WRITTEN / ORAL / ONLINE EXAMINATION KEY COVER SHEET

Examination Number/Title: PDA 17-1 NRC Exam SRO, Rev. 0		
Training Program: Operations		
Course/Lesson Plan Number(s): 60006		
Total Points Possible: 25	**PASS CRITERIA: ≥ 70%**	**Exam Time: 120**

	Yes	No		Yes	No
This is an alternate examination; verified at least 30% of the questions are different from other forms/versions of this exam (e.g., Forms A, B, C; continuing training exam versions for consecutive weeks). For LOCT weekly exams during a segment, verified ≥ 50% difference.	☐	☒	This is a remediation exam. Verified the questions are different from the failed exam by at least the following criteria listed below: • 70% for Maintenance/Technical • 90% for Operations training programs	☐	☒
This is an initial training examination; verified at least 30% of the questions are different from previous administration of the same exam.	☒	☐	This is a LOCT annual operating exam or biennial comprehensive remedial exam, verified the questions are 100% different from the failed exam.	☐	☒
This is a non-randomly generated exam from an electronic exam bank, printed out or administered online. Verified at least 30% of the questions are different from other forms/versions of this exam (e.g., Forms A, B, C; continuing training exam versions for consecutive weeks). For LOCT weekly exams during a segment, verified ≥ 50% difference.	☐	☒	This is a randomly generated exam from an electronic exam bank, printed out or administered online. Verified the exam bank has 3 questions per objective if one test item on exam for the objective. If 2 or more test items on exam for an objective, then 6 questions are in bank.	☐	☒
NOTE: • See TR-AA-230-1003, SAT Development, for exam development and review guidelines. • NRC exams may require additional information. Refer to fleet and site specific procedures.			Key should contain the following: ▪ Learning Objective Number ▪ Test Item ○ Question or Statement ○ All possible answers ○ Correct Answer Indicated ○ Point Value ○ References (if applicable)		

EXAMINATION REVIEW AND APPROVAL:	
Developed by:	**Date:**
Instructional Review of Written Exam (Qualified Instructor):	**Date:**
Technical Review (SME):	**Date:**
Approved by Training Supervisor:	**Date:**
Approved by Training Program Owner (or line designee):	**Date:**

Indicate in the following table if any changes are made to the exam after approval:

#	DESCRIPTION OF CHANGE	REASON FOR CHANGE	AR/TWR# (if applicable)	PREPARER / SUPERVISOR	DATE / DATE

Trainee Name:		
Employee Number:	**Site:**	**PDA**
Examination Number/Title: PDA 17-1 NRC Exam SRO, Rev. 0		
Training Program: Operations		
Course/Lesson Plan Number(s): 60006		
Total Points Possible: 25	**PASS CRITERIA: ≥ 70%**	**Grade: _____/25=_____%**
Graded by:		**Date:**
Co-graded by (if necessary):		**Date:**

EXAMINATION RULES

1.	References may not be used during this examination, unless otherwise stated.
2.	Read each question carefully before answering. If you have any questions or need clarification during the examination, contact the examination proctor.
3.	Conversation with other trainees during the examination is prohibited.
4.	Partial credit will not be considered, unless otherwise stated. Show **all** work and state **all** assumptions when partial credit may be given.
5.	Rest room trips are limited and only one examinee at a time may leave.
6.	For exams with time limits, you have **120 minutes** to complete the examination.
7.	Feedback on this exam may be documented on TR-AA-230-1004-F03, Examination Feedback Form. Contact Instructor to obtain a copy of the form.

EXAMINATION INTEGRITY STATEMENT

Cheating or compromising the exam will result in disciplinary actions up to and including termination.

"I acknowledge that I am aware of the Examination Rules stated above. Further, I have not given, received, or observed any aid or information regarding this examination prior to or during its administration that could compromise this examination."

Examinee's Signature: **Date:**

REVIEW ACKNOWLEDGEMENT

"I acknowledge that the correct answers to the exam questions were indicated to me following the completion of the exam. I have had the opportunity to review the examination questions with the instructor to ensure my understanding.

Examinee's Signature: **Date:**

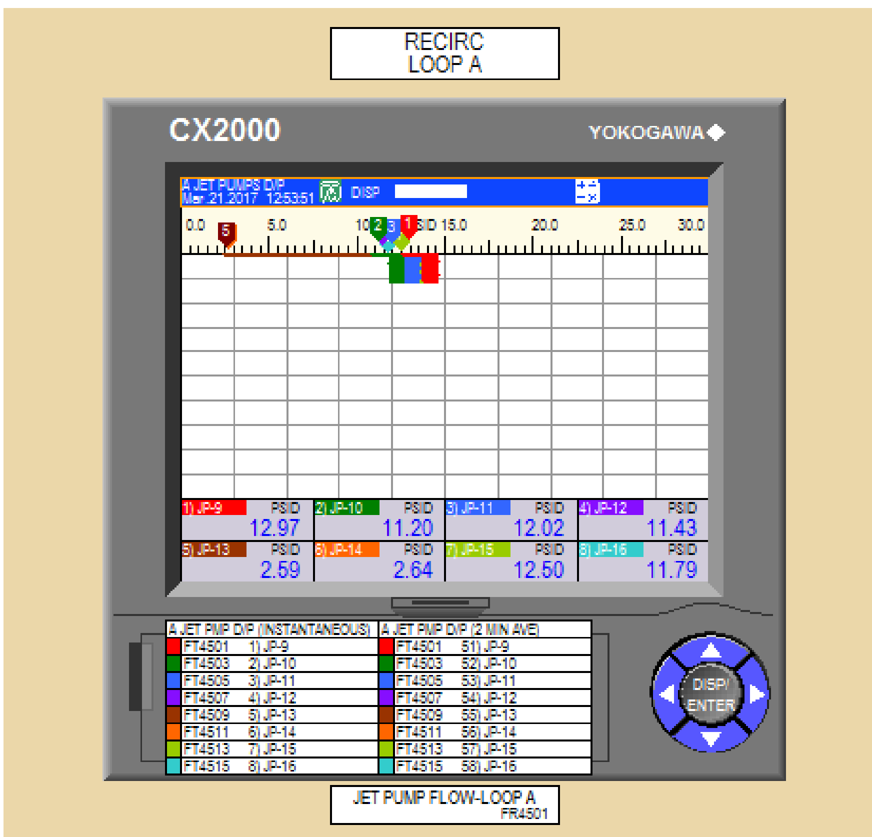
Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295001	AA2.04
	Importance Rating		3.1

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Individual jet pump flows: Not-BWR-1&2

Proposed Question: SRO Question 1 (76)

The plant was operating at full reactor power when an event occurred. The Operator at the Controls reported that Total Core Flow lowered from 49.5 Mlbm/hr to 44 Mlbm/hr.

The following indications were observed, after the event, at 1C38:



What action is required?

A. Establish single loop operations within 24 hours.

B. Be in MODE 3 within 12 hours.

C. TRIP the “A” Recirc pump within 2 hours

D. Be in MODE 2 within 9 hours and MODE 3 within 13 hours

Proposed Answer: B

Explanation (Optional):

A. Incorrect – The reduction in core flow and the lowering jet pump flows could lead the applicant to believe a recirc pump trip has occurred. IF the applicant assumes the indications are due to a pump trip, single loop operations are required to be established within 24 hours per LCO 3.4.1.

B. Correct –Jet pumps 13 and 14 flows are significantly less than that of the remaining Jet Pumps consistent with indications “Rams’s Head” failure. The SRO will determine the Jet Pumps are inoperable based upon this failure and consult Technical Specifications. LCO 3.4.2 This condition requires the plant to be in MODE 3 within 12 hours. This specification is only applicable to Modes 1 and 2, therefore entering MODE 3 exits the mode of applicability and satisfies the specification.

C. Incorrect – If the applicant believes a speed mismatch due to a recirc pump runback has occurred, the action would be to trip the running pump within 2 hours per LCO 3.4.1. This is plausible in that core flow has decreased and jet pump flows are changing.

D. Incorrect – The applicant may perceive an unanalyzed condition based upon the indications, the applicant may elect to apply LCO 3.0.3 which requires MODE 2 within 9 hours and Mode 3 within 13 hours. This condition is accounted for in Technical Specifications 3.4.2 and therefore LCO 3.0.3 is not required to be entered.

Technical Reference(s): AOP 255.2 Power/Reactivity Abnormal Change rev. 43 Bases LCO 3.4.2 Safety Analysis rev. 2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: L3.4.1, L3.4.2

Learning Objective: (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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295016	AA2.02
	Importance Rating		4.3

Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT: Reactor Water Level

Proposed Question: SRO Question 2 (77)

The plant was operating at full reactor power when a fire in the cable spreading room has resulted in heavy smoke in the control room.

- All efforts to extinguish the fire have been unsuccessful
- An immediate evacuation of the control room is required

Transfer of control to the remote shutdown panel must be completed within 20 minutes to ______________________.

A. prevent spurious operation of an SRV

B. prevent spurious operation of the SBDG

C. place RCIC system in service for RPV level control

D. prevent spurious operation of the ECCS Low Pressure Injection Pumps

Proposed Answer: A

Explanation (Optional):

A. Correct – Control of Time Critical Tasks states that transfer of 1C388 must be completed within 20 minutes to close a spuriously opened SRV to control RPV inventory in this condition...

B. Incorrect – Plausible since it is a mitigating action for loss of core flow and to promulgate flow through the core. In the given circumstance however, these conditions are not present. In addition this is not the basis for the time critical operator action.

C. Incorrect – while plausible since this is a mitigating action, it is not the basis for the time critical operator action.

D. Incorrect – while plausible since this is a mitigating action of the procedure, it is not the basis for the time critical operator action.

Technical Reference(s): ACP 103.10 Control of Time Critical Tasks, Rev. 11 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: (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

10 CFR Part 55 Content:	55.41	
	55.43	1

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295019	AA2.02
	Importance Rating		3.7

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Status of safety-related instrument air system loads

Proposed Question: SRO Question 3 (78)

The plant is operating at full reactor power with 1K4, CB/SBGTS Instrument Air Compressor, INOPERABLE.

- 1K4 Day 2 of maintenance for planned repairs
- At 1600 1A311, 1A3 Bus Feeder Breaker from SBDG 1G-31 indications have been lost
- At 1630 Electricians determine that the control power fuses for 1A311 have been blown
- Repairs cannot be completed for 6 hours

What action is the CRS first required to comply with?

A. Declare 1K3 INOPERABLE immediately

B. Declare 1K3 INOPERABLE by 2000

C. Restore 1K4 to OPERABLE within 7 days

D. Restore “A” SBDG 1G-31 to OPERABLE by 1600 the following day.

Proposed Answer: B

Explanation (Optional):

A. Incorrect – Prior to the SBDG being declared INOPERABLE the required redundant feature (1K4) was INOPERABLE. Technical Specifications allots 4 hours prior to declaring the supported equipment (1K3) INOPERABLE. Properly applying the specification, the applicant would determine the 4 hours are allowable to recover the INOPERABLE equipment before the supported piece of equipment is required to be declared INOPERABLE,

B. Correct – Indications provided in the stem demonstrate the “A” SBDG is INOPERABLE. 1K4 was already INOPERABLE (provided in the stem). The applicant would enter LCO 3.8.1 Condition B and perform required actions B.1, perform SR 3.8.1.1 within 1 hour and every 12 hours thereafter. The applicant would also perform required action B.2 which states 4 hours is allotted before the supported feature(1K3) must be declared INOPERABLE. This makes C correct and 1K3 the supported feature must be declared INOPERABLE within 4 hours or by 2000.

C. Incorrect – Although still in this specification, there are other actions that are required to be “first” taken.

D. Incorrect – This may be true as well however there are other actions that are required to be taken first.

Technical Reference(s):	LCO 3.8.1 AC Sources – Operating rev. 272, LCO 3.7.9 CB/SBGT Instrument Air System rev. 229	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: L3.7.9, L3.8.1

Learning Objective: (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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295023	2.2.22
	Importance Rating		4.7

Knowledge of limiting conditions for operations and safety limits: Refueling Accidents

Proposed Question: SRO Question 4 (79)

The plant is in MODE 5 with the fuel pool gates installed for a RFO with the following conditions:

- Fuel is being moved in the Spent Fuel Pool

Then, annunciator FUEL POOL COOLING PANEL 1C-65/1C-66 TROUBLE 1C04B (D-2) alarms. The following plant conditions exist:

- The cause of the alarm is Skimmer Surge Tank Low Level
- Spent Fuel Pool level is 35 feet and slowly lowering
- Refuel Floor ARMs have increased by 2 mrem / hr and **not** alarming

(1) What actions are required?

AND

(2) What EAL is required to be declared?

A. (1) Movement of irradiated fuel assemblies in the spent fuel storage pool may continue indefinitely provided contingencies in place.
(2) An Alert must be declared.

B. (1) Movement of irradiated fuel assemblies in the spent fuel storage pool must be suspended immediately.
(2) An Unusual Event must be declared.

C. (1) Movement of irradiated fuel assemblies in the spent fuel storage pool must be suspended immediately.
(2) An Alert must be declared.

D. (1) Movement of irradiated fuel assemblies in the spent fuel storage pool may continue indefinitely provided contingencies in place.
(2) An Unusual Event must be declared.

Proposed Answer: B

Explanation (Optional):

A. Incorrect –The LCO has been exceeded and UE is required IAW EAL-02, RU2.1, Unplanned valid Refuel Floor ARM reading increase with an uncontrolled loss of reactor cavity, fuel pool, or fuel transfer canal water level with all irradiated fuel assemblies remaining covered by water as indicated by any of the following:
  - Report to control room
  - Valid fuel pool level indication (LI-3413) LESS THAN 36 feet and lowering
  - Valid WR GEMAC Floodup indication (LI-4541) coming on scale

B. Correct – IAW ARP 1C04B A-4 Section 3.7 & 3.8. TS 3.7.8 LCO limit is 36 feet, action is required to immediately suspend fuel movement. An Unusual Event must be declared because fuel pool level indication (LI-3413) LESS THAN 36 feet and lowering and no alarming Refuel Floor ARMs exist..

C. Incorrect - A UE must be declared for the given conditions. Radiation Levels have not risen to the alarm setpoint in the EAL and therefore an ALERT threshold has not been met.

D. Incorrect – Movement of irradiated fuel must be suspended immediately.

Technical Reference(s): TS 3.7.8 Spent Fuel Storage Pool water level rev. 224, EPIP Form EAL-02 EAL Matrix - Cold MODE, Rev 9 (Attach if not previously provided)

.

Proposed References to be provided to applicants during examination: EPIP Form EAL-01
EPIP Form EAL-02

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # X (Note changes or attach parent)

New

Question History: Last NRC Exam: 2011 NRC

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 2

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295025	EA2.03
	Importance Rating		4.1

Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Suppression pool temperature

Proposed Question: SRO Question 5 (80)

The following conditions were present following a Reactor SCRAM:

- Reactor pressure is 910 psig and stable
- Torus water level is 10.4 feet and stable
- Torus water temperature is 160^{0}F and rising slowly
- Drywell pressure is 6 psig and rising slowly
- Drywell air temperature is 220^{0}F and rising slowly

The CRS will____________________.

A. anticipate emergency depressurization

B. emergency depressurize

C. lower reactor pressure 200 psig

D. vent the Drywell

Proposed Answer: B

Explanation (Optional):

A. Incorrect – The HCL curve has been exceeded. This requires emergency depressurization.

B. Correct – Since the HCL curve has been exceeded, emergency depressurization is required.

C. Incorrect – Since the HCL curve has been exceeded, emergency depressurization is required.

D. Incorrect – Group 3 isolation prevents venting the drywell. Venting containment is not yet required.

Technical Reference(s): EOP 1 RPV Control rev.20, EOP 2 Primary Containment Control rev. 18 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: EOP Graph 4 - HCL

Learning Objective: (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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295028	G2.4.8
	Importance Rating		4.5

Knowledge of how abnormal operating procedures are used in conjunction with EOPs.: High Drywell Temperature

Proposed Question: SRO Question 6 (81)

A loss of offsite power occurs and the following conditions are present:

- 1A3 and 1A4 4160VAC Essential Busses are being supplied by "A" and "B" SBDGs
- Drywell Pressure is 6 psig rising slowly
- Drywell air temperature is 2800F and rising slowly

What if any action(s) must the CRS direct?

A. No additional actions are required

B. Emergency Depressurize the RPV

C. Initiate Drywell Sprays

D. Restore power to 1B33, 1B43, 1B35, and 1B45

Proposed Answer: D

Explanation (Optional):

A. Incorrect – to restore drywell cooling, power must be restored to Well Water and to drywell cooling fans.

B. Incorrect – This would be true if drywell air temperature exceeded 340F

C. Incorrect – Outside the allowable Drywell spray initiation limit curve.

D. Correct – LOOP/LOCA Load shedding has occurred requiring restoration of power to 1B33,35,43, and 45. AOP 301 directs these actions.

Technical Reference(s):	AOP 301 Loss of essential electrical power, rev. 71 Restoration of Power to essential busses. Page 34 step d(1)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination:		EOP Graph 7 - DWSIL
Learning Objective:		(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	

Comments

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295037	G2.4.34
	Importance Rating		4.1

Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects: SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown

Proposed Question: SRO Question 7 (82)

- An Electrical ATWS has occurred.
- Both SBLC Pumps fail to run from the control room

The CRS shall direct __________________ to inject Boron to the RPV.

A. OI 153 Standby Liquid Control System

B. AIP 406 Injection with SBLC

C. SEP 304 Boron Injection using RWCU

D. OI 261 Reactor Water Cleanup System

Proposed Answer: C

Explanation (Optional):

A. Incorrect – The Operating instruction does not provide the procedural guidance for alternate injection of boron but does provide direction to inject boron to the RPV.

B. Incorrect – The AIP could inject Boron into the RPV however it requires the use of SBLC pumps which are not available

C. Correct – per EOP ATWS step /Q-

D. Incorrect – Although RWCU system is used to inject Boron to the RPV, the Operating instruction does not provide the procedural guidance for alternate injection of boron

Technical Reference(s): EOP ATWS rev. 23, SEP 304 Boron Injection using RWCU rev. 16 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:			(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	5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295007	G2.1.20
	Importance Rating		4.6

Ability to interpret and execute procedure steps: High Reactor Pressure

Proposed Question: SRO Question 8 (83)

- 10:00 While performing a plant walkdown at full reactor power, the Balance of Plant (BOP) Operator reports that the "B" EHC Pressure Regulator is in service.
- 10:02 The At the Controls Operator reports that Reactor Pressure is 1030 psig.
- 10:17 Attempts to lower Reactor Pressure were unsuccessful.

(1) The CRS will direct _________________ to mitigate this condition?

AND

(2) What is the first action that must be completed?

A. (1) AOP 693 Main Turbine/EHC Failures
(2) Be in MODE 3 within 12 hours

B. (1) AOP 693, Main Turbine/EHC Failures
(2) Reduce Thermal Power to <21.7% within 4 hours

C. (1) AOP 262, Loss of Reactor Pressure Control
(2) Be in MODE 3 within 12 hours

D. (1) AOP 262, Loss of Reactor Pressure Control
(2) Reduce Thermal Power to <21.7% within 4 hours

Proposed Answer: D

Explanation (Optional):

A. Incorrect – AOP 693 does not provide the guidance for a failed pressure regulator. The first action required to be completed is thermal power to <21.7% within 4 hours in accordance with LCO 3.2.2

B. Incorrect – AOP 693 does not provide the guidance for a failed pressure regulator.

C. Incorrect – The first action required to be completed is thermal power to <21.7% within 4 hours in accordance with LCO 3.2.2

D. Correct – AOP 262 provides the guidance for a failed pressure regulator. Step 6 directs lowering power <21.7% to comply with the MCPR spec 3.2.2.

Technical Reference(s):	AOP 262 Loss of Reactor Pressure Control, rev. 8 step 6, LCO 3.2.2 MCPR rev. 244, LCO 3.4.10 Reactor Steam Dome Pressure Rev. 224	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: L3.2.2 L3.4.10

Learning Objective: 94.54.00.01 (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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295010	G2.4.21
	Importance Rating		4.6

Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc. : High Drywell Pressure

Proposed Question: SRO Question 9 (84)

- **Station Blackout** conditions exist
- The RPV has been Emergency Depressurized and Reactor Pressure is being maintained 150-350 psig
- Torus Water level is 12.2 feet and rising slowly
- Containment Pressure is 40 psig and rising slowly
- All available support systems are operating as designed

The CRS directs ______________________ to lower containment pressure to 10 psig.

A. SEP 307, Rapid Depressurization with Bypass Valves

B. SEP 301.3, Torus Vent via Hard Pipe Vent

C. SEP 301.2, Drywell Vent via SBGT

D. SEP 301.1, Torus Vent via SBGT

Proposed Answer: B

Explanation (Optional):

A. Incorrect - This action would remove energy from the containment making it plausible.

B. Correct – With RPS lost, the other choices are not possible. The CRS would direct use of the Torus Hard Pipe Vent to lower containment pressure.

C. Incorrect – RPS is not available. Torus venting is preferable if available

D. Incorrect – RPS is not available.

Technical Reference(s): EOP 2 Primary Containment Control rev. 18, SEP 301.1 Torus Vent via SBGT rev. 10, SEP 301.2 Drywell Vent via SBGT rev. 6, (Attach if not previously provided)

SEP 301.3 Torus Vent via
Hardpipe vent, rev. 10

Proposed References to be provided to applicants during examination: N

Learning Objective: (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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295014	AA2.02
	Importance Rating		3.9

Ability to determine and/or interpret the following as they apply to INADVERTENT REACTIVITY ADDITION: Reactor period

Proposed Question: SRO Question 10 (85)

During a reactor startup, control rod 26-27 is withdrawn one notch. Reactor period is observed to change from 150 seconds to 50 seconds.

Which one of the following describes the appropriate action to be taken?

The CRS will direct the RO to ____________________.

A. immediately SCRAM the reactor

B. re-insert control rods as necessary to go to a subcritical hold point

C. shutdown the reactor until a thorough assessment can be performed

D. re-insert control rod 26-27 to obtain a stable period of greater than 60 seconds

Proposed Answer: D

Explanation (Optional):

A. Incorrect – Reactor Scram is not required at this time. The action to correct this condition is inserting the previously withdrawn control rod to the original position. This will insert the required negative reactivity to control the period.

B. Incorrect – Inserting control rods to a subcritical hold point is only valid if the reactor remained subcritical

C. Incorrect –.A reactor shutdown Is not required for this condition. Inserting the control rod will add the necessary negative reactivity to lengthen the period sufficiently to meet procedural requirements.

D. Correct – The CRS would direct the RO to reinsert control rods as necessary to maintain a manageable period. This action would be directed by the SRO overseeing the reactivity manipulation. It is not an immediate action in any procedure and will be directed by the person overseeing the reactivity manipulation.

Technical Reference(s): IPOI 2 Startup P&L 12 rev 154, (Attach if not previously provided)

Proposed References to be provided to applicants during examination:		N	
Learning Objective:			(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	
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	206000	A2.02
	Importance Rating		3.5

Ability to (a) predict the impacts of the following on the HIGH PRESSURE COOLANT INJECTION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Valve closures: BWR-2,3,4 [High Pressure Coolant Injection (HPCI) System]

Proposed Question: SRO Question 11 (86)

The plant is in MODE 2 and Reactor Pressure is 940 psig and stable

- A through wall piping leak is identified in the HPCI steam supply line.
- The leak has been isolated by closing MO2238 and MO2239 HPCI Steam Supply Isolation Valves.

Mode change to MODE 1 is __________________.

A. allowed by LCO 3.0.4a

B. allowed by LCO 3.0.4b

C. allowed by LCO 3.0.4c

D. not allowed

Proposed Answer: D

Explanation (Optional):

A. HPCI is INOPERABLE when isolated. MODE change to MODE 1 is not allowed.

B. HPCI is INOPERABLE when isolated. MODE change to MODE 1 is not allowed.

C. HPCI is INOPERABLE when isolated. MODE change to MODE 1 is not allowed.

D. Correct: A throughwall leak in the HPCI Steam Supply Line LCO 3.7.3 Condition A applies. Isolate the piping to exit the applicability to the action statement. Isolating HPCI makes HPCI INOPERABLE, MODE change to MODE 1 is not allowed with LCO 3.5.1 Condition .

Technical Reference(s): TRM 3.7.3 Structural Integrity rev. 8, LCO 3.5.1Emergency Core Cooling Systems (ECCS) 259, (Attach if not previously provided)

LCO 3.0.4 LCO Applicability rev. 266

Proposed References to be provided to applicants during examination: N

Learning Objective:	5.01.01.07, State when the HPCI System is required to be OPERABLE by Technical Specifications and describe the bases of the HPCI System LCO's	(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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	209001	G2.2.37
	Importance Rating		4.7

Ability to determine operability and/or availability of safety related equipment: Low Pressure Core Spray System

Proposed Question: SRO Question 12 (87)

- The plant is operating at full reactor power
- RHR Pump "B" supply breaker is racked down and removed for maintenance on the breaker
- During testing it is determined that the "B" Core Spray Pump could not develop more than 2650 gpm at 110 psig.

Which one of the following satisfies the most limiting action?

A. Enter LCO 3.0.3 immediately

B. No additional actions are required

C. Restore RHR Pump "B" to OPERABLE within 72 hours

D. Restore Core Spray Pump "B" to OPERABLE in 7 days

Proposed Answer: C

Explanation (Optional):

A. Incorrect – Since LCO 3.5.1 condition N is modified by a note that allows for condition C therefore performing this action is overly conservative and not in accordance with provisions allowed in the spec.

B. Incorrect – The data provided makes the "B" Core Spray Pump INOPERABLE therefore additional actions are required

C. Correct – With 2 Low Pressure ECCS Pumps INOPERABLE, a 72 hour LCO is required to be entered. Restoration of one of the 2 pumps to OPERABLE is the most restrictive action for the given conditions.

D. Incorrect - This is a required action however, it is not the most limiting action required at this time.

Technical Reference(s): LCO 3.5.1Emergency Core Cooling Systems (ECCS) rev.259 SR3.5.1.4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: L3.5.1

Learning Objective: (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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	212000	A2.02
	Importance Rating		3.9

Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: RPS bus power supply failure

Proposed Question: SRO Question 13 (88)

The plant was operating at full reactor power when an electrical transient occurred resulting in an automatic reactor scram.

The following are plant conditions:

- All control rods are fully inserted
- Reactor level has recovered to 178 inches and is slowly rising
- Reactor Pressure is 1000 psig and rising slowly
- The Main Turbine is tripped
- The SBDG's are running unloaded with the Essential Buses on their normal supply
- RPS is **deenergized**

The CRS shall direct which of the following pressure control strategies?

A. Cool down the RPV with the Main Turbine Bypass Valves utilizing EHC pressure set. Defeat isolation interlocks as necessary

B. Place RWCU in operation using SEP 302.1, RWCU in Recirc mode, or using SEP 302.2, RWCU in Drain mode

C. Place RCIC or HPCI in pressure control mode. Defeat isolation interlocks as necessary

D. Install Defeat 17 Hi Condenser Backpressure Bypass and utilize MSL Drains MO-4423 and MO-4424 for pressure control

Proposed Answer: C

Explanation (Optional):

A. Incorrect – A group 1 isolation is present with the loss of RPS power making this pressure control mode unavailable

B. Incorrect – RWCU is not available due to the loss of RPS power and the system being isolated

C. Correct – HPCI and RCIC are available in pressure control mode There is no 2 psig Drywell pressure signal making this a possible strategy

D. Incorrect – RPS power is lost. Cannot override the isolation signal.

Technical Reference(s): EOP 1 RPV Control rev.20. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 5.11.01.02, Evaluate plant conditions and control room indications and determine necessary mitigative strategies (AOP 358) (As available)

Question Source: Bank # 50632

Modified Bank # (Note changes or attach parent)

New

Question History: PDA 15-1 Workup Comp Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	217000	2.4.6
	Importance Rating		4.7

Knowledge of EOP mitigation strategies: Reactor Core Isolation Cooling (RCIC) System

Proposed Question: SRO Question 14 (89)

An event has occurred and the plant is in a station blackout condition. The following are plant conditions:

- Reactor pressure is being maintained 150 to 350 psig
- Reactor Core Isolation Cooling (RCIC) has been maximized for reactor vessel injection
- HPCI is unavailable
- Drywell air temperature will reach 340°F in 90 minutes
- Torus water temperature is rising such that Heat Capacity Limit (HCL) will be exceeded in 45 minutes
- Two unisolable steam leaks are raising local area temperatures and will exceed max safe in both areas in 55 minutes
- Portable Diesel Driven Fire Pump will have reactor vessel injection capabilities in 70 minutes

Given these plant conditions and times, when is the Control Room Supervisor **first** required to perform a full reactor vessel depressurization?

A full reactor depressurization will be required when ____________________.

A. alternate injections sources are available and lined up

B. two area temperatures are greater than max safe

C. the Heat Capacity Limit curve has been violated

D. drywell air temperature reaches 340°F

Proposed Answer: A

Explanation (Optional):

A. Correct – With the given information the applicant must determine that the Reactor has been depressurized (150 -300 psig) and is being maintained in this condition to allow the RCIC system to provide adequate core cooling through submersion. Full depressurization is not allowed until alternate injections sources are available Once Low Pressure alternate injection systems are aligned once conditions are met for an

Emergency Depressurization, this shall be accomplished. The reactor should not be fully depressurized to allow injection from these systems until it has been determined that low pressure alternate injections systems can be aligned and provide adequate core cooling through submersion. The first condition that would allow for a full depressurization is then determined to be once these systems are available for injection. The applicant should be able to determine from memory those systems that are considered Alternate Injection Systems. EOP 1 is modified by a table listing those systems that are considered Alternate Injection Systems.

B. Incorrect – While this is a condition where EMERGENCY DEPRESSURIZATION is required, based upon the conditions, fully depressurizing would render the RCIC system unavailable for injection. Therefore, waiting until the low pressure alternate injection systems are aligned is required prior to fully depressurizing the RPV.

C. Incorrect - While this is a condition where EMERGENCY DEPRESSURIZATION is required, based upon the conditions, fully depressurizing would render the RCIC system unavailable for injection. Therefore, waiting until the low pressure alternate injection systems are aligned is required prior to fully depressurizing the RPV.

D. Incorrect - While this is a condition where EMERGENCY DEPRESSURIZATION is required, based upon the conditions, fully depressurizing would render the RCIC system unavailable for injection. Therefore, waiting until the low pressure alternate injection systems are aligned is required prior to fully depressurizing the RPV.

Technical Reference(s): EOP Emergency RPV Depressurization, rev. 11 (Attach if not previously provided)
EOP 3 Secondary Containment / Rad Release Control, rev. 22
EOP 2 Primary Containment Control rev. 18

Proposed References to be provided to applicants during examination: N

Learning Objective: (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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	262001	A2.06
	Importance Rating		2.9

Ability to (a) predict the impacts of the following on the A.C. ELECTRICAL DISTRIBUTION ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Deenergizing a plant bus

Proposed Question: SRO Question 15 (90)

Following a loss of Essential AC power, the crew has entered the LOSS OF BOTH ESSENTIAL 4160V BUSES section of AOP 301, LOSS OF ESSENTIAL ELECTRICAL POWER.

Which ONE of the following correctly describes:

(1) The impact on Station Fire Protection?

AND

(2) The procedure which dictates the establishment of "compensatory" fire patrols?

A. (1) All Fire Detection Systems become NON-FUNCTIONAL
(2) EN-AA-202-1004, FIRE PROTECTION SCREENING

B. (1) Loss of Fire Barrier when fire doors are blocked open
(2) EN-AA-202-1004, FIRE PROTECTION SCREENING

C. (1) Loss of Fire Barrier when fire doors are blocked open
(2) ACP 1412.4, IMPAIRMENTS TO FIRE PROTECTION SYSTEMS

D. (1) All Fire Detection Systems become NON-FUNCTIONAL
(2) ACP 1412.4, IMPAIRMENTS TO FIRE PROTECTION SYSTEMS

Proposed Answer: C

Explanation (Optional):

A. Incorrect - Some fire detection systems are supplied by their own batteries and would not be INOPERABLE and therefore would not require "compensatory" fire patrols.

B. Incorrect - EN-AA-202-1004, FIRE PROTECTION SCREENING. and the DAEC Fire Plan describe the overall fire protection program at the Duane Arnold Energy Center. There is no specific guidance for impairments in EN-AA-202-1004, FIRE PROTECTION SCREENING..

C. Correct - IAW AOP-301, page 11, Consult with Security, establish natural/temporary ventilation by opening room/panel doors as follows: Battery Room doors, Nonessential

Switchgear Room doors, Control Room doors, and Essential Switchgear Room doors. A caution in this step is "This step affects the integrity of some plant fire zones. FPIRs and fire watches may be required." ACP-1412.4, Impairments to Fire Protection Systems, provides the requirement for establishing "compensatory" fire patrols.

D. Incorrect - Some fire detection systems are supplied by their own batteries and would not be INOPERABLE and therefore would not require "compensatory" fire patrols.

Technical Reference(s): AOP 301 Loss of Essential Electrical Power, Rev. 71 (Attach if not previously provided)

ACP 1412.4, Rev.

Proposed References to be provided to applicants during examination: N

Learning Objective: 5.04.01.02, Evaluate plant conditions and control room indications and determine the actions directed by AOP 301 (As available)

Question Source: Bank # 50469

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

55.43 2

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	201006	G2.2.40
	Importance Rating		4.7

Ability to apply Technical Specifications for a system: Rod Worth Minimizer System (Plant Specific)

Proposed Question: SRO Question 16 (91)

Reactor Power is at 8% with the Rod Worth Minimizer out of service. Operators inserting control rods inadvertently skipped several rod sequence control pages resulting in seven rods out of the allowable sequence.

(1) Which one of the following is the maximum time to correct the condition?

AND

(2) What is the basis for this action?

A. (1) 4 Hours
(2) ensures the 280 cal/gm fuel damage limit will not be violated during a Control Rod Drop Accident.

B. (1) 4 Hours
(2) maintains the capability to insert the control rods which provides assurance that the assumptions for scram reactivity and transient analyses are not violated.

C. (1) 8 Hours
(2) ensures the 280 cal/gm fuel damage limit will not be violated during a Control Rod Drop Accident.

D. (1) 8 Hours
(2) maintains the capability to insert the control rods which provides assurance that the assumptions for scram reactivity and transient analyses are not violated.

Proposed Answer: C

Explanation (Optional):

A. Incorrect - The applicant could consider LCO 3.1.3 condition D which has a 4 hour completion time for 2 or more INOPERABLE control rods not within the BPWS. In addition, the basis in this choice is for control rod operability and not for operating outside the allowable sequence.

B. Incorrect - The applicant could consider LCO 3.1.3 condition D which has a 4 hour completion time for 2 or more INOPERABLE control rods not within the BPWS

C. Correct – LCO 3.1.6 condition A required action A.1 move control rods to correct position within 8 hours. The basis for this action is “Operation within the Bank Position

Withdrawal Sequence ensures that the 280 cal/gm fuel limitation will not be exceeded during a Control Rod Drop Accident.

D. Incorrect – The basis in this choice is for control rod operability not being outside of the BPWS. This makes this choice incorrect.

Technical Reference(s): LCO 3.1.6 Rod Pattern Control rev. 224, LCO 3.1.3 Control Rod Operability rev. 271 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: L3.1.6, L3.1.3

Learning Objective: (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

Comments

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	216000	A2.07
	Importance Rating		3.5

Ability to (a) predict the impacts of the following on the NUCLEAR BOILER INSTRUMENTATION ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Reference leg flashing

Proposed Question: SRO Question 17 (92)

A normal plant Shutdown and Cooldown from full power operations is in progress.

- The reactor is shutdown
- RPV Pressure is 940 psig
- RPV Level is 190 inches and stable
- Cooldown has just started
- The GEMAC Reference Leg Backfill System has been out of service for 3 weeks

(1) What compensatory actions are directed to address these plant conditions?

AND

(2) Which procedure is directed if "NOTCHING" is observed?

A. (1) When RPV pressure reaches 500 psig direct the operating crew NOT to use the Yarway instruments on 1C05 for level indication
(2) OI 880, Non-Nuclear Instrumentation System

B. (1) When RPV pressure reaches 500 psig direct the operating crew NOT to use the Yarway Instruments on 1C05 for level indication
(2) IPOI 4, Shutdown

C. (1) Direct enhanced RPV Level monitoring during the plant cooldown
(2) OI 880, Non-Nuclear Instrumentation System

D. (1) Direct enhanced RPV Level monitoring during the plant cooldown
(2) IPOI 4, Shutdown

Proposed Answer: C

Explanation (Optional):

A. Incorrect - Yarways share instrument lines with the GEMAC instruments but these directions would be given for a rapid pressure decrease. OI 880 is the correct OI for the action required.

B. Incorrect - Yarways share instrument lines with the GEMAC instruments but these directions would be given for a rapid pressure decrease. IPOI 4 does direct action be performed IAW OI 880 for these conditions but does not have directions if Notching is observed.

C. Correct - Anytime the Reference Leg Backfill system is out of service for 7 days performance of OI-880, J-1 section 6.1 is required. Notching is expected when reducing RPV pressure during a shutdown. The Narrow range GEMAC level instruments are susceptible. OI-880 directs these actions for the given plant conditions.

D. Incorrect - IPOI 4 does direct action be performed IAW OI 880 for these conditions but does not have directions if Notching is observed. Enhanced level monitoring is the correct direction for these conditions.

Technical Reference(s): OI-880 Non Nuclear Instrumentation System section J-1, Rev.22 page 18 step 3 (Attach if not previously provided)

IPOI 4, Rev.

Proposed References to be provided to applicants during examination: N

Learning Objective: 4.18.03, Direct operator actions to control RPV level throughout the cooldown. (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2009 NRC

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	290001	G2.4.6
	Importance Rating		4.7

Knowledge of EOP mitigation strategies.: Secondary Containment

Proposed Question: SRO Question 18 (93)

With the plant operating at full power, a spent resin spill from the RWCU Phase Separator occurs in the reactor building. The following conditions are present:

- RWCU Spent Resin Room ARM, RI-9173, is reading 3,500 mR/hr
- RWCU Phase Separator Room ARM, RI-9177, is reading 850 mR/hr
- North Refuel Floor ARM, RI-9163, is reading 210 mR/hr

Which one of the following actions will be directed by the CRS?

A. Perform a fast power reduction and SCRAM the reactor

B. Anticipate Emergency Depressurization

C. Perform a plant shutdown per IPOI 3

D. Emergency Depressurize

Proposed Answer: C

Explanation (Optional):

A. A: Incorrect - would be true if reactor were discharging into the Secondary Containment (SC-3) . The conditions present represent a resin spill that would not be affected by lowering the energy state of the Reactor. The applicant may choose this condition due to exceeding Max Safe Operating Limits in ONE area per EOP 3, Secondary Containment Control/Rad Release Control.

B. Incorrect - would be true if reactor were discharging into the Secondary Containment (SC-3) and Max Safe Operating Limits were bing approached.

C. Correct Answer: It is required to perform a plant shutdown due to exceeding Max Safe Operating Limit in TWO areas. (SC-9)

D. Incorrect - would be true if reactor were discharging into the Secondary Containment. (SC-7)

Technical Reference(s):	Bases- EOP 3 / EOP 4 Secondary Containment / Rad Release Control, rev. 22	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: EOP Table 6

Learning Objective:	6.68.02.01, Justify the use of IPOI 3,4, or 5 to perform a plant shutdown while performing EOP 3.	(As available)

Question Source:	Bank #	
	Modified Bank # 52213	(Note changes or attach parent)
	New	

Question History:	PDA 17-1 EOP Comp	Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	
	55.43	4

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G1	G2.1.5
	Importance Rating		3.9

Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.

Proposed Question: SRO Question 19 (94)

The plant is operating at full reactor power during a weekend night shift. The Shift Technical Advisor's (STA) wife just called to say she is headed to the hospital to have a baby. The STA asks to leave the site to accompany his wife prior to watch relief.

Is this permitted per ACP 1410.1, Operations Working Standard?

A. No, a qualified relief must first be on-site and relieve the STA.

B. Yes, the only action necessary is to obtain the Operations Director's approval to secure the STA for the remainder of the shift.

C. Yes, as long as an immediate callout is made for watch relief and the relief arrives within 2 hours. The Operations Director must be notified.

D. Yes, as long as an immediate callout is made for watch relief and the relief arrives within 4 hours. The Operations Director must be notified.

Proposed Answer: C

Explanation (Optional):

A. Incorrect - It is allowed.

B. Incorrect - Ops Manager approval not necessary, just notification

C. Correct - IAW T.S. 5.2.2: Shift crew composition may be less than the minimum requirement for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements. IAW ACP 1410.1, Operations Director must be notified.

D. Incorrect - An immediate call out is necessary, and the relief must be on site within 2 hours.

Technical Reference(s): ACP 1410.1, Rev. 104, Sect. 3.2, Step (4) (a) & (b) LCO 5.2.2 Organization rev. 274 (Attach if not previously provided)

T.S. 5.2.2, Amendment
274

Proposed References to be provided to applicants during examination: N

Learning Objective:	1.11.03, ensure plant activities are performed in compliance with the ACP 1400 Manual.	(As available)

Question Source:	Bank #	50878	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History:	PDA 13-1 SRO DAEC Cert	Last NRC Exam:	

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

10 CFR Part 55 Content:	55.41	
	55.43	1

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			
	Group #			
	K/A #	G1	G2.1.36	
	Importance Rating			4.1

Knowledge of procedures and limitations involved in core alterations.

Proposed Question: SRO Question 20 (95)

Whose permission is required to commence Irradiated Fuel Handling?

A. Fuel Handling Supervisor

B. Operations Shift Manager

C. Reactor Engineer

D. Operations Director

Proposed Answer: B

Explanation (Optional):

A. Incorrect - The Fuel Handling Supervisor's permission is not required. Although this position is in authority in the field, ultimately it is the SM that approves the commencement of Fuel Handling.

B. Correct - The OSM must approve commencement of fuel handling per RFP 403 rev. 60 Attachment 1.

C. Incorrect - The Reactor Engineer's permission is not required to commence fuel handling. The reactor engineer does provide day to day support to the operating crew during normal operation and their approval is required during reactivity manipulations.

D. Incorrect - The Operations Director's permission is not required to commence fuel handling. This could be misconstrued by the applicant in that many operational decisions are overseen by the Operations Director. Their level within the organization high level and may be misconstrued by an operator that does not know the requirement as the correct answer.

Technical Reference(s): RFP 403 rev 60 Attachment 1 step 21. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	1.04.01.03, describe the information found in the Fuel Moving Plan and explain how minor changes are made to the Fuel Moving Plan,	(As available)

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

Question History:	PDA 15-1 SRO Workup Comp A	Last NRC Exam:	

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

10 CFR Part 55 Content:	55.41	
	55.43	6

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G2	G2.2.6
	Importance Rating		3.6

Knowledge of the process for making changes to procedures.

Proposed Question: SRO Question 21 (96)

While performing a section of Operating Instruction (OI) 410, River Water Supply System, the reactor operator comes across this step:

(27) Place Traveling Screen Wash Pump 1P-112A[B] in service by placing handswitch HS-2906A[B] to the AUTO position on breaker 1B9106 [1B2106].

This step is unable to be performed. The Reactor Operator (RO) has determined performance of the step is **NOT** required.

Which one of the following meets the approval requirements to place an "NA" in the block for this step?

A. Two Senior Reactor Operators

B. Only the Operations Shift Manager

C. The STA and the RO performing the procedure

D. The Control Room Supervisor and the RO performing the procedure

Proposed Answer: A

Explanation (Optional):

A. Correct: Two (2) SROs are required to review, approve, and initial an NA to a step in a work instruction or procedure that are for safety related equipment or equipment that supports technical specifications.

B. Incorrect: The OSM would need a second SRO

C. Incorrect: Two SROs are required to perform this action

D. Incorrect: The CRS would need a second SRO

Technical Reference(s): AD-AA-100-1006 Procedure Work Instruction Use and Adherence, page 28 section 4.8 step 5D, (Attach if not previously provided)

Proposed References to be provided to applicants during examination:	N		
Learning Objective:	1.11.02.10		(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	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G2	2.2.44
	Importance Rating		4.4

Ability to interpret control room indications to verify the status and | operation of a system, and understand how operator actions and directives affect plant and system conditions.

Proposed Question: SRO Question 22 (97)

The plant is operating at 100% power with the following conditions:

- Instrument and Control Technicians determine that all Torus Water Level indications are indicating 6 inches higher than actual Torus Water Level.
- Currently Torus Water Level indicates 10.35 Feet

What is required?

A. Fill the Torus

B. Scram the Reactor and enter EOP 1

C. Emergency Depressurize the RPV

D. Declare the Primary Containment INOPERABLE

Proposed Answer: A

Explanation (Optional):

A. Correct - Torus water level actual is 9.85 Feet which is below the entry condition of 10.1 feet in EOP 2. LCO 3.6.2.2 applies. This condition must be corrected within 2 hours.

B. Incorrect – This action is not required for the given conditions. Level is stable but low.

C. Incorrect – This action is not required for the given conditions. Level is stable but low.

D. Incorrect – This action in itself is not required. The required action is to fill the Torus to the appropriate level per Technical Specifications and EOP guidance.

Technical Reference(s): EOP 2, TS LCO 3.6.2.2
Suppression Pool Water Level rev. 224 (Attach if not previously provided)

Proposed References to be provided to applicants during examination:		L3.6.2.2 L3.6.1.1
Learning Objective:		(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	
Comments:		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G3	G2.3.5
	Importance Rating		2.9

Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question: SRO Question 23 (98)

An accident has occurred which has caused entry into the RPV Flood EOP. All control rods are fully inserted.

- Drywell area Rad monitors are now reading 450 and 500 R/Hr and slowly rising
- Torus area Rad monitors are now reading 45 and 50 R/hr and slowly rising
- The TSC is NOT manned at this time

Table 5	**Core Damage Indications**
Parameter	**Value**
Primary containment hydrogen concentration	Drywell OR torus H_2 concentration above 0.4% (minimum detectable)
Primary containment radiation	Drywell Area Hi Range Rad Monitor RIM-9184A/B above 7E+2 R/hr OR Torus Area Hi Range Rad Monitor RIM-9185A/B above 3E+1 R/hr
Reactor coolant activity	Chemistry samples above 300 µCi/gm dose equivalent I-131
Fuel damage assessment (PASAP 7.2)	At or above 5% fuel clad damage

Given these plant conditions and EOP Table 5, which of the following action(s) is(are) required?

A. Continue to perform RPV/F actions. No SAG entry conditions are present

B. Continue to perform RPV/F actions and perform SAG 1, Primary Containment Flooding actions concurrently

C. Continue to perform RPV/F actions until RPV reactor water level indications are observed. Once the TSC is operational, exit the EOPs and enter the SAGs

D. Enter the SAGs and transition from RPV/F to SAG 1, Primary Containment Flooding. Continue to monitor the plant for degrading conditions and report them to the TSC once manned and operational

Proposed Answer: C

Explanation (Optional):

A. Incorrect – The radiation conditions provided in the stem indicate core damage is present. The applicant should know that entry to SAGs is required when core damage indications are present while implementing this contingency.

B. Incorrect – SAG entry is required when indications of core damage are present (which is indicated by Torus radiation levels.) The crew is to perform only EOP actions and once the TSC is operational, The SAGs are entered at the appropriate point and directed by the TSC and then the EOPs are exited and EOP actions terminated.

C. Correct - Based upon the containment Rad levels during RPV flooding fuel damage is occurring. This requires entry to SAGs. The transition to SAGs is performed by continuing the EOP actions until there is evidence that the core is flooded or reactor water level indication is available and the TSC is operational. Once the TSC is operational, The SAGs are entered at the appropriate point and directed by the TSC and then the EOPs are exited and EOP actions terminated.

D. Incorrect - EOPs should NOT be exited until the TSC is operational. The transition to SAGs is performed by continuing the EOP actions until there is evidence that the core is flooded.

Technical Reference(s):	RPV/F RPV Flooding , Rev. 15 Table 5	(Attach if not previously provided)

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Proposed References to be provided to applicants during examination: N

Learning Objective:	6.85.06.01, discuss the process that is required to transition from EOPs to Severe Accident Management procedures	(As available)

Question Source:	Bank #	52275	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History:	PDA 15-1 SRO Workup Comp A	Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	
	55.43	4

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G4	G2.4.37
	Importance Rating		4.1

Knowledge of the lines of authority during implementation of the emergency plan.

Proposed Question: SRO Question 24 (99)

- An event has resulted in an unisolable leak and lowering level in the fuel pool.
- All onsite and offsite personnel have responded to their designated emergency response organizations.
- The Technical Support Center is staffed but **not** yet operational.
- The EOF is staffed but **not** yet operational.

It has been determined a task on the refuel floor is required to be performed by two operators.

Which one of the following personnel are required to approve access to the refuel floor?

A. Shift Manager

B. TSC Operations Supervisor

C. Site Radiation Protection Coordinator

D. Emergency Response and Recovery Director

Proposed Answer: A

Explanation (Optional):

A. Correct - EPIP 2.5 Control Room Emergency Response Operation, Revision 21 page 11: If suspected abnormal radiation conditions exist the Emergency Coordinator must approve the dispatch of all personnel into the plant.The SM is the EC until the TSC is fully staffed and has command and control

B. Incorrect – TSC Operations Supervisor does carry operational authority in decision making however with the TSC not yet operational, this position would not authorize this action. In addition, it would be the EC that would perform this authorization.

C. The SRPC, while this person does assess radiation conditions on and off site, the decision is made be the EC

D. The ERRD is not in command of the event at this point

Technical Reference(s): EPIP 2.5 Control Room Emergency Response Operation, Revision 21 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: (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 4

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G4	G2.4.38
	Importance Rating		4.4

Ability to take actions called for in the facility emergency plan, including supporting or acting as emergency coordinator if required.

Proposed Question: SRO Question 25 (100)

An event has occurred at the DAEC and the following conditions are now present:

- A Steam leak in the Steam Tunnel has caused temperatures to rise to 250°F
- Steam Tunnel Temperatures continue to rise
- The “A” Main Steam Line Isolation Valves failed to close automatically and will not close from the control room

(1) Which one of the following Emergency Action Level (EAL) declarations should be made?

AND

(2) Which one of the following conditions would result in an EAL escalation?

A. (1) FA1
(2) Drywell pressure rises above 2 psig

B. (1) FA1
(2) Torus Radiation Levels rise to 35 Rem/hr

C. (1) FS1
(2) Drywell pressure rises above 2 psig

D. (1) FS1
(2) Torus Radiation Levels rise to 35 Rem/hr

Proposed Answer: D

Explanation (Optional):

A. Incorrect: The conditions represented in the stem describe a failure of the Reactor Coolant System (Steam Leak with temperatures in excess of Max normal operating temperature) and the Primary Coolant System Fission Product barriers (Failure of the “A”Steam line to isolate). The student may incorrectly determine only the Primary Containment Barrier was making FA1 plausible. The stem describes conditions resulting in 2 of 3 barriers lost or potentially lost making Site Area Emergency the correct EAL. Drywell Pressure above 2 psig is indicative of a loss of reactor coolant system boundary and would not require escalation since it was already considered lost by the steam leak from Main Steam.

B. Incorrect: The conditions represented in the stem describe a failure of the Reactor Coolant System (Steam Leak with temperatures in excess of Max normal operating temperature) and the Primary Coolant System Fission Product barriers (Failure of the “A”Steam line to isolate). The student may incorrectly determine only the Primary Containment Barrier was making FA1 plausible. The Fuel Clad Barrier loss represented in Torus Radiation Levels would be an escalation criteria however the escalation would be from a Site Area not and Alert..

C. Incorrect: Plausible in that the candidate may properly classify the EAL however 2 psig in the drywell is indicative of a loss of Reactor Coolant System boundary which has already been accounted for. Therefore, escalation to the FG1 would not be required with these conditions.

D. Correct: The conditions represented in the stem describe a failure of the Reactor Coolant System (Steam Leak with temperatures in excess of Max normal operating temperature) and the Primary Coolant System Fission Product barriers (Failure of the “A”Steam line to isolate). The Fuel Clad Barrier loss as represented in Torus Radiation Levels would be an escalation criteria for loss of the Fuel Cladding Barrier which would result in loss of all 3 Fission Product Barriers and meet the General Emergency escalation.

Technical Reference(s): EPIP 1.1 Determination of Emergency Action Levels, Rev. 29 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: EPIP Form EAL-01

Learning Objective:	3.01.01.01, Explain the Responsibilities and Instructions contained in EPIP 1.1	(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

Comments: