QUESTION # 1

Examination Outline Cross-reference:	Level	RO	SRO
Partial or Complete Loss of AC / 6	Tier #		1
Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: (CFR: 41.10/ **43.5** / 45.13)	Group #		1
	K/A # **295003**		**AA2.04**
• System lineups	Importance Rating		**3.7**

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

- FW pumps are in 3-element control on the Master Feedwater Controller.
- 125 VDC Station Battery B is on equalize charge per OP-43A, 125 VDC Power System.
- UPS M-G set is on the DC Drive to "run-in" the DC motor brushes under load per OP-46B, 120 VAC Power System.
- EDG B and D are running unloaded per ST-9R, EDG System Quick-Start Operability Test and Offsite Circuit Verification.

Subsequently, the following valid annunciator alarm is received:

09-8-4-18, L26 600 V SUPP FDR BKR 12602 TRIP

All plant equipment responds per design.

Which <u>ONE</u> of the following is the PROMPT action directed by the CRS as a result of these plant conditions?

a) Take manual control of one FW pump on the MSC per AOP-21, Loss of UPS.

b) Lock the RWR Scoop Tubes to prevent a run back per AOP-21, Loss of UPS.

c) Reduce Station Battery B loads per AOP-19B, Loss of Switchgear L26.

d) Shutdown EDG B and D per AOP-19B, Loss of Switchgear L26.

Proposed Answer: d) Shutdown EDG B and D per AOP-19B, Loss of Switchgear L26.

Explanation (Optional): <u>**Justification:**</u>
Power is lost to L26 resulting in a loss of ESW pump B which supplies cooling water to the EDGs that were running. The UPS is still powered by the A DC battery system and the actions provided in the distracters A & B are not a prompt action to be taken for the plant conditions provided in the stem. Reducing DC loads is a subsequent action per AOP-19B.

QUESTION # 1 Continued

Distracters: a) Take manual control of one FW pump on the MSC per AOP-21, Loss of UPS is **incorrect:**: Take manual control of one FW pump on the MSC per AOP-21, Loss of UPS. This action is part of the response to a momentary loss of the UPS, per the plant conditions provided in the stem, the UPS is still powered from the DC drive. When a subsequent step is taken to transfer the UPS to the alternate AC source, this would be part of the actions to take.

b) Lock the RWR Scoop Tubes to prevent a run back per AOP-21, Loss of UPS is **incorrect:**: Lock the RWR Scoop Tubes to prevent a run back per AOP-21, Loss of UPS. Per the stem, the UPS is still powered from the DC Drive and has not lost power.

c) Reduce Station Battery B loads per AOP-19B, Loss of Switchgear L26 is **incorrect:** This is a subsequent action to the conditions provided in the stem.

Technical Reference(s): AOP-19B, AOP-21. (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: SDLP-71E EO-1.09.C (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

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

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5 Assessment of Facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: Plant Event 41987, JAF-CR-2005-3818, Momentary loss of UPS results in a RX Scram was used in part to supply the distracters for this question.

QUESTION # 2

Examination Outline Cross-reference:	Level	RO	SRO
Partial or Total Loss of DC Pwr / 6	Tier #		1
Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: (CFR: 41.10/ **43.5** / 45.13)	Group #		1
	K/A # **295004**		**AA2.02**
• Extent of partial or complete loss of D.C. power	Importance Rating		**3.9**

Proposed Question: The plant is at 100% power.

Feedwater Pump A & B are in 3 element control selected to RX WTR LVL COLUMN "A".

There are NO evolutions in progress when the following are noted:

Time 0:

- Annunciator 09-8-1-21, 125VDC BATT CHGR A DC GRD
- 125VDC Bus A GND DET meter on Panel 09-8 indicates +25 volts and steady

Time 1 minute:

- Annunciator 09-8-1-22, 125VDC BATT CHGR B AC SUPP TROUBLE

Time 4 minutes:

- Annunciator 09-8-1-23, 125VDC BATT B VOLT LO
- 125VDC Bus B Output Voltage meter on Panel 09-8 indicates 119VDC

Which ONE of the following identifies the resultant equipment status AND procedure used to respond to the above indications and alarms?

a) AC POWER breaker at 71BC-1A 125V DC BATTERY CHARGER A tripped, AOP-45, Loss of DC Power System A

b) 71BC-1B 125V DC BATTERY CHARGER B breaker tripped at 71MCC-262-OA1, AOP-46, Loss of DC Power System B

c) RX WTR LVL 06LI-94B indicates downscale, AOP-41, Feedwater Malfunction (Rising Feedwater Flow- High RPV Level)

d) RHR B initiation logic is inoperable, AOP-22, DC Power System A Ground Isolation

Proposed Answer: b) 71BC-1B 125V DC BATTERY CHARGER B breaker tripped at 71MCC-262-OA1, AOP-46, Loss of DC Power System B

Explanation (Optional): **Justification:** See ARP-09-8-1-22 causes & step 2, 2nd bullet, AOP-46, Loss of DC Power System B see symptom A-first bullets first dash- 09-8-1-22 is listed as 1 or more of the following annunciators in alarm.

QUESTION # 2 Continued

Distracters:

a) AC POWER breaker at 71BC-1A 125V DC BATTERY CHARGER A tripped is **incorrect** there is only a small ground on DC Bus A see ARP-09-8-1-21 this also means AOP-45, Loss of DC Power System A is an **incorrect** procedure to enter.

c) RX WTR LVL 06LI-94B indicates downscale is **incorrect** battery is supplying the bus at 119 VDC while this is a symptom of AOP-46, Loss of DC Power B, the voltage is still acceptable for the indicator to be normal. Conditions would **NOT** require entry into AOP-41, Feedwater Malfunction (Rising Feedwater Flow- High RPV Level) as the stem stipulates WTR Column "A" is selected.

d) RHR B initiation logic is unaffected by the loss of B DC. RHR A logic is powered by B DC.

Technical Reference(s): AOP-46, AOP-41, AOP-45, AOP-22, OP-46B (Attach if not previously provided)
ARPs:
09-8-1-19, 09-8-1-21,
09-8-1-22, 09-8-1-23,

Proposed references to be provided to applicants during examination: NONE

Learning Objective: SDLP-71B EO- 1.10.A.1 (As available)

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

Question History: Last NRC Exam

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

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

10 CFR Part 55 Content: 55.41
55.43 5 Assessment of Facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

QUESTION # 3

Examination Outline Cross-reference:	Level	RO	SRO
Main Turbine Generator Trip / 3	Tier #		1
Knowledge of EOP terms and definitions. (CFR: 41.10/ 43.5 / 45.13)	Group #		1
	K/A # 295005		G 2.4.17
	Importance Rating		3.8

Proposed Question: The plant is at 100% power.

I&C is working on 06LT-52C, Reactor Water Level Feedwater Control Level Transmitter when the following indications are noted on Panel 09-5:

- RX WTR LVL HI CHNL 'A' Amber Light is '<u>ON</u>',
- RX WTR LVL HI CHNL 'B' Amber Light is '<u>ON</u>',
- RX WTR LVL HI CHNL 'C' Amber Light is '<u>ON</u>'.

A correct automatic action occurs due to these indications. The CRS directs insertion of a manual scram and the SNO reports all rods in with the exception that:

- 3 rods are at position 02
- 1 rod is at position 48.

The CRS initially directs entry into AOP-1 (Reactor Scram) and the applicable EOP(s).

The automatic action that occurred was a _________ trip. The CRS must direct actions in _____________ remain shutdown under all conditions without boron.

a) Main Turbine, EOP-2 (RPV Control) because the reactor **WILL**

b) HPCI Pump, EOP-2 (RPV Control) because the reactor **WILL**

c) Main Turbine, EOP-3 (Failure to Scram) because the reactor will **NOT**

d) HPCI Pump, EOP-3 (Failure to Scram) because the reactor will **NOT**

Proposed Answer: a) Main Turbine, EOP-2 (RPV Control) because the reactor **WILL**

Explanation (Optional): **<u>Justification:</u>** Indications are directly part of Main Turbine & FW Pump trip circuitry. Per EP-1, EOP Entry and Use, section 4.7.2, the reactor will remain shutdown under all conditions without boron with one rod at 48 if all other rods are at 02 (or inserted).

QUESTION # 3 CONTINUED

Distracters: b) HPCI Pump trip, EOP-2 (RPV Control) because the reactor **WILL**
Justification: Indications are **NOT** part of HPCI trip circuitry. The second part of the distractor is correct..

c) Main Turbine Generator trip, EOP-3 (Failure to Scram) because the reactor will **NOT**
Justification: The first part is correct but the reactor will remain shutdown.

d) HPCI Pump trip, EOP-3 (Failure to Scram) because the reactor will **NOT**
Justification: The first part is incorrect, the second part is correct.

Technical Reference(s): EOP-2, EOP-3, EP-1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **EOP 2 and 3**

Learning Objective: MIT-301.11A EO- 1.04.i (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History: Last NRC Exam

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

Question Cognitive Level: Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 **5** Assessment of Facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

QUESTION # 4

Examination Outline Cross-reference:

Refueling Acc / 8

Ability to apply technical specifications for a system. **(10CFR 55.43.2/4/6/7)**

Level	RO	SRO
Tier #		**1**
Group #		**1**
K/A # **295023**		**G 2.1.12**
Importance Rating		**4.0**

Proposed Question: The plant has been shutdown for 2 days and is being refueled.

An irradiated fuel bundle is being moved from the core to the spent fuel storage pool. The bundle is over the core and being moved towards the fuel pool when level drops to 22 feet above the RPV flange and then continues to slowly decrease. The shift declares a radiological emergency per RAP-7.1.04B, Refueling Procedure.

i. The Technical Specification bases for maintaining a minimum water level over the flange is to insure___________.

ii. The Refuel Bridge SRO must direct the bundle to be placed in <u>any</u> empty location in the ________.

a) i. RHR Shutdown Cooling can maintain "Time to Boil" limitations
ii. core or the Fuel Pool storage rack

b) i. RHR Shutdown Cooling can maintain "Time to Boil" limitations
ii. Fuel Pool storage rack only

c) i. iodine release from the design refueling accident is retained by the water and off site doses are maintained within limits,
ii. core or the Fuel Pool storage rack

d) i. iodine release from the design refueling accident is retained by the water and off site doses are maintained within limits,
ii. Fuel Pool storage rack only

Proposed Answer: d) i. iodine release from the design refueling accident is retained by the water and off site doses are maintained within limits,
ii. Fuel Pool storage rack only

Explanation (Optional): **<u>Justification:</u>** In RAP 7.1.04B, section 7.3, the procedure allows, if a radiological emergency exists, for the bundle to be placed in an empty spent fuel rack location. Without this emergency, the bundle, if it can not be placed in its target location shall be returned to its prior location.

QUESTION # 4 Continued

Distracters:

a) i. RHR Shutdown Cooling can maintain "Time to Boil" limitations
   ii. core or the Fuel Pool storage rack

b) i. RHR Shutdown Cooling can maintain "Time to Boil" limitations
   ii. Fuel Pool storage rack only

c) i. iodine release from the design refueling accident is retained by the water and off site doses are maintained within limits,
   ii. core or the Fuel Pool storage rack

Justification for incorrect answers:

The incorrect portion of the distracters are "RHR Shutdown Cooling can maintain "Time to Boil" limitations" and the allowance to be able to store the fuel in any core location.
The level over the flange affects the ability of the water to adsorb heat and the time to boil (plausible distractor). However, the actual reason for the level is for iodine releases.
The fuel is allowed to be stored back into the core per RAP 7.1.04B but it must be returned to its prior location, not ANY location in the core.

Technical Reference(s): AOP-53, RAP-7.1.04.B, TS Bases 3.9.6 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: SDLP-08B EO- 1.17.a (As available)

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

Question History: Last NRC Exam

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

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

10 CFR Part 55 Content: 55.41

QUESTION # 4 Continued

55.43	**2**	2 -Facility operating limitations in the technical specifications and their bases.
	4	4 -Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.
	6	6 -Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming and determination of various internal and external effects on core reactivity.
	7	7-Fuel handling facilities and procedures.

Comments:

QUESTION # 5

Examination Outline Cross-reference:	Level	RO	SRO
High Drywell Pressure / 5	Tier #		1
Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE: **(10CFR 55.43.5)**	Group #		1
	K/A # **295024**		**EA 2.06**
• Suppression pool temperature	Importance Rating		**4.1**

Proposed Question:

<u>The initial plant indications were:</u>
- Reactor Power 100%
- Torus Water Temperature 83 °F
- Torus pressure 0 psig
- Drywell pressure 1.91 psig
- Safety Relief Valve "A" inadvertently opens.

<u>10 Minutes later plant indications are:</u>
- Torus Water Temperature 83 °F
- Torus pressure 11.40 psig
- Drywell pressure 10.90 psig

The above primary containment readings indicate that the suppression function is ____(1)____ and one of the procedures that the CRS is to implement is ____(2)____.

a) (1) working correctly,
(2) AOP-36, Stuck Open Relief Valve(s).

b) (1) **NOT** working correctly,
(2) AOP-1, Reactor Scram.

c) (1) working correctly,
(2) AOP-39, Loss of Coolant.

d) (1) **NOT** working correctly,
(2) AOP-9, Loss of Primary Containment Integrity.

Proposed Answer: b) (1) **NOT** working correctly,
(2) AOP-1, Reactor Scram.

Explanation (Optional): **<u>Justification:</u>** With the opening of the SRV torus temperature remains constant but both torus pressure and pressure rise. Torus pressure is 0.5 psig higher than drywell pressure which indicates that the torus is pressurizing and lifting the torus to drywell vacuum breakers. The high DW pressure caused a scram and AOP-1 entry.

QUESTION # 5 Continued

Distracters: a) (1) working correctly,
(2) AOP-36, Stuck Open Relief Valve(s).
Justification: The lack of a torus temperature rise with both a torus and DW press rise indicates bypass of torus pressure suppression function.
c) (1) working correctly,
(2) AOP-39, Loss of Coolant.
Justification: The lack of a torus temperature rise with both a torus and DW press rise indicates bypass of torus pressure suppression function.
d) (1) **NOT** working correctly,
(2) AOP-9, Loss of Primary Containment Integrity.
Justification: Although Primary Containment is not functioning properly, its integrity remains intact and the entry conditions are **NOT** met for AOP-9.

Technical Reference(s): AOP-1, AOP-9, AOP-36, AOP-39 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: SDLP- 16A EO- 1.09.e & f (As available)

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

Question History: Last NRC Exam

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

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

10 CFR Part 55 Content: 55.41

QUESTION # 5 Continued

55.43	**5**	5 - Assessment of Facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

QUESTION # 6

Examination Outline Cross-reference:	Level	RO	SRO
High Reactor Pressure / 3	Tier #		**1**
Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: **(10CFR 55.43.5)**	Group #		**1**
	K/A # **295025**		**EA 2.04**
• Suspression pool level	Importance Rating		**3.9**

Proposed Question:

The initial plant indications were:

- Reactor Power	100%
- RPV Pressure	1040 psig
- RPV Water Level	201.5 inches
- Torus Water Level	13.96 feet
- Torus Water Temperature	83 °F
- Drywell pressure	1.87 psig
- Steam Tunnel Temperature	120 °F

With NO operator actions 10 Minutes later plant indications are:

- Reactor Mode Switch	RUN
- RPS A & B Scram groups lights	ON
- ARI Valves are	OPEN
- RPV Pressure	A low of 800 psig – slowly rising
- RPV Water Level	A low of 150" - slowly rising
- Torus Water Level	14.12 feet - steady
- Torus Water Temperature	95 °F - steady
- Drywell pressure	1.87 psig - steady
- Steam Tunnel Temperature	140 °F – steady

Which one of the following caused the increase in Suppression Pool level and which procedure must the CRS implement?

a) A small break LOCA inside the drywell, AOP-39, Loss of Coolant.

b) A low vessel level, AOP-42, Feedwater Malfunction (Lowering Feedwater Flow).

c) A high RPV pressure, AOP-1, Reactor Scram.

d) A small main steam line break inside the steam tunnel, AOP 40, Main Steam Line Break.

Proposed Answer: c) A high RPV pressure, AOP-1, Reactor Scram.

QUESTION # 6 Continued

Explanation (Optional): **Justification:** high RPV pressure (determined by ARI valves indications open) resulted in SRV operation and subsequent ARI rod insertion. RPV level was low enough to cause a scram signal at 177" but an ATWS occurred as evidenced by the scram lights being on. This requires entry into AOP-1.

Distracters:
a) A small break LOCA inside the drywell, AOP-39, Loss of Coolant.
b) A low vessel level, AOP-42, Feedwater Malfunction (Lowering Feedwater Flow).
d) A small main steam line break inside the steam tunnel, AOP 40, Main Steam Line Break.

Justification:

a) There is no evidence of a DW leak. DW pressure remains normal 10 minutes into the event.
b) A low vessel level and entry into AOP-42 is appropriate but would not cause a signal to be generated to lift the SRVs which in turn would cause the high torus level.
d) A break in the steam tunnel could cause the MSIVs to go close and, with an ATWS, would cause SRV operation. However, there is no steam tunnel temperature isolation as evidenced by normal temperatures.

Technical Reference(s): AOP-1, AOP-36, AOP-39, AOP-40, (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: SDLP- 29 EO- 1.09.b (As available)

Question Source: Bank # ______

Modified Bank # ______ (Note changes or attach parent)

New **X**

Question History: Last NRC Exam ______

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

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis **X**

QUESTION # 6 Continued

10 CFR Part 55 Content:	55.41		
	55.43	5	5 - Assessment of Facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

QUESTION # 7

		RO	SRO
Examination Outline Cross-reference:	Level	RO	SRO
Reactor Low Water Level / 2	Tier #		1
Knowledge of the process for performing a containment purge. **(10CFR 55.43.4)**	Group #		1
	K/A # **295031**		**G 2.3.9**
	Importance Rating		**3.4**

Proposed Question: The Plant is in Mode 4. To support a maintenance activity, a vent and purge of the drywell is being established per OP-37, Containment Atmosphere Dilution System.

To ensure a purge can be established the _________ trip signal must be reset and, to minimize off-site releases, the CRS must direct the ___________ be used for the purge.

a) Low RPV Level (177 inches), Standby Gas Treatment system,

b) High RPV Pressure (1080 psig), Standby Gas Treatment system,

c) High RPV Level (222.5 inches), Drywell Ventilation and Cooling System,

d) High Drywell pressure (2.7 psig), Drywell Ventilation and Cooling System,

Proposed Answer: a) Low RPV Level (177 inches), Standby Gas Treatment system,

Explanation (Optional): **Justification:** Low RPV level isolates the purge valves and SGT has HEPA and charcoal filters to remove particulates and gaseous radioactive material.

QUESTION # 7 Continued

Distracters:

b) High RPV Pressure (1080 psig), Standby Gas Treatment system,

c) High RPV Level (222.5 inches), Drywell Ventilation and Cooling System,

d) High Drywell pressure (2.7 psig), Drywell Ventilation and Cooling System,

JUSTIFICATION:
Hi DW pressure is the only signal in these 3 distracters that isolates Containment purge. Drywell Ventilation and Cooling does not limit or reduce any airborne activity.

Technical Reference(s): AOP-15, OP-37 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: SDLP- 16C EO- 1.09.c (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

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

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 4 Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Comments:

QUESTION # 8

Examination Outline Cross-reference:	Level	RO	SRO
High Reactor Pressure / 3	Tier #		1
Ability to locate and operate components / including local controls. **(10CFR 55.43.5)**	Group #		2
	K/A # **295007**		**G 2.1.30**
	Importance Rating		**3.4**

Proposed Question: Given the following plant conditions:

Time 0:

APRM power	- 100%
Reactor pressure	- 1039 psig
Recirc Pump 'A' speed	- 86%
Recirc Pump 'B' speed	- 87%
Load Limit Limiting Light	- OFF
#4 Turbine Control Valve	- 40% open
Turbine Bypass Valves	- closed

Time + 3 minutes:

APRM power	- 105%
Reactor pressure	- 1047 psig
Recirc Pump 'A' speed	- 94%
Recirc Pump 'B' speed	- 87%
Load Limit Limiting Light	- ON
#4 Turbine Control Valve	- FULL open
#1 Turbine Bypass Valve	- 65% open

Which one of the following actions must the CRS direct to exit **ALL** active LCOs and return Reactor pressure and power to normal?

a) Run Load Limit up until the Turbine Bypass valves close.

b) Run Load Set up until the Turbine Bypass valves close.

c) Reduce Recirc Pump 'A' speed locally at the scoop tube.

d) Reduce Recirc Pump 'B' speed at panel 09-4.

Proposed Answer: c) Reduce Recirc Pump 'A' speed locally at the scoop tube.

QUESTION # 8 Continued

Explanation (Optional): **Justification:**
Recirc pump 'A' runaway has resulted in Recirculation flow mismatch and entry into LCO 3.4.1. Correct answer c will restore Recirc pump 'A' speed locally and bring the recirculation mismatch within limits.

Distracters: **Justification:**
Distracter d will result in making the mismatch greater. Distracter a & b will results in a raise in reactor pressure.

Technical Reference(s): LCO 3.4.1, OP-27, AOP-32, RAP-7.3.16, OP-9 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: SDLP-02I EO- 1.13. Given the procedure, discuss the procedure steps, administrative limitations, precautions, or cautions for the following: (As available)

c.- operate the scoop tube positioner using the hand crank (OP-27).

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

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

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5 Assessment of Facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

QUESTION # 9

Examination Outline Cross-reference:	Level	RO	SRO
High Reactor Water Level / 2	Tier #		1
Ability to determine and/or interpret the following as they apply to HIGH REACTOR WATER LEVEL: **(10CFR 55.43.5)**	Group #		2
	K/A # **295008**		**AA 2.04**
• Heatup rate: Plant-Specific	Importance Rating		**3.3**

Proposed Question: Reactor Scram from 100% power has just occurred 5 minutes ago. Current post scram plant conditions are as follows:

- RPV Water level dropped to 160 inches and was recovered **rapidly** to 220 inches
- Feedwater/HPCI/RCIC injection is secured to RPV
- RPV Pressure is 800 psig with a trend up at 10 psig per minute
- EHC pressure set is at 970
. psig
- Main Turbine Bypass Jack set is at 0% demand

With no operator action, over the next 5 minutes, RPV Water level will _________. To address the above conditions, the CRS must direct entry and actions of ___________.

a) Lower due to cooldown, AOP-1 "Reactor Scram"

b) Rise due to swell from an open Safety Relief Valve, OP-1 "Main Steam System"

c) Lower due to shrink from an open Main Turbine Bypass Valve, AOP-6 "Malfunction of EHC Pressure Regulator"

d) Rise due to heatup, EOP-2 "RPV Control"

Proposed Answer: d) Rise due to heatup, EOP-2 "RPV Control"

Explanation (Optional): **Justification:** The overfeeding upon the initial scram caused a cooldown and pressure reduction. With feed terminated, decay heat is causing the water to heat and the reactor to pressurize. Since level had dropped to less than 177", entry and actions of EOP-2 apply.

Distracters: a) Lower due to cooldown, AOP-1 "Reactor Scram"
b) Rise due to swell from an open Safety Relief Valve, OP-1 "Main Steam System"
c) Lower due to shrink from an open Main Turbine Bypass Valve, AOP-6 "Malfunction of EHC Pressure Regulator "

Justification:
a) With no feed or steam being drawn, decay heat will cause a heatup.
b) The rate of pressure rise over the next 5 minutes would 50 psig with total RPV pressure being 850 psig, less than the lift setpoint of an SRV.
c) Reactor pressure will be 850 psig in 5 minutes which is less than the 970 psig setpoint of EHC. Since 970 psig is the normal setpoint, there is no reason to believe that the controller has malfunctioned.

QUESTION # 9 Continued

Technical Reference(s): AOP-1, OP-1, AOP-6, EOP-2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: SDLP- 06 EO- 1.09.c (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

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

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5 Assessment of Facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

QUESTION # 10

Examination Outline Cross-reference:	Level	RO	SRO
Inadvertent Reactivity Addition / 1	Tier #		1
Ability to determine and/or interpret the following as they apply to INADVERTENT REACTIVITY ADDITION: **(10CFR 55.43.6)**	Group #		2
	K/A # **295014**		**AA 2.02**
• Reactor period	Importance Rating		**3.9**

Proposed Question: To complete the core refuel, the **last** new bundle of fuel is being lowered into the top guide of the core at location 17-34.
The fuel bundle experiences binding which results in a Slack Cable indication.
Subsequently the bundle frees itself from the obstruction and quickly slides partway into the core till the Hoist Loaded indication is met.
The bundle stops approximately 2/3rds of the way into the core with its full weight on the hoist.

INITIAL SRM INDICATIONS:

SRM 'A'	30 CPS & 90 sec period
SRM 'B'	20 CPS & Infinite period
SRM 'C'	20 CPS & Infinite period
SRM 'D'	15 CPS & Infinite period

SRM INDICATIONS DURING BUNDLE DROP:

SRM 'A'	65 CPS & 20 sec period
SRM 'B'	45 CPS & 90 sec period
SRM 'C'	25 CPS & 25 sec period
SRM 'D'	25 CPS & 120 sec period

Which SRM indications during the bundle movement incident require the evolution to be immediately stopped and the Refuel Bridge SRO to be notified per RAP-7.1.04.B, Neutron Instrumentation Monitoring During In-Core Fuel Handling?

a) SRM 'A' & 'B'.

b) SRM 'B' & 'C'.

c) SRM 'C' & 'D'.

d) SRM 'D' & 'A'.

Proposed Answer: a) SRM 'A' & 'B'.

Explanation (Optional): **Justification:** Per RAP-7.1.04.C Step 8.6 If loading fuel or withdrawing a control rod not immediately adjacent to a SRM/FLC AND count rate DOUBLES, THEN perform the following:
a) Immediately stop the evolution.
b) Notify Refuel Bridge SRO and SM.
This limitation was met with SRM 'A' & 'B' counts.

QUESTION # 10 Continued

Distracters: SRM 'B' & 'C'.
c) SRM 'C'&'D'.
d) SRM 'D' & 'A'.

JUSTIFICATION: Per RAP-7.1.04.C Step 8.6 If loading fuel or withdrawing a control rod not immediately adjacent to a SRM/FLC AND count rate DOUBLES, THEN perform the following:

a) Immediately stop the evolution.
b) Notify Refuel Bridge SRO and SM.

This limitation was met with SRM 'A' & 'B' counts.

Technical Reference(s): RAP-7.1.04.C (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **SDLP-07B Figure # 2**

Learning Objective: SDLP- 07B EO- 1.12.d (As available)

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

Question History: Last NRC Exam

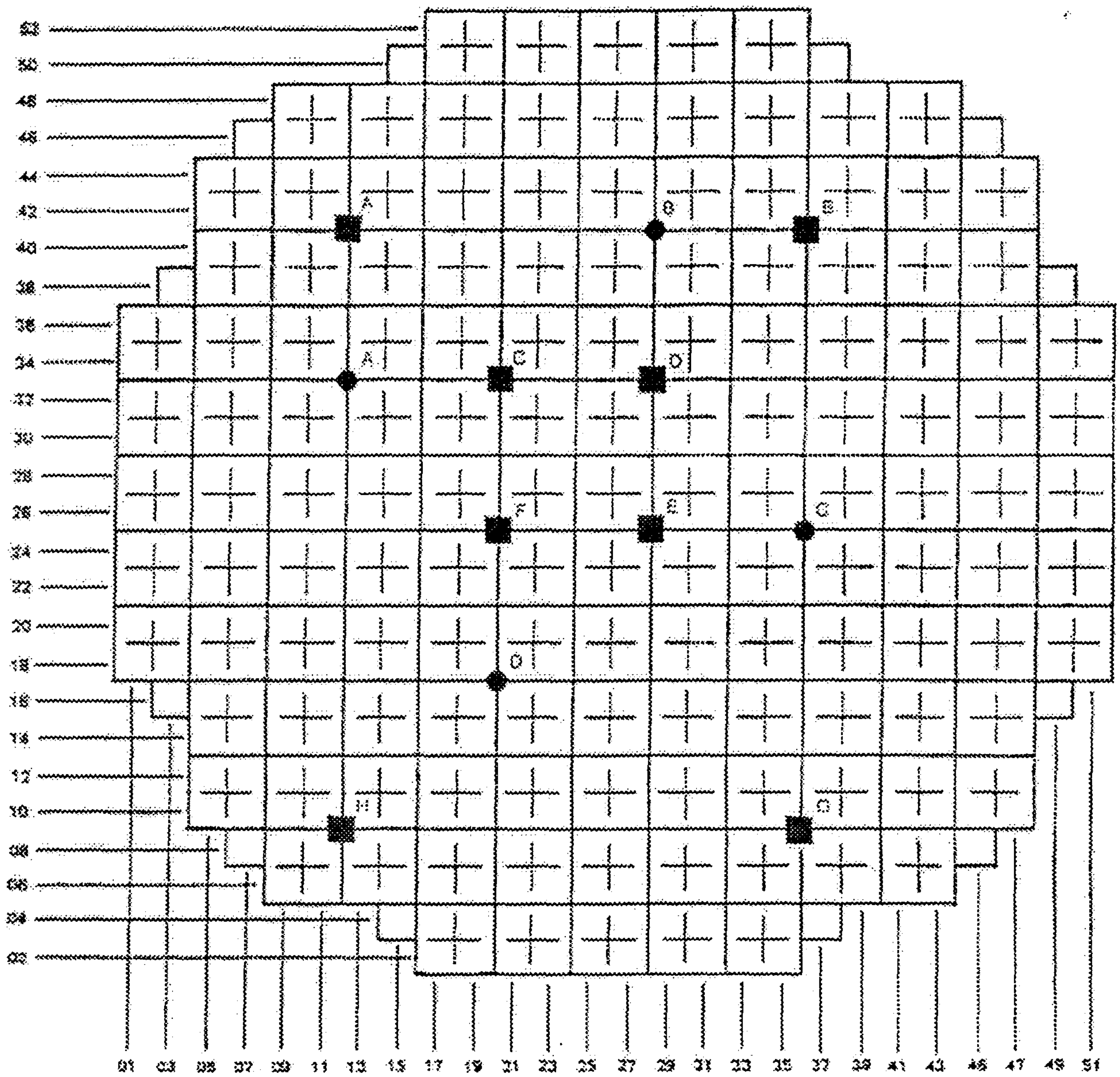
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41
55.43 6 Procedures & limitations involved in initial core loading, alterations in core configuration, control rod programming & determination of various internal & external effects on core reactivity.

Comments:

QUESTION # 10 Continued

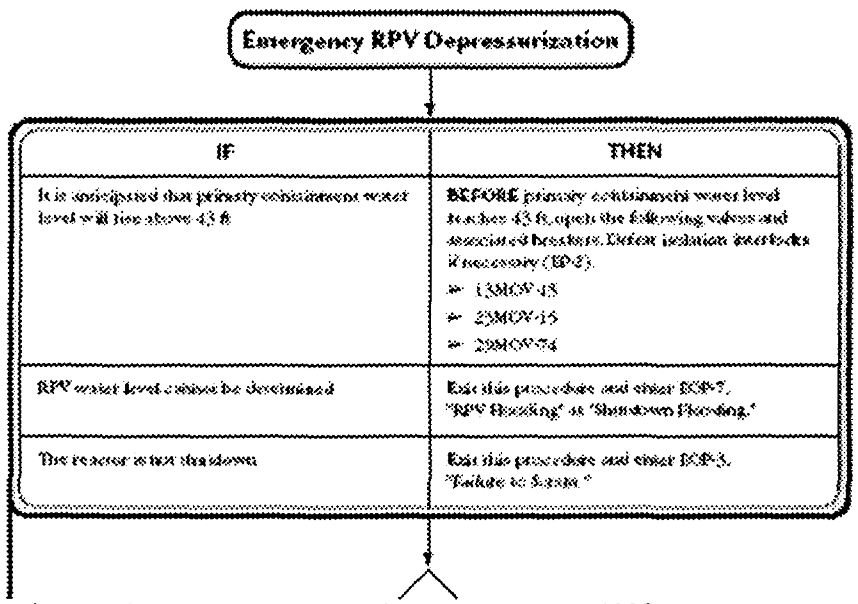


DETECTOR ASSEMBLY IN-CORE LOCATIONS

QUESTION # 11

Examination Outline Cross-reference:	Level	RO	SRO
HPCI	Tier #		2
Knowledge of EOP layout / symbols / and icons. **(10CFR 55.43.5)**	Group #		1
	K/A # **206000**		**G 2.4.19**
	Importance Rating		**3.7**

Proposed Question: Regarding the following EOP-2 step:



The instructions to manipulate the controls for 23MOV-15 are contained in a(n) _____________ and, when following the flowchart, the CRS is to _____________.

a) Major Decision Point, continue and complete the step whenever the "IF" condition is met

b) Override, continue and complete the step whenever the "IF" condition is met

c) Action Statement, stop at this step and wait for the "IF" condition to be met before continuing

d) Hold Point, stop at this step and wait for the "IF" condition to be met before continuing

Proposed Answer: b) Override, continue and complete the step whenever the "IF" condition is met

QUESTION # 11 Continued

Explanation (Optional): **Justification:** Per AP-02.02, EOP & SAOG, Step 5.7 this is an Override that provides guidance on the HPCI system among others. An override must be continuously evaluated during the execution of a series of procedure steps.

Distracters: a) Major Decision Point, continue and complete the step whenever the "IF" condition is met
c) Action Statement, stop at this step and wait for the "IF" condition to be met before continuing
d) Hold Point, stop at this step and wait for the "IF" condition to be met before continuing

Justification:
a). Major Decision Points are enclosed in diamonds.
c) Action statements are simple direct instructions enclosed in rectangles
d) Hold points are enclosed in octagons.

Technical Reference(s): AP-02.02, EOP-2, EOP-3, EOP-4 & EOP-7 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **EOP-2**

Learning Objective: MIT-301.11A, EO- 1.03.h (As available)

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

Question History: Last NRC Exam

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

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

10 CFR Part 55 Content: 55.41
55.43 **5** Assessment of Facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

QUESTION # 12

Examination Outline Cross-reference:	Level	RO	SRO
SLC	Tier #		2
Ability to analyze the effect of maintenance activities on LCO status. **(10CFR 55.43.2)**	Group #		1
	K/A # **211000**		**G 2.2.24**
	Importance Rating		**3.8**

Proposed Question: The Plant is in Mode-1 at 30% power during a startup.
Due to indications of bus overheating, L16 Bus was de-energized in preparation for corrective maintenance.
Compensatory actions have been taken per AOP-19A, Loss of Switchgear L16.
The following items supplied by this bus are being evaluated for Technical Specification LCO actions:
- 11P-2B B SLC Pump
- 01-125FN-1B Standby Gas Treatment Filter Train B Fan Motor
- 13MOV-15 RCIC Steam Supply Inbd Isol Valve

Which of the following is the Technical Specification required action to be taken regarding the evaluation of these three items?

a) Restore SLC B subsystem in 7 days

b) Restore SLC B subsystem in 8 hours

c) Enter LCO 3.03 Immediately

d) Be in Mode 3 in 12 hours with steam dome pressure < 150 psig in 36 hours

Proposed Answer: a) Restore SLC B subsystem in 7 days

Explanation (Optional): **Justification:** Refer to TS 3.1.7 Action A

Distracters: b) Restore SLC B subsystem in 8 hours (Refer to TS 3.1.7 Action B for 2 SLC Inop- only 1 is inop for evaluation) The loss of L16 also causes the loss of tank heater and heat tracing. However, temperatures are Tech. Spec limits and not the heaters.
c) Enter LCO 3.03 Immediately (Refer to TS 3.6.4.3 Action D for 2 SGTs inop- only one SGT is inop for evaluation)
d) Be in Mode 3 in 12 hours with steam dome pressure < 150 psig in 36 hours (Refer to TS 3.5.3 action B incorrect RCIC LCO allows 14 days the Containment Isol Valve is inop as it is failed open, TS 3.6.1.3 Action A.1 requires RCIC Steam Line to be isolated in 4 hours which would then make RCIC inoperable NOTE: this action was NOT provided as part of the distractor)

Technical Reference(s): AOP-19A, TS-3.1.7 Action A & B, TS- 3.6.4.3 Action D, TS- 3.5.3 action A & B, TS- 3.6.1.3 Action A.1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **Tech Specs- No bases**

Learning Objective: SDLP-11, EO- 1.16 (As available)

Question Source: Bank # ______

Modified Bank # ______ (Note changes or attach parent)

QUESTION # 12 Continued

New **X**

Question History: Last NRC Exam ______

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

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 ______

55.43 **2** Facility operating limitations in the tech specifications & their bases.

Comments:

QUESTION # 13

Examination Outline Cross-reference:	Level	RO	SRO
RCIC	Tier #		**2**
Ability to (a) predict the impacts of the following on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: **(10CFR 55.43.5)**	Group #		**1**
	K/A # **217000**		**A 2.11**
• Inadequate system flow	Importance Rating		**3.2**

Proposed Question: The Plant was at 100% when a Scram occurred. Reactor level is 120" and slowly decreasing. RCIC is injecting and has been running for 5 minutes with the following indications:

- RCIC Flow CNTRL 13FIC-91	- 375 gpm
- RCIC Room Temperature	- 100 deg. F
- TURB STM Supp VLV 13MOV-131	- Open
- INJ VLV 13MOV-21	- Open
- MIN FLOW VLV 13MOV-27	- Open
- VAC PMP 13P-3	- Running
- OIL CLR WTR SUPP 13MOV-132	- Open
- TEST VLV TO CST 13MOV-30	- Closed
- INBOARD STEAM SUPPLY VLV 13MOV-15	- Open
- OUTBOARD STEAM SUPPLY VLV 13MOV-16	- Open

With these indications it has been determined that RCIC is **NOT** operating normally.

Which one of the following OP-19 "Reactor Core Isolation Cooling System" procedural sections must the CRS direct to correct this situation?

a) Isolation Verification and Recovery

b) Man. Startup for RPV Pressure Control

c) Auto-Initiation Verification and Subsequent Actions

d) Manual Initiation Using Test Pot (Injection into RPV)

Proposed Answer: c) Auto-Initiation Verification and Subsequent Actions.

Explanation (Optional): **<u>Justification:</u>** RCIC flow is < 410 gpm, the only valve out of position is the Min Flow Valve 13MOV-27 which is Open and must be shut. The auto-initiation procedure has the SNO verify the valves are in the correct position. The SNO will report the mispositioned valve and the CRS will direct its closure.

QUESTION # 13 Continued

Distracters:
a) Isolation Verification and Recovery
b) Man. Startup for RPV Pressure Control
d) Manual Initiation Using Test Pot (Injection into RPV)

Justification: Choices B and D require the auto initiation signal of 126.5" to be clear before proceeding. Choice A does not correct the open MOV-27 valve and is not appropriate because RCIC room temp. is less that the isolation setpoint (~133 deg. F)

Technical Reference(s): AP-02.01, OP-19. Drawing FM-22A (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: SDLP-13, EO- 1.12.b (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

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

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5 Assessment of Facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

QUESTION # 14

Examination Outline Cross-reference:	Level	RO	SRO
SRVs	Tier #		2
Ability to (a) predict the impacts of the following on the RELIEF / SAFETY VALVES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: **(10CFR 55.43.5)**	Group #		1
	K/A # **239002**		**A 2.01**
• Stuck open vacuum breakers	Importance Rating		**3.3**

Proposed Question: While performing ST-22B, Manual Safety Relief Valve Operations and Valve Monitoring System Functional Test (IST), the following Plant conditions were noted:

- RHR is in full Torus Cooling
- Suppression Pool Temperature is 90°F trending down
- Annunciator 09-4-2-6 SRV Sonic Mon Alarm Hi is alarmed
- SRV Sonic Mon Channel 'A' meter is just in the RED region
- SRV 02RV-71A White Light is 'ON' on Panel 09-4
- SRV 02RV-71A Control Switch is in 'Auto' on Panel 09-4
- Torus Water Level is 13.9 feet and steady
- Torus Pressure is 0.03 psig and steady
- Drywell Pressure is 3.0 psig trending up
- Drywell Temperature is 97°F trending up
- Main Turbine Bypass Valves initially cycled closed about 10% when SRV 02RV-71A Control Switch was placed in 'Open'.
- Main Turbine Bypass Valves reopened approximately 7% when SRV 02RV-71A Control Switch was returned to 'Auto' from 'Open'.

The crew has entered AOP-36 Stuck Open Relief Valve(s).

Besides addressing the stuck open SRV, what other failure has occurred and what is the correct procedure to use?

a) SRV 02RV-71A vacuum breaker failed, EOP-4 Primary Containment Control
b) SRV 02RV-71A vacuum breaker failed, AOP-9 Loss of Primary Containment Integrity
c) Turbine Bypass Valves failed, AOP-6 Malfunction of EHC Pressure Regulator
d) Turbine Bypass Valves failed, EOP-2 RPV Control

Proposed Answer: a) SRV 02RV-71A vacuum breaker failed, EOP-4 Primary Containment Control

QUESTION # 14 Continued

Explanation (Optional): **Justification** The SRV is stuck partially open, it is discharging directly into the DW through the SRV vacuum breakers as noted by DW press & temp increases & it is NOT going into the TORUS as noted by Torus temp & pressure. The Main Turbine Bypass valves (BPV's) have responded to the change in SRV position, initially closing about 10% when the SRV was open & would be expected to re-open 10% if the SRV went full shut, in this case they went open only 7% due to the SRV being partly open so they are responding correctly. EOP-4 is entered to mitigate containment challenges from direct pressurization due to the SRV Vacuum breaker being open with the SRV partially open. EOP-2 is a correct procedure to enter but when tied with the BPV's failure it is the **wrong** choice. AOP-6 is wrong as the EHC Pressure regulator has NOT malfunctioned. AOP-9 Loss of Primary Containment Integrity Entry conditions are NOT met. From SDLP-02J, A failure of the vacuum Breakers to close would admit steam to the DW air space, resulting in rising DW press & temp, upon subsequent SRV opening.

Distracters:
b) SRV 02RV-71A vacuum breaker failed, AOP-9 Loss of Primary Containment Integrity
c) Turbine Bypass Valves failed, AOP-6 Malfunction of EHC Pressure Regulator
d) Turbine Bypass Valves failed, EOP-2 RPV Control

Justification: SRV is stuck partially open, it is discharging directly into the DW through the SRV vacuum breakers as noted by DW press & temp increases & it is NOT going into the TORUS as noted by Torus temp & pressure. The Main Turbine Bypass valves (BPV's) have responded to the change in SRV position, initially closing about 10% when the SRV was open & would be expected to re-open 10% if the SRV went full shut, in this case they went open only 7% due to the SRV being partly open so they are responding correctly. EOP-4 is entered to mitigate containment challenges from direct pressurization due to the SRV Vacuum breaker being open with the SRV partially open. EOP-2 is a correct procedure to enter but when tied with the BPV's failure it is the **wrong** choice. AOP-6 is wrong as the EHC Pressure regulator has NOT malfunctioned. AOP-9 Loss of Primary Containment Integrity Entry conditions are NOT met. From SDLP-02J, A failure of the vacuum Breakers to close would admit steam to the DW air space, resulting in rising DW press & temp, upon subsequent SRV opening.

Technical Reference(s): ST-22B, AOP-6, AOP-9, AOP-36, EOP-2, EOP-4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **None**

Learning Objective: SDLP-02J, EO- 1.09.f (As available)

Question Source: Bank # ____

Modified Bank # ____ (Note changes or attach parent)

New **X**

QUESTION # 14 Continued

Question History: Last NRC Exam ________

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

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		
	55.43	5	Assessment of Facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

QUESTION # 15

Examination Outline Cross-reference:	Level	RO	SRO
Reactor Water Level Control	Tier #		2
Knowledge of the process for controlling temporary changes. **(10CFR 55.43.3)**	Group #		1
	K/A # **259002**		**G 2.2.11**
	Importance Rating		**3.4**

Proposed Question: There is a tagout expected to be in place for greater than 90 days that tags out the "A" level column. In preparation for this tagout the 'B' level column is to be selected for Feedwater Level Control. The tagout is to support a proposed change to Technical Specifications to move TS 3.3.2.2 Feedwater and Main Turbine High Water Trip Instrumentation to the Technical Requirements Manual (TRM).

The selection of the 'B' Level Column and the associated tagout requires a _______. The addition of the Feedwater and Main Turbine High Water Trip Instrumentation to the TRM is controlled with _______.

a) Temporary Change to OP-2A Feedwater System, AP-02.04 Control of Procedures

b) Temporary Change to OP-2A Feedwater System, AP-01.02 License and Technical Specification Administration

c) 50.59 Screen per ST-1X Protected Tags and Temporary Alterations Audit, AP-02.01 Procedure Writers Manual

d) 50.59 Screen per ST-1X Protected Tags and Temporary Alterations Audit, AP-20.06 Final Safety Analysis Report (FSAR) Amendment Preparation and Control

Proposed Answer: d) 50.59 Screen per ST-1X Protected Tags and Temporary Alterations Audit, AP-20.06 Final Safety Analysis Report (FSAR) Amendment Preparation and Control

Explanation (Optional): 50.59 Screen is required for tagouts expected to be in place >30 days per ST-1X Protected Tags and Temporary Alterations Audit, AP-20.06 Final Safety Analysis Report (FSAR) Amendment Preparation and Control controls changes to the TRM. Temporary Change to OP-2A is NOT required as it has a section G.29 for swapping from Water Column 'A' to 'B'.

Distracters: a) Temporary Change to OP-2A Feedwater System, AP-02.04 Control of Procedures

b) Temporary Change to OP-2A Feedwater System, AP-01.02 License and Technical Specification Administration

c) 50.59 Screen per ST-1X Protected Tags and Temporary Alterations Audit, AP-20.06 Final Safety Analysis Report (FSAR) Amendment Preparation and Control

Justification: 50.59 Screen is required for tagouts expected to be in place >30 days per ST-1X Protected Tags and Temporary Alterations Audit, AP-20.06 Final Safety Analysis Report (FSAR) Amendment Preparation and Control controls changes to the TRM. Temporary Change to OP-2A is NOT required as it has a section G.29 for swapping from Water Column 'A' to 'B'.

Technical Reference(s): AP-20.6, ST-1X, AP-02.01, AP-01.02, AP-02.04, TRM, TS-3.3.2.2 (Attach if not previously provided)

QUESTION # 15 Continued

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: LP AP, EO- 1.01 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History: Last NRC Exam

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

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41

55.43 **3** Facility licensee procedures required to obtain authority for design and operating changes in the facility.

Comments:

QUESTION # 16

Examination Outline Cross-reference:	Level	RO	SRO
Control Rod and Drive Mechanism	Tier #		2
Ability to (a) predict the impacts of the following on the CONTROL ROD AND DRIVE MECHANISM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: **(10CFR 55.43.5)**	Group #		2
	K/A # **201003**		**A 2.01**
• *Stuck rod*	Importance Rating		**3.6**

Proposed Question: A plant startup and heatup is in progress with RPV pressure at 600 psig.

The following conditions were noted when the ROD MOVEMENT CNTRL switch was taken to "Out Notch" to move the selected rod to position 12:

09-5-2-1 RWM ROD BLOCK RPIS INOP – clear
09-5-2-2 ROD WITHDRAWAL BLOCK - clear
09-5-2-3 ROD DRIFT - clear
09-5-2-4 ROD OVER TRAVEL - clear
Rod 22-39 indicating light on Full Core Display - "ON"
Rod Out Perm light - "ON"
Rod 22-39 position – 10
Rod In Green light - cycled "ON" and "OFF"
Rod Out Red light - cycled "ON" and "OFF"
Rod Settle Amber light - cycled "ON" and "OFF"
IRMs - all mid scale Range 8 and steady with no change
<u>Panel 09-5 Indications:</u>
03PDI-302 CHG WTR Press – 1500 psig
03PDI-303 DRV WTR Diff Press – 650 psid
03PDI-304 CLG WTR Diff Press – 21 psid
03PDI-305 DRV WTR Flow – 0 gpm
03PDI-306 CLG WTR Flow – 60 gpm
<u>Local Indications:</u>
03FI-216 Stab Valves A & B Outlet Flow Ind – 6 gpm

The impact to the plant and equipment is _________ and the CRS is to enter __________.

a) damage to the drive mechanism seals, AOP-24 Stuck Control Rod

b) over-heating the drive mechanism seals, AOP-24 Stuck Control Rod

c) excessive control rod drive speeds, AOP-25 Uncoupled Control Rod

d) excessive reactivity addition rate, AOP-25 Uncoupled Control Rod

Proposed Answer: a) damage to the drive mechanism seals, AOP-24 Stuck Control Rod

QUESTION # 16 Continued

Explanation (Optional): Indications are rod did NOT move, both AOPs have symptoms of lack of NI response to rod movement while AOP-24 includes RPIS failure to indicate rod motion. The candidate must also determine Drive D/P is excessive, > 600 psid with RPV pressure < 650 psig & per procedure caution, this condition could damage the drive mechanism seals. Overheating the seals would only apply if the cooling water was isolated but this was NOT done per the stem. The distracters for AOP-25 are part of a caution in regards to individual Scram to re-couple.

Distracters:
b) over-heating the drive mechanism seals, AOP-24 Stuck Control Rod
c) excessive control rod drive speeds, AOP-25 Uncoupled Control Rod
d) excessive reactivity addition rate, AOP-25 Uncoupled Control Rod

Justification: Indications are rod did NOT move, both AOPs have symptoms of lack of NI response to rod movement while AOP-24 includes RPIS failure to indicate rod motion. The candidate must also determine Drive D/P is excessive, > 600 psid with RPV pressure < 650 psig & per procedure caution, this condition could damage the drive mechanism seals. Overheating the seals would only apply if the cooling water was isolated but this was NOT done per the stem. The distracters for AOP-25 are conditions that could occur an uncoupled rod and a stuck control rod. A successful recoupling and rod movement could cause excessive reactivity addition. Also a drive that does not have the weight of the blade on it would move at a faster rate than normal.

Technical Reference(s): AOP-24, AOP-25 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: LP AOP, EO- 1.04 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

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

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5 Assessment of Facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

QUESTION # 17

Examination Outline Cross-reference:	Level	RO	SRO
Nuclear Boiler Inst.	Tier #		2
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: **(10CFR 55.43.5)**	Group #		2
	K/A # **216000**		**A 2.03**
• Instrument line leakage	Importance Rating		**3.1**

Proposed Question: The plant is at 100%. Feedwater Level Control is in 3-element control and is selected to RPV Water Level Column 'B'.

A report is received from a Radiation Protection Technician that the Reactor Building 344 ft ARM is in ALARM and steam and water is leaking into Reactor Building 300'.

Coincident with the above the Control Room has the following indications:
- 09-5-1-28 RX WTR LVL ALARM HI OR LO – "ON"
- 09-5-2-29 FDWTR CNTRL A OR B OR C HI RX LVL TRIP – "ON"
- EPIC Pt #92- RFP HI WTR LVL A TRIP - "NORMAL"
- EPIC Pt #93- RFP HI WTR LVL B TRIP - "TRIPPED"
- EPIC Pt #94- RFP HI WTR LVL C TRIP - "NORMAL"

WITHOUT *any operator actions the plant will respond by* ________, The crew response, **PRIOR** to receiving any other alarms or indication changes shall be per _______.

a) Scramming, AOP-41 FEEDWATER MALFUNCTION (RISING FEEDWATER FLOW – HIGH RPV WATER LEVEL)and AOP-39 LOSS OF COOLANT

b) Scramming, AOP-42 FEEDWATER MALFUNCTION (LOWERING FEEDWATER FLOW) and EOP-5 SECONDARY CONTAINMENT CONTROL

c) Continuing operation at a **higher** RPV level, AOP-41 FEEDWATER MALFUNCTION (RISING FEEDWATER FLOW – HIGH RPV WATER LEVEL) and EOP-5 SECONDARY CONTAINMENT CONTROL

d) Continuing operation at a **lower** RPV level, AOP-42 FEEDWATER MALFUNCTION (LOWERING FEEDWATER FLOW) and AOP-39 LOSS OF COOLANT

Proposed Answer: b) Scramming, AOP-42 FEEDWATER MALFUNCTION (LOWERING FEEDWATER FLOW) and EOP-5 SECONDARY CONTAINMENT CONTROL

QUESTION # 17 Continued

Explanation (Optional): Stem symptoms indicate 06LT-52B RPV WTR LEVEL X-mitter has a reference side instrument line break, D/P went to 0 resulting in a indicated high level as confirmed by annunciators & EPIC point with a confirmation provided by the RP Tech that line break is outside the CNMT in the RB. Report stipulates ARM alarm which would be an EOP-5 entry & exit from AOP-9. Without operator action, RPV level would lower to the low SCRAM setpoint as FW backs down due to "B" level x-mitter in control & it would be the dominant control signal over Steam & Feed flow. Crew response prior to receiving further alarms is to enter AOP-42 based on indications & EOP-5 based on RP Tech report. NO entry symptoms are present for AOP-41, AOP-1 or AOP-39.

Distracters:

a) Scramming, AOP-41 FEEDWATER MALFUNCTION (RISING FEEDWATER FLOW – HIGH RPV WATER LEVEL)and AOP-39 LOSS OF COOLANT

c) Continuing operation at a **higher** RPV level, AOP-41 FEEDWATER MALFUNCTION (RISING FEEDWATER FLOW – HIGH RPV WATER LEVEL) and EOP-5 SECONDARY CONTAINMENT CONTROL

d) Continuing operation at a **lower** RPV level, AOP-42 FEEDWATER MALFUNCTION (LOWERING FEEDWATER FLOW) and AOP-39 LOSS OF COOLANT

Justification: Stem symptoms indicate 06LT-52B RPV WTR LEVEL X-mitter has a reference side instrument line break, D/P went to 0 resulting in a indicated high level as confirmed by annunciators & EPIC point with a confirmation provided by the RP Tech that line break is outside the CNMT in the RB. Report stipulates ARM alarm which would be an EOP-5 entry & exit from AOP-9. Without operator action, RPV level would lower to the low SCRAM setpoint as FW backs down due to "B" level x-mitter in control & it would be the dominant control signal over Steam & Feed flow. Crew response prior to receiving further alarms is to enter AOP-42 based on indications & EOP-5 based on RP Tech report. NO entry symptoms are present for AOP-41, AOP-1 or AOP-39.

Technical Reference(s): AOP-41, AOP-42, AOP-9, AOP-1, EOP-5 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **EOP 5**

Learning Objective: SDLP-06, EO- 1.10.d (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History: Last NRC Exam

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

Question Cognitive Level: Memory or Fundamental Knowledge

QUESTION # 17 Continued

	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:

QUESTION # 18

Examination Outline Cross-reference:	Level	RO	SRO
RHR/LPCI: Torus/Pool Cooling Mode	Tier #		**2**
Ability to (a) predict the impacts of the following on the RHR/LPCI: TORUS / SUPPRESSION POOL COOLING MODE; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: **(10CFR 55.43.5)**	Group #		**2**
	K/A # **219000**		**A 2.12**
• Valve logic failure: Plant-Specific	Importance Rating		**3.1**

Proposed Question: The plant is operating at 100% power. The SNO is operating RHR for surveillance and reports that 10MOV-66A "RHR HEAT EXCH A BYPASS VLV" will not close.

The CRS must declare the _________ mode of RHR inoperable and implement procedure________.

a) Torus cooling, AP-10.01 Work Order Processing

b) Torus cooling, AP-20.13 10CFR21 Reporting

c) LPCI flow, AP-05.13 Maintenance During LCOs

d) LPCI flow, AP-12.08 LCO Tracking and Safety Function Determination Program

Proposed Answer: a) Torus cooling, AP-10.01 Work Order Processing

Explanation (Optional): With RHR HEAT EXCH A BYPASS VLV failing to remain shut due to the valve logic, it will NOT meet required heat removal capacity in A RHR loop. Procedures that apply are AP-10.01 (initiate a Work request), AP-12.08 & AP-05.13 are applicable procedures to be used but are NOT correct choices due to RHR will provide the required LPCI flow. AP-20.13 is NOT correct as a simple valve failure does not meet the reporting requirements.

Distracters: b) Torus cooling, AP-20.13 10CFR21 Reporting
c) LPCI flow, AP-05.13 Maintenance During LCOs
d) LPCI flow, AP-12.08 LCO Tracking and Safety Function Determination Program

Justification:
AP-12.08 & AP-05.13 are applicable procedures to be used but are NOT correct choices due to RHR will provide the required LPCI flow. AP-20.13 is NOT correct as a simple valve failure does not meet the reporting requirements

Technical Reference(s): OP-13, OP-13B, TS-B3.5.1, B3.6.2.3, AP-10.01, AP-01.02, AP-05.13, AP-12.08 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: SDLP-10, EO- 1.09.d.2 (As available)

QUESTION # 18 Continued

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		

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

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

10 CFR Part 55 Content:	55.41		
	55.43	5	Assessment of Facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

QUESTION # 19

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
Knowledge of less than one hour technical specification action statements for systems. **(10CFR 55.43.2)**	Group #		1
	K/A # **2.1**		**G 2.1.11**
	Importance Rating		**3.8**

Proposed Question: The Plant is at 100% Power on 4/16/06.

- At 19:00 on 4/16/06 a loss of Lighthouse Hill-Fitzpatrick Line # 3 occurs.
- ST-9W Electrical Lineup and Power Verification was last performed at 17:00 on 4/16/06 per regularly scheduled surveillance frequency.

Applicable portions of ST-9R EDG System Quick-Start Operability Test and Offsite Circuit Verification must be performed next by__________.

a) 17:00 on 4/23/06

b) 11:00 on 4/25/06

c) 01:00 on 4/17/06

d) 20:00 on 4/16/06

Proposed Answer: d) 20:00 on 4/16/06

Explanation (Optional): Stem provided indications that one offsite power source was lost and thus requires entry into LCO 3.8.1 Action A requiring completion of SR 3.8.1.1 within 1 hour & once per 8 hours there-after to ensure that offsite power is available. LCO entry time is 19:00 making SR due by 20:00. The 17:00 on 4/23/06 is the normal 7 day frequency based upon last completing the SR at 17:00 on 4/16/06. The choice of 11:00 on 4/25/06 allows for 1.25 extension of the normal 7 day frequency based upon SR 3.02. The choice of 01:00 on 4/17/06 is an 8 hour time based upon last completing the SR at 17:00 on 4/16/06. SR 3.02 is NOT allowed for the, perform within 1 hour- see example 1.4-2 in TS.

Distracters: a) 17:00 on 4/23/06
b) 11:00 on 4/25/06
c) 01:00 on 4/17/06
Justification: Stem provided indications that one offsite power source was lost and thus requires entry into LCO 3.8.1 Action A requiring completion of SR 3.8.1.1 within 1 hour & once per 8 hours there-after to ensure that offsite power is available. LCO entry time is 19:00 making SR due by 20:00. The 17:00 on 4/23/06 is the normal 7 day frequency based upon last completing the SR at 17:00 on 4/16/06. The choice of 11:00 on 4/25/06 allows for 1.25 extension of the normal 7 day frequency based upon SR 3.02. The choice of 01:00 on 4/17/06 is an 8 hour time based upon last completing the SR at 17:00 on 4/16/06. SR 3.02 is NOT allowed for the, perform within 1 hour- see example 1.4-2 in TS.

QUESTION # 19 Continued

Technical Reference(s):	AOP-72, ST-9W, ST-9R, TS-SR 3.02, TS-3.8.1, SR-3.8.1.1, TS-B3.8.1	(Attach if not previously provided)

Proposed references to be provided to applicants during examination: **Technical Specs- No Bases**

Learning Objective:	SDLP-71D, EO- 1.16		(As available)
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		

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

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

10 CFR Part 55 Content:	55.41		
	55.43	2	Facility operating limitations in the technical specifications and their bases.

Comments:

QUESTION # 20

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		**3**
Ability to supervise and assume a management role during plant transients and upset conditions. **(10CFR 55.43.5)**	Group #		**1**
	K/A # **2.1**		**G 2.1.6**
	Importance Rating		**4.3**

Proposed Question: The Plant is at 70% power to allow removing 'A' Feedwater Pump from service for maintenance.

- All 3 Circ Water Pumps are in service.
- Tempering is in progress to maintain Cond Demin inlet temperature 95-100°F.
- Lake level went from 245.5 ft to 242.5 ft in the last 8 hours as noted on EPIC Log 1.
- The Outside NPO reports that there is NO ice formation on the traveling screens or intake structure.
- Traveling Screens are 'Continuous Run' with indications of debris in the Fish Basket.

In accordance with AOP-64 Loss of Intake Water Level, which of the following actions will be the next required action to perform?

a) Raise tempering flow per OP-4 Circulating Water System

b) Reduce Reactor Power to less than 65% per RAP-7.3.16 Plant Power Changes

c) Stop Circ Wtr Pump C 36P-1C per OP-4 Circulating Water System

d) Manually Scram the Plant per AOP-1 Reactor Scram

Proposed Answer: b) Reduce Reactor Power to less than 65% per RAP-7.3.16 Plant Power Changes

Explanation (Optional): Indications are provided that the loss of intake level is due to other than ice formation. AOP-64 requires power reduced to < 65% prior to stopping CW Pump C if lake level has lowered > 2 ft in the last 8 hours as stipulated in the stem. The plant is manually scrammed if lake level is < 240 ft which was NOT given in the stem. Raise tempering flow per OP-4 section G if Ice formation is the cause of the lowering level, this was NOT given in the stem, in fact NO indications reported locally of ice formation in the intake structure was provided in the stem.

QUESTION # 20 Continued

Distracters: a) Raise tempering flow per OP-4 Circulating Water System
c) Stop Circ Wtr Pump C 36P-1C per OP-4 Circulating Water System
d) Manually Scram the Plant per AOP-1 Reactor Scram

Justification: Indications are provided that the loss of intake level is due to other than ice formation. AOP-64 requires power reduced to < 65% prior to stopping CW Pump C if lake level has lowered > 2 ft in the last 8 hours as stipulated in the stem. The plant is manually scrammed if lake level is < 240 ft which was NOT given in the stem. Raise tempering flow per OP-4 section G if Ice formation is the cause of the lowering level, this was NOT given in the stem, in fact NO indications reported locally of ice formation in the intake structure was provided in the stem.

Technical Reference(s): AOP-64 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: LPAOP, EO- 1.03.a (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

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

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5 Assessment of Facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

QUESTION # 21

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
Knowledge of the process for managing maintenance activities during power operations. **(10CFR 55.43.5)**	Group #		2
	K/A # **2.2**		**G 2.2.17**
	Importance Rating		**3.5**

Proposed Question: The Plant is at 100% power on a Sunday night. The Operating Shift has determined that the Plant will be de-rated to 65% power to support emergent repair work.

Per procedure AP-12.13, "345/115 KV Transmission Line Operations and Interface", the NYPA Energy Control Center is required to be notified by the____________.

a) Shift Manager

b) Reactor Engineer

c) Operations Manager

d) Field Support Supervisor

Proposed Answer: a) Shift Manager

Explanation (Optional): AP-12.13 provides guidance for NYPA ECC interface for this situation while OP-65 has a step to notify ECC of a shutdown schedule, the plant is NOT being shutdown but is being de-rated. Per AP-12.13, Operations Manager is responsible for overall implementation of this procedure. Reactor Engineer (RE) is responsible to coordinate generation scheduling with ENN Power Marketing. **Advanced Scheduling:** All generation scheduling will be done between ENN Power Marketing & RE Dept. RE must notify ENN On-Call Scheduler at least 7 days in advance of any planned power changes. The Shift Manager (SM) is responsible for authorizing access to JAF Switchyard, communicating with transmission operator for resolving emergent issues. SM is responsible for changes to unit power scheduling with NYISO. Work Control Center Supervisor (WCCS)is responsible for ensuring that 115/345KV work with potential to affect operation of JAF, are scheduled on the weekly work schedule per AP-10.02 "12 Week Rolling Schedule", & coordinated by the JAF 115/345 KV Coordinator. JAF 115KV/345KV Coordinator is responsible to interface with Power Control & Regional Central Control to review, coordinate & schedule line outages & work that has potential for causing an unplanned line outage. WCCS is the 115/345 KV coordinator. **Real-Time Operations:** For unplanned down powers or delayed power restorations, JAF Ops is required to contact NYPA ECC for a "derate", the term "derate" must be used & give plant status. FOR the stem conditions, the only member of Operations present at Sunday night would be the SM as the OM is off.

QUESTION # 21 Continued

Distracters:
b) Reactor Engineer
c) Operations Manager
d) Field Support Supervisor

Justification: Per AP-12.13, Operations Manager is responsible for overall implementation of this procedure. Reactor Engineer (RE) is responsible to coordinate generation scheduling with ENN Power Marketing. **Advanced Scheduling:** All generation scheduling will be done between ENN Power Marketing & RE Dept. RE must notify ENN On-Call Scheduler at least 7 days in advance of any planned power changes. The Shift Manager (SM) is responsible for authorizing access to JAF Switchyard, communicating with transmission operator for resolving emergent issues. SM is responsible for changes to unit power scheduling with NYISO. **Real-Time Operations:** For unplanned down powers or delayed power restorations, JAF Ops is required to contact NYPA ECC for a "derate", the term "derate" must be used & give plant status. FOR the stem conditions, the only member of Operations present at Sunday night would be the SM as the OM is off. The Field Support Supervisor is part of a normal crew on Sunday but has not responsibility in AP-12.13.

Technical Reference(s): AP-12.13, EN-OP-115, RAP-7.3.16 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: LPAP, EO- 20.02 (As available)

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

Question History: Last NRC Exam

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

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

10 CFR Part 55 Content: 55.41

QUESTION # 21 Continued

55.43	**5**	Assessment of Facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

QUESTION # 22

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		**3**
Knowledge of the refueling process. **(10CFR 55.43.6)**	Group #		**2**
	K/A # **2.2**		**G 2.2.27**
	Importance Rating		**3.5**

Proposed Question: The plant is in a refuel outage and fuel movement is underway. The next move per SNM move sheet 06-058, step 275, is fuel bundle (YJX 230) which is being moved into core location 43-48 (Clip NE).

A Refuel Error per RAP-7.1.04B, Refueling Procedure, would occur if the bundle__________.

a) has its nose cone partially inserted into 43-48 (Clip SE) then is changed to 43-48 (Clip NE) prior to the start of the next move.

b) is inserted 30 inches into location 03-32 (Clip NE) and is subsequently removed and inserted into location 43-48 (Clip NE).

c) Is moved from core location 39-42 (Clip SE) and is returned to location 39-42 (Clip SE) due to poor visibility in location 43-48.

d) is fully inserted into 43-48 (Clip SE) with the grapple disengaged and then is changed to 43-48 (Clip NE) prior to the start of the next move.

Proposed Answer: b) is inserted 30 inches into location 03-32 (Clip NE) and is subsequently removed and inserted into location 43-48 (Clip NE).

Explanation (Optional): Per RAP-1.1.04B, 5.9 Refuel Error, the only choice that constitutes a refuel error is 'B' as the nose cone is partially inserted into the wrong location. Distractor 'A' & 'D' are wrong orientation that is corrected prior to start of next move. Distractor 'C' is <u>NOT</u> a refuel error per step 5.9.3.

5.9.1 A fuel bundle fully or partially placed (i.e., past the nose cone) in an incorrect location is a Refuel Error.

5.9.2 A mis-orientated fuel bundle <u>is not</u> considered a Refuel Error if the miss-orientation is corrected immediately. It is a Refuel Error if the mis-orientated bundle is identified after the start of the next move.

5.9.3 It is not a Refuel Error if a move cannot be completed for any reason and the bundle is returned to the starting location.

QUESTION # 22 Continued

Distracters: a) has its nose cone partially inserted into 43-48 (Clip SE) then is changed to 43-48 (Clip NE) prior to the start of the next move.
c) Is moved from core location 39-42 (Clip SE) and is returned to location 39-42 (Clip SE) due to poor visibility in location 43-48.
d) is fully inserted into 43-48 (Clip SE) with the grapple disengaged and then is changed to 43-48 (Clip NE) prior to the start of the next move.

Justification:

Per RAP-1.1.04B, 5.9 Refuel Error, the only choice that constitutes a refuel error is 'B' as the nose cone is partially inserted into the wrong location. Distractor 'A' & 'D' are wrong orientation that is corrected prior to start of next move. Distractor 'C' is NOT a refuel error per step 5.9.3.

5.9.1 A fuel bundle fully or partially placed (i.e., past the nose cone) in an incorrect location is a Refuel Error.

5.9.2 A mis-orientated fuel bundle is not considered a Refuel Error if the miss-orientation is corrected immediately. It is a Refuel Error if the mis-orientated bundle is identified after the start of the next move.

5.9.3 It is not a Refuel Error if a move cannot be completed for any reason and the bundle is returned to the starting location.

Technical Reference(s): RAP-7.1.04B (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: LPAP, EO- 73.04 (As available)

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

Question History: Last NRC Exam

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

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

10 CFR Part 55 Content: 55.41
55.43 6 Procedures & limitations involved in initial core loading, alterations in core configuration, control rod programming & determination of various internal & external effects on core reactivity.

Comments:

QUESTION # 23

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
Knowledge of the requirements for reviewing and approving release permits. **(10CFR 55.43.4)**	Group #		3
	K/A # **2.3**		**G 2.3.6**
	Importance Rating		**3.1**

Proposed Question: The plant is in cold shutdown and all equipment is operable. A liquid Radwaste discharge to the canal is about to occur. An independent review of the Canal Discharge Worksheet (attached) by the FSS shows that the discharge can NOT take place. The reason for this is that:

a) The Chemistry Superintendent's signature is required

b) An Independent Analysis signature is required

c) The radiation monitor, 17RM-350, alarm and isolation setpoints are to be set lower

d) The radiation monitor, 17RM-350, alarm and isolation setpoints are to be set higher

Proposed Answer: d) The radiation monitor, 17RM-350, alarm and isolation setpoints are to be set higher

Explanation (Optional): The canal discharge activity level is obtained from the discharge permit. The permit activity is larger than the number recorded on the worksheet. A larger activity number, if used on the worksheet, would calculate to a higher monitor setpoint. Thus, since a lower activity number was used, a lower alarm and setpoint were used.

Choice A is wrong because the Chemistry Superintendents signature is required if the minimum CW pumps (1) are not operating.
Choice B is wrong because an independent analysis is required if the radiation monitor is inoperable.
Choice C is wrong because the actual activity number, 3.8xE-4 is larger than the number on the worksheet of 2.8xE-5.

The question is higher order in that the candidate must analyze the calculations and required signatures to see what mistake was made. With the mistake determined to be an incorrect transcribe activity number he must analyze and determine how this affects the setpoint settings without having the formula to actually calculate the setpoint.

QUESTION # 23 Continued

Distracters:

Technical Reference(s): OP-49 Liquid Radioactive Waste System (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **Provide calculator and filled in discharge permit and worksheet.**

Learning Objective: SDLP-20, EO 1.13 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

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

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 4 Radiation hazards that may arise during normal & abnormal situations, including maintenance activities & various contamination conditions.

Comments:

LIQUID RADIOACTIVE WASTE DISCHARGE PERMIT Page 1 of 1

Tank: "A" WST	Release Batch No. 43	Sample Date/Time 6/18/06 0600

Section A To: Shift Manager From: Chemistry Department

Required Dilution Factor 100 DF	Gamma Activity: 3.8 x 10⁻⁴ µCi/ml	
Required Dilution Water Pumps	Circ Pumps: 1	Service Water: 2
Percent Tempering 0 %	Maximum Discharge Rate: 20 gpm	
Monitor is Calibrated and Source Checked SAT / UNSAT	Technician Initials: J.A.	
Technician Signature (Print/Sign): John Allen John Allen	Date	Time

Independent analysis is required for discharge without operable radiation monitor. Otherwise, mark NA.		
Independent Analysis by (Print/Sign): NA	Date 6/18/06	Time 0700

Section B To: Auxiliary Operator From: Shift Manager

Alarm Potentiometer Settings	HI 6.7	Hi/Hi 7.0
A MINIMUM OF ONE OPERATING CIRC PUMP REQUIRED FOR TANK DISCHARGE UNLESS AUTHORIZED BY CHEMISTRY SUPT.		Signature
Effluent Radiation Monitor OPERABLE / INOPERABLE		
Shift Manager Authorization (Print/Sign):	Date	Time

Section C To: Chemistry Department From: Shift Manager

Discharge Valve Line-Up Performed by:	Date	Time
If effluent radiation monitor is INOPERABLE, then an independent verification of discharge valve line-up by a qualified individual is required. Otherwise, mark NA.		
Independently Verified by (Print/Sign):	Date	Time

Flow Rate Instrument Daily Check Complete? YES NO

DISCHARGE DATA	Date	Time	Dilution Water Pumps Operating Circ Service	Level	Rate	Operator Initials
Start Pumpout						
End Pumpout						

During Discharge:

Rad Rcdr Reading 17RR-337 cps	Flw Rcdr Reading 20FR-441 gpm

Discharge valve line-up returned to normal in accordance with canal discharge shutdown lineup for applicable tank:

Tank:	Aux. Operator (Print/Sign):	Date	Time

FORWARD DISCHARGE PERMIT TO CHEMISTRY IMMEDIATELY FOLLOWING DISCHARGE

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CANAL DISCHARGE WORKSHEET Page 1 of 2

DATA

1. Number of running circulating water pumps (36P-1A/B/C) 1
2. Number of running service water pumps (46P-1A/B/C) 2
3. Tank Discharge Flow Rate (maximum) TDFR 20 gpm
4. Tank Activity (ACT) 2.8×10^{-5} µCi/ml (from discharge permit)
5. Required Dilution Factor (DF) 100 (from discharge permit)
6. Liquid rad monitor (17RM-350) reading 70 cps (EPIC-A-1209)

NOTE 1: Items 7 and 8 are obtained at panel 09-14

NOTE 2: Background should be maintained **LESS THAN** 1000 cps. It is recommended that the detector canister be flushed to levels below this prior to discharge.

7. Liquid rad monitor (17RM-350) background 43 cps
8. Liquid rad monitor (17RM-350) K-factor 2.09×10^{-7} µCi/ml/cps
9. Tempering gate/flow 0 % (EPIC-A-3547)

CALCULATIONS

10. CFR = [(#1 X 120,000)+(#2 X 18,000)] x [(1 - #9/100)] = 156,000 gpm

11. Calculate Canal Dilution Factor (CDF):

$$CDF = \frac{TDFR}{CFR} = \frac{\#3}{\#10} = 1.28 \times 10^{-4}$$

NOTE: F_L= Fraction of allowed dilution (dimensionless, must be less than 1.0 for discharge).

12. Calculate F_L:

F_L = CDF X DF = #11 X #5 = .01

13. Calculate Background Correction Activity (BCA) in µCi/ml:

BCA = (#6 - #7) X #8 = 5.6×10^{-6} µCi/ml

COMPLETED FORMS ARE ATTACHED TO THE DISCHARGE PERMIT

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CANAL DISCHARGE WORKSHEET Page 2 of 2

14. Calculate Hi/Hi setpoint in µCi/ml:

$$\text{Hi/Hi} = \frac{\text{(ACT)}}{2 \times F_L} = \frac{\#4}{2 \times \#12} + \#13 = \underline{1.4 \times 10^{-3}}\ \mu\text{Ci/ml}$$

15. Calculate Hi setpoint in µCi/ml:

$$\text{Hi} = \frac{\text{(ACT)}}{4 \times F_L} = \frac{\#4}{4 \times \#12} + \#13 = \underline{7.1 \times 10^{-4}}\ \mu\text{Ci/ml}$$

16. Obtain 17RM-350 potentiometer setting for Hi-Hi setpoint from Chemistry

Hi/Hi __v 7.0__

17. Obtain 17RM-350 potentiometer setting for Hi setpoint from Chemistry.

Hi __v 6.7__

18. Enter potentiometer settings for Hi and Hi-Hi setpoints on Discharge Permit Section B and attach this worksheet to the discharge permit.

Performed by (SM) Bob Jones Bob Jones 6/18/06
Print/Sign/Date

Independent Verification ____________________
Print/Sign/Date

COMPLETED FORMS ARE ATTACHED TO THE DISCHARGE PERMIT

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QUESTION # 24

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
Knowledge of how event-based emergency/abnormal operating procedures are used in conjunction with the symptom based EOPs. **(10CFR 55.43.5)**	Group #		4
	K/A # **2.4**		**G 2.4.8**
	Importance Rating		**3.7**

Proposed Question: The plant is at 100% power with two Service Water pumps running and the third one in "Standby" when a large unisolable Service Water rupture in the reactor building occurs. The Crew enters AOP-10 "Loss of Service Water Cooling" and EOP-5 "Secondary Containment Control". These procedures direct the following actions:

- AOP-10 - Ensure standby service water pump(s) start, manually scram the reactor
- EOP-5 – isolate all systems that are discharging into the area, shutdown the reactor

The CRS must direct that the reactor be ____________ and that all service water pumps must be ____________.

a) shutdown normally, started

b) shutdown normally, tripped

c) scrammed, started

d) scrammed, tripped

Proposed Answer: d) scrammed, tripped

Explanation (Optional): EP-1 states that other procedures may be used with EOPs but shall not contradict nor subvert actions specified in the EOPs. If SW was allowed to continue its operation, it would subvert the intent of "isolating all systems that are discharging into the area".
Per the EOP bases, the requirement to shutdown does not preclude a scram. With a loss of all SW, a scram is required.

Distracters: a) shutdown normally, started
b) shutdown normally, tripped
c) scrammed, started

Justification:
Choices A and C allow SW to continue running. In EOP-5 a reactor shutdown is required when both crescent area water levels are 18" or greater. Per the EOP bases, "a direct threat exists relative to secondary containment integrity, to equipment located in the reactor building and to continued safe operation of the plant." This, along with the EOP requirement to isolate the leak, requires that SW be tripped.

If SW is completely lost, then ESW automatically aligns to supply some ventilation cooling loads and can be aligned to supply cooling to essential RX Bldg loads. However, it does not cool loads that are required for power production. If choice B is selected, and the SW pumps are tripped, a normal shutdown would be impossible. Additionally, AOP-10 requires a reactor scram for a complete loss of SW.

Technical Reference(s): AOP-10, EOP-5, EP-1, MIT-301.11F (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: SDLP-46A, EO- 1.14.a (As available)

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

Question History: Last NRC Exam X Modified from 2005 Vermont Yankees SRO exam (question # 99, attached)

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

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

10 CFR Part 55 Content: 55.41
55.43 5 Assessment of Facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations..

Comments:

QUESTION # 25

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
Ability to perform without reference to procedure those actions that require immediate operation of system components and controls. **(10CFR 55.43.2)**	Group #		4
	K/A # **2.4**		**G 2.4.49**
	Importance Rating		**4.0**

Proposed Question: The Plant is at 100% power.

The following event and indications occur subsequently:
- Feedwater Pump A trips off
- RWR Pump A trips off
- RWR Pump B - speed 30%
- Annunciator 09-5-2-44 APRM UPSCALE – ON
- All APRM recorders – Cycling 65% to 77% every 2 seconds
- All SRM Period meters - Cycling minus 80 to plus 30 seconds every 1 ½ seconds
- Various LPRM Upscale Alarms – Alarming and clearing every 2 seconds

Which one of the following is immediately required?

a) Trip RWR Pump B

b) Insert CRAM Groups

c) Manually Scram the Reactor

d) Raise RWR Pump B speed and flow

Proposed Answer: c) Manually Scram the Reactor.

Explanation (Optional): AOP-8 Loss or Reduction of Reactor Coolant Flow requires immediate action to manually SCRAM if indications of thermal hydraulic instabilities (THI) are observed. Refer to Attachment 1 of AOP-8 for Indications of THI. Distracters B and D are possible AOP-8 actions but not in this situation. Tripping of the B RWR pump would make THI worse (high power with low flow).

Distracters: a) Trip RWR Pump B.
b) Insert CRAM Groups
d) Raise RWR Pump B speed and flow

Justification: AOP-8 Loss or Reduction of Reactor Coolant Flow requires immediate action to manually SCRAM if indications of thermal hydraulic instabilities (THI) are observed. Refer to Attachment 1 of AOP-8 for Indications of THI. Distracters B and D are possible AOP-8 actions but not in this situation. Tripping of the B RWR pump would make THI worse (high power with low flow).

Technical Reference(s): AOP-8, TS- Bases 3.4.1, (Attach if not previously provided)

CR-JAF-2000-06312 (SER 7-00, BWR Core Power Oscillations) Reference Only

Proposed references to be provided to applicants during examination: NONE

QUESTION # 25 Continued

Learning Objective: LPAOP, EO- 1.03.a (As available)

QUESTION # 25 Continued

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

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

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5 Assessment of Facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

QUESTION # 24 Attachment

Vermont Yankees 2005 SRO Question # 99

Select the correct answer:

While operating at power, a service water rupture in the reactor building has occurred and it can not be isolated. During implementation of procedures, the following directions conflict:

- OP 2181 - secure all SW pumps
- ON 3148 - manually scram the reactor, reduce SW pumps operating to two
- EOP-4 - complete Reactor Shutdown per OP 0105
- ARS (6-A-5) SERV WTR HDR PRESS LO - start all SW pumps, perform Reactor Shutdown

What action must be implemented first? Why?

a) Implement OP 2181; preventing pump damage is critical
b) Implement ON 3148; reactor scram is required to reduce heat loads
c) Implement EOP-4; EOP actions override low tier procedures
d) Implement ARS (6-A-5); controlled restoration of SW and plant shutdown is required

Answer: B