Facility:	Wolf Creek
Vendor:	WEC
Exam Date:	8/2009
Exam Type:	R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	103	K1.08	
	Importance Rating	3.6		

Knowledge of the physical connections and/or cause-effect relationships between the containment system and the following systems: SIS, including action of safety injection reset

Proposed Question: RO Question # 1

Which ONE of the following describes the operation of Containment Isolation Phase A (CISA)?

A. ONLY actuated on Containment Pressure HIGH-1. May be physically reset independently of any other safeguards actuation signal.

B. ONLY actuated on Containment Pressure HIGH-1. Contacts in the SI reset circuitry prevent Phase "A" reset prior to SI reset.

C. Actuated on any SI actuation. Contacts in the SI reset circuitry prevent Phase "A" reset prior to SI reset.

D. Actuated on any SI actuation. May be physically reset independently of any other safeguards actuation signal.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because CISA does actuate on HIGH-1. However, any SI actuation signal will also initiate CISA.
B. Incorrect. Plausible because CISA is always reset after SI in accordance with plant procedures, so it is logical to believe that it cannot physically be reset prior to SI.
C. Incorrect. Plausible because the first half is true, and also because of the same reason that option B is plausible, due to EOP usage.
D. Correct.

Technical Reference(s): SY 1301301, Rev 9 (Attach if not previously provided)

M-744-00025

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1301301 R3 (As available)

Question Source: Bank # WTSI 55090

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: VC Summer 2006

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 08/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	010	K1.08	
	Importance Rating	3.2		

Knowledge of the physical connections and/or cause-effect relationships between the PZR PCS and the following systems: PZR LCS

Proposed Question: RO Question # 2

Given the following:

- The plant is at 55% power.
- BB LT-459 is the controlling Pressurizer Level Transmitter.
- BB LT-459 fails high instantaneously.

Assuming NO action by the crew, which ONE of the following describes the **initial observable** effect on Chemical and Volume Control System (CVCS) and Pressurizer Heaters?

A. Charging flow lowers; Pressurizer Backup Heaters energize

B. Charging flow lowers; Pressurizer Backup Heaters de-energize

C. Letdown isolates; Pressurizer Backup Heaters de-energize

D. Letdown isolates; Pressurizer Backup Heaters energize

Proposed Answer: A

Explanation (Optional):

A. Correct. 5% level deviation HIGH will cause backup heaters to energize. Charging flow lowers because the controlling channel indicates a HIGH deviation from program setpoint.

B. Incorrect. Charging lowers but heaters will energize. Plausible because heaters eventually will de-energize when the backup channel is reduced below 17%.

C. Incorrect. Letdown will isolate and heaters will de-energize at a later time when backup channel is low (<17%) if no action is taken. This would immediately happen only for a **low** failure of the controlling channel.

D. Incorrect. Letdown will isolate later when backup channel is low if no action is taken. Heaters do energize for this event based on +5% deviation.

Technical Reference(s): SY 1301000 Rev 7 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1301000 R10 (As available)

Question Source: Bank # WTSI 58389

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Callaway 2007

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Facility:	Wolf Creek
Vendor:	WEC
Exam Date:	08/2009
Exam Type:	R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	003	K2.01
	Importance Rating	3.1	

Knowledge of bus power supplies to the following: RCPS
Proposed Question: RO Question # 3

Given the following:

- A power ascension is in progress.
- Power is 33%.
- Load increase is in progress in accordance with GEN 00-004, Power Operation.

Which ONE of the following describes the current power alignment for the RCPs?

A. Startup Transformer supplying Bus PA01, which supplies RCPs "A" and "C".

B. Unit Auxiliary Transformer supplying Bus PA01, which supplies RCPs "A" and "B".

C. Startup Transformer supplying Bus PA01, which supplies RCPs "A" and "B".

D. Unit Auxiliary Transformer supplying Bus PA01, which supplies RCPs "A" and "C".

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Startup transformer supplies bus PA01/PA02 prior to synchronization. Once load is increased, station loads are transferred to UAT. Also, PA01 supplies A and B RCPs, not A and C.

B. Correct.

C. Incorrect. Startup transformer supplies bus PA01/PA02 prior to synchronization. Once load is increased, station loads are transferred to UAT. Correct loads for PA01.

D. D is incorrect but plausible because of the correct supply to PA01. RCP C is supplied by PA02.

Technical Reference(s): SY 1300300, Rev 18
GEN 00-003, Rev 72
KD 7496
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1300300 R2 (As available)

Question Source: Bank #
Modified Bank # WTSI 57838 (Note changes or attach parent)
New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 3
55.43

Mechanical components and design features of reactor primary system.

Comments:

Modified from WTSI 57838, 2007 Wolf Creek NRC Exam. (Attached)

Facility: Wolf Creek

Vendor: WEC

Exam Date: 08/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	006	K2.02
	Importance Rating	2.5	

Knowledge of bus power supplies to the following: Valve operators for accumulators

Proposed Question: RO Question # 4

Given the following:

- The plant was at 100% power.
- A LOCA occurred.
- The feeder to 480 Volt Bus NG01 is tripped open and locked out.
- The crew is performing EMG ES-11, Post LOCA Cooldown and Depressurization.

Which ONE of the following describes the action required in relation to the SI Accumulators as the cooldown and depressurization continues in EMG ES-11?

Isolate SI Accumulators....

A. "B" and "C"; Vent SI Accumulators "A" and "D".

B. "B" and "D"; Vent SI Accumulators "A" and "C".

C. "A" and "D"; Vent SI Accumulators "B" and "C".

D. "A" and "C"; Vent SI Accumulators "B" and "D".

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because there are four accumulators with each NG bus providing power to two MOV's. Each choice is used twice on each side of the semi-colon.

B. Correct. NG01A provides power to SI Accumulator Isolation Valves EP HV-8808A and EP HV-8808C. These valves will be open, and the only way to prevent N2 injection during the depressurization is to vent those accumulators. NG02A supplies EP HV-8808B and EP HV-8808D

C. Incorrect. Plausible because there are four accumulators with each NG bus providing power to two MOV's. Each choice is used twice on each side of the semi-colon.

D. Incorrect. Plausible because there are four accumulators with each NG bus providing power to two MOV's. Each choice is used twice on each side of the semi-colon.

Technical Reference(s): EMG ES-11, Rev 17 Step 34/35 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1300600 R4 (As available)

Question Source: Bank # WTSI 59247

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: McGuire 2008

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	073	K3.01
	Importance Rating	3.6	

Knowledge of the effect that a loss or malfunction of the PRM system will have on the following: Radioactive effluent releases

Proposed Question: RO Question # 5

Given the following:

- The plant is operating at 100% power.
- Discharge of Waste Monitor Tank No. 1 (THB07B) is in progress.
- The following alarms are received approximately 15 seconds apart:

- ALR 00-061B, PROCESS RAD HI
- ALR 00-061A, PROCESS RAD HI-HI
- ALR 00-109F, LIQ RADWASTE TROUBLE
- ALR 00-061C, PROCESS RAD MON FAIL

- Report from Liquid Process Control Panel HB-115 indicates that the alarm received is RADIATION MONITOR
- Control Room indication for Liquid Radwaste Discharge Process Monitor, HB RE-18, is OFF-SCALE LOW.

Which ONE of the following describes the status of the liquid waste discharge for these conditions?

A. The discharge is automatically terminated; Liquid release may be resumed ONLY after HB RE-18 is returned to OPERABLE status.

B. The discharge is automatically terminated; Liquid release may be resumed upon satisfactory completion of actions required by AP 07B-003, Offsite Dose Calculation Manual, even if HB RE-18 remains INOPERABLE.

C. The discharge continues and must be manually terminated; Liquid release may be resumed ONLY after HB RE-18 is returned to OPERABLE status.

D. The discharge continues and must be manually terminated; Liquid release may be resumed upon satisfactory completion of actions required by AP 07B-003, Offsite Dose Calculation Manual, even if HB RE-18 remains INOPERABLE.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because automatic termination does occur as indicated by HI HI rad and the local waste panel alarm (RAD MON). Also, release may NOT continue unless actions described in options B and D are taken.

B. Correct. Automatic termination occurs on HI HI radiation. Release may not continue unless rad monitor is repaired or ODCM actions are taken.

C. Incorrect. Plausible because of same reason as option A, and also because a radiation monitor failure by itself will not close the release path. Applicant must also know that a failure of the rad monitor does not cause a HI HI alarm by itself.

D. Incorrect. Plausible because actions are correct, and also because the applicant must understand what a rad monitor failure does, and that a HI HI radiation alarm will cause automatic valve closure, independent of monitor failure.

Technical Reference(s):
ALR 00-061A Rev 16
ALR 00-061B Rev 16A
ALR 00-061C Rev 17
ALR 00-109F Rev 5
AP 07B-003 Rev 6

(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1406904 R9 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	11
	55.43	

Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Comments:

This is a new question but there are several similar items in our bank. For Wolf Creek, the alarms are plant specific but ODCM actions have been tested at other facilities

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	064	K3.01	
	Importance Rating	3.8		

Knowledge of the effect that a loss or malfunction of the ED/G system will have on the following: Systems controlled by automatic loader

Proposed Question: RO Question # 6

Given the following:

- A Loss of Off-Site power occurred.
- "A" EDG started but its output breaker will NOT close.
- Subsequently, a large steamline break resulted in a Safety Injection.
- All other equipment operated as designed and NO other abnormal conditions have occurred.

Which ONE of the following describes the operation of the Emergency Load Sequencer associated with the failed EDG?

A. Does NOT perform load shed or equipment loading sequence.

B. Load shed sequence is initiated by the Shutdown Sequencer upon receipt of loss of power; LOCA Sequencer sheds additional loads upon receipt of the SI signal; equipment loading sequence is NOT initiated.

C. LOCA sequencer will block the Shutdown Sequence load shed sequence when SI actuates, but load shedding will not occur until the EDG output breaker is closed.

D. Load shed sequence is initiated by the Shutdown Sequencer upon receipt of loss of power; LOCA Sequencer sheds additional loads upon receipt of the SI signal; equipment loading sequence occurs, but NO loads will be energized.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Load shedding is initiated on the UV signal. Plausible because Load Sequence is not initiated until the EDG output breaker is closed.

B. Correct. It is logical to believe that the sequencer operates when power is lost or SI is actuated. However, the actual load sequencing will not occur until the bus is re-energized. Load shedding is performed when the UV signal is received, then again when the SI signal is received.

C. Incorrect. Plausible because the LOCA sequencer does block the operation of the Shutdown Sequencer, but ONLY after EDG output breaker closure.

D. Incorrect. Plausible because the bus will be de-energized, so the fact that equipment will <u>not</u> run is true. The applicant must also understand that the sequences start from EDG output breaker closure.

Technical Reference(s): SY 1406401 Rev 10 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1406401 R5 (As available)

Question Source: Bank # WTSI 55095

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

WTSI 55095
This item was changed enough to be considered significantly modified, but we are calling it a bank item. Not from another NRC exam

Facility: Wolf Creek

Vendor: WEC

Exam Date: 08/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	076 K4.02	
	Importance Rating	2.9	

Knowledge of SWS design feature(s) and/or interlock(s) which provide for the following: Automatic start features associated with SWS pump controls

Proposed Question: RO Question # 7

Given the following:

- Reactor trip and safety injection have occurred.
- All equipment is operating as required.

Which ONE of the following describes the operation of Essential Service Water (ESW) Pumps for this event?

A. BOTH trains of ESW start immediately when the LOCA sequencers are actuated.

B. BOTH trains of ESW start simultaneously after a time delay as programmed by the LOCA sequencers.

C. Train "A" ESW starts after a time delay, and Train "B" ESW starts approximately 5 seconds later.

D. Train "B" ESW starts after a time delay, and Train "A" ESW starts approximately 5 seconds later.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because there is no loss of off-site power. Applicant may assume that since the EDG is not required, sequenced loading is not required.

B. Incorrect. Plausible because the 2 trains are independent, and applicant may assume that the sequencers operate in the same manner for each train.

C. Correct.

D. Incorrect. Plausible because the applicant may know that the time delay is slightly different for each train, therefore making this choice logical.

Technical Reference(s): SY 1408900 Rev 18 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1408900 R5 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 08/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	008	K4.01
	Importance Rating	3.1	

Knowledge of CCWS design feature(s) and/or interlock(s) which provide for the following: Automatic start of standby pump

Proposed Question: RO Question # 8

Given the following:

- The plant is at 100% power.
- "A" and "D" CCW Pumps are operating.
- "B" and "C" CCW Pumps are in AUTO.
- "A" Train CCW is aligned to the Service Loop.
- The following alarm is received and is locked in:

- ALR 00-52B, CCW PUMP A/C PRESS LO

Ten seconds later, which ONE of the following describes the Component Cooling Water (CCW) Pumps that are running?

A. "A" and "D" CCW Pumps ONLY.

B. "B" and "D" CCW pumps ONLY.

C. "A", "C", and "D" CCW Pumps.

D. "B", "C", and "D" CCW Pumps.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because the 'C' CCW pump does auto start after 4 seconds of low pressure. This pump will remain running. There is no indication that "A" CCW Pump tripped so it will remain running.

B. Incorrect. There is no indication that "A" CCW Pump tripped so it will remain running.

C. Correct.

D. Incorrect. Plausible because a LOW FLOW alarm will not provide for auto start, and applicant may confuse the 2 alarms. Also, 'B' CCW Pump would remain in standby because the low pressure alarm is on the opposite train.

Technical Reference(s): SY 1400800, Rev 19 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1400800 R7 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Facility:	Wolf Creek
Vendor:	WEC
Exam Date:	8/2009
Exam Type:	R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	005	K5.09	
	Importance Rating	3.2		

Knowledge of the operational implications of the following concepts as they apply the RHRS: Dilution and boration considerations

Proposed Question: RO Question # 9

Given the following:

- The plant is in Mode 4
- Plant cooldown is in progress in accordance with GEN 00-006, Hot Standby to Cold Shutdown.
- Refueling activities are planned.

Which ONE of the following describes the **minimum** requirement for RHR system boron concentration prior to placing RHR in operation for RCS cooldown in accordance with SYS EJ-120, Startup of a Residual Heat Removal Train?

Greater than or equal to the ...

A. current RWST boron concentration.

B. current RCS boron concentration.

C. minimum RCS boron concentration required for Hot Standby.

D. minimum RWST boron concentration allowed by Technical Specifications.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Shutdown margin for current plant conditions or Cold Shutdown, not RWST boron concentration, is limiting factor. Plausible because RWST boron concentration is an ITS LCO.

B. Correct. Prevents inadvertent dilution.

C. Incorrect. Plausible because the plant is currently in Hot Shutdown (Mode 4) However, requirement is current RCS concentration or Cold Shutdown concentration.

D. Incorrect. Plausible because RCS boron concentration requirement will rise as the RCS is cooled down to Mode 6. Applicant may believe that RCS and RHR must be borated to refueling concentration, and the Refueling cavity will be filled with RWST.

Technical Reference(s): SYS EJ-120, Rev 52 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1300500 R6 (As available)

Question Source: Bank # WTSI 52533

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Watts Bar 2004

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 5

55.43

Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limiitations and reasons for these operating characteristics.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	061	K5.03
	Importance Rating	2.6	

Knowledge of the operational implications of the following concepts as the apply to the AFW: Pump head effects when control valve is shut

Proposed Question: RO Question # 10

Given the following:

- All three AFW pumps are operating supplying feed to steam generators.
- AFW flow to each SG is approximately 300 GPM.

Which ONE of the following describes the affect on pump discharge pressure for each of the pumps as AL HV-6, TDAFW supply to SG "D", is throttled CLOSED?

	MDAFW "A" Discharge Pressure	MDAFW "B" Discharge Pressure	TDAFW Discharge Pressure
A.	unaffected	decreases	increases
B.	increases	increases	decreases
C.	decreases	unaffected	increases
D.	unaffected	unaffected	decreases

Proposed Answer: A

Explanation (Optional):

A. Correct. Flow from TDAFW will be lowered. MDAFW "B" discharge valve (smart valve MOV) will throttle open to maintain 300 GPM to the SG. The position of the valves on the discharge piping will cause pump discharge pressure from TDAFW Pump to rise as the resistance to flow rises as the discharge valve closes.

B. Incorrect. Plausible because it is logical to assume both MDAFW Pumps are affected if TDAFW Pump discharge is closed. In this case, closing the valve raises TDAFW discharge pressure. The effect is the opposite of actual.

C. Incorrect. Plausible because this is similar to correct answer, except for the fact that MDAFW Pump "B" supplies SG "D". The applicant may confuse AFW alignment.

D. Incorrect. Plausible because it is logical to assume both MDAFW Pumps are affected if TDAFW Pump discharge is affected. In this case, the discharge pressure effects would be correct if the valve being operated was on a discharge line common to all 3 pumps, and if the valve was opened to raise flow.

Technical Reference(s): SY 1406100 Rev 18 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1406100 R3, R5 (As available)

Question Source: Bank # WTSI 52706

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: VC Summer 2005

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

WTSI 52706

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	004	K6.13	
	Importance Rating	3.1		

Knowledge of the effect of a loss or malfunction on the following CVCS components: Purpose and function of the boration/dilution batch controller

Proposed Question: RO Question # 11

Given the following:

- The plant is operating at 100% power.
- All controls are in AUTO.
- VCT level reaches the AUTO makeup setpoint, and RCS makeup is automatically initiated.
- The air supply line to BG FCV-111A, Reactor Makeup Water to Boric Acid Blending Tee, is severed while the makeup is in progress.

Which ONE of the following describes the effect on the RCS and the RCS makeup for this condition?

RCS makeup could potentially result in an RCS...

A. dilution; the makeup is automatically terminated when ALR 00-041D, Boric Acid Flow Deviation alarm, is received.

B. dilution; the makeup is automatically terminated when ALR 00-041E, Total Makeup Flow Deviation alarm, is received.

C. boration; the makeup is automatically terminated when a ALR 00-041D, Boric Acid Flow Deviation alarm, is received.

D. boration; the makeup is automatically terminated when ALR 00-041E, Total Makeup Flow Deviation alarm, is received.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. BG FCV-111A is the total flow control valve. In automatic, it will regulate flow to 120 GPM for makeup, regardless of the Boric Acid controller. If air is lost to the total flow valve, it will fail closed. Boric acid flow will still be controlled by its own controller. Thgerefore, a total flow devaiation will occur, Boric acid flow will initially be unchanged, resulting in potential for boration for up to 30 seconds. BG FCV-110A for boric acid flow does fail open, and applicant may confuse the 2 valves.

B. Incorrect. Plausibility same as option A for reason dilution will not occur, and second part of option is correct for conditions listed.

C. Incorrect. First part is correct, and plausible for same reason as option A for why the boric acid flow deviation alarm will not cause the shutdown.

D. Correct.

Technical Reference(s): SY 1300401 Rev 16 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1300401 R2 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

Design, components, and function of reactivity control mechanisms and instrumentation.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 08/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	013	K6.01
	Importance Rating	2.7	

Knowledge of the effect of a loss or malfunction on the following will have on the ESFAS:
Sensors and detectors

Proposed Question: RO Question # 12

Given the following:

- A Containment Pressure that feeds both SI and Containment Spray fails LOW.
- In accordance with OFN SB-008, Instrument Malfunctions, the Containment Pressure HIGH-1 and HIGH-2 Bistables are placed in TRIP.
- The Containment Pressure HIGH-3 Bistable is placed in BYPASS.

Which ONE of the following identifies the correct ESF actuation logic for the **remaining** Containment pressure channels?

A. Safety Injection - 1/2; Containment Spray - 1/3

B. Safety Injection - 1/2; Containment Spray - 2/3

C. Safety Injection - 1/3; Containment Spray - 1/3

D. Safety Injection - 1/3; Containment Spray - 2/3

Proposed Answer: B

Explanation (Optional):

A. Incorrect. SI is normally 2 of 3 logic for Containment Pressure and CS is 2 of 4. When a protection channel that feeds both actuations is removed from service, bistables are tripped in all cases except for the AUTO CS actuation. (Placed in BYPASS) Thus, AUTO SI will occur if either of the two remaining bistables trip, but AUTO CS still needs two of the three remaining channels to trip.

B. Correct. See distractor explanations and attached references.

C. Incorrect. SI is 1 of 2. CS is 2 of 3. Plausible because PZR Pressure SI actuation is 2 of 4 logic, and the applicant may confuse the logic with the 2 of 3 logic provided by High CTMT pressure SI actuation. The applicant may also believe that all bistables, including CS actuation, are tripped when a channel has failed.

D. Incorrect. SI is 1 of 2 after removing a channel from service, because Containment Pressure High-1 provides 3 channel inputs to SI actuation. Plausible because Low PZR Pressure SI input would be 1 of 3.

Technical Reference(s): SY 1301301 Rev 9 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1301301 R3 (As available)

Question Source: Bank # WTSI 53947

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: BVPS-2 2002

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Similar items have been on other exams

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	012	A1.01
	Importance Rating	2.9	

Ability to predict and/or monitor Changes in parameters (to prevent exceeding design limits) associated with operating the RPS controls including: Trip setpoint adjustment

Proposed Question: RO Question # 13

The plant is at 80% power during a plant startup.

Assuming that RCS and flux distribution parameters remain on program/target, as power is raised to 100%, how will the over-temperature (OT) and over-power (OP) differential temperature (DT) Reactor Protection setpoints change?

	OTDT setpoint	OPDT setpoint
A.	increase	stay the same
B.	decrease	decrease
C.	decrease	stay the same
D.	stay the same	increase

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because the OPDT setpoint does remain the same, and the applicant may confuse the fact that the OTDT setpoint decreases with the term 'increase' meaning the actual value is closer to setpoint.

B. Incorrect. The OTDT setpoint does decrease. The OPDT setpoint never increases from its nominal value. It will, however decrease if T-avg deviates above its nominal

100% power program value. Since T-avg at 80% power is less than 100% power, the OPDT setpoint will be at its nominal full power value and thus, will not change from 80 to 100% power assuming T-avg stays on program.

C. Correct. Since T-avg at 80% power is less than 100% power, the OPDT setpoint will be at its nominal full power value and thus, will not change from 80 to 100% power assuming T-avg stays on program. The OTDT setpoint, on the other hand can increase or decrease from its nominal value. Since program T-avg will increase approximately 5 more degrees, the trip setpoint will become more limiting, decreasing to its nominal full power value.

D. Incorrect. OTDT decreases as referenced above in C. The OPDT setpoint never increases from its nominal value. It will, however decrease if T-avg deviates above its nominal 100% power program value

Technical Reference(s): SY 1301200 Rev 6 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1301200 R3 (As available)

Question Source: Bank # WTSI 59651

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Diablo Canyon 2007

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

Design, components, and function of reactivity control mechanisms and instrumentation.

Comments:

Diablo Canyon 2007 WTSI 59651

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	063	A1.01	
	Importance Rating	2.5		

Ability to predict and/or monitor changes in parameters associated with operating the dc electrical system controls including: Battery capacity as it is affected by discharge rate

Proposed Question: RO Question # 14

Given the following:

- A loss of all AC power has occurred.
- The crew is performing actions of EMG C-0, Loss of All AC Power.
- EMG C-0, Attachment C, DC Load Shedding, is performed.

Which ONE of the following describes the design capacity of the Class 1E batteries, and the effect of performing EMG C-0, Attachment C?

A. 2 hours; ensures that the 2 hour requirement will be met.

B. 2 hours; extends the life of the batteries up to 4 hours.

C. 4 hours; ensures that the 4 hour requirement will be met.

D. 4 hours; extends the life of the batteries beyond 4 hours.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because the ITS action time for DC Bus is 2 hours. Applicant must also know the purpose of load shedding and the design of the batteries to answer correctly.

B. Incorrect. Plausible for same reason as option A, and the actual 4 hour time is referred to.

C. Incorrect. Plausible because the design capacity is correct but the applicant may choose this option if they do not know the basis for performing the attachment.

D. Correct.

Technical Reference(s): EMG C-0 Rev 18, Step 27 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1506300 R2 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	059	A2.01	
	Importance Rating	3.4		

Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Feedwater actuation of AFW system

Proposed Question: RO Question # 15

Given the following:

- Plant startup is in progress.
- Reactor power is currently 29%.
- "A" Main Feed Pump turbine is in the tripped position, but available for service.

- The following event occurs:

- "B" Main Feed Pump turbine trips on low oil pressure.
- The Main Turbine automatically trips.
- Steam Generator Levels are as follows:

- "A" S/G 28% NR and decreasing slowly
- "B" S/G 29% NR and decreasing slowly
- "C" S/G 27% NR and decreasing slowly
- "D" S/G 28% NR and decreasing slowly

Which ONE of the following describes the current status of the AFW system; and how will the plant be stabilized?

A. ONLY Motor Driven AFW pumps have auto started; Control Rods, AFW, and Steam Dump are operated as required to stabilize the reactor.

B. AFW pumps must be MANUALLY started; the reactor must be manually tripped due to exceeding a Reactor Protection trip setpoint.

C. ONLY Motor Driven AFW pumps have auto started; the reactor must be manually tripped due to exceeding a Reactor Protection trip setpoint.

D. AFW pumps must be MANUALLY started; Control Rods, AFW, and Steam Dump are operated as required to stabilize the reactor.

Proposed Answer: A

Explanation (Optional):

A. Correct. Although SG LO-LO setpoint has not been reached, the MDAFW pumps have started due to BOTH Main Feed Pumps being tripped. Since below P-9 and no reactor trip, the unit is stabilized using OFN MA-001, Load Rejection/Turbine Trip.

B. Incorrect. Plausible because SG level has not reached the AFW auto start setpoint at this time. RPS setpoint has not been exceeded, but it is logical to assume that the reactor would be tripped on loss of all feed, and SG levels are approaching an RPS setpoint.

C. Incorrect. Plausible because the pumps have auto started, and also because SG levels are approaching an RPS setpoint.

D. Incorrect. Plausible because the SG level setpoint for AFW actuation hasn't been reached. Actions are also correct in accordance with OFN MA-001, Load Rejection/Turbine Trip.

Technical Reference(s): SY 1406100 Rev 18 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1406100 R2
LO 1732411 R3 (As available)

Question Source: Bank # WTSI 58546

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: McGuire 2007

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	10
	55.43	

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	022	A2.04
	Importance Rating	2.9	

Ability to (a) predict the impacts of the following malfunctions or operations on the CCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of service water

Proposed Question: RO Question # 16

Given the following:

- The plant was at 100% power.
- All Containment Coolers were in service, operating in fast speed.
- A momentary loss of off-site power occurred.
- The reactor is tripped.
- Off-Site Power is now available.
- "A" ESW Pump Discharge pressure and motor amps are experiencing large fluctuations.
- Containment temperature is 118°F and rising at 0.5°F per minute.
- The crew is referring to OFN EF-033, Loss of Essential Service Water.

Which ONE of the following describes (1) the effect on the Containment temperature Technical Specification, and (2) the action required in accordance with OFN EF-033?

(1) LCO 3.6.5, Containment Air Temperature...

A. has been exceeded;
(2) align Train "A" ESW to Service Water.

B. has been exceeded;
(2) Locally align NB02 to PA02 to SL Busses.

C. will be exceeded within five (5) minutes;
(2) Align Train "A" ESW to Service Water.

D. will be exceeded within five (5) minutes;
(2) Locally align NB02 to PA02 to SL Busses.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. TS limit on Containment temperature is 120°F. Plausible because the temperature given is above normal and rising. Action is correct for conditions given.

B. Incorrect. TS limit on Containment temperature is 120°F. Plausible because the temperature given is above normal and rising. Action given is incorrect but in the same RNO as the action required. This would be performed if neither ESW pump was operating and off-site power still lost.

C. Correct.

D. Incorrect. The time for exceeding the TS LCO is correct. The action is plausible because it is contained in the same procedure step as the required action, but incorrect because it is not required unless both ESW pumps are off and power is lost.

Technical Reference(s): OFN EF-033, Rev 10, step 2 RNO 2b; Attachment A (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732443 R3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	039	A3.02	
	Importance Rating	3.1		

Ability to monitor automatic operation of the MRSS, including: Isolation of the MRSS

Proposed Question: RO Question # 17

Given the following:

- A plant cooldown is in progress in accordance with GEN 00-006, Hot Standby to Cold Shutdown.
- RCS pressure is 1850 psig.
- RCS temperature is 505°F.
- All required procedural actions have been taken for the cooldown in accordance with plant procedures.

An event occurs:

- RCS pressure is 1700 psig and lowering at 10 psi per second.
- SG pressures are 700 psig and lowering at 25 psi per second.
- Containment pressure is 2.8 psig and rising.

Assuming all equipment operates as designed, which ONE of the following describes the ESF actuation status?

A. Safety Injection has occurred; Main Steam Line Isolation has occurred.

B. Safety Injection has NOT occurred; Main Steam Line Isolation has occurred.

C. Safety Injection has occurred; Main Steam Line Isolation has NOT occurred.

D. Safety Injection has NOT occurred; Main Steam Isolation has NOT occurred.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because the applicant may not know that Low PZR Pressure SI is blocked above the current pressure, and may confuse when the hight rate MSLI is active.

B. Correct.

C. Incorrect. Plausible because the applicant may fail to consider that Low PZR pressure SI is blocked, as this would be all of the plant response if PZR pressure was above P-11.

D. Incorrect. Plausible because the applicant may not know the rate at which steam pressure will initiate a MSLI, even if they know that Low PZR pressure is blocked, particularly because the normal MSLI setpoints for steam pressure and Containment pressure have not been reached.

Technical Reference(s): SY 1301301 Rev 9 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1301301 R3 (As available)

Question Source: Bank # WTSI 57052

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Sequoyah 2007

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	026	A3.01	
	Importance Rating	4.3		

Ability to monitor automatic operation of the CSS, including: Pump starts and correct MOV positioning

Proposed Question: RO Question # 18

Given the following:

- A LOCA has occurred.
- All equipment has functioned as designed.
- Containment pressure indicates the following:

- Channel I 25.9 psig
- Channel II 26.7 psig
- Channel III 27.1 psig
- Channel IV 27.4 psig

Which ONE of the following describes the status of the Containment Spray System?

A. Containment Spray Pump "A" OFF;
EN HV-6, Containment Spray Discharge Isolation Valve, CLOSED.

B. Containment Spray Pump "B" OFF;
EN HV-7, Containment Recirc Sump Isolation Valve, CLOSED.

C. Containment Spray Pump "A" ON;
EN HV-6, Containment Spray Discharge Isolation Valve, OPEN.

D. Containment Spray Pump "B" ON:
EN HV-7, Containment Recirc Sump Isolation Valve, OPEN.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because the applicant may believe that with Channels I and II below the setpoint, Train "A" may not have actuated yet.

B. Incorrect. Plausible because EN HV-7 is closed, and receives a close signal on CISA. Applicant may also see channels I and II below CISB setpoint and believe that has not actuated.

C. Correct. two out of 4 channels above setpoint will actuate bot trains of spray.

D. Incorrect. Plausible because the pump will be on, but EN HV-7 would have already received a close signal.

Technical Reference(s): SY 1302600 Rev 12 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1302600 R3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	062	A4.01	
	Importance Rating	3.3		

Ability to manually operate and/or monitor in the control room: All breakers (including available switchyard)

Proposed Question: RO Question # 19

Given the following:

- A Reactor Trip has occurred from 100% power.
- The crew has entered EMG E-0, Reactor Trip or Safety Injection.
- Ten seconds after the trip, the following indications are observed:

- RCS pressure is 1760 psig and lowering.
- Containment pressure is 4.5 psig and rising.
- RCS temperature is 542°F and lowering slowly.
- Main Generator output breakers indicate closed.

Which ONE of the following describes the event in progress, and the actions that will be required for the current plant conditions?

A. Main Turbine failed to trip; manually trip the turbine, verify generator output breakers and exciter field breaker open after approximately 30 seconds, and verify that plant parameters stabilize while continuing in EMG E-0.

B. Main Turbine failed to trip; immediately trip the turbine, open the generator output breakers and exciter field breaker, and verify SI has actuated.

C. A LOCA has occurred; verify SI has actuated and immediately open the generator output breakers and exciter field breaker.

D. A LOCA has occurred; verify SI has actuated and verify that the generator output breakers and exciter field breaker open within 30 seconds after the trip.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because generator output breakers being closed (but for >30 seconds) after a trip can be a sign of turbine trip circuitry failure. Additionally, RCS temperature is lowering but not fast enough to indicate a turbine trip failure. Containment pressure indicates a LOCA has occurred.

B. Incorrect. Plausible for same reasons as A, except that action is different. Turbine trip would be required if that was the failure, but generator output breakers open on a time delay to assist coastdown.

C. Incorrect. Failure is correct, and action is plausible because generator load is indicated, but applicant must know that generator output breakers do not open until after a time delay. If breakers do not open, then they will be manually opened

D. Correct. See explanations above.

Technical Reference(s): EMG E-0, Rev 20 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732313 R3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Facility:	Wolf Creek
Vendor:	WEC
Exam Date:	08/2009
Exam Type:	R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	078	A4.01	
	Importance Rating	3.1		

Ability to manually operate and/or monitor in the control room: Pressure gauges

Proposed Question: RO Question # 20

Given the following:

- The plant is at 100% power.
- An Instrument Air leak is occurring, resulting in a lowering air pressure.
- Instrument Air Header Pressure indication KA PI-40, Air Dryer Filter Discharge Pressure Indicator, indicates 85 psig and is stabilizing.
- Computer Point KAP0010 indicates 86 psig and is stabilizing.

Which ONE of the following describes the status of the Instrument Air System and the operation of plant equipment?

A. Service Air Isolation Valve, KA PV-11, is CLOSED; Main Feedwater Regulating Valves are fully controllable using the Instrument Air System.

B. Service Air Isolation Valve, KA PV-11, is CLOSED; Main Feedwater Regulating Valves are fully controllable using the Backup Nitrogen System.

C. Pressurizer Spray Valves are UNAVAILABLE to perform their function; Main Feedwater Regulating Valves are fully controllable using the Instrument Air System.

D. Pressurizer Spray Valves are UNAVAILABLE to perform their function; Main Feedwater Regulating Valves are fully controllable using the Backup Nitrogen System.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because KA PV-11 is closed at 110 psig decreasing, and pressure is slightly below 90 psig, when Backup N2 will allow control of Feedwater Regulating Valves and Auxiliary Feedwater Control Valves.

B. Correct.

C. Incorrect. Pressurizer Spray valves will operate until approximately 70 psig. At 85-90 psig, they are still available for use.

D. Incorrect. Pressurizer Spray valves will operate until approximately 70 psig. At 85-90 psig, they are still available for use. Backup N2 will allow control of Feedwater Regulating Valves and Auxiliary Feedwater Control Valves below 90 psig.

Technical Reference(s): OFN KA-019 Rev 13 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY1407800 R5 (As available)

Question Source: Bank #

Modified Bank # WTSI 57803 (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

WTSI 57803 Modified from Wolf Creek 2007 item (Attached)

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	007	2.4.50
	Importance Rating	4.2	

Emergency Procedures / Plan: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

Proposed Question: RO Question # 21

Given the following:

- The plant is at 100% power when Letdown Relief Valve BG V-8117 lifted.
- The crew subsequently corrected the condition causing the relief valve to lift.

- The following Pressurizer Relief Tank (PRT) parameters exist:

- PRT pressure has increased from 2.0 psig to 4.5 psig.
- PRT temperature has increased from 90°F to 138°F.
- PRT level has increased from 68% to 78%.

Which ONE of the following describes the annunciator that will alarm and the preferred action to return the PRT parameters to nominal values?

A. Annunciator 00-034D, PRT TEMP HI, will be in alarm. The PRT should be cooled down by spraying cool reactor makeup water into the PRT.

B. Annunciator 00-34F, PRT LVL HI-LO, will be in alarm. The PRT level should be reduced by draining to the Reactor Coolant Drain Tank (RCDT).

C. Annunciator 00-034D, PRT TEMP HI, will be in alarm. The PRT should be cooled down by recirculating the PRT water through the RCDT heat exchanger.

D. Annunciator 00-034E, PRT PRESS HI, will be in alarm. The PRT pressure should be reduced by venting the PRT to the Gas Decay Tanks (GDT).

Proposed Answer: C

Explanation (Optional):

A. Incorrect. The alarm is correct because it comes in at 120°F. Action is incorrect because level is already elevated. Plausible because spraying with cooler water would cool the tank

B. Incorrect. HI-LO level alarm is received at 88%. Plausible because level is elevated at 77%, and the action is correct if the level alarm is in.

C. Correct. If only temperature is above the alarm setpoint, then recirculation through the RCDT heat exchanger will be the action required

D. Incorrect. High Pressure alarm is not received until 6 psig. Plausible because if there was a high pressure alarm, the action would be correct

Technical Reference(s): SYS BB-202, Rev 40 section 6.5 (Attach if not previously provided)

ALR 00-034D Rev 9
ALR 00-034E Rev 9
ALR 00-034F Rev 7

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1300200 R2 (As available)

Question Source: Bank # WTSI 55547

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Farley 2006

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

WTSI 55547

Matches KA because all actions directed by alarm response procedures refer to the SYS procedure

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	063	2.4.6
	Importance Rating	3.7	

Emergency Procedures / Plan: Knowledge of EOP mitigation strategies.

Proposed Question: RO Question # 22

Given the following:

- The plant is operating at 100% power.
- The following alarm is received:

- 00-028C, NK04 TROUBLE

- NK04 voltage indication is 0 volts.

Which ONE of the following describes the effect on the plant and the actions required?

A. Reactor will trip; stabilize the plant using EMG E-0, Reactor Trip or Safety Injection, and refer to the appropriate OFN procedure to locally operate the Turbine Driven AFW Pump, if required.

B. Reactor will trip; stabilize the plant using EMG E-0, Reactor Trip or Safety Injection, and refer to the appropriate OFN procedure to place EDG "B" in LOCAL/MANUAL due to loss of field flash capability.

C. The reactor will remain at power; EDG "B" must be placed in LOCAL/MANUAL in accordance with the appropriate OFN procedure due to loss of field flash capability.

D. The reactor will remain at power; Turbine Driven AFW Pump will start due to loss of power and must be shut down manually using the appropriate OFN procedure.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Loss of NK04 will cause loss of field flash for EDG "B". Plausible because TDAFW is affected by loss of NK02. The reactor will trip because feedwater isolation occurs, and at 100% power, SG LO-LO level will cause a trip.

B. Correct.

C. Incorrect. Plausible because there is no direct reactor trip, and loss of control power to the reactor trip breaker does not cause it to open. The applicant must determine that feedwater isolation will result in a reactor trip at this power level. Actions for EDG "B" are correct.

D. Incorrect. Plausible because there is no direct reactor trip, and loss of control power to the reactor trip breaker does not cause it to open. The applicant must determine that feedwater isolation will result in a reactor trip at this power level. Actions for TDAFW are incorrect, since a loss of NK04 does not affect TDAFW, but loss of NK02 does.

Technical Reference(s): OFN NK-020, Rev 7, Attachment D (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732430 R3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 08/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	078 A3.01	
	Importance Rating	3.1	

Ability to monitor automatic operation of the IAS, including: Air pressure

Proposed Question: RO Question # 23

Given the following:

- The plant is operating at 100% power.
- Instrument Air System is in a normal alignment.
- Instrument Air System Sequence Selector Switch, KA HSS-310, is in position C-A-B.
- Instrument Air Pressure is 119 psig and slowly lowering due to normal system demand.

Which ONE of the following describes the LOWEST Instrument Air pressure that will be reached prior to recovering, and the number of air compressors that will be supplying the system at that pressure?

A. 116 psig; ONE Air Compressor operating.

B. 116 psig; TWO Air Compressors operating.

C. 112 psig; ONE Air Compressor operating.

D. 112 psig; TWO Air Compressors operating.

Proposed Answer: A

Explanation (Optional):

A. Correct. The LEAD air compressor will cycle between 116 and 125 psig

B. Incorrect. Plausible because lag and standby compressor starts are associated with lowering air pressure. In this case, the applicant must be aware of setpoints, because number of compressors in operation is not given in the stem

C. Incorrect. 112 psig is the setpoint for start of the standby compressor. Plausible because it is a valid system setpoint that will turn pressure around if dropping, because a compressor auto start is occurring

D. Incorrect. 112 psig is the setpoint for start of the standby compressor. Plausible because it is a valid system setpoint that will turn pressure around if dropping, because a compressor auto start is occurring, and additionally because it would be a second compressor running at this setpoint

Technical Reference(s): OFN KA-019 Rev 13
SY1407800 Rev 15 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1407800 R5 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 08/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	076	A1.02	
	Importance Rating	2.6		

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SWS controls including: Reactor and turbine building closed cooling water temperatures.

Proposed Question: RO Question # 24

Given the following:

- The plant is at 100% power.
- "A" Circ Water Bay tagged out for diver inspection.
- Service Water Pump "B" is running.
- Service Water Pump "C" has tripped.
- Service Water Header Pressure is 50 psig and lowering slowly.

Which ONE of the following describes the effect on the unit, and action that may be required?

A. Generator Hydrogen Cold Gas temperatures will rise; throttle open the Generator H2 cooler supply valves.

B. Condensate Pump seal temperature will rise; place a second Closed Cooling Water Pump in service.

C. Generator Hydrogen Cold Gas temperatures will rise; Reduce generator load and VAR load as necessary to maintain Cold Gas temperatures within limits.

D. Steam Packing Exhauster temperatures will rise; trip the main turbine.

Proposed Answer: C

Explanation (Optional):

A. Incorrect because outlets are throttle valves and changing position would rob flow from other vital components such as EHC. Plausible because this action could be performed to lower temperature for the component.

B. Incorrect because OFN AF-025 does not consider these conditions for limiting plant operation. Plausible because this could help cooling if Service Water was adequate

C. Correct

D. Incorrect because OFN AF-025 does not consider these conditions for limiting plant operation. Plausible because steam seal temperature rising is detrimental to the turbine

Technical Reference(s): OFN AF-025, Rev 27 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1407600 R5 (As available)

Question Source: Bank # WTSI 57790

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Wolf Creek 2007

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Facility:	Wolf Creek
Vendor:	WEC
Exam Date:	08/2009
Exam Type:	R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	005	K4.06	
	Importance Rating	2.7		

Knowledge of RHRS design feature(s) and/or interlock(s) which provide or the following:
Function of RHR pump miniflow recirculation

Proposed Question: RO Question # 25

RHR Pump "A" is started on recirculation in preparation for placing RHR in service in Mode 4.

Which ONE of the following describes the operation of minimum flow valve EJ FCV-610 when the pump is started on recirculation?

A. OPEN prior to pump start; CLOSES when pump flow reaches a setpoint of 816 GPM.

B. OPEN prior to pump start; CLOSES when pump flow reaches a setpoint of 1650 GPM.

C. CLOSED prior to pump start; OPENS when pump is running and flow is below 816 GPM; CLOSES when pump flow reaches a setpoint of 816 GPM.

D. CLOSED prior to pump start; OPENS when pump is running and flow is below 1650 GPM; CLOSES when pump flow reaches a setpoint of 1650 GPM.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because 816 GPM is the setpoint for valve opening as flow is lowered, and also because the valve is open when the pump is shut off. This valve is ONLY interlocked with a flow switch, and not pump start circuitry

B. Correct.

C. Incorrect. Would be correct if valve was interlocked with pump start circuitry, and plausible because plant systems have alignments where 'low flow' alarms are only active when a pump motor breaker is closed.

D. Incorrect. Would be correct if valve was interlocked with pump start circuitry, and plausible because plant systems have alignments where 'low flow' alarms are only active when a pump motor breaker is closed. In this case, the setpoint is correct.

Technical Reference(s): SY 1300500 Rev 15 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1300500 R2, R6 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Facility:	Wolf Creek
Vendor:	WEC
Exam Date:	8/2009
Exam Type:	R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	026	2.1.20	
	Importance Rating	4.6		

Conduct of Operations: Ability to interpret and execute procedure steps.

Proposed Question: RO Question # 26

Given the following:

- A LOCA has occurred.
- Containment pressure is 28 psig.
- The following annunciators are lit:

- 00-059A, CSAS
- 00-059B, CISB

- ESFAS Status Panel for CSAS and CISB white status lights are DARK on BOTH red and yellow trains.

In accordance with EMG E-0, Reactor Trip or Safety Injection, which ONE of the following describes the MINIMUM action required for the current plant conditions?

A. Rotate two Spray Actuation handswitches to ACTUATE one at a time.

B. Rotate two Spray Actuation handswitches to ACTUATE simultaneously.

C. Rotate two Spray Actuation handswitches to ACTUATE simultaneously. Trip all RCPs.

D. Rotate two Spray Actuation handswitches to ACTUATE one at a time. Trip all RCPs.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Rotating handswitches one at a time will not pick up the logic for containment spray actuation. Additionally, RCPs must be tripped as part of the minimum actions.

B. Incorrect. Action to initiate spray is correct, but RCPs must also be tripped.

C. Correct.

D. Incorrect. Rotating handswitches one at a time will not pick up the logic for containment spray actuation.

Technical Reference(s): EMG E-0, Rev 21, Attachment F (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732313 R3 (As available)

Question Source: Bank #

Modified Bank # WTSI 57804 (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

This was from 2007 NRC but stem was (significantly) editorially modified to allow for interpretation of conditions and answer options were modified.
WTSI 57804

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	012	A2.05	
	Importance Rating	3.1		

Ability to (a) predict the impacts of the following malfunctions or operations on the RPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Faulty or erratic operation of detectors and function generators

Proposed Question: RO Question # 27

Given the following:

- The plant is operating at 100% power.
- Rod Control is in Automatic.
- Power Range Channel N-41 begins rapidly oscillating between 90% and 105% power.
- Power Range N-41 Delta Flux Channel is oscillating between 0 and (-)20.
- All other power range and delta flux indications are stable.
- Main Generator Output is stable.
- Tavg and Tref are matched.

Which ONE of the following describes the effect on the rod control system, and the action that is required?

Automatic rod movement...

A. may occur because the High Flux Rod Stop automatically clears below the setpoint; bypass channel N-41 and trip associated bistables; I&C may attempt troubleshooting prior to removing control power fuses.

B. may occur because the High Flux Rod Stop automatically clears below the setpoint; bypass channel N-41; I&C may NOT attempt troubleshooting prior to tripping associated bistables.

C. will not occur because the High Flux Rod Stop must be manually reset when received; bypass channel N-41 and trip associated bistables; I&C may attempt troubleshooting prior to removing control power fuses.

D. will <u>not</u> occur because the High Flux Rod Stop must be manually reset when received; bypass channel N-41; I&C may <u>NOT</u> attempt troubleshooting prior to tripping associated bistables.

Proposed Answer: A

Explanation (Optional):

A. Correct. The only trip that must be reset after receipt is the high flux rate trip. If power is below the rod stop setpoint, rod movement will be available. Also, bistables must be tripped prior to troubleshooting by I&C.

B. Incorrect. First part is correct. second half is plausible because I&C may be performing the bistable trip as well as the troubleshooting, and it is logical to restrict troubleshooting activities prior to placing an RPS function is a safe condition.

C. Incorrect. First part is incorrect but plausible because high flux rate trips do require manual reset on the NI cabinet. Second part is correct.

D. Incorrect. Plausibility described in distractors above.

Technical Reference(s): OFN SB-008, Att R , Rev 24B (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732418 R3, R4 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	004	A4.18	
	Importance Rating	4.3		

Ability to manually operate and/or monitor in the control room: Emergency borate valve

Proposed Question: RO Question # 28

Given the following:

- A reactor trip has occurred.
- The crew is performing actions of EMG ES-02, Reactor Trip Response.
- Two Control Bank D rods are stuck at approximately 20 steps withdrawn.
- All other plant systems have responded as designed.
- All Critical Safety Function Status Trees are GREEN.

Which ONE of the following describes the FIRST action required in response to the current plant conditions in accordance with EMG ES-02?

A. Establish RCS boration at a minimum flow rate of 30 GPM from RWST using either RWST to CCP suction valve BN HIS-112D or BN HIS-112E.

B. Establish RCS boration at a minimum flow rate of 90 GPM from RWST using either RWST to CCP suction valve BN HIS-112D or BN HIS-112E.

C. Establish RCS boration at a minimum flow rate of 30 GPM from Boric Acid Tanks using Emergency Borate to Charging Pump Suction Valve BG HIS-8104.

D. Establish RCS boration at a minimum flow rate of 90 GPM from Boric Acid Tanks using Emergency Borate to Charging Pump Suction Valve BG HIS-8104.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Use of RWST valves is a contingency if the normal Charging flowpath is not available via BG HIS-8104. 30 GPM is the correct flowrate for flow through BG HIS-8104.

B. Incorrect. This action would be required at this flowrate if BG HIS-8104 was not available at 30 GPM. Also plausible because plant conditions do not appear to be severe enough for emergency boration with all CSFs green, so applicant may believe RWST boration is sufficient.

C. Correct.

D. Incorrect. Plausible because correct valve is used but minimum flowrate is 30 GPM. Only 90 GPM through RWST, which is at a lower boron concentration than the BATs.

Technical Reference(s): EMG ES-02, Rev 18 step 9 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732419 R3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 08/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	014	A4.01	
	Importance Rating	3.3		

Ability to manually operate and/or monitor in the control room: Rod selection control

Proposed Question: RO Question # 29

Given the following:

- The plant is operating at 88% power.
- Rod Control is in MANUAL.
- Control Bank D rods are at 200 steps.
- All Tavg channels are approximately 5.5°F higher than Tref.

Which ONE of the following describes the resulting rod control operation if the Rod Control System Mode Selector Switch is placed in AUTO prior to matching Tavg and Tref?

A. Step in at 8 SPM

B. Step in at 48 SPM

C. Step in at 64 SPM

D. Step in at 72 SPM

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Rod speed would be 8 steps per minute with a temperature error of 1.5°F to 3°F.

B. Incorrect. Rod speed increases from an error of 3°F to 5°F. Speed will rise linearly from 6 steps per minute to 72 steps per minute. 48 SPM is plausible because it is the speed at which Control Bank rods will move if Rod Control is in MANUAL

C. Incorrect. Rod speed increases from an error of 3°F to 5°F. Speed will rise linearly from 6 steps per minute to 72 steps per minute. 64 SPM is plausible because it is the speed at which Shutdown Bank rods will move if Rod Control is in MANUAL

D. Correct. For temperature errors above 5°F, Control Bank speed is 72 SPM if rod control is in AUTO

Technical Reference(s): SY 1300100 Rev 14 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1300100 R7 (As available)

Question Source: Bank # WTSI 57902

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: BVPS-1 2007

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

Design, components, and function of reactivity control mechanisms and instrumentation.

Comments:

Facility:	Wolf Creek
Vendor:	WEC
Exam Date:	08/2009
Exam Type:	R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	015	K6.02	
	Importance Rating	2.6		

Knowledge of the effect of a loss or malfunction on the following will have on the NIS: Discriminator/compensation circuits

Proposed Question: RO Question # 30

Given the following:

- The plant was initially at 95% power.
- The reactor has tripped.
- Compensating voltage on N-35, Intermediate Range NI, is failed LOW.

Which ONE of the following describes the response of Intermediate Range N-35 and the Reactor Protection System for this condition?

A. Indicates LOW; the Source Range HI FLUX TRIP will be reinstated when P-6 is satisfied by the other IR channel (N-36).

B. Indicates LOW; causing P-6 to reinstate the Source Range HI FLUX TRIP prematurely.

C. Indicates HIGH; preventing P-6 from automatically reinstating the Source Range HI FLUX TRIP.

D. Indicates HIGH; the Source Range HI FLUX TRIP will be reinstated by P-6 from the other IR channel (N-36).

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because the applicant may confuse overcompensation from undercompensation. This would be correct if the compensating voltage was set too high

B. Incorrect. Channel would indicate high because compensating voltage is set too low. Reinstatement of SR High Voltage on a trip is 2 of 2 logic with IR <P-6. Plausible because the applicant may confuse the logic with 1 of 2 logic required for SR High Flux trip above P-6.

C. Correct. See distractor analysis

D. Incorrect. Reinstatement of SR High Voltage on a trip is 2 of 2 logic with IR <P-6. The other channel alone will not reinstate the trip. Plausible because the applicant may confuse the logic with 1 of 2 logic required for SR High Flux trip above P-6.

Technical Reference(s): SY 1301501 R11
BD-OFN-SB-008 R8 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1301501 R5 (As available)

Question Source: Bank #
Modified Bank # WTSI 57581 (Note changes or attach parent)
New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Modification is to change failure mode to undercompensation, resulting in a different correct answer. See attached original from Wolf Creek 2007 NRC Examination

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	016	2.4.6
	Importance Rating	3.7	

Emergency Procedures / Plan: Knowledge of EOP mitigation strategies.

Proposed Question: RO Question # 31

Given the following:

- A reactor trip has occurred from 100% power.
- Following survey and acknowledgment of control room alarms, the following abnormal conditions are noted:

- Loop 2 Tavg meter, BB TI-422, indicates 590°F.
- Loop 2 Delta T meter, BB TI-421, indicates 0%.

Which ONE of the following describes RCS temperature control, and the NEXT action required for RCS temperature control, in accordance with EMG ES-02, Reactor Trip Response?

A. Steam Dumps are automatically controlling Tavg above the normal Tavg setpoint; EMG ES-02 directs manual Steam Dump to condenser to restore Tavg to setpoint.

B. Steam Dumps are automatically controlling Tavg above the normal Tavg setpoint; EMG ES-02 directs manual control of feed flow to stabilize RCS temperature at setpoint.

C. Steam Dumps have closed due to P-12 actuation; EMG ES-02 requires reset of Steam Dumps to regain control of secondary heat removal.

D. Steam Dumps have closed due to P-12 actuation; EMG ES-02 directs manual control of feed flow and manual operation of steam dumps/ARVs to control RCS temperature at setpoint.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because this could occur if P-4 (RTB) wasn't actuated to place steam dumps in plant trip mode. They would act as though they are in load reject mode and operate in a 2 degree dead band. Action would be logical if this were the failure.

B. Incorrect. Plausible because this could occur if P-4 (RTB) wasn't actuated to place steam dumps in plant trip mode. They would act as though they are in load reject mode and operate in a 2 degree dead band. Action would be logical if this were the failure, as raising feed flow would cause reduction in Tavg. (Same as A)

C. Incorrect. Failure mode is correct because the Tavg mode selects Auctioneered High Tavg. Although Delta T indicates low power, it is because Tcold failed high in this case. Plausible because bypassing P-12 may be performed for an RCS cooldown, but in this case, ES-02 does not provide this guidance because Tavg will be controlled at a higher value than where the P-12 interlock will actuate.

D. Correct.

Technical Reference(s): EMG ES-02, Rev 18, Step 1 RNO (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1504100 R3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 4

55.43

Secondary coolant and auxiliary systems that affect the facility.

Comments:

KA matched because this is the mitigation strategy for a failure of RCS temperature instrumenation post trip

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	045	K4.02	
	Importance Rating	2.5		

Knowledge of MT/G system design feature(s) and/or inter-lock(s) which provide for the following: Automatic shut of reheat stop valves as well as main control valves when tripping turbine

Proposed Question: RO Question # 32

Given the following:

- The plant is operating at 20% power during a plant startup.

Which ONE of the following describes the operation of the Turbine Emergency Trip System (ETS) if the Main Turbine were manually tripped from the control room?

A. ONLY the Electrical Trip System actuates to provide turbine trip function; Intermediate Stop Valves close, Intercept Valves remain open.

B. ONLY the Electrical Trip System actuates to provide turbine trip function; Intermediate Stop Valves AND Intercept Valves close.

C. BOTH the Electrical Trip System AND the Mechanical Trip System actuate to provide turbine trip function; Intermediate Stop Valves close, Intercept Valves remain open.

D. BOTH the Electrical Trip System AND the Mechanical Trip System actuate to provide turbine trip function; Intermediate Stop Valves AND Intercept Valves close.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. While the main trip is provided by the electrical trip system in this case, cross-trips are actuated to cause the mechanical trip system to operate as well.

Plausible because the mechanical trip system is typically identified with mechanical overspeed trip and manual trip from the front standard. Also, logical that intercept valves could remain open because the stop valves will interrupt flow to the LP turbine from MSRs. The intercept valves can be used to control the turbine, and are in the same casing as the stop valves.

B. Incorrect. While the main trip is provided by the electrical trip system in this case, cross-trips are actuated to cause the mechanical trip system to operate as well. Plausible because the mechanical trip system is typically identified with mechanical overspeed trip and manual trip from the front standard. Also, second part correctly describes operation of the valves.

C. Incorrect. The first part is correct, as cross-trips will actaute the mechanical governor. Second half is incorrect but plausible due to same reason as option A

D. Correct.

Technical Reference(s): SY 1504800 Rev 9 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1504800 R4 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 08/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	027	K2.01	
	Importance Rating	3.1		

Knowledge of bus power supplies to the following: Fans

Proposed Question: RO Question # 33

Which ONE of the following states the power supplies to Containment Atmospheric Control Fans "A" and "B"?

A. PG20 and PG25

B. PG19 and PG20

C. NG01 and NG02

D. NG03 and NG04

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because one power supply is correct, and the other supply provides power to equipment in Containment.

B. Correct.

C. Incorrect. Plausible because the supplies listed are load centers that supply the same voltage as the correct option, and it supplies other Containment ventilation equipment.

D. Incorrect. Plausible because the supplies listed are load centers that supply the same voltage as the correct option, and it supplies other Containment ventilation equipment.

Technical Reference(s): SY 1302800 Table 1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1302800 R2 (As available)

Question Source: Bank # WTSI 57839

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Wolf Creek 2007

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 8

55.43

Components, capacity, and functions of emergency systems.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	068	K1.07
	Importance Rating	2.7	

Knowledge of the physical connections and/or cause effect relationships between the Liquid Radwaste System and the following systems: Sources of liquid wastes for LRS

Proposed Question: RO Question # 34

Given the following:

- An event has occurred causing a Reactor Trip and Safety Injection Actuation.
- RCS pressure is 1450 psig and lowering.
- Containment pressure is 8.5 psig and rising.

Which ONE of the following describes the status of RCP seal leakoff under these conditions?

A. All RCP seal leakoff is isolated; #2 seal leakoff is directed to the PRT.

B. RCP #1 seal leakoff is directed to the PRT; #2 seal leakoff is directed to the RCDT.

C. RCP #1 seal leakoff is directed to the VCT; #2 seal leakoff is directed to the PRT.

D. All RCP seal leakoff is isolated; #2 seal leakoff is directed to the RCDT.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. All RCP leakoff is not isolated. RCP #1 seal leakoff isolates but is directed to PRT via relief valve, since there is a DP from the seal through the relief valve. Plausibility enhanced because #2 seal does drain to the RCDT, not the PRT.

B. Correct.

C. Incorrect. Plausible because normally #1 seal is directed to VCT, and applicant must infer that Containment Isolation Phase A (CISA) has occurred since SI is actuated, which would isolate the normal #1 seal leakoff flowpath. Applicant may confuse seal leakoff flowpaths and believe that #2 seal is directed to the PRT.

D. Incorrect. Same description as option A and C. #1 seal is directed to PRT via relief, even though isolation to VCT does occur when CISA actuates. #2 seal is directed to RCDT under either condition, whether normal or CISA conditions.

Technical Reference(s): SY 1300300 Rev 18 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1300300 R6 (As available)

Question Source: Bank # WTSI 58936

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Watts Bar 2007 NRC Exam (Exam was never administered)

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	028 A1.02	
	Importance Rating	3.4	

Ability to predict and/or monitor changes in parameter (to prevent exceeding design limits) associated with operating the HRPS controls including: Containment pressure

Proposed Question: RO Question # 35

Given the following:

- A LOCA has occurred.
- ECCS failures required entry to EMG FR-C1, Response to Inadequate Core Cooling.
- The crew is evaluating the operation of Hydrogen Recombiners.
- Containment pressure is 37 psig.
- Containment Hydrogen Concentration is 2.7%.

Which ONE of the following describes whether the Hydrogen Recombiners will be placed in operation, and the reason why?

A. Place Hydrogen Recombiners in service;

Containment Hydrogen concentration is above the level where risk of an explosion would result in unacceptable containment pressure rise.

B. Place Hydrogen Recombiners in service;

Containment Hydrogen concentration is within the range where Recombiners can effectively reduce containment hydrogen concentration, but below the range where ignition of hydrogen may occur.

C. DO NOT place Hydrogen Recombiners in service;

Containment Hydrogen concentration is above the level where risk of an explosion would result in unacceptable containment pressure rise.

D. DO NOT place Hydrogen Recombiners in service;

Containment Hydrogen concentration is below the range where Recombiners can effectively reduce containment hydrogen concentration, and risk of explosion is minimal.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. EMG FR-C1 background doc describes operation of Recombiners. If concentration is between 0.8% and 3.0%, Recombiners are placed in service. This is due to minimal risk of explosion and unwanted containment pressure increase. Plausible because the applicant may believe that recombiners must be placed in service due to the higher risk of explosion as H2 concentration rises. Incorrect because at this concentration, risk is lower, especially with 37 psig in containment.

B. Correct. At 4% hydrogen, ignition may occur in dry air. The recombiners are placed in service due to this consideration. Typical flammability limit discussions start at 3% hydrogen concentration, but with containment pressure at 37 psig, ignition is highly unlikely.

C. Incorrect. Plausible because the range is 0.8% to 3.0%. The applicant must understand the plant condition and can easily confuse the applicant, who may believe that Recombiners are started only if hydrogen concentration is below 0.8% to prevent ignition.

D. Incorrect. Plausible because the applicant may know that one setpoint in EMG FR-C1 is 3%, and can confuse that setpoint with being the limit for Recombiner operation. The option is logical if the applicant determines that 3% is the cutoff.

Technical Reference(s): EMG FR-C1, Rev 15 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732341 R3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:		Last NRC Exam:
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	10
	55.43	

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	086	A4.01	
	Importance Rating	3.3		

Ability to manually operate and/or monitor in the control room: Fire water pumps

Proposed Question: RO Question # 36

Given the following:

- A fire is in progress in the Upper Cable Spreading Room.
- The Motor Driven Fire Pump automatically started on low system pressure.
- The Diesel Driven Fire Pump remains in Standby.
- Ten minutes after initiation, the fire is extingushed, and the Fire Protection System pressure has returned to normal.

Which ONE of the following describes the operation of the Motor Driven Fire Pump during this event?

A. The pump will run indefinitely until stopped by using the local pump controls ONLY.

B. The pump will run indefinitely until stopped by using either the local pump controls or the control switch on panel KC-008.

C. The pump will automatically shut down 30 minutes after starting, unless the control switch on Panel KC-008 is placed in STOP.

D. The pump will automatically shut down 30 minutes after starting, unless the control switch on Panel KC-008 is placed in START.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. The pump will run indefinitely only if the local or remote control switch is placed in START. It will not run indefinitely if neither hand switch is operated.

B. Incorrect. Same reason as A, but also, control switch in Control Room only starts the pump, and is spring return to neutral with no stop feature.

C. Incorrect. Plausible because the pump will automatically shut down after 30 minutes, unless either the local controls are placed in start or stop, or the KC-008 control is placed in start.

D. Correct.

Technical Reference(s): SY 1408600 Rev 18 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1408600 R4 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Facility:	Wolf Creek
Vendor:	WEC
Exam Date:	8/2009
Exam Type:	R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	017	A3.02	
	Importance Rating	3.4		

Ability to monitor automatic operation of the ITM system including: Measurement of in-core thermocouple temperatures at panel outside control room

Proposed Question: RO Question # 37

Which ONE of the following describes the individual inputs to the Core Subcooling Monitor, and the location(s) where these inputs may be monitored?

A. Thermocouple inputs and reference junction temperatures; Process Cabinet CRT ONLY.

B. Thermocouple inputs and reference junction temperatures; Process Cabinet CRT AND Plant Computer.

C. Pressure inputs and Wide Range RTDs; Plant Computer ONLY.

D. Pressure inputs and Wide Range RTDs; Process Cabinet CRT ONLY.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. All of these inputs are also provided to NPIS.

B. Correct.

C. Incorrect. Plant computer has these indications, but the Processor Cabinet CRT has them also.

D. Incorrect. The plant computer also has these same indications available. Plausible because the applicant may know that the CRT has these indications, and may not be sure about the plant computer.

Technical Reference(s): SY 1300202 Rev 10 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1300202 R4 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 8

55.43

Components, capacity, and functions of emergency systems.

Comments:

Cabinet is not really outside of control room but near process 7300 racks away from the main control board. Found no other ref that would provide for indication elsewhere

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	041	A2.02	
	Importance Rating	3.6		

Ability to (a) predict the impacts of the following malfunctions or operations on the SDS; and (b) based on those predictions or mitigate the consequences of those malfunctions or operations: Steam valve stuck open

Proposed Question: RO Question # 38

Given the following:

- The plant is at 95% power.
- Rod Control is in AUTO.
- The following alarm is received and clears approximately 30 seconds later:

- 00-065D, TREF/T-AUCT HI

- Condenser Steam Dump valve, AB ZL-37, failed open.

Which ONE of the following describes the effect on the unit and the **minimum** action required in accordance with OFN AB-041, Steamline or Feedline Leak?

A. Rod withdrawal will occur; Place the affected Condenser Steam Dump Interlock Switch to OFF.

B. Rod withdrawal will occur; Place BOTH Condenser Steam Dump Interlock Switches to OFF.

C. Rod insertion will occur; Place the affected Condenser Steam Dump Interlock Switch to OFF.

D. Rod insertion will occur; Place BOTH Condenser Steam Dump Interlock Switches to OFF.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because rod motion is correct and also because interlock selector switches must be placed in OFF, but the procedure requires both, not 1 switch to be placed in OFF.

B. Correct.

C. Incorrect. See option A for description of interlock selector switch operation. Also plausible because the alarm is given, and does not indicate what is happening to RCS temperature. The applicant may know temperature is lowering or that power is rising, and they will have to choose which way rods will move.

D. Incorrect. Plausible because description of interlock selector switch operation is correct. Also, see option C for plausibility of first part of distractor.

Technical Reference(s): OFN AB-041 Rev 2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732451 R3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Similar questions exist for other KAs. This was developed from scratch and uses different parameters

Facility: Wolf Creek

Vendor: WEC

Exam Date: 08/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	055	EK1.02	
	Importance Rating	4.1		

Knowledge of the operational implications of the following concepts as they apply to the Station Blackout : Natural circulation cooling

Proposed Question: RO Question # 39

The crew is performing EMG CS-01, Loss of All AC Power Recovery Without SI Required, and are attempting to verify natural circulation with the following conditions:

- Offsite power is <u>not</u> available.
- Both emergency diesel generators are supplying their loads.
- RCS subcooling is 35°F.
- RCS pressure is lowering slowly.
- Steam generator levels and pressures are stable.
- Core exit thermocouples are rising slowly.
- RCS WR cold legs are at saturation temperatures for the steam generator pressures.

Which ONE of the following describes the correct operator actions based on the above conditions?

A. Open the steam dumps to lower steam generator pressures and cool the RCS.

B. Open the steam generator ARVs to lower steam generator pressures and cool the RCS.

C. Throttle closed the steam dumps to maintain the same steam generator pressures and conserve secondary inventory.

D. Throttle closed ARVs to maintain the same steam generator pressures and conserve secondary inventory.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Condenser is not available due to SBO. Plausible because dumping steam is the correct course of action.

B. Correct. Step 24 of EMG CS-01. ARVs must be used because condenser is not available, and RCS subcooling does not meet minimum required

C. Incorrect. Condenser is not available and procedure gives direction to dump steam at a higher rate if Natural Circulation is not established. Plausible because the ability to maintain inventory is a concern with no power to transfer water to CST.

D. Incorrect. Procedure gives direction to dump steam at a higher rate if Natural Circulation is not established. Plausible because the ability to maintain inventory is a concern with no power to transfer water to CST.

Technical Reference(s): EMG CS-01, step 24, Rev 17 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO1732330, R3 (As available)

Question Source: Bank # WTSI 60576

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Vogtle 2005

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

WTSI 60576

The question tests knowledge of NC verification criteria and actions to take when parameters are not reading values that indicate NC has been established. The implications of not yet having NC are to steam the SGs to induce NC. Therefore, the question is matching the KA testing operational implications (operator actions) of not having NC following a SBO

Facility:	Wolf Creek
Vendor:	WEC
Exam Date:	08/2009
Exam Type:	R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	054	AK1.02
	Importance Rating	3.6	

Knowledge of the operational implications of the following concepts as they apply to Loss of Main Feedwater (MFW): Effects of feedwater introduction on dry S/G

Proposed Question: RO Question # 40

Initial Conditions:

- The plant tripped from 100% power due to a loss of all feed flow.
- Multiple failures caused a loss of all AFW.

Current conditions:

- ALL SG Wide Range levels are 7% and slowly lowering.
- RCS Hot Leg Temperatures are 575°F and slowly lowering.
- Core Exit TCs are 580°F and slowly lowering.
- Local actions have restored the capability to feed from AFW.

Which ONE of the following describes the INITIAL method for recovery of Heat Sink in accordance with EMG FR-H1, Loss of Secondary Heat Sink, and the basis for that method?

A. Feed all four SGs at a minimum of 270,000 lbm/hr total flow;
Minimum flow to ensure adequate heat sink.

B. Feed all four SGs at a minimum of 270,000 lbm/hr total flow;
Ensure symmetric cooling during Natural Circulation.

C. Feed only ONE SG at less than or equal to 35,000 lbm/hr;
Minimum flow to ensure adequate heat sink.

D. Feed only ONE SG at less than or equal to 35,000 lbm/hr;
Limit thermal stresses to SG components.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible since to satisfy the CSF status tree 270K is required based on these plant conditions (NR levels in all SGs < 6%). The applicant who is not familiar with the criteria for a hot-dry SG is likely to select this distractor.

B. Incorrect. Flow rate is plausible as described above. ERG and plant procedures discuss maintaining symetric cooling, so this is plausible however the HOT-DRY criteria precludes feeding all SGs.

C. Incorrect. First part is correct as it is the guidance for HOT-DRY SGs. The second part is incorrect but plausible since 35K gpm is the minimum verifiable AFW flow at WCNOC; candidate is likely to associate 35K as correct for this reason (e.g. EMG C-21 establishes 30K to maintain minimum heat sink). For these plant conditons the concern is for feeding excessively not feeding a minimum amount.

D. Correct. EMG FR-H1 limits feeding to only ONE SG at a time and limits flow to 35K lbm/hr. The ERG basis is to limit thermal stresses to the SG.

Technical Reference(s): EMG FR-H.1, Rev 19A and background doc (Attach if not previously provided)

Proposed References to be provided to applicants during examination:

Learning Objective: LO1732346 R1 (As available)

Question Source: Bank # WTSI 62054

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: North Anna 2008

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:
WTSI 62054

Facility: Wolf Creek

Vendor: WEC

Exam Date: 08/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	011	EK1.01
	Importance Rating	4.1	

Knowledge of the operational implications of the following concepts as they apply to the Large Break LOCA: Natural circulation and cooling, including reflux boiling.

Proposed Question: RO Question # 41

Given the following:

- A LOCA has occurred approximately 20 minutes ago.
- All equipment is operating as designed.
- RCS pressure is 200 psig and slowly lowering.
- Steam generator pressures are approximately 800 psig and slowly lowering.

Which ONE of the following describes the heat removal mechanism(s) currently occurring, and the operator action(s) that will be effective to enhance core cooling?

A. Break flow only; ensure adequate ECCS flow for current RCS pressure.

B. Break flow only; ensure RHR flow is rising as RCS pressure lowers, ensure adequate AFW flow exists, and operate ARVs as necessary for plant cooldown.

C. Break flow and reflux boiling; ensure adequate ECCS flow for current RCS pressure ONLY.

D. Break flow and reflux boiling; ensure RHR flow is rising as RCS pressure lowers, ensure adequate AFW flow exists, and operate ARVs as necessary for plant cooldown.

Proposed Answer: A

Explanation (Optional):

A. Correct. For large break LOCAs, break flow is the heat removal mechanism. At this point in the event, SGs are a heat source, and steaming them will provide no benefit for RCS cooldown until SG pressure is below RCS pressure.

B. Incorrect. Steaming SGs will enhance reflux cooling, but SGs must be a heat sink for this to occur.

C. Incorrect. Plausible because the action is correct, and applicant may misunderstand reflux cooling mechanism.

D. Incorrect. Plausible because the action is correct if reflux cooling were occurring, and applicant may misunderstand reflux cooling mechanism.

Technical Reference(s): LO1610706 Rev 8 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO1610706 R5 (As available)

Question Source: Bank #

Modified Bank # WTSI 61377 (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43

Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

WTSI 61377, Modified from Diablo Canyon 2005 NRC exam (Attached)

Facility: Wolf Creek

Vendor: WEC

Exam Date: 08/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	E11 EK2.1	
	Importance Rating	3.6	

Knowledge of the interrelations between the (Loss of Emergency Coolant Recirculation) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Proposed Question: RO Question # 42

Which ONE of the following describes the action required in EMG C-11, Loss of Emergency Coolant Recirculation, when ALR 00-047B, RWST EMPTY, alarm is received, and RWST level is verified to be 5%?

A. Stop BOTH Containment Spray pumps and reduce ECCS flow to ONE train running.

B. Stop ALL pumps taking a suction from the RWST, align normal charging, and initiate secondary depressurization to facilitate SI Accumulator injection.

C. Stop BOTH Containment Spray pumps, and throttle SI and RHR flow in accordance with decay heat removal requirements.

D. Stop ONE Containment Spray Pump and reduce ECCS flow to ONE train running. Ensure sufficient Containment Fan Coolers aligned, then stop the second Containment Spray Pump.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because both Containment Spray pumps are stopped, and there is a step in procedure where the crew may reduce ECCS to one train running based on whether cooling requirements are being met.

B. Correct. With RWST empty, all pumps will lose suction if they are not tripped.

C. Incorrect. Plausibility same as A, and action to throttle ECCS is also contained in EMG C-11.

D. Incorrect. Plausibile because there is a table that evaluates reducing Containment Spray flow, but with RWST empty, all ECCS and Spray must be stopped.

Technical Reference(s): EMG C-11, Rev 20, Steps 44-58 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO1732332 R3 (As available)

Question Source: Bank #

Modified Bank # WTSI 53062 (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

WTSI 53062 Mod from North Anna 2006 NRC Retake exam

Mods simplified to memory level item and removed plant conditions that would make item open to interpretation and more than 1 correct answer

Facility: Wolf Creek

Vendor: WEC

Exam Date: 08/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	E12 EK2.1	
	Importance Rating	3.4	

Knowledge of the interrelations between the (Uncontrolled Depressurization of all Steam Generators) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Proposed Question: RO Question # 43

During the performance of EMG C-21, Uncontrolled Depressurization of All Steam Generators, the following conditions exist:

- RCS cooldown rate is determined to be 165° F/Hr.
- All SG NR levels are off-scale low.
- Total AFW flow is 300,000 lbm/hr.

Which ONE of the following describes how the crew is directed to control AFW flow?

A. Flow is reduced to 30,000 lbm/hr to each SG, and Thot is monitored to ensure secondary heat sink is maintained.

B. Flow is terminated to all but a single SG, which is fed at 30,000 lbm/hr, and Tcold is monitored for conditions that may result in Pressurized Thermal Shock.

C. Total flow is maintained >270,000 lbm/hr until ANY SG narrow range level is >6%, and Thot is monitored to ensure secondary heat sink is maintained.

D. Total flow is maintained >270,000 lbm/hr until ALL SG narrow range levels are >6%, and Tcold is monitored for conditions that may result in Pressurized Thermal Shock.

Proposed Answer: A

Explanation (Optional):

A. Correct. See EMG C-21 step 5 and basis

B. Incorrect. Plausible because flow is initiated to only 1 SG in EMG FR-H1.

C. Incorrect. Plausible because this flow is maintained under these conditions in EMG E-0 or in EMG C-21 if RCS cooldown rate is <100°F/Hr.

D. Incorrect. Plausible because second half is true, but with RCS cooldown rate >100°F/hr, AFW flow is minimized.

Technical Reference(s): EMG C-21, Rev 17, Step 5 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO1732334 R2, R3 (As available)

Question Source: Bank #

Modified Bank # WTSI 52615 (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

WTSI 52615 - From VC Summer Audit 2006. There are items in our bank with similar context.

Facility:	Wolf Creek
Vendor:	WEC
Exam Date:	08/2009
Exam Type:	R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	008	AK2.01
	Importance Rating	2.7	

Knowledge of the interrelations between the Pressurizer Vapor Space Accident and the following: Valves

Proposed Question: RO Question # 44

Given the following:

- A reactor trip and safety injection have occurred.
- RCS pressure is 1050 psig and lowering.
- Tavg is 550°F and lowering.
- Pressurizer level is 65% and rising rapidly.
- Containment pressure is 3 psig and rising.

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

A. Letdown line break.

B. SBLOCA on an RCS cold leg.

C. Stuck open pressurizer spray valve.

D. Stuck open pressurizer PORV.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. A letdown line break would result in pressurizer level lowering. Plausible because all other conditions presented by the stem would be available. If RCS pressure was rising, this could be true because SI could be refilling the RCS.

B. Incorrect. Same as A, but break could potentially be larger than a letdown line break.

C. Incorrect. Plausible because a stuck open spray valve would have characteristics of a vapor space break, except that Containment pressure would not be rising.

D. Correct. If a safety valve or PORV sticks open and the PRT ruptures, Containment pressure will rise. PZR level will lower until the RCS reaches saturation, when a bubble is created under the head. At this point (~1100 psig RCS pressure) RCS inventory is forced out of the break, which is in the top of the pressurizer, resulting in PZR level indication rising.

Technical Reference(s): LO1610722 Rev 9 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO1610722 R8 (As available)

Question Source: Bank # WTSI 57311

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: VC Summer 2007

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43

Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limiitations and reasons for these operating characteristics.

Comments:

WTSI 57311

Facility:	Wolf Creek
Vendor:	WEC
Exam Date:	08/2009
Exam Type:	R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	007	EK3.01
	Importance Rating	4.0	

Knowledge of the reasons for the following as the apply to a reactor trip: Actions contained in EOP for reactor trip

Proposed Question: RO Question # 45

Which ONE of the following describes an immediate action and the basis for the action in accordance with EMG E-0, Reactor Trip or Safety Injection?

A. Insert control rods in Manual; to insert negative reactivity by the most direct manner possible.

B. Insert control rods in Manual; to ensure that the only heat input to the RCS is from RCPs.

C. Place EHC Pumps in PULL-TO-LOCK; to prevent uncontrolled cooldown of the RCS.

D. Place EHC Pumps in PULL-TO-LOCK; to minimize secondary makeup requirements.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because these are immediate actions in EMG FR-S1, which is a transition from the immediate actions of E-0. In E-0, attempt to manually trip reactor, but rods are driven in EMG FR-S1. Reason is correct and logical for any reactor trip.

B. Incorrect. Plausible because these are immediate actions in EMG FR-S1, which is a transition from the immediate actions of E-0. In E-0, attempt to manually trip reactor, but rods are driven in EMG FR-S1. Reason is correct manual trip of reactor.

C. Correct.

D. Incorrect. The action is correct, and the reason is plausible because the statement by itself is true. However, the basis for the action is to minimize cooldown, not minimize the need for makeup.

Technical Reference(s): EMG E-0, Rev 20
BD EMG FR-S1 Rev 9
BD EMG E-0, Rev 10

(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732313 R2 (As available)

Question Source: Bank # WTSI 57329

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: VC Summer 2006

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Numerous similar items in bank. Wolf Creek item was initially randomly selected. There were enough items to reselect.

Facility: Wolf Creek

Vendor: WEC

Exam Date: 08/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	056	AK3.01	
	Importance Rating	3.5		

Knowledge of the reasons for the following responses as they apply to the Loss of Offsite Power: Order and time to initiation of power for the load sequencer

Proposed Question: RO Question # 46

Given the following:

- The plant tripped from 100% power due to loss of off-site power.
- Ten seconds after the EDG output breakers close, Safety Injection actuates.

Which ONE of the following describes the operation of AFW pumps "A" and "B" for this event, and the reason why?

A. BOTH AFW pumps start when AFAS is actuated; Both AFW pump breakers remain closed and will start as soon as the LOCA Sequencer initiates;

Ensures conditions assumed in the accident analysis for ALL Category III or IV events are satisfied.

B. BOTH AFW pumps start when AFAS is actuated; Both AFW pump breakers are stripped and will start as based on pre-determined priority when the LOCA Sequencer initiates;

Prevents EDG overload by allowing starting current from other equipment to decay.

C. NEITHER AFW pump starts until the pre-determined time after the LOCA Sequencer initiates;

Ensures conditions assumed in the accident analysis for ALL Category III or IV events are satisfied.

D. NEITHER AFW pump starts until the pre-determined time after the LOCA Sequencer initiates;

Prevents EDG overload by allowing starting current from other equipment to decay.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because AFW pumps do start on an AFAS signal after the reactor trips. The loss of power occurred prior to the AFAS signal being received, so the AFAS signal would be blocked by either Shutdown or LOCA Sequencer until the time delay is satisfied. AFW start is not a condition assumed for all Category III or IV events, although it is assumed for some.

B. Incorrect. The loss of power occurred prior to the AFAS signal being received, so the AFAS signal would be blocked by either Shutdown or LOCA Sequencer until the time delay is satisfied. Reason is correct.

C. Incorrect. Plausible because operation of the pumps is correct, but AFW start is not a condition assumed for all Category III or IV events, although it is assumed for some.

D. Correct.

Technical Reference(s): SY 1406401 Rev 10 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SI 1406401 R5 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 08/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	E05	EK3.4	
	Importance Rating	3.7		

Knowledge of the reasons for the following responses as they apply to the (Loss of Secondary Heat Sink) RO or SRO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated.

Proposed Question: RO Question # 47

Given the following:

- Reactor trip and safety injection have occurred.
- The crew has entered EMG FR-H1, RESPONSE TO LOSS OF SECONDARY HEAT SINK on a RED condition for the Heat Sink CSF Status Tree.
- ALL SG wide range levels are 30% and slowly trending down.
- Total AFW flow is 0 gpm.
- Main Feed Pumps will NOT reset.
- Startup Feed Pump tripped upon starting and CANNOT be started.

Which ONE of the following describes the actions required in accordance with EMG FR-H1?

Trip all RCPs to....

A. minimize heat input to the RCS; initiate secondary depressurization to establish Condensate flow.

B. prevent loss of RCS inventory due to damage to RCP seals; initiate secondary depressurization to establish Condensate flow.

C. minimize heat input to the RCS; return to EMG FR-H1, step 1 and re-attempt start of AFW Pumps.

D. prevent loss of RCS inventory due to damage to RCP seals; return to EMG FR-H1, step 1 and re-attempt start of AFW Pumps.

Proposed Answer: A

Explanation (Optional):

A. Correct. Reason for tripping RCPs in EMG FR-H1 is to reduce heat input to RCS to conserve SG inventory to delay requirement for bleed and feed. SG levels must have 3 below 26% wide range before initiation of bleed and feed.

B. Incorrect. Reason for RCPs is incorrect for current conditions, but would be correct for other EOPs, when Phase B isolation is required. (Loss of seal cooling) Action is correct for SG levels.

C. Incorrect. Plausible because reason for RCP trip is correct. The crew would attempt condensate prior to returning to step 1

D. Incorrect. Reason for RCPs is incorrect for current conditions, but would be correct for other EOPs, when Phase B isolation is required. (Loss of seal cooling) The crew would attempt condensate prior to returning to step 1.

Technical Reference(s): EMG FR-H1 Rev 19A
BD EMG FR-H1 Rev 8A (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732346 R1 (As available)

Question Source: Bank #
Modified Bank # WTSI 56116 (Note changes or attach parent)
New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Facility:	Wolf Creek
Vendor:	WEC
Exam Date:	08/2009
Exam Type:	R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	E04	EA1.2	
	Importance Rating	3.6		

Ability to operate and / or monitor the following as they apply to the (LOCA Outside Containment) Operating behavior characteristics of the facility.

Proposed Question: RO Question # 48

Given the following:

- A LOCA outside containment has occurred.
- The crew is performing the actions in EMG C-12, LOCA Outside Containment.
- CVCS has been verified isolated.

Which ONE of the following actions will be the FIRST to be attempted to isolate the break and which indication is used to determine if the leak has been isolated in accordance with EMG C-12?

A. Isolate RHR piping; RCS pressure is monitored, because SI flow will repressurize the RCS with the break isolated.

B. Isolate RHR piping; Pressurizer level is monitored, because with the break isolated, RCS inventory will rapidly rise.

C. Isolate the BIT Injection line; RCS pressure is monitored, because SI flow will repressurize the RCS with the break isolated.

D. Isolate the BIT Injection line; Pressurizer level is monitored, because with the break isolated, RCS inventory will rapidly rise.

Proposed Answer: A

Explanation (Optional):

A. Correct. CVCS is checked while re-energizing RHR Hot Leg suction valves to ensure they are closed. The first isolation occurs in low pressure piping, which is more likely to break than the BIT Injection line, which is designed for higher pressure.

B. Incorrect. Pressurizer level may be an indication of recovering inventory, but if it was off-scale low, it may take a long time to determine if the break was isolated, so it is not used for EMG C-12.

C. Incorrect. Plausible because the BIT is checked, and connected to RCS. BIT is just lower priority than RHR. Parameter is correct, as in A.

D. Incorrect. See descriptions above, same determination.

Technical Reference(s): EMG C-12, Steps 4-7 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732333 R4 (As available)

Question Source: Bank # WTSI 52232

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Robinson 2004 NRC

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Similar items available on several NRC exams.

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	038	EA1.19
	Importance Rating	3.4	

Ability to operate and monitor the following as they apply to a SGTR: MFW System status indicator

Proposed Question: RO Question # 49

Given the following:

- The plant is at 100% power.

Which ONE of the following describes the indications of a tube leak in "A" Steam Generator?

A. Steam Flow stable
Feed Flow rising
SG Level rising

B. Steam Flow rising
Feed Flow rising
SG Level rising

C. Steam Flow stable
Feed Flow lowering
SG Level stable

D. Steam Flow lowering
Feed Flow lowering
SG Level stable

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because it is logical that extra fluid into the SG would raise level, but the control system would respond to maintain level, by lowering feed flow. At 100% power, the mismatch will not be large, but based on plant conditions, it could be seen on SGWLC.

B. Incorrect. Plausible because if the applicant believes that steam flow would rise, it is logical to believe that feed flow would attempt to match it, and level would rise. Incorrect because the phenomena of a tube rupture would not be to increase steam flow, but to limit feed flow to maintain level with the RCS inventory contributing to SG mass.

C. Correct. Level will be maintained at program by SGWLC. With a large SGTR, feed flow would lower to limit SG level increase, creating a slight steam flow-feed flow mismatch.

D. Incorrect. Steam flow would be maintained by steam demand and be fairly equal to all other SGs. A tube rupture would not be large enough to appreciably change steam flow. Plausible because feed flow and SG level response are correct, and because a logical connection can be made to increased steam flow based on RCS inventory entering the SG.

Technical Reference(s): LO 1610722, Rev 9, Section 10.2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1610722 R10 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 14

55.43

Principles of heat transfer, thermodynamics and fluid mechanics.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	025	AA1.11
	Importance Rating	2.9	

Ability to operate and / or monitor the following as they apply to the Loss of Residual Heat Removal System: Reactor building sump level indicators

Proposed Question: RO Question # 50

Initial conditions:

- The plant is in Mode 5.
- Reduced Inventory Operations are in progress for "B" RCP repair.
- RHR Pump "A" and CCW Train "A" are in service.

Current conditions:

- RHR Pump "A" discharge flow begins fluctuating up to 1000 GPM.
- RCS temperature is rising slowly.
- Reactor Coolant Drain Tank level is rising at an increasing rate.
- The crew enters OFN EJ-015, Loss of RHR Cooling.
- RCS level indicates the following:

- Loop level indicates 16 inches.
- Tygon Tube level indicates 14 inches.

Which ONE of the following describes the location of the leak, and the FIRST action that will be taken?

The leak is located in...

A. the Auxiliary Building; initiate RCS makeup using an available SI Pump.

B. the Auxiliary Building; reduce RHR flow to stop RHR Pump cavitation.

C. Containment; initiate RCS makeup using an available SI Pump.

D. Containment; reduce RHR flow to stop RHR Pump cavitation.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Level is below minimum required for reduced inventory operations, so leakage must be occurring somewhere in the system, but with only RCDT level rising, the leak will be in Containment. Plausible because action is correct for CCP but not SIP, but only after other actions such as reducing flow are performed.

B. Incorrect. Level is below minimum required for reduced inventory operations, so leakage must be occurring somewhere in the system, but with only RCDT level rising, the leak will be in Containment. Plausible because action is correct (per foldout page) for the indication given.

C. Incorrect. Plausible because this indication would be seen, and the action to reduce letdown is an action that will be taken for the conditions given, but it is after the step (and foldout page) to reduce RHR flow to reduce cavitation.

D. Correct. See reference and see explanations above.

Technical Reference(s): OFN EJ-015 Rev 15A (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732425 R3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	027	AA2.13
	Importance Rating	2.8	

Ability to determine and interpret the following as they apply to the Pressurizer Pressure Control Malfunctions: Seal return flow

Proposed Question: RO Question # 51

Given the following:

- The plant is at 100% power.
- The controlling pressurizer pressure instrument fails high.

Which ONE of the following describes the response of the plant PRIOR to the first protective action that occurs?

A. Total RCP Seal Leakoff flow will rise slightly; unaffected OTDT channel setpoints rise.

B. Total RCP Seal Leakoff flow will rise slightly; unaffected OTDT channel setpoints lower.

C. Total RCP Seal Leakoff flow will lower slightly; unaffected OTDT channel setpoints rise.

D. Total RCP Seal Leakoff flow will lower slightly; unaffected OTDT channel setpoints lower.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Seal leakoff flow is driven by seal design, and will change with differential pressure across the #1 seal. The RCS is depressurizing due to open PORV and spray valves, and VCT pressure is constant. Therefore, differential pressure goes down, resulting in less impetus for flow. OTDT setpoint lowers due to the lowering RCS pressure. This is easily confused by the applicant because the plant is operating closer to setpoint.

B. Incorrect. Seal leakoff flow is driven by seal design, and will change with differential pressure across the #1 seal. The RCS is depressurizing due to open PORV and spray valves, and VCT pressure is constant. Therefore, differential pressure goes down, resulting in less impetus for flow. OTDT setpoint lowers due to the lowering RCS pressure, which is correct in this option.

C. Incorrect. First part is correct but second half is incorrect for same reason as option A.

D. Correct.

Technical Reference(s): SY 1301000 Rev 7 (Attach if not previously provided)

BD OFN SB-008 Rev 8

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1301000 R10 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 08/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	057	AA2.19
	Importance Rating	4.0	

Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: The plant automatic actions that will occur on the loss of a vital ac electrical instrument bus

Proposed Question: RO Question # 52

Given the following:

- In accordance with GEN 00-003, Hot Standby to Minimum Load, power has been stabilized between 13 and 16%.
- Preparations are being made to synchronize the Main Generator to the grid.
- Reactor power indicates the following:

- IR N-35 - $1X10^{-5}$ amps.
- IR N-36 - $1X10^{-5}$ amps.
- PR N-41 - 15%.
- PR N-42 - 14%.
- PR N-43 - 14%.
- PR N-44 - 14%.

120 VAC Inverter NN11 output breaker trips open.

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

A. A reactor trip due to the deenergization of Intermediate Range Channel N-35.

B. A Power Range Rod Stop (C-2) signal, with no change in reactor power.

C. A reactor trip due to the deenergization of Intermediate Range Channel N-36.

D. An Intermediate Range Rod Stop (C-1) signal, with no change in reactor power.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Power is above P-10, so the Intermediate Range High Flux Reactor Trips are not active. Plausible because this would be correct for 2 of 4 Power Range Channels below 10% power.

B. Correct. 1 of 4 coincidence is required for Power Range Overpower Rod Stop. Since the bistable for the rod stop de-energizes to actuate, this rod stop would be in effect for PR Channel N41.

C. Incorrect. Plausible because Inverter NN12 deenergizing below 10% power would make this correct.

D. Incorrect. Plausible because power below P-10, this would be correct if IR power went above limit. Similar to actual answer, but incorrect because of power level.

Technical Reference(s): SY 1301200 Rev 5 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1301200 R5 (As available)

Question Source: Bank # WTSI 57307

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: VC Summer 2007

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Facility:	Wolf Creek
Vendor:	WEC
Exam Date:	8/2009
Exam Type:	R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	077	AA2.07	
	Importance Rating	3.6		

Ability to determine and interpret the following as they apply to Generator Voltage and Electric Grid Disturbances: Operational status of engineered safety features

Proposed Question: RO Question # 53

Given the following:

- The plant is operating at 100% power.
- The switchyard is in a normal electrical lineup.
- Severe weather is causing voltage fluctuations on the grid, and observed in the Wolf Creek switchyard.
- Westar Transmission Services (WETS) predictive model has generated a LO ALRM for the past hour.
- Switchyard voltage is currently 334.05 KV.
- The crew is referring to OFN AF-025, Unit Limitations.

Which ONE of the following is (1) the operational concern caused by the current plant conditions, and (2) the action that will be required if the conditions continue?

A. (1) Operability of safety related equipment
(2) A unit load reduction or plant shutdown may be required

B. (1) Operability of safety related equipment
(2) EDGs will be started to supply NB01 and NB02 disconnected from the grid

C. (1) Overload of switchyard breakers and disconnects
(2) A unit load reduction or plant shutdown may be required

D. (1) Overload of switchyard breakers and disconnects
(2) Commence shutdown of unnecessary electrical loads

Proposed Answer: A

Explanation (Optional):

A. Correct. Concern is potential loss of off-site power with LOCA loads running.

B. Incorrect. Plausible because first half is correct, but OFN AF-025 does not direct starting EDGs with NB01/NB02 disconnected from grid. However, this is a common practice at Westinghouse facilities if a degraded voltage condition exists for an extended period.

C. Incorrect. Plausible because a lower voltage results in a higher current flow for a given generator output, or real load on the grid. Action is correct.

D. Incorrect. Plausible same reasons as option C. This action would be correct also, for problems with transformers.

Technical Reference(s): OFN AF-025 Rev 27, Attachment E (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO1732435 R3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Facility:	Wolf Creek
Vendor:	WEC
Exam Date:	8/2009
Exam Type:	R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	026	2.1.23	
	Importance Rating	4.3		

Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Proposed Question: RO Question # 54

Given the following:

- The plant is at 100% power.
- "B" Component Cooling Water (CCW) train is running supplying the Service Loop.
- "B" CCW surge tank is at 15% and lowering.
- "A" CCW surge tank indicates 55% and stable.

Which ONE of the following actions is required NEXT in accordance with OFN EG-004, CCW SYSTEM MALFUNCTIONS, while attempting to identify and isolate the leak?

A. Align ESW makeup to the CCW surge tank, and attempt to maintain level from 40% to 60%.

B. Place "B" CCW Pump in Pull-To-Lock, isolate CCW service loop supply and return valves, trip the reactor, trip RCPs.

C. Isolate CCW to the Spent Fuel Pool Cooling Heat Exchanger.

D. Isolate Letdown and reduce Charging flow to minimum while maintaining reactor power stable.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. An alternate train pump is first started, inventory is maintained to the system so that time is available to isolate the leak, and the alternate train is placed in service. In this case, the leak is on the service loop, so the next action is to open ESW supply to CCW, but Foldout page criteria is met to trip reactor based on 15% level in CCW Surge Tank.

B. Correct.

C. Incorrect. Isolating this component is not addressed until the end of the procedure, although it will be performed for a service loop leak at some point prior to identification of the leak.

D. Incorrect. This is part of foldout page action when the CCW Surge Tank lowers to 15% due to a service loop leak. More plausible because letdown is a service loop component. Reactor is tripped, power is not maintained stable.

Technical Reference(s): OFN EG-004, Rev 10, Foldout (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732414 R3 (As available)

Question Source: Bank #

Modified Bank # WTSI 48192 (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

WTSI 48192
Modified from Wolf Creek 2004

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	022	2.2.40	
	Importance Rating	3.4		

Equipment Control: Ability to apply technical specifications for a system.

Proposed Question: RO Question # 55

Given the following:

- The plant is operating at 100% power.
- NCP is in service.
- "A" CCP is tagged Out of Service.
- "B" CCP is in standby.

Which ONE of the following describes the Technical Specification action statements, if any, that apply for this condition?

A. NO Technical Specification Action Statements are in effect.

B. ONLY Technical Specification 3.5.2, ECCS, applies.

C. ONLY Technical Specification 3.5.4, RWST, applies.

D. BOTH Technical Specification 3.5.2 and 3.5.4 apply.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. With one CCP out of service, ITS 3.5.2 is entered. Plausible because the applicant may believe the NCP is a required component for ITS 3.5.2.

B. Correct.

C. Incorrect. Plausible because the applicant may confuse boration flowpath requirements of TRM 3.1.9 with operability of RWST. It is logical to believe that the RWST is not capable of performing its intended function, so the action must be performed for inoperability.

D. Incorrect. Same reason and logic as in Option C above.

Technical Reference(s): ITS 3.5.2 Amendment 123 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1300400 R23 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 8

55.43

Components, capacity, and functions of emergency systems.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	029	2.1.30
	Importance Rating	3.0	

Conduct of Operations: Ability to locate and operate components, including local controls.

Proposed Question: RO Question # 56

Given the following:

- The plant was operating at 100% power.
- A secondary load rejection resulted in a Reactor Protection System parameter exceeding a reactor trip setpoint.
- The reactor did NOT trip.
- Safety Injection is NOT actuated.
- The crew is performing action of EMG FR-S1, Response to Nuclear Power Generation/ATWS.

Which ONE of the following describes an immediate action performed in EMG FR-S1?

Dispatch an operator to locally...

A. open reactor trip and bypass breakers on 2026' Auxiliary Building.

B. open reactor trip and bypass breakers on 2000' Auxiliary Building.

C. de-energize Rod Drive MG Set feeder breakers on 2026' Auxiliary Building.

D. de-energize Rod Drive MG Set feeder breakers on 2000' Auxiliary Building.

Proposed Answer: A

Explanation (Optional):

A. Correct. Reactor Trip and Bypass Breakers are located in the Rod Drive MG Set room, which is on the 2026' Aux Building.

B. Incorrect. Wrong location for Reactor Trip and Bypass Breakers, but plausible because 2000' Aux Building is a location where Rod Drive MG Sets may be de-energized, by tripping breaker on PG-19.

C. Incorrect. This action is performed if Reactor Trip and Bypass Breakers will not trip. Location is correct for PG-20. Also plausible because the crew will open PG breakers from the control room as an immediate action.

D. Incorrect. This action is performed if Reactor Trip and Bypass Breakers will not trip. Location is correct for PG-19. Also plausible because the crew will open PG breakers from the control room as an immediate action.

Technical Reference(s): EMG FR-S1 Rev 15A Step 1 RNO (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732339 R3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	061	AK1.01
	Importance Rating	2.5	

Knowledge of the operational implications of the following concepts as they apply to Area Radiation Monitoring (ARM) System Alarms: Detector limitations

Proposed Question: RO Question # 57

Which ONE of the following describes the type of detector used, and the maximum reading, on Containment High Area Radiation Monitors (CHARMS) GT RE-59 and GT RE-60?

A. Geiger-Mueller;

10^5 mRem/Hr

B. Geiger-Mueller;

10^8 Rem/Hr

C. Ion Chamber;

10^5 mRem/Hr

D. Ion Chamber;

10^8 Rem/Hr

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Area Radiation Monitors are typically G-M detectors, but CHARMS are ion chambers due to the safety related requirement for higher dose measurement post accident.

B. Incorrect. Actual scale is correct but type of detector is for normal area radiation monitor

C. Incorrect. detector type is correct but scale is for normal area radiation monitors

D. Correct.

Technical Reference(s): SY 1407300, Rev 11 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1407300 R2 (As available)

Question Source: Bank # WTSI 56006

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 11

55.43

Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Comments:

WTSI 56006 (WBN Audit 2007)

Facility: Wolf Creek

Vendor: WEC

Exam Date: 08/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	036 AK2.01	
	Importance Rating	2.9	

Knowledge of the interrelations between the Fuel Handling Incidents and the following: Fuel handling equipment

Proposed Question: RO Question # 58

Given the following:

- The plant is in Mode 6.
- Core Alterations are in progress.
- While inserting a fuel assembly in the assigned Spent Fuel Pool rack, the following indications are observed:
  - Fuel Building Area Rad Monitor indications are rising on SD RI-37 & 38.
  - Fuel Building Atmosphere Monitor GG RIC-27 is in HI-HI alarm.
  - Fuel Building Atmosphere Monitor GG RIC-28 is trending up but NOT in alarm.
- The crew enters OFN KE-018, Fuel Handling Accident.

Which ONE of the following describes an action required in accordance with OFN KE-018, and the status of the Fuel Building Isolation System?

A. Fuel Transfer Cart must be in SFP with Fuel Transfer Tube Gate Valve, EC-V995, CLOSED; FBIS actuation has occurred.

B. Fuel Transfer Cart must be in SFP with Fuel Transfer Tube Gate Valve, EC-V995, CLOSED; FBIS actuation has NOT occurred.

C. BOTH AUX/FUEL Normal Exhaust Fans, CGL03A and CGL03B, must be placed in FAST Speed; FBIS actuation has occurred.

D. BOTH AUX/FUEL Normal Exhaust Fans, CGL03A and CGL03B, must be placed in FAST Speed; FBIS actuation has NOT occurred.

Proposed Answer: A

Explanation (Optional):

A. Correct. Although not explicitly called for in procedure for this event, the gate valve cannot be closed if the cart is in Containment. FBIS has occurred because AUTO actuation occurs with 1 out of 2 FHB Atmosphere monitors in alarm.

B. Incorrect. Plausible because first half is correct and also because the candidate may reason that only 1 rad monitor is in alarm in the entire FHB, which may not meet logic for actuation.

C. Incorrect. Plausible because 2nd half is correct, and also because 1 fan could be running in fast speed, but the running fan would be reduced to slow speed.

D. Incorrect. Plausible because the running fan may be in fast speed, but would be reduced to slow speed.

Technical Reference(s): OFN KE-018, Rev 11 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732428 R3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	E14 EK3.1	
	Importance Rating	3.2	

Knowledge of the reasons for the following responses as they apply to the (High Containment Pressure) Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics.

Proposed Question: RO Question # 59

Given the following:

- A LOCA has occurred.
- Due to several component failures, the crew was required to perform EMG C-11, Loss of Emergency Coolant Recirculation.
- A RED condition is received on the Containment CSF Status Tree.
- Containment pressure is 40 psig and STABLE.
- BOTH Containment Spray Pumps were STOPPED while performing EMG C-11.
- RWST Level is 30% and lowering slowly.

Which ONE of the following describes the reason for the current configuration and strategy for reducing Containment Pressure?

BOTH Containment Spray Pumps were turned off to...

A. prevent damage due to cavitation. START <u>both</u> Containment Spray Pumps in accordance with EMG FR-Z1 because Red CSF conditions take precedence over EMG C-11 actions.

B. prevent damage due to cavitation. Remain in EMG C-11 and do NOT perform actions of EMG FR-Z1, because restoration of core cooling takes precedence over maintenance of Containment parameters.

C. conserve RWST inventory. START <u>both</u> Containment Spray Pumps in accordance with EMG FR-Z1 because Red CSF conditions take precedence over EMG C-11 actions.

D. conserve RWST inventory. OPERATE Containment Spray Pumps in accordance with the guidance in EMG C-11, as directed by EMG FR-Z1.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. EMG FR-Z1 Step 3 RNO says operate Pumps IAW EMG C-11. Plausible because a red condition exists on the Containment CSF, and it is logical to start all spray pumps. First part is plausible because EMG C-11 directs stopping Containment Spray as as soon as it is no longer necessary. Cavitation may occur, but the reason for stopping spray is to conserve RWST inventory.

B. Incorrect. First part is plausible because EMG C-11 directs stopping Containment Spray as as soon as it is no longer necessary. Cavitation may occur, but the reason for stopping spray is to conserve RWST inventory. Restoring core cooling is a priority, but actions performed in higher red paths are performed not actions in ECA procedures.

C. Incorrect. First part is correct. Second part is plausible because a red condition exists on the Containment CSF, and it is logical to start all spray pumps.

D. Correct. See descriptions above

Technical Reference(s): EMG FR-Z1, Rev 13 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732350 R3 (As available)

Question Source: Bank # WTSI 48375

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Callaway 2005

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content:	55.41	10
	55.43	

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Changes were made and info was added to options in attempt to meet KA

Facility: Wolf Creek

Vendor: WEC

Exam Date: 08/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	051	AA1.04
	Importance Rating	2.5	

Ability to operate and / or monitor the following as they apply to the Loss of Condenser Vacuum: Rod position

Proposed Question: RO Question # 60

Given the following:

- The plant is at 90% power.
- The Main Turbine is on the load limiter.
- The steam dump select switch is selected to 'Tavg' mode.
- Tavg-Tref mismatch is 0.6°F.
- Bank D rods are at 225 steps with the Rod Control Bank Selector switch in AUTO.
- The following alarms are received:

- ALR 00-116B, COND A VAC LO
- ALR 00-117B, COND B VAC LO

- Turbine First Stage pressure is stable.

Prior to any action by the crew, which ONE of the following describes the plant response and operation of the Rod Control System?

Bank D Control Rods would....

A. INSERT because Tref would be lowering because turbine efficiency is lowering.

B. WITHDRAW because Tref would be rising because turbine efficiency is lowering.

C. REMAIN at 225 steps because Tref is unchanged by change in efficiency.

D. REMAIN at 225 steps because the Main Turbine is on the load limiter.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Tref will not be lowering because plant efficiency is changing. Minor changes in vacuum will not affect first stage impulse pressure on the HP turbine.

B. Incorrect. Tref will not be rising for same reason as option A. Tref derived from First Stage pressure and that will not be changing, since control valve position is not changing.

C. Correct.

D. Incorrect. Turbine on load limit will not prevent load from dropping due to changes in efficiency. Plausible because applicant may assume that load will be held stable, and therefore, rod position.

Technical Reference(s): OFN AF-025, Rev 27 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1610721 R7 (As available)

Question Source: Bank # WTSI 54346

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: McGuire 2003

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	059	AA2.04
	Importance Rating	3.2	

Ability to determine and interpret the following as they apply to the Accidental Liquid Radwaste Release: The valve lineup for a release of radioactive liquid

Proposed Question: RO Question # 61

Given the following:

- The plant is at 100% power.
- The following alarms are received in sequence in the Control Room:

- ALR 00-061B, PROCESS RAD HI
- ALR 00-061A, PROCESS RAD HI HI

- Radiation Monitoring Panel (RM-11R) indication for Steam Generator Blowdown Effluent Monitor, BE RE-52, is RED.
- ALL other Process Radiation monitor indications are GREEN.
- SG Blowdown Isolation Valves, BM HV-1 – BM HV-4, indication is GREEN.
- SG Blowdown Sample Isolation Valve indication is RED.
- ONE SG Blowdown Discharge Pump is running.

Which ONE of the following describes the status of the plant in relation to current plant conditions?

A. Indications are consistent with the alarms received, and the crew will immediately refer to OFN BB-07A, Steam Generator Tube Leakage.

B. Indications are consistent with the alarms received. The crew will refer to OFN BB-07A, Steam Generator Tube Leakage, only upon verification of rising trend on SG Blowdown Monitor or Main Steam Line monitor indication.

C. An Accidental radioactive release may be occurring. SG Blowdown Isolation Valves must be manually closed to terminate the release.

D. An Accidental radioactive release may be occurring. SG Blowdown Discharge Pump must be stopped to terminate the release.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. SG BD Isolation valves should be closed on HI HI Rad from RE-92. Plausible because the SG Blowdown Monitors are not in alarm, and applicant can easily misunderstand all inputs to valve closure circuitry. It is not unusual for sample valves to be closed, as they would only be open when a sample is being drawn. The OFN will be used based on the HI-HI Rad alarm response.

B. Incorrect. SG BD Isolation valves should be closed on HI HI Rad from RE-92. Plausible because the SG Blowdown Monitors are not in alarm, and applicant can easily misunderstand all inputs to valve closure circuitry. It is not unusual for sample valves to be closed, as they would only be open when a sample is being drawn. The OFN will be used based on the HI-HI Rad alarm response, but the step is an 'OR' step, not an 'AND' step for the various monitors.

C. Incorrect. SG Blowdown Valves should have automatically closed on High Radiation

D. Correct. The alarm on BE RE-52 should automatically stop the SG Blowdown Discharge Pump. Since Pump is still running and SG Blowdown Isolation Valves are open, a release may be occurring which may be stopped by stopping the SG Blowdown Discharge Pump..

Technical Reference(s): ALR 00-061A, Rev 16
OFN SP-010, Rev 7A (Attach if not previously provided)

SY 1407300 Rev 11

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1407300 R3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	11
	55.43	

Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	003	2.2.39
	Importance Rating	3.9	

Equipment Control: Knowledge of less than one hour technical specification action statements for systems.

Proposed Question: RO Question # 62

Given the following:

- The plant is at 100% power
- While withdrawing Bank D rods for Tavg control, ONE rod is dropped, and indicates 0 steps on DRPI.
- The crew stabilizes the plant in accordance with the applicable OFN procedure.

In accordance with Technical Specifications, which ONE of the following describes the action required within one hour of this event?

A. Reduce thermal power to less than or equal to 75% RTP.

B. Determine that Shutdown Margin requirements are satisfied OR initiate RCS boration to restore Shutdown Margin.

C. Verify the position of control rods using movable incore detectors OR reduce power to less than or equal to 50% RTP.

D. Determine that Shutdown Margin requirements are satisfied AND initiate RCS boration.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because this action is required if the rod cannot be realigned, but the action is required within 2 hours, not 1

B. Correct. Verify Shutdown Margin or initiate boration to restore shutdown margin

C. Incorrect. Plausible because the action statement is true for a DRPI failure IAW ITS 3.1.7, and action is to reduce power to less than 50% for that Technical Specification LCO.

D. Incorrect. Plausible because both of these actions could be taken, and the requirement is 1 hour. Wrong because either one or the other is performed, not both. Since power must be reduced within 2 hours, the applicant could logically assume that boration is required, and not a contingency

Technical Reference(s): ITS 3.1.4 Amendment 123 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732421 R4 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 2

Facility operating limitations in the technical specifications and their bases.

Comments:

This was written from scratch, but similar to a variety of dropped and misaligned control rod questions that evaluate knowledge of Shutdown Margin requirements

Facility:	Wolf Creek
Vendor:	WEC
Exam Date:	08/2009
Exam Type:	R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	076	AK2.01
	Importance Rating	2.6	

Knowledge of the interrelations between the High Reactor Coolant Activity and the following: Process radiation monitors

Proposed Question: RO Question # 63

With the unit at power, which ONE of the following describes the radiation monitor that will provide direct confirmation of High RCS activity in accordance with OFN-BB-006, RCS High Activity?

A. GT-RE31, Containment Atmosphere

B. AB-RE114-111, SG Steam Line Monitors

C. SJ-RE01, CVCS Letdown Monitor

D. GT-RE59, CTMT High Range Area Monitor (CHARMS)

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because the monitor could provide backup indication if there were a leak in Containment, and high RCS activity would provide a higher indication on this monitor if a leak existed.

B. Incorrect. Plausible because they could provide additional indication if a SG tube leak existed, and the procedure directs increased monitoring of these detectors if a SGTL does exist. However, an ARV would have to be open.

C. Correct

D. Incorrect. High RCS activity will have an effect on the indication of this monitor potentially if a LOCA occurred, or post accident. Fuel failure is determined by these monitors after an accident.

Technical Reference(s): OFN BB-006, Rev 5, entry conditions and note prior to step 1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732416, R2 (As available)

Question Source: Bank # WTSI 58368

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Callaway 2007

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 11

55.43

Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Comments:

Minor changes considered editorial

Facility:	Wolf Creek
Vendor:	WEC
Exam Date:	8/2009
Exam Type:	R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	028	AA2.03	
	Importance Rating	2.8		

Ability to determine and interpret the following as they apply to the Pressurizer Level Control Malfunctions: Charging subsystem flow indicator and controller

Proposed Question: RO Question # 64

Given the following:

- The plant is at 100% power.
- NCP is in service.
- The following alarm is received:

- 00-032C, PZR LO LEV DEV

- BB LI-459A indicates 50% and lowering.
- BB LI-460A indicates 59% and rising.
- BB LI-461 indicates 59% and rising.

Which ONE of the following describes (1) the event in progress, and (2) the operation of Charging Flow Control Valve BG FK-462?

A. (1) PZR level instrument reference leg is leaking
(2) BG FK-462 output is lowering

B. (1) PZR level instrument reference leg is leaking
(2) BG FK-462 output is rising

C. (1) Controlling PZR level channel is failing low
(2) BG FK-462 output is lowering

D. (1) Controlling PZR level channel is failing low
(2) BG FK-462 output is rising

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because one level instrument is reading slightly high, (reference leg leak causes high indication) but the controlling instrument is low, causing the alarm. If the controlling instrument is low, then Charging will be rising to increase level. Charging flow control output will be rising if demand for flow is rising. Output lowering is plausible because CVCS has a reverse acting controller for letdown backpressure. Additionally, the pressurizer level master controller provides input to the controller on low level.

B. Incorrect. Same reason as option A for plausibility of cause, and in this case, charging flow control is correct.

C. Incorrect. Correct failure, and plausible because the applicant may confuse controller output. Since the charging flow controller fails open, the applicant may assume that output lowers when valve opens.

D. Correct.

Technical Reference(s): SY 1301000 Rev 7 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1301000 R5 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 08/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	005	AK1.04	
	Importance Rating	3.0		

Knowledge of the operational implications of the following concepts as they apply to Inoperable / Stuck Control Rod: Definitions of axial imbalance, neutron error, power demand, actual power tracking mode, ICS tracking

Proposed Question: RO Question # 65

Given the following:

- The plant was operating at 100 % power.
- A secondary load rejection occurs.
- Power is reduced to 81% before stabilizing.
- Control rod M-4 in bank D failed to move with the rest of the bank.
- Rod Control Urgent Failure alarm has NOT actuated.
- The following alarms are received:

- ALR 00-078A, PR CHANNEL DEV.
- ALR 00-079C, RPI DEV OR PR TILT.
- ALR 00-080C, RPI ROD DEV.

- Control Bank D is at 202 steps.
- Rod M-4 in Control Bank D is at 222 steps.

Which ONE of the following describes the INITIAL actions required, and the reason for the time limit requirement for realignment of rod M-4?

A. Reactor power is adjusted to match the turbine using boration or dilution. The core is not analyzed for a misaligned control rod.

B. Turbine load is adjusted to match Tref with Tavg. The core is not analyzed for a misaligned control rod.

C. Reactor power is adjusted to match the turbine using boration or dilution. Local xenon redistribution may potentially cause excessive power peaking.

D. Turbine load is adjusted to match Tref with Tavg. Local xenon redistribution may potentially cause excessive power peaking.

Proposed Answer: D

Explanation (Optional):

A. Incorrect: Reactor power is held constant. The concern is xenon peaking.
Plausible: Raising turbine power could be construed as non-conservative. The unanalyzed condition is the reason for the rod bank alignment TS, as opposed to the reason for the short action statement. The core is analyzed for a single most reactive rod being stuck in the worst position.

B. Incorrect: The concern is xenon peaking.
Plausible: The unanalyzed condition is the reason for the rod bank alignment TS, as opposed to the reason for the short action statement. The core is analyzed for a single most reactive rod being stuck in the worst position.

C. Incorrect: Reactor power is held constant.
Plausible: Raising turbine power could be construed as non-conservative, and the applicant may consider primary control using boration as more conservative.

D. Correct.

Technical Reference(s): OFN SF-011, Rev 7, Steps 1-3, 17
TS 3.1.4 and basis (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 01732421 R4 (As available)

Question Source: Bank # WTSI 60465
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: Catawba 2004

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	10
	55.43	

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 08/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	1		
	K/A #	G1	2.1.25	
	Importance Rating	3.9		

Conduct of Operations: Ability to interpret reference materials, such as graphs, curves, tables, etc.

Proposed Question: RO Question # 66

With the plant operating at power, a hydrogen leak on the Main Generator has reduced hydrogen pressure from 75 psig to 60 psig.

Which ONE of the following represents a set of initial Turbine Generator operating parameters that, if remaining unchanged by the leak, are UNACCEPTABLE for continued operation of the Main Generator?

(Reference Provided)

A. Turbine Load of 1100 MWe; AND
Reactive Load of (+)475 MVARs OUT.

B. Turbine Load of 1200 MWe; AND
Reactive Load of 350 (+)MVARs OUT.

C. Turbine Load of 1100 MWe; AND
Reactive Load of (-)475 MVARs IN.

D. Turbine Load of 1200 MWe; AND
Reactive Load of (-)350 MVARs IN.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** When the initial conditions are plotted on Figure 1 - Generator Performance

Curve, the intersecting point is below the limiting Hydrogen Pressure line of the post-leak condition (i.e. 60 psig). Therefore, it is acceptable to operate at this point for both initial and after-leak conditions.

B. **Incorrect.** When the initial conditions are plotted on Figure 1 - Generator Performance Curve, the intersecting point is below the limiting Hydrogen Pressure line of the post-leak condition (i.e. 60 psig). Therefore, it is acceptable to operate at this point for both initial and after-leak conditions.

C. **Correct.** According to SY1504502 (p21-22, 47; Rev 010), Figure 8, the Generator Capability Curve, shows the allowable generator MWe load as a function of hydrogen pressure and reactive power. Reactive power is indicated by either power factor or as leading/lagging VAR flow. As an example, this curve shows that with a load of 1200 MW and a hydrogen pressure of 75 psig, the limit for lagging MVARS is 625 (i.e. the intersecting point of Reactive Load and Turbine Load is below the 75 psig Hydrogen Pressure line). According to OFN AF-025 (p1; Rev 27), the Unit Limitations procedure delineates the limitations on the Turbine Load for operating during various restrictive conditions (such as Reduced Generator Cooling Capability). According to Step 4 of OFN AF-025 (p5; Rev 27), the operator must determine, Using Figure 1 - Generator Performance Curve, that Turbine Generator parameters are Acceptable, and if NOT, employ a corrective strategy in accordance with Attachment G. According to OFN AF-025 (p32; Rev 27), Figure 1 - Generator Performance Curve, this is the only set of initial conditions that are Acceptable initially (i.e. below the 75 psig Hydrogen Pressure line) AND Unacceptable after the stabilized leak pressure of 60 psig (i.e. above the 60 psig Hydrogen Pressure line).

D. **Incorrect.** When the initial conditions are plotted on Figure 1 - Generator Performance Curve, the intersecting point is below the limiting Hydrogen Pressure line of the post-leak condition (i.e. 60 psig). Therefore, it is acceptable to operate at this point for both initial and after-leak conditions.

Technical Reference(s): SY1504502, Rev 10 (p21-22, 47)
OFN AF-025, Rev 27 (p1, 5, 32) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: OFN AF-025 Generator capability curve

Learning Objective: SY1504502-R9 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New	X	
Question History:	Last NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	10
	55.43	

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 08/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	1		
	K/A #	G1	2.1.13	
	Importance Rating	2.5		

Conduct of Operations: Knowledge of facility requirements for controlling vital / controlled access.

Proposed Question: RO Question # 67

You are assigned as an escort for visiting contractors.

Which ONE of the following describes the MAXIMUM number of visitors that you may provide escort for at one time, and the process required for entry to a Vital Area, in accordance with AP 27-001, Escort of Individuals Within the Protected Area?

You may escort a maximum of.....

A. Three (3) visitors; visitors place ACAD in card reader first, prior to entering a Vital Area.

B. Three (3) visitors; escort places ACAD in card reader first, prior to entering a Vital Area.

C. Five (5) visitors; visitors place ACAD in card reader first, prior to entering a Vital Area.

D. Five (5) visitors; escort places ACAD in card reader first, prior to entering a Vital Area.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** Up to 5 visitors may be escorted at one time. 3 is plausible, especially in the context of entry to a vital area, where control of visitors is important. Second half is correct.

B. **Incorrect.** Up to 5 visitors may be escorted at one time. 3 is plausible, especially in the context of entry to a vital area, where control of visitors is important. Second half is wrong but plausible because the escort goes through the turnstile first when entering the Protected Area.

C. **Correct.**

D. **Incorrect.** First part is correct, as an escort may control up to 5 individuals at a time. Second half is wrong but plausible because the escort goes through the turnstile first when entering the Protected Area.

Technical Reference(s): AP 27-001 Rev 10 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO1733210 R2 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 08/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	2		
	K/A #	G2	2.2.35	
	Importance Rating	3.6		

Equipment Control: Ability to determine Technical Specification Mode of Operation.

Proposed Question: RO Question # 68

With a plant startup in progress the following indications exist on the Power Range Nuclear Instrumentation Channels:

- N41 = 6.1%
- N42 = 4.7%
- N43 = 4.9%
- N44 = 4.9%

Subsequently, a plant transient occurs and is stabilized by the operating crew.

After plant stabilization the Power Range Nuclear Instrumentation Channels indicate as follows:

- N41 = 4.8%
- N42 = 4.8%
- N43 = 5.8%
- N44 = 5.6%

Which ONE of the following identifies the plant's Mode of Operation before the transient occurred, AND its Mode of Operation after the plant is stabilized?

A. Mode 1; AND
Mode 1.

B. Mode 1; AND
Mode 2.

C. Mode 2; AND
Mode 1.

D. Mode 2; AND
Mode 2.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible if the operator incorrectly believes that only one Power Range instrument reading greater than 5% is sufficient to satisfy the criteria for entry into Mode 1.

B. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible if the operator incorrectly believes that Mode 1 is entered when at least 3 of 4 Power Ranger channels are equal to or greater than 5%.

C. **Correct.** 1st part correct, 2nd part correct. According to LO1732700 (p18-20; Rev 7), A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel. Technical Specifications identifies the six operational Modes in Section 1.1 (Definitions), page 1.1-7. According to Table1.1-1, the difference between Mode 1 and Mode 2 is determined based on the existing % Rated Thermal Power (%RTP). When %RTP is greater than 5%, the plant is in Mode 1; and when %RTP is less than or equal to 5%, the plant is in Mode 2. According to SY1301501 (p35; Rev 11), there are four Power Range instruments designated N41, N42, N43 and N44. Because of this, some further definition of the Mode Change criteria is necessary to account for variations between the Power Range Instruments. According to LO1732700 (p18-20; Rev 7) and GEN 00-003 (p46; Rev 72) Mode 1 is entered when 2 of 4 power range channels indicate greater than 5% power, and according to LO1732700 (p18-20; Rev 7) and GEN 00-005 (p32; Rev 58), Mode 2 is entered when 3 of 4 power range channels indicate less than or equal to 5%. Based on this, it can be seen that initially the plant is in Mode 2, with only N41 indicating greater than 5%; and subsequently, the plant is in Mode 1, with both N43 and N44 greater than 5%.

D. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible if the operator incorrectly believes that Mode 1 is entered when at least 3 of 4 Power Ranger channels are greater than 5%.

Technical Reference(s):
LO1732700 (p18-20; Rev 7)
SY1301501 (p35; Rev 11)
GEN 00-003 (p46; Rev 72)
GEN 00-005 (p32; Rev 58)

(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO1732700 - R4 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 2

Facility operating limitations in the technical specifications and their bases.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	G2	2.2.36
	Importance Rating	3.1	

Equipment Control: Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.

Proposed Question: RO Question # 69

Given the following:

- The plant is in Mode 5.
- The crew is preparing to commence a plant heatup in accordance with GEN 00-002, Cold Shutdown to Hot Standby.
- TDAFW Pump is out of service for impeller repair.
- "A" EDG is out of service for maintenance on fuel injectors.

Which ONE of the following states the HIGHEST MODE that the plant may achieve with the current equipment alignment WITHOUT performing a Risk Assessment?

A. Mode 1

B. Mode 3

C. Mode 4

D. The plant must remain in Mode 5

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Although having 1 EDG inoperable does not require an immediate shutdown, TS 3.0.4 does not allow entry to a mode unless all equipment required for that mode is operable, or a Risk Assessment is performed.

B. Incorrect. Also plausible for same reasons as Option A. Applicant may confuse note about operability of TDAFW Pump as a required redundant feature.

C. Incorrect. Plausible because Applicant may confuse note about operability of TDAFW Pump as a required redundant feature, since that applies in Modes 1-3.
D. Correct.

Technical Reference(s): ITS 3.8.1 Amendment 155 (Attach if not previously provided)
ITS 3.8.2 Amendment 163
ITS 3.0.4 Amendment 173

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1406400 R10 (As available)

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

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41
55.43 2
Facility operating limitations in the technical specifications and their bases.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	3		
	K/A #	G3	2.3.14	
	Importance Rating	3.4		

Radiation Control: Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Proposed Question: RO Question # 70

Given the following:

- A Steam Generator Tube Rupture has occurred on SG "B".
- The crew has performed all actions of EMG E-3 up to the step to commence depressurization of the RCS.
- All equipment is functioning as designed.

Which ONE of the following describes the status of "B" SG Atmospheric Relief Valve, and the reason for the status?

A. CLOSED with controller in Manual; prevent radioactive release to atmosphere.

B. CLOSED with controller in Manual; ensures minimum RCS subcooling will be maintained when RCS depressurization is initiated.

C. Set at 1160 psig with controller in AUTO; prevent uncontrolled radioactive release due to SG safety valve lifting.

D. Set at 1160 psig with controller in AUTO; ensures minimum RCS subcooling will be maintained when RCS depressurization is initiated.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because it is logical to maintain valve closed but controller will not be in manual. Reason is correct.

B. Incorrect. Same reason as option A, and additionally, reason is plausible because if the ARV stuck open on a ruptured SG, the depressurization would also cause depressurization of the RCS. This would result in loss of RCS subcooling.

C. Correct.

D. Incorrect. Correct for status of valve, but reason is incorrect. Plausible because valve would be placed in manual and closed if it stuck open below 1160 psig, but this is not the reason that the valve is placed in AUTO. The remainder of the steps for SG isolation are correct for this reason.

Technical Reference(s): EMG E-3, Rev 21 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732325 R1 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

This is similar to other bank questions but not the same. Written from scratch.

Facility: Wolf Creek

Vendor: WEC

Exam Date: 08/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	3		
	K/A #	G3	2.3.15	
	Importance Rating	2.9		

Radiation Control: Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question: RO Question # 71

With the plant at 100% power, the following conditions exist:

- A .5 gpm Steam Generator Tube Leak develops.
- GE RE-92, Condenser Off Gas Monitor, enters the high-high alarm condition (Red).
- The operating crew enters the appropriate Off-Normal procedure.

Which ONE of the following will provide the quickest indication of the Steam Generator with the leaking tube?

A. A local radiation survey of the steam lines.

B. The Steam Line Radiation Monitors (AB RE 111-114).

C. The Steam Generator Blowdown Sampling Monitor (SJ RE-02).

D. The Steam Generator Blowdown Process Monitor (BM RE-25).

Proposed Answer: A

Explanation (Optional):

A. **Correct.** According to OFN BB-7A (p6; Rev 8), there are four primary means to identify the leaking Steam Generator; (1) unexpected increase in level (which is NOT plausible with a .5 gpm tube leak), (2) radiation from any SG steamline radiation monitor, (3) radiation from any SG steamline survey, or (4) radiation from any SG sample. According to BD-OFN BB-7A (p14; Rev 1), subsequent recovery actions require that the operator be able to distinguish between the intact Steam Generators and the Steam Generator with the tube leak to minimize primary to secondary leakage. With small leaks it may be necessary to use radiation monitors or to sample. The operator is instructed to try to identify the affected Steam Generator early in this guideline to allow the time necessary to sample if the leak is small. This implies that the use of installed or portable radiation monitors are preferred to the more time consuming sample. Since there is no flow through any of the Blowdown monitors, or steam flow by the Steam Line Monitors, the local survey of the steam lines will provide information necessary to identify the Steam Generator with the leaking tube.

B. **Incorrect.** According to LO1407300 (p28-29; Rev 11), the Steam Line Monitors continuously indicate, detect, and alarm the AREA of a steam release for gamma radiation. These monitors are also known as "plume" monitors or "shine" monitors. A SLM is mounted adjacent to each Main Steam Line Atmospheric Relief Valve (ARV), and will provide indication within the Control Room when there is flow in this line. Under the stated conditions, there is no flow through the atmospheric relief valves, and no abnormal radiation will be indicated.

C. **Incorrect.** According to LO1407300 (p24-25; Rev 11), GE RE-92 will actuate a BSPIS and close BM HV-1, 2, 3, 4 CTMT isolation valves and BM HV-5, 6, 7, and 8 sampling isolation valves. According to LO1407300 (p24-25; Rev 11), this BSPIS signal will also cause flow to be lost through SJ RE-02 and BM RE-25. This is plausible because according to LO1407300 (p16-17; Rev 11), SJ RE-02 will help plant personnel diagnose a steam generator tube leak, according to OFN BB-7A (p12; Rev 8), under certain circumstances the setpoints of SJ RE-02 can be readjusted, and the operator may incorrectly believe that it is, or can be made, available.

D. **Incorrect.** According to LO1407300 (p24-25; Rev 11), GE RE-92 will actuate a BSPIS and close BM HV-1, 2, 3, 4 CTMT isolation valves and BM HV-5, 6, 7, and 8 sampling isolation valves. According to LO1407300 (p24-25; Rev 11), this BSPIS signal will also cause flow to be lost through SJ RE-02 and BM RE-25. This is plausible because according to LO1407300 (p17; Rev 11), BM RE-25 will help plant personnel diagnose a steam generator tube leak, according to OFN BB-7A (p12; Rev 8), under certain circumstances the setpoints of BM RE-25 can be readjusted, and the operator may incorrectly believe that it is, or can be made, available.

Technical Reference(s):
OFN BB-7A (p6, 12; Rev 8)
BD-OFN BB-7A (p14; Rev 1)
LO1407300 (p16-17, 24-25, 28-29; Rev 11) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY1407300 - R2 (As available)

LO1732436 - R4

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

Question History: Last NRC Exam:

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	11
	55.43	

Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 08/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	4		
	K/A #	G4	2.4.25	
	Importance Rating	3.3		

Emergency Procedures / Plan: Knowledge of fire protection procedures.

Proposed Question: RO Question # 72

Given the following:

- The plant at 100% power.
- A fire occurs in the plant.
- The operating crew enters OFN KC-16, Fire Response.
- Position indication for PORV BB PCV-455A and its associated block valve are lost.

In accordance with OFN KC-016, which ONE of the following identifies the indication that determines whether the PORV is spuriously lifting, and the MAXIMUM amount of time allowed to close a spuriously lifting PORV?

A. Pressurizer level;
3 minutes

B. Pressurizer level;
10 minutes

C. Pressurizer pressure;
3 minutes

D. Pressurizer pressure;
10 minutes

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** This is plausible because spurious lifting of the PORV (if it stays open) can eventually depressurize the RCS until saturation is reached under the head, and water is forced out of the pressurizer, causing level to rise. Also, if a LOCA were occurring, level will initially lower.

B. **Incorrect.** Plausible same reason as option A, and also 10 minutes is plausible because it relates to time critical actions on other equipment (Containment Spray pumps.)

C. **Correct.**

D. **Incorrect.** **1**0 minutes is plausible because it relates to time critical actions on other equipment (Containment Spray pumps.) Also, the correct parameter is listed

Technical Reference(s): OFN KC-016 Rev 21 step 5 notes (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO1732426 R3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 08/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	4		
	K/A #	G4	2.4.11	
	Importance Rating	4.0		

Emergency Procedures / Plan: Knowledge of abnormal condition procedures.

Proposed Question: RO Question # 73

Which ONE of the following describes the Off-Normal Procedures (OFNs), AND their use when operating within the EMG network?

A. Generally, there are no immediate action steps in the OFNs; AND
OFNs may be implemented concurrently with the EMGs as long as the actions do not interfere with the performance of the EMGs.

B. Generally, there are no immediate action steps in the OFNs; AND
OFNs can NOT be implemented once the EMGs have been implemented, except when specifically directed by an EMG.

C. Several OFNs have immediate action steps; AND
OFNs may be implemented concurrently with the EMGs as long as the actions do not interfere with the performance of the EMGs.

D. Several OFNs have immediate action steps; AND
OFNs can NOT be implemented once the EMGs have been implemented, except when specifically directed by an EMG.

Proposed Answer: A

Explanation (Optional):

A. **Correct.** 1st part correct, 2nd part correct. According to AP 15C-003 (p4 &17; Rev 17), Steps 4.8.1, 6.4.1 and 6.4.2, operators shall be capable of performing procedure steps identified as immediate actions steps from memory or with the use of memory aids. According to AP 15C-003 (p19; Rev 17), Step 6.5.1, all procedure steps not identified as immediate actions steps should be performed by referring directly to the written procedure. According to AP 15C-003 (p18; Rev 17), Step 6.4.5, the recommended assignment of immediate actions is provided in Attachment A, and according to AP 15C-003 (p37; Rev 17), Attachment A identifies three procedures, all of them EMGs that have immediate actions. According to EMG E-0 (p7; Rev 20), a NOTE is provided before Step 1 identifying Steps 1-4 as immediate actions. Similarly, according to EMG FR-S.1 (p4; Rev 15A), a NOTE is provided before Step 1 identifying Steps 1-2 as immediate actions. Similarly, according to EMG C-0 (p4; Rev 18), a NOTE is provided before Step 1 identifying Steps 1-2 as immediate actions. According to EMG ES-0.2 (p4; Rev 18), Step 2 RNO, the operator is directed to use OFN BG-009. According to EMG ES-0.2 (p12; Rev 18), Step 6 RNO, a similar direction exists to OFN BB-005. According to OFN RP-017 (p16, 20 & 28; Rev 29), Attachments B, C and D have actions that are identified as Immediate Actions Steps. However, they are not to be taken upon entry into the OFN, and only when directed within the body of the procedure. Furthermore, this is the only one of the 40 OFNs that has identified immediate actions. Therefore, generally speaking, there are no immediate actions in the OFNs. According to AP 15C-003 (p15; Rev 17), Step 6.2.3, when operating within the EMGs, the operator may perform OFNs as long as the actions do not interfere with the performance of the EMGs.

B. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because there are several OFNs that are directed from steps within an EMG. For instance, according to EMG ES-0.2 (p4; Rev 18), Step 2 RNO, the operator is directed to use OFN BG-009. According to EMG ES-0.2 (p12; Rev 18), Step 6 RNO, a similar direction exists to OFN BB-005. Because of this, the operator may incorrectly believe that in order for an OFN to be performed while operating within an EMG, is to be specifically directed by the EMG to do so.

C. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because the operator may incorrectly believe that in order for an OFN to be performed while operating within an EMG, it must be specifically directed by the EMG.

D. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because according to AP 15C-003 (p22; Rev 17), Step 6.5.9.2, early performance of a step is allowable under certain conditions, and examples of these are given. For instance, starting CCW or an ESW Pump that does not automatically start when it is supposed to, or throttling AFW flow to limit plant cooldown. Based on this, the operator may incorrectly believe that these are immediate actions. However, this allowance does NOT mean that these actions are immediate actions, that MUST be performed from memory or with the use of memory aids; and the operator may incorrectly believe that in order for an OFN to be performed while operating within an EMG, it must be specifically directed by the EMG.

Technical Reference(s):	15C-003 (p4, 17-19, 22, & 37; Rev 17) EMG ES-0.2 (p4, 12; Rev 18) EMG E-0 (p7; Rev 20) EMG FR-S.1 (p4; Rev 15A) EMG C-0 (p4; Rev 18) LO1732312 (p10; Rev 6)	(Attach if not previously provided)

OFN RP-017 (p16, 20 & 28; Rev 29)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO1732312-R1/2 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 08/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	4		
	K/A #	G4	2.4.34	
	Importance Rating	4.2		

Emergency Procedures / Plan: Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.

Proposed Question: RO Question # 74

Given the following:

- The plant is at 100% power.
- A fire in the Control Room substantially reduces visibility in the room.
- The Shift Manager decides that the Control Room must be evacuated.
- The operating crew enters the appropriate OFN.
- The reactor is manually tripped.
- The MSIVs are closed.

Which ONE of the following identifies the AFW Pump started by the Reactor Operator, AND the affect that this will have on the plant?

A. Turbine Driven AFW Pump; AND
The "B" Steam Generator Level will increase.

B. Turbine Driven AFW Pump; AND
The "D" Steam Generator Level will increase.

C. "B" Motor Driven AFW Pump; AND
The "B" Steam Generator Level will increase.

D. "B" Motor Driven AFW Pump; AND
The "D" Steam Generator Level will increase.

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because both the TDAFW Pump and the B MDAFW Pump are started during this procedure. However, according to OFN RP-017 (p13; Rev 29), Steps A10 and A11, the SRO is directed to start the TDAFW Pump and align its discharge to the B Steam Generator.

B. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because both the TDAFW Pump and the B MDAFW Pump are started during this procedure. However, according to OFN RP-017 (p13; Rev 29), Steps A10 and A11, the SRO is directed to start the TDAFW Pump and align its discharge to the B Steam Generator. While the D Steam Generator level may raise if only the TDAFW Pump is started, this will NOT be the result of an action taken by the RO.

C. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because the operator may incorrectly believe that the B MDAFW Pump start will raise the B Steam Generator Level. According to LO1732427 (p4; Rev 9), the B Train Components are protected during this procedure. According to SY1406100 (p47; Rev 18), the discharge of each MDAFW Pump is normally aligned to two of the four Steam Generators. For instance, B MDAFW Pump is aligned to A and D Steam Generator, while the A MDAFW Pump discharge is aligned to the B and C Steam Generators. While there are cross-connect valves between the pump discharge lines, these valves are normally closed; and NOT opened during the implementation of OFN RP-017. Also, according to OFN RP-017 (p13; Rev 29), Steps A10 and A11, the SRO is directed to start the TDAFW Pump and align its discharge to the B Steam Generator. So the B Steam Generator level will increase during the performance of this procedure; just NOT as a result of the RO starting the B MDAFW Pump.

D. **Correct.** 1st part correct, 2nd part correct. According to OFN RP-017 (p7; Rev 29), Step 5, during the evacuation of the Control Room due to a fire event, the SRO is directed to the Auxiliary Shutdown Panel, and directed to perform the actions of Attachment A. According to OFN RP-017 (p8; Rev 29), Step 6, the Reactor Operator is directed to perform the actions of Attachment C. According to OFN RP-017 (p10; Rev 29), Steps A4 and A5, the SRO is directed to align the B MDAFW Pump to the D Steam Generator. According to OFN RP-017 (p23; Rev 29), Step C13, the RO is directed to start the B MDAFW Pump, after the SRO has aligned the pump discharge to the D Steam Generator in accordance with Steps A4 and A5. With the B MDAFW Pump started, and its discharge aligned to the D Steam Generator, the D Steam Generator Level will increase.

Technical Reference(s): OFN RP-017 (p7, 8, 10, 13 & 23; Rev 29)
SY1406100 (p47; Rev 18)
LO1732427 (p4; Rev 9)
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO1732724 - R3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

The KA is matched because the operator must demonstrate knowledge of RO tasks performed outside the main control room during an emergency (i.e. local start of the B MDAFW Pump) and the resultant operational effects (i.e. D SG Level will increase).

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	2		
	K/A #	G2	2.2.1	
	Importance Rating	4.5		

Equipment Control: Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.

Proposed Question: RO Question # 75

Initial conditions:

- A reactor Startup is in progress in accordance with GEN 00-003, Hot Standby to Minimum Load.
- The reactor is critical.
- Reactor Power is 2E-9 amps on Intermediate Range.

Current conditions:

- During Control Bank "C" withdrawal, ONE (1) Control Bank "C" rod drops to the bottom of the core.
- Reactor power is 8E-8 amps and trending down.

In accordance with GEN 00-003, which ONE of the following actions is required?

A. Trip the reactor.

B. Insert Control Bank "C" only, determine and correct the cause of the dropped rod, and re-perform the startup.

C. Insert ALL Control Banks, determine and correct the cause of the dropped rod, and re-perform the startup.

D. Stabilize reactor power by performing RCS boration/dilution, and enter the applicable OFN.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Reactor trip will not be required unless unexpected criticality occurs. Plausible because the plant is in an abnormal configuration for the event that occurs and tripping the reactor is considered a conservative response.

B. Incorrect. Plausible because Control Bank "C" has the dropped rod and it is inserted, but it is not the only bank inserted.

C. Correct.

D. Incorrect. Thsi action would be correct for conditions at power, as the OFN would first require stabilization of power

Technical Reference(s): GEN 00-003, Rev 72, Precaution 4.7 (Attach if not previously provided)

Proposed References to be provided to applicants during examination:

Learning Objective: LO 1732103 R3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	056	AA2.32
	Importance Rating		4.3

Ability to determine and interpret the following as they apply to the Loss of Offsite Power: Transient trend of coolant temperature toward no-load T-ave

Proposed Question: SRO Question # 76

Given the following:

- A loss of off-site power occurred from 100% power.
- All systems responded as designed.
- NO other failures have occurred.
- The crew entered EMG E-0, Reactor Trip or Safety Injection.
- Off-site power became available immediately after the reactor trip.
- The Shift Manager directed that a cooldown to Mode 5 will be performed due to a pre-existing Maintenance condition, prior to restart.
- The crew is currently performing EMG ES-02, Reactor Trip Response.

Which ONE of the following describes (1) the INITIAL trend of RCS temperature for the first 30 seconds following the trip, and (2) the procedure usage required after transition from EMG ES-02?

RCS Tavg immediately rises due to...

A. Tcold rising, before trending toward no-load Tavg; EMG ES-04, Natural Circulation Cooldown.

B. Tcold rising, before trending toward no-load Tavg; GEN 00-006, Hot Standby to Cold Shutdown.

C. Thot rising, before trending toward no-load Tavg; EMG ES-04, Natural Circulation Cooldown.

D. Thot rising, before trending toward no-load Tavg; GEN 00-006, Hot Standby to Cold Shutdown.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because Tcold would rise immediately after the trip. Procedure transition is plausible because it is the normal transition without RCPs. However, RCPs will be restarted in EMG ES-02 because power is available.

B. Correct. Immediately post trip Thot will lower and Tcold will rise as forced circulation stops. Once natural circulation is being established, Tcold will lower and thot will rise as thermal driving head rises for natural circulation to occur.

C. Incorrect. Plausible because Thot eventually will rise to a value consistent with decay heat and natural circulation flow. A delta T must exist between Thot and Tcold to promote natural circulation. RCPs would be restarted in ES-02, so the procedure transition to ES-04 is incorrect.

D. Incorrect. Plausible because Thot eventually will rise to a value consistent with decay heat and natural circulation flow. A delta T must exist between Thot and Tcold to promote natural circulation. Procedure use for cooldown is correct.

Technical Reference(s): EMG ES-02, Rev 18 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732317 R2 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	011	EA2.14
	Importance Rating		4.0

Ability to determine or interpret the following as they apply to a Large Break LOCA: Actions to be taken if limits for PTS are violated

Proposed Question: SRO Question # 77

Given the following:

- The crew has performed EMG E-0, Reactor Trip or Safety Injection.
- All equipment is running as designed.
- Upon transition to EMG E-1, Loss of Reactor or Secondary Coolant, the following conditions exist:

- RCS temperature is 170°F and lowering slowly.
- RCS pressure is 50 psig and stable.
- RHR flow indicates 6000 GPM.

Which ONE of the following describes the status of the RCS INTEGRITY Critical Safety Function Status Tree, and the NEXT action required upon exit from EMG E-0?

(Reference Provided)

A. A RED condition exists on the RCS INTEGRITY Critical Safety Function Status Tree. Reduce RHR flow and initiate an RCS temperature soak in accordance with EMG FR-P1, Response to Imminent Pressurized Thermal Shock Conditions.

B. A RED condition exists on the RCS INTEGRITY Critical Safety Function Status Tree. Verify RHR flow rate in EMG FR-P1 and then transition back to EMG E-1 due to the Large Break LOCA in progress.

C. An ORANGE condition exists on the RCS INTEGRITY Critical Safety Function Status Tree. Reduce RHR flow and initiate an RCS temperature soak in accordance with EMG FR-P1, Response to Imminent Pressurized Thermal Shock Conditions.

D. An ORANGE condition exists on the RCS INTEGRITY Critical Safety Function Status Tree. Verify RHR flow rate in EMG FR-P1 and then transition back to EMG E-1 due to the Large Break LOCA in progress.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because a RED condition exists on the Integrity CSF Status Tree because Tcold are to the left of Limit A. The applicant will have Limit A as a reference but must understand it's use, as no instructions will be given. Since all systems are operating as designed, RHR flow cannot be reduced and no soak will be required.

B. Correct.

C. Incorrect. Plausible because the applicant must interpret Limit A and the point of RCS pressure and temperature, and for conditions where RCS pressure is above 325 psig or RHR flow is <1000 GPM, this would be a correct action, as conditions represent a large cooldown in a short period of time.

D. Incorrect. Condition is RED because RCS temperature is to the left of limit A for the pressure provided. Action is to verify RHR flow (Large Break LOCA in progress) and return to procedure and step in effect.

Technical Reference(s): EMG F-0, Rev 16 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: EMG F-0, Limit A curve

Learning Objective: LO 1732349 R3, R4 (As available)

Question Source: Bank #

Modified Bank # WTSI 57821 (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis	X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	058	AA2.02
	Importance Rating		3.6

Ability to determine and interpret the following as they apply to the Loss of DC Power: 125V dc bus voltage, low/critical low, alarm

Proposed Question: SRO Question # 78

Given the following:

- The plant is in Mode 3.
- The following alarm is received in the control room:

- 00-025C, NK01 TROUBLE

- NK01 Voltage indicates 119 VDC and lowering slowly.
- Battery NK11 indicates 220 amps.
- Battery Charger NK21 DC Output Breaker is tripped open.

- The following alarms are displayed on NK01:

- 301-02B, CHARGER FAILURE
- 301-04B, CHARGER DC BREAKER OPEN
- 301-06A, CHARGER DC UNDERVOLTAGE

Which ONE of the following describes the operability of the DC Distribution System, and the action required?

A. Declare DC Source to Bus NK01 INOPERABLE because there is no Battery Charger connected. Enter OFN NK-020, Loss of Vital 125 VDC Bus NK01, NK02, NK03, NK04 to align an operable battery charger to Bus NK01.

B. Bus NK01 remains OPERABLE because bus voltage is within Technical Specification LCO limits. Enter OFN NK-020, Loss of Vital 125 VDC Bus NK01, NK02, NK03, NK04 to align an operable battery charger to Bus NK01.

C. Declare DC Source to Bus NK01 INOPERABLE because there is no Battery Charger connected. Align an operable Battery Charger to Bus NK01 in accordance with alarm response procedures and SYS NK-131, Energizing NK Busses.

D. Bus NK01 remains OPERABLE because bus voltage is within Technical Specification LCO limits. Align an operable Battery Charger to Bus NK01 in accordance with alarm response procedures and SYS NK-131, Energizing NK Busses.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because first half is correct, and because the OFN is used for DC Bus failure, In this case, the OFN is only used for a loss of the NK Bus, and not for charger trip.

B. Incorrect. Although battery voltage is above 110 VDC, it is still not operable due to failure of the charger. Technical Specifications 3.8.4, 3.8.6, and 3.8.9 are all reviewed and 3.8.4 is required for entry.

C. Correct.

D. Incorrect. Plausible because the procedure is correct. Wrong because even though the bus is energized by the battery, it is not connected to an operable Battery Charger.

Technical Reference(s): ALR 00-025C, Rev 12A (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1506300 R5 (As available)

Question Source: Bank # WTSI 57088

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Sequoyah 2007

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis	X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Also meets 10CFR55.43(b) item 2

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	E05	2.4.18
	Importance Rating		4.0

Emergency Procedures / Plan: Knowledge of the specific bases for EOPs.

Proposed Question: SRO Question # 79

Given the following:

- A Reactor Trip and Safety Injection have occurred.
- The crew is performing the actions of EMG E-0, Reactor Trip or Safety Injection.
- AFW flow cannot be established.
- SG Wide Range levels are as follows:

- 20% in SG 'A' and SG 'D'.
- 21% in SG 'B' and SG 'C'.

- RCS Pressure is 975 psig and stable.
- Containment pressure is 9 psig.
- Intact SG pressures are 875 psig and stable.
- All other systems are operating as designed.

Which ONE of the following describes the correct procedure entry and action required?

A. Enter EMG FR-H1, Loss of Secondary Heat Sink; Secondary heat removal is ineffective and RCS Bleed and Feed must be initiated.

B. Enter EMG FR-H1, Loss of Secondary Heat Sink; Secondary heat removal may be restored using Main Feed or Condensate prior to initiation of RCS Bleed and Feed.

C. Enter EMG E-1, Loss of Reactor or Secondary Coolant; RCS pressure is low enough that use of EMG FR-H1 is not required, and RCS cooldown will subsequently be performed using EMG ES-11, Post LOCA Cooldown and Depressurization.

D. Enter EMG E-1, Loss of Reactor or Secondary Coolant; RCS pressure is low enough that use of EMG FR-H1 is not required, and RCS cooldown will subsequently be performed using EMG ES-12, Transfer to Cold Leg Recirculation.

Proposed Answer: A

Explanation (Optional):

A. Correct. Bleed and feed criteria is 26% on 3 of 4 SGs. SG pressure is below RCS pressure, so EMG FR-H1 is required.

B. Incorrect. Bleed and feed is immediately required. The crew may continue attempts to initiate feed, condensate, and AFW, but bleed and feed should not be delayed.

C. Incorrect. Plausible because RCS pressure is below normal SG pressure and close to actual SG pressure. Procedure transition to EMG ES-11 is correct if the applicant misunderstands entry to EMG FR-H1.

D. Incorrect. Same as option C above, but EMG ES-11 would not be entered if all other ECCS was operating correctly. RWST level would not be low enough for a long period of time since the only flow to RCS at 975 psig would be Charging and SI. (lower flow than RHR to reduce RWST level)

Technical Reference(s): EMG F-0, Rev 16 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732346 R1 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Meets KA because it gets to the basis of the procedure and reason for using it.

We have somewhat similar items in bank that distinguish which procedure to use but none in this context.

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	009	2.1.25
	Importance Rating		4.2

Conduct of Operations: Ability to interpret reference materials, such as graphs, curves, tables, etc.

Proposed Question: SRO Question # 80

Given the following:

- Reactor Trip and Safety Injection have occurred.
- The crew is performing actions of EMG ES-11, Post LOCA Cooldown and Depressurization.
- The crew is evaluating operation of SI Pumps.
- RCS temperature is 535°F. (Highest CET)
- RCS pressure has gradually lowered, and is currently 1500 psig.
- Containment pressure is 5.8 psig and stable.
- Pressurizer level indicates 58% and slowly rising.
- Both SI Pumps are running.
- All actions have been taken as required in EMG ES-11 to this point.

Which ONE of the following describes the action that will be taken with regard to the SI pumps in accordance with EMG ES-11?

(Reference Provided)

A. BOTH SI Pumps must remain running until RCS cooldown establishes sufficient subcooling.

B. ONE SI Pump may be stopped at this time; the second SI Pump may be stopped after verifying sufficient subcooling still exists in EMG ES-11.

C. ONE SI Pump may be stopped at this time; the second SI Pump must remain in operation until stopped just prior to RHR cooling being placed in service.

D. BOTH SI Pumps must remain running until stopped just prior to RHR cooling being placed in service.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because the applicant may believe that RCPs were tripped due to LOCA. RCPs would NOT have been tripped based on RCS pressure >1400 psig. The applicant must infer that either RCPs were NOT tripped, or that if they were, they would have been restarted in EMG ES-11 prior to this point.

B. Correct. RCS subcooling is currently 61-62°F, and 60°F (adverse containment exists) is required for stopping the first SI Pump.

C. Incorrect. RCS cooldown will continue and SI Pump will be stopped to prevent overfilling the PZR at some point prior to placing RHR in service, as PZR is currently filling and adequate RCS subcooling remains. Plausible because conditions only allow trip of 1 SI Pump based on current subcooling value.

D. Incorrect. RCS cooldown will continue and SI Pump will be stopped to prevent overfilling the PZR at some point. Plausible because if applicant does not realize that RCPs are running, this is logical, since no SI Pumps would be tripped.

Technical Reference(s): EMG ES-11, Rev 17 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: EMG ES-11, Rev 17, page 36 of 94 (Step 21)

Learning Objective: LO 1732321 R3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	
	55.43	5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: S

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			1
	Group #			1
	K/A #	E12	2.4.2	
	Importance Rating			4.6

Emergency Procedures / Plan: Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.

Proposed Question: SRO Question # 81

Given the following:

- A reactor trip has occurred from 100% power.
- The crew has entered EMG E-0, Reactor Trip or Safety Injection.
- RCS pressure is 1880 psig and lowering.
- Containment pressure is 0.2 psig and stable.
- All SG pressures are 820 psig and lowering.
- All SG Narrow Range levels are off-scale low.
- The TDAFW pump tripped upon starting.
- All other equipment is running as required.

- Subsequently, Main Steam Isolation Valve ALL CLOSE button is pressed.
- RCS pressure and SG pressure trends continue as prior to the MSIVs being closed.
- The crew manually initiates Safety Injection.

Assuming the current trends continue, which ONE of the following describes the procedure usage to mitigate this event upon transition from EMG E-0?

A. EMG E-2, Faulted Steam Generator Isolation, to EMG E-1, Loss of Reactor or Secondary Coolant.

B. EMG E-2, Faulted Steam Generator Isolation, to EMG C-21, Uncontrolled Depressurization of All Steam Generators.

C. EMG ES-03, SI Termination, to GEN 00-006, Hot Standby to Cold Shutdown.

D. EMG E-1, Loss of Reactor or Secondary Coolant, to EMG E-2, Faulted Steam Generator Isolation, to EMG ES-03, SI Termination.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. EMG E-2 will require transition to EMG C-21 because none of the SG pressures are stable or rising. Plausible because this is the normal transition from EMG E-2 and the correct transition from EMG E-0.

B. Correct.

C. Incorrect. Plausible because SG and RCS pressures are high enough to consider SI Termination, but they are trending down. EMG ES-03 would also be a plausible transition directly from EMG E-0.

D. Incorrect. Plausible because RCS pressure is lowering and this transition comes after transition to EMG E-2.

Technical Reference(s): EMG E-0, Rev 20 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732324 R2 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	037	AA2.05
	Importance Rating		3.3

Ability to determine and interpret the following as they apply to the Steam Generator Tube Leak: Past history of leakage with current problem

Proposed Question: SRO Question # 82

Given the following:

- The plant is at 100% power.
- The following conditions have been observed:

- Condenser Air Discharge Radiation Monitor GE RE-92 went into alert 4 and 1/2 hours ago.
- The crew is performing OFN BB-07A, Steam Generator Tube Leakage.
- Primary to Secondary leakage into "B" SG has been identified as follows:

- 21 gallons per day 4 hours ago.
- 41 gallons per day 3 hours ago.
- 62 gallons per day 2 hours ago.
- 82 gallons per day for the last 60 minutes.

Which ONE of the following describes the required action in accordance with OFN BB-07A?

(Reference Provided)

A. Shutdown the plant in accordance with GEN 00-004, Power Operation or OFN MA-038, Rapid Plant Shutdown. Be in Mode 3 within 24 hours.

B. Hold power stable and direct Chemistry to perform STS CH-033, Primary to Secondary leakage determination, to determine the affected Steam Generator.

C. Shutdown the plant in accordance with OFN MA-038, Rapid Plant Shutdown. Be in Mode 3 in less than 6 hours.

D. Reduce power to 50% within 1 hour in accordance with OFN MA-038, Rapid Plant Shutdown. Be in Mode 3 within the next 2 hours.

Proposed Answer: A

Explanation (Optional):

A. Correct. For leak rates >75 GPD but increasing leak rate less than 30 GPD/Hr,a shutdown to Mode 3 is required within 24 hours. Either GEN 00-004 or OFN MA-038 may be used.

B. Incorrect. Plausible because Chemistry will be performing STS CH-033 during this event and continuously monitoring leakage. Typically an actual leak rate can only be determined if power and Tavg are held stable while performing inventory balance. Incorrect because leak rate is high enough that power will not be held stable.

C. Incorrect. Plausible because attachment C directs this action in step C6, but action is only required for leak rates >150 GPD.

D. Incorrect. Plausible because attachment C directs this action in step C5, but action is only required if leakage is increasing by >30 GPD/Hr sustained for 30 minutes.

Technical Reference(s): OFN BB-07A, Rev. 8, steps 1-9, and attachment C (Attach if not previously provided)

Proposed References to be provided to applicants during examination: OFN BB-07A, Attachment C

Learning Objective: LO 1732436 R3 (As available)

Question Source: Bank # WTSI 52741

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Harris 2004

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

WTSI 52741

Facility:	Wolf Creek
Vendor:	WEC
Exam Date:	8/2009
Exam Type:	S

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			1
	Group #			2
	K/A #	E10	EA2.1	
	Importance Rating			3.9

Ability to determine and interpret the following as they apply to the (Natural Circulation with Steam Void in Vessel with/without RVLIS) Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Proposed Question: SRO Question # 83

Given the following:

- A reactor trip has occurred due to a loss of offsite power.
- The operating crew is performing actions of EMG ES-04, Natural Circulation
- Cooldown.
- "A" Train RVLIS is out of service.
- The crew has commenced RCS cooldown and depressurization.
- RCS pressure is 1780 psig and trending DOWN.
- RCS Tavg is 448 °F and trending DOWN.
- All SG pressures are at 430 psig and trending DOWN slowly.
- Pressurizer level is 35% and trending UP slowly.
- All CRDM Cooling fans are in service.
- The Shift Manager directs that RCS cooldown rate MUST be performed at approximately 90 °F/hr due to secondary inventory concerns.

Based on current plant conditions, which ONE of the following actions will be required in accordance with EMG ES-04?

A. Remain in EMG ES-04 to continue cooldown and depressurization.

B. Actuate Safety Injection and go to EMG E-0, Reactor Trip or Safety Injection.

C. Transition to EMG ES-06, Natural Circulation Cooldown With Steam Void In Vessel (With RVLIS) to continue cooldown.

D. Transition to EMG ES-05, Natural Circulation Cooldown With Steam Void In Vessel (Without RVLIS) to continue cooldown.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. EMG ES-04 does not permit cooldown at the rate required (>50 °F/hr). (See step 8 of EMG ES-04, RCS cooldown rate is <50 °F/hr; step 13 and the NOTE prior to step 15. The NOTE directs performance of EMG ES-06 or EMG ES-05 if the cooldown rate must be performed at a rate > 50 °F/hr.)

B. Incorrect. RCS pressure is at or near the point where an SI would normally be required, the SI actuation criteria in EMG ES-04 is pressurizer level <6% or RCS subcooling <30°F, which are not met. With SG pressure and and RCS temperature lowering, the applicant has to determine whether the SG is leading the RCS or the RCS is causing the cooldown.

C. Correct. Since RCS cooldown must be performed at a rate greater than 50 °F/hr, transition to EMG ES-06 is required. (See ES-06, step 8, RCS cooldown rate is <100 °F/hr)

D. Incorrect. Plausible because this would be correct if RVLIS were unavailable, but one train of RVLIS is available (train "B").

Technical Reference(s): EMG ES-04, Rev 14 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732317 R3 and 11 (As available)

Question Source: Bank # WTSI 61875

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Harris 2008

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 08/2009

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	E06	2.1.28
	Importance Rating		4.1

Conduct of Operations: Knowledge of the purpose and function of major system components and controls.

Proposed Question: SRO Question # 84

Given the following:

- A LOCA has occurred.
- ECCS has NOT functioned as required.
- All RCP's are TRIPPED.
- CETC's indicate 626°F.
- RVLIS Natural Circulation Range is 39%.
- Containment pressure is 6 psig and rising slowly.
- PZR PORVs indicate closed.
- All SG pressures are approximately 1070 psig and stable.
- Total AFW flow is 280,000 lbm/Hr.
- SG NR levels are 13%, 18%, 11%, and 17%, respectively.

Which ONE of the following procedures will the crew implement for these conditions, and how will the PZR PORVs be operated in accordance with the procedure used?

A. EMG FR-C1, Response To Inadequate Core Cooling; PORVs are manually isolated by manually closing the associated block valve to prevent RCS inventory loss.

B. EMG FR-C1, Response To Inadequate Core Cooling; PORVs are allowed to operate in AUTO for RCS pressure control.

C. EMG FR-C2, Response To Degraded Core Cooling; PORVs are manually isolated by manually closing the associated block valve to prevent RCS inventory loss.

D. EMG FR-C2, Response To Degraded Core Cooling; PORVs are allowed to operate in AUTO for RCS pressure control.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. EMG-FR-C1 is plausible because the SRO applicant must have detailed knowledge of the Core Cooling CSF Status Tree, and EMG FR-C1 would be entered if core exit thermocouple temperature was >712°F. Also, PORV isolation is plausible because a LOCA is in progress.

B. Incorrect. While the option has the correct operation of the PORVs, the procedure entry is incorrect because of core exit thermocouple temperature.

C. Incorrect. PORVs are not isolated unless failed open. They are operated in accordance with EMG FR-C2 in AUTO. Procedure entry is correct for condition presented.

D. Correct.

Technical Reference(s): EMG FR-C2, Rev 12, Step 4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732341, R6 and R7 (As available)

Question Source: Bank # WTSI 57858

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: BVPS-1 2007

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	076	2.4.4
	Importance Rating		4.7

Emergency Procedures / Plan: Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.

Proposed Question: SRO Question # 85

Given the following:

- The plant is operating at 60% power following a secondary load rejection.
- Reactor power was stabilized 2 hours ago.
- The following alarm is received:

- 00-061B, PROCESS RAD HI

- Chemistry sample of the RCS indicates the following:

- 85 microcuries per gram Dose Equivalent Iodine 131
- 212 microcuries per gram Dose Equivalent Xe-133

Which ONE of the following describes the procedure required for the plant conditions, and the actions required?

A. Enter OFN BB-006, High Reactor Coolant Activity; maximize letdown and place the plant in Hot Standby within 6 hours.

B. Enter OFN BB-006, High Reactor Coolant Activity; isolate letdown and refer to AP 07B-003, Offsite Dose Calculation Manual, for additional compensatory actions.

C. Enter OFN SP-010, Accidental Radioactive Release; maximize letdown and place the plant in Hot Standby within 6 hours.

D. Enter OFN SP-010, Accidental Radioactive Release; isolate letdown and refer to AP 07B-003, Offsite Dose Calculation Manual, for additional compensatory actions.

Proposed Answer: A

Explanation (Optional):

A. Correct. ITS requires shutdown within 6 hours if DE I-131 is >60 microcuries/gram. For Letdown Monitor, correct procedure is OFN BB-006. See ITS 3.4.16 Action C.

B. Incorrect. Plausible because isolating Letdown is logical if high radiation is on the letdown line, and ODCM would be referred to for most radiation alarms.

C. Incorrect. Procedure is wrong but logical because the limit for RCS activity is based on off-site dose for a steam generator tube rupture. Action referred to in option C is correct.

D. Incorrect. Plausible for same reasons describes in options B and C.

Technical Reference(s): OFN BB-006, Rev 5 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732416, R2 and R3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:
Also meets 10CFR55.43(b) item 2

Facility:	Wolf Creek
Vendor:	WEC
Exam Date:	8/2009
Exam Type:	S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	007	A2.02
	Importance Rating		3.2

Ability to (a) predict the impacts of the following malfunctions or operations on the P S; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Abnormal pressure in the PRT

Proposed Question: SRO Question # 86

Given the following:

- A Reactor Trip and Safety Injection actuation have occurred.
- The crew is performing actions contained in EMG E-0, Reactor Trip or Safety Injection.
- RCS pressure is 1800 psig and increasing.
- Pressurizer level is 7% and increasing.
- SG pressures are 1000 psig and stable.
- SG levels are being controlled by AFW, with NR SG levels at 26%.
- Containment pressure is 0.3 psig and stable.
- PRT pressure is 24 psig and stable.
- PRT temperature is 200ºF and stable.
- PRT level is 71%.

Which ONE of the following describes the action and procedure use required for these conditions?

A. Continue in EMG E-0 until transition to ES-03, SI Termination, and determine cause of PRT conditions.

B. Transition to EMG E-1, Loss of Reactor or Secondary Coolant, due to RCS pressure abnormally low with PRT conditions abnormal.

C. Transition to EMG C-12, LOCA Outside Containment, due to RCS pressure abnormally low with Containment parameters normal.

D. Transition to EMG ES-11, Post LOCA Cooldown and Depressurization, to restore Charging and Letdown and secure ECCS pumps.

Proposed Answer: A

Explanation (Optional):

A. Correct. For these conditions, PRT parameters will be investigated as the crew continues through EMG E-0.

B. Incorrect. Plausible because EMG E-1 is the normal transition for conditions with abnormally low RCS pressure, but in this case, containment conditions do not exist for transition to EMG E-1.

C. Incorrect. Plausible because this procedure may also be entered from the same area of EMG E-0, but transition to EMG ES-03 would be performed first. EMG C-12 is logical when RCS pressure is lowering with no containment conditions being abnormal.

D. Incorrect. This procedure will accomplish the action required to reduce RCS temperature and pressure after a SBLOCA, but is not entered directly from EMG E-0.

Technical Reference(s): EMG E-0, Rev 21 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732313 R4 (As available)

Question Source: Bank # WTSI 57757

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Wolf Creek 2007

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 08/2009

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	010	A2.03
	Importance Rating		4.2

Ability to (a) predict the impacts of the following malfunctions or operations on the PZR PCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: PORV failures

Proposed Question: SRO Question # 87

Given the following:

- A Steam Generator Tube Rupture is in progress on SG "C".
- RCPs were tripped due to a loss of CCW flow.
- EMG E-3, Steam Generator Tube Rupture, is being performed.
- Cooldown to target temperature is complete.
- RCS is being depressurized using PORV BB PCV-455A.
- Upon closing BB PCV-455A, the following indications are observed:

- RCS pressure continues to lower.
- 00-035B, PORV OPEN, is LIT.
- PRT temperature and pressure are rising.
- PZR PORV discharge temperature is rising.

- The RO places the PORV Block Valve, HV-8000A, in CLOSE.
- RCS and PRT conditions continue to degrade.

Based upon current plant conditions, which ONE of the following describes the operational concern associated with this failure, and the NEXT transition that will be required?

A. RCS subcooling may be lost and ECCS pump may be cycled unnecessarily; Go to EMG C-31, SGTR with Loss of Reactor Coolant - Subcooled Recovery Desired.

B. RCS subcooling may be lost and ECCS pump may be cycled unnecessarily; Go to EMG C-32, SGTR with Loss of Reactor Coolant - Saturated Recovery Desired.

C. Rupture of the PRT may result in contamination of Containment; Go to EMG C-31, SGTR with Loss of Reactor Coolant - Subcooled Recovery Desired.

D. Rupture of the PRT may result in contamination of Containment; Go to EMG E-1, Loss of Reactor or Secondary Coolant.

Proposed Answer: A

Explanation (Optional):

A. Correct.

B. Incorrect. Correct operational concern for a PORV failure while depressurizing in EMG E-3. Incorrect procedure transition. Crew will go to EMG C-31 before any requirement to go to C-32 is met for such reasons as secondary inventory for cooldown is limited.

C. Incorrect. Plausible because a caution prior to the step warns of containment parameters changing if PRT ruptures, but this is not cause for transition to another procedure. PRT could rupture even if PORV closed as required, depending on how long it was open. Procedure transition is correct.

D. Incorrect. Plausible because a caution prior to the step warns of containment parameters changing if PRT ruptures, but this is not cause for transition to another procedure. PRT could rupture even if PORV closed as required, depending on how long it was open. Procedure transition is plausible because the note is placed in EMG E-3 to ensure that an inadvertent transition to EMG E-1 is not made.

Technical Reference(s): EMG E-3, Rev 21 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1301000 R9 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	
	55.43	5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Facility:	Wolf Creek
Vendor:	WEC
Exam Date:	8/2009
Exam Type:	S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	022	2.4.9
	Importance Rating		4.2

Emergency Procedures / Plan: Knowledge of low power / shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

Proposed Question: SRO Question # 88

Given the following:

- A Plant Shutdown is in progress.
- RCS pressure is 465 psig.
- RCS temperature is 370°F.
- RCPs 'B' and 'D' are in service.

- The following alarm is received:

- 00-032C, PZR LO LEV DEV

- Pressurizer level is 16% and lowering.

Which ONE of the following describes the procedure required to mitigate this event, and the strategy for operation of Containment Ventilation?

A. OFN BB-007, RCS Leakage High; Ensure all available Containment Fan Coolers are running at the required speed.

B. OFN BB-007, RCS Leakage High; Ensure at least ONE Containment Atmosphere Control Fan is running.

C. OFN BB-031, Shutdown LOCA; Ensure all available Containment Fan Coolers are running at the required speed.

D. OFN BB-031, Shutdown LOCA; Ensure at least ONE Containment Atmosphere Control Fan is running.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because this procedure would be used for an RCS leak in Modes 1 or 2. The Fan Coolers are required when Containment atmosphere is subject to a leak.

B. Incorrect. Same reason as option A, and atmosphere control fan may be operated, but is not required. Procedures requires evaluation of availability, not direction to start one.

C. Correct.

D. Incorrect. Plausible because correct procedure entry requirement, and atmosphere control fan may be operated, but is not required. Procedures requires evaluation of availability, not direction to start one.

Technical Reference(s): OFN BB-031, Rev 17 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732425 R7, R8 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	003	2.4.30
	Importance Rating		4.1

Emergency Procedures / Plan; Knowledge of events related to system operation / status that must be reported to internal organizations or external agencies, such as the state, the NRC, or the transmission system operator.

Proposed Question: SRO Question # 89

Given the following:

- The plant is operating at 55% power.
- The following alarm is received:

- 00-041C, SEAL WTR INJ TEMP HI

- Seal Water Injection Temperature, BG TB-216, indicates 135°F and rising slowly.

- Subsequently, the following alarm is received:

- 00-072A, RCP #1 SEAL FLOW HI

- RCP 'A' #1 seal leakoff, BG FR-157, indicates 8.1 GPM and rising.
- RCP 'A' #1 seal and bearing water temperature is 195°F and rising slowly.

Which ONE of the following describes the action required, and the MAXIMUM amount of time allowed prior to notifying the NRC?

(Reference Provided)

A. Controlled Plant Shutdown is required; trip the RCP when reactor power is below P-8; NRC must be notified within no greater than ONE hour after the RCP is tripped.

B. Controlled Plant Shutdown is required; trip the RCP when reactor power is below P-8; NRC must be notified within no greater than FOUR hours after the RCP is tripped.

C. Reactor Trip is required; NRC must be notified within no greater than ONE hour after the reactor is tripped.

D. Reactor Trip is required; NRC must be notified within no greater than FOUR hours after the reactor is tripped.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because each condition with the exception of #1 seal leakoff require a controlled plant shutdown, but there is no basis for a 1 hour report. This will be a reactor trip/RCP trip. Four hour call.

B. Incorrect. RCP does meet the immediate trip criteria of OFN BB-005. Reactor trip requires a four hour report.

C. Incorrect. Immediate RCP trip is required, and since reactor power is above P-8, a reactor trip will also be required. No 1 hour report is required for this event.

D. Correct.

Technical Reference(s): OFN BB-005, Rev 16, Attachment B (Attach if not previously provided)

Proposed References to be provided to applicants during examination: OFN BB-005, Att E

Learning Objective: LO 1732415 R3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	039	2.1.25
	Importance Rating		4.2

Conduct of Operations: Ability to interpret reference materials, such as graphs, curves, tables, etc.

Proposed Question: SRO Question # 90

Given the following:

- The plant is at 100% power.
- Engineering has identified calculation errors during the previous Main Steam Safety Valve (MSSV) testing for SG 'A'.
- The following describes the actual corrected setpoints for 'A' SG MSSVs.

- AB-V0055 1150 psig
- AB-V0056 1146 psig
- AB-V0057 1175 psig
- AB-V0058 1182 psig
- AB-V0059 1228 psig

- All other safety valves are operable

Which ONE of the following describes the HIGHEST allowable reactor power after action is taken in accordance with ITS, and the ITS basis for the response?

(Reference Provided)

A. 87% RTP;

Ensures that safety analysis assumptions for 10CFR100 dose limits will be met in the event of off-site release following a steam generator tube rupture.

B. 65% RTP;

Ensures that safety analysis assumptions for 10CFR100 dose limits will be met in the event of off-site release following a steam generator tube rupture.

C. 87% RTP;

prevent a challenge to RCS Pressure and Reactor Core Safety Limits for anticipated operational occurrences.

D. 65% RTP;

prevent a challenge to RCS Pressure and Reactor Core Safety Limits for anticipated operational occurrences.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. SVs AB-0056 and AB-0058 are not within lift settings +/-3%. Action for 3 OPERABLE SVs is reduce power to 65% and adjust trip setpoint to a maximum of the same value within 4 hours. Basis is for BOTH safety limits. The basis supplied is incorrect but plausible because it is the basis for RCS activity and the reason for reducing RCS temperature due to high RCS activity.

B. Incorrect. Correct power but incorrect basis. The basis supplied is incorrect but plausible because it is the basis for RCS activity and the reason for reducing RCS temperature due to high RCS activity.

C. Incorrect. Incorrect power but correct basis. Plausible because the value for 1 safety valve inoperable is provided, and the applicant may determine that only 1 safety valve is inoperable based on incorrect calculations.

D. Correct.

Technical Reference(s): TS 3.7.1, amendment 123 and basis (Attach if not previously provided)

TS bases 2.1.1 and 2.1.2

Proposed References to be provided to applicants during examination: ITS Table 3.7.1-1 and Table 3.7.1-2

Learning Objective: SY 1503900 R5 (As available)

Question Source:	Bank #	WTSI 59322	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam: McGuire 2008

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	
	55.43	2

Facility operating limitations in the technical specifications and their bases.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	075	A2.01
	Importance Rating		3.2

Ability to (a) predict the impacts of the following malfunctions or operations on the circulating water system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of intake structure

Proposed Question: SRO Question # 91

Given the following:

- The plant is operating at 100% power.
- The following alarms are received:

- 00-098B, SSE
- 00-098A, R SPCTRM SSE EXCEED
- 00-098D, OBE
- 00-098C, R SPCTRM OBE EXCEED
- 00-098E, SEISMIC RECORDER ON

- Effects of an earthquake were felt throughout the plant.
- Equipment inspection is complete.
- There is no visible damage inside the plant.

- Subsequently, Main Condenser Vacuum is 2.4 inches HgA and degrading slowly.
- Main Condenser temperature is 128°F and rising slowly.
- Cooling Lake level indicates approximately 1080 feet and decreasing.

Which ONE of the following describes the impact on the plant, and the action required?

A. Plant systems essential for shutdown may be damaged; maintain reactor power and turbine load stable and evaluate plant status in accordance with OFN SG-003, Natural Events.

B. Failure of the Cooling Lake Dam has occurred; Place Essential Service Water in service and initiate a plant shutdown in accordance with GEN 00-004, Power Operation, or OFN

MA-038, Rapid Plant Shutdown, as directed by OFN SG-003, Natural Events.

C. Plant systems essential for shutdown may be damaged; inspect systems required for safe shutdown in accordance with OFN SG-003, Natural Events, and go to OFN AF-025, Unit Limitations, for the rising condenser pressure.

D. Potential failure of the Cooling Lake Dam is anticipated; maintain reactor power and turbine load stable while placing Essential Service Water in service in accordance with OFN SG-003, Natural Events.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausble because the procedure is correct and the procedure has steps that require this action, but in this case, additional damage has caused imminent loss of Circ Water due to loss of the cooling lake.

B. Correct. With level less than 1080 feet, the intake structure will cease to function to provide Circulating Water.

C. Incorrect. Plausible because OFN AF-025 is the normal procedure for reducing load based on rising condenser pressure, and also, OFN SG-003 is required.

D. Incorrect. Plausible because all of the actions are correct except for maintaining load stable. Actions do require maintaining load stable within the procedure, but a plant shutdown is required by the conditions in the stem of the item.

Technical Reference(s): OFN SG-003, Rev 15 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732413, R2, R3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	
	55.43	5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	011	2.1.28
	Importance Rating		4.1

Conduct of Operations: Knowledge of the purpose and function of major system components and controls.

Proposed Question: SRO Question # 92

Which ONE of the following describes the applicability and Technical Specification Basis for the High Pressurizer Water Level reactor trip?

A. Modes 1 and 2; Provides primary protection for loss of load events and ensures that on a PZR level channel failure, the PZR safety valves will not lift prior to the PZR High Pressure reactor trip.

B. Modes 1 and 2; Provides a backup trip to PZR High Pressure reactor trip and ensures that water relief through the PZR safety valves will not occur.

C. Mode 1 above P-7; Provides a backup trip to PZR High Pressure reactor trip and ensures that water relief through the PZR safety valves will not occur.

D. Mode 1 above P-7; Provides primary protection for loss of load events and ensures that on a PZR level channel failure, the PZR safety valves will not lift prior to the PZR High Pressure reactor trip.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because the pressurizer high pressure trip actually provides this function, (as well as turbine trip-reactor trip) and the high water level trip is provided as a backup to pressurizer high pressure. Also, in modes 1 and 2, the reactor is at normal operating temperature and pressure.

B. Incorrect. Plausible because the basis is correct, but incorrect because the trip is required above P-7, not in Modes 1 and 2. Mode is plausible for same reason as option A.

C. Correct.

D. Incorrect. Actual requirement for operability is correct but the basis is related to the pressurizer high pressure reactor trip.

Technical Reference(s): TS 3.3.1 Amendment 123 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SY 1301200 R6 (As available)

Question Source: Bank # WTSI 56251

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Robinson 2007

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 2

Facility operating limitations in the technical specifications and their bases.

Comments:

56251

Facility:	Wolf Creek
Vendor:	WEC
Exam Date:	8/2009
Exam Type:	S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	002	A2.03
	Importance Rating		4.3

Ability to (a) predict the impacts of the following malfunctions or operations on the RCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of forced circulation

Proposed Question: SRO Question # 93

Given the following:

- A plant heatup is in progress in accordance with GEN 00-002, Cold Shutdown to Hot Standby.
- RCP 'D' is in operation.
- Preparations are being made to start RCP 'B'.

The following alarms are received:

- 00-070B, RCP VIB/SYS ALERT
- 00-070A, RCP VIB DANGER

- RCP "D" Frame Vibration indicates 5 mils and slowly rising.
- RCP "D" Shaft Vibration indicates 18 mils and slowly rising.

Which ONE of the following describes the action required, and the operational concern of this condition?

A. An immediate shutdown of RCP "D" is required in accordance with OFN BB-005, RCP Malfunctions, Attachment B; DNBR may be reduced below the limit required for safety analysis assumptions.

B. A Controlled shutdown of RCP "D" is required in accordance with OFN BB-005, RCP Malfunctions, Attachment C; DNBR may be reduced below the limit required for safety analysis assumptions.

C. An immediate shutdown of RCP "D" is required in accordance with OFN BB-005, RCP Malfunctions, Attachment B; RCS boron stratification may occur with loss of forced circulation.

D. A Controlled shutdown of RCP "D" is required in accordance with OFN BB-005, RCP Malfunctions, Attachment C; RCS boron stratification may occur with loss of forced circulation.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because DNBR is a Mode 1 and 2 basis for operability of RCS loops. Since GEN 00-002 is in progress, the plant is in Mode 4, where the basis becomes decay heat removal and prevention of boron stratification. Plausibility enhanced because the action described is correct.

B. Incorrect. Plausible for same reason as option A. Also, a controlled shutdown would be performed for the RCP if vibration was below 5 mils on the shaft. Frame vibration is less than the limit of 20 mils, and by itself, would require a controlled shutdown if frame vibration was below 5 mils.

C. Correct.

D. Incorrect. Plausibility is same same as second half of option B for controlled shutdown of the RCP. The TS basis, or operational concern, is correct.

Technical Reference(s): OFN BB-005, Rev 16 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732415 R3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	
	55.43	5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	G1	2.1.36
	Importance Rating		4.1

Conduct of Operations: Knowledge of procedures and limitations involved in core alterations.

Proposed Question: SRO Question # 94

Given the following:

Date	Time	Activity
7/31/2009	0000	Plant Shutdown to Mode 3 commenced.
7/31/2009	0630	Mode 3 Entry.
7/31/2009	1320	Mode 4 Entry.
7/31/2009	2210	Mode 5 Entry.
8/02/2009	2200	First Reactor Vessel Head Stud detensioned.
8/03/2009	0900	Reactor Vessel Head removed.

Which ONE of the following is the EARLIEST time that irradiated fuel movements may commence in accordance with GEN 00-009, Refueling?

A. 8/3/2009 at 1030.

B. 8/4/2009 at 0210.

C. 8/7/2009 at 0200.

D. 8/7/2009 at 1300.

Proposed Answer: A

Explanation (Optional):

A. Correct. Reactor must be subcritical for 76 hours prior to irradiated fuel movement. Mode 3 trip breakers open in the moment of subcriticality.

B. Incorrect. Plausible because it is 76 hours from a milestone in the shutdown to Refueling. In this case, Mode 5 conditions. Applicant may consider 76 hours from beginning of other plant conditions, such as Mode 5 (Water temperature is below boiling), Mode 6 (first head stud detensioned), and vessel head removed.

C. Incorrect. Plausible because it is 76 hours from a milestone in the shutdown to Refueling. In this case, Mode 6 conditions. Applicant may consider 76 hours from beginning of other plant conditions, such as Mode 5 (Water temperature is below boiling), Mode 6 (first head stud detensioned), and vessel head removed.

D. Incorrect. Plausible because it is 76 hours from a milestone in the shutdown to Refueling. In this case, Mode 6 conditions in preparation for cavity fill. Applicant may consider 76 hours from beginning of other plant conditions, such as Mode 5 (Water temperature is below boiling), Mode 6 (first head stud detensioned), and vessel head removed.

Technical Reference(s): GEN 00-009, Rev 22, Precaution 4.35 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732109 R3 (As available)

Question Source: Bank # WTSI 56261

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Robinson 2007

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 7

Fuel handling facilities and procedures.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		2
	K/A #	G2	2.2.43
	Importance Rating		3.3

Equipment Control: Knowledge of the process used to track inoperable alarms.

Proposed Question: SRO Question # 95

Given the following:

- The plant is in Mode 3.
- A problem with CCW flow switch BB FS-017 is causing an intermittent alarm on annunciator 00-070C, RCP A THRM BAR CCW FLOW.
- The alarm is determined to be a nuisance alarm.

Which ONE of the following describes the LOWEST level of authority required to remove the annunciator card, and the administrative process required for removal of the card, in accordance with AP 21F-001, Equipment Out of Service Control?

A. Manager - Operations or designee;
A Work Request must be generated and the disabled alarm must be entered into the Equipment Out of Service log.

B. Manager - Operations or designee;
If removal of the annunciator card is controlled by a procedure, an OOS sticker is sufficient to document the disabled alarm. NO log entry is required.

C. Work Control SRO;
A Work Request must be generated and the disabled alarm must be entered into the Equipment Out of Service log.

D. Work Control SRO;
If removal of the annunciator card is controlled by a procedure, an OOS sticker is sufficient to document the disabled alarm. NO log entry is required

Proposed Answer: A

Explanation (Optional):

A. Correct.

B. Incorrect. Plausible because an OOS sticker will be used in this condition, and the applicant may confuse the requirement for procedure control to alleviate a work request, with the relaxation of the requirement to log in EOOS log.

C. Incorrect. Work Control SRO would be involved in the process, but will not have responsibility for approval. Plausible also because action is correct.

D. Incorrect. Plausibility same reasons as provided for options B and C.

Technical Reference(s): AP 21F-001, Rev 16, Step 1.4 and 6.3.3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1733213 R5, R6 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 3

Facility licencee procedures required to obtain authority for design and operating changes in the facility.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		3
	K/A #	G3	2.3.13
	Importance Rating		3.8

Radiation Control: Knowledge of Radialogical Safety Procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high radiation areas, aligning filters, etc.

Proposed Question: SRO Question # 96

Given the following:

- The plant is in Mode 6.
- Core offload is in progress in accordance with GEN 00-009, Refueling.
- A fuel assembly being removed from the core appears to have visible damage.
- Subsequently, Containment radiation is rapidly rising.
- The crew enters OFN KE-018, Fuel Handling Accident.
- Personnel evacuation is being performed.

Which ONE of the following describes MINIMUM actions required to limit the radioactive release in accordance with the procedures in effect?

A. BOTH Containment Personnel Hatch doors closed; CTMT Equipment Hatch must be installed with a MINIMUM of TWO (2) bolts.

B. BOTH Containment Personnel Hatch doors closed; CTMT Equipment Hatch must be installed with a MINIMUM of SIX (6) bolts.

C. At least ONE Containment Personnel Hatch door closed; CTMT Equipment Hatch must be installed with a MINIMUM of TWO (2) bolts.

D. At least ONE Containment Personnel Hatch door closed; CTMT Equipment Hatch must be installed with a MINIMUM of SIX (6) bolts.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because this would limit radiation exposure, but neither action is the MINIMUM action required. OFN KE-018 requires ONE personnel hatch door closed, and ITS 3.9.4 requires 4 bolts for the equipment hatch, while GEN 00-009 requires 6.

B. Incorrect. Plausible for same reason as Option A, but 2nd half of this option is correct.

C. Incorrect. Plausible because both actions will limit radiation exposure and the first action is correct. Second option is below minimum required by ITS 3.9.4 and GEN 00-009.

D. Correct.

Technical Reference(s): OFN KE-018, Rev 11 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732428 R3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 4

Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #	G4	2.4.5
	Importance Rating		4.3

Emergency Procedures / Plan: Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions.

Proposed Question: SRO Question # 97

Given the following:

- The crew is responding to a LOCA with no available core injection flow.
- The crew is performing EMG FR-C1 in response to a severe challenge to the core cooling Critical Safety Function.
- All Secondary Heat Removal was lost earlier in the event.
- RCS vent paths have been opened.
- Core Exit Thermocouple (CETC) temperatures remain above 1200°F.

Which ONE of the following describes the NEXT procedure that will be required to mitigate the event?

A. EMG E-1, Loss of Reactor or Secondary Coolant.

B. Severe Accident Control Room Guideline No. 1 (SACRG-1).

C. EMG FR-H1, Response to Loss of Secondary Heat Sink.

D. Remain in EMG FR-C1 until CETC's indicate below 1200°F.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Transition to EMG E-1 will not be made until no Orange or Red CSF Status Trees exist

B. Correct.

C. Incorrect. If Core Cooling remains Red, Heat Sink will not be addressed at this time

D. Incorrect. IF CETs will not decrease below 1200, then SACRG-1 is initiated

Technical Reference(s): EMG FR-C1, Rev 15, steps 23-26 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732341 R3 (As available)

Question Source: Bank # WTSI 59718

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Diablo Canyon 2007

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Added stem information and modified one distractor. Called Bank but probably meets modified criteria

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		2
	K/A #	G2	2.2.6
	Importance Rating		3.6

Equipment Control: Knowledge of the process for making changes to procedures.

Proposed Question: SRO Question # 98

Given the following:

- The plant is in Mode 5.
- An Operations Dept. surveillance cannot be performed as written due to current plant conditions.
- The operator performing the surveillance requests initiation of an On The Spot Change (OTSC) for immediate use.
- The procedure being performed already has ONE documented OTSC in effect.

Which ONE of the following describes the restrictions on performance of this surveillance in accordance with AP-15C-004, Preparation, Review, and Approval of Procedures, Instructions, and Forms?

A. Only ONE active OTSC is allowed per procedure; a procedure revision must be approved prior to completion of the task.

B. Only ONE active OTSC is allowed per procedure; the new OTSC must also incorporate the existing OTSC, and the change must be approved by a minimum of ONE individual cognizant of the procedure in use, who must be a current or active SRO licensed supervisor.

C. Only ONE active OTSC is allowed per procedure; the new OTSC must also incorporate the existing OTSC, and the change must be approved by a minimum of TWO individuals cognizant of the procedure in use, ONE of which must be a current or active SRO licensed supervisor.

D. Multiple OTSCs are allowed in each procedure for a maximum of 14 days; The OTSCs must be approved by a minimum of TWO current or active SRO licensed supervisors, AND a procedure revision must be approved within 14 days of the second OTSC.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. 14 days is allowed for revision, although plausible because only 1 active OTSC is allowed per procedure.

B. Incorrect. Plausible because other departments require 1 SRO to approve an OTSC. In Operations, both signatures would likely be SROs, but only 1 required to be.

C. Correct.

D. Incorrect. Multiple OTSCs are not allowed, but the 14 day limit is plausible and may draw the applicant into believing this is correct.

Technical Reference(s): AP-15C-004, Rev 34, section 6.6.4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1733203 R18 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

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Facility licencee procedures required to obtain authority for design and operating changes in the facility.

Comments:

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		3
	K/A #	G3	2.3.4
	Importance Rating		3.7

Radiation Control: Knowledge of radiation exposure limits under normal or emergency conditions.

Proposed Question: SRO Question # 99

Given the following:

- A General Emergency exists.
- The TSC's Emergency Response Organization is manned and functional.
- An operator that was sent to close a valve in the Auxiliary Building was severely injured when the seal on the valve he was closing failed.
- His injuries appear to be life threatening.
- A Rescue Team is being organized to attempt to remove the operator from the area.
- Radiation levels are approximately 60 Rem/Hr in the General Area.

Which ONE of the following describes the maximum dose that is allowed to perform this operation and whose permission is required?

An individual may receive up to . . .

A. 10 Rem TEDE with permission from the duty Shift Manager.

B. 25 Rem TEDE with permission from the duty Shift Manager.

C. 10 Rem TEDE with permission from the Emergency Manager.

D. 25 Rem TEDE with permission from the Emergency Manager.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. With TSC manned, SM will not approve the dose for Emergency Exposures. This will be performed by the Emergency Manager.

B. Incorrect. Correct dose, wrong approval. Plausible because the Shift Manager typically approves actions related to Operations personnel.

C. Incorrect. Original task would allow 10 Rem, but current task allows 25.

D. Correct.

Technical Reference(s): AP 25A-011, Rev 13A (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1734021 R1 (As available)

Question Source: Bank # WTSI 58332

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Callaway 2007

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

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Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Comments:

This question was changed because emergency exposure limits are different for Wolf Creek. Left as bank question, but there were significant changes in the stem and values are different.

Facility: Wolf Creek

Vendor: WEC

Exam Date: 8/2009

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #	G4	2.4.45
	Importance Rating		4.3

Emergency Procedures / Plan: Ability to prioritize and interpret the significance of each annunciator or alarm.

Proposed Question: SRO Question # 100

Given the following:

A transient has occurred resulting in the following alarms:

- 00-082C, OP DELTA T ROD STOP
- 00-082A, PR OVER PWR ROD STOP
- 00-079A, ROD CTRL URG FAIL
- 00-083C, REACTOR PARTIAL TRIP
- 00-084C, OPDT RX TRIP

Reactor power indicates the following:

- N41 - 105.2%
- N42 - 106.2%
- N43 - 105.9%
- N44 - 106.1%
- Tavg is 575°F

Which ONE of the following has occurred, and which procedure(s) is/are required to be implemented?

A. Uncontrolled Rod Withdrawal; EMG E-0, Reactor Trip or Safety Injection.

B. Uncontrolled Rod Withdrawal; OFN SF-011, Realignment of Dropped Misaligned Rod(s) and Rod Control Malfunctions.

C. Secondary steam leak coincident with a rod control failure; EMG E-0, Reactor Trip or Safety Injection.

D. Secondary steam leak coincident with a rod control failure; OFN AB-041, Steam Line of Feedline Leak and OFN SF-011, Realignment of Dropped Misaligned Rod(s) and Rod Control Malfunctions.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because diagnosis must be performed on high power, which could lead to conclusion of rod withdrawal malfunction. RCS temperature is too low for this diagnosis. In this case, priority would be on the first out annunciator, so the procedure would be correct for this option

B. Incorrect. Plausible for same reason as option A, and under normal circumstances, this would be the correct procedure to refer to. Applicant must understand that the OPDT Reactor Trip annunciator is not that a trip setpoint has been exceeded, but that a reactor trip should occur on 2 of 4 channels

C. Correct. RCS temperature is low by approximately 10°F, indicating excess heat removal. Reactor power is high, resulting in a series of alarms. Diagnosis of the alarms would require use of EMG E-0.

D. Incorrect. Plausible because the diagnosis is correct, and under normal circumstances these would be the correct procedures to use.

Technical Reference(s): ALR 00-084C, Rev 4A (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 1732313 R1 (As available)

Question Source: Bank # WTSI 56978

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: McGuire 2007

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

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Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

This is SRO because none of the indications show a reactor trip setpoint exceeded. The only indication is an alarm that must be interpreted.