Omaha Public Power District
444 South 16th Street Mall
Omaha, NE 68102-2247

April 26, 2012
LIC-12-0055

Mr. Kelly Clayton
Chief Examiner, Region IV
U. S. Nuclear Regulatory Commission
1600 East Lamar Boulevard
Arlington, Texas 76011-4511

References: 1. Docket No. 50-285
2. NUREG 1021, "Operator Licensing Examination Standards for Power Reactors"

SUBJECT: NRC Licensed Operator Written Exam Reviews

In accordance with NUREG 1021, the Fort Calhoun Station Training Department has completed a review of the NRC licensed operator written exam that was conducted on April 13, 2012, at the Fort Calhoun Station.

Eleven questions on the reactor operator (RO) section of the exam were missed by 50% or more of the applicants. Six senior reactor operator questions were missed by 50% or more of the SRO applicants. These questions were reviewed for quality issues. In several cases, it was determined changes should be made to the exam or key. Attachment 1 discusses the issues associated with the questions and requested changes to the written exam and key.

For each question that was missed by 50% or more of the applicants the training material and training provided were reviewed.

All of the questions that were missed by 50% or more of the applicants were discussed with them on April 20, 2012. The applicants identified several issues that are included in the Attachment.

Fort Calhoun Station requests that the Attachment to this letter be withheld from public disclosure until after the next initial license exam.

If you require additional information, please contact Mr. Tom Giebelhausen at (402) 533-6015.

Sincerely,

D. J. Bannister
Vice President and CNO

DJB /epm

Attachment

c: M. S. Haire, Chief, Operations Branch, Region IV (w/o Attachment)
Document Control Desk (w/o Attachment)
J. C. Kirkland, NRC Senior Resident Inspector (w/o Attachment)

Attachment – FCS Exam Comments

Question 5

Given the following:

> A loss of off-site power has occurred.
> Both emergency diesel generators have started and are loaded on their respective buses.
> Breaker 1A4-10 trips, de-energizing 480V bus 1B4A

Which of the following LPSI isolation valves would **NOT** automatically reposition following a SIAS?

A. HCV-327
B. HCV-329
C. HCV-331
D. HCV-333

Key Answer: B

Fort Calhoun Station requests that this question be deleted from the exam per ES-403 D.1.b because this question is not linked to job requirements.

As discussed during exam validation, FCS believes that this question requires the recall of knowledge that is too specific for the closed reference test mode (i.e., it is not required to be known from memory.)

In the control room, operators can quickly determine the power supply to these valves by looking at the control switch labels. If an electrical bus is lost, operators are trained to use AOP-32, "LOSS OF 4160 VOLT or 480 VOLT BUS POWER," to determine the affected plant equipment. Section XII of AOP-32 addresses the loss of bus 1B4A. AOP-32 Attachment B provides a complete list of components powered from Bus 1B4A including MCC-4A1 (page 314). It also includes a list of components powered by MCC-4A1 including HCV-329. (page 324)

This K/A has a RO Importance factor of 2.7* (An importance factor barely above 2.5 with variability in the rating responses.) In referring to the asterisk, NUREG-1122, Rev 2, states "These marks indicate a need for examination developers to review plant-specific materials to determine whether or not that knowledge or ability is indeed appropriate for inclusion in any given examination." Since the power supplies are clearly indicated in the control room and the operators are trained in the use of AOP-32, requiring Operators to memorize the bus power supplied to MOVs would NOT increase protection of the health and safety of the public. Therefore, this knowledge is NOT appropriate for inclusion in the exam.

Question # _5__

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	_2___	_____
Group #	_1___	_____
K/A #	_005 K2.03______	
Importance Rating	_2.7*_	_____

K/A Statement: (005 Residual heat removal) Knowledge of the bus power supplies to the following: RCS pressure boundary motor-operated valves.

Proposed Question:

Given the following:

A loss of off-site power has occurred.
Both emergency diesel generators have started and are loaded on their respective buses.
Breaker 1A4-10 trips, de-energizing 480V bus 1B4A

Which of the following LPSI isolation valves would **NOT** automatically reposition following a SIAS?

A. HCV-327
B. HCV-329
C. HCV-331
D. HCV-333

Proposed Answer: __B___

Explanation: HCV-329 is powered by MCC-4A1, which is powered by the 1B4A 480V bus. Since that bus is deenergized, this motor operated valve would not reposition following a SIAS. The other three choices are the other three LPSI isolation valves and are plausible distracters if the applicant does not remember which motor control center powers each valve: HCV-327 is powered by MCC-3B1 (1B3B 480V bus), HCV-331 is powered by MCC-3A1 (1B3A 480V bus), and HCV-333 is powered by MCC-4C1 (1B4C 480V bus).

Technical Reference(s): STM Vol. 15 Emergency Core Cooling System Rev. 52, USAR Section 6.2, Paragraph 6.2.3.6, Rev. 35, USAR Figure 8.1-2 Rev. 1, STM Vol. 14 Electrical Distribution System Rev. 40_

(Attach if not previously provided) ________________
(including version/revision number) ________________

Proposed references to be provided to applicants during examination: __None__________

Learning Objective: ______________________ (As available)

Question Source:	Bank #	______	
	Modified Bank #	______	(Note changes or attach parent)
	New	___X___	

Question History: Last NRC Exam ___N/A______

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_X___
	Comprehension or Analysis	_____

10 CFR Part 55 Content:	55.41	__(7)__
	55.43	_____

Comments:

Attachment B

4160 Volt and 480 Volt Load List

8. Bus 1B3B-4B

CH-1C	Charging Pump
FW-8C	Condenser Evacuation Pump
HE-2	Aux Building Crane
SI-3C	CS Pump
VA-7D	Containment Vent Fan

9. Bus 1B3C

CA-1A	Air Compressor
HE-3	Turbine Building Crane
MCC-3C1	
MCC-3C2	
MCC-3C3	
SI-3A	CS Pump
T1B-3C-1	13.8KV/480V Emergency Power Transformer
T1C-3B	Outdoor Lighting Transformer Feeder Breaker

10. Bus 1B3C-4C

AC-3C	CCW Pump
MCC-3C4C-2	
VA-7C	Containment Vent Fan

11. Bus 1B4A

AC-3B	CCW Pump
CW-3B	Screen Wash Pump
FW-8B	Condenser Evacuation Pump
MCC-4A1	
MCC-4A2	
MCC-4A3	
PNL-MS	480V AC Feeder To Security Building Power Panel MS

Attachment B

4160 Volt and 480 Volt Load List

27. MCC-4A1

Proportional Heater Bank P2 Group 7
RC-3B Motor Space Heater
T1B-4A Cooling Fans

ATA-D1	D-1 Emergency 480V Feeder
ATA-D2	D-2 Normal 480V Feeder
EE-22	Rod Drive Control System Cabinet
EE-36	Valve Lapping Machine
EE-4T	Inverter 2 Bypass Transformer
EE-4U	Swing Inverter EE-8U Bypass Transformer
EE-8D	Battery Charger 2
HCV-1041C	MSIV Bypass Valve
HCV-151	PCV-102-1 Block Valve
HCV-2934	SI-6B Tank Discharge Valve
HCV-315	HPSI Loop Injection Valve
HCV-318	HPSI Loop Injection Valve
HCV-329	LPSI Loop Injection Valve
HE-7B	Auxiliary Building Elevator
RC-3B-1	RC-3B Oil Pump
VA-12B	Detector Well Cooling Fan
VA-2B	CEDM Cooling Fan
VA-45B	Room 81 Supply Fan
VA-46B	Control Room Ventilation Fan
VA-52B	D-2 Exhaust Fan
VA-71B	Battery Room 2 Exhaust Fan
WD-2B	Reactor Coolant Drain Tank Pump
WD-3B	Containment Sump Pump
VA-64B	Filter Heater

Attachment – FCS Exam Comments

Question 23

Current letdown heat exchanger outlet temperature is 113 °F as measured by temperature elements TE-2897A and TE-2897B. Assuming the 2" letdown controller TC-2897B setpoint is set at the bottom of its normal band, 8" letdown controller TC-2897A is set to its normal setpoint, and both controllers TC-2897A and TC-2897B are in the AUTO position, what is the expected response of letdown heat exchanger CCW outlet valves TCV-2897A and TCV-2897B?

A. TCV-2897A is closed; TCV-2897B is closed
B. TCV-2897A is closed; TCV-2897B is modulated open
C. TCV-2897A is modulated open; TCV-2897B is closed
D. TCV-2897A is modulated open; TCV-2897B is modulated open

Key Answer: B

Fort Calhoun Station requests that both choices B and D be accepted as correct per ES-403 D.1.b because the question stem did not provide all the information necessary to differentiate between choices B and D.

Applicant names Redacted , made this comment during the post-exam review.

The normal setpoint for TC-2897B is 110°F to 115°F with an allowable band of ± 2°F per OI-CH-1, Attachment 12. Thus the bottom of the band is 110°F.

Per OI-CH-1,TC-2897A is normally set approximately 5°F higher than TC-2897B or 115°F for the setpoint described in the stem. However, according to OI-CH-1, the setpoint could be set as high as 120°F. The allowed range for TC-2897A per IC-CP-01-2897 is ± 7.5°F, therefore its actual setpoint could be as low as 107.5°F, in which case, TCV-2897A would be open at 113°F. It could also be as high as 127.5°, in which case TCV-2897A would be closed at 113°F. Therefore both choices B and D could be correct depending on applicant assumptions.

FCS training staff concurs with this comment.

Question # _23_

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	__2__	_____
Group #	__1__	_____
K/A #	_008000 A3.04__	
Importance Rating	__2.9_	_____

K/A Statement: (008 Component Cooling Water) Ability to monitor automatic operation of the CCWS, including: Requirements on and for the CCWS for different conditions of the power plant

Proposed Question:

Current letdown heat exchanger outlet temperature is 113 °F as measured by temperature elements TE-2897A and TE-2897B. Assuming the 2" letdown controller TC-2897B setpoint is set at the bottom of its normal band, 8" letdown controller TC-2897A is set to its normal setpoint, and both controllers TC-2897A and TC-2897B are in the AUTO position, what is the expected response of letdown heat exchanger CCW outlet valves TCV-2897A and TCV-2897B?

A. TCV-2897A is closed; TCV-2897B is closed
B. TCV-2897A is closed; TCV-2897B is modulated open
C. TCV-2897A is modulated open; TCV-2897B is closed
D. TCV-2897A is modulated open; TCV-2897B is modulated open

Proposed Answer: __B___

Explanation: TCV-2897B is the smaller of the two valves and is used for fine adjustments in temperature. To minimize opening of the larger valve, its setpoint is set 5 degrees higher than the smaller valve. The bottom of the normal band for TCV-2897B's setpoint is 110 °F +/- 2 °F, therefore TCV-2897A's setpoint should be set to 115 °F. At 113°F, the controller for TCV-2897B will cause that valve to modulate open, but TCV-2897A will remain shut. Thus Answer "B" is correct. Answer "A" is plausible if the applicant believes that the normal setpoint is 115°F. Answer "C" is plausible if the applicant has the misconception that 2897A is the smaller of the two valves. Answer "D" is plausible if the applicant has the misconception that the normal setpoint of 110°F is for the larger valve and thus the smaller one is set 5 degrees cooler (105°F).

Technical Reference(s): __STM Vol. 8, "Component Cooling Water" Rev. 39, Lesson Plan 7-11-6, Rev. 15____________________

(Attach if not previously provided) ____________________

(including version/revision number) ____________________

Proposed references to be provided to applicants during examination: ______None_______

Learning Objective:	__7-11-6 1.2 **EXPLAIN** the operation of controls associated with the CCW System valves operated from the Control Room.__________________ (As available)		
Question Source:	Bank #	_______	
	Modified Bank #	_______	(Note changes or attach parent)
	New	__X__	
Question History:	Last NRC Exam	__N/A____	

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X__
10 CFR Part 55 Content:	55.41 __(7)___	
	55.43 _____	

Comments:

Attachment 12 – Letdown Temperature Control

PREREQUISITES (✓) INITIALS

1. Procedure Revision Verification

   Revision No.______ Date:________ ________

2. 120 Volt AC System is in normal operation per OI-EE-4. ________

3. Instrument Air is in operation per OI-CA-1. ________

PROCEDURE

NOTES

1. Letdown Temperature setpoint on TC-2897B is normally set between 110°F and 115°F with an allowable band of ± 2°F.

2. Letdown Temperature setpoint on TC-2897A is to be set approximately 5°F higher than TC-2897B, but at no time be greater than 120°F.

3. When Letdown Temperature is automatically controlled by TC-2897A, 8" Letdown CNTRLR, it may be necessary to place TC-2897B, 2" Letdown CNTRLR, in Manual Open to prevent oscillations from occurring between the two controllers.

1. Automatic Letdown Temperature Control

   a. IF raising Letdown Temperature, THEN perform the following steps:

      1) Verify that both Controllers are in (A) AUTOMATIC.

         - TC-2897A, 8" Letdown CNTRLR ___
         - TC-2897B, 2" Letdown CNTRLR ___ ________

      2) Adjust the SET Push-button on the 8" Letdown Temp Controller, TC-2897A to 5°F greater than the new desired Letdown Temperature setpoint. ________

      3) Raise Letdown Temperature by adjusting the SET Push-button on the 2" Letdown Temp Controller, TC-2897B to the new desired Letdown Temperature setpoint. ________

Attachment 9.3 - Data Sheet 1A

INPUT: TT-2897A*	OUTPUT: TC-2897A, Letdown Heat Exchanger CH-7 Outlet Temp Controller (CB-2)			
Applied	Desired	Allowed Range	As Found	As Left
Ohms	°F	°F	°F	°F
105.73	58.0	50.5 to 65.5		
120.38	125.0	117.5 to 132.5		
134.87	192.0	184.5 to 199.5		
120.38	125.0	117.5 to 132.5		
105.73	58.0	50.5 to 65.5		
TCV-2897A with TC-2897A in AUTO: Process above setpoint, valve opens Process below setpoint, valve closes			________ initials	________ initials
*Decade box connected at JB-761A, TB-1, Terminals 1 (H), 2 (L) & 3 (G).				

TEST EQUIPMENT			REMARKS
ID Number	Cert Date	Due Date	
I&C Tech & Date____________			

Attachment – FCS Exam Comments

Question 30

With the reactor at 100% power, main steam safety valve MS-280 fails open. What is the initial steam generator level response and the reason for that response?

A. Steam generator level decreases due to decreased recirculation flow
B. Steam generator level decreases due to increased recirculation flow
C. Steam generator level increases due to decreased recirculation flow
D. Steam generator level increases due to increased recirculation flow

Key Answer: D

Fort Calhoun Station requests that the key answer be changed to "C" per ES-403 D.1.b based on newly discovered technical information that supports a change to the answer key.

A calculation based on performance data for the FCS replacement steam generators indicates that the recirculation flow decreases from 90% to 100% power from 1.063×10^7 lb/hr at 90% power to 9.987×10^6 lb/hr at 100% power.

In the attached Table 2-9-1, values for circulation ratio, steam flow and feedwater flow at 90% and 100% power are given. In the table note, circulation ratio is defined as circulation flow rate / steam flow rate.

Therefore circulation flow rate = circulation ratio X steam flow rate.

STM 25 states "The recirculation flow plus the feed flow yield the circulation flow".

Therefore recirculation flow = circulation flow – feed flow.

Using these formulas recirculation flow @ 90% power = 1.063×10^7 lb/hr and recirculation flow @ 100% power = 9.987×10^6 lb/hr showing that recirculation flow decreases with an increase in steam flow supporting C as the correct answer.

The statement in the STM that indicates that recirculation flow increases with an increase in steam flow will be corrected.

Question # _30_

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	_2___	_____
Group #	_2___	_____
K/A #	_035 K5.03_	
Importance Rating	_2.8__	_____

K/A Statement: (035 Steam Generator) Knowledge of operational implications of the following concepts as they apply to the S/Gs: Shrink and swell concept

Proposed Question:

With the reactor at 100% power, main steam safety valve MS-280 fails open. What is the initial steam generator level response and the reason for that response?

A. Steam generator level decreases due to decreased recirculation flow
B. Steam generator level decreases due to increased recirculation flow
C. Steam generator level increases due to decreased recirculation flow
D. Steam generator level increases due to increased recirculation flow

Proposed Answer: __D___

Explanation: Per STM Vol. 25, on an excess steam demand (which occurs if an MSSV spuriously opens), steam generator pressure decreases, increasing recirculation flow and increasing downcomer level, making answer "D" the correct choice. Answer "A" would be correct if it was a load reduction instead. Answer "B" is plausible if the applicant had a misconception on the effect of recirculation flow on downcomer level. Answer "C" is plausible if the applicant had a misconception on the relationship between recirculation flow and steam demand, and the effect of recirculation flow on downcomer level.

Technical Reference(s): __STM Vol. 25 "Main Steam & Steam Generator System" Rev. 33, Lesson Plan 07-11-17, "Main Steam System" Rev. 15________

(Attach if not previously provided) ________________________________
(including version/revision number) ________________________________

Proposed references to be provided to applicants during examination: _None______

Learning Objective: _07-11-17 B. 1.1c Explain shrink and swell in a steam generator_____ (As available)

Question Source: Bank # ________
Modified Bank # ________ (Note changes or attach parent)

	New	__X____
Question History:	Last NRC Exam	__N/A___

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X__

10 CFR Part 55 Content:	55.41	__(5)__
	55.43	_____

Comments:

Page redacted
due to Proprietary
Information

surface occurs and a vapor film layer forms over the surface of the tube walls. The point at which film boiling occurs is called departure from nucleate boiling (DNB). The steam layer acts as an insulator, thus inhibiting heat transfer to the secondary side water. The mechanism of heat transfer that exists across the steam layer is radiative heat transfer. Since the radiative heat transfer mechanism is not efficient, higher heat source temperatures are required for the same rate of heat transfer, and the tube wall temperature increases abruptly when film boiling occurs.

1.41 Shrink and swell phenomenon:

- The water level in the steam generators varies during power changes above about 5% power. As transients occur, the recirculation ratio is perturbed, resulting in phenomena known as shrink and swell.
- At no load, the pressure and level in the tube wrapper is the same as in the downcomer. As steam is drawn off, a differential pressure develops across the steam separator. The level difference causes flow to the tube bundle region and results in a mass inventory decrease in the tube region. On a reactor trip the mass redistributes and results in a decrease in downcomer level.
- Another explanation is that the boiling process causes a frothing mixture of steam and water in the tube bundle region. Since a mass of steam occupies more volume than the same mass of water, the mixture height will tend to be very tall. The centrifugal separators separate out the liquid part of the flow as it passes through them.
- The liquid flow from the separators is called recirculation flow. The recirculation flow plus the feed flow yield the circulation flow. Relative to feed flow, recirculation flow is large, depending on power level. Feed flow varies directly with power level. Recirculation flow is fairly constant above 5% power, but decreases as power increases. The recirculation ratio is about 30 to 1 at 10% power, 12.5 to 1 at 50% power, and 5 to 1 at 100% power. A recirculation ratio of 5 to1 means that the dryers and separators are returning 4 pounds of water to the downcomer for each pound of steam that is leaving the steam generator. Circulation ratio (recirculation minus feedwater flow is 36 to 1 at 10% and 4 to 1 at 100%.
- Downcomer level depends more on recirculation flow than feed flow. Following a reactor trip, heat transfer to the steam generator decreases, and pressure in the steam generator increases. The flow leaving the tube bundle region decreases due to void collapse in the tube bundle region. Steam generator inventory does not change much during a trip, but the indicated level

	90%	100%
Circulation Ratio	4.56	3.99
Steam Flow	3.003E+06	3.361E+06
Feedwater Flow	3.065E+06	3.423E+06
Circulation Flow	1.369E+07	1.341E+07
Recirculation Flow	1.063E+07	9.987E+06

Attachment – FCS Exam Comments

Question 33

The reactor was just shutdown for refueling. OI-RM-1, RADIATION MONITORING is being performed to verify the area monitor setpoints for monitor RM-072, Containment Main Floor, East (1022'-10"). The as-found warn/alert setpoint was set at 20 mr/hr. The as-found high alarm setpoint was set at 30 mr/hr. These setpoint values are the same as the last time this check was performed, while the reactor was still at power.

Per OI-RM-1, the warn/alert setpoint ____(1)____ and the high alarm setpoint ____(2)____.

A. Needs adjustment; needs adjustment
B. Needs adjustment; does not need adjustment
C. Does not need adjustment; needs adjustment
D. Does not need adjustment; does not need adjustment

Key Answer: B

Fort Calhoun Station requests that this question be deleted from the exam per ES-403 D.1.b because this question is not linked to reactor operator job requirements and it is at the wrong license level.

As discussed during exam validation, this question is not linked to operator job requirements because reactor operators do not adjust the area radiation monitor setpoints.

This question has an asterisk following its importance rating. In referring to the asterisk, NUREG-1122, Rev 2, states "These marks indicate a need for examination developers to review plant-specific materials to determine whether or not that knowledge or ability is indeed appropriate for inclusion in any given examination." It is important that the operators know how to respond to a radiation monitor alarm. However, this question is not linked to operator job requirements because reactor operators do not adjust the radiation monitor setpoints. Setpoint adjustments are performed by I&C.

There is a task on the Licensed Operators task list for changing the setpoints for process radiation monitors but not for changing area monitor setpoints.

OP-3A does direct the SRO to notify I&C to adjust the area radiation monitor setpoints. In addition, the STA changes setpoints for the computer.

In addition, the K/A discusses the "Ability to manually operate and/or monitor in the control room: Alarm and interlock setpoint checks and adjustments. However, the question addresses a knowledge rather than an ability. Therefore, there is also a K/A mismatch.

Question # _33__

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_2___	_____
	Group #	_2___	_____
	K/A #	_072 A4.01_	
	Importance Rating	_3.0*_	_____

K/A Statement: **(072 Area Radiation Monitoring (ARM) System)** Ability to manually operate and/or monitor in the control room: Alarm and interlock setpoint checks and adjustments

Proposed Question:

The reactor was just shutdown for refueling. OI-RM-1, RADIATION MONITORING is being performed to verify the area monitor setpoints for monitor RM-072, Containment Main Floor, East (1022'-10"). The as-found warn/alert setpoint was set at 20 mr/hr. The as-found high alarm setpoint was set at 30 mr/hr. These setpoint values are the same as the last time this check was performed, while the reactor was still at power.

Per OI-RM-1, the warn/alert setpoint _____(1)_____ and the high alarm setpoint _____(2)_____.

A. Needs adjustment; needs adjustment
B. Needs adjustment; does not need adjustment
C. Does not need adjustment; needs adjustment
D. Does not need adjustment; does not need adjustment

Proposed Answer: ___B___

Explanation: Per OI-RM-1, the setpoints should be set to the values specified in TDB-IV-8. Per TDB-IV-8, Rev. 83, the power operation setpoints for RM-072 are 20 and 30 mr/hr. For refueling operations, these setpoints are 10 and 30 mr/hr. Since the reactor is shutdown, the as-found warn/alert setpoint is too high (20 mr/hr vs. 10 mr/hr). Therefore, the warn/alert setpoint needs adjusting. The high setpoint is correct for both power and refueling operations and therefore does not need adjusting. Thus answer "B" is correct. The other answers are incorrect but plausible if the student does not know that there are different values for this monitor since the values given in the stem are for the at power values, or if the student does not remember which values change.

Technical Reference(s): __OI-RM-1, "Radiation Monitoring", Rev. 57, TDB-IV.8, "Area Monitoring Setpoints" Rev. 83____________

(Attach if not previously provided) ______________________________
(including version/revision number) ______________________________

Proposed references to be provided to applicants during examination: _____None______

Learning Objective: ___07-12-3 4.0 **EXPLAIN** the operations, actuations and applications of the individual radiation monitors ___ (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New __X__

Question History: Last NRC Exam ___N/A_______

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X_
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __(11)___
55.43 _____

Comments:

Attachment 3 - RCS Cooldown from Mode 3 to Mode 4/5

PROCEDURE (continued) (✓) INIT/DATE

64. Stop using FC-1327 to track RCS inventory changes. ______/______

65. WHEN required,
THEN direct I&C to change Radiation Monitor setpoints from Operating to Refueling Shutdown. ______/______

66. Direct STA to reference TDB figure TDB-IV.8 and change Radiation Monitor setpoints from Operating to Refueling Shutdown. ______/______
STA

67. Contact System Engineering to evaluate the need to change taps on T1A-3 and T1A-4. ______/______

68. Align the Reactor Coolant Drain Tank for Shutdown Operation per OI-WDL-1. ______/______

Completed by____ Date/Time______/______

PROCEDURE

(✓) INIT/DATE

NOTES

1. Unless designated otherwise, all initials and signatures shall be those of the Shift Manager or Control Room Supervisor.
2. To allow for proper system operation, steps may be performed concurrently at the direction of the Shift Manager/CRS when plant conditions permit.
3. Attachment 2A is to be entered from EOP-01.

1. For Plant Shutdown from Mode 1 to Mode 2, perform Attachment 1. ____/____
2. For Plant Shutdown following a planned reactor trip perform Attachment 1A. ____/____
3. For Plant Shutdown from Mode 2 to Mode 3, perform Attachment 2. ____/____
4. For stabilizing Plant in Mode 3, perform Attachment 2A. ____/____
5. For Plant Cooldown from Mode 3 to Mode 4/5, perform Attachment 3. ____/____

REFERENCES/COMMITMENT DOCUMENTS

1. Technical Specifications:

- 2.1.1 Reactor Coolant System Operable Components
- 2.1.2 Reactor Coolant System Heatup and Cooldown Rate
- 2.1.8(3) Reactor Coolant System Vents
- 2.2 Chemical and Volume Control System
- 2.3(3) Emergency Core Cooling System Protection Against Low Temperature Overpressurization
- 2.5 Steam and Feedwater Systems
- 2.8 Refueling
- 2.10.2 Reactor Core Reactivity Control Systems and Core Physics Parameters Limits
- 3.2, Table 3-5, Item 23 Equipment and Sampling Tests

Attachment – FCS Exam Comments

Question 50

Given the following:

- Reactor has tripped
- S/G RC-2A is 540 psia and stable
- S/G RC-2A is at 30% WR level and stable
- S/G RC-2B is 480 psia and decreasing
- S/G RC-2B is 25% WR level and decreasing
- T-avg is 500°F and decreasing
- SGLS has isolated both S/Gs

What is the current expected response of AFW and the primary purpose for that response?

A. AFW will feed both steam generators to maintain heat sink
B. AFW will feed only S/G RC-2A to maintain heat sink of intact steam generator
C. AFW will feed only S/G RC-2A to minimize cooldown of RCS
D. AFW will NOT feed either steam generator to minimize cooldown of RCS

Key answer: C

Fort Calhoun Station requests that choices B and C be accepted as correct per ES-403 D.1.b because the unclear wording in the choices made choices B and C both correct.

This comment was made by Redacted) and supported by all other applicants during the post-exam review. Applicants stated they had difficulty deciding between choices B and C because both are important to reactor safety. Most picked B because the wording addressed feeding the intact steam generator rather than isolating the faulted steam generator.

The AFW system is designed to isolate the faulted S/G to minimize RCS cooldown while maintaining flow to the intact S/G to ensure RCS heat removal. AFW will be isolated from S/G RC-2B to minimize the cooldown of the RCS making the key answer, C, correct. AFW will be supplied to S/G RC-2A (i.e. both steam generators will not be isolated) to maintain the intact S/G as a heat sink to satisfy the RCS Heat Removal Safety Function making choice B equally correct.

The AFW STM supports both answers as being correct.

Per the Auxiliary Feedwater STM "The auxiliary feed actuation logic is designed to provide AFW to the intact steam generator. The additional cooldown from automatic auxiliary feedwater actuation is prevented by a low steam generator pressure condition in the ruptured steam generator and by an adequate level in the intact unit."

Question # _50_

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	__1__	_____
Group #	__1__	_____
K/A #	_000040 2.1.28_	
Importance Rating	_4.1_	_____

K/A Statement: (000040 Steam Line Rupture-Excessive Heat Transfer) Knowledge of the purpose and function of major system components and controls associated with Steam Line Rupture – Excessive Heat Transfer

Proposed Question:

Given the following:

Reactor has tripped
S/G RC-2A is 540 psia and stable
S/G RC-2A is at 30% WR level and stable
S/G RC-2B is 480 psia and decreasing
S/G RC-2B is 25% WR level and decreasing
T-avg is 500°F and decreasing
Both S/G's have been isolated

What is the current expected response of AFW and the primary purpose for that response?

A. AFW will feed both steam generators to maintain heat sink
B. AFW will feed only S/G RC-2A to maintain heat sink of intact steam generator
C. AFW will feed only S/G RC-2A to minimize cooldown of RCS
D. AFW will NOT feed either steam generator to minimize cooldown of RCS

Proposed Answer: ___C__

Explanation: Based on the above information, AFAS will start AFW flow to RC-2A only, RC-2B's pressure is less than 500 psia and so will not be fed by AFW. This is done to minimize cooldown of the RCS. AFW will also maintain the heat sink of the intact steam generator, but that is not the primary reason for the response at this time. By isolating the intact steam generator from the faulted one, this helps maintain inventory of the intact steam generator.

Technical Reference(s): _STM Vol. 19, Engineered Safeguards Control System, Rev. 37, STM Vol. 4, Auxiliary Feedwater System, Rev. 48, Lesson Plan 07-15-20, Excessive Heat Removal Events, Rev. 6__

(Attach if not previously provided) ____________________
(including version/revision number) ____________________

Proposed references to be provided to applicants during examination: __None_____

Learning Objective: __07-15-20 2.5 **EXPLAIN** the operator actions required to mitigate an Excessive Heat Removal Event.________________________ (As available)

Question Source: Bank # _______
Modified Bank # _______ (Note changes or attach parent)
New ___X___

Question History: Last NRC Exam __N/A__
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __(5)___
55.43 _____

Comments:

3.24 A complete loss of feedwater was assumed in the analysis, with a minimum RCS pressure at the beginning of the transient. No credit was taken for the low steam generator level trip. The high pressurizer pressure trip was assumed to terminate the event.

3.25 The results of the loss of feedwater flow incident analysis shows that the incident remains bounded by the loss-of-load event, and that no core damage or RCS overpressurization occurs.

••Main Steam Line Break Accident

3.26 A large break of a pipe in the Main Steam System causes a rapid depletion of steam generator inventory and an increased rate of heat extraction from the primary system. The resultant cooldown of the reactor coolant will cause a thermal margin low pressure (TMLP) trip of the reactor. A severe decrease in main steam pressure will also initiate a reactor trip on steam generator low pressure signal (SGLS) and cause the steam line isolation valves to trip closed.

3.27 If the steam line rupture occurs between the isolation valve and the steam generator outlet nozzle, blowdown of the affected steam generator would continue, but closure of the check valve in the ruptured steam line, and closure of both main steam isolation valves will terminate blowdown from the intact steam generator.

3.28 The auxiliary feed actuation logic is designed to provide AFW to the intact steam generator. The additional cooldown from automatic auxiliary feedwater actuation is prevented by a low steam generator pressure condition in the ruptured steam generator and by an adequate level in the intact unit. Refer to the Engineered Safeguards System Training Manual for further information. The design basis also assumes the turbine-driven AFW pump is unavailable after a main steam line break. The motor-driven AFW pump has sufficient capacity to fulfill the AFW flow requirements and prevent core damage provided the minimum flow recirculation is closed.

••Steam Generator Tube Rupture

3.29 A steam generator tube rupture incident is a failure of the barrier between the RCS and the Main Steam System. The major concern with respect to auxiliary feedwater would be the radiological release to atmosphere from the turbine-driven AFW pump exhaust.

3.30 The steam supply valve from the affected steam generator can be isolated to prevent unmonitored release. An air accumulator allows the valve to be closed even in the event of a loss of instrument air. Either steam generator is capable of supplying the required steam for operation of the turbine. Either AFW pump can supply the required AFW flow.

Attachment – FCS Exam Comments

Question 62

Given the following plant conditions:

A station blackout has occurred
D/G#2 has been restored and loaded
RCS pressure is 2090 psia
1 charging pump is running
WR S/G levels indicate 29% in both steam generators
FW-10 is mechanically bound
FW-54 has failed to start
Tcold has risen 9°F in the last few minutes and is continuing to increase

Which ONE of the following actions should the operators take next?

A. Start motor driven auxiliary feedwater pump FW-6 to provide feedwater to the steam generators
B. Use the demineralized water system to provide feedwater to the steam generators
C. Establish once-through cooling by opening PORV PCV-102-2 and starting HPSI pump SI-2B ONLY
D. Establish once-through cooling by aligning power as necessary to start two HPSI pumps and open both PORVs

Key answer: D

Fort Calhoun Station requests that both choices C and D be accepted as correct per ES-403 D.1.b due to an unclear stem that did not provide the applicants with information need to differentiate between choices "C" and "D."

The actions in choice "C" are part of the actions in choice "D."

The ultimate goal is to establish once-through-cooling using both PORVs and two HPSI pumps. However, If bus 1A3 is unavailable, EOP-20, HR-4 directs the operator to step 2 where he is directed to establish "partial once-through-cooling" by starting HPSI pump SI-2B and opening PCV-102-2. Therefore, choice "C" is clearly correct. In step 3, the operators are directed to crosstie buses and start HPSI pump SI-2C, Steps 8-10 directs the operators to crosstie breakers and open PORV, PCV-101-1 making the key answer "D" also correct.

Question # _62_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_1___	_____
	Group #	_2___	_____
	K/A #	_CE/E09 EK2.2_	
	Importance Rating	_3.7_	_____

K/A Statement: (CE/E09 Functional recovery) Knowledge of the interrelations between the Functional Recovery 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.

Proposed Question:

Given the following plant conditions:

A station blackout has occurred
D/G#2 has been restored and loaded
RCS pressure is 2090 psia
1 charging pump is running
WR S/G levels indicate 29% in both steam generators
FW-10 is mechanically bound
FW-54 has failed to start
Tcold has risen 9°F in the last few minutes and is continuing to increase

Which ONE of the following actions should the operators take next?

A. Start motor driven auxiliary feedwater pump FW-6 to provide feedwater to the steam generators
B. Use the demineralized water system to provide feedwater to the steam generators
C. Establish once-through cooling by opening PORV PCV-102-2 and starting HPSI pump SI-2B ONLY
D. Establish once-through cooling by aligning power as necessary to start two HPSI pumps and open both PORVs

Proposed Answer: __D___

Explanation: For the above conditions, there is currently a loss of offsite power (since DG 2 has been restored) coincident with a loss of all feedwater, since FW-6 is powered from 1A3, which would be powered by DG 1. Therefore answer "A", although plausible, is incorrect. Per EOP-20, if Tcold has an uncontrolled increase of greater than five degrees, once through cooling is initiated per HR-5. Since there is power to only one bus, power must be aligned to start a second HPSI pump and open the second PORV. FCS analysis shows that two HPSI pumps and both PORVs MUST be used in order to ensure adequate heat removal and inventory. Therefore, although there is only power to one bus, answer "C" is incorrect, because this does not establish once-through cooling (it's partial once-through cooling). This answer is plausible if the student does not know that both PORVs and two HPSI pumps must be used, and reasons that since there is only power to one bus, only those components can be operated. Answer "D" is correct based on the above discussion. Answer "B" is plausible but incorrect in that it is directed by procedure, but only if once through cooling fails for some reason, which is not provided in the stem of the question.

Technical Reference(s):	_Figure 8.1-1 "Simplified One Line Diagram Plant Electrical System P&ID", Rev. 141, TBD-EOP-20, "Functional Recover Procedure", Rev. 24, Lesson Plan 07-18-18 Rev. 17_
(Attach if not previously provided)	___________
(including version/revision number)	___________

Proposed references to be provided to applicants during examination: ____None_______

Learning Objective: __07-15-17 2.3 **EXPLAIN** the operator actions required during a total loss of feedwater event.__________________ (As available)

Question Source:	Bank #	_______
	Modified Bank #	07-15-17 025 (Note changes or attach parent)
	New	_______

Question History: Last NRC Exam ___________

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X__

10 CFR Part 55 Content:	55.41	__(5)_
	55.43	_____

Comments: This question was based off the given bank question, but the stem was modified slightly and all four answers are new.

16.0 RCS AND CORE HEAT REMOVAL

SAFETY FUNCTION: RCS and Core Heat Removal

SUCCESS PATH: Once-Through-Cooling: HR-4

RESOURCE TREE: Tree E

HR-4

INSTRUCTIONS | CONTINGENCY ACTIONS

NOTE

The action in this success path are intended to establish once-through-cooling in which both PORVs are open and at least two HPSI pumps are running. This is the desired mode of operation, however, other configurations may also be successful.

****

CAUTIONS

1. This is a heat removal path of last resort, it should only be utilized when S/G heat removal is not possible.
2. Opening PORVs may cause RCS pressure to drop.
3. Do not allow Diesel Generator loads to exceed power and current rating limits.

****

✘1. **IF** both Vital 4160 V Buses are energized,
THEN establish once-through-cooling by performing the following:

a. Stop all RCPs.

(continue)

1.1 **IF** once-through-cooling can **NOT** be established because a Vital 4160 V bus is deenergized,
THEN determine the appropriate method of recovery by performing step a, b, c or d:

(continue)

16.0 RCS AND CORE HEAT REMOVAL

HR-4

INSTRUCTIONS

✘1. (continued)

b. Deenergize all PZR Heaters.

c. Initiate PPLS by placing the following switches in "TEST":

- 86A/PPLS TEST SWITCH
- 86B/PPLS TEST SWITCH

d. Ensure **ALL** of the available pumps have started:

- Either HPSI Pumps, SI-2A/B or SI-2B/C
- Charging Pumps, CH-1A/B/C

e. Ensure all of the HPSI Loop Injection Valves are open.

f. Ensure **BOTH** of the PORV Block Valves are open:

- HCV-150
- HCV-151

(continue)

CONTINGENCY ACTIONS

1.1 (continued)

a. **IF** 1A3 is deenergized, **AND BOTH** of the PORV Block Valves are open:

- HCV-150
- HCV-151

THEN GO TO Step 2.

b. **IF** 1A4 is deenergized, **AND BOTH** of the PORV Block Valves are open:

- HCV-150
- HCV-151

THEN GO TO Step 12.

(continue)

16.0 RCS AND CORE HEAT REMOVAL

INSTRUCTIONS	CONTINGENCY ACTIONS
✖1. (continued)	1.1 (continued)
g. Open the PORVs. Time: ___________	c. **IF** 1A3 is deenergized, **AND ANY** of the PORV Block Valves are closed: • HCV-150 • HCV-151 **THEN** GO TO Step 21.
h. GO TO Step 60.	d. **IF** 1A4 is deenergized, **AND ANY** of the PORV Block Valves are closed: • HCV-150 • HCV-151 **THEN** GO TO Step 42.

16.0 RCS AND CORE HEAT REMOVAL

HR-4

INSTRUCTIONS	CONTINGENCY ACTIONS

✱ 2. Establish partial once-through-cooling by performing the following:

a. Stop all RCPs.

b. Deenergize all PZR Heaters.

c. Open **ALL** of the following HPSI Loop Isolation Valves:

- HCV-315
- HCV-318
- HCV-312
- HCV-321

d. Start SI-2B, HPSI Pump.

e. Open PCV-102-2, PORV.

Time: ____________

16.0 RCS AND CORE HEAT REMOVAL

INSTRUCTIONS	CONTINGENCY ACTIONS

3. Restore power to SI-2C, HPSI Pump, by performing the following:

    a. Place CA-1C, Air Compressor, in "PULL OUT".

    b. Open BT-1B3A.

    c. Close BT-1B4A.

    d. Start SI-2C, HPSI Pump.

    Time: ____________

4. Open **ALL** of the following breakers (East Switchgear Room):

    - 1B3C-4C-2, "MCC-3C4C-2 TURB BLDG (MEZZANINE)"
    - 1B3C-4C-3, "CONTAINMENT COOLING FAN VA-7C"
    - 1B3C-4C-4, "COMPONENT COOLING WATER PUMP AC-3C"

16.0 RCS AND CORE HEAT REMOVAL

INSTRUCTIONS	CONTINGENCY ACTIONS
5. Open **ALL** of the following breakers (East Switchgear Room): • 1B3C-8, "AIR COMPRESSOR CA-1A FEED TO LOCAL CONTACTOR" • 1B3C-6, "CONTAINMENT SPRAY PUMP SI-3A" • 1B3C-3, "OUTDOOR LIGHTING XFMR T1C-3B" • 1B3C-5, "MCC-3C3 SERVICE BLDG (3RD FLOOR)" • 1B3C-2, "MCC-3C2 AUX BUILDING (CORR 26)" • 1B3C-7, "TURB BLDG CRANE HE-3"	

16.0 RCS AND CORE HEAT REMOVAL

HR-4

INSTRUCTIONS | CONTINGENCY ACTIONS

6. Place **ALL** of the following breakers in "OFF" (East Upper Electrical Penetration Room):

- MCC-3C1-A2L, "EE-8E BATTERY CHARGER NUMBER 3"
- MCC-3C1-A2R, "AUX BLDG ROOF D-S, AUX BLDG ROOF STRESS TEST DISC SWITCH"
- MCC-3C1-A3L, "MPP-58/EE-98 MOTOR PROTECTION PANEL TRANSFORMER"
- MCC-3C1-A3R, "HE-12-DS/ STRESS GALL-DS ROOM 66 HOIST & STRESS GALLERY TEST DISC SWITCH"
- MCC-3C1-A4R, "EE-4Q INVERTER "C" EE-8K BYPASS TRANSFORMER"
- MCC-3C1-A05, "TRANSFORMER T1B-3C COOLING FANS"
- MCC-3C1-B01, "PRESSURIZER BACK-UP HEATERS BANK 2, GROUP 4"
- MCC-3C1-C01, "PRESSURIZER BACK-UP HEATERS BANK 2, GROUP 5 FUSES"

16.0 RCS AND CORE HEAT REMOVAL

INSTRUCTIONS	CONTINGENCY ACTIONS
7. Open **BOTH** of the 4.16 KV supply breakers to 1B3C: • T1B-3C • 1B3C	
8. Energize MCC-3C1 by performing the following: a. Close BT-1B4C. b. Close BT-1B3C. c. Ensure breaker 1B3C-1, "MCC-3C1 ELECT. PENET. AREA (RM 57E)" (East Switchgear Room) is closed.	
9. Ensure breaker MCC-3C1-A01, "PCV-102-1 PZR POWER OPERATED RELIEF VALVE" (East Upper Electrical Penetration Room) is closed.	
10. Ensure PCV-102-1, PORV, is open. Time: ____________	

Attachment – FCS Exam Comments

Question 67

Which one of the following instrument errors would cause the reactor power calculated by XC-105 to be greater than actual reactor power?

A. Feedwater pressure indicating lower than actual.
B. Feedwater temperature indicating higher than actual.
C. Feedwater flow indicating higher than actual.
D. Reactor Coolant Pump Power indicating lower than actual

Key Answer: C

Fort Calhoun Station requests that choices C and D be accepted as correct per ES-403 D.1.b because newly discovered technical information supports this change to the key.

This comment was made by Redacted) during the post-exam review.

The calorimetric equation is:

$$Q_{RX} = M_{fw}(h_{stm} - h_{FW}) - Q_{RCP} + Q_{losses}$$

Feedwater flow indicating higher than actual will cause the calculated reactor power to be greater than the actual reactor power making the key answer, "C," correct.

RCP power indicating lower than actual would also cause the calculated reactor power to be greater than actual reactor power making choice "D" also correct.

Choice D being correct is consistent with Question P2485 from the PWR GFE bank- August, 2011 (next page)

RE-CPT-RX-0003, Attachment 9.1 also supports choices C and D. It indicates that having feedwater flow greater than actual will have a conservative effect (XC-105 will indicate higher that actual power). It also indicates that having RCP power less than actual will have a conservative effect on XC-105.

FCS training staff concurs with this comment.

Question # _67_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	__3__	_____
	Group #	_____	_____
	K/A #	2.2.2____	
	Importance Rating	_4.6__	_____

K/A Statement: Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.

Proposed Question:

Which one of the following instrument errors would cause the reactor power calculated by XC-105 to be greater than actual reactor power?

A. Feedwater pressure indicating lower than actual.
B. Feedwater temperature indicating higher than actual.
C. Feedwater flow indicating higher than actual.
D. Reactor Coolant Pump Power indicating lower than actual

Proposed Answer: __C__

Explanation: This question's concept was taken from bank question 07-12-19 009 used on the 1999 NRC RO exam. The direction of XC-105 vs actual power was changed in the stem and the correct answer was changed as well as two distracters were changed. See parent question below and reference chart from RE-CPT-RX-0003, rev 17, page 11.

Technical Reference(s): RE-CPT-RX-0003, rev 17, page 11__
(Attach if not previously provided) ______________________
(including version/revision number) ______________________

Proposed references to be provided to applicants during examination: __None________

Learning Objective: __0712-19 (As available)

Question Source: Bank # ______
Modified Bank # ___07-12-009___
New ______

Question History: Last NRC Exam _______
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 (4)
55.43 ____

Comments:

Bank question from 1999 exam

07-12-19 009

Which one of the following instrument errors would cause the reactor power calculated by XC-105 to be less than actual reactor power?

A. Turbine first stage pressure indicating lower than actual.
B. Feedwater temperature indicating higher than actual.
C. Feedwater flow indicating higher than actual.
D. Main generator electrical output indicating less than actual.

Correct Answer: B

KA#: 015000 A1.01 Bank Reference #:
LP# / Objective: 0712-19 Exam Level:
Cognitive Level: HIGH Source: NRC FCS 1999
Reference: LP 0753.02 Handout: NONE

TOPIC: 193007
KNOWLEDGE: K1.06 [3.1/3.3]
QID: P2485 (B2684)

The power range nuclear instruments have been adjusted to 100 percent based on a heat balance calculation. Which one of the following will result in indicated reactor power being higher than actual reactor power?

A. The feedwater temperature used in the heat balance calculation was 20°F higher than actual feedwater temperature.

B. The reactor coolant pump heat input term was omitted from the heat balance calculation.

C. The feedwater flow rate used in the heat balance calculation was 10 percent lower than actual feedwater flow rate.

D. The ambient heat loss term was omitted from the heat balance calculation.

ANSWER: B.

TOPIC: 193007
KNOWLEDGE: K1.06 [3.1/3.3]
QID: P2685 (B2284)

The power range nuclear instruments have been adjusted to 100 percent based on a heat balance calculation. Which one of the following will result in indicated reactor power being lower than actual reactor power?

A. The feedwater temperature used in the heat balance calculation was 20°F higher than actual feedwater temperature.

B. The reactor coolant pump heat input term was omitted from the heat balance calculation.

C. The feedwater flow rate used in the heat balance calculation were 10 percent higher than actual flow rates.

D. The operator miscalculated the enthalpy of the steam exiting the steam generators to be 10 Btu/lbm higher than actual.

ANSWER: A.

Attachment 9.1 - XC105 Inputs and Conservative Effects

Parameter	ERF Point	Approximate 100% Value	Lowest Value	Highest Value	Conservative Effect	Reason
Feedwater Pressure	P1141	1000 psia	15.7 psia	2000 psia	Greater than Actual	Specific volume lowers which in turn raises lbm/hr flow from the SG. A higher enthalpy value lowers SG output, but does not offset the higher FW flow.
Feedwater Flow	F1395 (RC-2A)	360 in-H_2O 3.3E6 lbm/hr	1E-10 in H_2O	400 in H_2O	Greater than Actual	Greater output from SG.
	F1398 (RC-2B)		1E-10 in H_2O	400 in H_2O		
Steam Generator Pressure	P0902A (RC-2A)	820 psia	0.1 psia	1000 psia	Less than Actual	Higher steam enthalpy value raises SG output calculation and is not offset by the lower liquid enthalpy
	P0905A (RC-2B)		0.1 psia	1000 psia		
Feedwater Temperature	T1396 (RC-2A)	443 °F	35°F	700°F	Less than Actual	Specific volume lowers which in turn raises lbm/hr flow from the SG. A lower enthalpy also raises SG output.
	T1399 (RC-2B)		35°F	700°F		
SG Blowdown Flow	F1392 (RC-2A)	2.8 in-H_2O 2.0E4 lbm/hr	1E-10 in H_2O	6.32 in H_2O	Less than Actual	Reduces the blowdown loss from SG output calculation. In reality, a lower blowdown flow would have a corresponding reduction in FW flow.
	F1394 (RC-2B)		1E-10 in H_2O	6.32 in H_2O		
SG Blowdown Temperature	T1391 (RC-2A)	500 °F	200°F	600°F	Greater than Actual	Specific volume of the liquid rises which in turn reduces the mass flowrate of blowdown. The higher enthalpy also has a positive effect on SG output.
	T1393 (RC-2B)		200°F	600°F		
Reactor Coolant Pump Power	Y3268 (RC-3A)	2000 KW	0 kW	4200 kW	Less than Actual	A lower heat input from the reactor coolant pumps makes the perceived contribution of reactor higher and raises the overall power.
	Y3269 (RC-3B)		0 kW	4200 kW		
	Y3270 (RC-3C)		0 kW	4200 kW		
	Y3271 (RC-3D)		0 kW	4200 kW		

Attachment – FCS Exam Comments

Question 86

The reactor is currently at 100% power. It was determined at 0815 today, April 13th, that one safety injection tank had a boron concentration that was less than the refueling concentration. Assuming boron concentration cannot be restored to within limits, the plant must be in HOT SHUTDOWN no later than ______(1)______.

A. April 14th at 2015
B. April 15th at 0815
C. April 16th at 2015
D. April 17th at 0815

Key answer: C

Fort Calhoun Station requests that answers A and C be accepted as correct per ES-403 D.1.b because the question stem did not provide all of the information necessary to differentiate between choices A and C.

Numerous SRO applicants made this comment during the post-exam review.

The stem states that "assuming boron concentration cannot be restored to within limits," and does not provide a reason why normal SIT boration cannot be performed. Most applicants made a similar assumption that a problem with valves, interlocks or piping is preventing restoration of boron concentration.

Technical Specifications 2.3(2)e through h state:

e. Any valve, interlock or piping associated with the safety injection and shutdown cooling system which is not covered under d. above but which is required to function during accident conditions may be inoperable for a period of no more than 24 hours.

f. One safety injection tank may be inoperable for reasons other than g. or h. below for a period of no more than 24 hours.

g. Level and/or pressure instrumentation on one safety injection tank may be inoperable for a period of 72 hours.

h. One safety injection tank may be inoperable due to boron concentration not within limits for a period of no more than 72 hours."

Technical Specification 2.3(2) contains the following statement:

"If the system is not restored to meet the minimum requirements within the time period specified below, the reactor shall be placed in a hot shutdown condition within 12 hours."

Question 86 (continued)

Therefore, if an applicant assumed that there was a problem with piping, valves or interlocks, then Technical Specifications 2.3(2)e or 2.3(2)f would apply. Both of these situations are 24 hour LCO's.

If These LCO's are not met, then the plant must be placed in hot shutdown within 12 hours per Technical Specification 2.3(2) for a total of 36 hours to get to hot shutdown. Therefore, 4/13 @ 0815 + 36 hours = 4/14 @ 2015 making choice "A" also correct.

FCS Training staff concurs that the assumption made by several of the applicants is reasonable based on needed information not provided in the stem and that "A" should also be accepted as correct.

Question # _86___

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #	_____	__2___
Group #	_____	__1___
K/A #		006_G2.2.38_
Importance Rating	_____	__4.5__

K/A Statement: (006 Emergency Core Cooling) **2.2.38 Knowledge of conditions and limitation in the facility license.**

Proposed Question:

The reactor is currently at 100% power. It was determined at 0815 today, April 13th, that one safety injection tank had a boron concentration that was less than the refueling concentration. Assuming boron concentration cannot be restored to within limits, the plant must be in HOT SHUTDOWN no later than _____(1)_____.

A. April 14th at 2015
B. April 15th at 0815
C. April 16th at 2015
D. April 17th at 0815

Proposed Answer: __C__

Explanation: Per T.S. 2.3(2)(h), one safety injection tank may be inoperable due to boron concentration not within limits for a period of no more than 72 hours. Per T.S. 2.3(2), If system is not restored to meet the minimum requirements in time period specified, reactor shall be placed in hot shutdown within 12 hours. Per T.S. 2.3(1)(c), the safety injection tanks must have refueling boron concentration. Therefore, for the question stem above, the plant is in T.S. 2.3(2)(h). If the concentration cannot be restored, the plant must be in hot shutdown 84 hours after entry into T.S. This would make answer C correct. The other answers are incorrect but plausible due to using different LCO times, specifically, 24 hours instead of 72 as would be the case if the SIT was inoperable for reasons other than boron or level/pressure instrumentation, and 24 hours to shutdown instead of 12, as is the case for entering cold shutdown.

Technical Reference(s): _T.S. 2.3 (Amendment 221)
(Attach if not previously provided) ______________________
(including version/revision number) ______________________

Proposed references to be provided to applicants during examination: ______None_____

Learning Objective: ______________________ (As available)

Question Source:
Bank # ________
Modified Bank # ________ (Note changes or attach parent)
New ___X___

Question History: Last NRC Exam _____N/A___

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ____

Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 _____

55.43 _(1)__

Comments:

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.3 Emergency Core Cooling System (Continued)

(2) Modification of Minimum Requirements

During power operation, the Minimum Requirements may be modified to allow one of the following conditions to be true at any one time. If the system is not restored to meet the minimum requirements within the time period specified below, the reactor shall be placed in a hot shutdown condition within 12 hours. If the minimum requirements are not met within an additional 48 hours the reactor shall be placed in a cold shutdown condition within 24 hours.

a. One low-pressure safety injection train may be inoperable provided the train is restored to operable status within seven (7) days.

b. One high-pressure safety injection pump may be inoperable provided the pump is restored to operable status within 24 hours.

c. One shutdown heat exchanger may be inoperable for a period of no more than 24 hours.

d. Any valves, interlocks or piping directly associated with one of the above components and required to function during accident conditions shall be deemed to be part of that component and shall meet the same requirements as listed for that component.

*** e. Any valve, interlock or piping associated with the safety injection and shutdown cooling system which is not covered under d. above but which is required to function during accident conditions may be inoperable for a period of no more than 24 hours.

f. One safety injection tank may be inoperable for reasons other than g. or h. below for a period of no more than 24 hours.

g. Level and/or pressure instrumentation on one safety injection tank may be inoperable for a period of 72 hours.

h. One safety injection tank may be inoperable due to boron concentration not within limits for a period of no more than 72 hours.

***SEE TDB-VIII

Attachment – FCS Exam Comments

Question 89

The reactor was operating at full power when a fire was reported in Room 18 (CCW Heat Exchanger Room). The reactor has been tripped and plant status is as follows:

- Raw water pumps AC-10A and AC-10C are running
- CCW Heat Exchanger AC-1C is operating with a CCW exit temperature of 125 degrees F
- Procedure AOP-06 FIRE EMERGENCY FOR AUXILIARY BUILDING RADIATION CONTROLLED AREAS AND CONTAINMENT was entered

What should the control room supervisor direct the crew to do to achieve and maintain the plant in a safe shutdown condition and what procedure should be used to complete these actions?

A. Implement Procedure AOP-11, LOSS OF COMPONENT COOLING WATER and establish feed and bleed cooling for the CCW system components
B. Implement Procedure AOP-18, LOSS OF RAW WATER and use the fire protection system water to cool the CCW system components
C. Stay in Procedure AOP-06 and establish feed and bleed cooling for the CCW system components
D. Stay in Procedure AOP-06 and use Demineralized water to cool the CCW system components

Key answer: B

Fort Calhoun Station requests that the key answer be changed to "C" per ES-403 D.1.b because of newly discovered technical information that supports a change to the answer key.

Redacted.) made the following comment during the post-exam review.

AOP-18, Attachment B, "Fire Protection System Backup" steps 2 and 3 require that hoses be run from BOTH Fire Hose Cabinets FP-7C (Room 19) AND FP-7D (Corridor 4). Applicants eliminated the key answer ,"B," because for a fire in Room 18, Hose Cabinet FP-7D would be used for fire fighting and would not be available for Fire Protection System Backup.

AOP-18, Contingency step 12.1 states "IF the Fire Protection System is NOT available, THEN GO TO Step 14. Step 14 directs the establishment of feed and bleed cooling by opening a CCW drain valve. AOP-06 would not be exited, making choice "C" the correct answer.

The FCS Training staff concurs with this comment.

Question # _89___

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #	_____	__2___
Group #	_____	__1___
K/A #		008_A2.03_
Importance Rating	_____	__3.2__

K/A Statement: (008 CCWS) **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;** A2.03 High/Low CCW Temperature

Proposed Question:

The reactor was operating at full power when a fire was reported in Room 18 (CCW Heat Exchanger Room). The reactor has been tripped and plant status is as follows:

- Raw water pumps AC-10A and AC-10C are running
- CCW Heat Exchanger AC-1C is operating with a CCW exit temperature of 125 degrees F
- Procedure AOP-06 FIRE EMERGENCY FOR AUXILIARY BUILDING RADIATION CONTROLLED AREAS AND CONTAINMENT was entered

What should the control room supervisor direct the crew to do to achieve and maintain the plant in a safe shutdown condition and what procedure should be used to complete these actions?

A. Implement Procedure AOP-11, LOSS OF COMPONENT COOLING WATER and establish feed and bleed cooling for the CCW system components
B. Implement Procedure AOP-18, LOSS OF RAW WATER and use the fire protection system water to cool the CCW system components
C. Stay in Procedure AOP-06 and establish feed and bleed cooling for the CCW system components
D. Stay in Procedure AOP-06 and use Demineralized water to cool the CCW system components

Proposed Answer: ___B___

Explanation:

Answer B is correct because AOP-06 directs the crew to implement AOP-18 for loss of Raw Water because it contains the actions to use fire protection water since CCW equipment is not credited during a fire. AOP-06 does not contain the specific steps to align fire protection water to the CCW system (AOP-18 has these actions).

Distractor A is credible but not correct because CCW is the system lost and so it would be reasonable to assume that you should go to this procedure but it is not credited in fires. DIstractor C and D are incorrect because you must transition out of AOP-06 and itto AOP-18 in order to get cooling to the components in the CCW system via fire protection water.

Technical Reference(s): AOP-06-01 Rev. 1, AOP-11 "Loss of Component Cooling Water" Rev. 15, AOP-18 "Loss of Raw Water" Rev. 7

(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: __None______

Learning Objective: ____________________ (As available)

Question Source: Bank # _______
Modified Bank # _______ (Note changes or attach parent)
New ___X___

Question History: Last NRC Exam __N/A________

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 _____
55.43 _(5)__

Comments:

Attachment B

Fire Protection System Backup

INSTRUCTIONS | CONTINGENCY ACTIONS

1. Inform Security and the RP Technician that the door between Room 18 and Room 19 will be open.

NOTE

Hoses and couplings (180° coupling for AC-1A, straight hose for AC-1B and a 90° coupling for AC-1C or AC-1D) required for connecting RW/CCW HX drains to the Fire Protection System are located in the AI-100 Supply Cabinet.

2. Connect a hose from FP-418, "FIRE HOSE CABINET FP-7C 2 1/2" AUX HOSE CONNECTION VALVE" (Room 19), to **ONE** of the following RW/CCW HX Inlet Drain Valves:

   - RW-213, "CCW HEAT EXCHANGER AC-1A DRAIN VALVE" (Corridor 4)
   - RW-197, "CCW HEAT EXCHANGER AC-1B DRAIN VALVE" (Corridor 4)
   - RW-214, "CCW HEAT EXCHANGER AC-1C DRAIN VALVE" (Room 18)
   - RW-215, "CCW HEAT EXCHANGER AC-1D DRAIN VALVE" (Room 18)

Attachment B

Fire Protection System Backup

INSTRUCTIONS | CONTINGENCY ACTIONS

NOTE

Hoses and couplings (180° coupling for AC-1A, straight hose for AC-1B and a 90° coupling for AC-1C or AC-1D) required for connecting RW/CCW HX drains to the Fire Protection System are located in the AI-100 Supply Cabinet.

3. Connect a hose from FP-428, "FIRE HOSE CABINET FP-7D 2 1/2" AUX HOSE CONNECTION VALVE" (Corridor 4), to **ONE** of the following RW/CCW HX Inlet Drain Valves:

   - RW-213, "CCW HEAT EXCHANGER AC-1A DRAIN VALVE" (Corridor 4)
   - RW-197, "CCW HEAT EXCHANGER AC-1B DRAIN VALVE" (Corridor 4)
   - RW-214, "CCW HEAT EXCHANGER AC-1C DRAIN VALVE" (Room 18)
   - RW-215, "CCW HEAT EXCHANGER AC-1D DRAIN VALVE" (Room 18)

INSTRUCTIONS	CONTINGENCY ACTIONS

NOTE

Technical Specification 2.0.1, General Requirements, requires RCS T_C to be less than 300°F within six hours of the shutdown.

10. Commence RCS cooldown to less than 300°F PER OP-3A, Plant Shutdown.

11. Reduce CCW Heat Loads PER Attachment A, CCW System Heat Loads.

12. Establish alternate cooling for the CCW System from the Fire Protection System PER Attachment B, Fire Protection System Backup.

    12.1 **IF** the Fire Protection System is **NOT** available, **THEN** GO TO Step 14.

13. **IF** alternate cooling is established, **THEN** GO TO Step 15.

INSTRUCTIONS | CONTINGENCY ACTIONS

****

CAUTIONS

1. No more than one drain valve should be opened while performing CCW System feed and bleed. Makeup for the CCW Surge Tank is only supplied through 1½ inch line.

2. CCW is considered toxic and potentially radioactive.

****

14. **IF** adequate water inventory exists for makeup to the CCW system, **THEN** establish feed and bleed cooling for CCW by performing the following steps:

    a. Throttle open **ONE** of the following CCW drains:

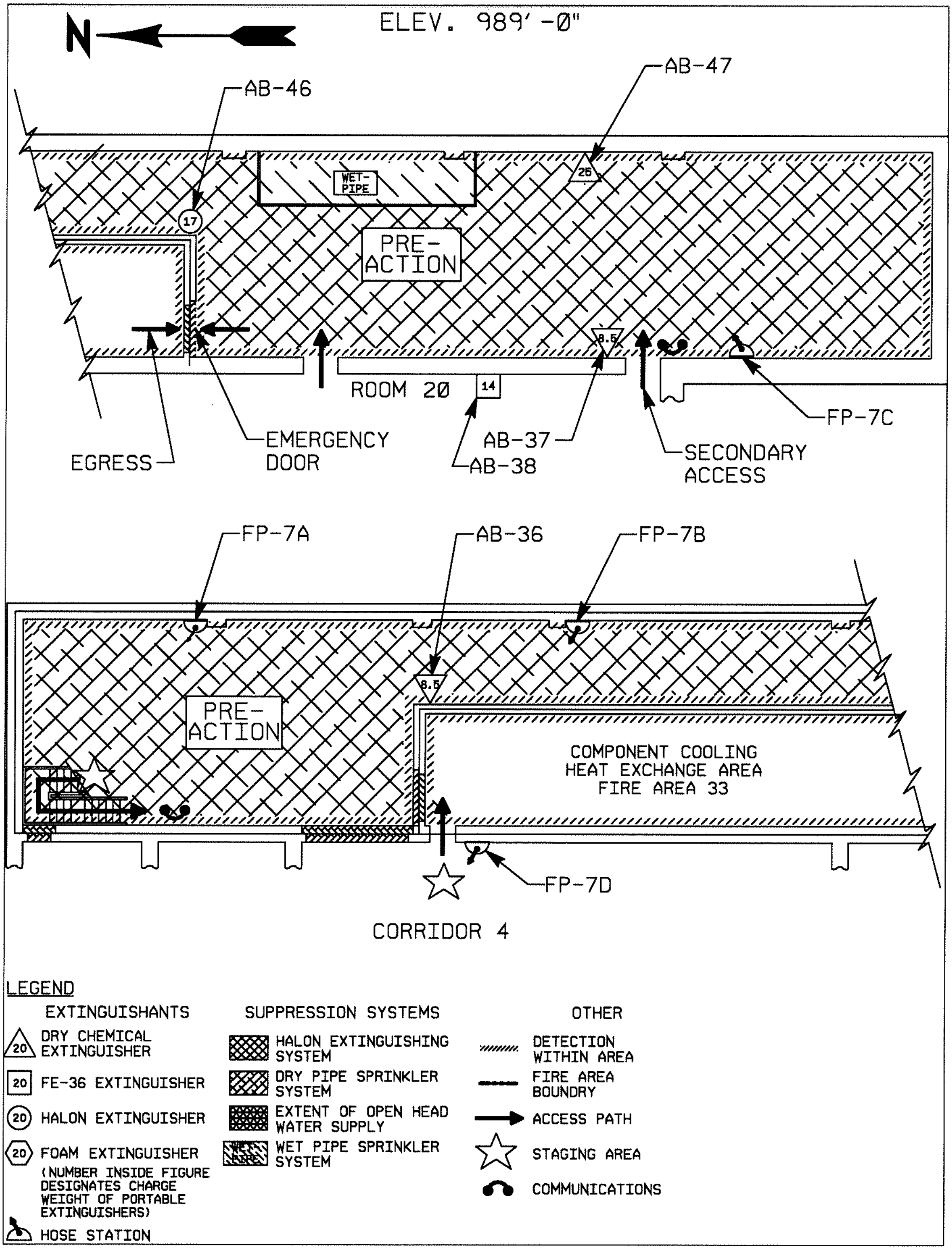
        - AC-225, "COMP COOLING HT EXCH AC-1C CCW INLET DRAIN VALVE" (Room 18)

        - AC-226, "COMP COOLING HT EXCH AC-1A CCW INLET DRAIN VALVE" (Corridor 4)

    b. Monitor the CCW Surge Tank's level and pressure.

Attachment 2 - Auxiliary Building Radiation Controlled Areas

Component Cooling Heat Exchanger Area (Rm 18)



Fort Calhoun Station's Initial Post Exam Comments submittal was Unsat because it did not analyze all questions that were missed by 50% or more of the class, which is required by NUREG-1021.

The chief examiner called FCS and requested a second submittal, which the station completed and provided via e-mail the following week on May 8th, 2012.

The pages behind this sheet are the e-mail with date submitted and the corresponding analysis with applicant names marked out.

From: GIEBELHAUSEN, THOMAS E
To: Clayton, Kelly
Subject: FW: Exam analysis with applicant comments
Date: Tuesday, May 08, 2012 8:38:24 AM
Attachments: Exam Analysis with applicant comments.pdf

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fyi

From: KOSKE, JERRY E
Sent: Thursday, May 03, 2012 12:01 PM
To: GIEBELHAUSEN, THOMAS E
Cc: CADE, RANDALL B
Subject: Exam analysis with applicant comments

Hi Tom

Here is the document to send to Kelly.

It is a pdf file that has the exam analysis followed by the applicant comments.

Jerry

2012 NRC WRITTEN EXAM GRADING and ANALYSIS

QUESTION #	ANSWER KEY	ALT. ANSWER	R1	R2	R3	U1	U2	I1	I2	I3	I4	I5
1	d		d	d	d	d	d	d	d	d	d	d
2	a		a	c	a	a	c	a	a	a	c	d
3	a		a	a	a	a	a	a	a	a	a	a
4	d		d	b	d	b	d	b	d	b	d	d
5	b		c	c	b	c	c	c	c	c	b	c
6	a		a	a	a	a	a	a	a	a	a	a
7	a		d	a	a	a	a	c	a	a	a	a
8	c		c	c	c	c	c	c	c	c	c	c
9	b		a	b	b	b	b	b	b	b	b	b
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14	b		b	b	b	b	b	b	b	b	b	b
15	c		c	c	c	c	c	c	c	c	c	c
16	a		a	a	a	a	a	a	a	a	a	a
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19	d		d	d	b	d	d	d	d	d	a	d
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21	d		d	d	d	b	d	d	d	d	d	d
22	b		b	b	a	b	b	a	a	b	b	b
23	b		d	b	c	b	b	c	d	b	d	c
24	c		c	c	a	c	c	a	c	a	c	c
25	b		b	b	b	b	b	b	b	b	b	b

	R1	R2	R3	U1	U2	I1	I2	I3	I4	I5	% CORRECT	ITEM ANALYSIS "a"	"b"	"c"	"d"
POINTS	75	75	75	100	100	100	100	100	100	100					
CORRECT	58	63	61	79	87	72	77	81	82	66					
SCORE	77.3%	84.0%	81.3%	79.0%	87.0%	72.0%	77.0%	81.0%	82.0%	66.0%					
1	1	1	1	1	1	1	1	1	1	1	100%	0	0	0	10
2	1	0	1	1	0	1	1	1	0	0	60%	6	0	3	1
3	1	1	1	1	1	1	1	1	1	1	100%	10	0	0	0
4	1	0	1	0	1	0	1	0	1	1	60%	0	4	0	6
5	0	0	1	0	0	0	0	0	1	0	20%	0	2	8	0
6	1	1	1	1	1	1	1	1	1	1	100%	10	0	0	0
7	0	1	1	1	1	0	1	1	1	1	80%	8	0	1	1
8	1	1	1	1	1	1	1	1	1	1	100%	0	0	10	0
9	0	1	1	1	1	1	1	1	1	1	90%	1	9	0	0
10	1	1	1	1	1	1	1	1	1	1	100%	0	10	0	0
11	1	1	1	1	1	1	1	1	1	1	100%	0	10	0	0
12	1	1	1	1	1	1	1	1	1	1	100%	0	10	0	0
13	1	1	1	1	1	1	1	1	1	1	100%	0	0	0	10
14	1	1	1	1	1	1	1	1	1	1	100%	0	10	0	0
15	1	1	1	1	1	1	1	1	1	1	100%	0	0	10	0
16	1	1	1	1	1	1	1	1	1	1	100%	10	0	0	0
17	1	1	1	1	1	1	1	1	1	0	90%	1	9	0	0
18	0	1	1	1	0	0	1	0	1	1	60%	0	6	0	4
19	1	1	0	1	1	1	1	1	0	1	80%	1	1	0	8
20	1	0	0	1	1	0	0	1	1	1	60%	1	3	0	6
21	1	1	1	0	1	1	1	1	1	1	90%	0	1	0	9
22	1	1	0	1	1	0	0	1	1	1	70%	3	7	0	0
23	0	1	0	1	1	0	0	1	0	0	40%	0	4	3	3
24	1	1	0	1	1	0	1	0	1	1	70%	3	0	7	0
25	1	1	1	1	1	1	1	1	1	1	100%	0	10	0	0

2012 NRC WRITTEN EXAM GRADING and ANALYSIS

QUESTION #	ANSWER KEY	ALT. ANSWER	R1	R2	R3	U1	U2	I1	I2	I3	I4	I5
26	c		c	c	c	c	c	c	c	c	c	c
27	c		c	c	c	b	c	c	c	c	c	c
28	a		a	a	a	a	a	a	a	a	a	a
29	b		b	b	b	b	b	b	c	b	b	b
30	d		c	c	d	c	d	d	d	a	c	a
31	a		a	a	a	a	a	a	a	a	a	a
32	a		a	a	a	a	d	a	a	a	a	a
33	b		d	b	a	d	a	d	a	a	a	d
34	d		d	d	d	d	d	d	d	d	d	d
35	d		d	b	a	a	d	d	d	d	d	b
36	d		d	d	d	d	d	d	d	a	d	d
37	b		b	d	b	d	b	c	d	c	a	a
38	c		a	c	c	c	c	c	c	a	c	c
39	b		b	b	b	b	b	b	b	b	b	b
40	c		a	c	c	a	c	c	c	c	c	d
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43	a		a	a	a	a	a	a	a	a	a	a
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47	c		c	c	b	c	c	c	c	c	c	c
48	a		a	a	a	a	a	a	a	a	a	a
49	b		b	b	b	b	b	b	b	b	b	b
50	c		c	b	b	b	c	b	a	c	b	b

	R1	R2	R3	U1	U2	I1	I2	I3	I4	I5	% CORRECT	"a"	"b"	"c"	"d"
POINTS	75	75	75	100	100	100	100	100	100	100					
CORRECT	58	63	61	79	87	72	77	81	82	66					
SCORE	77.3%	84.0%	81.3%	79.0%	87.0%	72.0%	77.0%	81.0%	82.0%	66.0%					
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37	1	0	1	0	1	0	0	0	0	0	30%	2	3	2	3
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40	0	1	1	0	1	1	1	1	1	0	70%	2	0	7	1
41	1	1	1	1	1	0	1	1	1	1	90%	0	0	1	9
42	1	1	1	1	1	0	1	1	1	0	80%	0	0	8	2
43	1	1	1	1	1	1	1	1	1	1	100%	10	0	0	0
44	1	1	1	1	1	1	1	1	1	1	100%	0	0	10	0
45	1	1	1	1	1	1	1	1	1	1	100%	0	0	10	0
46	1	1	1	1	1	1	1	1	1	0	90%	1	0	0	9
47	1	1	0	1	1	1	1	1	1	1	90%	0	1	9	0
48	1	1	1	1	1	1	1	1	1	1	100%	10	0	0	0
49	1	1	1	1	1	1	1	1	1	1	100%	0	10	0	0
50	1	0	0	0	1	0	0	1	0	0	30%	1	6	3	0

2012 NRC WRITTEN EXAM GRADING and ANALYSIS

QUESTION #	ANSWER KEY	ALT. ANSWER	R1	R2	R3	U1	U2	I1	I2	I3	I4	I5
51	a		a	a	a	a	a	a	a	a	a	a
52	c		c	c	c	c	c	c	c	c	c	c
53	a		c	a	a	a	a	a	a	a	a	a
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68	d		a	a	a	a	a	a	b	d	a	a
69	c		c	c	c	c	c	c	c	c	c	c
70	b		c	a	b	b	b	b	b	b	b	a
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72	c		c	c	c	c	c	c	c	c	c	c
73	c		c	c	c	c	c	c	a	c	c	a
74	d		d	d	d	c	d	c	d	c	d	d
75	c		c	c	c	c	b	c	c	c	c	c

	R1	R2	R3	U1	U2	I1	I2	I3	I4	I5	% CORRECT	ITEM ANALYSIS "a"	"b"	"c"	"d"
POINTS	75	75	75	100	100	100	100	100	100	100					
CORRECT	58	63	61	79	87	72	77	81	82	66					
SCORE	77.3%	84.0%	81.3%	79.0%	87.0%	72.0%	77.0%	81.0%	82.0%	66.0%					
51	1	1	1	1	1	1	1	1	1	1	100%	10	0	0	0
52	1	1	1	1	1	1	1	1	1	1	100%	0	0	10	0
53	0	1	1	1	1	1	1	1	1	1	90%	9	0	1	0
54	0	1	1	1	1	0	1	1	1	0	70%	3	0	0	7
55	1	1	1	1	1	0	1	1	1	1	90%	1	0	0	9
56	1	1	1	1	1	1	0	1	1	1	90%	0	1	0	9
57	1	0	0	0	1	1	1	0	1	0	50%	4	1	0	5
58	1	1	1	1	1	1	1	1	1	0	90%	0	9	0	1
59	1	1	1	1	1	1	1	1	1	1	100%	0	10	0	0
60	1	1	1	1	1	1	1	1	1	0	90%	0	1	0	9
61	0	1	0	1	1	1	0	0	0	0	40%	4	5	1	0
62	1	1	1	0	1	0	0	1	0	0	50%	0	0	5	5
63	0	1	1	1	1	0	1	1	1	1	80%	0	2	0	8
64	0	0	1	1	1	0	1	1	1	0	60%	4	0	6	0
65	0	1	0	1	1	0	0	0	1	0	40%	3	0	4	3
66	1	1	1	1	1	1	1	1	1	1	100%	0	10	0	0
67	1	1	0	1	1	1	0	0	0	1	60%	1	0	6	3
68	0	0	0	0	0	0	0	1	0	0	10%	8	1	0	1
69	1	1	1	1	1	1	1	1	1	1	100%	0	0	10	0
70	0	0	1	1	1	1	1	1	1	0	70%	2	7	1	0
71	1	1	1	1	1	1	1	1	1	0	90%	0	0	1	9
72	1	1	1	1	1	1	1	1	1	1	100%	0	0	10	0
73	1	1	1	1	1	1	0	1	1	0	80%	2	0	8	0
74	1	1	1	0	1	0	1	0	1	1	70%	0	0	3	7
75	1	1	1	1	0	1	1	1	1	1	90%	0	1	9	0

2012 NRC WRITTEN EXAM GRADING and ANALYSIS

	R1	R2	R3	U1	U2	I1	I2	I3	I4	I5
POINTS	75	75	75	100	100	100	100	100	100	100
CORRECT	58	63	61	79	87	72	77	81	82	66
SCORE	77.3%	84.0%	81.3%	79.0%	87.0%	72.0%	77.0%	81.0%	82.0%	66.0%

QUESTION #	ANSWER KEY	ALT. ANSWER	R1	R2	R3	U1	U2	I1	I2	I3	I4	I5
76	a					a	a	a	a	a	a	a
77	a					a	a	b	a	a	a	b
78	b					b	b	b	c	b	b	b
79	b					b	b	b	b	b	b	b
80	d					d	d	d	d	d	d	d
81	d					d	d	d	c	d	d	c
82	c					c	c	c	c	c	c	c
83	d					d	d	d	d	d	d	b
84	b					b	b	c	b	a	b	b
85	c					c	b	b	d	c	c	c
86	c					a	a	a	a	c	a	c
87	b					a	b	b	b	b	a	a
88	b					b	b	b	b	b	b	b
89	b					c	a	c	c	a	a	c
90	d					a	a	d	d	a	a	b
91	b					c	a	b	c	b	a	a
92	c					b	c	d	b	c	c	b
93	a					a	a	a	a	a	a	a
94	c					c	c	c	c	a	c	c
95	d					d	d	d	c	c	d	d
96	a					a	a	a	a	a	a	a
97	c					a	a	b	c	c	a	a
98	c					c	c	c	c	c	d	c
99	d					d	d	d	d	d	d	b
100	c					c	c	c	c	c	c	c

QUESTION #	R1	R2	R3	U1	U2	I1	I2	I3	I4	I5	% CORRECT	ITEM ANALYSIS "a"	"b"	"c"	"d"
76				1	1	1	1	1	1	1	100%	7	0	0	0
77				1	1	0	1	1	1	0	71%	5	2	0	0
78				1	1	1	0	1	1	1	86%	0	6	1	0
79				1	1	1	1	1	1	1	100%	0	7	0	0
80				1	1	1	1	1	1	1	100%	0	0	0	7
81				1	1	1	0	1	1	0	71%	0	0	2	5
82				1	1	1	1	1	1	1	100%	0	0	7	0
83				1	1	1	1	1	1	0	86%	0	1	0	6
84				1	1	0	1	0	1	1	71%	1	5	1	0
85				1	0	0	0	1	1	1	57%	0	2	4	1
86				0	0	0	0	1	0	1	29%	5	0	2	0
87				0	1	1	1	1	0	0	57%	3	4	0	0
88				1	1	1	1	1	1	1	100%	0	7	0	0
89				0	0	0	0	0	0	0	0%	3	0	4	0
90				0	0	1	1	0	0	0	29%	4	1	0	2
91				0	0	1	0	1	0	0	29%	3	2	2	0
92				0	1	0	0	1	1	0	43%	0	3	3	1
93				1	1	1	1	1	1	1	100%	7	0	0	0
94				1	1	1	1	0	1	1	86%	1	0	6	0
95				1	1	1	0	0	1	1	71%	0	0	2	5
96				1	1	1	1	1	1	1	100%	7	0	0	0
97				0	0	0	1	1	0	0	29%	4	1	2	0
98				1	1	1	1	1	0	1	86%	0	0	6	1
99				1	1	1	1	1	1	0	86%	0	1	0	6
100				1	1	1	1	1	1	1	100%	0	0	7	0
SRO Only %				72%	76%	72%	68%	80%	72%	60%					

Comments and responses during exam review with applicants.

Comments on the exam overall provided by numerous applicants:

This exam was very different from other exams we have seen. It required us to memorize a lot of minute details from our procedures and training material. FCS training staff acknowledged comments.

Question 5 : numerous applicants

Why would we need to memorize power supplies to these valves (LPSI loop injection valves)? The power supplies are on the control switch labels.

The question stem did not include the noun names for the valves.

FCS training staff replied that we argued that this question was minutia with the NRC but the NRC did not agree. We even wrote a replacement question, but it was not accepted.

Question 18 (55-41861)

The stem does not state which relays. Some energize and others de-energize.

FCS training staff said we would consider arguing for two correct questions. (After more detailed review, FCS training staff concluded that the key answer is the only correct answer. However, the bank question should be reworded to specify which relay.)

Question 23: (55-42109), (55-41777)

The correct answer depends on how the instruments are calibrated.

Are we expected to memorize setpoints and proportional bands for all instruments? Even I&C techs are not required to do that.

Shouldn't we have been provided with a reference?

FCS training staff said we will consider arguing for two correct answers. After further analysis of the operation and the flexibility of the setpoints allowed by procedure it is possible that both valves may be open at same time therefore FCS will argue that two answers are possible.

Comments and responses during exam review with applicants.

Question 24: (55-42112), (55-70783), (55-41861)

It's "LPSI stop and throttle criteria." I ruled out choice "C" because the LPSI pump should have automatically tripped. I picked "A" because the stem didn't state if minimum required injection flow was met or not.

FCS training staff said that all the question was looking for was recognition that the LPSI pumps should have tripped and they didn't. Choice "A" is incorrect because it requires either SI-2A and SI-2B or SI-2B and SI-2C. Applicants replied that early in EOP-03 it states to ensure SI flow is acceptable per attachment 3 and doesn't place limits on which pumps to use. FCS training staff acknowledged comment but further determined answer key correct.

Question 30: numerous applicants

What does recirculation flow have to do with operating the plant. That's more of an engineering concept. I understand shrink and swell. We were taught about recirculation ratio. This was a coin flip question.

Training staff responded that this question was based on a statement from the STM that said recirculation flow increased. (Following a detailed review of S/G design data, training staff concluded that recirculation flow actually decreases with an increase in steam flow.) This should have answer key changed or with inaccurate guidance in STM should be removed from the exam.

Question 33: numerous applicants

Operators don't change setpoints on area radiation monitors. Why was this question even on our exam?

Training staff responded that the NRC felt this met the K/A. FCS will argue that it is a SRO level question or does not match job requirements.

Comments and responses during exam review with applicants.

Question 37: numerous applicants

This one was just a guess.

Training staff responded that this was based on a note in OI-RC-2A which states that LI-119, LI-197, and LI-199 should normally agree to within six inches of one another by the time RCS level reaches 1011'.

Applicants replied, we are not expected to memorize every note in every procedure. We are expected to read the notes when we use the procedures.

Training staff acknowledged the comments and verified exam key correct.

Question 50: (55-41553) and other applicants

I knew that both B and C were correct. I spent a lot of time on this question trying to figure out which correct answer they wanted. I picked B because it addressed feeding and none of the choices addressed isolating the faulted S/G.

FCS training staff said that they agreed that there were two correct answers and they would argue for two correct answers.

Question 57: (55-42112)

This is a tech spec LCO question. It shouldn't be in a RO exam.

FCS training staff replied that although this AOP is based on tech spec required LCOs, RO's are required to be familiar with AOPs. Answer on key verified correct

Question 61: Several Applicants

This was a guess because I'm not that familiar with the DCS and I haven't memorized all the alarm setpoints.

FCS training staff replied that the setpoints are the same as they were before DCS and they should have focused on operation of the system and its requirements and not get hung up on whether it was a hard panel alarm or DCS alarm. Answer on key was correct

Comments and responses during exam review with applicants.

Question 62: Numerous applicants

How can "C" be incorrect and "D" correct? The actions in choice "C" are a subset of the actions in "D". You can't do "D" without doing "C" EOP-20 says to first establish partial once-through-cooling using PCV-102-2 and SI-2B and then take the steps required to establish full once-through-cooling.

FCS training staff said that they agreed that there were two correct answers and they would argue for two correct answers.

Question 67: (55-41861)

D is also correct. There are examples in the NRC's GFE exam bank.

FCS training staff responded that we will check in to that. Further analysis did show reactor pump power did have the same effect on XC-105 and FCS will argue for two correct answers.

Question 68: numerous applicants

The Shift Manager – Operations Standards signs the Operation Memo that states who can operate under instruction in the control room.

FCS Training staff replied that that is a training memo not an operations memo that directs the actions of Operations operating the plant which is much higher order as described in S.O. O-13. FCS verified that the correct answer was on key.

Question 70: Several Applicants

Doesn't maximize mean all?

FCS training staff replied EOP-03 step 8 specifies which combinations of 2 HPSI pumps is used to "maximize." Answer key was correct.

Comments and responses during exam review with applicants.

Question 75: (55-41553)

I picked choice "C" because the BOPO performs a detailed walkdown of the right side of CB-4 to check steam generator pressures, MSIV and MSIV bypass valve status, SGLS status and status of CB-4 annunciators for S/G level and FW control. Why is choice "C" incorrect?

FCS training staff responded that SO-O-1 specifies that for the Secondary LO, panels CB-10/11, 20, AI-66A and 66B and all backpanel areas will be walked down in detail.

. responded that the question stem did not refer to SO-O-1 and that good operator practice requires the BOPO to do a detailed walkdown of the secondary system controls, indications and annunciator status on CB-4.

FCS training staff acknowledged the comment and verified key answer was correct per SO-O-1.

Question 86: numerous SRO applicants

The stem does not say why boron concentration cannot be restored to within limits. Something must be broke so I went with the most conservative LCO. We should have been given the Tech Spec as a handout.

FCS training staff acknowledged comments and said we would consider a formal comment to this question. After thorough review and based on multiple students missing this for the same reason the stem provided poor and vague information resulting in two correct answers. FCS will argue for two answers. There was no lack of knowledge regarding the Technical Specification limit by applicants.

Comments and responses during exam review with applicants.

Question 89: (55-41777) and several other SRO applicants

I ruled out choice B. If there was a fire in room 18, some of the hose cabinets needed for AOP-18 fire protection backup would already be in use for fire fighting. Then we would have to use feed and bleed cooling.

There are so many details in the huge procedure. We should not be required to memorize them all.

FCS training staff acknowledged the comments and will do additional research. FCS completed analysis and determined that with fire main being used for the fire in the same area as the requirement to use it for raw water backup would require a deviation from procedures or choice "C" being correct. FCS will argue for choice "C" or question thrown out.

Question 90: Numerous SRO Applicants

This is a very complex tech spec. We should have been provided with a handout.

FCS Training staff acknowledged the comment. Applicants are required to know the basis of technical specifications

Question 91: Numerous SRO Applicants

A and B don't make any sense because sodium tetraborate will always raise the pH of the sump water. The "not" wording is very confusing. I picked A because if the pH was below 7, adding sodium tetraborate would raise it above 7.

FCS training staff acknowledged the comments and stated that the question wording was changed based on our recommendations. We still felt the wording was unclear and verbally proposed another question but further adjustments to the question was not used.

Comments and responses during exam review with applicants.

Question 92: Numerous SRO applicants

We don't operate FH-1.

FCS training staff stated that SRO are licensed to supervise fuel movement even if they don't operate it and therefore are required to know how it operates and is covered under the K/A catalog.

Question 95: (55-41861)

The stem doesn't say if the fuel assembly is already being inserted.

FCS training staff acknowledged comment but the answer is correct without that information.

Question 97: Numerous SRO applicants

We are required to verify RCS flow every shift per Tech Spec table 3.1 . A is also correct.

FCS Training staff stated that the question stem referred to tech spec 3. 10(7) which specifies monthly total flow measurement.

Applicants responded that the stem was not clear as to which flow verification was referred to.

FCS training staff acknowledged comment. Verified answer key is correct.