A: Incorrect. Lower rod height is correct. Xenon concentration greater is incorrect. Xenon concentration will be less but will be adding positive reactivity which will result in a lower rod height for criticality to be obtained.

B: Correct. Delay will affect core reactivity since Xenon is decaying, reducing the negative reactivity in the core. Rods will not have to be withdrawn as far to make the reactor critical.

C: Incorrect. Higher rod height is incorrect. Xenon concentration will be less but will be adding positive reactivity which will result in a lower rod height for criticality to be obtained.

D: Incorrect. Rods will not have to be withdrawn as far to make the reactor critical. Delay will affect core reactivity since Xenon is decaying, reducing the negative reactivity in the core. Candidate needs to demonstrate an understanding of the time that it takes Xenon to peak from a full power trip, which is typically the square root of the equilibrium power level.

G2.2.34 Equipment Control
Knowledge of the process for determining the internal and external effects on core reactivity.

Question Number: 70

Tier 3 Group 2

Importance Rating: 2.8

Technical Reference: Physics curve 60
Proposed references to be provided to applicants during examination: None
Learning Objective:

10 CFR Part 55 Content: 41.1

Comments: This KA tests the operator to evaluate core reactivity and the effects of xenon after a rx trip for a startup.

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

Source :	MODIFIED	Source if Bank:	SEQUOYAH 2004
Cognitive Level:	HIGHER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

71. G2.3.10 007

What precaution is required to be taken at the 121' Piping Penetration Room (PPR) prior to lowering RCS level to mid-loop IAW UOP-4.3, Mid Loop Operations?

A. The door to the 121' PPR must be locked.

B. Health physics (HP) must survey the 121' PPR.

C✓ A caution sign must be placed at the entrance of the 121' PPR.

D. All vent valves on systems in the 121' PPR penetrating containment must be verified closed.

UOP-4.3

2.24 Prior to reducing RCS level, a caution sign concerning the establishment of containment closure must be placed at the entrance of the following locations.

NOTE: The signs can be obtained from the Shift Support Supervisor and are normally stored in the CCW Storage Room on Unit 2.

_____	2.24.1	139' Electrical Penetration Room
_____	2.24.2	121' Piping Penetration Room
_____	2.24.3	100' Piping Penetration Room
_____	2.24.4	Main Steam Valve Room
_____	2.24.5	Personnel Access Hatch
_____	2.24.6	Auxiliary Access Hatch

A. Incorrect - Door not required to be locked SOP 0.0, 15.3.5
The following doors will be locked closed when unattended during unit operation in Modes 1 through 4:

- 139' Electrical Penetration Room Doors 317A/2317
- 121' Piping Penetration Room Doors 214/2214

B. Incorrect - Surveys are not required prior to reducing level.

C. Correct - per the above initial condition of UOP-4.3

D. Incorrect - Air to air barrier not required for midloop integrity (ctmt closure in 2 hrs)

G2.3.10 Radiation Control
Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.

Question Number: 71

Tier 3 Group 3

Importance Rating: 2.9

Technical Reference: Health Physics manual

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe how the rad monitoring system helps to protect the health and safety of plant workers and the public. (ESP52106D08)
10 CFR Part 55 Content: 43.4

Comments: This question tests the basic generic applicability of the KA at an RO level of knowledge.

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

Source :	BANK	Source if Bank:	FARLEY
Cognitive Level:	LOWER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

72. G2.3.9 001

The following conditions exist on Unit 2 while at 100% power:

- Containment Main Purge and Mini-Purge are secured.
- CTMT to ATMOS DP is currently 0.3 psid.
- All pre-requisites to perform a batch release of the containment atmosphere have been met.

Which ONE of the following describes where the containment purge system discharges to and how the system is operated to reduce containment pressure IAW SOP-12.2, Containment Purge and Pre-access Filtration System, Appendix 3, Batch Releases of Containment Atmosphere?

A. • Discharges directly to the plant vent stack;
• Open the Mini-Purge dampers, then start the Mini-Purge supply and exhaust fans to initiate the release.

B. • Discharges directly to the exhaust plenum;
• Open the Mini-Purge dampers to initiate the release. When CTMT to ATMOS DP is <0.25 psid, then start the Mini-Purge supply and exhaust fans.

C. • Discharges directly to the plant vent stack;
• Open Purge Filter Outlet Valve, V-294, then open the Mini-Purge dampers and start the Mini-Purge supply and exhaust fans. Then close V-294.

D✓ • Discharges directly to the exhaust plenum;
• Open Purge Filter Outlet Valve, V-294, then open the Mini-Purge dampers. When CTMT to ATMOS DP is <0.25 psid, then close V-294 and start the Mini-Purge supply and exhaust fans.

DISTRACTOR ANALYSIS:

A Incorrect. not the correct order, the incorrect release path and v294 is not being used as required.

B Incorrect. Incorrect order and v294 is not being used as required.

C Incorrect. incorrect release path and the pressure has to be checked < .25 psid to start the fans.

D Correct. per the procedure below and the prints

REFERENCES:
1. SOP-12.2 Containment Purge and Pre-access filtration system, Rev. 34/27

3.2 Open N2P13V294, PURGE FILTER COOLING OUTLET VALVE.

3.3 WHEN performing the following valve manipulations, THEN note the start time for recording purposes:

3.3.1 Place the following CTMT Purge DMPRS hand switches to MINI to initiate CTMT Batch Release:

_____ HS-3196

_____ HS-3198

_____ 3.3.2 Record start date/time data in Part III of the Batch Gaseous Waste Release Permit.

3.4 WHEN CTMT DIFF PRESSURE decreases to = 0.25 psid, THEN perform the following, noting the fan start time and containment-to-atmosphere delta pressure for recording purposes:

_/_ IV 3.4.1 Close N2P13V294, PURGE FILTER COOLING OUTLET VALVE.

_____ 3.4.2 Start MINI PURGE SUPP/EXH FAN.

_____ 3.4.3 Record fan start date/time data in Part III of the Batch Gaseous Waste Release Permit.

G2.3.9 Radiation Control
Knowledge of the process for performing a containment purge.

Question Number: 72

Tier 3 Group 3

Importance Rating: 2.5
Technical Reference: SOP-12.2, step 4.4 version 33.0
P & IDs:
5.1.1 D-205010, sheets 1 and 2, Containment Cooling and Purge System P & ID
5.1.2 D-207783, Elem. Diag., Containment Mini-Purge Fans
5.1.3 D-207204, Elem. Diag., Containment Purge Iso. Dampers Train A
5.1.4 D-207199, Elem. Diag., Containment Purge Iso. Dampers Train B
5.1.5 D-207236, Elem. Diag., Containment Purge Air Handling Unit Fan
5.1.6 D-207237, Elem. Diag., Containment Purge Exhaust Fan
5.1.7 D-204654, Conn. Diag., Containment Purge Starter Panels

Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 43.4

Comments:
This is according to SOP-12.2 appendix 3 for the batch release.
This question requires knowledge of process (procedure) for equalizing containment pressure with atmospheric pressure when initiating a containment purge. One answer choice (distractor) may result in system damage.

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

Source :	BANK	Source if Bank:	FARLEY
Cognitive Level:	LOWER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

73. G2.4.22 002

Given the following:

- FRP-H.1, Response to Loss of Secondary Heat Sink, is in progress in response to a Red Heat Sink condition.
- The crew is still progressing through FRP-H.1 when critical safety function status tree conditions are reported as follows:
  - Subcriticality: Orange
  - Core Cooling: Green
  - Heat Sink: Yellow
  - Integrity: Green
  - Containment: Red
  - Inventory: Yellow

Which ONE of the following describes the actions the crew should take in response to the conditions given above?

A. Complete FRP-H.1, then transition to FRP-S.1.

B.✓ Complete FRP-H.1, then transition to FRP-Z.1.

C. Immediately exit FRP-H.1 and transition to FRP-S.1.

D. Immediately exit FRP-H.1 and transition to FRP-Z.1.

A is incorrect. S.1 is higher priority CSF, but lower challenge (Orange).

B is correct. Entered H.1 on red condition, must complete prior to any other lower procedure.

C is incorrect. Even though heat sink is no longer red, would not immediately leave.

D same as C, except that current conditions indicate a transition to Z.1 is required.

QUESTIONS REPORT
for 75 RO Questions

G2.4.22 Emergency Procedures / Plan
Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.

Question Number: 75

Tier 3 Group 4

Importance Rating: 3.0

Technical Reference: FRP-H.1 and EOP Users Guide
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.10

Comments:

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	B A A C B A D C C B	Scramble Range: A - D	
Source :	BANK			Source if Bank:	NORTH ANNA	
Cognitive Level:	HIGHER			Difficulty:		
Job Position:	RO			Plant:	FARLEY	
reviewed:	GTO			Previous 2 NRC exams:	NO	

74. G2.4.34 001

Given the following:

- A fire required evacuation of the control room.
- The crew is performing actions of AOP-28.2, Fire in the Control Room.
- HSD Panel A is manned and functional.

The crew is at the step to adjust HIK-122, CHG FLOW, to maintain pressurizer level within the required band when the following is reported:

- HIK-122 on the HSD panel is not controlling FCV-122 properly.
- Pressurizer level is 16% and trending down.

Which ONE of the following contains a **correct method** for controlling PRZR level and the **correct location** of the components to be operated IAW AOP-28.2?

A. Close LCV-459 or 460, LTDN LINE ISO, from the HSD panel.
Control charging flow using the bypasses around FCV-122 locally in the 100 ' Piping Penetration Room entrance.

B. Close HV-8149A and 8149B or C, LTDN ORIF ISO, at the HSD panel.
Control charging flow using the bypasses around FCV-122 locally in the 100' hallway BIT area.

C. Close LCV-459 or 460, LTDN LINE ISO, from the HSD panel.
Control charging flow using MOV-8803A or B, HHSI TO RCS CL, locally in the 100' Piping Penetration Room entrance.

D.✓ Close HV-8149A and 8149B or C, LTDN ORIF ISO, at the HSD panel.
Control charging flow using MOV-8803A or B, HHSI TO RCS CL, locally in the 100' hallway BIT area.

A. incorrect. LCV-459 or 460 can not isolated at the HSDP and would not be isolated procedurally per the note below. Control of charging using the bypass valves is an option, actually the first option, (step 14.6) but in this case the first part of the distracter is not correct and the location is correct.
AOP-28.2
NOTE: Isolation of letdown due to low pressurizer level (15%) will unnecessarily complicate plant recovery (LCV 459 & 460 cannot be re-opened from the HSDP, Reactor head vents must then be used for removing mass from the primary system). Therefore, emphasis should be placed on controlling charging flow to establish a stable or slowly rising pressurizer level that compensates for any effect on level due to cooldown.

B. incorrect. Placing 2 orifices on service is correct at step 25 in the procedure for controlling level, and bypassing FCV-122 is correct, but the location is not correct.

C. incorrect. LCV-459 or 460 can not isolated at the HSDP and would not be isolated procedurally per the note below.
control charging flow using MOV8803A or B, HHSI TO RCS CL, is correct but the location is not correct.

D. Correct. Placing 2 orifices on service is correct step 25 in the procedure for controlling level, and control charging flow using MOV8803A or B, HHSI TO RCS CL, is correct and the location is correct.

G2.4.34 Emergency Procedures / Plan
Knowledge of RO tasks performed outside the main control room during emergency operations including system geography and system implications.

Question Number: 74

Tier 3 Group 4

Importance Rating: 3.8

Technical Reference: FNP-1-AOP-28.2 step 14.6 and the note above as well as step 25.2
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.10

Comments: This question tests the knowledge of an RO task outside the control room during an emergency and tests geography, plant location as well as procedural guidance and operational implications, which include the letdown portion of the question and the note describing why letdown is not isolated.

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

QUESTIONS REPORT
for 75 RO Questions

Source :	NEW	Source if Bank:	
Cognitive Level:	LOWER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO

75. G2.4.49 002

Given the following:

- The crew is performing the actions of FRP-H.1, Loss of Secondary Heat Sink, following a reactor trip due to a loss of feedwater.
- RCS pressure is 2280 psig.
- Containment pressure is 1 psig.

Which one of the following sets of steam generator wide range level parameters meet the FRP-H.1 foldout page criteria for feed and bleed for the conditions given?

A. 1A SG - 28%
1B SG - 29%
1C SG - 0%

B. 1A SG - 0%
1B SG - 13%
1C SG - 15%

C. 1A SG - 31%
1B SG - 29%
1C SG - 32%

D.✓ 1A SG - 11%
1B SG - 11%
1C SG - 14%

A. Incorrect. Plausible, 2 of 3 SG WR levels < 31% is the figure for adverse containment initiation of feed and bleed.

B. Incorrect. plausible since 2 of 3 SG WR levels < 28% was chosen since candidate could confuse with the adverse containment figure for SG NR level control of 28%. Also this could be chosen if the candidate did not remember 2 of 3 and thought it was 1 of 3 for non adverse numbers.

C. Incorrect. Plausible, 2 of 3 SG WR levels < or equal to 31% is the setpoint for a Dry SG during adverse containment. Possible candidate may confuse the setpoint. Also this could be chosen if the candidate did not remember 2 of 3 and thought it was 1 of 3 for adverse numbers or did not remember greater than 31%.

D. Correct. 2 of 3 SG WR levels < 12% requires feed and bleed with normal containment pressure conditions.

FRP-H.1 Foldout page requirements

1 Monitor bleed and feed criteria. (applicable steps 1 thru 11 only)

1.1 Check at least two SG wide range levels - GREATER THAN **12%**{31%}.	1.1 Perform the following. 1.1.1 Stop all RCPs. RCP [] 1A [] 1B [] 1C 1.1.2 Proceed to Step 12

G2.4.49 Emergency Procedures / Plan
Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

Question Number: 73

Tier 3 Group 4

Importance Rating: 4.0
Technical Reference: FRP-H.1
Proposed references to be provided to applicants during examination: None
Learning Objective:
10 CFR Part 55 Content: 41.10

Comments: meets the KA in that the question tests the FO page of H.1 which are IOAs of that procedure when those conditions require entry. This is required RO knowledge.

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

Source :	BANK	Source if Bank:	VOGTLE
Cognitive Level:	LOWER	Difficulty:	
Job Position:	RO	Plant:	FARLEY
reviewed:	GO	Previous 2 NRC exams:	NO