Question # _1__

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	__2___	_____
	Group #	__1___	_____
	K/A #	_003 K4.04_______	
	Importance Rating	__2.8___	_____

K/A Statement: (003 Reactor Coolant Pump) Knowledge of RCPS design feature(s) and/or interlock(s) which provide for the following: Adequate cooling of RCP motor and seals.

Proposed Question:

Which of the following reactor coolant pump RC-2B radial bearing temperatures, if exceeded, would require immediate shutdown of this reactor coolant pump in accordance with OI-RC-9, REACTOR COOLANT PUMP OPERATION, Revision 73?

A. 205°F
B. 210°F
C. 225°F
D. 230°F

Proposed Answer: __D___

Explanation: Per OI-RC-9, Precaution 18, a valid temperature indication for the radial bearing of 230°F for RC-3B shall not be exceeded, or the reactor will be tripped and the reactor coolant pump immediately shutdown. Therefore, the maximum temperature indication prior to exceeding 230°F is 230°F. Answer C of 225°F is plausible because the applicant could believe that the pump must be shutdown prior to reaching 230°F. Answer B is plausible because that is the temperature requirement for reactor coolant pumps RC-3A, RC-3C, and RC-3D. Answer A is plausible if the applicant believes that the pump must be shutdown prior to reaching 210°F, which is the temperature for the other three reactor coolant pumps.

Technical Reference(s): _OI-RC-9_______________________________________
(Attach if not previously provided) ____
(including version/revision number) ____

Proposed references to be provided to applicants during examination: __None___________

Learning Objective: __7-11-20 1.7b List the design parameters for the RCP (As available)

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

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	__X__
	Comprehension or Analysis	_____
10 CFR Part 55 Content:	55.41 __(3)__	
	55.43 _____	

Comments:

Question # _2___

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_2___	_____
	Group #	_1___	_____
	K/A #	_004 A2.15_______	
	Importance Rating	_3.5__	_____

K/A Statement: (004 Chemical and Volume Control) Ability to (a) predict the impacts of high or low PZR level on the CVCS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of high or low PZR level.

Proposed Question:

Given the following:

Reactor power is 100%
T_{ave} is on program
Pressurizer level is 56.9%
Pressurizer level controller is in cascade

The Control Room is using Attachment 2 for "AUTOMATIC PRESSURIZER LEVEL CONTROL" from procedure OI-RC-8, REACTOR COOLANT SYSTEM LEVEL CONTROL NORMAL OPERATION. Assuming no operator action has been taken up to this point, what should operators do to verify correct operation of the pressurizer level controller?

A. Verify that two charging pumps are running and letdown flow rate is 26 gpm
B. Verify that two charging pumps are running and letdown flow rate is 36 gpm
C. Verify that three charging pumps are running and letdown flow rate is 26 gpm
D. verify that three charging pumps are running and letdown flow rate is 36 gpm

Proposed Answer: ___A___

Explanation: OI-RC-8, R15, page 7 on Attachment 2 for automatic operation, has the operator verify pressurizer level is maintaining 48% to 60% within a deviation from setpoint of plus or minus 4% as indicated by LR-101X. Pressurizer Level is in control because it is within the 4% however the operator would be expected to verify that the correct number of charging pumps and letdown flow are responding correctly for the controller and its mode of operation.

Based on the "Level Control Christmas Tree" diagram, the second charging pump starts when level is 2.9% below setpoint, and letdown flow is minimized when level is 2.0% below setpoint. The third charging pump starts at 3.3% below setpoint. Letdown flow is 36 gpm when the pressurizer is at setpoint. At 100% power, with T_{ave} on program, the pressurizer level setpoint is 60%. 56.9% is 3.1% below setpoint. Therefore, answer A is correct. Answer B is plausible because 36 gpm is also located on the diagram as letdown flow at pressurizer setpoint. Letdown flow decreases to the minimum value of 26 gpm at 2% below setpoint. Answer C is plausible because three pumps run, but not until level is 3.3% below setpoint. Answer D is plausible based on the same explanations for answers A and C.

Technical Reference(s): System Training Manual Vol. 12 Chemical and Volume Control System, pgs. 14, 15, 69, 70__and________
(Attach if not previously provided) ______OI-RC-8, R15, page 7.________
(including version/revision number) ____

Proposed references to be provided to applicants during examination: _None____________

Learning Objective: _7.11.02 4.1 Explain how RCS volume is controlled automatically and manually.

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:

Question # __3__

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	__2__	_____
	Group #	__1__	_____
	K/A #	_039 A3.02_____	
	Importance Rating	_3.1__	_____

K/A Statement: (039 Main and reheat Steam) Ability to monitor automatic operation of the MRSS, including: Isolation of the MRSS

Proposed Question:

Given the following plant conditions:

- The RCS is being cooled down
- RCS Pressure is 1650 psia
- Pressure in both Steam Generators is 490 psia
- PPLS has been blocked
- SGLS has been blocked

Which one of the following will cause automatic closure of the MSIVs, HCV-1041A and HCV-1042A?

A. Containment pressure rises to 5.5 psig
B. RCS Pressure rises to 1705 psia
C. Level in RC-2A lowers to 30% WR
D. Pressure in RC-2A rises to 555 psia

Proposed Answer: __A___

Explanation:
For Answer A, SGIS will actuate due to CPHS at 5 psig containment pressure and MSIVs will close, therefore A is correct.
For distractor B, PPLS will unblock above 1700 psia but that will not affect the MSIVs. PPLS does normally close the MSIV's, too.
For distractor C, AFAS will initiate below 32% wide range but that will not affect the MSIVs.
For distractor D, SGLS will automatically unblock above 550 psia, but SGIS will not occur unless S/G pressure then falls to 500 psia.

Technical Reference(s): _System Training Manual Vol. 25 Main Steam and Steam Generator System REV, PAGE, Also ESFAS STM-19, rev 17.

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

Proposed references to be provided to applicants during examination: __None__________

Learning Objective: __7-11-17 1.0 Apply operating principles to diagnose Main Steam System response to specific plant conditions (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:

Question # _4__

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_2___	_____
	Group #	_1___	_____
	K/A #	_008 K3.01______	
	Importance Rating	_3.4_	_____

K/A Statement: (008 Component Cooling Water) Knowledge of the effect that a loss or malfunction of the CCWS will have on the following: Loads cooled by CCWS.

Proposed Question:

The set point for automatic component cooling water isolation, coincident with a CIAS, to the reactor coolant pumps and the control element drive mechanisms occurs when component cooling water pressure drops below __1___ for more than __2___.

A. 34 psig, 15 seconds
B. 34 psig, 30 seconds
C. 46 psig, 15 seconds
D. 46 psig, 30 seconds

Proposed Answer: _D____

Explanation: Answer D is correct per pg. 31 of STM Vol. 8, Component Cooling Water System. CCW is automatically isolated if there's a CIAS AND pressure drops below 46 psig for more than 30 seconds. The applicant could incorrectly determine that the water pressure value is the same as the pressure at which CCW must be declared inoperable (34 psig), per OI-CC-1. The applicant could choose 15 seconds if they don't remember the value of the time delay. Fifteen seconds is also the required action range of one of the isolation valves (HCV-438D).

Technical Reference(s): _STM Vol. 8 Component Cooling Water System, Procedure OP-ST-CCW-3004 , "Component Cooling Category A and B Valve Exercise Test", Rev. 22, and Procedure OI-CC-1, "Component Cooling System Normal Operation", Rev. 67__

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

Proposed references to be provided to applicants during examination: __________________

Learning Objective: __________________________ (As available)

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

New __X____

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 __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __(4)___
55.43 _____

Comments:

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, USAR Section 6.2, Paragraph 6.2.3.6, USAR Figure 8.1-2, STM Vol. 14 Electrical Distribution System_

(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:

Question # _6__

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_2___	_____
	Group #	_1___	_____
	K/A #	_010 K1.01_	
	Importance Rating	_3.9__	_____

K/A Statement: (010 Pressurizer Pressure Control) Knowledge of the physical connections and/or cause-effect relationships between the PZR PCS and the following systems: RPS

Proposed Question:

- The plant is at 100% power. On the previous shift both the High Pressure and TM/LP (Thermal Margin/Low Pressure) trip units on RPS (Reactor Protection System) Channel "C" had to be placed in the "BYPASS" condition due to an instrument failure

- During troubleshooting, an I&C Technician incorrectly placed the RPS Channel "A" TM/LP trip unit in the "TRIPPED" condition.

What is the resulting trip logic for these conditions?

A. The RPS will be in a 2 out-of 3 trip logic on High Pressure, and a 1 out-of-2 trip logic on TM/LP

B. The RPS will be in a 1 out-of 2 trip logic on High Pressure, and a 1 out-of-2 trip logic on TM/LP

C. The RPS will be in a 2 out-of 3 trip logic on High Pressure, and a 2 out-of-3 trip logic on TM/LP

D. The RPS will be in a 1 out-of 2 trip logic on High Pressure, and a 2 out-of-3 trip logic on TM/LP

Proposed Answer: __A___

Explanation:
Thermal Margin Low Pressure is part of the Pressurizer Pressure Control System. RPS logic with no failed or bypassed signals is 2 out-of-4 logic, while TM/LP logic is 1 out-of4 with no failed or bypassed signals.

Per STM 38, Revision 20, with one channel lost due to bypass:

1. RPS becomes 2 out-of-3 logic to trip and TM/LP becomes 1out of 3 logic to trip.
2. With the "A" channel in a tripped state it changes the configuration for TM/LP only (not RPS) to 1 out-of-2 logic.

For this reason answer A is correct. The other distracters are incorrect but credible because they are logic combinations that are possible depending on the failures and how they fail, but not in these circumstances.

Technical Reference(s): ________STM 38 Rev 20 page 41__________________

Proposed references to be provided to applicants during examination: _None______

Learning Objective: __07-12-25 02.02 ________

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

Question History: Last NRC Exam _None_________

Pulled from FCS Bank in 2011 on page 1492 with the following information:
Ent. 2/26/90 Reference: 7-12-25,2.2
New-Bannister Rev. 8/16/91
KA#: 012000 Bank Reference #:
LP# / Objective: 07-12-25 02.02

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

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

Comments:
It appears as if the question were written in 1991 with Learning objectives identified above. These need to be verified as current and correct by licensee.
I modified the distracters to be an even 2 X 2 matrix and increased the credibility of the distracters.

Question # _7__

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_2___	_____
	Group #	_1___	_____
	K/A #	_059 A1.03___	
	Importance Rating	_2.7*_	_____

K/A Statement: (059 Main Feedwater) 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.

Proposed Question:

Per OP-4, LOAD CHANGE AND NORMAL POWER OPERATION, Rev. 44, Attachment 2, "Power Reduction", during a plant shutdown:

1. Main feedwater control is transferred to the main feedwater bypass valve control when reactor power is reduced to less than ___1___ %.
2. The operating main feedwater pump recirculation valve(s) must be failed open to ensure adequate pump cooling when reactor power is reduced to less than ___2___ %.

A. 30; 10
B. 10; 50
C. 30; 50
D. 10; 10

Proposed Answer: __A___

Explanation: Per OP-4, Attachment 2, pg. 26, Note: "Auto transfer or operator intervention may be used to switch between the Bypass and Main Feed Reg Valves below 30% power but above 20% power. Step 9 states: "When less than 30% power, then ensure main feedwater control is transferred to main feedwater bypass valve control per OI-FW-3". The Caution on the same page states: "The operating feedwater pump recirc valves must be failed open to ensure adequate pump cooling when suction flow is less than 3000 gpm (10-25% reactor power)." Therefore, answer A is correct. B is incorrect but plausible because the system technical manual states that main feedwater reg valves function to maintain programmed steam generator level between 15 and 100 % reactor power. Also, the lesson plan states that automatic transfer to the bypass valves occurs at approximately 16% during a power reduction. Therefore, 10% is plausible, but incorrect. Failing open the feedwater pump recirc valves at 50% power is not in accordance with the caution, but is plausible because that is when one main feedwater pump is taken out of service, so the applicant could have the misconception that the valves need to be failed open at the time that the pump becomes the only operating pump.

Question for FCS – is digital feedwater control system in full auto during a normal down-power and if so does the auto transfer occur as stated in lesson plans.

Technical Reference(s): _OP-4, “Load Changes and Normal Power Operation” Rev. 44, Lesson Plan 07-11-11 Rev. 14, STM Vol. 20 Feedwater and Condensate System Rev. 46

(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 __(5)__

55.43 _____

Comments:

Question # _8__

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	__2__	_____
Group #	__1__	_____
K/A #	_064 K1.03___	
Importance Rating	_3.6__	_____

K/A Statement: (064 Emergency Diesel Generator) Knowledge of the physical connections and/or cause-effect relationships between the ED/G system and the following system: Diesel fuel oil supply system

Proposed Question:

Pick the **ONE** response which describes the flow path of fuel oil being supplied to **ONE** emergency diesel generator from onsite bulk storage. Notice the use of "common" to refer to components that are intended to serve both diesel generators and "dedicated" to refer to components that serve only the associated diesel generator.

A. From the dedicated underground fuel oil storage tank through common fuel oil transfer pumps, the dedicated auxiliary day tank, the dedicated engine base tank, the dedicated fuel oil pumps, and to the engine
B. From the dedicated underground fuel oil storage tank through dedicated fuel oil transfer pumps, the dedicated auxiliary day tank, the dedicated engine base tank, the dedicated fuel oil pumps, and to the engine
C. From the common underground fuel oil storage tank through dedicated fuel oil transfer pumps, the dedicated auxiliary day tank, the dedicated engine base tank, the dedicated fuel oil pumps, and to the engine
D. From the common underground fuel oil storage tank through common fuel oil transfer pumps, the dedicated auxiliary day tank, the dedicated engine base tank, the dedicated fuel oil pumps, and to the engine

Proposed Answer: __C___

Explanation: Per the system training manual, C is the correct choice. The other choices are plausible if the candidate cannot remember which tanks/pumps are dedicated and which are not.

Technical Reference(s): _STM Vol. 16 Emergency Diesel Generator System____
(Attach if not previously provided) ______________________________
(including version/revision number) ______________________________

Proposed references to be provided to applicants during examination: __None_____

Learning Objective: __07-13-05 01.04________ (As available)

Question Source: Bank # ___07-13-05 053 ___
Modified Bank # _______ (Note changes or attach parent)

	New	_______
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 _(8)___	
	55.43 _____	

Comments:

Question # _9__

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_2___	_____
	Group #	_1___	_____
	K/A #	_005 K3.01_____	
	Importance Rating	_3.9__	_____

K/A Statement: (005 Residual Heat Removal) Knowledge of the effect that a loss or malfunction of the RHRS will have on the following: RCS

Proposed Question:

Given the following:

Shutdown cooling is in service
Pressurizer manway has been removed
Shutdown cooling heat exchanger flow control valve, HCV-341, has lost control power

If no operator action is taken, the bulk reactor coolant system temperature is expected to _____1_____ because HCV-341 has failed _____2_____.

A. Stay the same; as is
B. Increase; closed
C. Decrease; open
D. Increase; to minimum flow position

Proposed Answer: __B__

Explanation: HCV-341 determines the amount of flow that goes through the heat exchangers. If control power to the valve is removed, it fails closed, thus isolating the heat exchangers and causing RCS temperature to increase. Thus B is correct. A is plausible if the applicant believes that this valve fails as is on loss of control power. C is plausible because that would happen if the valve failed open. D is plausible if the applicant believes that the valve fails to some minimum flow position, which it does not (it fails closed).

Technical Reference(s): _STM Vol. 15 Emergency Core Cooling System Rev. 52, Lesson plan 07-11-22 Rev. 33___

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

Proposed references to be provided to applicants during examination: _________________

Learning Objective: _________________________ (As available)

Question Source: Bank # _______

	Modified Bank #	_______ (Note changes or attach parent)
	New	___X___
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	__(7)__
	55.43	_____

Comments:

Question # _10_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_2___	_____
	Group #	_1___	_____
	K/A #	_004 2.1.20_	
	Importance Rating	_4.6__	_____

K/A Statement: (004 Chemical and Volume Control) Ability to interpret and execute procedure steps for chemical volume and control system.

Proposed Question:

At 100% power, when placing a second charging pump in service, per OI-CH-1, CHEMICAL AND VOLUME CONTROL SYSTEM NORMAL OPERATION, how are charging and letdown flows matched to minimize the pressurizer level transient?

A) The letdown control valves, LCV-101-1 or LCV-101-2, are controlled manually when more than one charging pump is in operation to match flows.
B) The level bias potentiometer on letdown flow controller, HIC-101-1/101-2, is manually adjusted to match flows.
C) The pressure setpoint on PIC-210 is manually adjusted until charging and letdown flows are matched.
D) Charging and letdown are matched automatically with no change in pressurizer level.

Proposed Answer: __B___

Explanation: Per OI-CH-1 attachment 3, the bias on HIC-101-1 and 1-102 is adjusted to stabilize pressurizer level. Answer A is plausible because the valves can be controlled in manual if there's a problem with HIC-101-1, but a failure of this is not given in the question statement. Answer C is plausible because the pressure setpoint is adjusted, but it's to maintain letdown pressure around 300 psig. Answer D is plausible in that if no operator action is taken, the pressurizer level control system would automatically make changes to charging and letdown given the change in pressurizer level, but pressurizer level changes would occur.

Technical Reference(s): _STM 12 Chemical and Volume Control, Rev. 44, OI-CH-1, "Chemical and Volume Control System Normal Operation", Rev. 88_____

(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 # __071102 010____

	Modified Bank #	_______ (Note changes or attach parent)
	New	_______
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 __(10)___	
	55.43 _____	

Comments:

Question # _11_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_2___	_____
	Group #	_1___	_____
	K/A #	_062 K2.01_	
	Importance Rating	_3.3__	_____

K/A Statement: (062 AC Electrical Distribution) Knowledge of bus power supplies to the following: Major system loads

Proposed Question:

Given the following conditions:

The plant is at 100% power
4160V bus 1A4 develops a ground fault and all feeder breakers that could supply it are locked out, causing a reactor trip.

Which of the following loads is lost as a result of deenergizing 1A4?

A) CCW pump AC-10C
B) CCW pump AC-10B
C) Feed water pump FW-4B
D) Circ Water pump CW-1B

Proposed Answer: ___B__

Explanation: Per STM 20, and Fig. 8.1-1, CCW pump AC-10B is powered from this bus, and the others are not, making choice B correct.

Technical Reference(s): __STM Vol. 20, Feedwater and Condensate System, Rev. 46, Fig. 8.1-1 "Simplified One Line Diagram Plant Electrical System P&ID", Rev. 141_______

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

Proposed references to be provided to applicants during examination: _______None___

Learning Objective: __07-13-02 01.00________ (As available)

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

Question History: Last NRC Exam __NA____

(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:

Question # _12__

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_2___	_____
	Group #	_1___	_____
	K/A #	_013 A1.07__	
	Importance Rating	_3.6_	_____

K/A Statement: (013 Eng. Safety features Actuation) Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ESFAS controls including: Containment radiation

Proposed Question:

Given the following:

Plant is in Mode 3

The following have just automatically actuated:

Waste gas tank discharge secures

Control room ventilation shifts to filtered makeup

Containment cooling and filtering units shift to charcoal filtered mode

Containment ventilation paths isolate

Assuming no other automatic actuations or operator actions have occurred, which of the following caused the above plant response?

A. Containment particulate monitor, RM-050, is in alarm (reads above 5E6 cpm)
B. Containment noble gas monitor, RM-051, is in alarm (reads above 5E6 cpm)
C. Containment pressure switches A/PC-742-1 and C/PC-742-1 read above 5 psig
D. Containment pressure switches A/PC-742-1 and C/PC-742-2 read above 5 psig

Proposed Answer: __B__

Explanation: B is correct because the automatic actuations are from a VIAS, and since no other actuations have occurred, there is no SIAS or CIAS. A high containment radiation signal is a 1 out of 3 logic, so if RM-051 reads above the alarm setpoint (5E6 cpm per TDB-IV.7), then a VIAS is actuated. A is incorrect but plausible because RM-050 also detects radiation levels in containment but only has an alarm function and does not have any automatic actuations. C is incorrect but plausible because it generates a CIAS and SIAS. The CIAS would then initiate the VIAS, but since no other actuations have occurred, it is incorrect. D is incorrect but plausible in that a high containment pressure signal is initiated in 2 out of 4 logic, and 2 switches read high, but they are in separate logic trains (A/PC-742-1 is in CPHS-A while C/PC-742-2 is in CPHS-B) and thus no actuation would occur.

Technical Reference(s): _STM Vol. 19 Engineered Safeguards Control System, Technical Data Book TDB-IV.7, STM Vol. 33 Radiation Monitoring System

(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 _(11)___
55.43 _____

Comments:

Question # _13_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_2___	_____
	Group #	_1___	_____
	K/A #	__007 A1.02___	
	Importance Rating	_2.7__	_____

K/A Statement: (007 Pressurizer Relief Tank/Quench Tank System) Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating PRTS controls including: maintaining quench tank pressure

Proposed Question:

Given the following:

The gas volume in the pressurizer quench tank (PQT) is sufficient to limit the maximum tank pressure after ___________.

A. A zero to 112 percent reactor power swing with both PORV's ONLY discharging to the PQT
B. A zero to 112 percent reactor power swing with both PORVs and both code safety valves discharging to the PQT
C. A one hundred percent load rejection with both PORVs ONLY discharging to the PQT
D. A one hundred percent load rejection with both PORVs and both code safety valves discharging to the PQT

Proposed Answer: __B___

Explanation: Per the STM-37, page 111, the correct answer is "B," the quench tank purpose is to quench the steam discharged by all four pressurizer relief valves (ie two PORV and two code safeties) during a 0-112% power swing without letdown or spray with the maximum pressure expected to be around 50psig. The distracters are credible but not correct because FCS did have a 100% load rejection in 1976 and the two PORV's prevented the two code safeties from lifting but the rupture disc on the PQT was blown out (> 75psig) for this event.

Technical Reference(s): STM Vol. 37 Reactor Coolant System, Rev 42, page 111, page 103, and USAR-4.3, Rev 35, page 35.
(Attach if not previously provided) ____________________
(including version/revision number) ____________________

Proposed references to be provided to applicants during examination: None_________

Learning Objective: _0711-20_________________________ (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 _(5)__	
	55.43 _____	

Comments:

Question # _14__

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_2___	_____
	Group #	_1___	_____
	K/A #	_063 K1.02_	
	Importance Rating	_2.7__	_____

K/A Statement: (063 DC Elect Distribution) Knowledge of the physical connections and/or cause-effect relationships between the DC electrical system and the following systems: AC electrical system

Proposed Question:

Plant electrical system is in its normal alignment for power operation except for Instrument Bus A, which is currently powered by its bypass power source.

A loss of 480V Bus 1B3B has just occurred.

Assuming no operator actions have occurred, what is the current status of Instrument Bus A?

A. Deenergized
B. Energized from battery
C. Energized from bypass power source
D. Energized from crosstie with Instrument Bus C

Proposed Answer: __A___

Explanation: With Inverter A bypassed, it cannot automatically transfer back to its instrument bus per OI-EE-4. Per the plant electrical system diagram, the bypass power source for instrument bus A is MCC-3B1 off of 480V bus 1B3B. Bus 1B3B also powers the battery charger for Battery #1 and instrument buses A, C, and 1 via their respective inverters. Therefore, if Bus 1B3B is lost, the bypass power to bus A is also lost. Since it can't automatically transfer to the battery, and the normal electrical alignment has the crossties with Bus C open, Instrument Bus A will be deenergized. B is plausible if the applicant believes that the bus will automatically transfer. C is plausible if the applicant does not remember that the bypass source for bus A is on the 480V bus 1B3B. D is plausible if the applicant does not remember that the cross ties with instrument bus C are normally open.

Technical Reference(s): _STM Vol. 14 Electrical Distribution System, OI-EE-4, "Operating Instruction – 120 Volt AC System Normal Operation" Rev. 48, Dwg. FIG 8.1-1 "Simplified One Line Diagram Plant Electrical System P&ID" Rev. 141___

(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 __(8)__
55.43 _____

Comments:

Question # _15_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_2___	_____
	Group #	_1___	_____
	K/A #	_078 K1.03____	
	Importance Rating	_3.3*_	_____

K/A Statement: (078 Instrument Air) Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems: Containment Air

Proposed Question:

Which if the following will automatically close instrument air isolation valves PCV-1849A and PCV-1849B?

A. Only PCV-1849A closes on instrument air pressure <70 psig
B. Only PCV-1849B closes on CIAS
C. Both PCV-1849A and B valves close if CIAS present AND instrument air pressure <70 psig
D. Both PCV-1849A and B valves close if CIAS present OR instrument air pressure <70 psig

Proposed Answer: _C___

Explanation: Per STM Vol. 43, CIAS automatically closes instrument air isolation valves PCV-1849A/B if air pressure upstream of the valves is less than 70 psig. Therefore answer C is correct. Answer A is plausible but incorrect if the candidate believes that only one valve closes if instrument air is less than 70 psig. Answer B is plausible but incorrect if the candidate believes that only one valve closes on CIAS. Answer D is plausible but incorrect if candidate does not remember that both a CIAS AND instrument air <70 psig must be present.

Technical Reference(s): _STM Vol. 43 “Service and Instrument Air System”, AOP-17 “Loss of Instrument Air”____________________

(Attach if not previously provided) ______________________________

(including version/revision number) ______________________________

Proposed references to be provided to applicants during examination: _________________

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 _(9)__
55.43 _____

Comments:

Question # _16_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_2___	_____
	Group #	_1___	_____
	K/A #	_073 K4.01_____	
	Importance Rating	_4.0_	_____

K/A Statement: (073 Process Radiation Monitors) Knowledge of PRM system design feature(s) and/or interlock(s) which provide for the following: Release termination when radiation exceeds setpoint.

Proposed Question:

A liquid radioactive release is in progress when Effluent Radiation Monitor, RM-055, alarms. As a result of the alarm, which of the following will occur?

A. HCV-691 and HCV-692 (Overboard Discharge Control Valves) will close and the Monitor Tank Pumps will trip.
B. HCV-691 and HCV-692 (Overboard Discharge Control Valves) and HCV-673 and HCV-679 (Monitor Tank Inlet Valves) will close and the Monitor Tank Pumps will trip.
C. HCV-691 and HCV-692 (Overboard Discharge Control Valves) will close and the Monitor Tank Pumps will continue to operate in recirculation mode.
D. HCV-691 and HCV-692 (Overboard Discharge Control Valves) and HCV-673 and HCV-679 (Monitor Tank Inlet Valves) will close and the Monitor Tank Pumps will continue to operate in recirculation mode.

Proposed Answer: _A___

Explanation: A is correct based on STM Vol. 33 Radiation monitoring system and Vol. 48, Radioactive waste disposal system. B is incorrect but plausible if the applicant believes all four valves close. C is incorrect but plausible if the applicant does not know that the pumps are tripped. D is incorrect but plausible for the same as the previous two.

Technical Reference(s): __STM Vol. 33 Radiation Monitoring System, Vol. 48, Radioactive Waste Disposal System, __________

(Attach if not previously provided) ______________________________

(including version/revision number) ______________________________

Proposed references to be provided to applicants during examination: _________________

Learning Objective: _07-11-32 01.02A____________ (As available)

Question Source: Bank # _______

	Modified Bank #	071132 025 (Note changes or attach parent)
	New	_______

Question History: Last NRC Exam __1997____

(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	_(13)__
	55.43	_____

Comments:

Parent Question from 1997 RO exam

07-11-32 001
An alarm on Effluent Radiation Monitor, RM-055, will cause which of the following to occur?

A. HCV-691 and HCV-692 (Overboard Discharge Control Valves) will close and the Monitor Tank Pumps will trip.
B. HCV-672 and HCV-678 (Monitor Tank Inlet Valves) will close and the Monitor Tank Pumps will trip.
C. HCV-691 and HCV-692 (Overboard Discharge Control Valves) will close and the Monitor Tank Pumps will continue to operate.
D. HCV-672 and HCV-678 (Monitor Tank Inlet Valves) will close and the Monitor Tank Pumps will continue to operate.

Answer: B

KA#: 068000 A2.04 Bank Reference #:
LP# / Objective: 0711-32 01.02A Exam Level:
Cognitive Level: LOW Source: NRC 97 EXAM
Reference: LP 0711-32 Handout: NONE

Question # _17_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_2___	_____
	Group #	_1___	_____
	K/A #	_061 K6.01_	
	Importance Rating	_2.5__	_____

K/A Statement: Knowledge of the effect of a loss or malfunction of controllers and positioners will have on AFW components

Proposed Question:

Following a reactor trip with an auxiliary feed actuation signal (AFAS), turbine driven auxiliary feedwater pump FW-10 overspeed limit overpressure protection circuit, PS-1122, momentarily senses maximum indicator pressure before returning to normal. An operator then attempts to manually restart the pump from panel AI-66B in the Control Room. Assuming no other operator action is taken, what is expected response of FW-10 and the reason for that response?

A. FW-10 restarts because stop valve YCV-1045 reopens following operator attempt
B. FW-10 does not restart because stop valve YCV-1045 remains closed following operator attempt
C. FW-10 restarts because steam supply valves YCV-1045A & B reopen following operator attempt
D. FW-10 does not restart because steam supply valves YCV-1045A & B remain closed following operator attempt

Proposed Answer: __B___

Explanation: Per STM Vol. 4, An overpressure condition at the pump discharge will actuate the trip relay to reposition an air solenoid valve on the inlet tubing to the YCV-1045 actuator, thus closing YCV-1045. The overpressure trip will override all starts of FW-10 (auto, manual, AFAS). Overpressure 1600 +/- 35 psig sensed by PS-1122 actuates overpressure trip. After an overpressure trip, restart of FW-10 will not be available until the trip is manually reset (which it is not in this question). Therefore, YCV-1045 closes and remains closed. A is incorrect but plausible if the applicant does not know that the trip must be manually reset. C is incorrect but plausible if the applicant does not know that the trip must be manually reset and that it's the stop valve, and not the steam supply valves that close on an overpressure condition. D is incorrect but plausible in that the pump does not restart but the steam supply valves don't close on overpressure, the stop valve (YCV-1045) does.

YCV-1045 is the stop valve
YCV-1045A and B are steam supply valves

Technical Reference(s): _STM Vol. 4 "Auxiliary Feedwater System" Lesson Plan 7-11-1 Auxiliary Feedwater System_______________

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

Proposed references to be provided to applicants during examination: __None_______

Learning Objective: __7-11-1 1.7 Explain the automatic operations of AFW System components_ (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:

Question # _18_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	__2__	_____
	Group #	__1__	_____
	K/A #	__012 A3.03___	
	Importance Rating	_3.4__	_____

K/A Statement: (012 Reactor Protection) Ability to monitor automatic operation of the RPS, including: power supply

Proposed Question:

DC power supply PS8 fails, de-energizing matrix relays AC3 and AC4. This will trip "M" coils _____1_____ and de-energize clutch power supplies _____2_____.

A. M3 and M4; PS2 and PS4
B. M2 and M4; PS2 and PS4
C. M1 and M3; PS1 and PS3
D. M1 and M2; PS1 and PS3

Proposed Answer: __A___

Explanation: Per STM Vol. 38, Figure 2-35, power supply PS8 provides the power for the "C" side of the "AC" logic ladder (similar to Figure 2-26 which is for the "AB" logic ladder). Therefore, failure of PS8 will cause AC3 and AC4 matrix relays to de-energize, Per Figure 2-29, this will result in a trip of "M" coils M3 and M4, which in turn de-energize PS2 and PS4, making "A" the correct choice. Answer "D" is plausible in that this is what would occur if power supply PS7 were to fail and de-energize matrix relays AC1 and AC2. Answer "B" is plausible if the applicant had the conceptual error that the "M" coil designations match the clutch power supply designations (i.e., M2 energizes PS2). PS2 and PS4 would be deenergized, but not M2. Answer "C" is plausible for the same reason as answer "B", except that the applicant would also have to incorrectly assume that matrix relays AC1 and AC2 had failed (which would occur if PS7 was lost).

Technical Reference(s): _STM Vol. 38, "Reactor Protective System and Diverse Scram System", Lesson Plan 07-12-25_____

(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	__(7)__
	55.43	_____

Comments:

Question # _19_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_2___	_____
	Group #	_1___	_____
	K/A #	_026 A2.08___	
	Importance Rating	_3.2__	_____

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

Proposed Question:

A main steam line break on steam generator RC-2B has occurred inside containment. Standard post-trip actions have been completed and the transition has been made to EOP-05 UNCONTROLLED HEAT EXTRACTION. Actions have been completed to Steps 24 and 25, which determine if containment spray can be reduced or terminated.

Current containment status:

Containment spray pumps SI-3A and SI-3B are running
Containment pressure is 5 psig and slowly lowering
Containment air cooler filters systems VA-3A and VA-3B are NOT in service
Containment air coolers VA-7C and VA-7D are in service

Based on the above status, EOP-05 Steps 24 and 25 require which of the following?

A. Termination criteria is met: stop both containment spray pumps
B. Reduction criteria is met: stop one containment spray pump and ensure containment spray flow is at least 2300 gpm
C. Reduction criteria is met: stop one containment spray pump and ensure containment spray flow is at least 2800 gpm
D. Neither reduction nor termination criteria met: continue operation of both containment spray pumps

Proposed Answer: _D___

Explanation: Per EOP-05, containment pressure must be less than 3 psig and stable or lowering with containment spray not required for containment cooling in order to terminate containment spray (stop both pumps). If two pumps are operating (they are in this question), containment pressure is less than 60 psig (it is), at least one containment air cooler is in service (both are), and at least one containment air cooler filter system is in service (it isn't), then one pump would be secured and one would ensure that flow rate is at least 2300 gpm. 2800 gpm is the runout flow rate for one containment spray pump. Based on this, answer "D" is correct because the criteria for termination of containment spray is not met (5 psig is greater than 3 psig), and the criteria to reduce containment spray is not met (no containment air cooler filter systems in service). Answer A is plausible if the applicant believes that 5 psig is the termination point instead of 3 psig. Answer B is plausible in that it would be correct if one containment air cooler filter system was in service. Answer C is plausible if the applicant believed that 2800 gpm was the required flowrate (this is the pump maximum flowrate).

Technical Reference(s): ___EOP-05 “Uncontrolled Heat Extraction” Rev. 24, STM Vol. 15 ECCS Rev. 52_______

(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 __(5)__
55.43 _____

Comments:

Question # _20_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_2___	_____
	Group #	_1___	_____
	K/A #	_003 K5.01_____	
	Importance Rating	_3.3_	_____

K/A Statement: (003 Reactor Coolant Pump) Knowledge of the operational implications of the following concepts as they apply to the RCPS: The relationship between the RCPS flow rate and the nuclear reactor core operating parameters (quadrant power tilt, imbalance, DNB rate, local power density, difference in loop T-hot pressure).

Proposed Question:

Three weeks before the next scheduled refueling outage, the rotor for reactor coolant pump 2C seizes. Prior to the CEA insertion due to the reactor trip on low reactor coolant flow, initial expected plant response is:
Reactor power will _____1_____ and DNBR will _____2_____ .

A. Increase; increase
B. Increase; decrease
C. Decrease; increase
D. Decrease; decrease

Proposed Answer: _D___

Explanation: Per USAR section 14.6 and Lesson plan 07-15-16, for a seized rotor event, coolant flow will quickly decrease to the reactor trip setpoint. However, due to RPS processing and CEA holding coil release delays, the reactor is still at power, for about 1.3 seconds until CEAs begin to insert. During this time, core flow decreases and core temperature increase. For a positive temperature coefficient (which occurs at the beginning of core life), this would result in an increase in reactor power (described in the USAR for worst case scenario). However, in this case, since it's at the end of core life, it's a negative temperature coefficient and would result in a slight decrease in reactor power prior to the CEA's insertion. Answer "B" is plausible because this is the accident discussed in the USAR for a positive temperature coefficient. Answers A and C are plausible if the applicant has the misconception that an increase in DNBR leads to approaching the DNBR limit.

Technical Reference(s): _USAR 14.6, Lesson Plan 07-15-16__________
(Attach if not previously provided) ______________________________
(including version/revision number) ______________________________

Proposed references to be provided to applicants during examination: __________________

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	_X___
	55.43	_____

Comments:

Question # _21_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	__2__	_____
	Group #	__1__	_____
	K/A #	_076 2.4.4_____	
	Importance Rating	__4.5_	_____

K/A Statement: (076 Service Water) Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Proposed Question:

Which of the following would result in entering AOP-18, LOSS OF RAW WATER?

A. One raw water header flow indicates less than 3500 gpm
B. Both raw water header flows indicate less than 3500 gpm
C. One raw water header flow indicates less than 1000 gpm
D. Both raw water header flows indicate less than 1000 gpm

Proposed Answer: _D___

Explanation: Entry conditions into AOP-18 include BOTH raw water header flows being low. Per the STM, annunciators come in when raw water flow indicates less than 1000 gpm. Therefore, answer D is correct. Answer C is plausible but incorrect if the candidate believes only one header flow must be low before entering AOP-18. Answers A and B are plausible but incorrect because 3500 gpm is the low flow alarm for CCW, not raw water.

Technical Reference(s): ______STM Vol. 35 Raw Water System, Rev. 26, AOP-18, "Loss of Raw Water" Rev. 17

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

Proposed references to be provided to applicants during examination: __________________

Learning Objective: __________________________ (As available)

Question Source: Bank # ________
Modified Bank # __071119 013_ (Note changes or attach parent)
New ______

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 __(10)__
55.43 _____

Comments: Changed stem so that it matched K/A. Slightly changed answers.

Question # _22_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	__2__	_____
	Group #	__1__	_____
	K/A #	__103 K3.02____	
	Importance Rating	__3.8_	_____

K/A Statement: (103 Containment) Knowledge of the effect that a loss or malfunction of the containment system will have on the following: Loss of containment integrity under normal operations

Proposed Question:

During surveillance testing HCV-1560A, Dearated Water Supply to Containment Isolation Valve was declared inoperable for faulty control room indications. AOP-12, LOSS OF CONTAINMENT INTEGRITY has been entered.

Using the attached Figure Q22-1, what additional actions are required prior to exiting AOP-12?

A. Isolate air from HCV-1560A and depressurize its associated regulator
B. Isolate air from HCV-1560B and depressurize its associated regulator
C. Verify HCV-1560B is operable and closed
D. Direct EM to remove control power fuses for HCV-1560A

Proposed Answer: __B__

Explanation: Per step 5 of AOP-12 if containment integrity is lost due to excessive leakage or faulty control room indication for a containment isolation valve, then containment integrity is established by disabling the redundant valve for HCV-1560A, which in this case is HCV-1560B. This is done by step 5a or step 5b. Using step 5a, choice B is then the correct answer, Isolate air from HCV-1560B and depressurize its associated regulator.

Distractor A is not correct because the procedure directs disabling the redundant valve, not the valve with the faulty indication or leak. Distractor C is incorrect because it is not part of the procedure direction and because this valve, if manipulated, would cause a loss of containment integrity again. DIstractor D is incorrect because the procedure directs disabling the redundant valve, which is HCV-1560B, not HCV-1560A. These three distracters are all credible because they are correct in step 6 of AOP-12 when these steps are taken when inoperable automatic isolation valve(s) are inoperable for reasons other than excessive leakage or faulty control room indication.

Technical Reference(s): AOP-12, “Loss of Containment Integrity” Rev. 6

Proposed references to be provided to applicants during examination: Attached Figure Q22-1__

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	__(9)___
	55.43	_____

Comments:

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 controller setpoints are set at the normally selected value and 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. TCV-2897B's normal setpoint is 110 °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): ______________________________
(Attach if not previously provided) ______________________________
(including version/revision number) ______________________________

Proposed references to be provided to applicants during examination: _______________

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 __(7)___
55.43 _____

Comments:

Question # _24__

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	__2__	_____
	Group #	__1__	_____
	K/A #	__006 K6.03__	
	Importance Rating	__3.6_	_____

K/A Statement: (006 Emergency Core Cooling) Knowledge of the effect of a loss or malfunction the following will have on the ECCS: Safety Injection Pumps

Proposed Question:

A small break LOCA has occurred. The following plant conditions exist:

Pressurizer pressure is 750 psia and slowly raising
Pressurizer level is 0%
Reactor Vessel level is 46% and slowly raising
RCS subcooling is 23 °F
Both steam generator pressures are 700 psia
Narrow range steam generator levels are 29%
Containment pressure is 6 psig and stable
All containment air coolers are running
All charging pumps are currently running
HPSI pump SI-2A is currently running
HPSI pump SI-2B has tripped
No containment spray pumps are running
LPSI pump SI-1A is currently running
LPSI pump SI-2B has tripped
A recirculation actuation signal has occurred

Per EOP-03, LOSS OF COOLANT ACCIDENT, what action should be taken?

A. Start HPSI pump SI-2C to meet minimum required safety injection flow
B. Stop HPSI pump SI-2A as HPSI trip and throttle criteria have been met
C. Stop LPSI pump SI-1A as LPSI trip and throttle criteria have been met
D. Start CS pump SI-3A to commence cooling of recirculation flow

Proposed Answer: __C___

Explanation: Per EOP-03, and STM Vol. 15, LPSI pumps are supposed to trip following a RAS to ensure that they aren't operating at their shutoff head (and are not required for accident mitigation following a RAS). In this case, the pressurizer pressure is still above the LPSI shutoff head, and since the RAS has occurred, the recirculation flow path back to the SIRWT has been isolated. Therefore, LPSI pump SI-2A should have tripped and needs to be stopped to prevent damage to the pump. The LPSI trip and throttle criteria (per the EOP floating steps) has been reached. Answer "A" is plausible if minimum required safety injection flow was not being met, but due to the fact that pressurizer pressure and reactor vessel water level are raising, adequate injection is being maintained. Answer “B” is plausible in that HPSI trip and throttle criteria would be met if pressurizer level was greater than 10%, which it is not here. One HPSI pump is still required per the floating steps for a LOCA, and all HPSI pumps are only stopped for a MSLB. Answer “D” is plausible in that it is a viable method to cool recirculation flow following a RAS, however, the requirements for entering shutdown cooling, per EOP-03, have not yet been met. The containment spray pumps are not started until shutdown cooling is entered.

Technical Reference(s): __EOP-03 “Loss of Coolant Accident”, EOP/AOP Floating Steps, EOP/AOP Attachments, STM Vol. 15 ECCS__

(Attach if not previously provided) ______________________________

(including version/revision number) ______________________________

Proposed references to be provided to applicants during examination: ______________

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 __(7)___
55.43 _____

Comments:

Question # _25_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	__2__	_____
	Group #	__1__	_____
	K/A #	_076 K2.01_	
	Importance Rating	_2.7*_	_____

K/A Statement: (076 Service Water) Knowledge of bus power supplies to the following: Service water (Raw water for FCS)

Proposed Question:

Which raw water pumps are powered from the 1A3 bus?

A. AC-10A and AC-10B
B. AC-10A and AC-10C
C. AC-10B and AC-10D
D. AC-10C and AC-10D

Proposed Answer: __B___

Explanation: Per STM Vol. 35, Raw water system, bus 1A3 supplies pumps AC-10A and AC-10C, while bus 1A4 powers pumps AC-10B and AC-10D. Therefore, answer B is correct. The other answers are incorrect yet plausible as they are possible combinations of raw water pumps.

Technical Reference(s): __STM Vol. 35 "Raw Water System" Rev. 26, 11405-E-7 Sheets 1 and 2_

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

Proposed references to be provided to applicants during examination: _________________

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:

Question # _26_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_2__	_____
	Group #	_1___	_____
	K/A #	_062 A2.05_	
	Importance Rating	_2.9__	_____

K/A Statement: (062 AC Elect Dist.) Ability to (a) predict the impacts of the following malfunctions or operations on the ac distribution system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: methods for energizing a dead bus

Proposed Question:

The reactor is currently shutdown for refueling with a reactor coolant temperature of 110 $^{\circ}$F. 480VAC bus 1B3C has been deenergized for maintenance, which is now complete. However, due to other ongoing maintenance, 480VAC bus 1B3C will be energized from its alternate supply. All other 4160V and 480V buses are currently energized.

Current breaker status:

BT-1B3C is open
1B3C is open
All breakers on bus 1B3C are open
1B4C is closed, powering bus 1B3C-4C

Per OI-EE-2, 480 VOLT AC SYSTEM NORMAL OPERATION, what additional actions are necessary to energize bus 1B3C from its alternate source?

A. Ensure T1B-3C is open and close breaker 1B3C
B. Ensure T1B-3C is closed and close breaker 1B3C
C. Ensure T1B-3C is open and close breaker BT-1B3C
D. Ensure T1B-3C is closed and close breaker BT-1B3C

Proposed Answer: __C___

Explanation: Per OI-EE-2, Attachment 4, intentional realignment of the 480 VAC buses is prohibited at RCS temperatures greater than 300 degrees. Current RCS temperature is 110 degrees, so the bus can be aligned to its alternate source. Per Step 4 of Attachment 4, Steps 4.e and 4.h state to ensure breaker T1B-3C is open and close breaker BT-1B3C, making answer “C” correct. Answer “B” is plausible because those are the actions that would be taken if the bus was being powered from its normal supply (4160V bus 1A3). Answers “A” and “C” are incorrect but plausible if the applicant cannot remember which breaker goes to which supply (BT-1B3C connect the bus to the 4160VAC bus 1A4, 1B3C connects it to 1A3). T1B-3C supplies the transformer from the 1A3 bus to the 1B3C bus.

Technical Reference(s): __OI-EE-2, "480 Volt AC System Normal Operation" Rev. 94, Figure 8.1-1, "Simplified One-Line Diagram Plant Electrical System" Rev. 141____________________

(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 _(5)__
55.43 _____

Comments:

Question # _27_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	__2__	_____
	Group #	__1__	_____
	K/A #	_022 A4.01_	
	Importance Rating	_3.6__	_____

K/A Statement: (022 Containment Cooling) Ability to manually operate and/or monitor in the control room: CCS fans

Proposed Question:

Per OP-ST-VA-008, SURVEILLANCE TEST – CONTAINMENT VENTILATION SYSTEM CONTAINMENT FANS EXERCISE TEST, containment air cooling and filtering units, VA-3A/3B are considered acceptable if amperage is less than ____1____ amps, and containment air cooling units, VA-7C/D, are acceptable if amperage is less than ____2____.

A. 130; 130
B. 130; 250
C. 250; 130
D. 250; 250

Proposed Answer: __C___

Explanation: Per OP-ST-VA-008 acceptance criteria, VA-3A/B are acceptable as long as amperage is less than 250 amps and VA-7C/7D are acceptable as long as amperage is less than 130 amps, making C correct. Answer "B" is plausible if the individual gets the coolers reversed. Answer "A" is plausible if the individual has the misconception that the units have to meet the same requirement and remembers the amperage for VA-7C/D. Answer "D" is plausible if the individual thinks the units have the same amperage requirement and remembers the amperage requirement for VA-3A/3B

Technical Reference(s): __OP-ST-VA-0008, R10_________________
(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:

Question # _28_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_2__	_____
	Group #	_1___	_____
	K/A #	_063 A4.03_	
	Importance Rating	_3.0*_	_____

K/A Statement: (063 DC Elect Dist.) Ability to manually operate and/or monitor in the control room: Battery discharge rate

Proposed Question:

A loss of offsite power occurred seventy-five minutes ago. At that time, diesel generator 1 failed to start. Diesel generator 2 started successfully and is currently supplying power to its 4160 VAC bus. EOP-02, LOSS OF OFF-SITE POWER / LOSS OF FORCED CIRCULATION, is in progress.

What action is currently required by EOP-02 and the EOP/AOP FLOATING STEPS to minimize DC loads?

A. Stop LO-4, the DC oil pump
B. Transfer AI-41A power source to DC bus #2
C. Stop LO-12B, the DC seal oil pump
D. Transfer DC bus 1 power source to battery charger #3

Proposed Answer: __A___

Explanation: At this point, floating step BB applies, which requires minimizing DC loads if any DC bus is not energized by its charger (DC Bus 1). Per EOP Attachment 6, that requires stopping the DC oil pump after the turbine stops turning, approximately one hour into the event). Since it's greater than one hour since the turbine tripped, the DC oil pump should be stopped and answer "A" is correct. Answer "B" is a correct action for the loss of a DC bus (AOP-16), but the bus isn't lost, it's just lost AC power. Answer "C" is plausible in that after two hours, the operator is directed to secure this pump but two hours has not yet elapsed. Answer "D" is plausible in that the power supply can be cross-tied to battery charger #3, but these actions are not proceduralized.

Technical Reference(s): EOP-02, EOP Attachment 6 rev 29, page 28._____
(Attach if not previously provided) ________________________________
(including version/revision number) ________________________________

Proposed references to be provided to applicants during examination: _________________

Learning Objective: ________________________ (As available)

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

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 _(7)____
55.43 _____

Comments:

Question # _29_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_2___	_____
	Group #	_2___	_____
	K/A #	_086 A1.01_	
	Importance Rating	_2.9__	_____

K/A Statement: (086 Fire Protection) Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with Fire Protection System operating the controls including: Fire header pressure

Proposed Question:

The following conditions were noted during preparations for a surveillance test of fire protection system motor-driven pump FP-1A:

- A local operator was stationed at local panel AI-183 and reading fire main header pressure to the control room, which was steady at 120 psig
- FP-1A and FP-1B (the fire protection system diesel-driven pump) start switches were in the STOP position in the control room at CB-10/11
- Both FP-1A and FP-1B local controller hand switches were in AUTO

Suddenly, the local operator reports that pressure instantly dropped to 95 psig.

Based on these indications:

A. FP-1A started automatically after 10 seconds
FP-1B started automatically after 10 seconds

B. FP-1A started immediately
FP-1B started immediately

C. FP-1A started immediately
FP-1B started automatically after 10 seconds

D. FP-1A started automatically after 10 seconds
FP-1B started immediately

Proposed Answer: _B___

Explanation: Per STM Vol. 21, Fire Protection System, FP-1A is the electric driven fire pump and starts automatically when fire main pressure drops below 110 psig. FP-1B will start after 10 seconds if FP-1A fails to start. FP-1B also starts immediately if fire main pressure drops below 100 psig. Therefore, both start immediately and answer “B” is correct. Answer A is plausible if the applicant thinks that both pumps have a time delay. Answer C is plausible because this would happen if the fire main pressure dropped below 110 psig (but above 100 psig) and pump A failed to run. Answer D is plausible if the applicant reverses which pump has a time delay circuit, and does not recall the automatic start if header pressure drops below 100 psig.

Technical Reference(s): __STM Vol. 21 Fire Protection System" Rev. 29, OI-FP-1, "Operating Instruction – Fire Protection System Water System" Rev. 77, Lesson Plan 7-11-12, "Fire Protection" Rev. 16

(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 __(5)___
55.43 _____

Comments:

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:

Question # _31_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_2___	_____
	Group #	_2__	_____
	K/A #	_015 A3.02_	
	Importance Rating	_3.7__	_____

K/A Statement: (015 Nuclear Instrumentation) Ability to monitor automatic operation of the NIS, including: Annunciator and alarm signals

Proposed Question:

While operating at 100% power, the "Rod Drop Nuclear Instrumentation Channel" annunciator alarms on CB-4. Which of the following events will cause that alarm?

A. One of the power range NI channels has detected that power has dropped 8% in 8 seconds
B. One of the power range NI channels has detected that power has dropped 10% in 12 seconds
C. One of the wide range NI channels has detected that power has dropped 8% in 8 seconds
D. One of the wide range NI channels has detected that power has dropped 10% in 12 seconds

Proposed Answer: __A___

Explanation: Per STM Vol. 29 and Lesson plan 07-12-19, under the section describing power range safety channels, it states that a difference of 8% in 8 seconds between the prompt and delayed NI power signals developed in the rod drop detection circuit, it actuates a rod drop nuclear instrumentation channel alarm, making answer "A" correct. Answer "B" is plausible if the student does not remember that the power drop must occur such that 1% per second over an eight second interval is met. Answers "C" and "D" are plausible if the student believes that it is the wide range NI channels that have input into the rod drop bistable. The wide range NI's generate the start-up rate signal and the permissive for zero power mode bypass, as well as inputting to the SCEAPIS (CEA position indication) digital control system.

Technical Reference(s): STM Vol. 29, "Nuclear Instrumentation System", Lesson Plan 07-12-19___

(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 # __X___ (Note changes or attach parent)
New ________

Question History: Last NRC Exam ___2002_
(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:

Question # _32_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_2___	_____
	Group #	_2___	_____
	K/A #	_011 A1.03_	
	Importance Rating	_2.8_	_____

K/A Statement: (011 Pressurizer Level Control) Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PZR LCS controls including: VCT level

Proposed Question:

CVCS is in normal at power alignment except for mode selector switch HC-218-1 in AUTO. VCT level is at 86 percent when pressurizer level control channel LI-101X fails high. No boration or dilution is in progress.

VCT level will _____1_____, then _____2_____.

A. Increase to 94.4%; letdown would divert to the RWTS
B. Increase to 91.5%; automatic makeup would stop
C. Decrease to 84%; charging pump suction would swap to SIRWT
D. Decrease to 82.5%; automatic makeup would start

Proposed Answer: __A___

Explanation: Per STM Vol. 12, Chemical and Volume Control System, both the VCT inlet valve and VCT makeup valves are in manual mode for normal at power alignment. Per the question stem, the VCT inlet valve is in automatic mode. Therefore, when VCT level reaches 94.4%, it will automatically divert letdown to the radioactive waste treatment system, making “A” the correct answer. Answer “B” is incorrect but plausible because that is the setting at which the inlet valve will return letdown to the VCT. Answer “C” is incorrect but plausible because that would be the action if makeup was in automatic (not used at FCS). Answer “D” is incorrect but plausible because 82.5% (per EM-219) is when auto makeup will stop.

Technical Reference(s): __STM Vol. 12 CVCS, Rev. 44____
(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	_(5)__
	55.43	_____

Comments:

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 is currently 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:

Question # _34_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_2__	_____
	Group #	_2___	_____
	K/A #	_033 A2.02_	
	Importance Rating	_2.7_	_____

K/A Statement: (033 Spent Fuel Pool Cooling) Ability to (a) predict the impacts of the following malfunctions or operation on the Spent Fuel Pool Cooling System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of SFPCS

Proposed Question:

Given the following:

- The plant is in a refueling outage
- A full core offload was just completed
- Shutdown cooling is in service
- Refueling cavity is flooded
- Spent fuel pool cooling pump, AC-5A, is out of service for maintenance
- Spent fuel pool cooling pump, AC-5B, has just tripped and cannot be restarted

What is the expected rate of temperature increase of the spent fuel pool and what actions are taken per AOP-36, LOSS OF SPENT FUEL POOL COOLING, to establish alternate spent fuel pool cooling?

A. 6.3°F/hr; Align fuel transfer canal drain pumps, AC-13A and AC-13B, to circulate water between the spent fuel pool and the safety injection and refueling water storage tank, SI-5
B. 6.3°F/hr; Align the low pressure safety injection pumps, SI-1A and SI-1B, to circulate water between the spent fuel pool and the shutdown cooling system
C. 12.6°F/hr; Align fuel transfer canal drain pumps, AC-13A and AC-13B, to circulate water between the spent fuel pool and the safety injection and refueling water storage tank, SI-5
D. 12.6°F/hr; Align the low pressure safety injection pumps, SI-1A and SI-1B, to circulate water between the spent fuel pool and the shutdown cooling system

Proposed Answer: __D___

Explanation: Per Lesson Plan 07-11-24, Rev.12, a loss of cooling capability immediately after a complete core unload would result in a temperature rise of the pool water at about 12.6°F/hr. If only 1/3 of the core had been unloaded, this would result in a rise of about 6.3°F/hr, making it a plausible distractor. Per AOP-36, with the RCS vented and SDC in service, the operators are directed to align the LPSI pumps to circulate water between the pool and SDC. The fuel transfer canal pumps, in conjunction with the fuel pool circulating pumps, are used to circulate water between the spent fuel pool and the SIRWT when the reactor is critical or when PPLS is

NOT blocked. In this case, the fuel pool pumps are both unavailable and so this method could not be used, but it is a method in AOP-36, and so is plausible. Therefore, answers "A", "B", and "C" are all plausible but incorrect and answer "D" is correct.

Technical Reference(s): __Lesson Plan 07-11-24, Rev. 12, ARP-AI-100/A52, Rev. 8, AOP-36 "Loss of Spent Fuel Pool Cooling", 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____

(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 _(5)____

55.43 _____

Comments:

Question # _35_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_2___	_____
	Group #	_2___	_____
	K/A #	_014 A4.02_	
	Importance Rating	_3.4__	_____

K/A Statement: (014 Rod Position Indication) Ability to manually operate and/or monitor in the control room: Control rod mode-select switch

Proposed Question:

During a reactor startup, as group 4 CEAs are being withdrawn (from 50 inches) with the Control Rod Drive Mode Selector switch in the "Manual-Sequential" position, an automatic reactor trip occurred due to high startup rate. One group 4 CEA failed to insert because its clutch failed to disengage. How will the CEA Rod Rundown (RRD) feature function in this situation?

A. The RRD function will immediately begin to insert the withdrawn CEA
B. The RRD function will begin to insert the withdrawn CEA as soon as the mode selector switch is placed in the "Manual Individual" position
C. The RRD function will NOT insert the withdrawn CEA due to the zero power mode bypass being enabled
D. The RRD function will NOT insert the withdrawn CEA due to the CEA position deviation

Proposed Answer: __D___

Explanation: Per STM. 11, the RRD feature is the only automatic function still enabled for FCS rod drive system. From the STM: "Rod rundown is designed to start all CEDMs driving in when a reactor trip occurs. Rod rundown is defeated when the mode selector switch is in OFF or any rod block actuates." The assumption is that a CEA position deviation causes the rod block which then causes RRD to be defeated.

Technical Reference(s): __STM 11, "Control Rod Drive System" Rev. 19, Lesson Plan 07-12-26, Rev. 12, __________

(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 #	_071702 025____
	Modified Bank #	_______ (Note changes or attach parent)
	New	_____

Question History: Last NRC Exam ___2005_____

(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:

Question # _36_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	__2__	_____
	Group #	__2__	_____
	K/A #	__075 K4.01_	
	Importance Rating	__2.5_	_____

K/A Statement: (075 Circ Water) Knowledge of circulating water system design feature(s) and interlock(s) which provide for the following: Heat sink

Proposed Question:

The INTERLOCKS that must be satisfied to start a Circulating Water pump are:

A. Discharge MOV closed, suction sluice gate open
B. Discharge MOV open, seal water supply valve open
C. Discharge check valve closed, seal water supply valve open
D. Discharge check valve closed, suction sluice gate open

Proposed Answer: __D___

Explanation: Per STM Vol. 7, Lesson Plan 07-11-03, and OI-CW-1, the pump suction inlet sluice gate (CW-15C), must be open and the discharge check valve must be closed before the circulating water pump will start.

Technical Reference(s): _STM Vol. 7 “Circulating Water System” Rev. 37, Lesson Plan 07-11-03, “Circulating Water System”, Rev. 12, OI-CW-1, “Circulating Water System Normal Operation” Rev. 62

(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 #	___071103 016___
	Modified Bank #	_______ (Note changes or attach parent)
	New	_______

Question History: Last NRC Exam __N/A___
Pulled from FCS bank, written in 1989 but it has not been used on an NRC exam.

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

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

Comments:

Question # _37_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	__2__	_____
	Group #	__2__	_____
	K/A #	__002 K6.03_	
	Importance Rating	__3.1_	_____

K/A Statement: (002 Reactor Coolant) Knowledge of the effect of a loss or malfunction on the following RCS components: Reactor vessel level indication

Proposed Question:

The RCS is being drained to mid-loop using OI-RC-2A, RCS FILL AND DRAIN OPERATIONS, Attachment 4, "RCS Draining". The following plant conditions exist:

Fuel is in the reactor vessel
The reactor vessel head is in place
The pressurizer manway has been removed
RCS level indicators, LI-119, LI-197, and LI-199 are in service and all indicating 1014'
Both trains of RVLMS are unavailable
Two CETs are providing temperature indication

How large of a discrepancy between LI-119, LI-197, and LI-199 would require an evaluation to determine the need to stop draining the RCS?

A. 6 inches when RCS level reaches 1014'
B. 6 inches when RCS level reaches 1011'
C. 3 inches when RCS level reaches 1014'
D. 3 inches when RCS level reaches 1011'

Proposed Answer: ___B__

Explanation: Answer "A" is incorrect, LI-119, LI-197, and LI-199 should normally agree by the time RCS Level reaches 1011' (not 1014'). (Step 2 note). Answer "B" is correct. If there is a large discrepancy (6 inches) during draining below 1011', then evaluate the need to stop draining and notify I&C. Answer "C & D" are incorrect, the discrepancy is less than 6 inches.

Technical Reference(s): _OI-RC-2A, "RCS Fill and Drain Operations" Rev. 74__
(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	__(7)___
	55.43	_____

Comments:

Question # _38_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	__2__	_____
	Group #	__2__	_____
	K/A #	__001 K5.28_	
	Importance Rating	__3.5_	_____

K/A Statement: (001 Control Rod Drive) Knowledge of the following operational implications as they apply to the CRDS: Boron reactivity worth vs. boron concentration, i.e., amount of boron needed (ppm) to change core reactivity to desired amount.

Proposed Question:

The group 4 CEAs will be partially inserted into the core for an approved core physics test. During the CEA insertion, reactor power will be held constant and RCS T-cold will be maintained at its programmed value. Assuming the reactivity change due to CEA insertion is the same, conducting the test at the end of cycle will require a _____1_____ change in boron concentration and _____2_____ dilution water than conducting this test at the beginning of cycle.

A. Larger; more
B. Larger; less
C. Smaller; more
D. Smaller; less

Proposed Answer: _C____

Explanation: At EOC, the boron worth is greater so it takes a smaller change in boron concentration but it also takes much more dilution water per ppm change, making answer "C" correct. Answer "A" is plausible if boron worth was less. Answer "B" is plausible if it was thought that it would take less dilution water (and that boron worth is less at EOC). Answer "D" is plausible if the applicant thought it would take less dilution water.

Technical Reference(s): ___TDB-V.12, "Miscellaneous Formula Sheet, Rev. 10
The boron concentration information is in GFECBT-R04-LP.doc pg. 61 and 77 (proprietary, so not saving to our drive)

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

Proposed references to be provided to applicants during examination: ___None_________

Learning Objective:	_07-11-02 03.05 Calculate the amount of concentrated boric acid or water required to adjust Reactor Coolant System boron concentration_ (As available)	
Question Source:	Bank #	__071102 059____
	Modified Bank #	_______ (Note changes or attach parent)
	New	_______
Question History:	Last NRC Exam	__2007______

(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 __(1)___	
	55.43 _____	

Comments:

Question # _39_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	__1__	_____
	Group #	__1__	_____
	K/A #	_000055 EK3.02___	
	Importance Rating	_4.3__	_____

K/A Statement: (055 Station Blackout) Knowledge of the reasons for the following responses as they apply to the SBO: Actions contained in EOP for loss of offsite and onsite power

Proposed Question:

Given the following:

Station blackout has occurred.
FW-10 is currently supplying AFW flow to both S/G's
Condenser vacuum is 18 inches Hg and lowering
Work is in progress to restore offsite power and DG #1
EOP-07, STATION BLACKOUT, has been entered and actions through Step 4 completed

Per Step 5 of EOP-07, what are the preferred actions the operator should take and the basis for those actions?

A. Use steam dumps and turbine bypass valves for cooldown because vacuum levels are currently sufficient to use the condenser as the preferred cooldown method
B. Ensure that steam dumps and turbine bypass valves are closed to allow continued use of the atmospheric dump valve for cooldown
C. Close the main steam isolation valves and bypass valves to prevent pressurization of the condenser and use the main steam safety valves for cooldown
D. Isolate steam generator blowdown to minimize the loss of condenser vacuum and extend the time that the condenser can be used for cooldown

Proposed Answer: __B__

Explanation: When condenser vacuum is less than 19 inches Hg, the procedure directs isolating the condenser by ensuring the steam dumps and turbine bypass valves are closed. This allows use of additional cooldown methods, including the ADV, making "B" correct. "A" is incorrect but plausible if condenser vacuum was 19 inches Hg or greater. "C" is incorrect but plausible because that action is directed by procedure if the MSIV's are closed. "D" is incorrect but plausible because the EPG directs this action because its reference plant has blowdown discharge to the condenser. At FCS, it discharges to the raw water system and isolation is not necessary.

Technical Reference(s): ___Lesson Plan 7-18-17, EOP-07, Station Blackout, TBD-EOP-07, Station Blackout, Rev. 14_______

(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 _(10)___
55.43 _____

Comments:

Question # _40_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	__1__	_____
	Group #	__1___	_____
	K/A #	_000011 EK2.02__	
	Importance Rating	_2.6*__	_____

K/A Statement: (Large Break LOCA) Knowledge of the interrelations between the following and the Large Break LOCA: Pumps

Proposed Question:

During a LOCA, the EOPs may direct that all RCPs be secured. Which one of the following is the basis for taking this action?

A. To slow the rate of RCS depressurization
B. To prevent pump damage from running under voided conditions
C. To allow stratification of fluid phases and reduce rate of inventory loss
D. To remove the RCPs as a source of heat input to the RCS

Proposed Answer: __C___

Explanation: Per lesson plan 07-15-23, RCPs are tripped once NPSH can no longer be maintained in order to conserve RCS inventory. Instead of having a beneficial transition from liquid to steam flow, a steam-water mixture would continue to flow out of the break. It causes less water loss at the beginning of the transient but results in a larger loss over the whole transient due to the slower pressure decrease. Therefore, answer C is correct. Answer A is incorrect but plausible in that tripping RCP's does not slow the RCS depressurization. Answer B is incorrect but plausible because tripping RCP's would prevent damage to the pump, but is not the basis for taking the action. Answer D is incorrect but plausible because this would be the reason for tripping the RCP's for a loss of all feedwater event, but not a LOCA.

Technical Reference(s): ______________________________
(Attach if not previously provided) ______________________________
(including version/revision number) ______________________________

Proposed references to be provided to applicants during examination: _________________

Learning Objective: _________________________ (As available)

Question Source:	Bank #	__071813 002___
	Modified Bank #	_______ (Note changes or attach parent)
	New	_______

Question History:	Last NRC Exam	__1995____

(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:

Question # _41_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_1___	_____
	Group #	_1___	_____
	K/A #	_000008 AK1.02_	
	Importance Rating	_3.1_	_____

K/A Statement: (008 PZR Vapor Space Accident) Knowledge of the operational implications of the following concepts as they apply to a Pressurizer Vapor Space Accident: Change in leak rate with change in pressure

Proposed Question:

Pressurizer safety valve RC-141 leakage increases to the point where it exceeds the capacity of the charging pumps. The plant has been tripped, standard post trip actions have been completed, safety injection is initiated, and operators have transitioned to EOP-03 “Loss of Coolant Accident”.

Operators will commence a _____1_____ plant cooldown and depressurization using the steam generators. Following this cooldown and depressurization, the break flow is _____2_____.

A. Controlled; unchanged
B. Controlled; reduced
C. Rapid; unchanged
D. Rapid; reduced

Proposed Answer: __D___

Explanation: Per Lesson Plan 07-15-23 and TBD-EOP-03, if the break is unisolable, the plant is rapidly cooled and depressurized to minimize break flow (and maximize SI flow). Since this leakage is through a safety valve, it is unisolable, therefore answer “D” is correct. Answer “B” would be correct is the break flow was isolable, which it is not in this case. Answers “A” and “C” are incorrect but plausible if the applicant does not understand the relationship between pressure and break flow.

Technical Reference(s): __Lesson Plan 07-15-23 “Loss of Coolant Accidents” Rev. 10, TBD-EOP-03 “Loss of Coolant Accident” Rev. 36__

(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 __(8)___
	55.43 _____

Comments:

Question # _42_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_1___	_____
	Group #	_1___	_____
	K/A #	_000022 AA1.01_	
	Importance Rating	_3.4_	_____

K/A Statement: (00022 Loss of Reactor Coolant makeup) Ability to operate and/or monitor the following as they apply to the Loss of Reactor Coolant Makeup: CVCS letdown and charging

Proposed Question:

AOP-33, CVCS LEAK has been entered for a suspected CVCS leak.

Current plant parameters are as follows:

RCS pressure 2050 psia and slowly decreasing
Pressurizer level 55% and slowly decreasing
Tave is on program
VCT level is 45% and slowly increasing
Reactor coolant drain tank level is stable
Pressurizer quench tank level is stable
Auxiliary building sump tank level is slowly increasing

Actions have been completed through Step 4 of AOP-33 (as listed in bold below).

Step 1: Place all Charging pump control switches in "PULL-TO-LOCK"

Close ALL of the following valves:
- **TCV-202, Letdown Isolation Valve**
- **HCV-204, Letdown Isolation Valve**
- **HCV-238, Loop 1 Charging Isolation**
- **HCV-239, Loop 2 Charging Isolation**
- **HCV-240, PZR Auxiliary Spray Isolation Valve**
- **HCV-249, PZR Auxiliary Spray Isolation Valve**

Step 2: Implement the Emergency Plan

Step 3: If pressurizer level is lowering at an abnormal rate, then….. (it is not therefore this step is not applicable)

Step 4: IF VCT level is lowering,
THEN close LCV-218-2, VCT Outlet Valve

Based on the above, which ONE of the following relief valve failures is the most probably location of the leak?

A. Relief valve CH-224, downstream of letdown heat exchanger CH-7
B. Relief valve CH-208, RCP seal leakage piping
C. Relief valve CH-180, charging pump CH-1A suction piping
D. Relief valve CH-183, charging pump CH-1A discharge piping

Proposed Answer: __C___

Explanation: Per AOP-33, LCV-218-2 is closed only if VCT level is decreasing following isolation of letdown and CVCS. This indicates that the leak is not in the letdown line. Therefore, answer "A" is incorrect. If VCT level recovers following closure of LCV-218-2, then the leak is not in the VCT, therefore answer "B" is incorrect. The leak must be in the charging line. Per STM Vol. 12 and Fort Calhoun P&ID Sheets 11, 18, 136, and 137, the discharge piping relief valve relieves back to the suction side of the pump. The suction piping relief valve relieves to the equipment drain header and thus to the aux building sump, whose levels are increasing. Therefore, answer "D" is incorrect and answer "C" is correct.

Technical Reference(s): _STM Vol. 12, "Chemical and Volume Control System" Rev. 44, AOP-33 "CVCS Leak", TBD-AOP-33 "CVCS Leak Rev. 7, Sheets 11, 18, 136 and 137 from FCS P&ID sheets______

(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 _(7)__
55.43 _____

Comments:

Question # _43__

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_1___	_____
	Group #	_1___	_____
	K/A #	_000026 2.4.35__	
	Importance Rating	_3.8_	_____

K/A Statement: (000026 Loss of CCW) For Loss of Component Cooling Water, knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.

Proposed Question:

A loss of component cooling water has occurred. The reactor is tripped and standard post trip actions complete. Operators are currently continuing in AOP-11, LOSS OF COMPONENT COOLING WATER, to manually align raw water to provide cooling to desired components. Control room ventilation fan VA-46A is in service and a local operator has been sent to establish raw water flow to this component.

Per Step 12.i of AOP-11, the operator will establish flow by:

A. Unlocking and releasing hand-jacks from both inlet and outlet valves and opening these valves by placing the three-way manual control valve to “OPEN”
B. Unlocking and releasing hand-jacks from the inlet valve ONLY and opening this valve by placing the three-way manual control valve to “OPEN”
C. Placing the three-way manual control valve to “OPEN” for both inlet and outlet valves
D. Placing the three-way manual control valve to “OPEN” for the inlet valve ONLY

Proposed Answer: __A___

Explanation: Per AOP 11 Step 12.i, RW flow is established to VA-46A by unlocking and releasing hand-jacks from both HCV-2898C and HCV-2898D (the inlet and outlet valves) and placing IA-HCV-2898C/D-TV to “OPEN”. . Therefore, answer “A” is correct. Answer “B” is plausible if it’s thought that the outlet is already opened or is interlocked with the inlet valve similar to CCW valve manipulations in the control room for this cooler. Distractors C” and “D” are plausible if the applicant does not remember that both of these valves are locked closed and must be unlocked and hand-jack released to open.

Technical Reference(s): __TBD-AOP-11 “Loss of Component Cooling Water” Rev. 15, Lesson Plan 7-17-11 “Loss of Component Cooling Water AOP-11” Rev. 8, STM Vol. 8, “Component Cooling Water System”, Rev. 39__

(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 _(10)____
55.43 _____

Comments:

Question # _44_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_1___	_____
	Group #	_1___	_____
	K/A #	_000038 EK3.06_	
	Importance Rating	_4.2__	_____

K/A Statement: (000038 SGTR) Knowledge of the reasons for the following responses as they apply to the SGTR: Actions contained in EOP for RCS water inventory balance, S/G tube rupture, and plant shutdown procedures

Proposed Question:

Given the following conditions:

Steam generator tube rupture has occurred on RC-2B
Reactor is tripped
Safety injection has actuated
Standard post trip actions are complete
EOP-04 has been entered.
RCS pressure is 1350 psig and stable
RCS Th is 545°F

Per EOP-04, STEAM GENERATOR TUBE RUPTURE, the next steps the operators will take will be to cooldown the RCS to less than _____1_____ because _____2_____.

A. 510 degrees F; this meets the 50 degrees subcooling requirement after depressurization
B. 525 degrees F; this meets the 20 degrees subcooling requirement after depressurization
C. 510 degrees F; this value is less than the saturation temperature to lift a main steam safety valve
D. 525 degrees F; this value is less than the saturation temperature to lift a main steam safety valve

Proposed Answer: __C___

Explanation: Per TBD-EOP-04, page 17, the goal of this step is to reduce the RCS hot leg temperature to less than 510°F prior to isolating the affected S/G. This reduction in temperature is to prevent lifting main steam safety valves in the affected S/G. The temperature in the isolated S/G will be essentially Thot since it is no longer being used as a heat sink.
The first MSSVs open at 1000 psia which corresponds to a saturation temperature of 545°F. The next major step to prevent lifting the MSSVs will be to depressurize the RCS to less than 1000 psia. A hot leg temperature of 545°F and 1000 psia RCS pressure would result in a saturated RCS. A Thot less than 525°F would allow 20°F subcooling but this would not meet the RCP NPSH requirements. The 510°F is used because it is less than the saturation temperature to lift main steam safety valves and is low enough to meet the NPSH requirements for the RCPs

when RCS pressure is reduced to 1000 psia. This makes Distractor B and D incorrect because 525 degrees F is the wrong temperature. Distractor A is incorrect because the subcooling for RCP NPSH at 1000 psig is 20 degrees per the TBD-EOP basis document.

Technical Reference(s): _EOP-04, “Steam Generator Tube Rupture”, TBD-EOP-04, “Steam Generator Tube Rupture”, Rev. 26, EOP/AOP Attachments, Attachment 3, Rev. 29, Lesson plans 7-15-33, Rev. 5 and 7-18-14, Rev. 17___

(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 _(10)___
55.43 _____

Comments:

Question # _45_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_1___	_____
	Group #	_1___	_____
	K/A #	_000065 AA2.08__	
	Importance Rating	_2.9*_	_____

K/A Statement: (000065 Loss of Instrument Air) Ability to determine and interpret the following as they apply to the Loss of Instrument Air: failure modes of air-operated equipment

Proposed Question:

An instrument air line blockage occurs that results in a loss of instrument air to Condensate Makeup valve, LCV-1190, and Condensate Dump valve, LCV-1193. Assuming no operator actions were taken, how would Condensate Storage Tank level respond over the next six hours?

A. Condensate Storage Tank level would steadily lower
B. Condensate Storage Tank level would steadily rise
C. Condensate Storage Tank level would steadily lower for approximately 4 hours, then begin to lower at a faster rate
D. Condensate Storage Tank level would steadily rise for approximately 4 hours, then begin to rise at a faster rate

Proposed Answer: __C___

Explanation: Per STM 20, Feedwater and Condensate System, LCV-1190 fails open on a loss of air after the accumulators run out, which takes approximately 4 hours. LCV-1193 fails closed on a loss of instrument air. Because the recirculation to the condensers continues through LCV-1172, this results in a slow lowering of CST level. Once the accumulator runs out of pressure, LCV-1190 fails open, which will allow CST flow to the condenser hotwell, which will cause the CST level to lower at an even faster rate. This makes answer C correct. Answer A is plausible but incorrect because this is what would occur if there were no accumulators on LCV-1190. Answer B is incorrect but plausible if there were no accumulators and the candidate mistakes which valve fails closed on a loss of air and which fails open. Answer D is incorrect but plausible if the candidate remembers the accumulators but mistakes which valve fails closed on a loss of air and which fails open.

Technical Reference(s): _STM Vol. 20, “Feedwater and Condensate System” Rev. 46_
(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 #	___07-11-04-002___	
	Modified Bank #	_______ (Note changes or attach parent)	
	New	_______	

Question History:	Last NRC Exam	___2001______

Used with KA 074.EA1.21 on 2001 FCS NRC exam.

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

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

Comments:

Question # _46_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_1___	_____
	Group #	_1___	_____
	K/A #	_00009 EA2.39__	
	Importance Rating	_4.3_	_____

K/A Statement: (00009 Small Break LOCA) Ability to determine or interpret the following as they apply to a small break LOCA: Adequate core cooling.

Proposed Question:

Given the following:

A small break LOCA occurred in containment ten minutes ago
Reactor is tripped
Both trains of SI has actuated
Both steam generators are removing decay heat
One RCS pump per loop is operating
Pressurizer level is 15% and stable
RVLMS is 43% and stable
Pressurizer pressure is 1200 psia and slowly decreasing
CET's are reading 553°F and stable

Per EOP Floating Step A, HPSI STOP AND THROTTLE CRITERIA, can the operators stop/throttle a HPSI pump based on the above conditions?

A. Yes, all criteria are met
B. No, pressurizer level criterion is NOT met
C. No, RVLMS level criterion is NOT met
D. No, RCS subcooling criterion is NOT met

Proposed Answer: _D____

Explanation: Per EOP/AOP Floating Step A, a HPSI pump can be stopped if all of the following are met: at least one S/G available for heat removal (there are two in this question); PZR level greater than or equal to 10% and not lowering (yes); RVLMS indicates level is at or above top of the Hot Leg, 43% (yes); and RCS subcooling greater than or equal to 20°F. Per the steam tables, the saturation temperature at 1200 psia is 567°F. To meet the stop/throttle criteria, the CET's would need to indicate less than 547°F, which they don't (only 13°F of subcooling). Thus, answer "D" is correct. Answers "B" and "C" are incorrect but plausible if the applicant doesn't remember the HPSI stop and throttle criteria correctly. Answer "A" is plausible the applicant doesn't remember that the RCS must be 20°F subcooled (not just less than saturation).

Technical Reference(s): __Steam Tables, Lesson plans 07-18-13 and 07-15-23, EOP/AOP Floating Steps, Rev. 1__

(Attach if not previously provided) ______________________________

(including version/revision number) ______________________________

Proposed references to be provided to applicants during examination: __Steam Tables______

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 _(5)____
55.43 _____

Comments:

Question # _47_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	__1__	_____
	Group #	__1__	_____
	K/A #	_000027 AK2.03_	
	Importance Rating	_2.6_	_____

K/A Statement: (027 PZR Pressure Control Malfunction) Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following: Controllers and positioners

Proposed Question:

The reactor tripped 20 minutes ago. The following conditions are observed:

“PRESSURIZER PRESSURE OFF NORMAL HI-LO” channel X and Y are in alarm
“PRESSURIZER LEVEL OFF NORMAL HI-LO” channel X and Y are in alarm
PRC-103Y (controlling channel) indicates 1980 psia and slowly lowering
PRC-103X indicates 1978 psia and slowly lowering
All backup heaters are in auto and deenergized
LRC-101Y (controlling channel) indicates 29% and steady
LRC-101X indicates 28% and steady
Letdown flow is 36 gpm
One charging pump is running
Tcold indicates 533°F,Thot indicates 534°F, both are stable

What action should be taken to restore RCS pressure to normal?

A. Select PRC-103X as the controlling pressure channel
B. Take manual control of PRC-103Y to raise pressurizer pressure
C. Select LRC-101X as the controlling level channel
D. Place all pressurizer heater control switches in the “ON” position

Proposed Answer: __C___

Explanation: With this pressurizer level, more charging pumps should be operating and letdown should be less, thus the level controller is not functioning properly and choice C is correct. The heaters are deenergized due to low pressurizer level causing RCS pressure to lower slowly. Therefore, choices A, B, D are incorrect.

Technical Reference(s): _STM Vol. 36 “RCS Instrumentation & Reactor Regulating Systems” Rev. 23, STM Vol. 37 “Reactor Coolant System” Rev. 42__

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

Proposed references to be provided to applicants during examination: __None_______

Learning Objective: ___07-11-20 4.00______ (As available)

Question Source: Bank # _07-11-20 156_
Modified Bank # _______ (Note changes or attach parent)
New _______

Question History: Last NRC Exam ___2007_____
(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:

Question # _48_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	__1__	_____
	Group #	__1__	_____
	K/A #	_000056 AA2.54_	
	Importance Rating	_2.9_	_____

K/A Statement: (000056 Loss of Off-site Power) Ability to determine and interpret the following as they apply to the Loss of Offsite Power: Breaker position (remote and local)

Proposed Question:

The plant was operating at full power with a normal electrical lineup and no equipment out of service. The following sequence of events occurred:

- The reactor tripped following a fault on bus 1A1.
- A few seconds later, 161 KV offsite power was lost.

What is the expected control switch indication for breaker 1A44 on CB-20 one minute later? (Note: no operator actions have been taken and all equipment operated as designed)

A. Red Flagged with Green light on
B. Red Flagged with Red light on
C. Green Flagged with Green light on
D. Green Flagged with Red light on

Proposed Answer: __A___

Explanation: Per drawing Ref. Figure 8.1-1 “Simplified One Line Diagram – Plant Electrical System PI&D” Rev. 141 and STM-14, rev 40, page 110,

The fault on bus 1A1 caused a RCP to trip which resulted in a loss of flow reactor trip. The reactor trip caused a turbine-generator trip that opened breakers 3451-4 and 3451-5 and disconnected 345 KV power to the plant.

Bus 1A4 is normally powered from 161 KV. When the reactor tripped, Breaker 1A44 remained closed. When 161 KV was lost, power was also lost to bus 1A4 and breaker 1A44 tripped allowing breaker 1AD2 to close when D/G-2 reached rated speed. Therefore answer “A” is correct.

Technical Reference(s): __Ref. Fig. 8.1-1 “Simplified One Line Diagram Plant Electrical System P&ID” Rev. 141____

(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 _(4)___
55.43 _____

Comments:

Question # _49_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_1___	_____
	Group #	_1___	_____
	K/A #	_000057 2.2.37_	
	Importance Rating	_3.6__	_____

K/A Statement: (0057 Loss of Vital AC Instrument Bus) Ability to determine operability and/or availability of safety related equipment associated with a Loss of Vital AC Instrument Bus.

Proposed Question:

With the plant operating at 25% power, the "INVERTER A TROUBLE" alarm annunciated. Just prior to the alarm, the ERF indicated Inverter "A" was at 117 volts and 25 amps.

Which of the following combination of indications would confirm that AI-40A (Instrument Bus A) is being supplied by the Bypass Transformer of Inverter "A"?

A. Inverter output voltage zero, inverter output current zero, instrument bus current zero
B. Inverter output voltage zero, inverter output current zero amps, instrument bus voltage 117 V
C. Inverter output voltage 117 V, inverter output current 25 amps, instrument bus voltage 117 V
D. Inverter output voltage 117 V, instrument bus voltage zero, instrument bus current zero

Proposed Answer: __C___

Explanation: Per Lesson Plan 0713-04, "125 VDC and 120 VAC Distribution", inverter output voltage should read between 115-118 V, inverter output current between 20-40 amps, and instrument bus voltage ~115 V. Therefore, answer C is correct. The other answers are plausible but incorrect based on the candidate misunderstanding the electrical scheme.

Technical Reference(s): LP 0713-04 "125 VDC and 120 VAC Distribution" Rev. 16, pages 31-32, 36 _________________

(Attach if not previously provided) ______________________________

(including version/revision number) ______________________________

Proposed references to be provided to applicants during examination: ___None__________

Learning Objective: _ 0713-04 01.04 __________ (As available)

Question Source: Bank # 07-13-04 003 ______

Modified Bank # _______ (Note changes or attach parent)

New _______

Question History: Last NRC Exam __1997_

(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:

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: _________________________ (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:

Question # _51_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_1___	_____
	Group #	_1___	_____
	K/A #	_CE/E02 2.4.2_	
	Importance Rating	_4.5__	_____

K/A Statement: (CE/E02 Reactor Trip-Stab-Recovery) Knowledge of system set points, interlocks, and automatic actions associated with EOP entry conditions for Reactor Trip, Stabilization and Recovery

Proposed Question:

While at 100% power, RCP RC-3D trips due to high motor current. No other failures occur.

In addition to the reactor trip on low RCS flow, which of the following will occur automatically for this case?

1) AFW initiates
2) Steam dumps and turbine bypass valve open
3) Generator breakers trip
4) SI actuates
5) Turbine stop and intercept valves close
6) Diesels start

A. 2, 3, 5 ONLY
B. 1, 3, 4 ONLY
C. 1, 5, 6 ONLY
D. 2, 4, 6 ONLY

Proposed Answer: __A__

Explanation: Per EOP-00, based on no failures other than an RCP pump failure, only the steam dumps and turbine bypass valves open, generator breakers trip, and turbine stop and intercept valves close. The others should not occur. This makes A correct. The other answers are combinations that include items that should not automatically occur for these conditions and don’t include items that should automatically occur.

Technical Reference(s): __EOP-00, Standard Post Trip Actions, Rev. 27, STM Vol. 38 Reactor Protective System and Diverse Scram System, Rev. 20, STM Vol. 19 Engineered Safeguards Control System, Rev. 37___

(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:

Question # _52_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_1___	_____
	Group #	_1___	_____
	K/A #	_000025 AK2.02_	
	Importance Rating	_3.2*_	_____

K/A Statement: (000025 Loss of RHR system) Knowledge of the interrelations between the Loss of Residual Heat Removal System and the following: LPI or Decay Heat Removal/RHR pumps.

Proposed Question:

Which of the following correctly describes the cooling flowpath that would be used in a loss of shutdown cooling caused as a result of an inoperable LPSI header downstream of FCV-326?

A. Spent fuel pool cooling pumps take a suction from the discharge of the LPSI pumps and discharge through the spent fuel pool heat exchanger to the suction of a HPSI pump. The HPSI pump discharges flow back to the RCS
B. Charging pumps take a suction from the RCS loop and discharge through the shutdown cooling heat exchanger back to the RCS
C. A containment spray pump takes a suction from the LPSI pump suction and discharges through the shutdown cooling heat exchanger to the HPSI pump suction. The HPSI pump discharge flows back to the RCS
D. A containment spray pump takes a suction on the LPSI pump suction and discharges through the shutdown cooling heat exchanger back to the RCS

Proposed Answer: _C___

Explanation: Answer “C” is correct as that uses the HPSI header for shutdown cooling. Answer “D” is what would be done if the LPSI header was intact, which it is not for this case. A possible means of cooling is by using the refueling pool, not the spent fuel pool, which could be a misconception. Thus answer “A” is plausible but incorrect. Charging pumps, as part of CVCS, do take a suction off the RCS, but don’t discharge through the shutdown cooling heat exchanger, nor is this action proceduralized. A portion of Attachment V does use the charging pumps, with a suction off the SIRWT, to restore RCS level if necessary, prior to using LPSI and HPSI via the charging header, but not with the charging pumps. Therefore, answer “B”, is plausible but incorrect.

Technical Reference(s): __TBD-AOP-19, “Loss of Shutdown Cooling” Rev. 16__
(Attach if not previously provided) ____
(including version/revision number) ____

Proposed references to be provided to applicants during examination: ___None___

Learning Objective: _0717-19 1.02A___ (As available)

Question Source: Bank # _______
Modified Bank # 07-17-19 002 (Note changes or attach parent)
New _______

Question History: Last NRC Exam __1997__

(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: Reworded stem for clarity. Changed distracters to make them more plausible.

Question # _53_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_1___	_____
	Group #	_1___	_____
	K/A #	_CE/E06 EK3.4_	
	Importance Rating	_3.2_	_____

K/A Statement: (CE/E06 Loss of Main feedwater) Knowledge of the reasons for the following responses as they apply to the Loss of Feedwater: RO or SRO function with the control 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.

Proposed Question:

A loss of all feedwater has occurred. The reactor is tripped, standard post trip actions have been taken, and EOP-06, LOSS OF ALL FEEDWATER has been entered.

Per EOP-06, which ONE of the following actions will the ATC take and what is the reason for that action?

A. Trip all RCPs to remove heat input to the RCS and maximize time to regain feedwater
B. Trip all RCPs to prevent damage due to loss of NPSH from once-through cooling initiation
C. Trip 1 RCP in each loop to reduce heat input while maintaining forced circulation
D. Trip 1 RCP in each loop to minimize damage due to loss of NPSH from once-through cooling initiation while maintaining forced circulation

Proposed Answer: ___A__

Explanation: Per TBD-EOP-06, the operator will trip all RCPs to remove that heat input to the RCS and maximize the time to regain feedwater cooling. Answer "C" is plausible if the applicant thinks that forced circulation must be maintained to provide adequate cooling (natural circulation is sufficient). Answers "B" and "D" are incorrect but plausible if the applicant misunderstands the reason for securing the pumps.

Technical Reference(s): __TBD-EOP-06, "Loss of All Feedwater" Rev. 17______
(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	_(10)__
	55.43	_____

Comments:

Question # _54_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_1___	_____
	Group #	_1___	_____
	K/A #	_000015/17 AK1.02_	
	Importance Rating	_3.7_	_____

K/A Statement: (00015/17 RCP Malfunctions) Knowledge of the operational implications of the following concepts as they apply to Reactor Coolant Pump Malfunctions (Loss of RC Flow): Consequences of an RCPS failure

Proposed Question:

For the first five to fifteen minutes after a reactor trip caused by the loss of all four reactor coolants pumps, RCS ΔT is _____1_____. It then _____2_____ as natural circulation is established.

A. Decreasing; remains steady at less than the normal full power ΔT
B. Decreasing; increases to greater than the normal full power ΔT
C. Increasing; remains steady at greater than the normal full power ΔT
D. Increasing; decreases to less than the normal full power ΔT

Proposed Answer: __D___

Explanation: Per Lesson Plan 07-15-16, RCS ΔT increases to a maximum in about 5 to 15 minutes, then decreases to less than the normal full power ΔT. Therefore answer "D" is correct. The other answers are incorrect but plausible if the student has a misconception on how ΔT is affected by a loss of forced circulation and the change in flow transit time for natural circulation.

Technical Reference(s): __Lesson Plan 07-15-16 "Loss of Flow Events", Rev. 5__
(Attach if not previously provided) ______________________________
(including version/revision number) ______________________________

Proposed references to be provided to applicants during examination: ______None_____

Learning Objective: __0715-16 2.04_______ (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 _(8)____
55.43 _____

Comments:

Question # _55_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_1___	_____
	Group #	_1___	_____
	K/A #	_000058 AA1.01_	
	Importance Rating	_3.4*_	_____

K/A Statement: (Loss of DC Power) Ability to operate and/or monitor the following as they apply to the Loss of DC Power: Cross-tie of the affected dc bus with the alternate supply

Proposed Question:

Given the following:

Instrument air pressure is 92 psig
CA-1B is running
Bus Power Failure DC distribution panel 1 light is OFF
Bus Power Failure DC distribution panel 2 light is ON
DC Bus #1 LOW VOLT annunciator is in alarm
CA-1A local selector switch in STANDBY
CA-1A local load transfer switch in position 1
CA-1A control switch is in AFTER-STOP
Both CA-1A and CA-1C have failed to start automatically

Per AOP-16, LOSS OF INSTRUMENT BUS POWER, which ONE of the following actions is required to be able to start CA-1A?

A. Take the local selector switch at the air compressor starter cabinet from STANDBY to CS
B. Take the local load transfer switch at the air compressor starter cabinet to position 2
C. Take the CA-1A control switch at CB-10/11 to START
D. Use the 1B3C-4C Emergency MTS button, PB2/1B3C-4C-MTS, to switch to the emergency DC power source

Proposed Answer: __D___

Explanation: Per AOP-16, the operator is directed to switch over to the emergency DC power source. This will restore control power to CA-1A and enable it to start based on current instrument air pressure. Answer “A” is plausible, in that it would start the air compressor regardless of IA pressure, but only if control power is available. Answer “B” is plausible if the applicant did not know what the load transfer switch does (it can be in either position 1 or position 2 for operating an air compressor). Answer “C” is plausible, because that would start an air compressor, but only with DC control power and only if the local selector switch is in CS, not STANDBY.

Technical Reference(s): _STM Vol. 43 “Service and Instrument Air System” Rev. 10, TBD-AOP-16 “Loss of Instrument Bus Power”, Rev. 17

(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 _(7)__
55.43 _____

Comments:

Question # _56_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_1___	_____
	Group #	_1___	_____
	K/A #	_000077 AK1.03_	
	Importance Rating	_3.3_	_____

K/A Statement: (077 Generator Voltage and Electric Grid Disturbances) Knowledge of the operational implications of the following concepts as they apply to Generator Voltage and Electric Grid Disturbances: Under-excitation

Proposed Question:

With the plant at 100% power and approximately 550 MWe loaded on the main generator a large electrical storm started causing grid disturbances on the grid around the Nebraska area. The load dispatcher calls the control room and requests that the main generator be adjusted for a leading power factor at negative (-) 180MVAR. What is the potential concern with the adjustments necessary to accomplish these parameters?

A. Under-excitation may cause excessive field heating
B. Under-excitation may cause excessive armature coil end heating
C. Over-excitation may cause excessive field heating
D. Over-excitation may cause excessive armature coil end heating

Answer:

Proposed Answer: B

Explanation:
From the STM Volume 24 page 59, rev 23 to adjust to a leading power factor (negative values of MVAR on the left side of the meter's zero value at the 12 noon position, with positive MVAR to the right of the meter's zero value, the generator must be placed in an under-excited condition (or to the left side of zero). The main concern with an under-excited generator is armature coil end heating. See also drawing OI-ST-1, page 14, Rev 18.

Answer A is credible because under-excitation is the correct way to get a generator into a leading power factor condition however the concern with over-excitation is excessive field heating, therefore this is incorrect. Answer C and D are credible because you have to know that leading corresponds to under-excitation not over-excitation.

Technical Reference(s): _STM-24, rev 23, OI-ST-1, rev 18.____________
(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: None

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

10 CFR Part 55 Content: 55.41 _4___
55.43 _____

Comments:

Question # _57_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_1___	_____
	Group #	_2___	_____
	K/A #	_000005 2.4.11__	
	Importance Rating	_4.0_	_____

K/A Statement: (00005 Inoperable/Stuck Control Rod) +Knowledge of abnormal condition procedures associated with Inoperable/Stuck Control Rod.

Proposed Question:

CEA 4-40 is currently at 105 inches. The other CEAs in group 4 are at 120 inches. This CEA was not recovered within one hour and has been declared inoperable. Actions are continuing in AOP-02 CEA AND CONTROL SYSTEM MALFUNCTIONS, Section III "Misaligned Group 4 CEA".

Current status is as follows:

Reactor power has been lowered to less than 70% ΔT power
Rod control mode selector switch is in manual individual
Control rod group selector switch has been selected to group 4

Per AOP-02, which ONE of the following actions will be performed?

A. Borate to maintain power level, select rod 40 on the rod selector switch, bypass the rod block, and slowly withdraw the CEA
B. Select rod 40 on the rod selector switch, bypass the rod block, slowly withdraw the CEA, and increase turbine load as necessary to maintain T_c stable
C. Dilute to maintain power level, select rod 1 on the rod selector switch, bypass the rod block, and slowly insert CEA to align to rod 40. Repeat for CEA's 4-38, 4-39, and 4-41
D. Select rod 1 on the rod selector switch, bypass the rod block, slowly insert CEA to align to rod 40, and reduce turbine load as necessary to maintain T_c stable. Repeat for CEA's 4-38, 4-39, and 4-41

Proposed Answer: __D__

Explanation: Per AOP-02, a group 4 control rod that is misaligned more than 12 inches, but less than 18 inches, when the CEA has been declared inoperable, Step 17 of Section III, states to attempt to align the remainder of the CEAs in the group to the misaligned CEA. Specifically, the steps state to select CEA to be moved, bypass rod block, slowly insert the CEA, reduce turbine load to maintain Tcold stable, and repeat these steps until all Group 4 CEAs are within 12 inches of misaligned CEA. Therefore, answer "D" is correct. Answer "A" is incorrect but plausible because this is what the procedure directs if the misalignment was greater than 18 inches, which it is not in this case. Answers "B" and "C" are incorrect but plausible if the candidate mixed up the two cases.

Technical Reference(s): _TBD-AOP-02 “CEA and Control System Malfunctions” Rev. 7, TDB-VI “COLR”, Rev. 39 (Cycle 26), TDB-I.A.3 “CEDM Locations” 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______

(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 __(10)_
55.43 _____

Comments:

Question # _58_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	__1__	_____
	Group #	__2__	_____
	K/A #	_000067 AK3.04__	
	Importance Rating	_3.3_	_____

K/A Statement: (000067 Plant Fire on-site) Knowledge of the reasons for the following responses as they apply to the Plant Fire on Site: Actions contained in EOP for plant fire on site

Proposed Question:

For a fire located in the BAST area (Corridor 26 West), AOP-06-01, FIRE EMERGENCY-AUXILIARY BUILDING RADIATION CONTROLLED AREAS AND CONTAINMENT, directs operators to isolate letdown. Which ONE of the following is the reason for isolating letdown for a fire located in this area?

(Note: TCV-202 is the Letdown Temperature Control Valve, HCV-204 is the Letdown Heat Exchanger Inlet Isolation valve, and HCV-206 is the Outboard Reactor Coolant Pump Controlled Bleed-off Isolation Valve).

A. A loss of VCT level control
B. Potential spurious operation of TCV-202
C. Potential spurious operation of HCV-204 and HCV-206
D. Minimize radiation levels for firefighting efforts

Proposed Answer: __C__

Explanation: Per AOP-06-01, Section IX, letdown is isolated due to the potential for spurious operation of HCV-204 and HCV-206. Answers “A” and “D” are plausible because they are reasons given in the AOP for isolating letdown if the fire occurs in a different area. Answer “B” is a reason given in the AOP for the possible loss of letdown if the fire occurs in Room 71 (Aux. Bldg. 1025’ Work Area), but no direction is given to isolate letdown for a fire occurring there.

Technical Reference(s): _TBD-AOP-06-01 “Fire Emergency – Auxiliary Building Radiation Controlled Areas and Containment” Rev. 1__

(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 (10)___
55.43 ______

Comments:

Question # _59_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_1___	_____
	Group #	_2___	_____
	K/A #	_CE/A 16 AA1.1__	
	Importance Rating	_3.4_	_____

K/A Statement: (CE/A 16 Excess RCS Leakage) Ability to operate and/or monitor the following as they apply to the Excess RCS Leakage: Components and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual failures

Proposed Question:

A small RCS leak has been identified and the following plant parameters are observed:

Pressurizer level and pressure are steady
Containment pressure, dew point and sump level are steady
VCT level and pressure are stable
Pressurizer quench tank level, pressure, and temperature are increasing
CCW surge tank level and pressure are steady
Aux building sump levels are steady
Safety injection tank levels and pressures are steady
All secondary radiation monitors read background

Which ONE of the following actions may result in isolating the leak?

A. Closing SIT loop injection valves
B. Isolating charging and letdown
C. Closing RCS sample lines
D. Isolating CCW to the reactor coolant pumps

Proposed Answer: _B___

Explanation: Letdown relief valve CH-223 relieves to the quench tank, so is a possible source of the leakage, making answer “B” correct. “A” is incorrect as the SIT levels and pressures are steady. “C” is incorrect because quench tank levels are increasing, and the sample line doesn’t drain to the quench tank. “D” is incorrect because CCW surge tank level and pressure are steady.

Technical Reference(s): _TDB-AOP-22, STM 12__
(Attach if not previously provided) ______________________________
(including version/revision number) ______________________________

Proposed references to be provided to applicants during examination: ____None________

Learning Objective: ___0717-33 01.02_ (As available)

Question Source: Bank # _07-17-33 003
Modified Bank # _______ (Note changes or attach parent)
New _______

Question History: Last NRC Exam _2007___
(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:

Question # _60_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_1___	_____
	Group #	_2___	_____
	K/A #	_000037 AK3.10_	
	Importance Rating	_3.3*_	_____

K/A Statement: (000037 Steam gen tube leak) Knowledge of the reasons for the following responses as they apply to the Steam Generator Tube Leak: Automatic actions associated with high radioactivity in S/G sample lines.

Proposed Question:

The reactor is currently at 100% power when a high radiation alarm occurs on RM-054A, which is in its normal lineup. This automatically isolates _____(1)_____ in order to _____(2)_____.

A. HCV-1387A and HCV-1387B; isolate steam generator blowdown from steam generator RC-2A ONLY
B. HCV-1387A and HCV-1387B; isolate steam generator blowdown from BOTH steam generators
C. HCV-1387A and HCV-1388A; isolate steam generator blowdown from steam generator RC-2A ONLY
D. HCV-1387A and HCV-1388A; isolate steam generator blowdown from BOTH steam generators

Proposed Answer: __D___

Explanation: Per STM Vol. 33, a high alarm on RM-054A closes the inside containment blowdown isolation valves HCV-1387A and HCV-1388A to isolate both steam generators. This makes answer “D” correct. Answer “A” is plausible because RM-054A normally monitors steam generator RC-2A (although it has the capability to be cross-connected to RC-2B). Therefore, the applicant might suppose that it would only isolate that generator, which would be by closing either HCV-1387A or HCV-1387B. The other two are plausible if the applicant does not recall which blowdown isolation valves isolate which steam generator (e.g. answer “C” if the applicant believes RC-2A should just be isolated, but does not recall that 1388A isolates RC-2B; answer “B” if applicant knows both steam generators are isolated but does not recall that HCV-1387B isolates RC-2A).

Technical Reference(s): _ STM Vol. 33 “Radiation Monitoring System” Rev. 17__
(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:

Question # _61_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_1___	_____
	Group #	_2___	_____
	K/A #	_000051 2.4.50_	
	Importance Rating	_4.2__	_____

K/A Statement: (000051 Loss of Condenser vacuum) Ability to verify system alarm setpoints and operate controls identified in the alarm response manual associated with Loss of Condenser Vacuum.

Proposed Question:

Given the following conditions:

Reactor power is 50%
Condenser pressure indicator, PI-975A, reads 23.5"Hg
Condenser pressure indicator, PI-975B, reads 24.5"Hg
All condenser evacuation pumps are operating

Which CB-10/11 Panel A9 annunciator alarm(s) should be lit and what action is required by AOP-26, TURBINE MALFUNCTIONS, for the above conditions?

A. A-4U "Exhaust A Pressure HI" ONLY; commence a reactor shutdown to restore vacuum
B. BOTH A-4U "Exhaust A Pressure HI" AND A-4L "Exhaust B Pressure HI"; commence a reactor shutdown to restore vacuum
C. A-4U "Exhaust A Pressure HI" ONLY; trip the reactor and enter EOP-00
D. BOTH A-4U "Exhaust A Pressure HI" AND A-4L "Exhaust B Pressure HI"; trip the reactor and enter EOP-00

Proposed Answer: __A___

Explanation: Per ARP-CB-10,11/A9, the setpoint for the alarm is <23.85"Hg. Therefore, A-4U should be lit, but A-4L should NOT be lit. Actions are taken once vacuum reaches 25"Hg, so if the student has a misconception, they might believe that 25"Hg is when the alarm will annunciate. If condenser vacuum is <23.85"Hg (which it is here), AND generator load is <150MW (approximately 30% power), then the reactor must be tripped and EOP-00 entered. However, in this case the generator is operating at 50% power, and the actions directed by AOP-26 are to commence a reactor shutdown to restore vacuum. Thus "A" is correct. Answers "B", "C", and "D" are incorrect yet plausible based on the above discussion.

NOTE: Check the values for the upper and lower condenser pressures to ensure that they are operationally valid with this spread of 1 dp.

Technical Reference(s): _ARP-CB-10,11/A9 "Annunciator Response Procedure A9 Control Room Annunciator A9" Rev. 29, STM Vol. 20

"Feedwater and Condensate System" Rev. 46, TBD-AOP-26, "Turbine Malfunctions" 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______
(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 _(10)__
55.43 _____

Comments:

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_

(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 # 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.

Question # _63_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	__1__	_____
	Group #	__2__	_____
	K/A #	_000068 AA2.06_	
	Importance Rating	_4.1_	_____

K/A Statement: (000068 Control Room Evac) Knowledge of the operational implications of the following concepts as they apply to the Control Room Evacuation: RCS pressure

Proposed Question:

The control room has been evacuated. Steps are complete per AOP-07, EVACUATION OF CONTROL ROOM, to establish control at the alternate shutdown panel and the auxiliary feedwater panel. You have been directed to maintain RCS pressure between 2050 and 2150 psia per AOP-07, Step 8.

How is RCS pressure control accomplished in this case?

A. Pressurizer backup heater Bank 1 is controlled remotely from the alternate shutdown panel
B. Pressurizer backup heater Bank 1 is controlled locally at the motor control center
C. Pressurizer backup heater Bank 4 is controlled remotely from the alternate shutdown panel
D. Pressurizer backup heater Bank 4 is controlled locally at the motor control center

Proposed Answer: __D__

Explanation: Per Step 8 of AOP-07, RCS pressure is maintained by controlling pressurizer backup heater Bank 4 locally at the motor control center. Per the STM, there's a handswitch on this motor control center to operate Bank 4. Therefore, answer "D" is correct. The other answers are plausible if the student does not remember which bank of backup heaters is used or that local operation is required. The STM does state that the remote/local switch on the alternate shutdown panel must be switched from remote to local to be able to use the local heater handswitch, however, the stem of the question states that control has been established at the alternate shutdown panel, which indicates that this switch has been placed in local. Therefore, remote control from the alternate shutdown panel is plausible, but incorrect.

Technical Reference(s): __TBD-AOP-07 "Evacuation of Control Room" Rev. 13, STM Vol. 37 "Reactor Coolant System" Rev. 42___

(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	_(5)__
	55.43	_____

Comments:

Question # _64_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_1___	_____
	Group #	_2___	_____
	K/A #	_000036 AK1.01_	
	Importance Rating	_3.5_	_____

K/A Statement: (000036 Fuel handling Accident) Knowledge of the operational implications of the following concepts as they apply to Fuel Handling Incidents: Radiation exposure hazards

Proposed Question:

Given the following initial conditions:

Refueling outage is in progress
The reactor is being defueled
The 105th fuel assembly has been removed from the upender and is being transferred to its storage location in the SFP
Stack radiation monitor RM-052 has power with pumps energized
One auxiliary building supply fan is in service
One auxiliary building exhaust fan is in service

Plant conditions then change:

Moving fuel assembly drops from FH-12
Portable radiation monitor near FH-12 pegs high
Area monitors alarm

Which of the following additional actions MUST be taken to meet requirements of AOP-08, FUEL HANDLING INCIDENT?

1. Start second auxiliary building supply fan
2. Start second auxiliary building exhaust fan
3. Ensure charcoal filter VA-66 in filtered mode

A. 3 ONLY
B. 1 and 3 ONLY
C. 2 and 3 ONLY
D. 1, 2, and 3

Proposed Answer: __C___

Explanation: Per AOP-08, one supply fan and two exhaust fans need to be running to ensure negative pressure (prevent any radiation from being released from the auxiliary building unmonitored). Given the initial conditions, one more exhaust fan must be started. Only one supply fan needs to be running and already is, so another one does not need to be started. The charcoal filter must also be placed in service. Therefore, answer “C” is correct. Answer “A” is incorrect but plausible if the student does not remember the need for the second exhaust fan. Answer “B” is incorrect but plausible if the student gets the two fans mixed up. Answer “D” is incorrect but plausible if the student thinks that two of each fan is required.

Technical Reference(s): __STM Vol. 40 “Refueling System” Rev. 20, TBD-AOP-08 “Fuel Handling Incident” Rev. 9_

(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 __(8)__
55.43 _____

Comments:

Question # _65_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	__1__	_____
	Group #	__2__	_____
	K/A #	_CE/A 11 AA1.2_	
	Importance Rating	_3.2__	_____

K/A Statement: (CE/A11 RCS Overcooling) Ability to operate and/or monitor the following as they apply to the RCS Overcooling: Operating behavior characteristics of the facility.

Proposed Question:

Which ONE of the following initial conditions will result in the greatest reactivity addition for a steam line break upstream of the MSIV?

A. Beginning of cycle, zero power
B. Beginning of cycle, full power
C. End of cycle, zero power
D. End of cycle, full power

Proposed Answer: ___C__

Explanation: At the end of cycle, the moderator temperature coefficient is more negative due to the decrease of boron concentration. Since a steam line break results in overcooling of the RCS, this will result in a greater reactivity addition. At zero power, the steam generator has more inventory than at full power, therefore there is more water in the steam generator that can steam and cool down the RCS, causing a greater cooldown and thus a greater reactivity addition than at full power. Therefore answer "C" is correct. Answer "A" is plausible if the student's logic is incorrect for BOC vs. EOC and the cooldown effects. Answer "D" is plausible if the student thinks that there is more of a cooldown from full power than zero power. Answer "B" is plausible for the same reasons as "A" and "D".

Technical Reference(s): __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 B1.5 Explain how initial power level and time in core cycle affect the reactivity added by 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	__(7)__
	55.43	_____

Comments:

Question # _66_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_3__	_____
	Group #	___	_____
	K/A #	2.3.13__	
	Importance Rating	_3.4_	_____

K/A Statement: Knowledge of radiological safety procedures pertaining to license operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Proposed Question:

A room contains a radiation source with a 2R/hr dose rate 30 cm from this source.

Which ONE of the following meets the requirements of SO-G-101,"Standing Order – Radiation Worker Practices" for entry into this area?

A. Individuals must wear a radiation monitoring device which continuously indicates the radiation dose rate in the area
B. Individuals must be accompanied by an individual qualified in radiation protection procedures equipped with a dose rate instrument
C. A Radiation Protection Technician must provide continuous coverage and be equipped with a dose rate instrument
D. Entry into this area is prohibited without a sound operational or safety reason

Proposed Answer: ___C__

Explanation: Based on the stem of the question, this meets the requirements of a restricted high radiation area (RHRA). Therefore, per SO-G-101, answer "C" is correct. Answers "A" and "B" are requirements for a high radiation area (HRA) entry, but are not sufficient for entry into an RHRA. Answer "D" would be correct if it was a very high radiation area (VHRA), but this area does not meet that definition and is therefore incorrect.

Technical Reference(s): __SO-G-101 "Standing Order – Radiation Worker Practices" Rev. 34_____

(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 _(12)__
55.43 _____

Comments:

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 # ______

New __X____

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

Question # _68_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	__3__	_____
	Group #	_____	_____
	K/A #	_2.1.15_	
	Importance Rating	_2.7_	_____

K/A Statement: Knowledge of the administrative requirements for temporary management directives, such as standing orders, night orders, Operations memos, etc.

Proposed Question:

Assuming that no alternates have been designated, which ONE of the following individuals has the authority to issue operations memorandums per SO-O-13 OPERATIONS MEMORANDUMS?

A. Shift Manager – Operations Standards
B. Supervisor – Operations Control Center
C. Manager – Shift Operations
D. Manager – Fort Calhoun Station

Proposed Answer: __D___

Explanation: Per SO-O-13, either the Manager-Fort Calhoun Station or Manager-Operations issues Operation Memoranda, or their designee. In the stem, it states that no alternates have been designated, so one of these two must issue the memos. Therefore answer "D" is correct. The shift manager-operations standards is specifically referenced in the procedure and receives a copy of the memos, but has no authority, making answer "A" plausible but incorrect. The supervisor-OCC has responsibility for Night Notes, not operations memos, making answer "B" plausible but incorrect. The Manager-Operations does have authority, but this is the Manager-Shift Operations, and although specifically referenced in the procedure, receives a copy of the memo but has no authority for issuance, making answer "C" plausible but incorrect.

Technical Reference(s): _SO-O-13 "Operations Memorandums" Rev. 19, SO-O-1, "Conduct of Operations" Rev. 84 (pg. 55)_____

(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	_(10)__
	55.43	_____

Comments:

Question # _69_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_3___	_____
	Group #	_____	_____
	K/A #	_2.2.13_	
	Importance Rating	_4.1_	_____

K/A Statement: Knowledge of Tagging and clearance procedures

Proposed Question:

AC-3A has been tagged out for repairs. A Temporary Lift was done to allow testing. Following the testing, it was determined that additional repairs were required.

Identify the proper sequence for removing the Temporary Lift.

A. Remove the Temporary Lift tags, independently verify equipment positions, and rehang the Danger tags
B. Remove the Temporary Lift tags and rehang the Danger tags
C. Remove the Temporary Lift tags, reposition equipment as necessary, rehang Danger tags, and independently verify equipment positions
D. Reposition equipment as necessary, rehang Danger tags, independently verify equipment positions, and remove the Temporary Lift tags

Proposed Answer: __C___

Explanation:

Technical Reference(s): SO-G-20A
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: ___None_________

Learning Objective: ADM-CONTROL 01.00

Question Source: Bank # ADM-CONTROL 003
Modified Bank # _______ (Note changes or attach parent)
New _______

Question History: Last NRC Exam 2005 RO 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	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41 (10)	
	55.43	

Comments:

Question # _70_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	__3__	_____
	Group #	_____	_____
	K/A #	_2.4.42________	
	Importance Rating	_2.6__	_____

K/A Statement: Knowledge of emergency response facilities

Proposed Question:

When the TSC ventilation is switched to the EMERGENCY Mode following TSC activation, which ONE of the following is the expected damper response for exhaust damper VA-108A and emergency fresh air damper VA-107H?

A. VA-108A closes to ensure acceptable airborne radioactivity levels in the TSC; VA-107H closes to ensure acceptable airborne radioactivity levels in the TSC
B. VA-108A closes to ensure acceptable airborne radioactivity levels in the TSC; VA-107H is controlled to maintain positive pressure in the TSC
C. VA-108A is controlled to maintain system duct static pressure; VA-107H is controlled to ensure acceptable airborne radioactivity levels in the TSC
D. VA-108A is controlled to maintain system duct static pressure; VA-107H is controlled to maintain positive pressure in the TSC

Proposed Answer: __B___

Explanation: Per Lesson Plan 4-23-23 slide 96, when the TSC ventilation is put into the EMERGENCY mode, damper 107H is controlled to maintain TSC at 1/8 inch pressure. Damper 108A closes to ensure acceptable airborne radioactivity levels in the TSC. Therefore, answer "B" is correct. VA-108A can be used during normal operation to maintain system duct pressure. Also, VA-107H is closed when outside temperatures are less than 60 degrees or if there is a toxic gas event and ventilation is in TOXIC GAS mode. Therefore answers "A", "C", and "D" are plausible but incorrect.

Technical Reference(s): _Lesson Plan 4-23-23 slide 96_____
(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	(10)
	55.43	______

Comments:

Question # _71_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	__3__	_____
	Group #	_____	_____
	K/A #	_2.3.11____	
	Importance Rating	_3.8_	_____

K/A Statement: Ability to control radiation releases

Proposed Question:

A normal release of Monitor Tank WD-22A is to be performed. Using Form FC-211, "Waste Liquid Tank Release Permit", the administrative release rate is:

A. 60 percent of the maximum release rate
B. 70 percent of the maximum release rate
C. 80 percent of the maximum release rate
D. 90 percent of the maximum release rate

Proposed Answer: __D__

Explanation: Per OI-WDL-3, the administrative release rate is 90% of the max release rate. Therefore answer "D" is correct. Answers B, C, and D are plausible if the student can't remember the value.

Technical Reference(s): __OI-WDL-3 "Liquid Waste Disposal Release" Rev. 21__
(Attach if not previously provided) ______________________________
(including version/revision number) ______________________________

Proposed references to be provided to applicants during examination: ______None_______

Learning Objective: 0711-32 01.02A________________ (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 _(13)__
55.43 _____

Comments:

Question # _72_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_3__	_____
	Group #	_____	_____
	K/A #	_2.1.4________	
	Importance Rating	__3.3_	_____

K/A Statement: Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, “no-solo” operation, maintenance of active license status, 10CFR55 etc.

Proposed Question:

Per OPD-3-11, LICENSE ACTIVATION AND WATCHSTATION MAINTENANCE, what are the MINIMUM watch requirements to maintain an active RO license?

A. Five 8-hr shifts in a calendar quarter
B. Five 8-hr shifts in ANY three month span
C. Five 12-hr shifts in a calendar quarter
D. Five 12-hr shifts in ANY three month span

Proposed Answer: __C___

Explanation: Per 10 CFR 55.53(e), either five 12-hour or seven 8-hour watches must be performed every calendar quarter (i.e. January through March). Therefore answers A, B, and D are incorrect but plausible if the applicant forgets the requirement. Answer C is correct.

Technical Reference(s): _10 CFR 55.53(f), OPD-3-11 “License Activation and Watchstander Maintenance” Rev. 16___________

(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 __(10)__
55.43 ______

Comments:

Question # _73_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	__3__	_____
	Group #	_____	_____
	K/A #	__2.4.13	__________
	Importance Rating	__4.0_	_____

K/A Statement: Knowledge of crew roles and responsibilities during EOP usage.

Proposed Question:

Per OPD-3-03, CONTROL ROOM OPERATOR POSITIONING DURING EMERGENCY/ABNORMAL PLANT CONDITIONS, during an emergency the ATCO normally transitions to a position that is responsible for manipulating all equipment controls located on which of the following panels?

A. CB-1/2/3, CB-4, AI-65A/B ONLY
B. CB-1/2/3, CB-4, AI-65A/B, AI-66A/B ONLY
C. CB-1/2/3, CB-4, AI-65A/B, AI-33A/B/C ONLY
D. CB-1/2/3, CB-4, AI-65A/B, AI-66A/B, AI-33A/B/C ONLY

Proposed Answer: __C___

Explanation: Per OPD-3-03, Step 4.1, the ATCO normally transitions to the primary LO position during an emergency. Per 4.1.1, the primary LO has responsibility for equipment manipulation for controls located on CB-1/2/3, CB-4, AI-65A/B, and AI-33A/B/C. Therefore answer “C” is correct. Answer “A” is plausible if the candidate forgets AI-33A/B/C would be one of the panels, and answers “B” and “D” are plausible if the candidate believes that they would have responsibility for panel AI-66A/B (which is the responsibility of the secondary LO position).

Technical Reference(s): _OPD-3-03 “Control Room Operator Positioning During Emergency/Abnormal Plant Conditions” Rev. 5, SO-O-1 “Conduct of Operations”, Rev. 84__

(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	_10___
	55.43	_____

Comments:

Question # _74_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_3___	_____
	Group #	_____	_____
	K/A #	_2.1.3_______	
	Importance Rating	_3.7_	_____

K/A Statement: Knowledge of shift or short-term relief turnover practices.

Proposed Question:

As part of the shift turnover, the Secondary Plant RO is required to perform a detailed walkdown of ________________.

A. CB-4, CB-10/11, AI-66A, and AI-66B
B. CB-10/11, AI-65A, and AI-65B
C. CB-4, CB-10/11, AI-66A, and AI-66B
D. CB-10/11, AI-66A, and AI-66B

Proposed Answer: __D___

Explanation:
Modified Bank question used on 2005 NRC exam (attached).

Technical Reference(s): ___SO-O-1 “Conduct of Operations” Rev. 84__________
(Attach if not previously provided) ______________________________
(including version/revision number) ______________________________

Proposed references to be provided to applicants during examination: ____None________

Learning Objective: 0762-01 01.00 ____________ (As available)

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

Question History: Last NRC Exam __2005 RO___

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

10 CFR Part 55 Content: 55.41 __10_
55.43 _____

Comments:

Bank question # 07-62-01 077
07-62-01 077

As a part of the shift turnover, the Secondary Plant RO is required to perform a detailed walkdown of ______________________.

A. CB-10/11, CB-20, AI-65A and AI-65B.
B. CB-10/11, CB-20, AI-66A and AI-66B.
C. CB-1,2,3, CB-4, CB-10/11 and CB-20.
D. CB-20, AI-65A, AI-65B, AI-66A and AI-66B.

Answer: B

Question 67 K/A # 000000 2.1.03
Knowledge of shift turnover practices.
RO Importance 3.0 SRO Importance 3.4 10 CFR 55 Section 41.10 / 45.13
FCS Lesson Plan / Objective 0762-01 01.00
STATE the major sections of the Standing Orders.
KA#: 000000 2.1.03 Bank Reference #:
LP# / Objective: 0762-01 01.00 Exam Level: RO
Cognitive Level: LOW Source: NRC 2005 EXAM
Reference: SO-O-1 Handout: NONE

Question # _75_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	__3__	_____
	Group #	_____	_____
	K/A #	_2.4.37________	
	Importance Rating	_3.0_	_____

K/A Statement: Knowledge of the lines of authority during implementation of the emergency plan.

Proposed Question:

The ATCO determines that an immediate action must be taken to protect the health and safety of the public during an emergency event. This action is contrary to plant procedures, and there are no immediately apparent actions that can be taken in accordance with procedures to adequately protect the public. Which ONE of the following describes the appropriate response to this situation?

A. This action cannot be taken without an approved procedure. Implement a rapid procedure change to allow this action to be taken
B. This action can be taken by the ATCO based on their own judgment with no additional actions required
C. This action can be taken by the ATCO provided they obtain permission from at least a licensed SRO prior to taking the action
D. This action can be taken by the ATCO provided they notify the shift manager as soon as possible after taking the action

Proposed Answer: __C___

Explanation: Per 10 CFR 50.54(x), "A licensee may take reasonable action that departs from a license condition or a technical specification in an emergency when this action is immediately needed to protect the public health and safety and no action consistent with license conditions and technical specifications that can provide adequate or equivalent protection is immediately apparent." 10 CFR 50.54(y) states, in part, "Licensee action permitted by paragraph (x) of this section shall be approved, as a minimum, by a licensed senior operator." Therefore, answer "C" is correct. Answer "A" is plausible if the candidate does not remember the allowance of 10CFR 50.54(x). Answer "B" is plausible if the candidate does not remember the requirement of 10 CFR 50.54(y), that it requires SRO approval. Answer "D" is plausible if the candidate does not remember that SRO approval must come PRIOR to the action being taken.

Technical Reference(s): _Title 10 CFR Part 50.54(x) and (y)________________
(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	__(10)_
	55.43	_____

Comments:

Question # _76___

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	__1___
	Group #	_____	__1___
	K/A #		0009.EA2.36_
	Importance Rating	_____	__4.6__

K/A Statement: **Ability to determine and interpret the following as they apply to a small break LOCA:** EA2.36 Difference between overcooling and LOCA indications

Proposed Question:

The reactor tripped fifteen minutes ago. Standard post trip actions of EOP-00 are complete and you are now diagnosing the event to determine what emergency operating procedure to implement.

The current plant conditions are:

Pressurizer pressure: 995 psia and decreasing slowly
Pressurizer level: 0%
RVLMS Level: 100%
RC-2A pressure: 900 psia and steady
RC-2B pressure: 895 psia and steady
Containment pressure: 4.5 psig and slowly decreasing
RM/054 A, B and RM/057: No high alarm
RM/091 A, B: No alarms
Safety/PORV tailpipe temperatures: 200°F
RCP's have been tripped
Tcold: 533°F and slowly decreasing
Thot: 544°F and slowly decreasing
RC-2A level: 28% NR and slowly increasing
RC-2B level: 29% NR and slowly increasing
HPSI pumps are injecting flow per Attachment 3

Based on the above conditions which EOP should be implemented?

A) EOP-03, Loss of Coolant Event
B) EOP-04, Steam Generator Tube Rupture Event
C) EOP-05, Uncontrolled Heat Extraction Event
D) EOP-20, Functional Recovery Procedure

Proposed Answer: __A___

Explanation: Per the diagnosis chart in EOP-00, everything is answered yes, except for the RCP's, until the second page. On the second page, it asks if pressurizer pressure is >1800 psia, which it is not. It then has items for determining if it's an UHE, SGTR, or LOCA. You do have a subcooled margin >20 degrees, but no low steam generator pressure, so the answer is no. The steam generators are essentially at the same level, so no tube rupture. So, this page has you consider a LOCA. Because there's just the one event, you would enter EOP-03, LOCA. Answer "A" is therefore correct. SGTR indications are of secondary plant activity (not provided here) and a difference in steam generator level. The two levels are different, but very close and within the margin for error. Therefore, this answer, although plausible, is incorrect. An uncontrolled heat extraction has high subcooling margin and low steam generator pressure, which is not the case here, although containment pressure is visibly high, and there are no radiation alarms in containment (due to the small size of the RCS break). Therefore this answer, although incorrect, is plausible if the candidate does not use the guidance from the diagnostic flow chart and instead looks at containment pressure and radiation levels. The functional recovery procedure might be entered a few different ways. The candidate might diagnose more than one event (i.e. UHE and LOCA or SGTR), for the above reasons. The candidate might also be unable to diagnose an event and determine that the functional recovery must be entered. However, by following the flow chart, LOCA should be diagnosed.

Note: Jerry ran this with his simulation program for a LOCA beyond the capacity of the charging pumps but too small to remove decay heat. At 15 minutes after the trip, Pressurizer Level was 0% but RVLMS was still 100%. At this point leak out the break exceeded the injection rate so inventory was decreasing. The operators would have tripped the RCPs. Natural circulation is being established. The hot leg is saturated. Containment radiation levels will be increased but not enough to cause RM-091A/B to alert (40 R/hr) or high alarm (6500 R/hr).

Technical Reference(s): _TBD-EOP-00, "Standard Post Trip Actions", Rev. 27, and Lesson Plan 0715-23, Loss of Coolant Accidents, Rev. 10

(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 _(5)__

Comments:

Might be able to change the distracters if needed by adding the LOOP/LOFC procedure in place of one of the other distracters due to the RCP's being tripped.

Question # _77___

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	__1___
	Group #	_____	__1___
	K/A #		0022_G2.4.21_
	Importance Rating	_____	__4.6__

K/A Statement: (000022 Loss of Rx Coolant Makeup) 2.4.21 Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.

Proposed Question:

The following plant conditions exist:

Plant is in Mode 3.
Trip from 100% power occurred five (5) minutes ago.
CEAs 40 and 41 are stuck fully withdrawn.
Charging pump trips on a loss of suction pressure.
Operators are unable to start the other charging pumps
Current reactor power is greater than 10^{-5} %.

Which one (1) of the following operator actions is procedurally required under these conditions?

A. Lower RCS pressure and emergency borate via HPSI.
B. Emergency borate via gravity feed suction path.
C. Lower RCS temperature and initiate SIAS.
D. Use auxiliary spray and emergency borate.

Proposed Answer: __A___

Explanation: Per RC-3 of EOP-20 where reactivity control safety function is not met from RC-1 or RC-2 because no boration with charging pumps can be performed, answer A is correct. Distractor B, is plausible gravity feed is an option in E-0 but won't work with no charging pumps, while distractor C and D are plausible steps in EOP-20 for other issues but not to satisfy reactivity control safety function RC-3.

Technical Reference(s): _________EOP-20, rev 24, Reactivity control safety function, page 12-14, and page 51.________

(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 Cognitive Level:	Memory or Fundamental Knowledge	______	
	Comprehension or Analysis	__X__	
10 CFR Part 55 Content:	55.41 ______		
	55.43 _(5)__		

Comments:

Question # _78___

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	__1___
	Group #	_____	__1___
	K/A #		0038_EA2.14_
	Importance Rating	_____	__4.6__

K/A Statement: (000038 Steam Generator Tube Rupture) **Ability to determine and interpret the following as they apply to a SGTR:** Magnitude of atmospheric radioactive release if cool down must be completed using steam dumps or if atmospheric reliefs lift.

Proposed Question:

In USAR Section 14.14 for a steam generator tube rupture event, the worst case release of radioactivity if cool down must be completed using atmospheric reliefs is

A. Direct flow that flashes from the hot side break of the faulted generator for the two hour dose at the exclusion area boundary (EA-B dose)
B. Direct flow that flashes from the cold side break of the faulted generator for the two hour dose at the exclusion area boundary (EA-B dose)
C. Direct flow that flashes from the hot side break of the faulted generator for the duration of event dose at the low population zone (LPZ dose)
D. Direct flow that flashes from the cold side break of the faulted generator for the duration of event dose at the low population zone (LPZ dose)

Proposed Answer: ___A___

Per pages 14.14 (page 7) hot side break is worst for radiological consequences and per table on page 9 the EA-B dose is the worst dose for the event (also inferred in RG 1.183).

Technical Reference(s): ___USAR 14.14___
(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 Cognitive Level:	Memory or Fundamental Knowledge	______
	Comprehension or Analysis	__X__
10 CFR Part 55 Content:	55.41 ______	
	55.43 __4__	

Comments:

Question # _79___

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	__1___
	Group #	_____	__1___
	K/A #		0040_EA2.1_
	Importance Rating	_____	__4.0__

K/A Statement: **Ability to determine or interpret the following as they apply to the (Excess Steam Demand):** EA2.1 Facility conditions and selection of appropriate procedures during abnormal and emergency operations

Proposed Question:

The reactor has just tripped and you are attempting to diagnose the event to determine which optimal recovery or function recovery procedure to implement. Plant conditions are:

Buses 1A3 and 1A4 are energized
Buses 1A1 and 1A2 are deenergized
T-cold: 492°F and decreasing
Pressurizer pressure 1620 psia and decreasing
Pressurizer level is 18% and decreasing
Containment pressure is 17 psig and increasing
No radiation monitors are in alarm
RC-2A level is 25% NR and decreasing
RC-2B level is 50% WR and decreasing
RC-2A pressure is 620 psia and decreasing
RC-2B pressure is 570 psia and decreasing

Which procedure should be implemented and what event is in progress?

A) Implement EOP-03 for a loss of coolant accident inside containment
B) Implement EOP-05 for an uncontrolled heat extraction event inside containment
C) Implement EOP-07 for a station blackout event
D) Implement EOP-20 for multiple events occurring

Proposed Answer: ___B__

Explanation: Based on the diagnosis flow chart with the note, a LOOP plus another event is not considered multiple events, therefore EOP-20 does not need to be entered. Due to the low T-cold, the low steam generator pressures, and the low steam generator level on RC-2B, it's an uncontrolled heat extraction event on RC-2B, therefore answer "B" is correct. The others are plausible based on misdiagnosis (due to the decreasing pressurizer level and pressure for a LOCA, and if they misdiagnose a station blackout instead of a LOOP for answer "C"). Answer "D" is plausible if the individual does not know the note about the LOOP.

Technical Reference(s): ___
(Attach if not previously provided) _TBD-EOP-00, "Standard Post Trip Actions", Rev. 27___

(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 _____

Comments:

Question # _80___

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	__1___
	Group #	_____	__1___
	K/A #		056_G2.2.22_
	Importance Rating	_____	__4.7__

K/A Statement: (000056 Loss of Off-site Power) 2.2.22 Knowledge of limiting conditions for operations and safety limits.

Proposed Question:

The control room is raising power from 50% toward 100%. An electrical disturbance causes the 161 KV line voltage to drop to 161.0 KV. The 345 KV grid voltage is unaffected. What actions should the control room take?

A. Continue the reactor power increase.
B. Transfer Bus 1A3 and 1A4 to the Unit Aux Transformer, then continue the power increase.
C. Stop the load increase and hold power steady until the 161 KV voltage is restored.
D. Restore 161 KV line to operable within 72 hours or be in Mode 3 within an additional 12 hours.

Proposed Answer: ___D___

Explanation: Per AOP-31, if the 161 KV line voltage falls below 161.3 kV, the line is considered inoperable. The inoperability of the 161 KV incoming line renders both transformer T1A-3 and T1A-4 inoperable per Tech Spec 2.7(2)c. This requires restoring the 161 KV line to operable within 72 hours or be in Mode 3 within an additional 12 hours. The other answers are plausible because they are reasonable answers depending on the candidate's memory of the technical specifications.

Technical Reference(s): _______Tech Spec 2.7(2)c. AOP-31, "161 kV Grid Malfunctions" Rev.10 ________________
(Attach if not previously provided) ____
(including version/revision number) ____

Proposed references to be provided to applicants during examination: _________________

Learning Objective: _________________________ (As available)

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

	New	___X____
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	_(2)__
	55.43	_____

Comments:

Question # _81___

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	__1___
	Group #	_____	__1___
	K/A #		0055_2.4.8_
	Importance Rating	_____	__4.5__

K/A Statement: (000055 Station Blackout) 2.4.8 Knowledge of how AOP's are used in conjunction with EOP's

Proposed Question:

The plant is currently in hot shutdown with a T-cold at 532°F, preparing for a startup following refueling when a station blackout occurs.

What ONE of the following actions should be taken?

A) Implement AOP-32, "Loss of 4160 Volt or 480 Volt Bus Power", then transition to EOP-07, "Station Blackout"
B) Implement AOP-31, "161 KV Grid Malfunctions", AOP-32, "Loss of 4160 Volt or 480 Volt Bus Power", and AOP-17, "Loss of Instrument Air"
C) Enter EOP-00, "Standard Post Trip Actions", then transition to EOP-07, "Station Blackout", and reference AOP-31, "161 KV Grid Malfunctions" to restore power to 4160 buses once off-site power is restored
D) Enter EOP-00, "Standard Post Trip Actions", then transition to EOP-07, "Station Blackout", and reference AOP-17, "Loss of Instrument Air"

Proposed Answer: __D___

Explanation: Per S-O-1, if T-cold is greater than 525 degrees, EOP-00 is entered anytime a reactor trip occurs or would have occurred. Therefore, EOP-00 would be entered, and the transition made to EOP-07 per diagnostics. EOP-07 specifically references the operators to use AOP-17 to help mitigate the consequences due to the concurrent loss of instrument air. Therefore "D" is correct. Answers "A" and "B" are plausible if the candidate does not understand the guidance in S-O-1 and believes that because the plant is shutdown, the EOP's are not entered. Answer "C" is plausible if the candidate believes that the AOP would be referenced for restoration of power to the buses. This is not the case as the EOP has specific steps to energize 1A3 and 1A4 from offsite sources, so AOP use is not required.

Technical Reference(s): _S-O-1, "Conduct of Operations" Rev. 84 pg. 62, TBD-EOP-07, "Station Blackout" Rev. 14, AOP-31, "161 KV Grid Instability" Rev. 10, AOP-32, "Loss of 4160 Volt or 480 Volt Bus Power" Rev. 16____

(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 _(5)__

Comments:

Question # _82___

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	__1___
	Group #	_____	__2___
	K/A #		0003_2.2.37_
	Importance Rating	_____	__4.6__

K/A Statement: (000003 Dropped Control Rod) 2.2.37 Ability to determine operability of safety related equipment.

Proposed Question:

OP-2A, Plant Startup, is in progress. The crew has just entered Attachment 4, Hot Standby, Mode 2 to Minimum Load, Mode 1. A full length CEA drops to the bottom of the reactor core.

What, if any, Technical Specification requirement(s) applies to the dropped rod?

1. Verify Shutdown Margin satisfied within one hour.
2. Realign the dropped CEA with the other CEA's in its group within one hour.
3. Declare the CEA inoperable within one hour.
4. There is no Technical Specification requirement at this time.

A) Both 1 and 2 apply.
B) Both 1 and 3 apply
C) Either 2 or 3 apply.
D) 4

Proposed Answer: ___D___

Explanation: 1, 2, and 3 are all possible Tech Spec requirements for a Full Lenth CEA Position during Power Operation. Power Operation is defined in Tech Specs as greater than 2% power. Attachment 4 is entered when power is less than 1% (prerequisite); therefore Tech Spec does not apply.

Technical Reference(s): _______Tech Specs 2.10.2, Tech Spec Definitions, OP - 2A, Attach 4 Prerequisites.________________________

(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 ____________
(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 _____
55.43 __(2)_

Comments:

Question # _83___

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	__1___
	Group #	_____	__2___
	K/A #		0069_2.2.40_
	Importance Rating	_____	__4.7__

K/A Statement: (000069 Loss of Containment Integrity) Ability to apply technical specifications for a system.

Proposed Question:

At 1022 on July 6th, following a containment entry, you are notified by the operators that the PAL door interlock was failed and both doors could be opened at the same time. The outer PAL door was closed and padlocked shut at 1031.

Assuming that the door interlock cannot be repaired, when is the latest that the plant MUST be in cold shutdown per Technical Specification 2.6?

A) 2322 July 7th
B) 522 July 8th
C) 1622 July 9th
D) 2222 July 9th

Proposed Answer: __D__

Explanation: Per the technical specification bases, any mechanism on the personnel air lock other than a door is under 2.6(1)b(ii). If both doors are declared inoperable, 2.6(1)a applies. The definition for operability, in part, is "A device is capable of performing its specified safety function and all necessary controls or other auxiliary equipment required for the device to perform its safety function are also capable of performing their related support functions(s)." In this case a mechanism on the personnel air lock is inoperable (the door interlock), and therefore 2.6(1)b(ii) applies. Per this technical specification, the mechanism must be repaired and the personnel air lock restored to operable status within 24 hours, or be in hot shutdown within the following 6 hours and cold shutdown 30 hours after that, which is 2222 July 9th. Answer "B" is plausible if the individual believes that both PAL doors are inoperable due to the failure of the door interlock, in which case 2.6(1)a would apply, in which if it is not restored to operable within 1 hour, must be in hot shutdown 6 hours after that, subcritical and <300°F 6 hours after that, and cold shutdown 30 hours after that. The other two distracters are plausible if the individual leaves out the six hours (30 hours AFTER the six hours has elapsed, but they do 30 hours including the six hours).

Technical Reference(s): __T.S. 2.6___________
(Attach if not previously provided) ________________________
(including version/revision number) ________________________

Proposed references to be provided to applicants during examination: ______T.S. 2.6_____

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 __(2)__

Comments:

Question # _84___

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	__1___
	Group #	_____	__2___
	K/A #		A13_AA2.2_
	Importance Rating	_____	__3.8__

K/A Statement: (CE/A13 Natural Circ) **Ability to determine and interpret the following as they apply to the (Natural Circulation Operations)** AA2.2 Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

Proposed Question:

A loss of offsite power has occurred. Standard post trip actions have been performed and subcooled natural circulation confirmed. EOP-02, "Loss of Offsite Power / Loss of Forced Circulation" has been entered and actions are currently in progress to cooldown to shutdown cooling entry conditions using natural circulation. An operator has begun the cooldown using the atmospheric dump valve.

Parameters are as follows:

Charging pumps CH-1A and CH-1B are operating
T-hot: 530°F and decreasing
T-cold: 475°F and decreasing
CET's: 525°F and decreasing
Pressurizer pressure: 2000 psia
Pressurizer level: 20%
Steam Generator RC-2A level: 55% WR and increasing
Steam Generator RC-3B level: 57% WR and increasing

Which ONE of the following describes the current status of subcooled natural circulation and the action that you would direct to be taken in accordance with EOP-02, EOP/AOP Floating Steps, and their bases?

A) Subcooled natural circulation is being maintained. Secure from the cooldown and wait five to fifteen minutes for reactor coolant loop temperatures to stabilize, verify subcooled natural circulation, then continue the cooldown and depressurization.
B) Subcooled natural circulation is being maintained. Increase steam generator feed rate to match steam flow rate to maintain steam generator levels.
C) Subcooled natural circulation is NOT being maintained. Start charging pump 1C to restore RCS inventory during two phase natural circulation.
D) Subcooled natural circulation is NOT being maintained. Depressurize RCS using auxiliary spray and start all available HPSI pumps to restore RCS inventory during two phase natural circulation.

Proposed Answer: __A__

Explanation: As provided in the stem, an operator is currently cooling down the plant using the ADV. Per the bases for EOP-02, EOP/AOP Floating Steps, and lesson plan 07-15-16, loop transit times are approximately 5-15 minutes. During cooling, because of this transit time, the 50 degrees ΔT can be exceeded, making the operator think that they have lost subcooling, but they haven't, because the T-cold is decreasing due to heat removal through the steam generators, but this has not yet reached T-hot, which is at its previous value. By stopping the cooldown and waiting for the loop transit time, this enables getting a true value for ΔT. Therefore answer "A", although not specifically described in the procedure, would be the appropriate response for this occurrence, to verify that it is just a loop transit time issue and not an actual loss of subcooled natural circulation. Answer "B" is plausible in that one would expect to require more feed if you're steaming the steam generator, however, the EOP-02 bases cautions against overfeeding the steam generators due to excessive RCS cooldown, pressurized thermal shock, and the shrink that might make steam generator level actually decrease before level begins to increase. Also, in this case, although the steam generator level is outside the band, it is trending in the correct direction, so no action needs to be taken. Therefore it is incorrect. Answer "C" is plausible because that action is directed if subcooled natural circulation is lost (as indicated by ΔT going over 50 degrees), but the bases caution that loop transit times can affect the accuracy of T-hot and T-cold. Because the operator is currently cooling down the RCS, loop transit times affect how quickly that effect is seen, and time should be given to see the effect on ΔT after the loop transit time. Answer "C" is therefore plausible but incorrect, in that it takes action that is not necessary based on current information. Answer "D" is plausible because another action for two phase natural circulation is to use HPSI and LPSI pumps to get adequate flow per the SI flow curves. The procedure does not state to depressurize the RCS so that it will inject however, and the answer is therefore incorrect because of this and for the same reason as "C".

Technical Reference(s): __Lesson Plan 07-15-16, Rev. 5, TBD-EOP/AOP Floating Steps Rev. 1, TBD-EOP-02, "Loss of Offsite Power / Loss of Forced Circulation" Rev. 18_

(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 _(5)_

Comments:

Question # _85___

Examination Outline Cross-Reference:

		RO	SRO
	Level	RO	SRO
	Tier #	_____	__1___
	Group #	_____	__2___
	K/A #		A16_AA2.1_
	Importance Rating	_____	__3.5__

K/A Statement: (CE/A16 Excess RCS Leakage) **Ability to determine and interpret the following as they apply to the (Excess RCS Leakage)** AA2.1 Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Proposed Question:

The RCS is leaking into SG RC-2A at a constant rate of 0.15 GPM. Which is the least restrictive (longest time) requirement that meets both Technical Specifications and Procedure directions?

A) Commence a power reduction per OP-4, Load Change and Normal Power Operation, to be in Mode 3 within 24 hours.
B) Commence a power reduction per OP-4, Load Change and Normal Power Operation, to be in Mode 3 within 12 hours.
C) Commence a power reduction per AOP-05, Emergency Shutdown, to be in Mode 3 within 6 hours.
D) Commence a power reduction per AOP-05, Emergency Shutdown, to be below 50% power within one hour and in Mode 3 within three hours.

Proposed Answer: __C__

Explanation: Per AOP-22, Attachment B, Step 11, if primary to secondary leakage exceeds 150 gpd (0.10 gpm), commence a plant shutdown per AOP-5 and enter Mode 3 within 6 hours. Per T.S. 2.1.4, if primary to secondary leakage exists outside the T.S. limit of 150 gpd through any one steam generator, be in Mode 3 within 6 hours. Therefore, answer C is correct. Answer A is incorrect but plausible because those are the actions driven by procedure if the primary to secondary leakage exceeds 75 gpd (0.05 gpm) sustained for 1 hour with less than 30 gpd increase in a 1 hour period of time (Step 9). Answer B is incorrect but plausible because 12 hours is a reasonable period of time to require entry into Mode 3 and is in between two procedurally directed times (24 hours per Step 9 and 6 hours per Step 11). Answer D is incorrect but plausible because these are the actions driven by procedure is the primary to secondary leakage exceeds 75 gpd AND a greater than 30 gpd increase in leak rate over an hour period of time.

Technical Reference(s): ____TS 2.1.4, AOP-22, Attach B, Step 11_
(Attach if not previously provided) ______________________________

(including version/revision number) ____

Proposed references to be provided to applicants during examination: ________________

Learning Objective: _________________________ (As available)

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

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 _____
55.43 _(5)__

Comments:

Question # _86___

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	__2___
	Group #	_____	__1___
	K/A #		006_G2.2.5_
	Importance Rating	_____	__3.2__

K/A Statement: (006 Emergency Core Cooling) 2.2.5 Knowledge of process for making design or operating changes to the facility

Proposed Question:

It has been identified that the EOP/AOP Floating Steps contain incorrect guidance on the throttling of LPSI. If the procedure is worked as written, this would result in a reduction in the ECCS flow rate below allowable values and, consequently, would reduce subcooling margin below 20°F.

Per SO-G-74, "Fort Calhoun Station EOP/AOP Generation Program", and SO-G-30, "Procedure Changes and Generation", which ONE of the following actions would be taken to address this procedure issue?

A) A Temporary Procedure Change would be implemented immediately; then the permanent procedure change would be processed as a Priority 1 procedure change.
B) A Temporary Procedure Change would be implemented immediately; then the permanent procedure change would be processed as a Priority 2 procedure change.
C) A Temporary Procedure Change would NOT be implemented; and the permanent procedure change would be processed as a Priority 1 procedure change.
D) A Temporary Procedure Change would NOT be implemented; and the permanent procedure change would be processed as a Priority 2 procedure change.

Proposed Answer: __C__

Explanation: Per SO-G-30, temporary procedure changes CANNOT be performed if the change is to an EOP, AOP, RERP, or ACSO, making answers "A" and "B" incorrect, but plausible if the student believes that the procedure must be corrected immediately. Per SO-G-74, a Priority 1 procedure change is where the procedure provides guidance which is impossible to implement, or guidance which, if implemented, would produce deteriorating or unsafe plant conditions. Priory 2 is when the procedure does not contain sufficient guidance to complete the required instruction/contingency action or is deficient with regard to procedure flow, structure, or usability (Operator convenience). From this, one determines that the above stem should be a Priority 1 procedure change as the guidance, if implemented, would result in deteriorating plant conditions due to the reduced ECCS flow and subcooling margin. Therefore answer "C" is correct. Answer "D" is plausible is the student doesn't apply the SO-G-74 guidance correctly.

Technical Reference(s): SO-G-30, “Procedure Changes and Generation” Rev. 121, SO-G-74, “Fort Calhoun Station EOP/AOP Generation Program” Rev. 14_

(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 _____
55.43 _(3)__

Comments:

Question # _87___

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	__2___
	Group #	_____	__1___
	K/A #		010_G2.2.37_
	Importance Rating	_____	__4.6__

K/A Statement: (010 Pressurizer Pressure Control) 2.2.37 Ability to determine operability and/or availability of safety related equipment.

Proposed Question:

The minimum Technical Specification capacity is ____and basis is _____for Pressurizer Heaters.

A) 150 KW, to overcome the Pressurizer Spray bypass flow and maintain the RCS boron concentration within 50 ppm of the Pressurizer.
B) 150 KW, to be available during a loss of offsite power to maintain natural circulation at Hot Shutdown.
C) 225 KW, to overcome the Pressurizer Spray bypass flow and maintain the RCS boron concentration within 50 ppm of the Pressurizer.
D) 225 KW, to be available during a loss of offsite power to maintain natural circulation at Hot Shutdown.

Proposed Answer: ___B___

Explanation: The 225KW distractor was chosen because 2 backup heater banks are rated at 225KW and 2 are rated at 150KW. The value and basis are right out of TS.

Technical Reference(s): _____TS 2.1.7____________________
(Attach if not previously provided) ____________________
(including version/revision number) ____________________

Proposed references to be provided to applicants during examination: _________________

Learning Objective: _________________________ (As available)

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

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 __X__

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 _____

55.43 _(2)_

Comments:

Question # _88___

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	__2___
	Group #	_____	__1___
	K/A #		013_A2.06_
	Importance Rating	_____	__4.0__

K/A Statement: (013 Engineered Safety Features Actuation) **Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations;** A2.06 Inadvertent ESFAS actuation

Proposed Question:

Given the following plant conditions:

Reactor has tripped
SIAS has NOT actuated
Alarms “86/A OPLS TRIP”, “86/B OPLS TRIP”, “LOAD SHED OFF NORMAL” are lit
No loss of offsite power has occurred
Buses 1A3 and 1A4 are powered from their respective diesel generators
No 4160 Volt engineered safeguards equipment can be started
Standard Post Trip Actions are completed and no other event is in progress

Based on the above, which ONE of the following procedures should be used FIRST to restore engineered safeguards equipment?

A) AOP-23 “Reset of Engineered Safeguards”
B) AOP-32 “Loss of 4160 Volt or 480 Volt Bus Power”
C) EOP/AOP Floating Step H “Reset of Engineered Safeguards”
D) EOP/AOP Floating Step JJ “Disabling Safeguards Relays”

Proposed Answer: __A___

Explanation: Based on the given conditions, an inadvertent OPLS has occurred, preventing safeguards equipment from operating. Therefore, AOP-23 should be entered to reset OPLS (and this guidance is given in the ARP for the “LOAD SHED OFF NORMAL” alarm). Answer “A” is correct. Answers “B” and “D” are incorrect but plausible because they are referenced in AOP-23 and used. AOP-32 is used following reset of OPLS to restore power to 480 V MCC’s. Floating Step JJ is a contingency step if the preferred method of reset does not work. However, the question asks for which procedure should be used first. Answer “C” is incorrect but plausible because it does reset ESF actuations, but it is used for legitimate actuations, not inadvertent.

Technical Reference(s): __AOP-23 “Reset of Engineered Safeguards”, Rev. 9, EOP/AOP Floating Steps, Rev.1, STM Vol. 19 Engineered Safeguards Control System Rev. 37, ARP-AI-30A/A33-1, Rev. 21, ARP-AI30A/A33-2 Rev. 23, ARP-AI-30B/A34-1, Rev. 22, ARP-AI-30B/A34-2, Rev. 23___

(Attach if not previously provided) ____

(including version/revision number) ____

Proposed references to be provided to applicants during examination: ___None_______

Learning Objective: __0717-23 01.00 Use Reset of Engineered Safeguards Procedure to mitigate the consequences of an inadvertent safeguards actuation.______ (As available)

Question Source:	Bank #	_______
	Modified Bank #	07-17-23 009 (Note changes or attach parent)
	New	_______

Question History: Last NRC Exam __2010___

(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	_(5)__

Comments: Changed the stem for a different inadvertent actuation (OPLS versus CPHS) and changed distracter accordingly.

Question # _89___

Examination Outline Cross-Reference:	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 an 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. Transition to Procedure AOP-11, “Loss of Component Cooling Water” and establish feed and bleed cooling for the CCW system components.
B. Transition to 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 the fire protection system 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, AOP-11, AOP-18
(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:

Question # _90___

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	__2___
	Group #	_____	__1___
	K/A #		064_G2.2.42_
	Importance Rating	_____	__4.6__

K/A Statement: (064 Emergency Diesel Generator) Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

Proposed Question:

A local operator has just finished his rounds and reports the following:

Diesel fuel oil inventory in FO-1 is 13,000 gallons
Diesel fuel oil inventory in FO-10 is 10,000 gallons
Diesel generator DG-1 lube oil inventory is 504 gallons
Diesel generator DG-2 lube oil inventory is 554 gallons
Neither diesel generator is undergoing any maintenance.

Based on the above report, which ONE of the following technical specifications applies?

A) Enter T.S. 2.7(3)(a) and restore required inventory within 48 hours.
B) Declare diesel generator DG-1 inoperable, enter T.S. 2.7(2)(j), and restore to operable within 7 days.
C) Declare diesel generator DG-2 inoperable, enter T.S. 2.7(2)(j), and restore to operable within 7 days.
D) Declare both diesel generators inoperable, enter T.S. 2.0.1, be in hot shutdown within 6 hours, and cold shutdown within the following 30 hours.

Proposed Answer: __D___

Explanation: Per T.S. 2.7(3)(f) if one or more diesels have diesel fuel oil, lube oil or required starting air subsystem not within limits for reasons other than a, b, c, d, or e, declare the associated diesel inoperable immediately. Per 2.7(3)(a), if FO-1 is less than 16,000 and/or FO-10 is less than 10,000 gallons, but the combined inventory in FO-1 and FO-10 is greater than 23,350 gallons, restore required inventory within 48 hours. The combined inventory in the stem is 23,000 gallons, and therefore 2.7(3)(a) does not apply, making T.S. 2.7(3)(f) apply. Since FO-1 is common to both tanks (as is FO-10 as it fills FO-1), both diesels must be declared inoperable and T.S. 2.01 entered. That makes answer D correct. Answer A is incorrect but plausible if the student does not recall the value of the combined inventory in the T.S. and applies T.S. 2.7(3)(a). Answers B and C are incorrect but plausible if the applicant believes that FO-1 and FO-10 each supply a separate diesel generator and based on which one they believe to be low. They would choose answer B if they recognize that FO-1 is low, and answer C if they believed that FO-10 is low. In this case, FO-10 is not low, as it meets the minimum T.S. value, but FO-1 is low, and is 3,000 gallons less than the T.S. minimum value.

Technical Reference(s): ___T.S. 2.0.1, T.S. 2.7(2), T.S. 2.7(3)_______________
(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 _(2)____

Comments:

Question # _91___

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	__2___
	Group #	_____	__2___
	K/A #		027_2.2.25_
	Importance Rating	_____	__4.2__

K/A Statement: (027 Containment Iodine Removal) **Knowledge of the bases in Technical Specifications for LCO's and safety limits.**

Proposed Question:

According to the Technical Specifications (and their associated bases), what is required for gaseous iodine removal from the air in containment during the design basis Loss of Coolant Accident?

A. One VA-7 cooling unit and one VA-3 cooler and filtering unit because sodium tetraborate only reduces particulate iodine.
B. One VA-7 cooling unit and one VA-3 cooler and filtering unit because sodium tetraborate only reduces gaseous iodine.
C. One containment spray pump and one VA-3 cooler and filtering unit because it contains a charcoal filter.
D. One containment spray pump and one VA-7 cooling unit because it contains a HEPA filter.

Proposed Answer: __B___

Explanation:

Answer B is correct, One VA-7 cooling unit and one VA-3 cooler and filtering unit because sodium tetraborate only reduces gaseous iodine not particulate.

Answer A is credible but not correct because the sodium tetraborate reduces the amount of iodine that becomes volatile from the RCS water in the sump by reducing the pH but can't reduce the particulate iodine. Answer C and D are not correct because the Containment Spray pumps are not credited in the TS bases for iodine removal (or at all for the LOCA DBA, only during the MSLB DBA are they required), although they would remove some iodine if started during the LOCA by scrubbing it from the air.

Technical Reference(s): TS 2.3 and 2.4 bases
(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 Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 _____
55.43 __2__

Comments:

Question # _92___

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	__2___
	Group #	_____	__2___
	K/A #		034_K4.02_
	Importance Rating	_____	__3.3__

K/A Statement: (034 Fuel Handling Equipment) **Knowledge of design feature(s) and/or interlock(s) which provide for the following:** K4.02 Fuel movement

Proposed Question:

The FH-1 refueling machine hoist cannot be raised if:

A) Hoist encoder position is greater than the software value for the Hoist Up Limit
B) Hoist encoder compare error is detected
C) Hoist is in the Upender Zone with the upender not vertical
D) Hoist is above the Upper Grapple Operate Zone with the grapple closed

Proposed Answer: __B___

Explanation: Per STM Vol. 40 "Refueling System" rev. 20, the hoist is interlocked such that if a hoist encoder compare error is detected, the interlock is not met and the hoist can be neither raised nor lowered. Therefore "B" is correct. Answer "A" is plausible yet incorrect in that the hoist cannot be raised if the encoder position is LESS than the software value, not greater than the software value. Answer "C" is plausible but incorrect because they are true for hoist lower, not hoist raise. Answer "D" is incorrect because that is an interlock that must be met to allow hoist raise.

Technical Reference(s): ___STM Vol. 40 "Refueling System", Rev. 20_______
(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 ______
55.43 _b(7)____

Comments:
Similar to bank questions 07-11-13 008, 07-11-13 030, and 07-11-13 042

Question # _93___

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	__2___
	Group #	_____	__2___
	K/A #		071_A2.02_
	Importance Rating	_____	__3.6__

K/A Statement: (071 Waste Gas Disposal) **Ability to (a) predict the impacts of the following malfunctions or operations on the Waste Gas Disposal System; and (b) based on those predictions, use procedures, to correct, control, or mitigate the consequences of those malfunctions or operations:** A2.02 Use of waste gas release monitors, radiation, gas flow rate, and totalizer

Proposed Question:

A waste gas release is planned. FR-758, the Stack Total Flowrate Recorder on AI-44, is not working. Per the ODCM, what additional actions are required to perform this release due to the recorder being inoperable?

A) Stack flow readings must be manually recorded on the gas discharge log at least every four hours
B) No additional actions are required as long as FR-532, Waste Gas Release Rate Recorder on AI-100, is operable
C) Stack flow must be determined at least every four hours by multiplying the number of running AB exhaust fans by a value given in the ODCM
D) The release is not allowed until FR-758 is repaired, has been recalibrated, and has passed a functional test

Proposed Answer: __A___

Explanation: Per the ODCM, a minimum of one operable channel is required for the waste gas discharge header (FR-532) AND Auxiliary Building Stack (FR-758). Therefore this requirement is not met. Note that the question stem states that the flow recorder is not working but does not state that the flow measurement device is not working. The required action (Action 7) states that releases may continue provided the flow rate is estimated or recorded manually at least once per 4 hours during the release. Therefore B and C are incorrect yet plausible. Answer "C" is incorrect in that the ODCM does not provide a value by which the number of exhaust fans can be multiplied to provide an estimated flow rate. The ODCM does provide the maximum flow rate with all three exhaust fans running and 2 containment purge fans running. Therefore "C" is incorrect but plausible. Answer "A" is in-line with the requirements of the ODCM and is therefore correct.

Technical Reference(s): _CH-ODCM-0001, "Offsite Dose Calculation Manual" Rev. 21, OI-WDG-2 "Waste Gas Disposal System Release" Rev. 24___

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

Proposed references to be provided to applicants during examination: ____None______

Learning Objective: _1950-04 2.02 State the action to be taken in the event liquid and gaseous effluent instrumentation is not operable_____ (As available)

Question Source:	Bank #	_19-50-04 001_
	Modified Bank #	_______ (Note changes or attach parent)
	New	_______

Question History: Last NRC Exam __2004_____

(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	_(5)__

Comments: Slightly changed wording of distracter C to make it more plausible. Deleted word automatic from stem (changed from "automatic waste gas release" to waste gas release).

Question # 94

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	__3__
	Group #	_____	_____
	K/A #	_____	2.1.35
	Importance Rating	_____	__3.9_

K/A Statement: Knowledge of the fuel-handling responsibilities of SROs.

Proposed Question:

You took the watch as the fuel handling coordinator at 1815.
During shift turnover, the offgoing fuel handling coordinator informed you that shutdown cooling was lost for approximately one hour when LPSI pump SI-1B tripped off-line at 1513. LPSI pump SI-1A is currently providing shutdown cooling while troubleshooting is continuing on LPSI pump SI-1B.
At 1950, an irradiated fuel assembly has been removed from the upender and is currently being indexed to the required core location. At this time, you receive notification from the control room that LPSI pump SI-1A has tripped off-line and cannot be restarted. The control room is working on getting a containment spray pump aligned to provide shutdown cooling.

Based on the above information, which ONE of the following actions will you direct to be taken?

A) Complete insertion of current fuel assembly and then stop further fuel movement until shutdown cooling can be restored.
B) Stop movement of current fuel assembly at its current location and suspend further fuel movement until shutdown cooling can be restored.
C) Return current fuel assembly to the upender and then stop further fuel movement until shutdown cooling can be restored.
D) Complete insertion of current fuel assembly and continue fuel loading provided that shutdown cooling is restored to operation by 2050 hours.

Proposed Answer: __C__

Explanation:
Per T.S. 2.8.1(3), Note 1: SDC can be secured for </= 1 hour per 8 hour period. The stem of the question states that SDC was lost for an hour at 1513, so the 8 hour period would go until 2313. SDC was again lost at 1950, which is within the 8 hour period of time. Therefore, this Note does NOT apply. Therefore, the required action is to immediately suspend loading of irradiated fuel assemblies into the reactor core. This is an irradiated assembly, so they need to suspend loading of this fuel assembly. They haven't started insertion yet, as they are still indexing, so they cannot insert it. This makes answer "A" plausible, but incorrect, if the candidate believes that they can finish this insertion and then suspend fuel movement. Per the definitions section of the Tech Spec, for both Refueling operations and core alterations, suspension of these shall not prevent completions of movement of a component to a safe, conservative position. Also, per OP-12, Precaution 4, the Fuel Handling Coordinator can direct the assembly to be returned to the Upender area if the fuel assembly is in transit to the core. In

this case, the fuel assembly is in transit to the core. Therefore, the conservative action is not to leave the fuel assembly where it currently is, but to return the fuel assembly to the upender. This makes answer “C” correct, and answer “B” incorrect but plausible if the candidate does not know that these options are available. Answer “D” is plausible if the candidate does not know or understand that Note 1 applies to any 8 hour window and that just because the shift changed, does not reset the Note 1 clock.

Technical Reference(s): _T.S. Definitions, T.S. 2.8.1(3), OP-12 “Fueling Operations”, Rev. 60_

(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 _(2), (6)_

Comments:

Question # 95_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	__3__
	Group #	_____	_____
	K/A #	_____	_2.1.37_
	Importance Rating	_____	__4.6_

K/A Statement: Knowledge of procedures, guidelines, or limitations associated with reactivity management.

Proposed Question:

During performance of surveillance test OP-ST-CEA-003, “Control Element Assembly Partial Movement Check”, at 100% power, CEA 38 was successfully inserted six inches to 114”, but then would not withdraw back to its original position of 120”. The rest of that group’s CEAs are located at 120”. It has been determined that the CEA is trippable and there is no indication of excess friction or mechanical interference. All other CEAs have passed the surveillance test satisfactorily.

Which ONE of the following is directed by AOP-02 “CEA and Control System Malfunctions”, based on the above information?

A) Trip the reactor
B) Commence a reactor shutdown
C) Lower reactor power to less than 70% ΔT power
D) Continue operation at current power level

Proposed Answer: __D__

Explanation: CEA 38 is a group 4 control rod. Per TDB-VI, COLR, Rev. 39, the long term steady state insertion limit for group 4 at 100% power is 104.5”. This CEA is above that limit. It’s trippable, within 12” of the rest of the group, and not mechanically bound or stuck. Only one CEA is inoperable, since the others passed their surveillance. Therefore, operation can continue in this configuration and answer “D” is correct. Answer “A” is plausible because it is required by procedure, but only if a CEA is misaligned >18”. Answer “B” is plausible because it is required by procedure, but only if more than 1 CEA is inoperable, if the CEA is untrippable, mechanically bound (stuck), and if the CEA is >12” but <18” misaligned and cannot be aligned within one hour after power level reduced to 70%. None of these are applicable in this case. Answer “C” is plausible because that is the initial action directed if the CEA is >12” but <18” misaligned.

Technical Reference(s): _TBD-AOP-02 “CEA and Control System Malfunctions” Rev. 7, TDB-VI “COLR”, Rev. 39 (Cycle 26), TDB-I.A.3 “CEDM Locations” 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______

(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 _____
55.43 _(6)_

Comments:

Question # 96_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	__3__
	Group #	_____	_____
	K/A #	_____	_2.2.6_
	Importance Rating	_____	__3.6_

K/A Statement: Knowledge of the process for making changes to procedures.

Proposed Question:

Which of the following procedure changes requires a 50.59 evaluation?

A. Adding a new valve stroke time to check
B. Correcting step numbers in a quality procedure note
C. Adding a drawing or figure for clarification
D. Marking optional procedure steps Not Applicable

Proposed Answer: __A____

Explanation:
All three distracters (B, C, and D) are administrative in nature and therefore do not require a 50.59 evaluation.

Technical Reference(s): ____10 CFR 50.59______________________________
(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 # ___X___
Modified Bank # _______ (Note changes or attach parent)
New _______

Question History: Last NRC Exam Clinton 2001______

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

10 CFR Part 55 Content: 55.41 _____
55.43 __3__

Comments: taken from INPO Exam bank (Attached) . I modified distractor D since it did not seem credible

Question # 97_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	__3__
	Group #	_____	_____
	K/A #	_____	_2.2.12_
	Importance Rating	_____	__4.2_

K/A Statement: Knowledge of Surveillance Procedures.

Proposed Question:

In accordance with Technical Specification Surveillance Requirement 3.10(7) for "Reactor Core Parameter" Departure from Nucleate Boiling (DNB), some of the required parameters required to be verified and their associated frequencies for this surveillance are:

A. Cold Leg Temperature — once per shift
Pressurizer Pressure — once per shift
Total Reactor Vessel Coolant Flow Rate — once per shift

B. Cold Leg Temperature — once per shift
Pressurizer Pressure — once per shift
Total Reactor Vessel Coolant Flow Rate — once per week

C. Cold Leg Temperature — once per shift
Pressurizer Pressure — once per shift
Total Reactor Vessel Coolant Flow Rate — once per month

D. Cold Leg Temperature — once per week
Pressurizer Pressure — once per week
Total Reactor Vessel Coolant Flow Rate — once per month

Proposed Answer: ___C___

Explanation:

C is correct, Temp and Pressure required to be done shiftly while flow is required monthly. DIstractors are combinations of frequencies that are not correct but plausible.

Technical Reference(s): _TS 3.10(7) a and b________________________
(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 Cognitive Level:	Memory or Fundamental Knowledge		__X__
	Comprehension or Analysis		______
10 CFR Part 55 Content:	55.41 ______		
	55.43 __2__		

Comments:

Question # 98_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	__3__
	Group #	_____	_____
	K/A #	_____	_2.3.4_
	Importance Rating	_____	__3.7_

K/A Statement: Knowledge of radiation exposure limits under normal or emergency conditions.

Proposed Question:

An accident has occurred in the plant which has resulted in core damage. A General Emergency has been declared based on projected radiological conditions offsite. A release is in progress. The TSC has been activated and the Site Director has completed EPIP-OSC-2, “Command and Control Position Actions/Notifications”, Attachment 6.7, “Relief Checklist”. The EOF is still in the process of activating.

It has been determined that the release can be terminated by isolation of a manual valve. It is estimated that it will take 12 minutes to isolate the valve. The radiation field at the valve is 30 R/hr.

Per EPIP-EOF-11, “Dosimetry Record, Exposure Extensions and Habitability”, who can authorize this dose extension?

A) Control Room Coordinator, Site Director, or Emergency Director
B) Control Room Coordinator ONLY
C) Site Director ONLY
D) Emergency Director ONLY

Proposed Answer: __C___

Explanation: Per EPIP-EOF-11, the emergency facility directors in their facility can authorize exposures up to 5 REM TEDE. With the given values in the stem above, the expected dose that will be received is 6 REM. For any authorizations above 5 REM (as in this case), it must be approved by the current command and control position in charge. In this case, based on the stem information, the TSC has the command and control function for the event. Therefore answer “C” is correct. Answer “A” is incorrect but plausible because this would be true if the anticipated dose was less than 5 REM, which it is not. Answers “B” and “D” are incorrect but plausible because they could be true depending on who had the command and control function at this time.

Technical Reference(s): _EPIP-OSC-2, “Command and Control Position Actions/Notifications” Rev. 52, EPIP-EOF-11, “Dosimetry Record, Exposure Extensions and Habitability” Rev. 26_

(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 # ADM-EP 062 (Note changes or attach parent)
New _______

Question History: Last NRC Exam __2004____
(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 _(4)__

Comments: Modified two conditions in the stem, one significantly: who had command and control at the time of the question, and one that changed the dose rate and time to completed the task. I changed distracters to make them more plausible.

Question # 99_

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	__3__
	Group #	_____	_____
	K/A #	_____	_2.4.8_
	Importance Rating	_____	__4.5_

K/A Statement: Knowledge of how abnormal operating procedures are used in conjunction with EOPs.

Proposed Question:

In the “Conduct of Operations Procedure” SO-O-1, guidance is offered for consistent, conservative response to events that include the use of both Emergency Operating Procedures and Abnormal Operating Procedures. One circumstance where an Abnormal Operating Procedure would take priority over Emergency Operating Procedure E-0 “Standard Post Trip Actions” is:

A. E-0 must always be performed first
B. During an OP-3A plant shutdown with an RPS activation
C. Only when directed by the shift manager
D. When the reactor coolant system is on shutdown cooling

Proposed Answer: ___D__

Explanation:

On page 63 of SO-O-1 it stated that the SPTA procedure E-0 shall be followed following any RPS actuation with two exceptions: 1) If the RPS actuation is part of OP-3A plant shutdown or a pre-planned maintenance or surveillance activity, and 2) if the RCS is on shutdown cooling (Tcold < 350 degrees F).

Distracter A is credible because this is the only exception and if not known then A would be correct. Distractor B is credible but not correct because no RPS actuation occurred. Distractor C is credible but not correct because shift managers can override certain things but not for this situation.

Technical Reference(s): __SO-O-1 Rev 84____(page 63)________________
(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 Cognitive Level:	Memory or Fundamental Knowledge	______	
	Comprehension or Analysis	__X__	
10 CFR Part 55 Content:	55.41	______	
	55.43	__5__	

Comments:

Question # 100

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	_____	__3__
	Group #	_____	_____
	K/A #	_____	_2.4.09_
	Importance Rating	_____	__4.2_

K/A Statement: Knowledge of low power / shutdown implications in accident (e.g. Loss of coolant accident loss of residual heat removal) mitigation strategies.

Proposed Question:

Given the following plant initial conditions:

- A refueling outage is in progress.
- 106 fuel assembles have been loaded into the core
- The 107th fuel assembly (a new fuel assembly) is being moved over the core
- The Equipment Hatch, Room 66 roll-up doors, Room 66 construction access Opening, and both PAL doors are all open

Plant conditions then change:

- The moving fuel assembly drops from FH-1
- No radiation monitor alarms are received

Which one of the following actions meets the requirements of AOP-08, “FUEL HANDLING INCIDENT” if performed within one hour?

A. Initiate CIAS using the EMERGENCY OPERATE switches. Close the equipment hatch, all applicable containment penetrations, and at least one of the PAL doors.

B. Initiate CIAS using the EMERGENCY OPERATE switches. Close the Room 66 Roll-up doors and both of the PAL doors.

C. Initiate VIAS using the CRHS test switches. Close the equipment hatch, all applicable containment penetrations and at least one of the PAL doors.

D. Initiate VIAS using the CCRHS test switches. Close the Room 66 Roll-up doors and both of the PAL doors.

Proposed Answer: __C___

Explanation:

C is correct per AOP-08, Initiate VIAS using the CRHS test switches and Close the equipment hatch, all applicable containment penetrations and at least one of the PAL doors. DIstractors A and B are incorrect because CIAS is not initiated. Distractor D is incorrect because the equipment hatch must be closed.

Technical Reference(s): ___AOP-08___
(Attach if not previously provided) ______
(including version/revision number) ______

Proposed references to be provided to applicants during examination: ______

Learning Objective: __0711-08 02.03______ (As available)

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

Question History: Last NRC Exam _NRC 2009 Exam______

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

10 CFR Part 55 Content: 55.41 ______
55.43 __7__

Comments: