QUESTION 1 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	1
	K/A #	295001 AK2.02		IR 3.2

Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION and the following: Nuclear boiler instrumentation.

Proposed Question:

The plant was operating at 100% power when a transient occurs resulting in the following conditions:

- RPV level 35 inches and stable
- Reactor Power is 73% and stable
- Total Core Flow is 51.5 x 10^6 lbm/hr and stable

The cause of this transient was the receipt of a signal from the ___.

A. ATWS/ARI logic.

B. EOC-RPT logic.

C. Recirc Flow Control Valve Runback logic

D. Recirc Pump Cavitation Interlock Circuitry

Proposed Answer: C

Explanation
A. The ATWS logic uses high RPV pressure and/or low RPV water level to trip the Recirc pumps to OFF. Core flow would indicate the Recirc pumps are still running.
B. This logic uses Turbine stop valve/first stage pressure to trip the Recirc pumps to SLOW, however, Slow speed Recirc Pumps would have altered Reactor power and core flow to less than given in stem.
C. Correct – The Reactor power and the Total Core Flow are consistent with a FCV Runback.
D. The cavitation interlocks will trip the Recirc pumps to slow speed; Reactor power and core flow are inconsistent with Recirc pumps in slow speed.

Technical Reference(s): R-STM-0053, Rev 13, p. 40 of 76
AOP-0024, Thermal Hydraulic Stability, Rev 25 Attachment 1

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0053, Obj 2

Question Source: Bank # October 2000 NRC exam #26

Question History: Last NRC Exam October 2000

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 2

10 CFR Part 55 Content: 55.41.b.7

Comments:

QUESTION 2 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑ SRO ☐
	Tier #	1	Group # 1
	K/A #	295004 AK1.04	IR 2.8

Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF DC POWER : Effect of battery discharge rate on capacity.

Proposed Question:

A 125 VDC bus has experienced a trip of its battery charger supply breaker. Efforts to restore the battery charger to service have been unsuccessful. Assuming battery loading remains un-changed; predict the change in bus voltage under these conditions.

Voltage will _____.

A. decrease sharply and then level off.

B. decrease in a linear fashion.

C. slowly lower and then drop sharply.

D. stair step down as individual cells are depleted.

Proposed Answer: C

Explanation
A. See "C"
B. See "C"
C. Correct – Battery voltage will decrease slowly then drop sharply as battery voltage reversal occurs as seen in SER 3-99.
D. See "C"

Technical Reference(s): RPPT-STM-0305-ILO Rev 2, Slide 95; SER 3-99

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0305, Obj 8

Question Source: Bank # March 2014 NRC exam #48

Question History: Last NRC Exam March 2014 Exam #48

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3

10 CFR Part 55 Content: 55.41.b.8

Comments: Appeared on one of last 2 NRC exams (1 of 2)

QUESTION 3 Rev 1

Examination Outline Cross-Reference: Level RO ☑ SRO ☐
Tier # 1 Group # 1
K/A # 295005 AK2.07 IR 3.6

Knowledge of the interrelations between MAIN TURBINE GENERATOR TRIP and the following: Reactor pressure control

Proposed Question:

The plant is performing a Startup, operating at 30% power when a break occurs in the EHC Hydraulic Supply line to the #2 Main Stop Valve Actuator. The Standby EHC pump starts on low pressure and the main turbine trips.

In this transient, with no operator action, the Turbine Bypass Valves will _(1)_ and the Reactor will scram due to _(2)_.

A. (1) open ; (2) high pressure.

B. (1) fail open ; (2) MSIV closure after a low steam line pressure.

C. (1) fail closed ; (2) high pressure.

D. (1) open ; (2) turbine stop valve fast closure.

Proposed Answer: A

Explanation
A. Correct - BPVs remain available for automatic pressure control due to an independent hydraulic system; due to the turbine trip, pressure will rise and the BPVs will try to control pressure by opening. If power were above 30.92%, then the Control Valve Fast closure scram would be enabled to send a trip signal to RPS, but this feature is not enabled so pressure will rise until the high pressure scram setpoint.
B. BPVs remain available for automatic pressure control due to an independent hydraulic system.
C. BPVs remain available for automatic pressure control due to an independent hydraulic system.
D. See A.

Technical Reference(s): R-STM-0509, pp. 15, 54 of 81

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0509, Obj 10d

Question Source: Bank # RBS-OPS-3478

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3

10 CFR Part 55 Content: 55.41.b.7

Comments: Replaced question to better match the KA

QUESTION 4 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	1
	K/A #	295006 AK3.03		IR 3.8

Knowledge of the operational implications of the following concepts as they apply to SCRAM: Reactor pressure response

Proposed Question:

A reactor plant has been operating at 100% power for an extended period when a scram occurs. Operators are executing AOP-0001, Reactor Scram. For a period of several minutes after the scram, there will be a __(1)__ due to __(2)__.

A. (1) drop in RPV level; (2) Recirc pumps shifting to slow speed

B. (1) rise in RPV level; (2) rise in void content

C. (1) rise in reactor pressure; (2) decay heat

D. (1) drop in reactor pressure; (2) automatic operation of SRV's

Proposed Answer: C

Explanation
A. Part 1 is correct, but it is due to an automatic feature of the feed water level control system; not because of Recirc pumps shifting.
B. Part 1 is seen in plant operation due to the setpoint set down feature of the Feedwater level control system; it is not because of void content.
C. Correct – reactor pressure does go up and must be controlled with one of the following: the turbine, turbine bypass valves, or safety relief valves (IAW AOP-1); the reason pressure rises for several minutes after a scram is because of decay heat: 7% for first 8-10 seconds, then drops to about 1% after 1 hour.
D. Both parts are incorrect: for part 1 see "C"

Technical Reference(s): GFES - Reactor Operational Physics, pp. 63-64 of 83; AOP-0001

Proposed references to be provided to applicants during examination: None

Learning Objective: GFES Reactor Operational Physics, Obj 31

Question Source: New

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3

10 CFR Part 55 Content: 55.41.b.1

Comments:

QUESTION 5 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 1	Group # 1	
	K/A # 295016 AA1.06	IR 4.0	

Knowledge of the operational implications of the following concepts as they apply to CONTROL ROOM ABANDONMENT: Reactor water level

Proposed Question:

The Main Control Room must be abandoned due to Halon actuation; there is **not** a fire in the MCR. In accordance with AOP-0031, Shutdown from Outside the Main Control Room, what systems will be initiated for RPV level control?

A. LPCS, HPCS, and RCIC

B. HPCS, and RCIC only

C. RCIC only

D. All Division 1 ECCS systems

Proposed Answer: A

Explanation

A. Correct – Subsequent actions for abandonment of the control room without a fire direct initiation of HPCS, LPCS, and RCIC.
B. See "A" this answer does not include LPCS
C. See "A" this answer does not include LPCS or RCIC
D. See "A" this answer does not include HPCS, and should not include RHR-A

Technical Reference(s): AOP-0031, Rev 322 p.10 of 122

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-OPS-AOP-0031, Obj 4

Question Source: New

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 6 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	1
	K/A #	295018 AA2.04	IR 2.9	

Ability to determine and/or interpret system flow as it applies to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER.

Proposed Question:

With the plant operating at 100%, the following annunciators are received:

- TURB CMPNT CLG WATER SYSTEM LOW PRESSURE
- TURB CMPNT CLG WTR SYS SURGE TK LOW LEVEL

On H13-P870, the unit operator observes:

- All 3 CCS pumps are running with elevated amps
- MWS-AOV132 TPCCW SURGE TK MAKE-UP VALVE is OPEN

Which of the following is the cause of the above?

A. CCS piping failure has caused a loss of inventory.

B. GSN-PCV1C, MANIFOLD REGULATOR (Nitrogen Pressure Control Valve to TPCCW Surge Tank) has failed closed.

C. CCS-LT113, SURGE TANK LEVEL TRANSMITTER has failed low.

D. CCS-PV111, MINIMUM FLOW AND PRESSURE CONTROL VALVE, has failed OPEN.

Proposed Answer: A

Explanation

A. Correct - A large leak would cause all the indications, low pressure - starts stby pump, max amps as the pumps operate in runout condition, and loss of inventory causes the surge tank low level condition **and** opening of the makeup valve.

B. The TPCCW surge tank is blanketed with 18 psig nitrogen for NPSH and for corrosion control; if this valve fails closed it would lower system pressure, but not surge tank level.

C. The surge tank level transmitter failing low would cause the surge tank low level alarm and the opening of the surge tank makeup valve, but would not cause all pumps to run with maximum amps, nor would it cause a system low pressure alarm.

D. This would lower system pressure, but would not account for a low Surge Tank level.

Technical Reference(s): ARP-P870-55-B01, C02, PID 09-07A

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-116 Obj 2, 5

Question Source: Bank # RBS Audit Exam, March 2010 #6

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3

10 CFR Part 55 Content: 55.41.b.4 Comments:

QUESTION 7 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑ SRO ☐
	Tier #	1	Group # 1
	K/A #	295019 G2.4.47	IR 4.2

Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

Proposed Question:

The plant is operating at 100% power when the following alarms annunciate:

P870-51A-A01, INSTRUMENT AIR COMPRESSOR TROUBLE

P870-51A-B01, SAS COMPRESSOR ALIGNED TO IA SYSTEM

The Unit Operator notices the following:

- Instrument Air Header Pressure on IAS-PI105 is trending downward at a rate of 1 psig every 10 seconds.
- A and B IAS compressors amber lights lit; C IAS compressor red light is lit

Assuming the situation continues to degrade, which of the following represents the correct sequence of events?

A. Feedwater Reg Valves Lock-up; then
SAS-AOV133, Service Air Header Block Valve, closes; then
MSIV's fail shut

B. Instrument Air to the Lower Fuel Pools Gate Seals is replaced by the passive nitrogen back-up system; then
Feedwater Reg Valves Lock-up; then
Condensate and Heater Drain Pump Recirc valves fail open

C. P870-51A-B02, "Instrument Air Header Pressure Low" Annunciator Alarms; then
SAS-AOV133, Service Air Header Block Valve, closes; then
Feedwater Reg Valves Lock-up

D. Feedwater Reg Valves Lock-up; then
P870-51A-B02, "Instrument Air Header Pressure Low" Annunciator Alarms; then
MSIV's fail shut

Proposed Answer: C

Explanation

A. (also B. and D.) See "C"

C. Correct – the cross tie valve opens at 113 psig; the low air header pressure alarm and the service air block valve closing both occur at 110 psig ; FRVs lock-up at 85 psig; MSIV's go closed at 65 psig; Fuel Pool Gates switch to nitrogen at 32 psig.

Technical Reference(s): AOP-0008, Loss of Instrument Air, Rev 37, p. 4 of 21

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-0527, Obj 4

Question Source: Modified Bank # RBS December 2008 NRC Exam #8

Question History: Last NRC Exam December 2008 NRC Exam #8

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3

10 CFR Part 55 Content: 55.41.b.7 Comments:

QUESTION 8 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	1
	K/A #	295021 AK1.03		IR 3.9

Knowledge of the operational implications of the following concepts as they apply to LOSS OF SHUTDOWN COOLING : Adequate core cooling

Proposed Question:

The plant has been in MODE 4 for 25 hours, cooling down for a refueling outage with the following conditions:

- RHR-B is in Shutdown Cooling mode of operation
- Recirc Pump A is tagged out
- Recirc Pump B is running in slow speed
- Reactor Coolant Temperature is 190˚F
- RPV level +38" and steady

RHR Pump B trips on motor overload and RHR Pump A will not start.

Which of the following operator actions would assure adequate core cooling?

A. Align RWCU for Alternate Shutdown Cooling.

B. Align for Main Steam Line Flooding

C. Align SPC/ADHR in Configuration 1

D. Raise RPV level to greater than 75 inches

Proposed Answer: C

Explanation
A. RWCU is only aligned for ADHR in mode 5 with coolant temperature below 125˚F
B. MSL Flooding can only be used between 120 and 170˚F
C. Correct – SPC/ADHR is available when reactor coolant temperature is below 200˚F
(Configuration 1 takes a suction from SDC and discharges through LPCI-C injection line)
D. Raising RPV level to +75 ensures natural circulation to prevent stratification and inadvertent pressurization, but does not assure adequate core cooling

Technical Reference(s): OSP-0041, Alternate Decay Heat Removal, Rev 306 pp 8, 18, 40
AOP-0051, Loss of Decay Heat Removal, Rev 312 pp 5&6 of 30

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-543, Obj 5

Question Source: Bank 2010 Grand Gulf NRC exam Q#40

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 9 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	1
	K/A #	295023 AK2.05		IR 3.5

Knowledge of the interrelations between REFUELING ACCIDENTS and the following: Secondary containment ventilation

Proposed Question:

Refueling operations are in progress when an irradiated fuel bundle is dropped in the spent fuel pool. The refueling team on the lower bridge reports bubbles rising from the dropped fuel bundle and the following annunciators are received in the MCR:

- H13-P863-75A-H01, DIV I Fuel Bldg Exh PAM Gaseous Radn Alarm
- RMS-DSPL230-1GE005, Fuel Build Stack/Vent Exhaust A – High
- RMS-DSPL230-2GE005, Fuel Build Stack/Vent Exhaust A – High

Which of the following describes the Fuel Building ventilation lineup after the conditions given above?

A. Fuel Building Ventilation is completely isolated

B. Supply air is via normal supply fans and exhaust is through the Div 1 charcoal filter trains

C. Supply air is via Fuel Receiving Area and exhaust is through both Div 1 and Div 2 charcoal filter trains

D. Supply air is via Fuel Receiving Area and exhaust is through the Div 1 charcoal filter trains

Proposed Answer: C

Explanation
A. A high radiation condition in the fuel building does not isolate the ventilation system; it isolates the normal supply and exhaust fans, and starts the charcoal filtration system.
B. A high radiation condition in the fuel building isolates the normal supply and exhaust fans, and starts the charcoal filtration system.
C. Correct - RMS-RE5A is the instrument that drives all three annunciators; this instrument will start the Div 1 filter train only, however a low flow signal will also start the Div 2 train.
D. This instrument will send a start signal to the Div 1 filter train only, but both will start (see C)

Technical Reference(s): R-STM-0406, pp 19 & 41 of 50, ARP-H13-P863-75A-H01
Proposed references to be provided to applicants during examination: None
Learning Objective: RLP-STM-0406, Obj 11
Question Source: New Question History: Last NRC Exam NA
Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3
10 CFR Part 55 Content: 55.41.b.7
Comments:

QUESTION 10 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	1
	K/A #	295024 EK3.06		IR 4.0

Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE: Reactor Scram

Proposed Question:

What is the reason for the reactor scram that occurs due to a High Drywell Pressure?

A. To minimize the possibility of fuel damage due to a reactor coolant pressure boundary leak by reducing the amount of energy being added to the coolant.

B. To ensure the Pressure Suppression function of the containment is maintained in the event Emergency Depressurization is required.

C. To ensure that offsite dose limits are not exceeded during a reactor coolant pressure boundary leak.

D. To avoid clearing of the suppression pool vents due to high drywell pressure.

Proposed Answer: A

Explanation
A high drywell pressure condition results due to a leak of the primary system. Due to the loss of coolant, an inability to cool the fuel may result. A reactor scram occurs to minimize the energy being produced. The pressure suppression function of the containment is based on containment pressure not drywell pressure. Offsite dose limits are prevented from being exceeded by the high drywell pressure containment isolation, not the high drywell pressure reactor scram. Although the scram signal will reduce the energy being leaked into the drywell, and may avoid clearing of the suppression pool vents, this is not the reason for the scram.

Technical Reference(s): R-STM-0508, RPS, Rev 6 p. 46 of 59

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0508 Obj 2

Question Source: Bank # 2008 NRC Exam Q#11

Question History: Last NRC Exam RBS 2008

Cognitive Level: Memory or Fundamental Knowledge ☑ 3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.6

Comments: Rejected KA: EK3.04 ; Randomly selected new KA statement EK3.06
(RBS has no guidance to Emergency Depressurize for a High DW D/P

QUESTION 11 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	1
	K/A #	295025 EA1.05		IR 3.7

Ability to operate and/or monitor the following as they apply to HIGH REACTOR PRESSURE: RCIC: Plant-Specific

Proposed Question:

The plant is operating at 93% power.
A RCIC lube oil change has just been completed and RCIC has been started with the "slow-roll startup" section of SOP-0035 in the RPV Pressure Control lineup (CST to CST). All MSIV's go closed and the reactor scrams.

What is the status of RCIC after the scram?

A. RCIC will isolate on steam supply line high flow.

B. RCIC will re-align to inject to the RPV.

C. RCIC will isolate on high steam line pressure.

D. RCIC will remain in a CST to CST lineup.

Proposed Answer: D

Explanation
A. High steam line flow can be a cause for RCIC isolation, but MSIV isolation does not cause this.
B. RCIC initiation occurs at RPV Level 2 (-43")
C. Low Steam Line Pressure, not high, can cause a RCIC isolation
D. Correct – There is no logic setpoints reached to cause a realignment of RCIC

Technical Reference(s): R-STM-0209, Rev 10, pp. 26,27,43 of 52

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0209, Obj 7, 12

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3

10 CFR Part 55 Content: 55.41.b.7

Comments:

QUESTION 12 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑ SRO ☐
	Tier #	1	Group # 1
	K/A #	295026 EA2.03	IR 3.9

Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Reactor pressure

Proposed Question:

Following a loss of offsite power, the plant is experiencing an ATWS. Suppression Pool level is currently 19 feet, 11 inches.

In accordance with the Pressure leg of EOP-1A, RPV Control, ATWS, which of the following pressure band/suppression pool temperature combinations would result in the need for emergency depressurization?

	RPV Pressure	Supp Pool Temperature
A.	800 - 1090 psig	131˚F
B.	800 - 1090 psig	137˚F
C.	500 - 700 psig	139˚F
D.	500 – 700 psig	142˚F

Proposed Answer: B

Explanation
B. Correct – The pressure leg of EOP-1A has an override step in RPA-4 that directs maintaining RPV pressure below the HCTL: Suppression pool level, stated as 19'11" in the stem, requires the examinee to use the 19'6" line on the HCTL curve (not allowed to interpolate). The pressure bands are common given bands for the given plant condition and the top of the band must be used to determine if the HCTL will be reached. The lowest s.p. temperature that would require ED for 700 psig is 145˚F; the lowest temperature for 1090 psig is 133˚F.

Technical Reference(s): EOPs, Heat Capacity Temperature Limit Curve

Proposed references to be provided to applicants during exam: Heat Capacity Temperature Limit Curve

Learning Objective: RLP-HLO-517, Obj 2

Question Source: New

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 2

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 13 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	1
	K/A #	295027 G2.4.20		IR 3.8

Knowledge of the operational implications of EOP warnings, cautions, and notes

Proposed Question:

A suppression pool temperature of 180˚F may result in ___.

A. direct pressurization of containment from RCIC turbine exhaust during operation.

B. damage to the RCIC turbine during operation due to reduced lube oil cooling.

C. inability to monitor suppression pool temperature due to instrument being off scale high.

D. inability to monitor suppression pool level due to temperature being greater than instrumentation environmental ratings.

Proposed Answer: B

Explanation

A. Direct pressurization is a concern with a high suppression pool level, not temperature (EOP Caution 4)

B. Correct- the lube oil and control oil for RCIC is cooled by process flow; the max allowable cooling water temperature for RCIC lube oil is 140˚F (EOP Caution 3)

C. Design temperature of containment is 185˚F, some of the RTD's are designed for Post Accident Monitoring and are tested to readings of 200˚F. (STP-555-4203)

D. See explanation in C; EOP Caution 7 warns of the inability to trust suppression pool temperatures with suppression pool water level too low.

Technical Reference(s): EOP-1 Caution 3, Bases p. B-5-10

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-OPS-HLO-511, Obj 6

Question Source: New

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 4 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.9

Comments:

QUESTION 14 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	1
	K/A #	295028 EK1.01		IR 3.5

Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE: Reactor water level measurement

Proposed Question:

EOP-1 Caution 1, part 2 identifies RPV levels above which RPV level instruments may be used when the containment or drywell temperature near the reference legs is at the specified limits. At these elevated RUN TEMPERATURES, the instrument would ___.

A. continue to indicate level on-scale when actual RPV level went off-scale high (above the indicating range).

B. fail off-scale low.

C. continue to indicate level on-scale when actual RPV level went below the variable leg tap.

D. provide erratic level indication when actual RPV level went off-scale low due to loss of the variable leg.

Proposed Answer: C

Explanation
A. Caution 1 deals with low RPV level, not high
B. see C.
C. Correct- indicated level would be on-scale while actual level would be below the variable leg tap.
D. Erratic indication would be a sign of boiling in the reference leg.

Technical Reference(s): EOP Caution 1, Bases p. B-5-2

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-OPS-HLO-511, Obj 6

Question Source: Bank # RBS-NRC-706

Question History: Last NRC Exam Audit Exam March 2008

Cognitive Level: Memory or Fundamental Knowledge ☑ 2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.2

Comments:

QUESTION 15 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 1	Group # 1	
	K/A # 295030 EK2.08	IR 3.5	

Knowledge of the interrelations between LOW SUPPRESSION POOL WATER LEVEL and the following: SRV discharge submergence

Proposed Question:

For normal SRV operations, which of the following suppression pool levels is based on direct pressurization of containment air space?

A. 13 feet

B. 15 feet, 5 inches

C. 21 feet, 3 inches

D. 21 feet, 6 inches

Proposed Answer: A

Explanation
A. Correct – This is the elevation of the top of the SRV discharge device below which opening of an SRV may cause pressurization of the containment air space
B. 15'5" is 2 feet above the horizontal vents; this level is associated with a leak from the DW passing through the horizontal vents.
C. 21'3" is the SRV tail pipe level limit; operation of an SRV above this limit may cause damage to the SRV discharge lines.
D. 21'6" is based on the Pressure Suppression Pressure (PSP) Curve; the PSP pressure is a function of suppression pool level.

Technical Reference(s): EOP Bases p. B-6-56

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-514, Obj 5

Question Source: New

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.9

Comments:

QUESTION 16 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	1
	K/A #	295031 EK2.03		IR 4.2

Knowledge of the interrelations between REACTOR LOW WATER LEVEL and the following: Low pressure core spray

Proposed Question:

An STP is being performed that directs the operator to close E21-MOV F011, LPCS Min Flow Valve; all other valves are in their normal, standby positions. A valid RPV low level, Low Pressure Core Spray (LPCS) initiation signal is received.

Assuming no operator action, how will the LPCS system respond?

A. The LPCS Min Flow Valve will remain closed until the initiation signal is reset.

B. The LPCS Min Flow Valve will open until system flow rises above 875 gpm.

C. The LPCS Pump will not receive a start signal because of the pump protection logic permissive not being met.

D. The LPCS Pump will start and will eventually overheat.

Proposed Answer: B

Explanation
A. The reset pushbutton is in the LPCS initiation sequence logic; the min flow valve is not part of this logic.
B. Correct – the design logic for the min flow valve is to OPEN when the pump breaker is closed and the sensed flow is less than 875 gpm.
C. The pump start logic does not include a permissive from the min flow valve
D. The pump would overheat without a flow path providing at least 500 gpm; the min flow valve will open with the given conditions.

Technical Reference(s): R-STM-0205 pp.13-14,16 of 33 ; STP-205-4201, Rev 301

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0205, Obj 4,9

Question Source: Modified Bank # RBS-OPS-5554

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 2

10 CFR Part 55 Content: 55.41.b.7

Comments:

QUESTION 17 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	1
	K/A #	295037 EA1.06	IR 4.1	

Ability to operate and/or monitor the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Neutron monitoring system

Proposed Question:

The plant was operating at 80% power when a transient in the Fancy Point Switchyard resulted in a Main Turbine trip. All control rods did not fully insert.

Current parameters:

- Reactor Power 22.5%
- Reactor Pressure 900 psig
- RPV Level -50 inches (Wide Range)
- Feed Flow 2.79 Mlbm/hr
- IRM and SRM detectors have been inserted
- Main Steam Bypass Valves are both full open
- Two SRVs are open

To avoid exceeding the Heat Capacity Temperature Limit Curve, the CRS has directed pressure lowered to 700 psig using SRVs.

Immediately following the opening of SRVs, indicated reactor power will ___.

A. rise due to the lowering of the reactor coolant temperature adding positive reactivity.

B. rise due to the water level inside the core rising, causing more neutron moderation.

C. lower due to voiding as the water in the core flashes to steam.

D. lower due to the moderator temperature rising with the low flow in the core.

Proposed Answer: C

Explanation

A. reactor power would lower; later as pressure stabilizes, and the water that flashed to steam is replaced with Feedwater, coolant temperature will add positive reactivity, but not immediately.

B. reactor power would lower; voiding causes less moderation

C. Correct – lowering pressure will flash the water in the core and therefore causing void volume to rise. Neutron moderation would decrease, thereby lowering power.

D. reactor power will lower, but it is due to voiding, not a temperature change

Technical Reference(s): GFES Components BC070

Proposed references to be provided to applicants during examination: None

Learning Objective: BC07022 Question Source: Bank # RBS-NRC-569

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3

10 CFR Part 55 Content: 55.41.b.1

Comments:

QUESTION 18 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 1	Group # 1	
	K/A # 295038 EA2.03	IR 3.5	

Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: Radiation levels

Proposed Question:

The radioactivity release control leg of EOP-0003, Secondary Containment and Radioactive Release Control is entered when radiation monitors reach the rate levels corresponding to the ___ action level defined in the Site Emergency Plan.

A. NOUE

B. ALERT

C. SITE AREA EMERGENCY

D. GENERAL EMERGENCY

Proposed Answer: B

Explanation
A. NOUE is below the entry level for EOP-0003
B. Correct- Entry into EOP-0003 radioactive release leg corresponds to the ALERT action level; radiation monitors' setpoints are set accordingly.
C. See B
D. See B

Technical Reference(s): EOP-0003

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-515, Obj 2

Question Source: New

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 19 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	1
	K/A #	600000 G2.4.11		IR 4.0

Knowledge of abnormal condition procedures

Proposed Question:

The plant is operating at 100% power. Standby Gas Treatment Filter Train "A" is being operated for surveillance testing.

The Auxiliary Building Operator reports smoke coming from the "A" Standby Gas Treatment Filter Train and the filter train case is glowing red.

Which one of the following describes the method to combat a fire in the Standby Gas Treatment Filter Train?

A. The Fire Protection System will initiate the automatic deluge system and fill the filter train with water.

B. The Fire Protection System will automatically open a deluge isolation valve, however, valves must be manually opened to admit water to the filter train.

C. The Fire Protection system Deluge Valve will have to be manually initiated via the pull station to admit water to the filter train.

D. The Fire Protection System at the filter train must be manually valved in to admit water to the filter train.

Proposed Answer: D

Explanation
A. There is no automatic fire suppression for the GTS train
B. The isolation valves must be manually opened, however there is no auto fire suppression for GTS
C. Some deluge systems on site have this capability but not for the GTS train
D. The isolation valves must be manually opened, no deluge valve actuation is needed.

Technical Reference(s): Pre-Fire Strategy AB-141-531 SGTS Filter A Room Fire Area AB-14

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0257, Obj 4

Question Source: Bank # RBS-NRC-654

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 20 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 1	Group # 1	
	K/A # 700000 AK1.03	IR 3.3	

Knowledge of the operational implications of the following concepts as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Under-excitation

Proposed Question:

Under-excitation of the Main Generator results in ___.

A. Turbine rotor overheating.

B. Generator field overheating.

C. the Voltage Regulator shifting to Manual.

D. Generator armature overheating.

Proposed Answer: D

Explanation
A. A loss of generator field causes "turbine torque oscillation" which leads to the Turbine rotor overheating.
B. Over-excitation causes the Generator field to overheat.
C. An exciter field overcurrent causes the Voltage Regulator to shift to Manual.
D. Correct – Under-excitation causes overheating of the armature.

Technical Reference(s): R-STM-0310, Rev 8 p. 29 of 76

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0310, Obj 3,10

Question Source: New

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41b.5

Comments:

QUESTION 21 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	2
	K/A #	295007 AK2.01	IR 3.5	

Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following: Reactor/turbine pressure regulating system

Proposed Question:

The plant is starting up and is currently operating at 60% power when the A Recirc. Flow Control valve ramps open.

How will the EHC pressure control system respond to this condition?

The EHC pressure control system will cause generator load to _(1)_ and reactor pressure will _(2)_

A. (1) rise (2) lower

B. (1) rise (2) remain steady

C. (1) lower (2) lower

D. (1) lower (2) remain steady

Proposed Answer: B

Explanation
As the FCV opens at this power level reactor power will rise and reactor pressure will follow. The pressure regulating system will open the turbine control valves causing generator load to rise. Reactor pressure will remain the same as the turbine control valves open to control reactor pressure. B is the correct answer.

Technical Reference(s): R-STM-0509, Rev 14 p.51 of 81
Proposed references to be provided to applicants during examination: None
Learning Objective: RLP-STM-0509, Obj 10
Question Source: New
Question History: Last NRC Exam NA
Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 2
10 CFR Part 55 Content: 55.41.b.5
Comments:

QUESTION 22 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	2
	K/A #	295008 AA2.05		IR 2.9

Ability to determine and/or interpret the following as they apply to HIGH REACTOR WATER LEVEL: Swell

Proposed Question:

River Bend was operating at full power with RCIC tagged out. A loss of offsite power occurred and HPCS recovered RPV level. Operators have closed the HPCS injection valve with an RPV water level of +20 inches. Current conditions are as follows:

- MSIV's are closed
- Reactor pressure is 700 psig and rising at 10psig per minute
- Time after scram is 5 minutes
- Drywell pressure is 0.3 psid

What is the expected RPV water level response over the next 10 minutes and why?

A. RPV level will rise due to swell from decay heat

B. RPV level will rise due to feed regulating valve leak by

C. RPV level will lower due to cool down

D. RPV level will lower due to injection being secured

Proposed Answer: A

Explanation
A. Correct- Level rises due to expansion cause by the heat up from decay heat
B. Feedwater is not available due to the loss of offsite power
C. RPV water level will rise not lower
D. RPV water level will rise not lower

Technical Reference(s): RLP-HLO-168, Rev 06, Thermodynamics: Steam, pp. 35-37 of 73

Proposed references to be provided to applicants during examination: None

Learning Objective: HLO-167, Obj 2

Question Source: Modified Bank # RBS-NRC-01108

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3

10 CFR Part 55 Content: 55.41.b.14

Comments:

QUESTION 23 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑ SRO ☐
	Tier #	1	Group # 2
	K/A #	295011	G2.4.04 IR 4.5

Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures: High Containment Temperature

Proposed Question:

Based on the following conditions, which one of the following describes the Emergency Operating Procedure(s) that should be entered?

- Reactor Power 0% (all rods in)
- RPV Pressure 970 psig
- RPV Level 20 inches
- Containment Temperature 92˚F
- Containment Pressure 0.25 psig
- Drywell Temperature 140˚F
- Drywell Pressure 0.75 psid
- Annulus Differ. Pressure -1.3 in WC

A. EOP-1, RPV Control ONLY

B. EOP-1, RPV Control AND EOP-2, Primary Containment Control

C. EOP-2, Primary Containment Control ONLY

D. EOP-2, Primary Containment Control AND EOP-3, Secondary Containment and Radioactivity Release

Proposed Answer: C

Explanation
A. No EOP-1 Entry Condition exists.
B. No EOP-1 Entry Condition exists
C. Correct- EOP-0002 Entry Condition is Containment Temperature above 90˚F
D. No EOP-3 Entry Condition exists.

Technical Reference(s): EOP-1, -2, and -3

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-514, Obj 3

Question Source: Modified Bank # RBS-NRC-444

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments: KA rejected because there is no Alarm or ARP for Cont. High Temperature.
Redrew from the 50 possible choices in the G2.4 category to get 2.4.4

QUESTION 24 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	2
	K/A #	295012 AK1.02		IR 3.1

Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE : Reactor power level control

Proposed Question:

Prior to Emergency Depressurization on High Drywell Temperature, EOP-1 is directed to be entered to assure, if possible, ___.

A. that RPV pressure is lowered as low as possible by steam loads before RPV depressurization is initiated.

B. that the reactor is scrammed and shutdown by control rod insertion before RPV depressurization is initiated

C. that RPV water level is raised as high as possible by Feed water before RPV depressurization is initiated

D. that the suppression pool is cooled and lowered by RHR before RPV depressurization is initiated

Proposed Answer: B

Explanation
A. While a lower pressure is desirable it is not specifically directed by the EOP prior to ED
B. Correct- Per EOP Bases the reactor is scrammed and shutdown if possible to minimize the amount of energy sent to containment
C. While a higher level is desirable to minimize inventory loss it is not specifically directed by the EOP prior to ED
D. While a cooler suppression pool is desired it is not directed by EOP-0001

Technical Reference(s): EOP BASES p. B-8-6

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-514, Obj 5

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.9

Comments:

QUESTION 25 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 1	Group # 2	
	K/A # 295013 AK2.01	IR 3.6	

Knowledge of the interrelations between HIGH SUPPRESSION POOL TEMPERATURE and the following: Suppression pool cooling

Proposed Question:

A safety relief valve has opened and AOP-0035, SAFETY RELIEF VALVE STUCK OPEN has been entered. Following reduction in reactor power to 89%, the CRS has directed you to place RHR A in suppression pool cooling mode.

Why is suppression pool cooling initiated?

A. to obtain localized suppression pool temperature

B. to minimize containment atmospheric activity

C. to reject suppression pool level

D. to establish bulk mixing of the suppression pool

Proposed Answer: D

Explanation
A. T.S. and TRM actions are based on average temperature not localized temperature
B. SP cooling does not significantly affect containment activity levels and the reason for establishing SP cooling is to establish bulk mixing and to cool the pool
C. While reject may become required, the reason for establishing SP cooling is to establish bulk mixing and to cool the pool
D. Correct- RHR A is placed in suppression pool cooling mode to establish bulk mixing of the suppression pool such that an average suppression pool temperature may be obtained

Technical Reference(s): AOP-0035, Safety Relief Valve Stuck Open, Rev 19 p. 5 of 10

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-OPS-AOP035, Obj 4

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 26 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	2
	K/A #	295029 EK3.03		IR 3.04

Knowledge of the reasons for the following responses as they apply to HIGH SUPPRESSION POOL WATER LEVEL : Reactor SCRAM

Proposed Question:

EOP-2, Suppression Pool Level Control, requires entry into EOP-1, RPV Control if suppression pool level cannot be maintained below 21 feet.
Entering EOP-1 ___.

A. will allow actions to take place that support terminating injection of water from sources external to containment.

B. will force a scram and shutdown of the reactor; this will make the requirement of maintaining Suppression Pool Level below 21 feet no longer applicable.

C. will allow for a reactor shutdown prior to flooding the drywell.

D. is required so the required emergency depressurization can be accomplished.

Proposed Answer: A

Explanation
A. Correct- EOP bases states that prior entry to EOP-1 requires control of RPV level and reactor power and therefore, facilitates making the determination if a system taking suction from a source external to the containment is needed for RPV injection or to shutdown the reactor
B. EOP-2 is still applicable when the reactor is shutdown
C. EOP-1 entry on SP level is based on SRV component loading and injection source termination.
D. EOP-1 entry on SP level is based on SRV component loading and injection source termination

Technical Reference(s): EOP Bases p. B-8-21

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-514, Obj 5

Question Source: Bank # RBS-NRC-20

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 27 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	2
	K/A #	295032 EA1.01		IR 3.6

Ability to operate and/or monitor the following as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE : Area temperature monitoring system

Proposed Question:

The plant is at 100% power.

Alarm P601/21A/H02, AIR TEMP MON R608 RCIC RM TEMP HIGH is received. The CRS has directed you to monitor and trend the value for the high area temperature.

What temperature monitoring equipment will be used to obtain this information?

A. ERIS computer data via the Plant Data Server System

B. H13-P632 Vent Diff Temperature recorder

C. H13-P632 Area Temperature recorder

D. Local temperature monitoring by a building operator

Proposed Answer: C

Explanation
A. The ERIS computer does not monitor this area of secondary containment
B. This recorder only supplies unit cooler differential temperature indication
C. The alarm response procedure directs the use of H13-P632 Area Temperature recorder
D. While this is not disallowed the alarm setpoint for this alarm is 144ºF and is a high dose area. This would be a safety and rad concern to enter the area to obtain temperatures.

Technical Reference(s): P601-21A-H02 PDS data point list

Proposed references to be provided to applicants during examination: None

Learning Objective: None Identified

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 28 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	203000 K4.03		IR 3.2

Knowledge of RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) design feature(s) and/or interlocks which provide for the following: Pump minimum flow protection

Proposed Question:

The reactor has been shut down due to a transient.

Current plant conditions are:

- Reactor pressure is 528 psig and slowly decreasing
- Reactor water level is -120 inches and slowly decreasing
- Drywell pressure is 1.78 psid and slowly increasing
- All low pressure ECCS systems are running as designed

What is the status of the RHR C system?

A. Injection valve (F042C) is open and pump minimum flow valve F064C is closed.

B. Injection valve (F042C) is open and pump minimum flow valve F064C is open.

C. Injection valve (F042C) is closed and pump minimum flow valve F064C is closed.

D. Injection valve (F042C) is closed and pump minimum flow valve F064C is open.

Proposed Answer: D

Explanation
A. and B are not correct, the injection valve, F064C will not open until reactor pressure is below 487 psig.
C. With the injection valve closed, system flow will be 0 gpm, the min flow valve is a normally open valve and is interlocked to open when flow is <1100 gpm.
D. Correct, see C

Technical Reference(s): R-STM-0204, RHR, Rev 11 pp. 12,14,15 of 63

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0204, Obj 6

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3

10 CFR Part 55 Content: 55.41.b.7

Comments: Rejected KA K4.09 ; for inability to write psychometrically sound question; Randomly selected K4.03
Wording of KA suggests How do design features/interlocks provide for STPs

QUESTION 29 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	205000 K5.02		IR 2.8

Knowledge of the operational implications of the following concepts as they apply to SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) : Valve operation

Proposed Question:

The plant is shut down. RHR A is operating in Shutdown Cooling mode (SDC) with an RPV water level of 75 inches.
Operation from the main control room of which of the following valves would lead to an OPDRV (Operations with a Potential to Drain the Reactor Vessel)?

A. E12-F004A, RHR PUMP A SUP PL SUCTION VALVE

B. E12-F006A, RHR PUMP A SDC SUCTION VALVE

C. E12-F024A, RHR PUMP A TEST RTN TO SUP PL

D. E12-F042A, RHR PUMP A LPCI INJECTION ISOL VALVE

Proposed Answer: C

Explanation
A. Interlocks prevent opening of E12-F004A with E12-F006A already open
B. closure of E12-F006A would cause to the RHR pump to trip but would not cause an OPDRV
C. Correct- There are no interlocks preventing operation and opening of E12-F024 would direct RPV water to the suppression pool
D. Open of E12-F042A would redirect water through the LPCI injection line but would not cause an OPDRV

Technical Reference(s): R-STM-0204, RHR, Rev 11 pp. 16,21 of 63

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0204, Obj 6

Question Source: Bank # RBS-NRC-182

Question History: Last NRC Exam NRC Exam 7/1997

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 2

10 CFR Part 55 Content: 55.41.b.7

Comments:

QUESTION 30 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	209001 K6.03		IR 3.3

Knowledge of the effect that a loss or malfunction of the following will have on the LOW PRESSURE CORE SPRAY SYSTEM : Torus/suppression pool water level

Proposed Question:

The Low Pressure Core Spray (LPCS) pump should **not** be run with suction from the suppression pool if level is less than 13 ft 3in, **except** in accordance with the EOPs because of the need to ____.

A. install an EOP-0005 enclosure to allow operation.

B. assure adequate net positive suction head.

C. provide cooling to the pump bearings.

D. maximize pump injection time following a LOCA.

Proposed Answer: B

Explanation
A. There are NO interlocks between suppression pool level and LPCS operation
B. Correct- SOP-0032 Precaution and Limitation 2.10 Lists 13’3” as the necessary level to assure NPSH
C. The pump bearings are NOT cooled by system flow
D. While a high suppression pool level will allow longer pump run time prior to loss of NPSH it is not the reason behind the limitation of 13’3”

Technical Reference(s): SOP-0032, LPCS, Rev 23 P&L 2.10 (p. 3 of 30)

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0205, Obj 8

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 31 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	209001 A3.03		IR 3.5

Ability to monitor automatic operations of the LOW PRESSURE CORE SPRAY SYSTEM including: System pressure

Proposed Question:

The Low Pressure Core Spray (LPCS) system is running in the test return to the suppression pool mode at 5050 gpm.

A leak has caused drywell pressure to increase to 1.95 psid.
Reactor water level is -62 inches
Reactor pressure is 450 psig

Which of the following identifies the expected AUTOMATIC response?

A. The Test Return Valve to the Suppression Pool (E21-F012) closes, Injection Isolation Valve (E21-F005) opens and discharge pressure rises.

B. The Test Return Valve to the Suppression Pool (E21-F012) closes, Injection Isolation Valve (E21-F005) opens and discharge pressure lowers.

C. The Test Return Valve to the Suppression Pool (E21-F012) remains open, Injection Isolation Valve (E21-F005) remains closed and discharge pressure rises.

D. The Test Return Valve to the Suppression Pool (E21-F012) remains open, the Injection Isolation Valve (E21-F005) remains closed and discharge pressure lowers.

Proposed Answer: A

Explanation
A. Upon a LOCA signal of 1.68 drywell diff pressure the LPCS system automatically aligns to inject into the RPV, the test return valve closes, if open, the injection valve opens, if RPV pressure is below 487 psig and the min flow valve opens if flow drops below 875 gpm. Since the stem RPV pressure is below the injection valve interlock pressure but above the pump shut off head (282 psig) the injection valve will open but the pump will not flow into the RPV. The min flow valve will open, flow will drop from the 5050 gpm and discharge pressure will rise. For these reasons all other choices are incorrect.
B, C, D. See A

Technical Reference(s): R-STM-0205, LPCS, Rev 5 pp. 17-18

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0205, Obj 9

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 4

10 CFR Part 55 Content: 55.41.b.7

Comments:

QUESTION 32 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	209002 A1.08		IR 3.1

Ability to predict and/or monitor changes in parameters associated with operating the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) controls including: System lineup: BWR-5,6

Proposed Question:

Following a loss of normal Feedwater, HPCS initiated and is restoring level. At 0 inches in the RPV, the operator closes the HPCS injection valve. How will the HPCS Discharge Pressure and HPCS Flow Rate parameters change?

	Discharge Pressure	Flow Rate
A.	Rise	Rise
B.	Rise	Lower
C.	Lower	Rise
D.	Lower	Lower

Proposed Answer: B

Explanation
A. See B
B. Correct- as the pumps discharge is constrained to the min flow capacity of 500 gpm, discharge pressure will rise and flow rate will lower
C. See B
D. See B

Technical Reference(s): R-STM-0203, HPCS, Rev 8 pp.15,16,20

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0203, Obj 9

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 2

10 CFR Part 55 Content: 55.41.b.8

Comments:

QUESTION 33 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 2	Group # 1	
	K/A # 211000 K5.06	IR 3.0	

Knowledge of the operational implications of the following concepts as they apply to STANDBY LIQUID CONTROL SYSTEM: Tank level measurement

Proposed Question:

An ATWS has occurred and the CRS has directed SLC injection. The hard card directs you to monitor SLC tank level and to report when Hot Shutdown Boron Weight is achieved. If the initial SLC tank level is 2666 gallons, which of the following is the highest tank level at which Hot Shutdown Boron Weight should be reported as having been injected?

A. 1986 gallons

B. 2036 gallons

C. 1866 gallons

D. 1966 gallons

Proposed Answer: D

Explanation
A. See D
B. See D
C. See D
D. Correct- Note above tank levels in OSP-0053 states that "When tank level falls between values, then the smaller value should be used", with an initial value of 2666 gallons the level must be the highest level that is below 1974 gallons which is 1966 gallons

Technical Reference(s): OSP-0053, Emergency & Transient Response, Attachment 13, Rev 22

Proposed references to be provided to applicants during examination: OSP-53, Attach. 13 p. 2 of 2 only

Learning Objective: RLP-STM-0201, Standby Liquid Control, Obj 1

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 2

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 34 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	211000 A2.03		IR 3.2

Ability to (a) predict the impacts of the following on the STANDBY LIQUID CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A.C. power failures

Proposed Question:

Following a LOSS of RSS #1 Plant conditions are as follows:

- All control rods did not insert
- Reactor power is 38%
- Division 1 DG started but tripped
- Division 2 DG is in standby
- Division 3 DG is running supplying its bus

Which of the following is a correct statement with regard to SLC operation?

A. SLC injection is required using SLC Pump A

B. SLC injection is required using SLC Pump B

C. EOP-0005 Enclosure 15 for alternate SLC is required

D. SLC injection is not required

Proposed Answer: B

Explanation
A. No power is available to SLC pump A
B. Correct - 38% power exceeds the drain and bypass valve capacity so SLC injection is required; power is only available to Div 2 (SLC pump B)
C. Power is available to SLC pump B, Alternate SLC injection is not warranted.
D. See B

Technical Reference(s): EOP-1, RPV Control ;
R-STM-0201, SLC, Rev 7 pp. 22,32

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-513, Obj 3

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 35 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	212000 A3.06		IR 4.2

Ability to monitor automatic operations of the REACTOR PROTECTION SYSTEM including: Main turbine trip: Plant-Specific

Proposed Question:

Following a main turbine trip from 100% power, the reactor will scram first on which of the following signals?

A. Low water level due to shrink

B. High pressure due to control valve closure

C. High neutron flux due to pressure rise

D. Turbine stop valve position

Proposed Answer: D

Explanation
A. See D
B. See D
C. See D
D. Correct - This scram anticipates the rise in reactor pressure, neutron flux and heat flux resulting from the loss of heat sink on a turbine trip

Technical Reference(s): R-STM-0508, RPS, Rev 6 p. 47 of 59

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0508, Obj 2

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.6

Comments:

QUESTION 36 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 2	Group #	1
	K/A # 215003 A4.05		IR 3.4

Ability to manually operate and/or monitor in the control room: Trip bypasses

Proposed Question:

A plant startup is in progress. Reactor power is on range 5 of the IRMs.

IRM-F amplifier card fails, causing an INOP condition on IRM-F and a ½ Scram.

The CRS directs you to bypass IRM-F and reset the ½ Scram.

To bypass the IRM you should:

A. Rotate the IRM draw mode switch to the STANDBY position

B. Fully withdraw IRM-F and down range to range 1

C. Rotate the IRM draw mode switch to the TRIP/TEST position

D. Place the IRM bypass Select Switch to the IRM-F position

Proposed Answer: D

Explanation
A. Rotating this switch to standby generates an INOP condition for the IRM and does not change the status
B. This will only bypass the IRM if the reactor mode switch is in RUN
C. See A
D. Correct - This is the only method for the control room operator to bypass IRM-F in these conditions

Technical Reference(s): SOP-0074, Neutron Monitoring, Rev 306
R- STM-0503 , Neutron Monitoring, Rev 8 p. 42 of 112

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.2

Comments:

QUESTION 37 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	215004 K2.01		IR 2.6

Knowledge of electrical power supplies to the following: SRM channels/detectors

Proposed Question:

The power supply to SRM ‘C’ detector is ___.

A. RPS A

B. RPS B

C. VBS-PNL01A1

D. VBS-PNL01B1

Proposed Answer: C

Explanation
A. Correct - RPS A is the power supply to SRM A & C
B. RPS B is the power supply to SRM B & D
C. VBS-PNL01A1 is the power supply to neutron monitoring recorders
D. VBS-PNL01B1 is the power supply to neutron monitoring recorders

Technical Reference(s): R-STM-0503, Neutron Monitoring, Rev 8 p.84 of 112

Proposed references to be provided to applicants during examination: None

Learning Objective: R-STM-0503, Obj 7

Question Source: Modified Bank # Nov 2010 Audit Q#36

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.2

Comments:

QUESTION 38 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑ SRO ☐
	Tier #	2	Group # 1
	K/A #	215004 G2.4.4	IR 4.5

Knowledge of annunciator alarms, indications, or response procedures. SRMs

Proposed Question:

During a reactor startup, the ATC has been withdrawing SRMs in accordance with GOP-0001, Plant Startup. All SRMs **except** C indicate full out; SRM C is reading offscale high and indicates “driving out.” The following alarm is in:

- SRM UPSCALE OR INOPERATIVE

Which of the following additional conditions will initiate a Control Rod Withdraw Block?

A. Mode switch in RUN and all IRMs on Range 8

B. Mode switch in STARTUP/HOT STBY and all IRMs on Range 7

C. Mode switch in STARTUP/HOT STBY and SRM C Bypassed

D. Mode switch in RUN and SRM C Selector switch in Standby

Proposed Answer: B

Explanation
A. and D. The rod withdrawal block is bypassed when the mode switch in run
B. Correct- If IRMs are not on Range 8 or above in Mode 2, an SRM UPSC or INOP initiates a rod withdrawal block .
C. The rod withdrawal block is bypassed when the affected SRM is bypassed

Technical Reference(s): ARP-P680-05-C05

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0503, Obj 5,7

Question Source: Modified Bank # RBS-NRC-53

Question History: Last NRC Exam NA (original question was on the 1997 NRC exam)

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 2

10 CFR Part 55 Content: 55.41.b.6

Comments: Rejected KA: There is no AOP or EOP entry conditions having to do with SRMs.
The only G2.4 category applicable to SRMs is G2.4.31

QUESTION 39 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	215005 K1.09		IR 3.6

Knowledge of the physical connections and/or cause-effect relationships between AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM and the following: Reactor recirculation system: BWR-5,6

Proposed Question:

Core flow rate is provided to the APRMs for use by the flow control trip reference cards to establish control rod withdrawal blocks and scram trip setpoints, this flow rate is obtained from which of the following sources?

A. the calibrated jet pumps

B. the above and below core plate differential pressure

C. the elbow taps on both recirculation loops

D. the steam flow feed flow differential summer

Proposed Answer: C

Explanation
A. See C
B. See C
C. Correct- A d/p signal is provided to the APRMs from elbow taps on both recirculation loops
D. See C

Technical Reference(s): R-STM-0503, Neutron Monitoring, Rev 7 p. 60 of 112

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0503, Obj 27

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.2

Comments:

QUESTION 40 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 2	Group # 1	
	K/A # 217000 K2.01		IR 2.8

Knowledge of electrical power supplies to the following: Motor operated valves

Proposed Question:

The normal power supply to E51-F063, RCIC STEAM SUPPLY INBD ISOL VALVE is ____.

A. EHS-MCC2L

B. EHS-MCC2D

C. ENB-MCC1

D. BYS-SWG01B

Proposed Answer: B

Explanation
A. EHS-MCC2L is an alternate source of power to E51-F063 utilized for AOP-0031
B. Correct- EHS-MCC2D is the normal power source to E51-F063
C. DC supply ENB-MCC1 is the source of power to many E51 MOVs but NOT to E51-F063
D. DC supply BYS-SWG01B is the source of power to the RCIC gland seal compressor NOT to E51-F063

Technical Reference(s): R-STM-0209, RCIC, Rev 10 p. 42 of 52

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0209, Obj 13

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.8

Comments:

QUESTION 41 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 2	Group # 1	
	K/A # 218000 K3.02	IR 4.5	

Knowledge of the effect that a loss or malfunction of the AUTOMATIC DEPRESSURIZATION SYSTEM will have on following: Ability to rapidly depressurize the reactor

Proposed Question:

The plant is operating in the Emergency Operating Procedures following a significant transient. During the transient, a short resulted in the loss of ENB-PNL02A.

Plant conditions require Emergency Depressurization per the Emergency Operating Procedures.

Which of the following represents the method that should be used to accomplish Emergency Depressurization?

A. At H13-P601, open 7 ADS/SRVs.

B. At H13-P631, open 7 ADS/SRVs.

C. Arm and depress the Division 1 ADS Manual Initiate pushbuttons.

D. Alternate depressurization methods listed in the EOPs should be utilized due to SRV failure.

Proposed Answer: B.

Explanation:
A. Loss of power to ENB-PNL02A prevents use of Div 1 SRV solenoids; the switches on P601 are Div 1.
B. Correct – Div 2 solenoids are still available to open the SRVs.
C. Div 1 solenoids are de-energized.
D. This would only be required if SRVs could not be opened. Div 2 solenoids are still available.

Technical Reference(s): R-STM-0202, ADS, Rev 2 pp. 20,21 of 36
R-STM-0109, Main Steam, Rev 13 p. 12 of 95

Proposed references to be provided to applicants during examination: None

Learning Objective: R-STM-0202 Obj. 6,12

Question Source: New

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3

10 CFR Part 55 Content: 55.41b.3

Comments:

QUESTION 42 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	218000 A4.02		IR 4.2

Ability to manually operate and/or monitor in the control room: ADS logic initiation

Proposed Question:

The plant has scrammed due to a loss of offsite power.
Neither HPCS nor RCIC will start.
Approximately 5 minutes later, RPV water level decreases below -143 inches, the "DIV 2 ADS LOGIC TIMER INITIATED" annunciator illuminates.
The Unit Operator is directed to "INHIBIT ADS" per EOP-0001.
Later the Operator ARMS and Depresses the ADS B MANUAL INITIATION pushbuttons.
What is the response of the ADS System in this situation?

ADS will initiate:

A. immediately, if any Div 2 low pressure ECCS subsystem pressure permissive is satisfied.

B. in 105 seconds, if any Div 2 low pressure subsystem pressure permissive is satisfied.

C. immediately, regardless of low pressure ECCS subsystem status.

D. In 105 seconds, regardless of low pressure ECCS subsystem status.

Proposed Answer: A

Explanation
A. Correct- ADS Logic only requires subsystem pressure permissive prior to manual initiation
B. See A
C. See A
D. See A

Technical Reference(s): R-STM-0202, ADS, Rev 2 p.38 of 41 (Figure 2)
Proposed references to be provided to applicants during examination: None
Learning Objective: RLP-STM-0202, Obj 7
Question Source: New Question History: Last NRC Exam NA
Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 2
10 CFR Part 55 Content: 55.41.b.8
Comments:

QUESTION 43 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 2	Group # 1	
	K/A # 223002 K4.01	IR 3.0	

Knowledge of PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF design feature(s) and/or interlocks which provide for the following: Redundancy

Proposed Question:

Why do systems penetrating containment which are part of the reactor coolant boundary have redundant isolation valves?

A. To allow one valve to be removed from service for maintenance activities.

B. To ensure that multiple failure will not prevent at least a single valve isolation of the process line

C. To allow for manual isolation from the control room in case the automatic isolation does not work

D. To ensure that a single failure will not prevent at least a single valve isolation of the process line.

Proposed Answer: D

Explanation
A. See D
B. Multiple failures could still prevent penetration isolation
C. Both valves must be automatic isolations per design criteria
D. Correct- 10CFR50, General Design Criteria 55

Technical Reference(s): R-STM-0058, CRVICS, Rev 9 p. 5 of 63

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0058, Obj 11

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.3

Comments:

QUESTION 44 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	239002 K5.04		IR 3.3

Knowledge of the operational implications of the following concepts as they apply to RELIEF/SAFETY VALVES : Tail pipe temperature monitoring

Proposed Question:

The plant had been operating at 100% power when a severe over-pressure transient resulted in ALL Safety Relief Valves (SRV) opening in the "relief" mode.

- RPV pressure peaked at 1200 psig
- Current reactor pressure is 500 psig and lowering
- One SRV remains stuck open

Which of the following describes the resulting tailpipe temperature trend as the plant cools down and depressurizes through the stuck open SRV? (Assume containment pressure is 0 psig and remains constant.)

SRV tailpipe temperature will ___ and will then slowly fall, following reactor pressure during the depressurization below 500 psig.

A. Start at 285°F, currently 280°F

B. Start at 285°F, currently 330°F

C. Start at 330°F, currently 280°F

D. Start at 330°F, currently 320°F

Proposed Answer: B

Explanation
A, Correct starting point, however tail pipe temperature will rise to 330ºF as pressure falls.
B. Correct - At onset of this event the tail pipe temperature would be 285ºF (the isentropic value for 1200 psig and ~25 psia)(14.7 plus 10 feet of water over the tail pipe quencher). Tail pipe temperature will rise to 330ºF as pressure falls to 500 psig.
C and D cannot be true due to temperature dropping

Technical Reference(s): HLO-168 Steam Tables

Proposed references to be provided to applicants during examination: Mollier Diagram (large)

Learning Objective:

Question Source: Modified Bank # RBS-NRC-459

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3

10 CFR Part 55 Content: 55.41.b.14

Comments:

QUESTION 45 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 2	Group # 1	
	K/A # 259002 K6.03		IR 3.1

Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR WATER LEVEL CONTROL SYSTEM : Main steam flow input

Proposed Question:

While operating at 100% power, the 'C' Main Steam Line flow transmitter fails low. What is the expected response of the Feedwater level control system with no operator actions taken?

A. RPV level will lower until it reaches Level 3 and the reactor scrams.

B. RPV level will lower and stabilize at a new lower level above Level 3.

C. RPV level will rise and stabilize at a new higher level below Level 8.

D. RPV level will rise until it reaches Level 8 and the reactor scrams.

Proposed Answer: B

Explanation
A. Only first part is correct; see B
B. Correct – A loss of 1 out of 4 steam flow transmitters results in a sensed reduction in steam flow of 25%; this sensed reduction will cause the FWLC system to close the Feed Reg Valves and consequently RPV level will lower. Level will stabilize below its normal level (but above the Level 3 scram setpoint) because the FWLC system is level dominant.
C. This is the expected response for a single steam line transmitter failing high
D. This is the expected response for a single feed flow transmitter failing low

Technical Reference(s): R-STM-0107, Feedwater, Rev 27 p. 71 of 105

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0107, Obj B14

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3

10 CFR Part 55 Content: 55.41.b.4

Comments:

QUESTION 46 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 2	Group # 1	
	K/A # 261000 A1.01		IR 2.9

Ability to predict and/or monitor changes in parameters associated with operating the STANDBY GAS TREATMENT SYSTEM controls including: System flow

Proposed Question:

A leak has occurred in the Drywell and Drywell pressure is 2.14 psid and stable. Both trains of Standby Gas Treatment (GTS) started and consequently OSP-0053 Hard Card Attachment 21, Operating Auxiliary Building Ventilation was used to reduce to one train of SBGT running; GTS-A is running. Subsequently, a loss RSS#1 occurred and the Div 1 Diesel Generator failed to start.

What is the status of GTS-B following the loss of RSS#1?

A. GTS-B may only be started in High Volume Purge mode due to power failure.

B. GTS-B must be manually initiated due to manually securing.

C. GTS-B will restart automatically due to low flow in GTS-A.

D. GTS-B will restart automatically due to undervoltage trip of GTS-A.

Proposed Answer: C

Explanation
A. Power is only lost to Div 1; Div 2 is still available
B. Securing per the hard card places GTS-B in standby;
C. Correct - DW 1.68 is still locked in, so when low flow occurs in Div 1 due to the loss of power, then the Div 2 will automatically restart.
D. The interlock that will start GTS-B is low flow in the running GTS; not undervoltage.

Technical Reference(s): R-STM-0257, Standby Gas Treatment, Rev 5 p. 15 of 28

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0257, Obj 5,12

Question Source: New   Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐   Comprehension or Analysis ☑ 3

10 CFR Part 55 Content: 55.41.b.7

Comments:

QUESTION 47 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	262001 A2.11		IR 3.2

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: Degraded system voltages

Proposed Question:

The plant is operating at 100% power when alarm P808-86A, Grid Trouble is received.
SPI-REC102 on P808 indicates 230 KV system voltage at Fancy Point to be 223.3 KV.
The SOC contacts the control room and indicates that a transient has occurred and Fancy Point stability cannot be determined at this time.
The synchroscope for Division I D/G is indicating erratically.

(1) What is the impact to plant equipment?
(2) What actions should be taken to mitigate this event?

A. (1) Amps will increase on the running equipment.
(2) Perform a normal start of the D/G, parallel it to offsite power and disconnect the bus from the grid.

B. (1) Amps will increase on the running equipment.
(2) Perform an emergency start of the D/G and open the safety related bus supply breaker.

C. (1) Amps will decrease on the running equipment.
(2) Perform an emergency start of the D/G and open the safety related bus supply breaker.

D. (1) Amps will decrease on the running equipment.
(2) Perform a normal start of the D/G, parallel it to offsite power and disconnect the bus from the grid.

Proposed Answer: B

Explanation
Plant indications show that Fancy Point is NOT stable(second note on page 64 of AOP-64)
A. Would be correct if Fancy Point was stable
B. Correct per steps:5.5.3.1 and 5.5.3.4
C. and D. Running equipment amps will rise as voltage lowers. Normal voltage is 230KV the stem gives 223.3KV

Technical Reference(s): AOP-0064, Degraded Grid, Rev 6 p 8 of 11
Proposed references to be provided to applicants during examination: AOP-0064 Degraded Grid, Rev 6
Learning Objective: ________________________ (As available)
Question Source: New Question History: Last NRC Exam NA
Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 4
10 CFR Part 55 Content: 55.41.b.5
Comments:

QUESTION 48 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	262002 A3.01		IR 2.8

Ability to monitor automatic operations of the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) including: Transfer from preferred to alternate source

Proposed Question:

BYS-INV01A displays the following information:

- Rectifier Output 0 amps
- Battery Output Normal at 135 VDC
- Inverter Output Normal at 122 VAC

What is the status of the loads normally supplied by this inverter?

A. Loads are currently de-energized

B. Loads are currently being supplied by the battery via the inverter

C. Loads are currently being supplied by the normal AC source via the inverter

D. Loads have automatically swapped to the alternate AC source

Explanation
A. The inverter output has power, so the loads have power
B. Correct – The normal AC source is unavailable based on the rectifier output; output of the battery and inverter reveal that the battery is supplying power to the inverter and the inverter is powering the loads
C. Rectifier output reveals that the normal AC power source is de-energized
D. Alternate AC power source is provided through a Manual Bypass Switch (which has not been turned)

Proposed Answer: B

Technical Reference(s): R-STM-0300, AC Electrical Distr., Rev 26 pp. 28-29 of 105

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0300, Obj 12

Question Source: Bank # RBS Nov 2008 Audit Q#49

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3

10 CFR Part 55 Content: 55.41.b.7

Comments:

QUESTION 49 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 2	Group #	1
	K/A # 263000 A4.01		IR 3.3

Ability to manually operate and/or monitor in the control room: Major breakers and control power fuses: Plant- Specific

Proposed Question:

A breaker test is being performed on ENS-ACB03, E12-C002A RHR A PUMP breaker. The breaker will be in the TEST position with control power fuses INSTALLED. The breaker will then be CLOSED.
Which of the following represents the expected control room light indications for the RHR Pump A breaker when the test conditions mentioned above are established?

A. Red light OFF, Green light OFF, White light OFF

B. Red light OFF, Green light OFF, White light ON

C. Red light ON, Green light OFF, White light OFF

D. Red light ON, Green light OFF, White light ON

Proposed Answer: C

Explanation
A. B. and D. See C
C. While in the test position the control room breaker indication for the red and green lights will be the same as the local indication however the white power available light will not be lite even with the control power fuses installed unless the breaker is in the 'connect' position. All other combination of light indication s are incorrect

Technical Reference(s): OSP-0052, Breaker Racking, Rev 18 p

Proposed references to be provided to applicants during examination: None

Learning Objective: Not available

Question Source: Bank # RBS-OPS-07756

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 50 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑ SRO ☐
	Tier #	2	Group # 1
	K/A #	264000 K1.05	IR 3.2

Knowledge of the physical connections and/or cause-effect relationships between EMERGENCY GENERATORS (DIESEL/JET) and the following: Emergency generator fuel oil supply system

Proposed Question:

The correct fuel oil supply flowpath for operation of the Div 1 Standby Emergency Diesel Generator is from its associated storage tank through the _____, to the injectors.

A. fuel oil transfer pump, strainer, duplex filter, fuel oil day tank

B. strainer, fuel oil transfer pump, fuel oil day tank, duplex filters

C. strainer, fuel oil transfer pump, fuel oil day tank, booster pump

D. fuel oil transfer pump, strainer, fuel oil day tank, booster pump

Proposed Answer: D

Explanation
A. the filter is located after the day tank
B. the strainer is after the transfer pump
C. the strainer is after the transfer pump
D. Correct – The full flowpath of fuel oil is storage tank, transfer pump, day tank, duplex strainer, booster pump, duplex filter, injector pump, injectors

Technical Reference(s): R-STM-0309S, Rev 14, pp. 17 & 20 of 117

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0309S, Obj 2

Question Source: New

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.8

Comments:

QUESTION 51 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	264000 G2.1.31		IR 4.6

Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup. Emergency Diesel Generators

Proposed Question:

The Div 2 Emergency Diesel Generator is operating and tied to ENS-SWG 1B. The unit operator rotates the DG Voltage Regulator Control Switch to raise. Which indication shows the largest change (1) if offsite is paralleled to the bus, and (2) if offsite is <u>NOT</u> paralleled to the bus.

	(1)	(2)
A.	Reactive Load (KVAR)	Frequency (Hz)
B.	Real Load (KW)	Voltage (VAC)
C.	Reactive Load (KVAR)	Voltage (VAC)
D.	Real Load (KW)	Frequency (Hz)

Proposed Answer: C

Explanation
A. (1) is correct however, Frequency is controlled by the Vovernor control switch when not in parallel
B. Generator load (KW) is adjusted with the Governor Control switch when the EDG is in parallel
C. Correct – The voltage regulator control switch varies excitation current and will adjust generator reactive load when in parallel and adjusts voltage when not in parallel.
D. Generator load (KW) is adjusted with the Governor Control switch when the EDG is in parallel

Technical Reference(s): R-STM-0309S, Rev 13 pp. 43-44 of 117

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0309S, Obj 8

Question Source: Modified Bank # RBS-OPS-3255

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 2

10 CFR Part 55 Content: 55.41.b.7

Comments:

QUESTION 52 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	300000 K2.01		IR 2.8

Knowledge of electrical power supplies to the following: Instrument air compressor

Proposed Question:

The Instrument air compressors A, B, and C are powered from ___, ___, and ___ respectively.

A. NJS-SWG 1G, 1H, and 1F

B. NJS-SWG 1E, 1J, and 1K

C. NHS-MCC 1L, 1M, and 1M

D. NHS-MCC 1L, 1L, and 1M

Proposed Answer: A

Explanation
A. Correct- these are the power supplies to the IAS compressors
B. these are the power supplies to the Service Air compressors
C. these are the power supplies to the IAS trim coolers
D. these are the power supplies to the SAS trim coolers

Technical Reference(s): R-STM-0121, Rev 16 p. 9 of 69

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0121, Obj 10

Question Source: New

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.4

Comments: rejected KA K2.02; re-selected K2.01 – RBS has diesel operated emergency air compressor

QUESTION 53 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	400000 K3.01		IR 2.9

Knowledge of the effect that a loss or malfunction of the CCWS will have on the following: Loads cooled by CCWS

Proposed Question:

The Division 2 isolation valves in the Reactor Plant Component Cooling Water System (CCP) have closed on a 56 psig isolation signal.

What equipment is still receiving cooling water from the CCP system?

A. Recirc pumps, RWCU NRHXs, RWCU pumps

B. RWCU pumps, RHR Pump A Seal Coolers, RWCU NRHXs

C. RWCU pumps, Reactor Plant Sampling panel, Fuel Pool Cooler 1A

D. Recirc pumps, RHR Pump A Seal Coolers, Reactor Plant Sampling panel

Proposed Answer: A

Explanation
A. Correct – The Safety loop isolates which secures cooling to the RHR Pump Seal Coolers, Fuel pool coolers, and CRD pumps. Cooling is not isolated to the three loads listed.
B. RHR Pump Seal Coolers (both divisions) are isolated as part of the safety loop
C. Fuel Pool Coolers (both divisions) are isolated as part of the safety loop
D. RHR Pump Seal Coolers (both divisions) are isolated as part of the safety loop

Technical Reference(s): R-STM-0115, Rev 6 pp. 12-14 of 35 and Figure 1

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0115, Obj 4,8,11

Question Source: Bank # RBS-OPS-1789

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.7

Comments:

QUESTION 54 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	2
	K/A #	201003 A4.02		IR 3.5

Ability to manually operate and/or monitor in the control room: CRD mechanism position: Plant-Specific

Proposed Question:

The ATC operator just selected a control rod to move in accordance with an approved reactivity maneuver plan when the Control Rod Withdrawal Block Alarm annunciates.

The RC & IS Operator Control Module (OCM) also displays the following:

- DATA FAULT pushbutton is backlit
- CH DATA 1 / CH DATA 2 are consecutively lighting and extinguishing
- CHANNEL DISAGREE is backlit

The rod position displayed on the full core display for the selected rod is flashing alternately between 08 and - - .

What operator action would allow this rod to be moved?

A. Isolating the rod electrically within the Rod Gang Drive Cabinet

B. Depressing the DATA MODE pushbutton on the OCM

C. Depressing the DATA SOURCE pushbutton on the OCM

D. Depressing the ENTER SUBST pushbutton on the OCM

Proposed Answer: D

Explanation

A. Electrically isolating the rod in RGDC will prevent rod movement; bypassing the rod in RACS (Rod Action Control System would allow rod movement.

B. This pushbutton works in conjunction with the Data Source pushbutton to allow you to choose either single channel or alternating channel(s) to display on the full core display; this would make the alternating rod position indication to stop, but not clear the rod block.

C. This pushbutton allows you to choose which of the the 2 channels to display on the full core display; pushing this button would show first one channel then pushing it again would show the other.

D. Correct – This pushbutton forces the single channel of bad data to be replaced with the other channels good data; thereby allowing rod movement.

Technical Reference(s): R-STM-0500, Rev 3 pp. 19-25 of 46; SOP-0071, Rev 29, Section 5.12
Proposed references to be provided to applicants during examination: None
Learning Objective: RLP-STM-0500, Obj 7,9
Question Source: New Question History: Last NRC Exam NA
Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3
10 CFR Part 55 Content: 55.41.b.6
Comments:

QUESTION 55 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	2
	K/A #	201005 A1.01		IR 3.2

Ability to predict and/or monitor changes in parameters associated with operating the ROD CONTROL AND INFORMATION SYSTEM (RCIS) controls including: First stage shell pressure/turbine load: BWR-6

Proposed Question:

Given the following plant conditions:

- Reactor power 45%
- Generator load 480 MWe

Power ascension is in progress. The next step of the Reactivity Maneuvering Plan is to select and continuously withdraw control rod 28-49 from position 12 to position 24.

Just prior to withdrawing the rod, the Main Turbine First Stage Shell Pressure transmitter output signal fails upscale.

When the withdraw button is pushed, control rod 28-49 will ___.

A. remain at position 12

B. withdraw to position 16 and settle

C. withdraw to position 20 and settle

D. withdraw to position 24 and settle

Proposed Answer: B

Explanation

A. If turbine first stage shell pressure was indicating failed low, the Rod Pattern Controller logic would be enabled and could cause a rod withdraw blow (depending on the current rod pattern)

B. Correct- the rod withdrawal limitations are dependent on reactor power as sensed by First Stage Shell Pressure. An upscale failure would indicate reactor power above the high power setpoint to RC&IS. The Rod Withdrawal Limiter will then limit rod withdrawals to 2 notches (12-16).

C. See B

D. See B

Technical Reference(s): R-STM-0500, Rev 3 p.16 of 46

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0500, Obj 22

Question Source: Bank # RBS-NRC-665

Question History: Last NRC Exam Dec 2008

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3

10 CFR Part 55 Content: 55.41.b.6

Comments:

QUESTION 56 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	2
	K/A #	202001 K4.02		IR 3.1

Knowledge of RECIRCULATION System design feature(s) and/or interlocks which provide for the following: Adequate recirculation pump NPSH

Proposed Question:

When total Feedwater flow drops below the Reactor Recirc system interlock level, the Reactor Recirc Pumps downshift from fast speed to slow speed. This interlock ___.

A. prevents flow velocity effects on the wide range level indication.

B. prevents thermal stress on the Recirc pump

C. prevents cavitation in the Recirc pumps

D. adds negative reactivity in anticipation of an imminent reactor scram

Proposed Answer: C

Explanation

A. the interlock associated with wide range flow velocity effects is the low reactor level interlock

B. the interlock associated with thermal stress is the Steam Dome to Vessel Bottom Head delta T

C. Correct – Cavitation is prevented by inhibiting high speed operation of pumps with feed flow < 19.9%

D. the EOC-RPT interlock anticipates a reactor scram based on a turbine trip causing a pressure transient, which will rapidly add positive reactivity.

Technical Reference(s): R-STM-0053, Rev 13 p. 27 of 76

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0053, Obj 17

Question Source: Modified Bank # RBS-NRC-168

Question History: Last NRC Exam 1995

Cognitive Level: Memory or Fundamental Knowledge ☑ 2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.2

Comments:

QUESTION 57 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	2
	K/A #	202002 K1.09		IR 3.1

Knowledge of the physical connections and/or cause-effect relationships between RECIRCULATION FLOW CONTROL SYSTEM and the following: Reactor water level

Proposed Question:

The plant is operating at 100% power. The "A" narrow range level channel is selected as the input to the Feedwater Level Control System. A leak has developed in the reference leg of the "A" narrow range level transmitter.

The ATC operator promptly placed the Master Feedwater Level Controller in MANUAL. As a result of this condition, both Recirc pumps will ___.

A. remain at their present speed, and the Recirc Flow Control Valves will runback to 60% drive flow position.

B. transfer to SLOW speed operation, and the Recirc Flow Control Valves will remain at their present position.

C. transfer to SLOW speed operation, and the Recirc Flow Control Valves will runback to 60% drive flow position.

D. remain at their present speed, and the Recirc Flow Control Valves will remain at their present position.

Proposed Answer: D

Explanation

A. first part is correct, but the reference leg leak will cause a false high level indication; FCV's will not runback due to a high level

B. and C. The reference leg leak will cause a false high level indication, Recirc pumps will not downshift due to a high RPV level.

D. Correct – Recirc Pumps and FCV's logic receive level signal from the narrow range selected (A); the reference leg leak will cause transmitter differential pressure to rise, therefore the Recirc system will not receive a low RPV level (Recirc pumps downshift at level 3 and FCVs runback at level 4)

Technical Reference(s): R-STM-0107B, Rev 27 pp. 58-59

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0107B, Obj 10

Question Source: Bank # Dec 2008 NRC Exam Q#57

Question History: Last NRC Exam Dec 2008

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3

10 CFR Part 55 Content: 55.41.b.2

Comments:

QUESTION 58 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	2
	K/A #	216000 K6.01		IR 3.1

Knowledge of the effect that a loss or malfunction of the following will have on the NUCLEAR BOILER INSTRUMENTATION: A.C. electrical distribution

Proposed Question:

A loss of RPS Bus A causes which RPV level indication on the 680 panel to fail downscale?

A. B21-R604, Wide Range meter

B. C33-R606A, Narrow Range Channel A meter

C. C33-R608B, Narrow Range recorder

D. C33-R608R, Upset Range recorder

Proposed Answer: A

Explanation
A. Correct – power supply to the wide range meter comes from RPS –A and fails downscale
B. power supplied by inverter 1VBN-PNL01B1
C. power supplied by inverter 1VBN-PNL01B1
D. The narrow range/upset range recorder has a single power supply (1VBN-PNL01B1)

Technical Reference(s): R-STM-0051, Rev 5 p. 23 of 47

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0051, Obj 13; RLP-STM-0508, Obj 7

Question Source: Bank # RBS-NRC-808

Question History: Last NRC Exam Feb 2003

Cognitive Level: Memory or Fundamental Knowledge ☑ 2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.7

Comments:

QUESTION 59 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑ SRO ☐
	Tier #	2	Group # 2
	K/A #	219000 K2.02	IR 3.1

Knowledge of electrical power supplies to the following: Pumps (RHR-Supp Pool Cooling)

Proposed Question:

The electrical power supplies to (1) RHR Pump C, and (2) its' associated line fill pump is ___ and ___.

A. (1) ENS-SWG1A ; (2) ENS-SWG1A

B. (1) ENS-SWG1B ; (2) EJS-SWG1A

C. (1) ENS-SWG1B ; (2) EJS-SWG1B

D. (1) ENS-SWG1A ; (2) ENS-SWG1B

Proposed Answer: C

Explanation
A. Both part 1 and 2 are incorrect; See C
B. First part is correct; See C for part 2
C. Correct – RHR pumps B & C power supply is ENS-SWG1B; the line fill pump for Div 2 is EJS-SWG1B
D. Part 1 is incorrect; See C

Technical Reference(s): R-STM-0204, p. 25 of 63 ; EE-001AC

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0204, Obj 11

Question Source: New

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.8

Comments:

QUESTION 60 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	2
	K/A #	223001 G2.1.28		IR 4.1

Knowledge of the purpose and function of major system components and controls. Primary Containment Systems

Proposed Question:

The function of the ___ System is to control hydrogen concentration that would be generated during a Design Basis Accident. If hydrogen is detected in the drywell or containment, proper operation of this system alone is sufficient to consume the hydrogen being generated.

A. Containment Hydrogen Purge

B. Hydrogen Igniter

C. Hydrogen Mixing

D. Hydrogen Recombiner

Proposed Answer: B

Explanation
A. This system designed as a backup to the H_2 Recombiners; there is no EOP/SAP guidance for its use
B. Correct – The H_2 igniters are designed to handle 75% of the Metal-Water Reaction rate; they are designed to mitigate the consequences of a generation event more severe than a design basis LOCA. The Igniter system bounds the DBA scenario used in sizing the H_2 Mixing and H_2 Recombiner Systems.
C. Primary function is to mix containment and drywell atmospheres thereby temporarily diluting the H_2
D. Recombiners are started 14 days after a DBA

Technical Reference(s): EOP Bases, Rev 16, p. B-8-30, 31

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0057, Obj 20

Question Source: Modified Bank # RBS-OPS-06288

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.9

Comments:

QUESTION 61 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	2
	K/A #	239001 K3.15		IR 3.5

Knowledge of the effect that a loss or malfunction of the MAIN AND REHEAT STEAM SYSTEM will have on following: Reactor water level control

Proposed Question:

The plant is operating at 70% reactor power when one inboard MSIV fails closed.

RPV level will ___.

A. (1) increase and stabilize at a higher level.

B. (1) decrease and stabilize at a lower level.

C. (1) decrease and then return to normal level.

D. (1) increase and then return to normal level.

Proposed Answer: C

Explanation

A. RPV level indication measures water level in the downcomer region and due to steam flow across the dryers, level is approximately 7 inches higher in the downcomer than the core. When an MSIV closes, pressure will go up momentarily and concurrently collapse some of the steam void in the core. Water levels equalize and then FWLC will compensate and get level back to the the setpoint on the tape set

B. First part is correct, but RPV level is being controlled by FWLC system and will return to whatever the tape set is dialed to (setpoint setdown is not activated)

C. Correct – RPV level initially drops due to a pressure transient causing voids to collapse; water from downcomer flows into the core and then FWLC compensates and level returns to normal.

D. See A

Technical Reference(s): RPPT-HLO-0316, Slide 46 of 72

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0109, Obj 2, 19; RLP-HLO-0316, Obj 1, 2

Question Source: Bank # RBS-NRC-308

Question History: Last NRC Exam June 1995

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3

10 CFR Part 55 Content: 55.41.b.5

Comments:

QUESTION 62 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 2	Group # 2	
	K/A # 241000 A2.01		IR 3.5

Ability to (a) predict the impacts of the following on the REACTOR/TURBINE PRESSURE REGULATING SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of turbine inlet pressure signal

Proposed Question:

The plant is starting up and reactor power is currently 8%. EHC Pressure Regulator Channel A is selected for pressure control. Pressure Regulator Channel B is in TEST.

If the Averaging Manifold Pressure transmitter for Pressure Regulator Channel A fails to 0 psig, the Turbine Control Valves will __(1)__ and the steam Bypass Valves __(2)__.

A. (1) fully open; (2) remain closed

B. (1) fully close; (2) fully open

C. (1) fully close; (2) remain closed

D. (1) fully open; (2) fully open

Proposed Answer: C

Explanation
A. The pressure regulator sensing a low pressure, will close (not open) the TCVs trying to raise pressure.
B. TCVs will close causing pressure to rise, but the BPVs use the same pressure transmitter and therefore would remain closed (not open).
C. Correct - Turbine Control Valves will close and the BPVs use the same pressure transmitter and therefore would also remain closed
D. The pressure regulator sensing a low pressure, will close (not open) the TCVs trying to raise pressure

Technical Reference(s): R-STM-0509, Rev 14 p.51 of 81

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0509, Obj 10

Question Source: Modified Bank # RBS-LOR-1252

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 2

10 CFR Part 55 Content: 55.41.b.5

Comments:

QUESTION 63 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	2
	K/A #	272000 K4.02		IR 3.7

Knowledge of RADIATION MONITORING System design feature(s) and/or interlocks which provide for the following: Automatic actions to contain the radioactive release in the event that the predetermined release rates are exceeded

Proposed Question:

The plant is operating at 100% power. Annulus Pressure Control system is in operation with HVR-FN16A, Annulus Pressure Control (APC) Fan A running.

RMS-RE11A, Div 1 Annulus Exhaust Radiation Monitor goes into High Alarm (reading greater than 3.89E-5 µci/cc). Which of the following describes the ventilation lineup after this event?

A. HVR-FN16A trips; Both Standby Gas Treatment Trains are running

B. HVR-FN16A stays running; Both Standby Gas Treatment Trains are running

C. HVR-FN16A stays running; ONLY Standby Gas Treatment Train A is running

D. HVR-FN16A trips; ONLY Standby Gas Treatment Train A is running

Proposed Answer: A

Explanation
A. Correct - RMS-RE11A sends a signal to trip FN16A and start GTS-FNA; GTS-FNB will start due to a low APC System flow
B. RMS-RE11A sends a signal to trip FN16A
C. RMS-RE11A sends a signal to trip FN16A
D. Part 1 is correct; Part 2 is wrong because GTS-FNB will start due to a low APC System flow

Technical Reference(s): ARP-DSPL230-1GP011, Rev 9
R-STM-0511, Radiation Monitoring, Rev 15 p. 46 of 48

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0511, Obj 4,6

Question Source: New

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3

10 CFR Part 55 Content: 55.41.b.11

Comments:

QUESTION 64 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 2	Group # 2	
	K/A # 288000 A3.01		IR 3.8

Ability to monitor automatic operations of the PLANT VENTILATION SYSTEMS including: Isolation/initiation signals

Proposed Question:

Containment Unit Coolers 1A and 1B are running.
Which of the following signals will cause (1) chilled water to the Containment Unit Coolers to isolate, and (2) Service Water to the Containment Unit Coolers to align?

A. (1) High Negative D/P between Containment and Annulus (greater than -12”) ;
(2) RPV level 2 (-43”)

B. (1) High Drywell D/P (1.68) ;
(2) RPV level 1 (-143”) after 60 sec time delay

C. (1) RPV level 1 (-143”) ; (2) RPV level 1 (-143”) after 60 sec time delay

D. (1) RPV level 2 (-43”) ; (2) RPV level 2 (-43”)

Proposed Answer: B

Explanation
A. Part 1 is correct ; SW aligns on RPV level 1 after 60 time delay
B. Correct – chill water isolates on RPV level2, DW D/P, and -12”; Service Water does not align until level 1 is reached and then after a time delay of 60 seconds to allow separation between CW and SW.
C. Part 1 is incorrect – Level 2 signal isolates CW;
D. Part 1 is correct ; SW aligns on RPV level 1 after 60 time delay

Technical Reference(s): R-STM-0403, Reactor Building HVAC, Rev 8 pp. 7-11 of 50

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0403, Obj 6

Question Source: Modified Bank # RBS-OPS-2307

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.7

Comments:

QUESTION 65 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 2	Group # 2	
	K/A # 290003 K5.01		IR 3.2

Knowledge of the operational implications of the following concepts as they apply to CONTROL ROOM HVAC: Airborne contamination (e.g., radiological, toxic gas, smoke) control

Proposed Question:

A high-high radiation condition has been detected by RMS-RE13A and 13B, Main Control Room Local Intake A and B.

Which of the following represents the effect on the Control Building HVAC system?

	HVC-MOV1A/B <u>CR AHU Outside Supply</u>	HVC-AOD19C/D/E/F <u>HVC Local Air Intake</u>
A.	OPEN	OPEN
B.	OPEN	CLOSE
C.	CLOSE	OPEN
D.	CLOSE	CLOSE

Proposed Answer: C

Explanation
A high-high rad detected by RMS-RE13 will cause the following: HVC-MOV1 to close, HVC-AOD19s to open, and HVC-FN1 to start, Making C correct.

Technical Reference(s): ARP-P863-74A-H03 and –H08, Rev 24

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0402, Obj 7

Question Source: New

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 2

10 CFR Part 55 Content: 55.41.b.11

Comments:

QUESTION 66 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	3		
	K/A #	G2.1.7		IR 4.4

Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation

Proposed Question:

OSP-0053, Emergency and Transient Response Support Procedure, contains Post-Scram Level Control Strategies.

In non-ATWS conditions, when MSIVs are closed, the prescribed level band is __(1)__.

In an ATWS condition, with Reactor power at 30% and the Turbine on-line, the prescribed level band is __(2)__.

A. (1) 10 to 51 inches; (2) -60 to -140 inches

B. (1) -20 to 51 inches; (2) -100 to -140 inches

C. (1) 10 to 51 inches; (2) -100 to -140 inches

D. (1) -20 to 51 inches; (2) -60 to -140 inches

Proposed Answer: D

Explanation

A. Part one would only be correct if MSIVs were open

B. Part 1 is correct, part two would be correct if SRVs were required for pressure control

C. see A

D. Correct – The prescribed level band for a non-ATWS condition is normally 10 to 51", except when certain plant conditions described in the EOP bases are present. One such condition is closure of MSIVs where the expanded band is prescribed. In an ATWS when conditions are not present that would challenge containment (SRVs are not required for pressure control), then the level control band should remain -60 to -100 inches.

Technical Reference(s): OSP-0053, Rev 22 pp. 16-19 of 74
EOP Bases pp. B-6-12 & -13; B-7-21 & -22

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-OPS-0512, Obj 5

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 67 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	3		
	K/A #	G2.1.31		IR 4.6

Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup

Proposed Question:

The main turbine has just tripped while operating at 40% power.

The operator will see ___.

A. all 8 Scram Pilot Solenoid Valve indicating lights out on H13-P691.

B. Turbine Bypass Valves indicating open on H13-P680.

C. ENS-SWG1A aligned to NNS –SWG1B on H13-P808.

D. the main generator output breakers tripped and the exciter field breaker closed on H13-P680.

Proposed Answer: B.

Explanation:
A. The 8 scram pilot lights are on H13-P680. P691 contains the RPS trip unit that input to the trip logic.
B. Correct-Bypass valve indications are on H13-P680 and they would be open as a result of the turbine trip.
C. ENS-SWG1A is normally aligned to NNS-SWG1A. This transient would not change that lineup.
D. Exciter field breaker would also be tripped for this transient.

Technical Reference(s): AOP-0001, Reactor Scram, Rev 28 p. 4 of 10;
AOP-0002, Turbine Trip, Rev 26 p. 6 of 10

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0310 Obj. 8

Question Source: Bank # RBS-NRC-01183

Question History: Last NRC Exam RBS Dec. 2010 Q# 67

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 2

10 CFR Part 55 Content: 55.41b.4

Comments:

QUESTION 68 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 3		
	K/A # G2.1.40		IR 2.8

Knowledge of refueling administrative requirements

Proposed Question:

During refueling activities, Main Control Room personnel are required to maintain constant communication with the ___.

A. Fuel Movement Supervisor

B. Bridge Driver/Fuel Handler

C. IFTS Operator

D. Refuel Senior Reactor Operator

Proposed Answer: D

Explanation: Required by FHP-0003, Roles and Responsibilities of Refuel SRO Step 2.1

Technical Reference(s): TR 3.9.11, Communications during core alts
EN-FAP-OU-108, Fuel Handling Process, Rev 5 pp. 3,4 of 39

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-OPS-0410, Obj 1

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 4 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 69 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier #	3	
	K/A #	G2.2.6	IR 3.0

Knowledge of the process for making changes to procedures

Proposed Question:

In accordance with RBNP-001, DEVELOPMENT AND CONTROL OF RBS PROCEDURES, a Comment PAR, may be used for which of the following?

A. To make minor changes that result in a change of intent.

B. To correct typographical errors to a procedure that is being implemented.

C. To make suggestions for future procedure improvements.

D. To change acceptance criteria.

Proposed Answer: C

Explanation
A. A procedure “Revision” is required for changes that result in a change of intent.
B. This requires an Editorial change.
C. Correct - This choice describes a “Comment”.
D. A change in acceptance criteria also results in a change of intent, so would require a “Revision”

Technical Reference(s): RBNP-001, Development and Control of Procedures, Rev 35 p. 19 of 43

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-0202, Obj 1

Question Source: Bank # RBS-OPS-1667

Question History: Last NRC Exam RBS April 2010

Cognitive Level: Memory or Fundamental Knowledge ☑ 2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 70 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier #	3	
	K/A #	G2.2.14	IR 3.9

Knowledge of the process for controlling equipment configuration or status

Proposed Question:

A decon technician has called the RO requesting permission to open an MWS supply valve to obtain water for deconning purposes. What action is required?

A. The RO can authorize the manipulation and logging is not required as long as the tech calls back before the end of shift stating that the valve is closed

B. The OSM/CRS must authorize the manipulation and it must be logged in the control room log or manipulated device log book.

C. The RO can authorize the manipulation and it must be logged in the control room log or manipulated device log book

D. The OSM/CRS must authorize the manipulation and logging is not required as long as the tech calls back before the end of shift stating that the valve is closed.

Proposed Answer: B

Explanation
A. See B
B. Correct OSP-0014 requires OSM/CRS approval and logging for configuration control purposes when manipulating plant components outside of procedures
C. See B
D. See B

Technical Reference(s): OSP-0014, Administrative Control of Equipment, Rev 304 p. 6 of 16

Proposed references to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 71 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier #	3	
	K/A #	G2.2.35	IR 3.6

Ability to determine Technical Specification Mode of Operation

Proposed Question:

The following plant conditions exist:

- Reactor Coolant temperature is 115˚F
- The mode switch position is Shutdown
- Refueling operations have been completed and the last head bolt has been tensioned

What mode of operation is the plant in?

A. Mode 2 Startup

B. Mode 3 Hot Shutdown

C. Mode 4 Cold Shutdown

D. Mode 5 Refueling

Proposed Answer: C

Explanation
A. See C
B. See C
C. Correct – Per Tech Spec Definitions Table 1.1-1
D. See C

Technical Reference(s): Tech Spec Definitions Table 1.1-1

Proposed references to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 2

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 72 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	3		
	K/A #	G2.3.11		IR 3.8

Ability to control radiation releases

Proposed Question:

Due to a steam leak, the Main Steam Tunnel area temperatures have caused automatic isolations to occur as designed. Because of the location of the leak, no LOCA signals have been generated by RPV level or DW pressure. An ALERT has been declared due to offsite release rate.

Which one of the following will reduce the UNMONITORED release rate?

A. Shutdown the Turbine Building Ventilation System, if operating.

B. Shutdown the Fuel Building Ventilation System, if operating.

C. Start the Turbine Building Ventilation System, if not operating.

D. Start the Fuel Building Charcoal Ventilation System, if not operating.

Proposed Answer: C

Explanation
A. This action would raise the unmonitored release rate
B. The fuel building normal and charcoal filtration systems are both monitored; this would not affect unmonitored release
C. Correct – IAW EOP-0003
D. See B

Technical Reference(s): EOP-0003 p. B-10-3

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-515, Obj 6

Question Source: Bank # RBS-NRC-797

Question History: Last NRC Exam RBS 2003 NRC

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 73 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier #	3	
	K/A #	G2.3.13	IR 3.4

Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc

Proposed Question:

For which of the following evolutions is the licensed operator in the control room procedurally required to notify Radiation Protection prior to performance?

A. Suppression Pool reject to Radwaste with Residual Heat Removal (RHR)

B. Placing Heater Drain pumps in the PUMP FORWARD mode

C. Startup of Circulating Water Blowdown

D. Reactor Core Isolation Cooling (RCIC) slow roll startup

Proposed Answer: D

Explanation:
A. See D. (SOP-0031)
B. See D. (SOP-0010)
C. See D. (SOP-0006)
D. Although notifying RP prior to any of the 4 choices demonstrate good teamwork, only RCIC slow roll startup specifically requires notification per SOP-0035.

Technical Reference(s): SOP-0035, RCIC, Rev 47 p. 16 of 76

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0209 Obj 10c

Question Source: Bank # RBS Oct 2012 Audit Q#72

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 4 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 74 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	3		
	K/A #	G2.4.12		IR 4.0

Knowledge of general operating crew responsibilities during emergency operations

Proposed Question:

To which of the following Emergency Response Organization facilities is the Fire Brigade assigned to when not performing fire fighting functions?

A. Operations Support Center (OSC)

B. Technical Support Center (TSC)

C. Emergency Operating Facility (EOF)

D. Joint Information Center (JIC)

Proposed Answer: A

Explanation
A. Correct – per EIP-2-016.
B. C. and D. See A

Technical Reference(s): EIP-2-016, Operations Support Center, Rev 29 p. 5 of 31

Proposed references to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # RBS Oct 2008 Audit Q#75

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 75 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier #	3	
	K/A #	G2.4.26	IR 3.1

Knowledge of facility protection requirements, including fire brigade and portable fire fighting equipment usage

Proposed Question:

According to EN-OP-115, Conduct of Operations, a Fire Brigade of at least _(1)_ members shall be maintained on site at all times; the fire brigade members shall NOT include the _(2)_.

A. (1) Four ; (2) Shift Technical Advisor (STA)

B. (1) Five ; (2) Duty Manager

C. (1) Five ; (2) Shift Technical Advisor (STA)

D. (1) Four ; (2) Duty Manager

Proposed Answer: C

Explanation
A. EN-OP-115 requires a minimum of 5 Fire Brigade Members
B. Part 1 is correct; The Duty Manager is not designated as being precluded by procedure
C. Correct – The fire brigade shall not include the OSM, CRS, STA, ATC, and 1 NEO
D. EN-OP-115 requires a minimum of 5 Fire Brigade Members

Technical Reference(s): EN-OP-115, Conduct of Operations, Rev 15 p. 78 of 89

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-0206, Obj 7

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments: Similar to Bank Question #RBS-OPS-3382

QUESTION 76 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	1	Group #	1
	K/A #	295004 G2.2.44		IR 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 (Partial or Complete Loss of DC)

Proposed Question:

During normal plant operation, several alarms are received on H13-P601 Insert 16, including:

- DIV III 125VDC SYSTEM TROUBLE
- DIV III BATTERY CHARGER TROUBLE

In addition to these alarms, the (2) 125VDC white power available lights on H13-P601 High Pressure Core Spray area are extinguished.

Which of the following procedures should the CRS enter to mitigate this condition?

A. SOP-0030, HIGH PRESSURE CORE SPRAY SYSTEM

B. SOP-0049, 125 VDC SYSTEM

C. AOP-0014, LOSS OF 125 VDC

D. AOP-0042, LOSS OF INSTRUMENT BUS

Proposed Answer: C.

Explanation:
A. This procedure is for normal operation of the High Pressure Core Spray System. The stem indicates abnormal conditions.
B. This procedure is for normal operation of the 125 VDC System. The stem indicates abnormal conditions.
C. Correct- The conditions in the stem indicate that 125 VDC to the HPCS has been lost. The correct procedure for this condition is AOP-0014.
D. This procedure deals with the loss of various uninterruptible power supplies. These UPS systems are associated with Div 1, Div 2 and non-safety related components.

Technical Reference(s): AOP-0014

Proposed references to be provided to applicants during examination: None

Learning Objective:	RLP-OPS-0532 Obj 2	
Question Source:	Bank# April 2010 NRC Exam	
Question History:	Last NRC Exam April 2010 Q#77	
Cognitive Level:	Memory or Fundamental Knowledge ☐	Comprehension or Analysis ☑ 3
10 CFR Part 55 Content:	55.43.b.5	
Comments:		

QUESTION 77 Rev 1

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	1	Group #	1
	K/A #	295006 G2.4.34		IR 4.2

Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects. (SCRAM)

Proposed Question:

The plant has experienced an ATWS. The following conditions exist at the 680 panel:

- The Mode Switch is in Shutdown
- ARI has been initiated
- Approximately 50% of the withdrawn rods fully inserted
- All eight white scram solenoid lights are extinguished
- The SDV Vent and Drain Valve position green lights are on and red lights are off
- Annunciator 680-05A-C08, SCRAM Pilot Valve Air Header Low Pressure NOT lit

In accordance with EOP Enclosure 26, Control Rod Insertion Method Determination, which procedure should be directed to allow insertion of control rods?

A. EOP Enclosure 10, De-Energizing SCRAM Solenoids

B. EOP Enclosure 11, Venting SCRAM Air Header

C. EOP Enclosure 13, Opening Individual SCRAM Test Switches

D. EOP Enclosure 17, Venting CRD Overpiston Volumes

Proposed Answer: B

A. The stem conditions are indicative the failure of the scram air header to vent; scram solenoids are already de-energized.
B. Correct – this enclosure will vent the scram air header and allow the rods to be inserted.
C. and D. These two enclosures are used after the scram air header is depressurized.

Technical Reference(s): EOP-0001A RPV Control - ATWS;
EOP-0005 Enclosure 26, Control Rod Insertion Method Determination

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

Learning Objective: RLP-HLO-0513, Obj 5

Question Source: Modified Bank# RBS-OPS-06265

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3

10 CFR Part 55 Content: 55.43.b.5

Comments: Original KA was G2.4.2, re-rolled (drew G2.4.34) ; The original KA was not conducive to writing an SRO level question.

QUESTION 78 Rev 1

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	1	Group #	1
	K/A #	295021 AA2.04		IR 3.6

Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING : Reactor water temperature

Proposed Question:

The plant is Shutdown in Mode 4 following 400 days of continuous operation.

A spurious Group 3 isolation occurs and the logic cannot be reset.
Reactor water temperature rises to 205˚F

What is the time limit for notification to the NRC?

A. Immediate notification (within 1 hour)

B. 4 hour notification

C. 8 hour notification

D. 24 hour notification

Proposed Answer: A
Explanation
A Correct – the conditions are indicative of a NOUE; a group 3 isolation would cause a loss of shutdown cooling; temperature rose enough for a mode change; notification is required immediately
B 4 hours is a valid notification time but not for this condition.
C. 8 hours would be for an valid ESF actuation/isolation
D. 24 hours is a valid notification time but not for this condition.
Technical Reference(s): EIP-2-001, Classification of Emergencies, Rev 24, Attach 10
EIP-2-002, Classification Actions, Rev 31, p. 7 of 20

Proposed references to be provided to applicants during examination: EIP-2-001, Attach 10 only
Learning Objective: (As available)
Question Source: Bank# GGNS 2013 AUDIT Q# 79
Question History: Last NRC Exam NA
Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 4
10 CFR Part 55 Content: 55.43.b.5
Comments: Replaced question to make this SRO level

QUESTION 79 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	1	Group #	1
	K/A #	295028 G2.4.21		IR 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 Temperature)

Proposed Question:

A transient occurred that caused the Control Room Team to enter the Alternate Level Control Leg of EOP-1. While executing the ALC steps, a large break LOCA, ALL RPV level instruments indicated off-scale low.

Five seconds later, the Fuel Zone Level instruments returned on scale. The following conditions now exist:

- Containment temperature 91°F (at EL 119 ft)
- Drywell temperature 285°F (at EL 145 ft)
- RPV Pressure 10 psig
- Fuel Zone Level indication -290 inches, rising and falling rapidly

With these conditions, Fuel Zone Level indication ____ be used to determine RPV level ; the CRS should direct ____.

A. CANNOT ; Emergency Depressurizing per EOP-1

B. CANNOT ; exiting EOP-1 and entering EOP-4

C. CAN ; restoring and maintaining RPV level above -162 inches per EOP-1

D. CAN ; exiting the ALC leg and entering the Steam cooling leg of EOP-1

Proposed Answer: B

Explanation

A. Incorrect per override RC-2 of EOP-1

B Correct per override RC-2 EOP-4 must be entered

C, D - Incorrect, FZ level indication is unreliable due to boiling in the instrument line run

Technical Reference(s): EOP-1, RPV Control, Rev 26

Proposed references to be provided to applicants during examination: Caution 1 of EOP-1

Learning Objective: (As available)

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3

10 CFR Part 55 Content: 55.43.b._

Comments:

QUESTION 80 Rev 1

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	1	Group #	1
	K/A #	295031 EA2.04		IR 4.8

Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL: Adequate core cooling

Proposed Question:

The following conditions exist:

- Reactor Pressure is 943 psig
- RHR pumps A and C are lined up for injection
- HPCS injection has commenced
- Reactor water level is -190” and slowly lowering

Which of the following is correct concerning cooling of the core, and
Which procedural action should the SRO direct next?

A. Adequate core cooling exists and Steam Cooling is required.

B. Adequate core cooling exists and Spray Cooling is required.

C. Adequate core cooling does not exist and Emergency Depressurization is required.

D. Adequate core cooling does not exist and Containment Flooding is required.

Proposed Answer: C

Explanation
A. -200 is the water level for steam cooling without injection
B. -211 is the water level for spray cooling, at 943 psig HPCS will not develop 5000 gpm
C. Correct - adequate core cooling does not exist, ED is necessary to allow low pressure ECCS to inject
D. Containment flooding would be required if adequate core cooling could not be restored and maintained.

Technical Reference(s): EOP Bases
Proposed references to be provided to applicants during examination: None
Learning Objective: (As available)
Question Source: New Question History: Last NRC Exam NA
Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3
10 CFR Part 55 Content: 55.43.b.5
Comments: Revised question to include procedural choice to make this SRO level

QUESTION 81 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	1	Group #	1
	K/A #	600000 AA2.16		IR 3.5

Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: Vital equipment and control systems to be maintained and operated during a fire

Proposed Question:

The control room has received a report of a fire on the east side crescent area of the auxiliary building. The reactor has been shut down and the current conditions are:

- Reactor Recirc Pumps both in slow speed
- Reactor pressure 125 psig and lowering
- RPV level 36” and steady

What direction should the CRS give concerning Shutdown Cooling (SDC) operation?

A. Place RHR-B in normal SDC per SOP-0031, RHR System

B. Place RHR-A in normal SDC per SOP-0031, RHR System

C. Place RHR-C in alternate SDC per AOP-0020, Alternate Decay Heat Removal

D. Place RHR-B in alternate SDC per AOP-0020, Alternate Decay Heat Removal

Proposed Answer: B

Explanation
A. C. and D. Per AOP-0052, Attachment 1 Div 2 equipment is **not** available
B. Correct- Per AOP-0052, Attachment 1 this equipment is available

Technical Reference(s): AOP-0052, Fire Outside the MCR, Rev 25 Attachment 1

Proposed references to be provided to applicants during examination: Attachment 1 of AOP-52

Learning Objective: (As available)

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 2

10 CFR Part 55 Content: 55.43.b.5

Comments:

QUESTION 82 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	1	Group #	1
	K/A #	700000 G2.2.37		IR 4.6

Ability to determine operability and/or availability of safety related equipment (Generator Voltage and Electric Grid Disturbances)

Proposed Question:

The plant is operating at rated power.

A grid disturbance occurs that causes:
Alarm P808-86A-H09, PFD STA XFMR 1RTX-XSR1C VOLTAGE LOW
ENS-ACB06 ENS-SWG1A PFD Supply BKR trips on under voltage
Div I DG starts and ties to the bus

An attempt to re-close ENS-ACB06 was unsuccessful
A CR has been initiated concerning the status of the ENS-ACB06 ENS-SWG1A PFD Supply BKR.

What Operability Code should be assigned to the initial screening of this CR in PCRS?
AND
Which T.S., if any, should be entered?

A. INOPERABLE and T.S. 3.8.1. condition A

B. OPERABLE and no T.S. entry

C. INOPERABLE-OP EVAL and T.S. 3.8.1. condition A

D. OPERABLE-COMP MEAS and no T.S. entry

Proposed Answer: A

Technical Reference(s): AOP-0064, Degraded Grid, Rev 6
ENS-DC-199, Offsite Power Supply Design Requirements, Attach 9.3

Proposed references to be provided to applicants during examination: None
Learning Objective: (As available)
Question Source: New Question History: Last NRC Exam NA
Cognitive Level: Memory or Fundamental Knowledge ☑ Comprehension or Analysis ☐
10 CFR Part 55 Content: 55.43.b.2
Comments:

QUESTION 83 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	1	Group #	2
	K/A #	295015 AA2.02		IR 4.2

Ability to determine and/or interpret the following as they apply to INCOMPLETE SCRAM : Control rod position

Proposed Question:

Following a high drywell differential pressure scram signal, the following conditions exist:

- 142 control rods indicate Full-In
- 3 control rods are not full in and are at various positions
- Reactor power indicates 0%
- SRMs are inserted and count rates are lowering

(1) What is the status of the reactor and (2) what procedure does the CRS direct?

A. The reactor will remain shutdown without boron; Enter EOP-1, RPV Control

B. It can NOT be determined that the reactor will remain shutdown under all conditions without boron; Enter EOP-1, RPV Control and transition to EOP-1A, RPV Control, ATWS.

C. It can NOT be determined that the reactor will remain shutdown under all conditions without boron; Enter EOP-1, RPV Control, EOP-1A is NOT applicable with reactor power less than 5%.

D. The reactor will remain shutdown without boron; Enter EOP-1, RPV Control and transition to EOP-1A, RPV Control, ATWS

Proposed Answer: B

Explanation

A. and D. With more than one rod out, it can not be determined that the reactor will remain shutdown. IAW EOP Bases, "For the current fuel design and core load, the MSBWP is all control rods fully inserted."

B. Correct - With more than one rod out, it can not be determined that the reactor will remain shutdown under all conditions without boron; A high drywell differential pressure scram signal is an EOP-0001 entry condition, transition to EOP-1A is required due to override step RC-2 in EOP-1.

C. First part correct, however, EOP-1 override RC-2 directs exiting EOP-1 and entering EOP-1A

Technical Reference(s): EOP Bases p. B-6-5
Proposed references to be provided to applicants during examination: None
Learning Objective: (As available)
Question Source: Modified Bank # 2008 Audit Q#25
Question History: Last NRC Exam NA
Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 2
10 CFR Part 55 Content: 55.43.b.5
Comments:

QUESTION 84 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	1	Group #	2
	K/A #	295029 G2.2.25	IR 4.2	

Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits (High Suppression Pool Wtr Lvl)

Proposed Question:

Technical Specifications (TS) require suppression pool water level to be ≤ 20 ft 0 inches. What is the TS basis for this upper limit?

A. Above 20 ft 0 inches, excessive pressure could build up in the drywell before the drywell horizontal vents are cleared during a DBA LOCA.

B. Water levels above 20 ft 0 inches would cause low pressure ECCS systems to operate outside of their design NPSH values.

C. Water levels above 20 ft 0 inches would overflow the drywell weir wall and cause inaccurate drywell leakage indications due to excessive pedestal sump discharge flow.

D. Above 20 ft 0 inches, SRV tail pipe pressure could build up too high during SRV operation and damage the tail pipes.

Proposed Answer: D

Explanation

A. A higher DW pressure buildup would occur as suppression pool level rises but this is within design analysis and “excessive” pressure buildup would not occur. This condition is not mentioned in the bases.
B. Levels above 20 ft would raise NPSH of the low pressure ECCS pumps but would not cause them to operate outside of design parameters.
C. The weir wall would overflow at 21 ft 3 inches and cause the pedestal sump to cycle but this is not mentioned as a basis for the upper limit
D. Correct - TS 3.6.2.2 Bases Statement; If the suppression pool water level is too high, it could result in excessive clearing loads from S/RV discharges”.

Technical Reference(s): Tech Spec 3.6.2.2 and Bases

Proposed references to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.43.b.2

Comments:

QUESTION 85 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	1	Group #	2
	K/A #	500000 EA2.03		IR 3.8

Ability to determine and / or interpret the following as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS: Combustible limits for drywell

Proposed Question:

The plant has experienced a LOCA. The following conditions exist:

- RPV pressure is 250 psig
- RPV level is +10 inches and slowly rising
- Containment pressure is 1.5 psig
- Drywell temperature is 200°F
- Drywell hydrogen concentration is 2 percent
- Containment hydrogen concentration is 2 percent

Based on this, the SRO should ________.

A. Enter EOP-2 at Hydrogen Control and Verify start of Div. I and Div. II H_2 monitoring systems

B. Continue EOP-1 at Pressure Control and initiate shutdown cooling

C. Enter EOP-2 at Hydrogen Control and operate all hydrogen igniters

D. Exit all EOPs and enter the SAPs

Proposed Answer: C

Explanation
With hydrogen concentration above 1.5%, all igniters should be operated
A – Hydrogen mixing not authorized in EOP-2
B – Shutdown cooling interlock not clear
D – Drywell / Containment H_2 concentration is not high enough

Technical Reference(s): EOP Bases p. B-8-26 & 27

Proposed references to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.43.b.2

Comments:

QUESTION 86 Rev 0

Examination Outline Cross-Reference:	Level	RO ☐	SRO ☑
	Tier # 2	Group # 1	
	K/A # 205000 A2.03	IR 3.2	

Ability to (a) predict the impacts of the following on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A.C. failure

Proposed Question:

The plant is shutdown, in Mode 4 with RHR A in SDC mode and no Recirc Pumps running. A leak in the drywell causes RPV Level to lower. An operator begins to line up RHR A for Injection mode.
When E12-F006A, RHR Pump A SDC Suction Valve begins to close, electrical power is lost to the valve. The operator closes E12-F008 and -F009, RHR Shutdown Cooling Outboard and Inboard Isolation valves.

The CRS will direct ___ ; and RHR A Shutdown Cooling is ___.

A. Manually closing E12-F006A ;
Operable

B. Aligning RHR Injection through E12-F0053A, RHR A SDC Injection Valve ;
Operable

C. Manually closing E12-F006A ;
Inoperable

D. Aligning RHR Injection through E12-F0053A, RHR A SDC Injection Valve ;
Inoperable

Proposed Answer: A

Explanation
A. Correct – The RHR Suppression Pool Injection Suction Valve (F004) cannot be opened with the F006 valve not full closed, but the F006 valve can be manually closed thereby allowing the F004 valve to be opened; Without opening closing the F006 and opening the F004, there is no suction path available to allow for injecting in any path.
RHR is operable because the LCO statement reads: “An OPERABLE RHR shutdown cooling subsystem consists of one OPERABLE RHR pump, two heat exchangers in series, and the associated piping and valves. Each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat.”

Technical Reference(s): AOP-0051, Loss of Decay Heat Removal, Rev 312
Tech Spec Bases 3.4.10 p. 3.4-48 and 49

Proposed references to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3

10 CFR Part 55 Content: 55.43.b.2 Comments:

QUESTION 87 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	2	Group #	1
	K/A #	212000 G2.2.12	IR 4.1	

Knowledge of surveillance procedures (RPS)

Proposed Question:

The plant is operating at 90% power. STP-508-0201, Manual Scram Channel Functional Test and LSFT is being performed. The last division is being restored.
When the operator attempted to reset the half-scram, status light H13-P680-11E1, RPS DIV 4 SCRAM SOV VALVES OPEN – CR 4B white light is **NOT** on.
All other SOV status lights are lit.

A loss of RPS Bus B occurs; what is the expected plant response and what AOP(s) should be entered?

A. All of the control rods will scram ;
AOP-0001, Reactor Scram only

B. Approximately ¼ of the control rods will scram ;
AOP-0001, Reactor Scram only

C. Approximately ¼ of the control rods will scram ;
AOP-0001, Reactor Scram, and AOP-0010, Loss of RPS Bus

D. None of the control rods will scram ;
AOP-0010, Loss of RPS Bus only

Proposed Answer: D
Explanation
A. See D
B. See D
C. See D
D. Correct – The loss of RPS B will only cause a half-scram signal; The Div 4 SOV is powered by RPS-B, so a loss of RPS B would have de-energized the CR4B which is already de-energized;
IF the transient had been a loss of RPS A, it would have caused approximately ¼ of the rods to scram due to de-energizing the second SOV in Div 4 and actions would need to be taken to insert a manual scram and enter AOP-1.
AOP-0010 must be entered in order to restore power to the bus and reset isolation signals.

Technical Reference(s): R-STM-508, RPS, Rev 5 p. 23 of 59; STP-508-0201, Rev 14 pp 19-20 of 26
Proposed references to be provided to applicants during examination: None
Learning Objective: RLP-STM-0508, Obj 8
Question Source: Modified Bank # RBS-NRC-937
Question History: Last NRC Exam Sept 2004 NRC
Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3
10 CFR Part 55 Content: 55.43.b.5
Comments:

QUESTION 88 Rev 1

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	2	Group #	1
	K/A #	215005 A2.02		IR 3.7

Ability to (a) predict the impacts of the following on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Upscale or downscale trips

Proposed Question:

GOP-0001 Plant Startup requires verification that all APRMs are >5% prior to taking the mode switch to RUN. This step was signed and then a shift turnover occurred.

The mode switch is taken to run. APRMs B & C are both reading <5%.
All other APRMs are operable and are reading >5%.

The result of this condition is a ___; the CRS will direct ___.

A. reactor scram ; actions according to EOP-0001, RPV Control.

B. Control Rod Withdrawal Block ; bypassing APRMs B & C to clear the rod block per EN-OP-115, Conduct of Operations

C. Control Rod Withdrawal Block ; be in Mode 3 within 12 hours in accordance with Tech Specs.

D. half scram ; transfer the Mode Switch back to START & HOT STBY per GOP-0002, Power Decrease/Plant Shutdown.

Proposed Answer: B

Explanation

A. There will be a rod withdraw block not a scram.

B. Correct - With mode switch in run and APRMs <5%, a rod withdraw block will occur; not a scram. With all other APRMs operable, B & C can be bypassed to clear the rod block. Tech Specs allow bypassing one channel from each trip system. Rods can be withdrawn to bring B & C above 5%.

C. First Part is correct, but this Tech Specs required action is for when one or more required channels are inoperable; Tech Specs only requires 3 channels per trip system (which in the given condition, is met)

D. A half scram is not present for two APRMs < 5%.

Technical Reference(s): SOP-0074, Neutron Monitoring, Rev 306 p.20 of 74
R-STM-0503, RPS, Rev 7, pp. 63,68,106 of 112

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0503, Obj 26,28

Question Source: Bank # 2008 RBS AUDIT Q#89

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3

10 CFR Part 55 Content: 55.43.b.5

Comments: Question revised to force procedure selection to make it SRO level

QUESTION 89 Rev 0

Examination Outline Cross-Reference:	Level	RO ☐	SRO ☑
	Tier # 2	Group #	1
	K/A # 239002 A2.06		IR 4.3

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: Reactor high pressure

Proposed Question:

While operating at 100% power, a loss of condenser vacuum occurs..
After a reactor scram, the following plant conditions exist:

- Reactor pressure spiked to 1140 psig
- Main Turbine has tripped
- Condenser vacuum 0”

(1) What is the current pressure control band for SRV 51D?
(2) What direction does the CRS give to control Reactor pressure?

A. (1) 1063 psig to 956 psig
   (2) Stabilize pressure with SRVs, then establish a band of 500 to 1090 pounds in accordance with OSP-0053, Emergency and Transient Response.

B. (1) 1063 psig to 956 psig
   (2) Install Enclosure 9, Defeating MSIV Isolation Interlocks, and open MSIVs

C. (1) 1103 psig to 966 psig
   (2) Stabilize pressure with SRVs, then establish a band of 500 to 1090 pounds in accordance with OSP-0053, Emergency and Transient Response.

D. (1) 1103 psig to 966 psig
   (2) Install Enclosure 9, Defeating MSIV Isolation Interlocks, and open MSIVs

Proposed Answer: A

Explanation
A. Correct - set points for 51D ; SRVs are the only available system due to low vacuum
B. First part is correct; Encl 9 is only allowed for alternate pressure control in this condition
C.. and D. First part is incorrect - these are the LLS set points for SRV 51C

Technical Reference(s): R-STM-0109, Main Steam, Rev 13 pp. 36,37,60 of 95
EOP-1, RPV Control

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0109, Obj 22

Question Source: New   Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐   Comprehension or Analysis ☑ 3

10 CFR Part 55 Content: 55.43.b.5   Comments:

QUESTION 90 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	2	Group #	1
	K/A #	262001 G2.2.25		IR 4.2

Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits (AC Electrical Distribution)

Proposed Question:

Due to a loss of EJS*LDC2A, you have entered TS 3.8.9 condition A (One or more Division I or II AC electrical power distribution subsystems inoperable). The required action is to Restore Division I and II AC electrical power distribution subsystems to OPERABLE status.

The 8 hour completion time for this LCO action is based on:

A. the unit is more vulnerable to a complete loss of AC power

B. the plant is significantly more vulnerable to a complete loss of all uninterruptible power.

C. the plant is significantly more vulnerable to a complete loss of DC power.

D. the plant is significantly more vulnerable to a complete loss of D/G power.

Proposed Answer: A

Explanation
A. Correct- copied from the bases for condition A
B. This applies to condition B
C. This applies to condition C
D. This applies to TS 3.8.1 condition A

Technical Reference(s): Tech Spec Bases 3.8.9 p. B 3.8-81

Proposed references to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 4 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.43.b.2

Comments:

QUESTION 91 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	2	Group #	2
	K/A #	201005 G2.1.23		IR 4.4

Ability to perform specific system and integrated plant procedures during all modes of plant operation (RCIS)

Proposed Question:

GOP-001, Plant Startup is in progress currently in Section C (Approach to Critical: Instructions). Pulling control rods began two hours ago. The dedicated reactor operator has just informed you that a single control rod was pulled out of sequence (4 notches).

What will you direct the reactor operator to do?

A. Manually scram the reactor and enter AOP-0001, Reactor Scram

B. Perform a MON calculation to determine margin to Preconditioning Envelope per GOP-0001, Plant Startup.

C. Select and continuously insert the control rod to position 00 per AOP-0061 Mispositioned Control Rod.

D. Suspend all rod motion and enter AOP-0061, Mispositioned Control Rod.

Proposed Answer: D

Explanation
A. This action is directed in AOP-61 when more than one rod has scrammed or been fully inserted.
B. This would be a required action if a control rod is found in an unintended position or a single rod scram were to have occurred.
C. This is the required action for a drifting rod.
D. Correct – GOP-1 Caution for step 11 directs IF an out of sequence rod should be withdrawn, THEN suspend rod motion, notify the OSM, and reference AOP-0061. AOP-61, Step 5.2 directs returning the rod to its required position and notify Reactor engineering.

Technical Reference(s): GOP-0001, Plant Startup, Rev 82 p. 34 of 103
AOP-0061, Mispositioned Control Rod, Rev 7 p. 5 of 8

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-0711, Obj 4

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 2

10 CFR Part 55 Content: 55.43.b.5

Comments:

QUESTION 92 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	2	Group #	2
	K/A #	204000 A2.14		IR 3.2

Ability to (a) predict the impacts of the following on the REACTOR WATER CLEANUP SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: System High Temperature

Proposed Question:

A partial loss of Reactor Plant Component Cooling Water (CCP) has resulted in reduced cooling to the Reactor Water Cleanup System (RWCU).

How will the RWCU system respond, and what actions are required to mitigate that response?

A. G33-F001, RWCU Pumps Inbd Suction Valve will close when NRHX outlet temperature ≥140°F ;
Transfer resin to the holding tank until CCP restored per SOP-0090.

B. G33-F001, RWCU Pumps Inbd Suction Valve will close when NRHX outlet temperature ≥140°F ;
Secure RWCU per SOP-0090.

C. G33-F004, RWCU Pumps Outbd Suction Valve will close when NRHX outlet temperature ≥140°F ;
Transfer resin to the holding tank until CCP restored per SOP-0090.

D. G33-F004, RWCU Pumps Outbd Suction Valve will close when NRHX outlet temperature ≥140°F ;
Secure RWCU per SOP-0090.

Proposed Answer: D

Explanation
A. See D
B. See D
C. Part 1 is correct; See D for Part 2
D. Correct – G33-F001 and F004 are interlocked to close on SLC injection, but only F004 is interlocked to close on Filter/Demin inlet temperature; The ARP requires either reducing flow through demins (Alarm at 130°F) or securing the system

Technical Reference(s): ARP-680-01-B01 & -C01; R-STM-0601, RWCU, Rev 8 pp. 15-16, 25-26 of 51
Proposed references to be provided to applicants during examination: None
Learning Objective: RLP-STM-0601, Obj 7
Question Source: New Question History: Last NRC Exam NA
Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 2
10 CFR Part 55 Content: 55.43.b.5
Comments: *Original KA was A2.02, re-rolled (drew 2.14). RBS does not have Low Press RWCU.*

QUESTION 93 Rev 0

Examination Outline Cross-Reference:	Level	RO ☐	SRO ☑
	Tier # 2	Group # 2	
	K/A # 290001 A2.04	IR 3.7	

Ability to (a) predict the impacts of the following on the SECONDARY CONTAINMENT; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High airborne radiation

Proposed Question:

A RWCU leak has occurred in the RWCU pump room and control room attempts to isolate the leak have failed. Work management is establishing a team to enter the Aux Building to attempt to drive the isolation valves closed from the motor control centers.

Area temperature monitors show the following readings:

RWCU PUMP ROOM	210˚F	RHR EQUIPMENT AREA A	90˚F
RCIC EQUIPMENT AREA	150˚F	RHR EQUIPMENT AREA B	85˚F

The Digital Radiation Monitoring System (DRMS) shows the following indications:

RMS-RE110, AUX BLDG VENT - Alarm (RED)
RMS-RE213, RHR EQUIPMENT ROOM A - Normal (Green)
RMS-RE214, RHR EQUIPMENT ROOM B - Normal (Green)
RMS-RE215, RHR EQUIPMENT ROOM C - ALERT (Yellow)
RMS-RE217, HPCS PENETRATION AREA - Normal (Green)
RMS-RE218, LPCS PENETRATION AREA - Normal (Green)
RMS-RE219 RCIC EQUIPMENT ROOM - ALERT (Yellow)

What impact could these radiation levels have and what procedural actions should be directed

A. High Radiation levels will have no impact they only provide indication that water from a primary system may be discharging into secondary containment;
the RPV should be emergency depressurized.

B. High radiation levels may preclude personnel access and lead to equipment failure;
the Aux building should be isolated per OSP-0053 Attachment 21 hard card.

C. High Radiation levels may preclude personnel access and lead to equipment failure;
the RPV should be emergency depressurized.

D. High Radiation levels will have no impact they only provide indication that water from a primary system may be discharging into secondary containment;
the Aux building should be isolated per OSP-0053 Attachment 21 hard card.

Proposed Answer: B

Explanation

A. While high radiation levels are indication that there may be a primary system discharging into secondary containment there is still the potential for radiation levels to preclude access and cause equipment failure. Criteria for ED have not been met.

B. Correct – High radiation levels have the potential to preclude access and cause equipment failure, with RMS-RE110 in alarm the auxiliary building should be isolated and exhaust processed through the standby gas system to minimize release to the public.

C. Criteria for ED have not been met.

D. While high radiation levels are indication that there may be a primary system discharging into secondary containment there is still the potential for radiation levels to preclude access and cause equipment failure

Technical Reference(s): EOP BASES

Proposed references to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 4

10 CFR Part 55 Content: 55.43.b.5

Comments:

QUESTION 94 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	3		
	K/A #	G2.1.20		IR 4.6

Ability to interpret and execute procedure steps

Proposed Question:

Refueling is in progress. Reactor Recirculation and Reactor Water Cleanup System are secured for system maintenance.

How is TR 3.4.13, Chemistry satisfied for reactor coolant conductivity during this portion of a refueling outage?

A. The conductivity of the reactor coolant is recorded continuously.

B. An in-line conductivity measurement is obtained every 4 hours.

C. An in-line conductivity measurement is obtained every 24 hours.

D. Not required to be met in this mode.

Proposed Answer: C.

Explanation:
A. With both Reactor Recirculation and RWCU secured, continuous monitoring is not available.
B. This would be required in Modes 1, 2, & 3.
C. Correct – Continuous monitoring is not available so an inline conductivity measurement must be obtained every 24 hours.
D. Required at all times.

Technical Reference(s): TR 3.4.13

Proposed references to be provided to applicants during examination: TR 3.4.13

Learning Objective: RLP-STM-0601 Obj. 9

Question Source: Bank # Dec 2010 NRC Q#95

Question History: Last NRC Exam Dec 2010 NRC

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 2

10 CFR Part 55 Content: 55.43.b.2

Comments:

QUESTION 95 Rev 0

Examination Outline Cross-Reference:	Level	RO ☐	SRO ☑
	Tier #	3	
	K/A #	G2.1.36	IR 4.1

Knowledge of procedures and limitations involved in core alterations

Proposed Question:

During a refuel outage, the Outage Control Center notifies the main control room that core alterations are scheduled to commence next shift.

Select the procedure below which the CRS utilizes to ensure that <u>all</u> requirements for core alterations have been met.

A. STP-000-0005, DAILY REFUELING LOGS

B. GMP-0102, REACTOR VESSEL DISASSEMBLY

C. FHP-0001, CONTROL OF FUEL HANDLING AND REFUELING OPERATIONS

D. FHP-0003, REFUEL PLATFORM OPERATION

Proposed Answer: C.

Explanation:
A. See C.
B. See C.
C. Correct – FHP-0001 Attachment 2 contains a list of Applicalbe Mode 5 Tech Specs which includes (section 2) a list of requirements to commence core alterations.
D. See C.

Technical Reference(s): FHP-0001, Control of Fuel Handling & Refueling Ops, Rev 35 pp. 4, 35 of 40

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0055 Obj 6 & 9

Question Source: Bank # Nov 2012 NRC Q#95

Question History: Last NRC Exam Nov 2012 NRC

Cognitive Level: Memory or Fundamental Knowledge ☑3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.43.b.7

Comments: Appeared on one of last 2 NRC exams (2 of 2):

QUESTION 96 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	3		
	K/A #	G2.2.11		IR 3.3

Knowledge of the process for controlling temporary design changes

Proposed Question:

An event has occurred which presents an imminent threat to the safety of the plant. In order to mitigate the event a temporary modification must be installed as directed by the Shift Manager. Whose concurrence is required to implement the emergency temporary modification?

A. Another SRO

B. The RBS Vice President (VP)

C. The General Manager Plant Operations (GMPO)

D. The Engineering Director or his designee

Proposed Answer: D

Explanation
A. See D
B. See D
C. See D
D. Correct – Per EN-DC-136, the concurrence of the Engineering Director, or designee is required

Technical Reference(s): EN-DC-136, Temporary Modifications, Rev 10 Section 5.3

Proposed references to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.43.b.3

Comments:

QUESTION 97 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	3		
	K/A #	G2.2.38		IR 4.5

Knowledge of conditions and limitations in the facility license

Proposed Question:

Which of the Limiting Conditions for Operation (LCOs) listed below provide guidance concerning when changes in mode or other specified conditions are allowed when an LCO is not met?

A. LCO 3.0.2

B. LCO 3.0.3

C. LCO 3.0.4

D. LCO 3.0.6

Proposed Answer: C.

Explanation:
A. 3.0.2 establishes that upon discovery to meet an LCO, the associated ACTIONS must be met.
B. 3.0.3. establishes the actions that must be met when an LCO is not met and the actions and completions times are not met and no other actions apply.
C. Correct
D. 3.0.6 establishes exception to 3.0.2 for support systems that have an LCO in the Tech Specs.

Technical Reference(s): LCO 3.0.4 Bases

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-416 Obj 15.

Question Source: Bank # Nov 2010 Audit Q#96

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑4 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.43.b.1

Comments:

QUESTION 98 Rev 1

Examination Outline Cross-Reference:	Level	RO ☐	SRO ☑
	Tier # 3		
	K/A # G2.3.14		IR 3.8

Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities

Proposed Question:

Which of the following statements explains the significance of the Maximum Safe Area Radiation Level being reached in the areas identified in EOP-0003, Secondary Containment and Radioactive Release Control?

A. This is the value that requires entry into EOP-0003.

B. This is the value that requires isolating all systems discharging into the area except systems (1) required for damage control, and (2) required to be operated by EOPs.

C. This value in any one area requires the plant to be shutdown per GOP-0002 Plant Shutdown IAW EOP-0003.

D. This value along with a primary system discharging into secondary containment requires entry into EOP-0001 RPV Control IAW EOP-0003.

Proposed Answer: D

Explanation
A. The maximum normal operating value is the entry condition for EOP-0003.
B. The max <u>normal</u> operating value is used as the stop sign before this action in EOP-3.
C. EOP-0003 requires two areas to be at the Maximum Safe value prior to shutting down per GOP-0002 Plant Shutdown
D. Correct -

Technical Reference(s): EOP Bases

Proposed references to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.43.b.5

Comments: Distractor B replaced due to plausibility

QUESTION 99 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	3		
	K/A #	G2.4.16		IR 4.4

Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, and severe accident management guidelines

Proposed Question:

The plant is at 100% power when a manual reactor scram is inserted due to a steam leak on Moisture Separator Reheater B.

The Control Room Supervisor should:

A. Enter AOP-0001, REACTOR SCRAM, AOP-0002, TURBINE TRIP and EOP-0001, RPV CONTROL concurrently. EOP-0001 may be exited when it is determined that an emergency no longer exists.

B. Enter AOP-0001, REACTOR SCRAM and AOP-0002, TURBINE TRIP. After AOP immediate actions and required subsequent actions are complete, enter EOP-0001, RPV CONTROL. EOP-0001 may be exited when it is determined that an emergency no longer exists.

C. Enter EOP-0001, RPV CONTROL and exit when it is determined that an emergency no longer exists. Then enter AOP-0001, REACTOR SCRAM and AOP-0002, TURBINE GENERATOR TRIP.

D. Enter AOP-0001, REACTOR SCRAM, AOP-0002 TURBINE GENERATOR TRIP, and EOP-0001, RPV CONTROL concurrently. EOP-0001 may only be exited when directed by exit steps in the procedure.

Proposed Answer: A.

Explanation:
A. Correct – The AOPs and EOP are entered concurrently. The EOP is exited when the emergency no longer exits.
B. The listed procedures are executed concurrently.
C. The listed procedures are executed concurrently.
D. First part is correct, but the EOP may be exited when the emergency no longer exists. Exit steps normally only are provided for escalating conditions.

Technical Reference(s): OSP-0009, Section 10.2

Proposed references to be provided to applicants during examination: None

Learning Objective: R-LPOPS-HLO218 Obj 4; R-LP-OPS-HLO520 Obj 2; R-LPOPS-HLO512 Obj 4

Question Source: Bank # Oct 2012 Audit #99

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.43.b.5 Comments:

Comments:

QUESTION 100 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	3		
	K/A #	G2.4.44		IR 4.4

Knowledge of emergency plan protective action recommendations

Proposed Question:

Following the declaration of a General Emergency, Protective Action Recommendations (PARs) are issued by ______________.

A. Parish Officials.

B. State Officials.

C. Federal Emergency Management Agency.

D. Emergency Director.

Proposed Answer: D.

Explanation:
A. See D.
B. See D.
C. See D.
D. The Emergency Director issues PARs. Offsite agencies determine whether or not to act upon the recommendation.

Technical Reference(s): EIP-2-007, Protective Action Recommendations, Rev 25 p. 3 of 9

Proposed references to be provided to applicants during examination: None

Learning Objective: None identified.

Question Source: Bank # RBS-Audit 2010 #99

Question History: Last NRC Exam RBS NRC 2007

Cognitive Level: Memory or Fundamental Knowledge ☑ 3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.43.b.5

Comments:

RBS December 2014 NRC Initial Written Exam

Modified Question – Parent Questions

Q#	Parent Question
7	RBS December 2008 NRC Exam #8
16	RBS-OPS-5554
22	RBS-NRC-01108
23	RBS-NRC-444
37	Nov 2010 Audit Q#36
38	RBS-NRC-53
44	RBS-NRC-459
51	RBS-OPS-3255
56	RBS-NRC-168
60	RBS-OPS-06288
62	RBS-LOR-1252
64	RBS-OPS-2307
77	RBS-OPS-06265
83	2008 Audit Q#25
87	RBS-NRC-937

Exam Question # 7

Modified from Parent Q# RBS December 2008 NRC Exam #8

QUESTION 8 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	1
	Group #	1
	K/A #	295019 AA1.03
	Importance Rating	3.0

Proposed Question:

Due to severe weather in the area around River Bend Station, a loss of offsite power has occurred.

All three diesel generators have started and are supplying safety related loads.

What is the status of the Plant Air Systems?

a. The Service Air compressors will be de-energized. Instrument Air compressors will be supplied by a safety related power source.

b. The Instrument Air compressors will be de-energized. SAS-AOV134 cross-connect valve will open to maintain IAS header pressure with the SAS compressors.

c. Instrument Air and Service Air compressors will both be de-energized. The diesel air compressor will automatically start to supply plant air needs.

d. Instrument Air and Service Air compressors will both be de-energized. The diesel air compressor must be manually started to supply plant air needs.

Proposed Answer: D.

Exam Question # 16

Modified from Parent Q# RBS-OPS-5554

As part of a routine surveillance, valve timing is being performed on the Low Pressure Core Spray (LPCS) system valves. Currently all valves are in their normal standby position except for E21-MOVF001, Suppression Pool Suction Valve, which is closed. A valid high drywell pressure, Low Pressure Core Spray (LPCS) initiation signal is received.

Assuming no operator action, which of the following statements identifies the expected response of LPCS?

A. LPCS Pump will start and run with pump suction valve E21-MOVF001 closed.
B. LPCS Pump will start and then immediately trip because it has **no** suction path.
C. LPCS Pump will **<u>not</u>** receive a start signal because the suction path permissive is not met.
D. E21-MOVF001 receives an automatic open signal and the LPCS Pump starts when the valve is full open.

Answer: A

Exam Question # 22

Modified from Parent Q# RBS-NRC-01108

River Bend was operating at full power with RCIC tagged out. A trip of all Reactor Feedwater Pumps caused RPV level to drop and a Reactor scram to occur on low RPV water level. HPCS recovered RPV level and E22-F004, HPCS INJECT ISOL VALVE was closed when RPV water level reached +35 inches. Conditions are currently:

- Vessel is isolated from the condenser
- Reactor pressure 700 psig (rising at 10 psig per minute)
- Time after scram 5 minutes
- CRD pumps both tripped
- Drywell pressure 0.3 psid

If no additional operator actions are taken other than the ones listed, what is the expected RPV water level response over the next 10 minutes and why?

A. rise due to swell from decay heat.

B. rise due to Feed Reg. Valve leakage exceeding heat requiements.

C. lower due to cooldown.

D. lower due to steam loads reducing RPV water inventory.

Answer: A

Exam Question # 23

Modified from Parent Q# RBS-NRC-444

Given the following plant conditions:

- Reactor Power 0% (all rods in)
- Reactor Level +33 inches
- Reactor Pressure 890 psig
- Drywell Pressure 1.8 psid
- Drywell Temperature 138°F
- Containment Temperature 88°F
- Containment Pressure 0.35 psig
- Annulus Differential Pressure -4.5 in.WC

Based on the above conditions, which one of the following describes the Emergency Operating Procedures that should be entered?

A. EOP-1 ONLY

B. EOP-1 and 2

C. EOP-2 ONLY

D. EOP-2 and 3

Answer: B

Exam Question # 37

Modified from Parent Q# Nov 2010 Audit Q# 36

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group # 1	
	K/A #	215004	IR 2.6	

Knowledge of the electrical power supplies to SRM channels/detectors.

Proposed Question:

The power supply to SRM ‘A’ detector is _______________.

A. VBS-PNL01A

B. VBS-PNL01B

C. RPS A

D. RPS B

Proposed Answer: C.

Exam Question # 38

Modified from Parent Q# RBS-NRC-53

During a reactor startup the ATC has been withdrawing SRM detectors per GOP-0001. All SRMs except A indicate full out. SRM A has an upscale high and an upscale High-high trip indicated and is reading off-scale high. The P680 indications show the detector is "driving out."

The ATC should . . .

A. Immediately insert control rods to return SRM A readings on-scale.
B. Insert a Div I half scram and continue with the plant startup.
C. Check the SRM A drive power fuses, if the problem is not corrected, obtain reactor engineering assistance..
D. Since the other drive OUT lights are on, SRM A drive has power therefore contact I & C for assistance.

Answer: C

Exam Question # 44

Modified from Parent Q# RBS-NRC-459

Given the following conditions:

- The plant had been operating at 100% power
- A severe over-pressure transient resulted in ALL Safety Relief Valves (SRV) opening in the "relief" mode and then lifting in their "safety" mode
- RPV pressure peaked at 1200 psig
- All valves have closed (reseated) with the exception of one SRV that remains open in its "safety" mode
- The required actions of AOP-0035, "Safety Relief Valve Stuck Open", have been taken including scramming the reactor but the SRV has not closed.

Which of the following describes the resulting tailpipe temperature trend as the plant cools down and depressurizes through the stuck open SRV? (Assume containment pressure is 0 psig and remains constant.)

SRV tailpipe temperature will:

A. start at 260 °F, rise to approximately 290 °F and then will slowly fall following reactor pressure during the depressurization below 500 psig.
B. start at 525 °F and will slowly fall following reactor pressure during the depressurization.
C. start at 285 °F, rise to approximately 325 °F and then will slowly fall following reactor pressure during the depressurization below 500 psig.
D. start at 305 °F and will slowly fall following reactor pressure during the depressurization.

Answer: C

Exam Question # 51

Modified from Parent Q# RBS-OPS-3255

The "A" Standby Diesel Generator is operating in parallel with the offsite power supply.

If the operator takes the DG Voltage Regulator Control Switch to raise, which of the following indications should be showing the LARGEST change?

A. Frequency (HZ)
B. Real Load (KW)
C. Engine Speed (RPM)
D. Reactive Load (KVAR)

Answer: D

Exam Question # 56

Modified from Parent Q# RBS-NRC-168

If total feedwater flow drops below the reactor recirculation system interlock level, the Reactor Recirculation pumps will downshift to slow speed.

What is the PRIMARY reason for this interlock?

A. pump cavitation.
B. flow control valve cavitation.
C. excessive axial thrust on the pump.
D. inaccurate wide range level indication.

Answer: B

Exam Question # 60

Modified from Parent Q# RBS-OPS-06288

The primary means of hydrogen control following a LOCA utilizes:

A. Hydrogen Igniters
B. Hydrogen Recombiners
C. Containment Purge subsystem
D. Drywell Purge subsystem

Answer: A

Exam Question # 62

Modified from Parent Q# RBS-LOR-1252

The plant is operating at 80% power with EHC Pressure Regulator Channel A selected for pressure control. Pressure Regulator Channel B is in TEST.

Which one of the following describes the plant response if the Pressure Regulator Channel A Averaging Manifold Pressure transmitter fails to 0 psig?

A. Turbine control valves fully open, steam bypass valves fully open, reactor pressure lowers until MSIVs close.
B. Turbine control valves fully close, steam bypass valves fully open, reactor pressure rises until a reactor scram occurs.
C. Turbine control valves fully open, steam bypass valves remain closed, reactor pressure lowers until MSIVs close.
D. Turbine control valves fully close, steam bypass valves remain closed, reactor pressure rises until a reactor scram occurs.

Answer: D

Exam Question # 64

Modified from Parent Q# RBS-OPS-2307

The HVN Chilled Water system Supply and Return valves to Containment (HVN-MOV127, 128, 129, 130, 102) will AUTO-CLOSE upon __________.

A. RPV Level 1 (-143")
B. RPV Level 2 (-43")
C. Drywell to Annulus d/p greater than 12 inches
D. Opening of Service Water supply and Return valves to Containment

Answer: B

Exam Question # 77

Modified from Parent Q# RBS-OPS-06265

The plant has experienced an ATWS. The following conditions exist at P680:

- All eight white scram solenoid lights are extinguished
- Annunciator P680-05-C08 SCRAM PILOT VLV AIR HEADER LOW PRESSURE is alarming
- SDV Vent and Drain valve position lights indicate all four valves are closed
- Approximately 20% of the withdrawn control rods fully inserted
- CRD cooling water differential pressure has been maximized
- ARI has been inititated

Which of the following methods for alternate control rod insertion should be attempted next?

A. venting the scram air header.
B. resetting and reinitiating ARI.
C. removing the scram solenoid power fuses.
D. resetting the scram and intiating a manual scram.

Answer: D

Exam Question # 83

Modified from Parent Q# 2008 Audit Q# 25

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	1
	Group #	2
	K/A #	295015 AA2.02
	Importance Rating	4.1

Proposed Question:

Following a Level 3 scram signal, the following conditions exist:

143 control rods indicate full-in
2 control rods are not full-in
Reactor power indicates 0%

Which of the following concerning reactor status and required procedure implementation is correct?

a. It can NOT be determined that the reactor will remain shutdown under all conditions without boron. Transition to EOP-0001A is required.

b. It can be determined that the reactor will remain shutdown without boron, EOP-0001A is NOT applicable.

c. It can NOT be determined that the reactor will remain shutdown under all conditions without boron, but with reactor power less than 5%, EOP-0001A is NOT applicable.

d. It can NOT be determined that the reactor will remain shutdown under all conditions without boron, but no EOP entry condition exists.

Proposed Answer: A.

Exam Question # 87

Modified from Parent Q# RBS-NRC-937

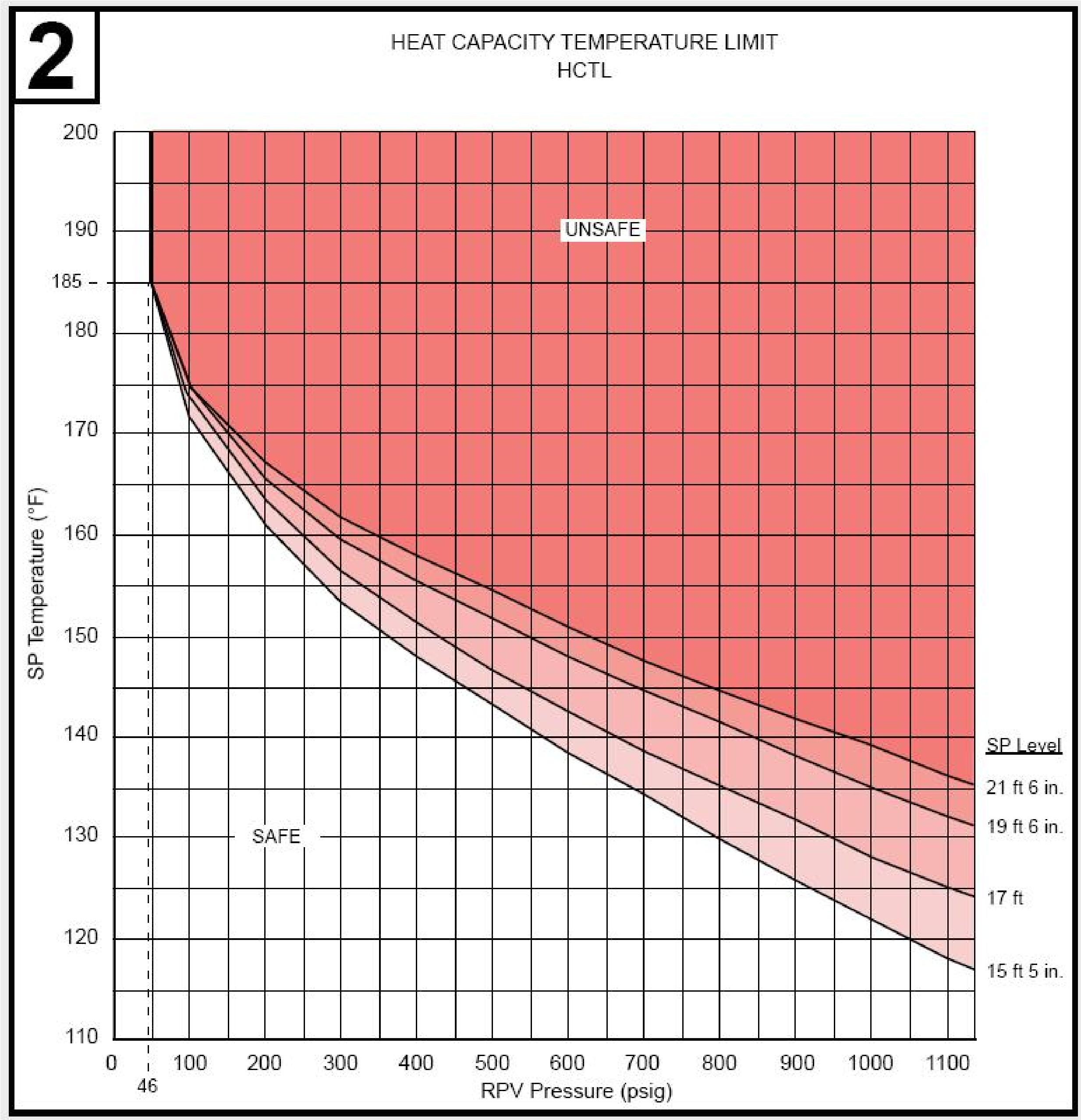
The plant is operating at 85% power. The Manual Scram Pushbutton Surveillance is being performed. After arming and depressing the last Manual Scram Pushbutton for DIV 4, the half scram is reset. However, a failure of the K14D relay contacts to reclose when re-energized results in the "B" RPS DIV 4 SCRAM SOV VALVES OPEN light above the DIV 4 Manual Scram Pushbutton on P680 remaining out. All other RPS scram solenoid valve white lights on P680 are lit.

If a loss of RPS Bus A occurs at this time, which one of the following is the expected response and appropriate action to be taken?

A. None of the control rods will scram. Transfer RPS Bus A to Alternate power per AOP-0010, Loss of RPS Bus.
B. Approximately 1/4 of the control rods will scram. Manually scram the reactor and enter AOP-0001 and AOP-0010.
C. Approximately 1/4 of the control rods will scram. Transfer RPS Bus A to Alternate power per AOP-0010, Loss of RPS Bus.
D. All of the control rods will scram. Manually scram the reactor and enter AOP-0001 and AOP-0010.

Answer: B

Proposed handouts for the Draft Written Exam submittal follow this coversheet.


2
HEAT CAPACITY TEMPERATURE LIMIT
HCTL
UNSAFE
SAFE
SP Temperature (°F)
200
190
185 –
180
170
160
150
140
130
120
110
0
46
100
200
300
400
500
600
700
800
900
1000
1100
RPV Pressure (psig)
SP Level
21 ft 6 in.
19 ft 6 in.
17 ft
15 ft 5 in.

INITIATING STANDBY LIQUID CONTROL

STANDBY LIQUID CONTROL INJECTION REQUIREMENTS

TANK LEVEL PRIOR TO INJECTION GAL	TANK LEVEL AFTER INJECTION OF 69 lb Boron (approximately 16 min inj time) GAL	TANK LEVEL AFTER INJECTION OF 166 lb Boron (approximately 38 min inj time) GAL
1531	905	0
1550	924	19
1600	974	69
1700	1074	169
1800	1174	269
1900	1274	369
2000	1374	469
2100	1474	569
2200	1574	669
2300	1674	769
2400	1774	869
2500	1874	969
2600	1974	1069
2700	2074	1169
2800	2174	1269
2900	2274	1369
3000	2374	1469
3100	2474	1569
3200	2574	1669
3300	2674	1769
3400	2774	1869
3500	2874	1969
3600	2974	2069
3700	3074	2169
3800	3174	2269
3900	3274	2369
4000	3374	2469
4100	3474	2569
4200	3574	2669

NOTE

WHEN tank level falls between values,
THEN the smaller value should be used.

CONTINUOUS USE

*G12.1.28

ENTERGY

RIVER BEND STATION
STATION OPERATING MANUAL
*** ABNORMAL OPERATING PROCEDURE**

**DEGRADED GRID*

PROCEDURE NUMBER:	***AOP-0064**
REVISION NUMBER:	***006**
Effective Date:	***04/13/2011**

NOTE : SIGNATURES ARE ON FILE.

*INDEXING INFORMATION

TABLE OF CHANGES

LETTER DESIGNATION TRACKING NUMBER	DETAILED DESCRIPTION OF CHANGES
AOP-0064R005EC-A	• Added procedural reference to Step 5.3. • Added clarifying bulleted steps to Step 5.4 and a clarifying Note to Step 5.5.2. • Corrected sentence structure in Step 5.5.3.5.

TABLE OF CONTENTS

SECTION PAGE NO.

1 PURPOSE/DISCUSSION

1.1 The purpose of this procedure is to:

- Provide instructions in the event of a degraded grid condition or Entergy transmission suspension of monitoring of the Entergy transmission grid.
- Create a consistent mechanism for timely notification of potential or developing grid changes that could impact ENS off-site power supply design requirements.
- Provide guidelines to Main Control Room Operations to ensure that grid reliability is maintained.
- Establish communications protocol during degraded grid notification between the System Operations Center and the Main Control Room Operations Staff.

1.2 Definitions

1.2.1. System Operations Center (SOC) – That part of the Entergy Transmission organization responsible for overall operation and monitoring of the Entergy transmission grid.

1.2.2. Transmission Operating Center (TOC) – That part of Entergy Transmission organization responsible for the day to day operation and monitoring of the local grid. For RBS Pine Bluff

1.2.3. Fancy Point Maintenance Point Of Contact (POC) – RBS site person who is the main point of contact (liaison) between RBS and SOC/TOC for matters concerning issue resolution.

1.2.4. OFF-LINE Monitoring – The process of using daily analyses for the purpose of projecting ENS site post trip voltages during the period of interest. If required this allows the application of corrective measures and notification to the affected ENS site(s) of post-trip voltages below the allowable limits of that facility.

1.2.5. ON-LINE Monitoring – The process of providing both near real-time contingency analysis monitoring of ENS post-trip projected voltages and real-time monitoring of present conditions.

1.2.6. Voltage Support Directives – MVAR (MegaVoltAmpReactive) directives received from the SOC/TOC to maintain the voltage of the grid. Directives differ from normal orders in that they will contain the phrase “This is a directive”.

1.3 Responsibilities

1.3.1. SOC Shift Supervisor – monitors the Entergy Transmission grid, and provides timely notification to the River Bend Station Main Control Room if projected nuclear offsite power supply design requirements are not maintained within specified limits.

1.3.2. Main Control Room-Operations Shift Manager (MCR-OSM) –

- Communicating with the SOC Shift Supervisor when notified that near real time on line monitoring or projected nuclear offsite power supply design requirements are not maintained within the specified limits. This responsibility includes scheduling any necessary follow-up calls with the SOC Shift Supervisor. Ref ENS-DC-201

- Perform initial assessment of Transmission notifications and initiate appropriate actions as necessary. This responsibility includes initiation of Condition Reports and performance of Operability Determinations, in accordance with plant procedures.

- Maintains awareness of the site specific design requirements for offsite power and the basis for those requirements in the operation of the station.

- Initiates appropriate actions as necessary in order to maintain grid reliability in accordance with procedures and protocol.

- Comply with directives unless such actions would violate safety, equipment, or regulatory or statutory requirements. Under these circumstances, the SOC shall be immediately informed of the inability to perform the directive so that the Reliability Coordinator may implement alternate remedial actions.

- Review procedures for performance and/or system configuration realignments, ensure vulnerabilities to common cause and common mode failures are evaluated for current plant conditions to protect safety sources and safety trains. (SOER 03-1 Recommendation 2 Emergency Power Reliability)(Ref. 6.3)

- Take actions as needed during events of grid instability and voltage degradation or the grid operator's prediction of insufficient off-site voltage levels following a plant trip by placing safety systems on backup power supplies and conservatively placing the plant in a safe operating or shutdown condition. The applicable procedures shall be used to manually configure electrical buses when automatic bus transfers fail to actuate or when manual alignment of emergency power is necessary. If offsite power is lost, the timely restoration of offsite power and resetting of safety system electrical sequencing equipment is the most significant operator action to mitigate core damage. The operating crew's immediate focus should be stabilizing the plant before consideration is given to returning the plant to power operation. (SOER 99-1 Recommendations 2a,b,c,d Loss Of Grid)(Ref. 6.5)

1.4 IF an SOC Notification of "ON-LINE Monitoring Tool Not Available" occurs, THEN closely monitor the voltages and frequencies on SPI-REC102 recorder on H13-P808 and alarms discussed in Section 2. This does not constitute grid degradation.

2 SYMPTOMS

2.1 The SOC has notified the OSM/CRS that grid degradation currently exists or notification that post-trip or actual switchyard voltage does not support accident loading.

2.2 Grid Oscillations of the main generator frequency and/or voltage

2.3 Annunciator, P808-86A-H01, GRID TROUBLE

- High Frequency 62 Hz
- Low Frequency 57 Hz
- Low Voltage 225.86 Kv (98.2%)

2.4 Low voltage alarms

- Annunciator, P808-86A-G09, PFD STA XFMR 1RTX-XSR1E VOLTAGE LOW (12kvolts)
- Annunciator, P808-86A-H09, PFD STA XFMR 1RTX-XSR1C VOLTAGE LOW (3780 volts)
- Annunciator, P808-88A-G01, PFD STA XFMR 1RTX-XSR1F LOW VOLTAGE (12kvolts)
- Annunciator, P808-88A-H01, PFD STA XFMR 1RTX-XSR1D LOW VOLTAGE (3780 volts)

3 AUTOMATIC ACTIONS

3.1 None

4 IMMEDIATE OPERATOR ACTIONS

4.1 None

5 SUBSEQUENT OPERATOR ACTIONS

NOTE

Steps may be performed concurrently as appropriate.

5.1 IF Annunciator, P808-86A-H01, GRID TROUBLE alarms, THEN contact the SOC Pine Bluff (primary) or TOC Beaumont (backup) to determine the reason and extent of the Degraded Grid.

- Primary: SOC Pine Bluff Ringdown / Phone 1-870-536-6935 / Fax 1-870-541-4594
- Backup: TOC Beaumont Phone 1-409-981-3061 / Fax 1-409-981-3083
  8-633-3061

5.2 Contact the Duty Manager.

NOTE

The Main Generator Emergency MVARS limit is
-60 MVARS to +359 MVARS.

5.3 IF possible, THEN maintain the Main Generator MVARS between 0 MVARS and +230 MVARS per SOP-0080.

5.4 Maintain the grid voltage at the RBS switchyard between 224.25 KV and 242 KV by:

- Adjusting MVARS per Step 5.3.
- Contacting the SOC or TOC for grid voltage assistance.

5.5 IF it is determined that grid voltage can not be maintained above 224.25 KV (97.5%), THEN perform the following:

5.5.1. Consult with SOC/TOC regarding stability of the grid near RBS.

5.5.2. IF SOC/TOC indicates that the Fancy Point is sufficiently stable, THEN

NOTE

It is preferred that Div 3 DG is transferred first and the division that has HVK running transferred last to prevent alternating HVK twice.

1. Perform a Normal start of all three Emergency Diesel Generators.

NOTE

If the synchroscope for the Diesel Generator(s) shows erratic movement, then the grid should be considered unstable and synchronizing the Diesel Generator to the grid should be avoided.

2. Use the synchroscopes used for paralleling the Diesel Generators to the safety busses as an instantaneous indication of instability on the grid.

NOTE

An unstable grid will result in swings in at least one of the following data points.

3. Use the following to trend the grid stability before paralleling the Diesel Generator to the grid by comparing the trends with historical data, inspecting for erratic behavior such as sudden spikes and dips:

   - IF the Main Generator Voltage Regulator is in Auto, THEN trend the PDS data for Main Generator MVARs (PDS Point – SPGEA02).

   - IF the Main Generator Voltage Regulator is in Manual, THEN trend the PDS data for Main Generator Voltage output (PDS Point – SPGEA01).

   - Trend PDS data for the Main Turbine Speed (PDS Point – N31EA001).

4. IF the grid is verified stable, THEN Parallel the Emergency Diesel Generators with offsite power AND disconnect the bus from the grid.

5.5.3. IF SOC/TOC OR plant indications used in step 5.5.2 indicate that the Fancy Point is NOT sufficiently stable, THEN:

1. Emergency Start each diesel generator.

2. Verify diesel system parameters are normal.

3. Ensure all isolations are reset for the applicable safety related bus prior to continuing to the next bus.

CAUTION

IF RPS is being supplied from the Alternate source, THEN deenergizing the divisional safety related bus will cause a RPS half SCRAM and Inboard or Outboard BOP isolation.

NOTE

It is preferred that Div 3 DG is transferred first and the division that has HVK running transferred last to prevent alternating HVK twice.

4. Open supply breaker to a single divisional safety related bus.

5. Restore isolations and systems as necessary.

- Verify alarms caused by loss of power clear upon power restoration.
- Restore RWCU and SPC, IF isolated on loss of power to leak detection.
- Reset half isolations for RWCU and MSL Drains per AOP-0003, Automatic Isolations.
- Restart Rad Monitors.
- Reset RPS Alternate EPA Breakers.
- Alternate Divisions of HVK if necessary.
- Reset SRV Accoustic Monitors (Division 2).
- Reset Bearing Lift Pumps following loss of power per ARP-680-15A-A2 (Division 2).
- Restore Generator Seal Oil to normal lineup per SOP-0019, Generator Seal Oil System (Division 2).

6. Repeat step 5.5.3.3 through 5.5.3.6 for remaining safety related busses.

5.6 Perform STP-302-0102, Power Distribution System Operability Check to verify operability of safety system buses.

5.7 Refer to the following TS for Operability.

- Modes 1, 2 or 3; 3.8.1, AC Sources Operating and 3.8.9, Distribution Systems Operating
- Modes 4 or 5; 3.8.2, AC source Shutdown and 3.8.10, Distribution Systems Shutdown

5.8 The OSM shall perform an initial assessment and initiate appropriate actions when notified. This responsibility includes the following:

- Scheduling any necessary follow-up calls with the SOC.
- Initiating a Condition Report and perform an Operability Determination.
- Contacting Engineering for assistance in evaluating grid conditions and options.

5.9 In response to a SOC grid degradation notification, the OSM should perform the following:

- Promptly initiate the requested actions (megawatts, megavars, etc.) in response to SOC grid urgency directive.
- Perform a risk assessment using ADM-0096, Risk Management Program Implementation and On-Line Maintenance Risk Assessment.
- Terminate and/or suspend all activities that may reduce generating capability, cause a unit shutdown, or impact critical electrical distribution.

NOTE

Discretion should be used in evaluation of plant testing and/or operational activities that raises the risk of unit loss. Mandatory testing and/or operational activities may be allowed, provided such activity is necessary due to time constraints. Maintenance that is performed on unit auxiliaries to ensure unit reliability, but raises the risk of unit loss, requires evaluation.

- Avoid actions that raise plant risks.
- Maintain or return Emergency Diesel Generators (EDGs) to a standby status as soon as practical.

- Maintain generator parameters in accordance with SOC requests and generator operating procedures.

- During normal working hours notify the Site Fancy Point Maintenance Point Of Contact (POC) of the condition. This person will act as a point of contact for any actions to take place in the switchyard or the plant. Outside normal working hours or if the POC can not be contacted, THEN Operations will be the point of contact.

- Verify all Limiting Conditions for Operation (LCOs) are entered when conditions require.

- Verify all abnormal procedures are entered when conditions require.

- Monitor grid voltage and frequency on recorder SPI-REC102 on H13-P808 and annunciator H13-P808/86A/H01, GRID TROUBLE and take action as directed by ARP-808-86.

- IF notified by the SOC that analysis indicates that the grid will NOT be able to provide the required post-trip voltage to support offsite power operability in the event of a reactor scram/generator trip, THEN declare the offsite power system INOPERABLE and take the required actions to evaluate the condition and/or return to Operable status.

- Initiate a Condition Report

5.10 System Operations Center/Main Control Room Interface

5.10.1. The SOC shall provide the MCR-OSM with early warning of potential or developing grid issues through direct notification. This notification/discussion should include: (SOER 99-1 Recommendation 1b & 1c Loss Of Grid)(Ref. 6.4)

- Expected duration of degraded condition

- Actions being taken and planned to restore grid to an acceptable condition

- Establishment of periodic communication until the grid is restored to the desired condition

5.10.2. The SOC will notify the OSM when grid degradation exists currently **OR** that conditions are projected to occur at a specific time **OR** that Entergy transmission monitoring of grid stability has been suspended.

5.11 In response to a SOC grid degradation notification, the OSM should avoid performing the following:

- Implementing actions that raise plant risks.
- Paralleling EDGs with unstable off-site power sources (may cause severe load swings and overload the EDGs).
- Proactive operation of plant equipment that may raise plant risks by bypassing or circumventing automatic protective features.

5.12 Refer to the following corporate procedures:

- ENS-DC-201, ENS Transmission Grid Monitoring

NOTE

ENS-DC-199 Attachment 9.3 identifies the operational requirements for the RBS offsite power supply. These requirements are part of the RBS design basis and licensing basis. Failure to meet these requirements may render the offsite power supply inoperable, thus requiring the operating RBS unit to shutdown. Failure to meet these requirements must be immediately communicated to RBS and its operating staff for operability determination.

- ENS-DC-199, Off-Site Power Supply Design Requirements (SOER 99-1 Recommendation 4 Loss Of Grid)(Ref. 6.6)
  - With the RBS units off-line, the 230 kV offsite power source should be capable of providing 84.7 MW and 59.7 MVAR to RBS for start-up, or for safe shutdown and design basis accident mitigation.

6 REFERENCES

6.1 ENS-DC-201, ENS Transmission Grid Monitoring

6.2 ENS-DC-199, Off-Site Power Supply Design Requirements

6.3 SOER 03-1 Recommendation 2 Emergency Power Reliability

6.4 SOER 99-1 Recommendation 1b & 1c Loss Of Grid

6.5 SOER 99-1 Recommendation 2a,b,c,d Loss Of Grid

6.6 SOER 99-1 Recommendation 4 Loss Of Grid

EMERGENCY CLASSIFICATION USER AID 1 MODES 1 - 3

Plant Modes (white boxes indicate applicable modes)	1	Power Operations	2	Startup	3	Hot Shutdown	4	Cold Shutdown	5	Refuel	D	Defueled

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENT — OPERATING MODES: 1 2 3 4 5 D

Radiological Effluents

GENERAL EMERGENCY — AG1 (1 2 3 4 5 D)

Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity > 1000 mR TEDE or 5000 mR thyroid CDE for the actual or projected duration of the release using actual meteorology

Emergency Action Level: (1 or 2 or 3)

NOTE: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, the classification should be based on EAL #2 instead of EAL #1. Do not delay declaration awaiting dose assessment results.

1. VALID reading on any of the radiation monitors in Table R1 > the GENERAL EMERGENCY reading for ≥ 15 minutes

OR

2. Dose assessment using actual meteorology indicates doses > 1000 mR TEDE or 5000 mR thyroid CDE at or beyond the SITE BOUNDARY

OR

3. Field survey results indicate closed window dose rates > 1000 mR/hr expected to continue for ≥ 60 minutes; or analyses of field survey samples indicate thyroid CDE > 5000 mR for one hour of inhalation, at or beyond the SITE BOUNDARY

SITE AREA EMERGENCY — AS1 (1 2 3 4 5 D)

Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity > 100 mR TEDE or 500 mR thyroid CDE for the actual or projected duration of the release

Emergency Action Level(s) (1 or 2 or 3)

NOTE: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, the classification should be based on EAL #2 instead of EAL #1. Do not delay declaration awaiting dose assessment results.

1. VALID reading on any of the radiation monitors in Table R1 > the SITE AREA EMERGENCY reading for ≥ 15 minutes

OR

2. Dose assessment using actual meteorology indicates doses > 100 mR TEDE or 500 mR thyroid CDE at or beyond the SITE BOUNDARY

OR

3. Field survey results indicate closed window dose rates > 100 mR/hr expected to continue for ≥ 60 minutes; or analyses of field survey samples indicate thyroid CDE > 500 mR for one hour of inhalation, at or beyond the SITE BOUNDARY

ALERT — AA1 (1 2 3 4 5 D)

Any release of gaseous or liquid radioactivity to the environment > 200 times the ODCM limit for ≥ 15 minutes

Emergency Action Level(s): (1 or 2 or 3)

NOTE: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

1. VALID reading on any of the radiation monitors in Table R1 > the Alert reading for ≥ 15 minutes

OR

2. For RMS-RE107 effluent monitor:
   EITHER
   VALID reading > 200 times the alarm setpoint established by a current radioactivity discharge permit for ≥ 15 minutes
   OR
   VALID reading > 1.27E-01 µCi/ml for ≥ 15 minutes

OR

3. Confirmed sample analysis for gaseous or liquid releases indicate concentrations or release rates > 200 times the ODCM limit for ≥ 15 minutes

NOUE — AU1 (1 2 3 4 5 D)

Any release of gaseous or liquid radioactivity to the environment > 2 times the ODCM limit for ≥ 60 minutes

Emergency Action Level(s): (1 or 2 or 3)

NOTE: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

1. VALID reading on any of the radiation monitors in Table R1 > the NOUE reading for ≥ 60 minutes

OR

2. VALID reading on RMS-RE107 effluent monitor > 2 times the alarm setpoint established by a current radioactivity discharge permit for ≥ 60 minutes

OR

3. Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 2 times the ODCM limit for ≥ 60 minutes

Abnormal Radiation Levels

ALERT — AA2 (1 2 3 4 5 D)

Damage to irradiated fuel or loss of water level that has resulted or will result in the uncovering of irradiated fuel outside the reactor vessel

Emergency Action Level(s): (1 or 2)

1. A water level drop in the reactor refueling cavity, spent fuel pool or fuel transfer canal that will result in irradiated fuel becoming uncovered

OR

2. A VALID reading on any of the following radiation monitors due to damage to irradiated fuel or loss of water level:
   - RMS-RE140 2000 mR/hr
   - RMS-RE141 2000 mR/hr
   - RMS-RE192 2000 mR/hr
   - RMS-RE193 2000 mR/hr
   - RMS-RE5A 1.64E+03 µCi/sec
   - RMS-RE5B (GE) 5.29E-04 µCi/ml

NOUE — AU2 (1 2 3 4 5 D)

UNPLANNED rise in plant radiation levels

Emergency Action Level(s): (1 or 2)

1. a. UNPLANNED water level drop in a reactor refueling pathway as indicated by any of the following:
   - Water level drop in the reactor refueling cavity, spent fuel pool, or fuel transfer canal indication on Control Room Panel 870
   - Personnel observation by visual or remote means.

   AND

   b. UNPLANNED VALID area radiation monitor alarm on any of the following:
   - RMS-RE140
   - RMS-RE141
   - RMS-RE192
   - RMS-RE193

OR

2. UNPLANNED VALID area radiation monitor readings or survey results indicate a rise by a factor of 1000 over normal* levels

NOTE: For area radiation monitors with ranges incapable of measuring 1000 times normal levels, classification shall be based on VALID full scale indications unless surveys confirm that area radiation levels are below 1000 times normal* within 15 minutes of the area radiation monitor indications going full scale.*

* Normal can be considered the highest reading in the past 24 hours excluding the current peak value.

ALERT — AA3 (1 2 3 4 5 D)

Rise in radiation levels within the facility that impedes operation of systems required to maintain plant safety functions

Emergency Action Level: (s)

1. Dose rate > 15 mR/hr in any of the following areas requiring continuous occupancy to maintain plant safety functions:
   - Main Control Room
   - CAS

Table R1 EAL THRESHOLD

Method	GENERAL DRMS	GENERAL Threshold	SITE AREA DRMS	SITE AREA Threshold	ALERT DRMS	ALERT Threshold	NOUE DRMS	NOUE Threshold
Main Plant Primary	4GE125	4.70E+08 µCi/sec	4GE125	4.70E+07 µCi/sec	4GE125	3.06E+07 µCi/sec	4GE125	3.06E+05 µCi/sec
Main Plant Secondary	N/A		N/A		1GE126	2.82E-01 µCi/ml	1GE126	5.26E-03 µCi/ml
FB Vent Primary	4GE005	6.70E+07 µCi/sec	4GE005	6.70E+06 µCi/sec	4GE005	2.19E+06 µCi/sec	4GE005	2.19E+04 µCi/sec
FB Vent Secondary	N/A		N/A		5GE005	2.82E-01 µCi/ml	5GE005	4.65E-03 µCi/ml
RW Vent Primary	N/A		N/A		4GE006	2.58E+06 µCi/sec	4GE006	2.58E+04 µCi/sec
RW Vent Secondary					5GE006	6.84E-02 µCi/ml	5GE006	6.84E-04 µCi/ml

FISSION PRODUCT BARRIER — OPERATING MODES: 1 2 3 4 5 D

FPB Loss / Potential Loss

GENERAL EMERGENCY — FG1 (1 2 3 4 5 D)

Loss of ANY two barriers AND loss or potential loss of third barrier

Emergency Action Level(s):

1. Loss of any two barriers
   AND
   Loss or potential loss of the third barrier

SITE AREA EMERGENCY — FS1 (1 2 3 4 5 D)

Loss or potential loss of ANY two barriers

Emergency Action Level(s):

1. Loss or potential loss of any two barriers

ALERT — FA1 (1 2 3 4 5 D)

ANY loss or ANY potential loss of EITHER fuel clad or RCS

Emergency Action Level(s):

1. Any loss or any potential loss of fuel clad
   OR
   Any loss or any potential loss of RCS

NOUE — FU1 (1 2 3 4 5 D)

ANY loss or ANY potential loss of containment

Emergency Action Level(s):

1. Any loss or any potential loss of containment

FUEL CLAD (FC) Barrier

Parameter	Loss	Potential Loss
FC1 Primary coolant activity level	Coolant activity > 300 µCi/gm dose equivalent I_{131}	None
FC2 Reactor vessel water level	RPV water level cannot be restored and maintained above -186 inches	RPV water level cannot be restored and maintained above -162 inches or cannot be determined
FC3 Primary containment radiation monitors	Containment radiation monitor RMS-RE16 reading > 3,000 R/hr	None
FC4 Emergency Director judgment	Any condition in the opinion of the Emergency Director that indicates loss of the Fuel Clad barrier	Any condition in the opinion of the Emergency Director that indicates potential loss of the Fuel Clad barrier

REACTOR COOLANT SYSTEM (RC) Barrier

Parameter	Loss	Potential Loss
RC1 Drywell pressure	Drywell pressure > 1.68 psid with indications of reactor coolant leak in drywell.	None
RC2 Reactor vessel water level	RPV water level cannot be restored and maintained above -162 inches or cannot be determined	None
RC3 RCS Leak Rate	1. UNISOLABLE main steam line break as indicated by the failure of both MSIVs in any one line to close AND High MSL flow annunciator (P601-19A-A2) OR 2. Indication of an UNISOLABLE HPCS, feedwater, RWCU or RCIC break OR 3. Emergency RPV depressurization is required	1. RCS leakage > 50 gpm inside the drywell OR 2. UNISOLABLE RCS leakage outside PC as indicated by exceeding either of the following: a. Max Normal Operating Temperature (Table F2) OR b. Max Normal Area Radiation (Table F2)
RC4 Drywell radiation	Drywell radiation monitor RMS-RE20 reading > 100 R/hr due to reactor coolant leakage	None
RC5 Emergency Director judgment	Any condition in the opinion of the Emergency Director that indicates loss of the RCS barrier	Any condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier

PRIMARY CONTAINMENT (PC) Barrier

Parameter	Loss	Potential Loss
PC1 Primary containment conditions	1. Rapid unexplained loss of PC pressure following initial pressure rise OR 2. PC pressure response not consistent with LOCA conditions	1. PC pressure > 15 psig and rising OR 2. a. PC hydrogen in the unsafe zone of HDOL curve OR b. DW hydrogen concentration > 9% OR 3. RPV pressure and suppression pool temperature cannot be maintained below the HCTL
PC2 Reactor vessel water level	None	Entry into PC flooding procedures SAP-1 and SAP-2
PC3 Primary containment isolation failure or bypass	1. a. Failure of both valves in any one line to close AND b. Direct downstream pathway to the environment exists after PC isolation OR 2. Intentional PC venting per EOPs or SAPs OR 3. UNISOLABLE RCS leakage outside PC as indicated by exceeding either of the following: a. Max Safe Operating Temperature (Table F1) OR b. Max Safe Area Radiation (Table F1)	None
PC4 Primary containment radiation monitors	None	Containment radiation monitor RMS-RE16 reading > 10,000 R/hr
PC5 Emergency Director judgment	Any condition in the opinion of the Emergency Director that indicates loss of the Primary Containment barrier	Any condition in the opinion of the Emergency Director that indicates potential loss of the Primary Containment barrier

TABLE F2 — RC 3 Potential Loss of RCS

Parameter	Area Temperature (Isolation temperature alarm)	DRMS Grid 2	Max Normal Operating Value
RHR A equip area	117° F (P601-20A-B4)	1213	8.2E+01 mR/hr
RHR B equip area	117° F (P601-20A-B4)	1214	8.2E+01 mR/hr
RHR C equip area	N/A	1215	8.2E+01 mR/hr
RCIC room	182° F (P601-21A-B6)	1219	1.20E+02 mR/hr
MSL Tunnel	173° F (P601-19A-A1/A3/B1/B3)		N/A
RWCU pump room 1 (A) / 2 (B)	165° F (P680-1A-A2/B2)		N/A

TABLE F1 — PC 3 Loss of Primary Containment

Parameter	Area Temperature Max Safe Operating Value	DRMS Grid 2	Max Safe Operating Value
RHR A equip area	200° F	1213	9.5E+03 mR/hr
RHR B equip area	200° F	1214	9.5E+03 mR/hr
RHR C equip area	N/A	1215	9.5E+03 mR/hr
RCIC room	200° F	1219	9.5E+03 mR/hr
MSL Tunnel	200° F		N/A
RWCU pump room 1 (A) / 2 (B)	200° F		N/A

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY — OPERATING MODES: 1 2 3 4 5 D

Security

GENERAL EMERGENCY — HG1 (1 2 3 4 5 D)

HOSTILE ACTION resulting in loss of physical control of the facility

Emergency Action Level(s): (1 or 2)

1. A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions

OR

2. A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and IMMINENT fuel damage is likely for a freshly off-loaded reactor core in pool

SITE AREA EMERGENCY — HS1 (1 2 3 4 5 D)

HOSTILE ACTION within the PROTECTED AREA

Emergency Action Level(s):

1. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the RBS security shift supervision

ALERT — HA1 (1 2 3 4 5 D)

HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat

Emergency Action Level(s): (1 or 2)

1. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the RBS security shift supervision

OR

2. A validated notification from NRC of an airliner attack threat within 30 minutes of the site

NOUE — HU1 (1 2 3 4 5 D)

Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant

Emergency Action Level(s): (1 or 2 or 3)

1. A SECURITY CONDITION that does NOT involve a HOSTILE ACTION as reported by the RBS security shift supervision

OR

2. A credible site specific security threat notification

OR

3. A validated notification from NRC providing information of an aircraft threat

Discretionary

GENERAL EMERGENCY — HG2 (1 2 3 4 5 D)

Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency

Emergency Action Level(s):

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area

SITE AREA EMERGENCY — HS2 (1 2 3 4 5 D)

Other conditions exist which in the judgment of the Emergency Director warrant declaration of a SITE AREA EMERGENCY

Emergency Action Level(s):

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts: (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the SITE BOUNDARY

ALERT — HA2 (1 2 3 4 5 D)

Other conditions exist which in the judgment of the Emergency Director warrant declaration of an ALERT

Emergency Action Level(s):

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels

NOUE — HU2 (1 2 3 4 5 D)

Other conditions exist which in the judgment of the Emergency Director warrant declaration of a NOUE

Emergency Action Level(s):

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs

Control Room Evacuation

SITE AREA EMERGENCY — HS3 (1 2 3 4 5 D)

Control room evacuation has been initiated and plant control cannot be established

Emergency Action Level(s):

1. a. Control room evacuation has been initiated
   AND
   b. Control of the plant cannot be established in accordance with AOP-0031, Shutdown from Outside the Main Control Room, within 15 minutes

ALERT — HA3 (1 2 3 4 5 D)

Control room evacuation has been initiated

Emergency Action Level(s):

1. AOP-0031, Shutdown from Outside the Main Control Room requires Control Room evacuation

Fire

ALERT — HA4 (1 2 3 4 5 D)

FIRE or EXPLOSION affecting the operability of plant safety systems required to establish or maintain safe shutdown

Emergency Action Level(s):

1. FIRE or EXPLOSION resulting in VISIBLE DAMAGE to any of the structures or areas in Table H2 containing safety systems or components or Control Room indication of degraded performance of those safety systems

NOUE — HU4 (1 2 3 4 5 D)

FIRE within the PROTECTED AREA not extinguished within 15 minutes of detection or EXPLOSION within the PROTECTED AREA

Emergency Action Level(s): (1 or 2)

NOTE: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the duration has exceeded, or will likely exceed, the applicable time.

1. FIRE not extinguished within 15 minutes of Control Room notification or verification of a Control Room FIRE alarm in any Table H2 structure or area

OR

2. EXPLOSION within the PROTECTED AREA

Toxic or Flammable Gases

ALERT — HA5 (1 2 3 4 5 D)

Access to a VITAL AREA is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of operable equipment required to maintain safe operations or safely shutdown the reactor

Emergency Action Level(s):

NOTE: If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.

1. Access to Main Control Room, Auxiliary Building, or 95' Control Building is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shutdown the reactor

NOUE — HU5 (1 2 3 4 5 D)

Release of toxic, corrosive, asphyxiant or flammable gases deemed detrimental to NORMAL PLANT OPERATIONS

Emergency Action Level(s): (1 or 2)

1. Toxic, corrosive, asphyxiant or flammable gases in amounts that have or could adversely affect NORMAL PLANT OPERATIONS

OR

2. Report by West Feliciana Parish for evacuation or sheltering of site personnel based on an offsite event

Natural or Destructive Phenomena

ALERT — HA6 (1 2 3 4 5 D)

Natural or destructive phenomena affecting VITAL AREAS

Emergency Action Level(s): (1 or 2 or 3 or 4 or 5 or 6)

1.a. Seismic event > Operating Basis Earthquake (OBE) as indicated by:
   Annunciator "Seismic Tape Recording System Start" (P680-02A-D06)
   AND
   Event Indicator on ERS-NBI-102 is white
   AND
   Receipt of EITHER 1 OR 2:
   1. Annunciator "Seismic Event High" (P680-02A-C06)
   2. Annunciator "Seismic Event High-High" (P680-02A-B06) AND amber light(s) on panel NBI-101

   AND

   b. Earthquake confirmed by any of the following:
   - Earthquake is felt in plant
   - National Earthquake Center
   - Control Room indication of degraded performance of systems required for safe shutdown of the plant

OR

2. Tornado striking resulting in VISIBLE DAMAGE to any of the Table H2 structures or areas containing safety systems or components or Control Room indication of degraded performance of those safety systems

OR

3. Internal flooding in Auxiliary Building 70 ft elevation resulting in an electrical shock hazard that precludes access to operate or monitor safety equipment or Control Room indication of degraded performance of those safety systems

OR

4. Turbine failure-generated PROJECTILES resulting in VISIBLE DAMAGE to or penetration of any of the Table H2 structures or areas containing safety systems or components or Control Room indication of degraded performance of those safety systems

OR

5. Vehicle crash resulting in VISIBLE DAMAGE to any of the Table H2 structures or areas containing safety systems or components or Control Room indication of degraded performance of those safety systems

OR

6. Hurricane or high SUSTAINED wind conditions ≥ 74 mph within the PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the Table H2 structures or areas containing safety systems or components or Control Room indication of degraded performance of those safety systems

NOUE — HU6 (1 2 3 4 5 D)

Natural or destructive phenomena affecting the PROTECTED AREA

Emergency Action Level(s):(1 or 2 or 3 or 4 or 5)

1. Seismic event identified by any 2 of the following:
   - Seismic event confirmed by activated seismic switch as indicated by receipt of EITHER a OR b:
     a. Annunciator "Seismic Tape Recording SYS Start" (P680-02A-D06)
     b. Event Indicator on ERS-NBI-102 is white
   - Earthquake felt in plant
   - National Earthquake Center

OR

2. Tornado striking within the PROTECTED AREA boundary

OR

3. Internal flooding that has the potential to affect safety related equipment required by Technical Specifications for the current operating mode in any Table H1 area

OR

4. Turbine failure resulting in casing penetration or damage to turbine or generator seals

OR

5. Severe weather or hurricane conditions with indication of SUSTAINED high winds ≥ 74 mph within the PROTECTED AREA boundary

Table H2 — Structures Containing Functions or Systems Required for Safe Shutdown

Reactor Building	Standby Cooling Tower
Auxiliary Building	Diesel Generator Building
Control Building	Tunnels (B, D, E, F, G)
Fuel Building	

Table H1 — Uncontrolled Flooding Threshold Area Water Level

Affected Location / Parameter	Max Safe Operating Value / Indicator
Aux Bldg Crescent Area 70' EL	6 inches above floor (must be verified locally)
HPCS Room 70'EL	4 inches above floor (P870-51A-G4)
RHR A Room 70'EL	4 inches above floor (P870-51A-G4)
RHR B Room 70'EL	4 inches above floor (P870-51A-G4)
RHR C Room 70'EL	4 inches above floor (P870-51A-G4)
LPCS Room 70'EL	4 inches above floor (P870-51A-G4)
RCIC Room 70'EL	4 inches above floor (P870-51A-G4)

EVENTS RELATED TO ISFSI — OPERATING MODES: 1 2 3 4 5 D

Cask Damage

NOUE — E-HU1 (1 2 3 4 5 D)

Damage to a loaded cask CONFINEMENT BOUNDARY

Emergency Action Level(s):

1. Damage to a loaded cask CONFINEMENT BOUNDARY

SYSTEM MALFUNCTION — OPERATING MODES: 1 2 3 4 5 D

Loss of AC Power

GENERAL EMERGENCY — SG1 (1 2 3 4 5 D)

Prolonged loss of all offsite and all onsite AC power to emergency busses

Emergency Action Level(s):

1. a. Loss of all offsite and all onsite AC power to Div I, II and III ENS busses
   AND
   b. Either of the following:
   - Restoration of at least one emergency bus in < 4 hours is not likely
     OR
   - RPV level cannot be maintained > -162 inches.

SITE AREA EMERGENCY — SS1 (1 2 3 4 5 D)

Loss of all offsite and all onsite AC power to emergency busses for ≥ 15 minutes

Emergency Action Level(s):

NOTE: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

1. Loss of all offsite power and all onsite AC power to Div I, II and III ENS busses for ≥ 15 minutes

ALERT — SA1 (1 2 3 4 5 D)

AC power capability to emergency busses reduced to a single power source for ≥ 15 minutes such that any additional single failure would result in station blackout

Emergency Action Level(s):

NOTE: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

1. a. AC power capability to Div I and Div II ENS busses reduced to a single power source for ≥ 15 minutes
   AND
   b. Any additional single failure will result in a station blackout

NOUE — SU1 (1 2 3 4 5 D)

Loss of all offsite AC power to emergency busses for ≥ 15 minutes

Emergency Action Level(s):

NOTE: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

1. Loss of all offsite AC power to Div I and II ENS busses for ≥ 15 minutes

Failure of Reactor Protection System

GENERAL EMERGENCY — SG3 (1 2 3 4 5 D)

Automatic scram and all manual actions fail to shutdown the reactor and indication of an extreme challenge to the ability to cool the core exists

Emergency Action Level(s):

1.a. An automatic scram failed to shutdown the reactor
   AND
   b. All manual actions do not shutdown the reactor as indicated by reactor power ≥ 5%
   AND
   c. Either of the following exist or have occurred due to continued power generation:
   - Core cooling is extremely challenged as indicated by RPV level cannot be restored and maintained > -186 inches
     OR
   - Heat removal is extremely challenged as indicated by RPV pressure and Suppression Pool temperature cannot be maintained in the EOP Heat Capacity Temperature Limit (HCTL) Safe Zone

SITE AREA EMERGENCY — SS3 (1 2 3 4 5 D)

Automatic scram fails to shutdown the reactor and the manual actions taken from the reactor control console are not successful in shutting down the reactor

Emergency Action Level(s):

1.a. An automatic scram failed to shutdown the reactor
   AND
   b. Manual actions taken at the reactor control console do not shutdown the reactor as indicated by reactor power ≥ 5%

ALERT — SA3 (1 2 3 4 5 D)

Automatic scram fails to shutdown the reactor and the manual actions taken from the reactor control console are successful in shutting down the reactor

Emergency Action Level(s):

1. a. An automatic scram failed to shutdown the reactor
   AND
   b. Manual actions taken at the reactor control console successfully shutdown the reactor as indicated by reactor power < 5%

Loss of DC Power

SITE AREA EMERGENCY — SS4 (1 2 3 4 5 D)

Loss of all vital DC power for ≥ 15 minutes

Emergency Action Level(s):

NOTE: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

1. < 105 VDC on all vital DC busses ≥ 15 minutes

Loss of Annunciation / Indication

SITE AREA EMERGENCY — SS6 (1 2 3 4 5 D)

Inability to monitor a SIGNIFICANT TRANSIENT in progress

Emergency Action Level(s):

NOTE: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

1. a. UNPLANNED loss of > approximately 75% of the following for ≥ 15 minutes:
   - Control Room safety system annunciation
     OR
   - Control Room safety system indication

   AND

   b. A SIGNIFICANT TRANSIENT is in progress

   AND

   c. Compensatory indications are unavailable

ALERT — SA6 (1 2 3 4 5 D)

UNPLANNED loss of safety system annunciation or indication in the control room with either (1) a SIGNIFICANT TRANSIENT in progress, or (2) compensatory non-alarming indicators are not available

Emergency Action Level(s):

NOTE: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

1. a. UNPLANNED loss of > approximately 75% of the following for ≥ 15 minutes:
   - Control Room safety system annunciation
     OR
   - Control Room safety system indication

   AND

   b. Either of the following:
   - A SIGNIFICANT TRANSIENT is in progress
     OR
   - Compensatory indications unavailable

NOUE — SU6 (1 2 3 4 5 D)

UNPLANNED loss of safety system annunciation or indication in the Control Room for ≥ 15 minutes

Emergency Action Level(s):

NOTE: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

1. UNPLANNED loss of > approximately 75% of the following for ≥ 15 minutes:
   a. Control Room safety system annunciation
   OR
   b. Control Room safety system indication

RCS Leakage

NOUE — SU7 (1 2 3 4 5 D)

RCS leakage

Emergency Action Level(s): (1 or 2)

NOTE: A relief valve that operates and fails to close per design should be considered applicable if the relief valve cannot be isolated.

1. Unidentified or pressure boundary leakage > 10 gpm

OR

2. Identified leakage > 35 gpm

Loss of Communication

NOUE — SU8 (1 2 3 4 5 D)

Loss of all onsite or offsite communication capabilities

Emergency Action Level(s): (1 or 2)

1. Loss of all of the following onsite communications methods affecting the ability to perform routine operations:
   - Plant radio system
   - Plant paging system
   - Sound powered phones
   - In-plant telephones

OR

2. Loss of all of the following offsite communications methods affecting the ability to perform offsite notifications:
   - All telephones
   - NRC phones
   - State of Louisiana Radio
   - Offsite notifications system and hotline

Cladding Degradation

Table S1

Flow (cfm)	Dose Rate (mR/hr)
≤15	9000
>15-17	8000
>17-20	7000
>20-25	5000
>25-30	4000
>30-60	2000
>60-140	1000
>140-200	700

NOUE — SU9 (1 2 3 4 5 D)

Fuel clad degradation

Emergency Action Level(s): (1 or 2)

1. Offgas pre-treatment radiation monitor reading > the Table S1 Dose Rate Limit for the actual indicated offgas flow indicating fuel clad degradation > T.S. allowable limits

OR

2. Reactor coolant sample activity value indicating fuel clad degradation > T.S. allowable limits
   - > 4.0 µCi/gm dose equivalent I-131
     OR
   - > 0.2 µCi/gm dose equivalent I-131 for > 48 hours

Inadvertent Criticality

NOUE — SU10 (1 2 3 4 5 D)

Inadvertent criticality

Emergency Action Level(s):

1. UNPLANNED sustained positive period observed on nuclear instrumentation

TECH SPEC Time Limit Exceeded

NOUE — SU11 (1 2 3 4 5 D)

Inability to reach required operating mode within Technical Specification limits

Emergency Action Level(s):

1. Plant is not brought to required operating mode within Technical Specifications LCO Action Statement time

EMERGENCY CLASSIFICATION USER AID 2 MODES 4, 5, and Defueled

Plant Modes (white boxes indicate applicable modes)	1	Power Operations	2	Startup	3	Hot Shutdown	4	Cold Shutdown	5	Refuel	D	Defueled

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENT — OPERATING MODES: 1 2 3 4 5 D

Radiological Effluents

GENERAL EMERGENCY

AG1 1 2 3 4 5 D

Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity > 1000 mR TEDE or 5000 mR thyroid CDE for the actual or projected duration of the release using actual meteorology

Emergency Action Level: (1 or 2 or 3)

***NOTE:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, the classification should be based on EAL #2 instead of EAL #1. Do not delay declaration awaiting dose assessment results.*

1. VALID reading on any of the radiation monitors in Table R1 > the GENERAL EMERGENCY reading for ≥ 15 minutes

OR

2. Dose assessment using actual meteorology indicates doses > 1000 mR TEDE or 5000 mR thyroid CDE at or beyond the SITE BOUNDARY

OR

3. Field survey results indicate closed window dose rates > 1000 mR/hr expected to continue for ≥ 60 minutes; or analyses of field survey samples indicate thyroid CDE > 5000 mR for one hour of inhalation, at or beyond the SITE BOUNDARY

SITE AREA EMERGENCY

AS1 1 2 3 4 5 D

Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity > 100 mR TEDE or 500 mR thyroid CDE for the actual or projected duration of the release

Emergency Action Level(s) (1 or 2 or 3)

***NOTE:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, the classification should be based on EAL #2 instead of EAL #1. Do not delay declaration awaiting dose assessment results.*

1. VALID reading on any of the radiation monitors in Table R1 > the SITE AREA EMERGENCY reading for ≥ 15 minutes

OR

2. Dose assessment using actual meteorology indicates doses > 100 mR TEDE or 500 mR thyroid CDE at or beyond the SITE BOUNDARY

OR

3. Field survey results indicate closed window dose rates > 100 mR/hr expected to continue for ≥ 60 minutes; or analyses of field survey samples indicate thyroid CDE > 500 mR for one hour of inhalation, at or beyond the SITE BOUNDARY

ALERT

AA1 1 2 3 4 5 D

Any release of gaseous or liquid radioactivity to the environment > 200 times the ODCM limit for ≥ 15 minutes

Emergency Action Level(s): (1 or 2 or 3)

***NOTE:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.*

1. VALID reading on any of the radiation monitors in Table R1 > the Alert reading for ≥ 15 minutes

OR

2. For RMS-RE107 effluent monitor:
   EITHER
   VALID reading > 200 times the alarm setpoint established by a current radioactivity discharge permit for ≥ 15 minutes
   OR
   VALID reading > 1.27E-01 µCi/ml for ≥ 15 minutes

OR

3. Confirmed sample analysis for gaseous or liquid releases indicate concentrations or release rates > 200 times the ODCM limit for ≥ 15 minutes

NOUE

AU1 1 2 3 4 5 D

Any release of gaseous or liquid radioactivity to the environment > 2 times the ODCM limit for ≥ 60 minutes

Emergency Action Level(s): (1 or 2 or 3)

***NOTE:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.*

1. VALID reading on any of the radiation monitors in Table R1 > the NOUE reading for ≥ 60 minutes

OR

2. VALID reading on RMS-RE107 effluent monitor > 2 times the alarm setpoint established by a current radioactivity discharge permit for ≥ 60 minutes

OR

3. Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 2 times the ODCM limit for ≥ 60 minutes

Abnormal Radiation Levels

AA2 1 2 3 4 5 D

Damage to irradiated fuel or loss of water level that has resulted or will result in the uncovering of irradiated fuel outside the reactor vessel

Emergency Action Level(s): (1 or 2)

1. A water level drop in the reactor refueling cavity, spent fuel pool or fuel transfer canal that will result in irradiated fuel becoming uncovered

OR

2. A VALID reading on any of the following radiation monitors due to damage to irradiated fuel or loss of water level:
   RMS-RE140 2000 mR/hr
   RMS-RE141 2000 mR/hr
   RMS-RE192 2000 mR/hr
   RMS-RE193 2000 mR/hr
   RMS-RE5A 1.64E+03 µCi/sec
   RMS-RE5B (GE) 5.29E-04 µCi/ml

AU2 1 2 3 4 5 D

UNPLANNED rise in plant radiation levels

Emergency Action Level(s): (1 or 2)

1. a. UNPLANNED water level drop in a reactor refueling pathway as indicated by any of the following:
   - Water level drop in the reactor refueling cavity, spent fuel pool, or fuel transfer canal indication on Control Room Panel 870
   - Personnel observation by visual or remote means.

   AND

   b. UNPLANNED VALID area radiation monitor alarm on any of the following:
   RMS-RE140
   RMS-RE141
   RMS-RE192
   RMS-RE193

OR

2. UNPLANNED VALID area radiation monitor readings or survey results indicate a rise by a factor of 1000 over normal* levels

***NOTE:** For area radiation monitors with ranges incapable of measuring 1000 times normal* levels, classification shall be based on VALID full scale indications unless surveys confirm that area radiation levels are below 1000 times normal* within 15 minutes of the area radiation monitor indications going full scale.*

* Normal can be considered the highest reading in the past 24 hours excluding the current peak value.

AA3 1 2 3 4 5 D

Rise in radiation levels within the facility that impedes operation of systems required to maintain plant safety functions

Emergency Action Level: (s)

1. Dose rate > 15 mR/hr in any of the following areas requiring continuous occupancy to maintain plant safety functions:
   Main Control Room
   CAS

Table R1 EAL THRESHOLD

Method	GENERAL DRMS	GENERAL Threshold	SITE AREA DRMS	SITE AREA Threshold	ALERT DRMS	ALERT Threshold	NOUE DRMS	NOUE Threshold
Main Plant Primary	4GE125	4.70E+08 µCi/sec	4GE125	4.70E+07 µCi/sec	4GE125	3.06E+07 µCi/sec	4GE125	3.06E+05 µCi/sec
Main Plant Secondary	N/A		N/A		1GE126	2.82E-01 µCi/ml	1GE126	5.26E-03 µCi/ml
FB Vent Primary	4GE005	6.70E+07 µCi/sec	4GE005	6.70E+06 µCi/sec	4GE005	2.19E+06 µCi/sec	4GE005	2.19E+04 µCi/sec
FB Vent Secondary	N/A		N/A		5GE005	2.82E-01 µCi/ml	5GE005	4.65E-03 µCi/ml
RW Vent Primary	N/A		N/A		4GE006	2.58E+06 µCi/sec	4GE006	2.58E+04 µCi/sec
RW Vent Secondary					5GE006	6.84E-02 µCi/ml	5GE006	6.84E-04 µCi/ml

EVENTS RELATED TO ISFSI — OPERATING MODES: 1 2 3 4 5 D

Confinement Boundary damage/ loss

NOUE

E-HU1 1 2 3 4 5 D

Damage to a loaded cask CONFINEMENT BOUNDARY

Emergency Action Level(s):

1. Damage to a loaded cask CONFINEMENT BOUNDARY

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY — OPERATING MODES: 1 2 3 4 5 D

Security

HG1 1 2 3 4 5 D

HOSTILE ACTION resulting in loss of physical control of the facility

Emergency Action Level(s): (1 or 2)

1. A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions

OR

2. A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and IMMINENT fuel damage is likely for a freshly off-loaded reactor core in pool

HS1 1 2 3 4 5 D

HOSTILE ACTION within the PROTECTED AREA

Emergency Action Level(s):

1. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the RBS security shift supervision

HA1 1 2 3 4 5 D

HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat

Emergency Action Level(s): (1 or 2)

1. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the RBS security shift supervision

OR

2. A validated notification from NRC of an airliner attack threat within 30 minutes of the site

HU1 1 2 3 4 5 D

Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant

Emergency Action Level(s):(1 or 2 or 3)

1. A SECURITY CONDITION that does NOT involve a HOSTILE ACTION as reported by the RBS security shift supervision

OR

2. A credible site specific security threat notification

OR

3. A validated notification from NRC providing information of an aircraft threat

Discretionary

HG2 1 2 3 4 5 D

Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency

Emergency Action Level(s):

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area

HS2 1 2 3 4 5 D

Other conditions exist which in the judgment of the Emergency Director warrant declaration of a SITE AREA EMERGENCY

Emergency Action Level(s):

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts: (1) toward site personnel or equipment that could lead to the likely failure of or: (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the SITE BOUNDARY

HA2 1 2 3 4 5 D

Other conditions exist which in the judgment of the Emergency Director warrant declaration of an ALERT

Emergency Action Level(s):

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels

HU2 1 2 3 4 5 D

Other conditions exist which in the judgment of the Emergency Director warrant declaration of a NOUE

Emergency Action Level(s):

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs

Control Room Evacuation

HS3 1 2 3 4 5 D

Control room evacuation has been initiated and plant control cannot be established

Emergency Action Level(s):

1. a. Control room evacuation has been initiated
   AND
   b. Control of the plant cannot be established in accordance with AOP-0031, Shutdown from Outside the Main Control Room, within 15 minutes

HA3 1 2 3 4 5 D

Control room evacuation has been initiated

Emergency Action Level(s):

1. AOP-0031, Shutdown from Outside the Main Control Room requires Control Room evacuation

Fire

HA4 1 2 3 4 5 D

FIRE or EXPLOSION affecting the operability of plant safety systems required to establish or maintain safe shutdown

Emergency Action Level(s):

1. FIRE or EXPLOSION resulting in VISIBLE DAMAGE to any of the structures or areas in Table H2 containing safety systems or components or Control Room indication of degraded performance of those safety systems

HU4 1 2 3 4 5 D

FIRE within the PROTECTED AREA not extinguished within 15 minutes of detection or EXPLOSION within the PROTECTED AREA

Emergency Action Level(s): (1 or 2)

***NOTE:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the duration has exceeded, or will likely exceed, the applicable time.*

1. FIRE not extinguished within 15 minutes of Control Room notification or verification of a Control Room FIRE alarm in any Table H2 structure or area

OR

2. EXPLOSION within the PROTECTED AREA

Toxic or Flammable Gasses

HA5 1 2 3 4 5 D

Access to a VITAL AREA is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of operable equipment required to maintain safe operations or safely shutdown the reactor

Emergency Action Level(s):

***NOTE:** If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.*

1. Access to Main Control Room, Auxiliary Building, or 95' Control Building is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shutdown the reactor

HU5 1 2 3 4 5 D

Release of toxic, corrosive, asphyxiant or flammable gases deemed detrimental to NORMAL PLANT OPERATIONS

Emergency Action Level(s) (1 or 2)

1. Toxic, corrosive, asphyxiant or flammable gases in amounts that have or could adversely affect NORMAL PLANT OPERATIONS

OR

2. Report by West Feliciana Parish for evacuation or sheltering of site personnel based on an offsite event

Natural or Destructive Phenomena

Table H2 Structures Containing Functions or Systems Required for Safe Shutdown

Reactor Building	Standby Cooling Tower
Auxiliary Building	Diesel Generator Building
Control Building	Tunnels (B, D, E, F, G)
Fuel Building	

HA6 1 2 3 4 5 D

Natural or destructive phenomena affecting VITAL AREAS

Emergency Action Level(s): (1 or 2 or 3 or 4 or 5 or 6)

1.a. Seismic event > Operating Basis Earthquake (OBE) as indicated by:
   Annunciator "Seismic Tape Recording System Start" (P680-02A-D06)
   AND
   Event Indicator on ERS-NBI-102 is white
   AND
   Receipt of **EITHER 1 OR 2**:
   1. Annunciator "Seismic Event High" (P680-02A-C06)
   2. Annunciator "Seismic Event High-High" (P680-02A-B06) **AND** amber light(s) on panel NBI-101

   AND

   b. Earthquake confirmed by any of the following:
   - Earthquake is felt in plant
   - National Earthquake Center
   - Control Room indication of degraded performance of systems required for safe shutdown of the plant

OR

2. Tornado striking resulting in VISIBLE DAMAGE to any of the Table H2 structures or areas containing safety systems or components or Control Room indication of degraded performance of those safety systems

OR

3. Internal flooding in Auxiliary Building 70 ft elevation resulting in an electrical shock hazard that precludes access to operate or monitor safety equipment or Control Room indication of degraded performance of those safety systems

OR

4. Turbine failure-generated PROJECTILES resulting in VISIBLE DAMAGE to or penetration of any of the Table H2 structures or areas containing safety systems or components or Control Room indication of degraded performance of those safety systems

OR

5. Vehicle crash resulting in VISIBLE DAMAGE to any of the Table H2 structures or areas containing safety systems or components or Control Room indication of degraded performance of those safety systems

OR

6. Hurricane or high SUSTAINED wind conditions ≥ 74 mph within the PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the Table H2 structures or areas containing safety systems or components or Control Room indication of degraded performance of those safety systems

HU6 1 2 3 4 5 D

Natural or destructive phenomena affecting the PROTECTED AREA

Emergency Action Level(s):(1 or 2 or 3 or 4 or 5)

1. Seismic event identified by any 2 of the following:
   - Seismic event confirmed by activated seismic switch as indicated by receipt of **EITHER a OR b**:
     a. Annunciator "Seismic Tape Recording SYS Start" (P680-02A-D06)
     b. Event Indicator on ERS-NBI-102 is white
   - Earthquake felt in plant
   - National Earthquake Center

OR

2. Tornado striking within the PROTECTED AREA boundary

OR

3. Internal flooding that has the potential to affect safety related equipment required by Technical Specifications for the current operating mode in any Table H1 area

OR

4. Turbine failure resulting in casing penetration or damage to turbine or generator seals

OR

5. Severe weather or hurricane conditions with indication of SUSTAINED high winds ≥ 74 mph within the PROTECTED AREA boundary

Table H1 Uncontrolled Flooding Threshold Area Water Level

Affected Location / Parameter	Max Safe Operating Value / Indicator
Aux Bldg Crescent Area 70' EL	6 inches above floor (must be verified locally)
HPCS Room 70'EL	4 inches above floor (P870-51A-G4)
RHR A Room 70'EL	4 inches above floor (P870-51A-G4)
RHR B Room 70'EL	4 inches above floor (P870-51A-G4)
RHR C Room 70'EL	4 inches above floor (P870-51A-G4)
LPCS Room 70'EL	4 inches above floor (P870-51A-G4)
RCIC Room 70'EL	4 inches above floor (P870-51A-G4)

COLD SHUTDOWN / REFUELING — OPERATING MODES: 1 2 3 4 5 D

Loss of RCS/RPV Inventory

CG1 1 2 3 4 5 D

Loss of RCS/RPV inventory affecting fuel clad integrity with containment challenged

Emergency Action Level(s): (1 or 2)

1. a. RPV level < -162 inches (TAF) for ≥ 30 minutes
   AND
   b. Any containment challenge indication in Table C1

OR

2. a. RCS level cannot be monitored for ≥ 30 minutes with core uncovery indicated by any of the following:
   - RMS-RE16 reading > 100 R/hr
   - Erratic source range monitor indication
   - Unexplained rise in floor or equipment sump level, Suppression Pool level, vessel make-up rate or observation of leakage or inventory loss

   AND

   b. Any containment challenge indication in Table C1

Table C1 Containment Challenge Indications

- CONTAINMENT CLOSURE not established
- Explosive mixture inside containment
- UNPLANNED rise in containment pressure
- Secondary containment area radiation monitor above EOP Max Safe Operating Value below:

Area	DRMS Grid 2	Max Safe Operating Value
RHR Equip Rm A	1213	9.5E+03 mR/hr
RHR Equip Rm B	1214	9.5E+03 mR/hr
RHR Equip Rm C	1215	9.5E+03 mR/hr

CS1 1 2 3 4 5 D

Loss of RCS/RPV inventory affecting core decay heat removal capability

Emergency Action Level(s): (1 or 2 or 3)

***NOTE:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. With CONTAINMENT CLOSURE not established, UNPLANNED RPV level < -49 inches

OR

2. With CONTAINMENT CLOSURE established, RPV level < -162 inches (TAF)

OR

3. RCS level cannot be monitored for ≥ 30 minutes with a loss of RCS inventory as indicated by any of the following:
   - RMS-RE16 reading > 100 R/hr
   - Erratic Source Range Monitor indication
   - Unexplained rise in floor or equipment sump level, Suppression Pool level, vessel make-up rate or observation of leakage or inventory loss

CA1 1 2 3 4 5 D

Loss of RCS inventory

Emergency Action Level(s): (1 or 2)

***NOTE:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. UNPLANNED loss of RCS inventory as indicated by RPV level < -43 inches (Level 2)

OR

2. RCS level cannot be monitored for ≥ 15 minutes with a loss of RCS inventory as indicated by an unexplained rise in floor or equipment sump level, Suppression Pool level, vessel make-up rate or observation of leakage or inventory loss

CU1 1 2 3 4 5 D

RCS leakage

Emergency Action Level(s):

***NOTE:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. RCS leakage results in the inability to maintain or restore RPV level > +9.7 inches (Level 3) for ≥ 15 minutes

CU2 1 2 3 4 5 D

UNPLANNED loss of RCS/RPV inventory

Emergency Action Level(s): (1 or 2)

***NOTE:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. UNPLANNED RCS level drop as indicated by either of the following:
   a. RCS water level drop below the RPV flange for ≥ 15 minutes when the RCS level band is established above the RPV flange
   OR
   b. RCS water level drop below the RPV level band for ≥ 15 minutes when the RCS level band is established below the RPV flange

OR

2. RCS level cannot be monitored with a loss of RCS inventory as indicated by an unexplained rise in floor or equipment sump level, Suppression Pool level, vessel make-up rate or observation of leakage or inventory loss

Loss of Decay Heat Removal

CA3 1 2 3 4 5 D

Inability to maintain plant in cold shutdown

Emergency Action Level(s):(1or 2)

1. An UNPLANNED event results in RCS temperature > 200 °F > the specified duration in Table C2

OR

2. An UNPLANNED event results in RCS pressure rise > 10 psig due to a loss of RCS cooling

Table C2 RCS Reheat Duration Thresholds

RCS	Containment Closure	Duration
Intact	N/A	60 minutes*
Not intact	Established	20 minutes*
	Not Established	0 minutes

*If and RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.

CU3 1 2 3 4 5 D

UNPLANNED loss of decay heat removal capability with irradiated fuel in the RPV

Emergency Action Level(s): (1 or 2)

***NOTE:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. An UNPLANNED event results in RCS temperature exceeding 200°F

OR

2. Loss of all RCS temperature and RCS/RPV level indication for ≥ 15 minutes

Loss of AC Power

CA5 1 2 3 4 5 D

Loss of all offsite and all onsite AC power to emergency busses for ≥ 15 minutes

Emergency Action Level(s):

***NOTE:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. Loss of all offsite and all onsite AC power to Div I and Div II ENS busses for ≥ 15 minutes

CU5 1 2 3 4 5 D

AC power capability to emergency busses reduced to a single power source for ≥ 15 minutes such that any additional single failure would result in station blackout

Emergency Action Level(s):

***NOTE:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. a. AC power capability to Div I and Div II ENS busses reduced to a single power source for ≥ 15 minutes
   AND
   b. Any additional single power source failure will result in station blackout

Loss of DC Power

CU6 1 2 3 4 5 D

Loss of required DC power for ≥ 15 minutes

Emergency Action Level(s):

***NOTE:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. < 105 VDC on required Vital DC busses for ≥ 15 minutes

Inadvertent Criticality

CU7 1 2 3 4 5 D

Inadvertent criticality

Emergency Action Level(s):

1. UNPLANNED sustained positive period observed on nuclear instrumentation

Loss of Communications

CU8 1 2 3 4 5 D

Loss of all onsite or offsite communications capabilities

Emergency Action Level(s): (1 or 2)

1. Loss of all of the following onsite communication methods affecting the ability to perform routine operations:
   Plant radio system
   Plant paging system
   Sound powered phones
   In-plant telephones

OR

2. Loss of all of the following offsite communication methods affecting the ability to perform offsite notifications:
   All telephones
   NRC phones
   State of Louisiana Radio
   Offsite notification system and hotline

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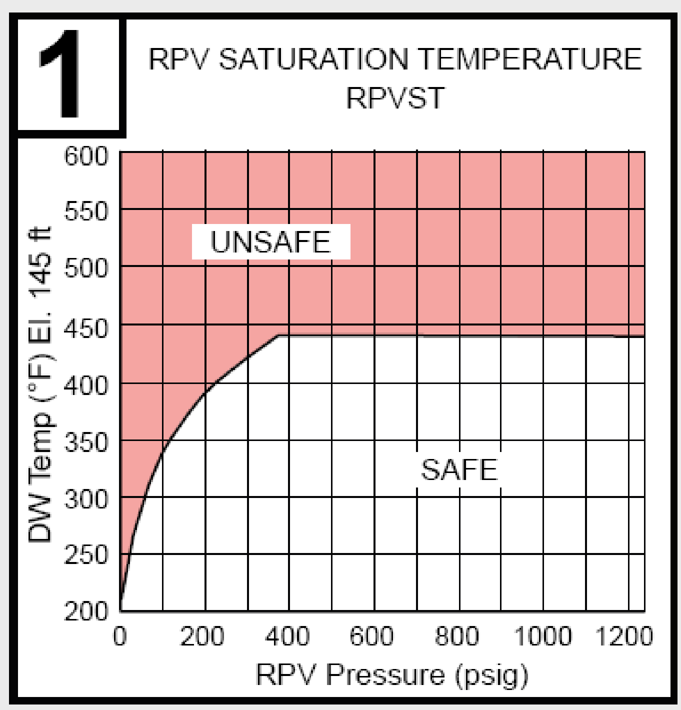

CAUTIONS

CAUTION #1

RPV water level indications are affected by instrument run temperatures and RPV pressure:

1. If the temp near **any** instrument run is above the RPVST (Fig. 1), the instrument may be unreliable due to boiling in the run

   Max DW Temp at El. 145 ft, 440°F

   Max CTMT Temp at El. 119 ft, 200°F

2. Each of the following instruments may be used to determine RPV water level only when the instrument reads above the Minimum Indicated Level associated with the highest temperature near the instrument reference leg vertical run.

Containment Temperature (°F) El. 119 ft

Containment Temperature (°F)	Narrow	Wide	Fuel Zone
100 – 200	1 in.	-142 in.	-310 in.
70 – 100	0 in.	-159 in.	-310 in.

Drywell Temperature (°F) El. 145 ft

Drywell Temperature (°F)	Upset	Shutdown
300 – 440	53 in.	55 in.
200 – 300	20 in.	29 in.
100 – 200	2 in.	16 in.

AVAILABLE SAFE SHUTDOWN SYSTEMS

Fire Area:	AB-1 West Side Crescent	AB-2 HPCS & HPCS Hatch	AB-3 RHR B	AB-4 RHR C, RCIC & RWCU	AB-5 RHR A	AB-6 LPCS & LPCS Hatch	AB-7 D-Tunnel	AB-10 MS Tunnel South End	AB-13 Standby Gas Treatment B
Electrical Power									
Div. I		X	X	X	X			X	X
Div. II	X		X		X	X	X	X	X
Div. III		X	X	X	X			X	X
Offsite									
RPV Level Control									
HPCS (Div. III)			X	X	X			X	X
RCIC (Div. I)			X					X	X
LPCS (Div. I)		X	X	X	X			X	X
LPCI-RHR C (Div. II)	X				X	X	X	X	X
RPV Pressure Control									
Div. I SRVs		X	X	X	X			X	X
Div. II SRVs	X		X		X	X	X	X	X
Decay Heat Removal									
RHR A Suppression Pool Cooling		X	X	X				X	X
RHR A Alternate Shutdown Cooling		X	X	X				X	X
RHR A Normal Shutdown Cooling									
RHR B Suppression Pool Cooling	X				X	X	X	X	X
RHR B Alternate Shutdown Cooling	X				X	X	X	X	X
RHR B Normal Shutdown Cooling									
Mech./Environ. Support									
Div. I Standby Service Water		X	X	X	X			X	X
Div. I HVAC Systems		X	X	X	X			X	X
Div. I Containment/RPV Monitoring		X	X	X	X			X	X
Div. II Standby Service Water	X		X		X	X	X	X	X
Div. II HVAC Systems	X		X		X	X	X	X	X
Div. II Containment/RPV Monitoring	X		X		X	X	X	X	X
Normal Service Water									
Equipment availability depends upon fire location in Fire Area									
Local Manual Action Required?	N	Y	Y	N	N	N	Y	N	N

AVAILABLE SAFE SHUTDOWN SYSTEMS

Fire Area:	AB-14	AB-15	AB-17	AB-18	C-1	C-2	C-3	C-4	
	Standby Gas Treatment A	East Side Crescent	Cont. Vent Flt. Train Rm.	D-Tunnel Cable Chase	Cable Chase I (NE)	Cable Chase II (SE)	Rm. North of ACU Rooms	ACU Rooms East	West
Electrical Power									
Div. I		X	X	X	X	X	X	X	
Div. II	X		X	partial			X		X
Div. III		X	X	X			X	X	X
Offsite									
RPV Level Control									
HPCS (Div. III)			X	X			X	X	X
RCIC (Div. I)			X		X		X	X	
LPCS (Div. I)		X	X	X	X	X	X	X	
LPCI-RHR C (Div. II)	X		X				X		X
RPV Pressure Control									
Div. I SRVs		X	X	X	X	X	X	X	
Div. II SRVs	X		X				X		X
Decay Heat Removal									
RHR A Suppression Pool Cooling		X	X	X	X	X	X	X	
RHR A Alternate Shutdown Cooling		X	X	X	X	X	X	X	
RHR A Normal Shutdown Cooling		X							
RHR B Suppression Pool Cooling	X		X				X		X
RHR B Alternate Shutdown Cooling	X		X				X		X
RHR B Normal Shutdown Cooling									
Mech./Environ. Support									
Div. I Standby Service Water		X	X	X	1 pump	1 pump	X	X	
Div. I HVAC Systems		X	X	X	X	X	X	X	
Div. I Containment/RPV Monitoring		X	X	X	X	X	X	X	
Div. II Standby Service Water	X		X				X		X
Div. II HVAC Systems	X		X				X		X
Div. II Containment/RPV Monitoring	X		X				X		X
Normal Service Water									
Equipment availability depends upon fire location in Fire Area								X	
Local Manual Action Required?	N	Y	N	Y	Y	Y	N	N	N

AVAILABLE SAFE SHUTDOWN SYSTEMS

Fire Area:	C-5	C-6	C-7	C-9	C-10	C-11	C-13E	C-13W	C-14
	Cable Area S. of ACU	Remainder of El. 70'	Post-Accid. Rad. Monit.	Cable Chase III (NW)	Cable Chase IV (SW)	Vestibule (NW Corner)	HVK Chiller East Side	HVK Chiller West Side	Div. II Stby. Swgr. Rm.
Electrical Power									
Div. I	X	X					X		X
Div. II			X	X	X	X		X	
Div. III						X	X	X	
Offsite									
RPV Level Control									
HPCS (Div. III)						X	X	X	
RCIC (Div. I)	X						X		X
LPCS (Div. I)	X	X					X		X
LPCI-RHR C (Div. II)			X	X	X	X		X	
RPV Pressure Control									
Div. I SRVs	X	X					X		X
Div. II SRVs			X	X	X	X		X	
Decay Heat Removal									
RHR A Suppression Pool Cooling	X	X					X		X
RHR A Alternate Shutdown Cooling	X	X					X		X
RHR A Normal Shutdown Cooling									
RHR B Suppression Pool Cooling			X	X	X	X		X	
RHR B Alternate Shutdown Cooling			X	X	X	X		X	
RHR B Normal Shutdown Cooling									
Mech./Environ. Support									
Div. I Standby Service Water	1 pump	1 pump					X		1 pump
Div. I HVAC Systems	X	X					X		X
Div. I Containment/RPV Monitoring	X	X					X		X
Div. II Standby Service Water			X	X	X	X		X	
Div. II HVAC Systems			X	X	X	X		X	
Div. II Containment/RPV Monitoring			X	X	X	X		X	
Normal Service Water									
Equipment availability depends upon fire location in Fire Area									
Local Manual Action Required?	Y	Y	N	N	N	N	N	N	Y

AVAILABLE SAFE SHUTDOWN SYSTEMS

Fire Area:	C-15	C-16	C-17	C-18	C-19	C-21	C-22	C-24	C-27
	Div. I Stby. Swgr. Rm.	Remote S/D Room	Control Bldg. Vent. Room	ENB A Batt. & Inv/Chgr.	ENB B Batt. & Inv/Chgr.	HPCS Batt. & Chgr. Rms.	HPCS Swgr. Rm.	Remainder of El. 116'	East Pipe Chase
Electrical Power									
Div. I			X		X	X			X
Div. II	X	X		X		X	X	X	
Div. III			X	X	X				X
Offsite									
RPV Level Control									
HPCS (Div. III)				X	X				X
RCIC (Div. I)			X		X	X			X
LPCS (Div. I)					X	X			X
LPCI-RHR C (Div. II)	X	X		X		X	X	X	
RPV Pressure Control									
Div. I SRVs			X		X	X			X
Div. II SRVs	X	X		X		X	X	X	
Decay Heat Removal									
RHR A Suppression Pool Cooling			X		X	X			X
RHR A Alternate Shutdown Cooling					X	X			X
RHR A Normal Shutdown Cooling			X						
RHR B Suppression Pool Cooling	X	X		X		X	X	X	
RHR B Alternate Shutdown Cooling	X	X		X		X	X	X	
RHR B Normal Shutdown Cooling									
Mech./Environ. Support									
Div. I Standby Service Water			X		X	1 pump			X
Div. I HVAC Systems			X		X	X			X
Div. I Containment/RPV Monitoring			X		X	X			X
Div. II Standby Service Water	X	X		X		X	X	X	
Div. II HVAC Systems	X	X		X		X	X	X	
Div. II Containment/RPV Monitoring	X	X		X		X	X	X	
Normal Service Water									
Equipment availability depends upon fire location in Fire Area									
Local Manual Action Required?	N	N	Y	N	N	Y	N	Y	N

AVAILABLE SAFE SHUTDOWN SYSTEMS

Fire Area:	C-29	C-30	DG-1	DG-2	DG-3	DG-4	DG-5	DG-6	DG-7
	NW Vestibule	Stairwell No. 1	Div. II Fuel Storage Tk.	Div. III Fuel Storage Tk.	Div. I Fuel Storage Tk.	Div. II Diesel Gen. Room	Div. III Diesel Gen. Room	Div. I Diesel Gen. Room	Div.II Diesel Elec. Tunnel
Electrical Power									
Div. I	X	X	X	X		X			X
Div. II	X	X		X	X		X	X	
Div. III	X	X	X		X	X		X	X
Offsite									
RPV Level Control									
HPCS (Div. III)	X	X	X		X	X		X	X
RCIC (Div. I)	X	X	X	X		X			X
LPCS (Div. I)	X	X	X	X		X			X
LPCI-RHR C (Div. II)	X	X		X	X		X	X	
RPV Pressure Control									
Div. I SRVs	X	X	X	X		X			X
Div. II SRVs	X	X		X	X		X	X	
Decay Heat Removal									
RHR A Suppression Pool Cooling	X	X	X	X		X			X
RHR A Alternate Shutdown Cooling	X	X	X	X		X			X
RHR A Normal Shutdown Cooling									
RHR B Suppression Pool Cooling	X	X		X	X		X	X	
RHR B Alternate Shutdown Cooling	X	X		X	X		X	X	
RHR B Normal Shutdown Cooling									
Mech./Environ. Support									
Div. I Standby Service Water	X	X	X	1 pump		X			X
Div. I HVAC Systems	X	X	X	X		X			X
Div. I Containment/RPV Monitoring	X	X	X	X		X			X
Div. II Standby Service Water	X	X		X	X		X	X	
Div. II HVAC Systems	X	X		X	X		X	X	
Div. II Containment/RPV Monitoring	X	X		X	X		X	X	
Normal Service Water									
Equipment availability depends upon fire location in Fire Area									
Local Manual Action Required?	N	N	N	N	N	N	N	N	N

AVAILABLE SAFE SHUTDOWN SYSTEMS

Fire Area:	ET-1	ET-2	ET-3	ET-4	ET-5	ET-6	FB-1	MG-1
	B-Tunnel East (DG)	B-Tunnel West (FB)	T-Tunnel West El.67'6"	T-Tunnel West El.95'	B-Tunnel South	C-Tunnel	Fuel Building	LFMG Building
Electrical Power								
Div. I		X	X	X	X	X	X	X
Div. II	X		X	X	X	X	X	X
Div. III		X	X	X	X	X	X	X
Offsite								
RPV Level Control								
HPCS (Div. III)			X	X	X	X	X	X
RCIC (Div. I)			X	X	X	X	X	X
LPCS (Div. I)		X	X	X	X	X	X	X
LPCI-RHR C (Div. II)	X		X	X	X	X	X	X
RPV Pressure Control								
Div. I SRVs		X	X	X	X	X	X	X
Div. II SRVs	X		X	X	X	X	X	X
Decay Heat Removal								
RHR A Suppression Pool Cooling		X	X	X	X	X	X	X
RHR A Alternate Shutdown Cooling		X	X	X	X	X	X	X
RHR A Normal Shutdown Cooling								
RHR B Suppression Pool Cooling	X		X	X	X	X	X	X
RHR B Alternate Shutdown Cooling	X		X	X	X	X	X	X
RHR B Normal Shutdown Cooling								
Mech./Environ. Support								
Div. I Standby Service Water		X	X	X	X	X	X	X
Div. I HVAC Systems		X	X	X	X	X	X	X
Div. I Containment/RPV Monitoring		X	X	X	X	X	X	X
Div. II Standby Service Water	X		X	X	X	X	X	X
Div. II HVAC Systems	X		X	X	X	X	X	X
Div. II Containment/RPV Monitoring	X		X	X	X	X	X	X
Normal Service Water								
Equipment availability depends upon fire location in Fire Area								
Local Manual Action Required?	N	Y	N	N	N	N	Y	N

AVAILABLE SAFE SHUTDOWN SYSTEMS

Fire Area:	PH-1	PH-2	PH-3	PH-4	PH-5	PT-1	PT-2
	Stby. Cooling Twr. (Div. I)	Stby. Cooling Twr. (Div. II)	Remote Air Intake Room	Div. II SCT Fans	Div. I SCT Fans	E, F & G-Tunnels	E-Tunnel (SWP/CCP)
Electrical Power							
Div. I		X	X	X	X	X	X
Div. II	X		X	X	X		X
Div. III	X	X	X	X	X	X	X
Offsite						X	
RPV Level Control							
HPCS (Div. III)	X	X	X	X	X	X	X
RCIC (Div. I)		X	X	X	X	X	X
LPCS (Div. I)		X	X	X	X	X	X
LPCI-RHR C (Div. II)	X		X	X	X		X
RPV Pressure Control							
Div. I SRVs		X	X	X	X	X	X
Div. II SRVs	X		X	X	X		X
Decay Heat Removal							
RHR A Suppression Pool Cooling		X	X	X	X	X	X
RHR A Alternate Shutdown Cooling		X	X	X	X	X	X
RHR A Normal Shutdown Cooling							
RHR B Suppression Pool Cooling	X		X	X	X		X
RHR B Alternate Shutdown Cooling	X		X	X	X		X
RHR B Normal Shutdown Cooling							
Mech./Environ. Support							
Div. I Standby Service Water		X	X	X			X
Div. I HVAC Systems		X	X	X	X	X	X
Div. I Containment/RPV Monitoring		X	X	X	X	X	X
Div. II Standby Service Water	X		X		X		X
Div. II HVAC Systems	X		X	X	X		X
Div. II Containment/RPV Monitoring	X		X	X	X		X
Normal Service Water						X	
Equipment availability depends upon fire location in Fire Area							
Local Manual Action Required?	N	N	N	N	N	N	N

AVAILABLE SAFE SHUTDOWN SYSTEMS

Fire Area:	RC-2	RC-3	RC-4	RC-6		RDW-1	
	MS Tunnel, N of Imp. Wall	Reactor Bldg East All Elev	Reactor Bldg West All Elev	Annulus Area East	West	Drywell East	West
Electrical Power							
Div. I	X	X		X		X	
Div. II	X		X		X		X
Div. III	X			X	X	X	X
Offsite							
RPV Level Control							
HPCS (Div. III)	X	X		X	X	X	X
RCIC (Div. I)							
LPCS (Div. I)	X	X	X	X		X	X
LPCI-RHR C (Div. II)	X		X		X		X
RPV Pressure Control							
Div. I SRVs	X	X		X		X	
Div. II SRVs	X		X		X		X
Decay Heat Removal							
RHR A Suppression Pool Cooling	X	X		X		X	
RHR A Alternate Shutdown Cooling	X	X		X		X	
RHR A Normal Shutdown Cooling							
RHR B Suppression Pool Cooling	X		X		X		X
RHR B Alternate Shutdown Cooling	X		X		X		X
RHR B Normal Shutdown Cooling							
Mech./Environ. Support							
Div. I Standby Service Water	X	X		X		X	
Div. I HVAC Systems	X	X		X		X	
Div. I Containment/RPV Monitoring	X	X		X		X	
Div. II Standby Service Water	X		X		X		X
Div. II HVAC Systems	X		X		X		X
Div. II Containment/RPV Monitoring	X		X		X		X
Normal Service Water							
Equipment availability depends upon fire location in Fire Area							
Local Manual Action Required?	N	N	N	N	N	N	N

TR 3.4.13 CHEMISTRY

TLCO 3.4.13 The chemistry of the reactor coolant system shall be maintained within the limits specified in Table 3.4.13-1.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. In MODE 1, with the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.4.13-1.	A.1 Restore to within limits.	72 hours
B. Required Action A.1 and associated Completion Time not met. <u>OR</u> Conductivity or chloride concentration exceeds the limit specified in Table 3.4.13-1 while in MODE 1 for > 336 hours in any 365 day period.	B.1 Be in Mode 2.	6 hours
C. In MODE 2 and 3 with the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.4.13-1.	C.1 Restore to within limits.	48 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action C.1 and associated Completion Time not met. OR The identification while in MODE 1, that conductivity exceeds 10 µmho/cm at 25°C or chloride concentration exceeds 0.5 ppm	D.1 Be in Mode 3. AND D.2 Be in Mode 4.	12 hours 36 hours
E. At all times other than MODE 1, 2 or 3, with the conductivity or pH exceeding the limit specified in Table 3.4.13-1.	E.1 Restore the conductivity and pH to within the limit.	72 hours
F. At all times other than MODE 1, 2 or 3, with the chloride concentration exceeding the limit specified in Table 3.4.13-1.	F.1 Restore chloride concentration to within limit.	24 hours
G. Required Action F.1 and associated Completion Time not met.	G.1 Perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system.	Prior to exceeding 200°F RCS temperature.

SURVEILLANCE REQUIREMENTS

------------------------------------NOTE------------------------------------
The reactor coolant shall be determined to be within the specified chemistry limit by performance of the following:

SURVEILLANCE		FREQUENCY
TSR 3.4.13.1	Determine reactor coolant to be within the specified chemistry limit by analyzing a sample of the reactor coolant for chlorides.	72 hours AND --------NOTE------ When conductivity is greater than the limit in Table 3.4.13-1. ------------------ 8 hours
TSR 3.4.13.2	Determine reactor coolant to be within the specified chemistry limit by analyzing a sample of the reactor coolant for conductivity.	72 hours.
TSR 3.4.13.3	----------------NOTE--------------------- Not required to be met when conductivity is $\leq$ 1.0 μmhos/cm @25°C. --- Determine reactor coolant to be within the specified chemistry limit by analyzing a sample of the reactor coolant for pH.	72 hours AND --------NOTE------ When conductivity is greater than the limit in Table 3.4.13-1. ------------------ 8 hours
TSR 3.4.13.4	-----------------NOTE-------------------- Not required to be met when obtaining in-line conductivity measurements per TSR 3.4.13.5 --- Record the conductivity of the reactor coolant.	Continuously

Continued

SURVEILLANCE		FREQUENCY
TSR 3.4.13.5	-------------------NOTE------------------- Not required to be met when the continuous recording conductivity monitor is operable. -- Obtain an in-line conductivity measurement.	4 hours in MODE 1, 2 or 3 AND 24 hours
TSR 3.4.13.6	---------------------NOTE------------------ Not required to be met when obtaining in-line conductivity measurements per TSR 3.4.13.5. -- Perform a CHANNEL CHECK of the continuous conductivity monitor with an in-line flow cell.	7 days AND ------NOTE------- When conductivity is greater than the limit in Table 3.4.13-1. ----------------- 24 hours

TABLE 3.4.13-1
REACTOR COOLANT SYSTEM
CHEMISTRY LIMITS

MODE	CHLORIDES	CONDUCTIVITY (μmhos/cm @25°C)	pH
1	≤ 0.2 ppm	≤ 1.0	5.6 ≤ pH ≤ 8.6
2 and 3	≤ 0.1 ppm	≤ 2.0	5.6 ≤ pH ≤ 8.6
At all other times	≤ 0.5 ppm	≤ 10.0	5.3 ≤ pH ≤ 8.6