QUESTION RO 1

When communicating the status of a Critical Parameter the format should be ____

A. Value – Channel – Trend

B. Parameter – Value – Trend

C. Parameter – Channel – Value

D. Parameter – Channel – Trend

QUESTION RO 1

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	Generic	2.1.17
	Importance Rating	3.9	

K&A: Ability to make accurate, clear, and concise verbal reports.

Generic

Explanation: **Answer B –** IAW NOP-OP-2001, Conduct Of Operations, crew members will communicate the status of a parameter by describing the Parameter, Value, and Trend.

A – Incorrect – The Channel is not required to be identified.

C – Incorrect – The Channel is not required to be identified.

D – Incorrect – The Channel is not required to be identified.

Technical Reference(s): NOP-OP-1002 Rev. 14	Reference Attached: NOP-OP-1002 p. 26

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3039-01-A

Question Source:	Bank #	Clinton 2002 # RO-37
	Modified Bank #	
	New	
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 2

The plant is in a General Emergency.

The EOF is fully staffed and the Emergency Coordinator has been established there.

The __(1)__ has the authority to deviate from the normal clearance process as described in NOP-OP-1001, Clearance/Tagging Program, if compliance would __(2)__ .

	(1)	(2)
A.	Shift Manager	result in significant hazard to personnel
B.	Shift Manager	increase the potential for or severity of damage to safety-related equipment
C.	Emergency Coordinator	result in significant hazard to personnel
D.	Emergency Coordinator	increase the potential for or severity of damage to safety-related equipment

QUESTION RO 2

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	Generic	2.1.26
	Importance Rating	3.4	

K&A: Knowledge of industrial safety procedures (such as rotating equipment, electrical, high temperature, high pressure, caustic, chlorine, oxygen and hydrogen).

Generic

Explanation: **Answer C –** IAW NOP-OP-1001, the Emergency Coordinator has the authority to deviate from the normal clearance process.

A – Incorrect – plausible because in an emergency the Shift Manager is the designated Emergency Coordinator until the EOC is manned and the Shift Manager has turned over Emergency Coordinator responsibilities to the EOC Emergency Coordinator. The second part is correct, one of the two reasons why the EC can deviate from the normal clearance process.

B – Incorrect - plausible because in an emergency the Shift Manager is the designated Emergency Coordinator until the EOC is manned and the Shift Manager has turned over Emergency Coordinator responsibilities to the EOC Emergency Director/Emergency Coordinator. The second part is correct, one of the two reasons why the EC can deviate from the normal clearance process.

D – Incorrect – plausible in that the EC is the one who can authorize deviating from the normal clearance process during an emergency; however, the second part is not one of the reasons for deviating from the normal clearance process. The two reasons are (1) compliance would result in significant hazard to personnel and (2) an increase in the potential for or severity of a radioactive offsite release.

Technical Reference(s): NOP-OP-1001 Rev. 31	Reference Attached: NOP-OP-1001 p. 81

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3304-01-D.15

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 3

Which of the following is a responsibility of the Reactor Operator during core alterations?

A. Monitor SRM count rate and period

B. Authorize commencement of fuel movements

C. Verify required refueling surveillances are current

D. Ensure the Control Room fuel tag board is maintained current

QUESTION RO 3

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	Generic	2.1.40
	Importance Rating	2.8	

K&A: Knowledge of refueling administrative requirements.

Generic

Explanation: **Answer A –** IAW IOI-9, Refueling, P&L # 2.3.2, flux levels and Rx period are monitored during core alterations. This ATC operator fulfills this responsibility by monitoring SRMs.

B – Incorrect – Plausible since the RO is notified when fuel movements are commencing. However, authorizing commencement of fuel movement is the responsibility of the Unit Supervisor.

C – Incorrect – Plausible since the BOP RO using Tech Spec Rounds, checks various conditions during core alts to ensure TS LCOs are met. However, verifying surveillances current and authorizing commencement of fuel movement are the responsibility of the Unit Supervisor.

D – Incorrect – Plausible since the ATC RO is in communication with the tag board when fuel movements are in progress. The control room tag board is outside the horseshoe area and not normally manned by an RO.

Technical Reference(s): IOI-9 Rev. 43 and NOP-OP-1002 Rev. 14	Reference Attached: IOI-9 p. 7 and NOP-OP-1002 p. 40

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3039-01-A

Question Source:	Bank #	Perry 2013 # RO-02
	Modified Bank #	
	New	

Question History: Previous 2 NRC Exams? No

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

10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 4

The plant is shutdown for a refueling outage with the following conditions:

- The Mode switch is locked in the Shutdown position
- Reactor water temperature is 90 °F
- IOI-9 Section 4.5, Transfer to Mode 5 and Reactor Head Removal, is in progress

The plant will enter MODE 5, Refueling, when _____.

A. the Reactor Mode Switch is placed in REFUEL

B. a reactor vessel head closure bolt is not fully tensioned

C. the last reactor vessel head closure bolt is not fully tensioned

D. a reactor vessel head closure bolt is not fully tensioned and the Reactor Mode Switch is placed in REFUEL

QUESTION RO 4

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	Generic	2.2.35
	Importance Rating	3.6	

K&A: Ability to determine Technical Specification Mode of Operation.

Generic

Explanation: **Answer B –** Per T.S. Definitions, Mode 5 is entered when one or more head bolts are detensioned.

A – Incorrect – plausible if the applicant believes the Mode Switch is involved with declaring Mode 5.

C – Incorrect – plausible if the applicant believes that Mode 5 is entered when the last bolt is being detensioned.

D – Incorrect – plausible if the applicant believes that Mode 5 is entered only if in REFUEL position with a head bolt detensioned.

Technical Reference(s): TS 1.1 Rev. Amend. 69	Reference Attached: TS 1.1 p. 1.0-7

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3037-02-2

Question Source:	Bank #	Perry 2009 # RO-05
	Modified Bank #	
	New	

Question History:	Previous 2 NRC Exams?	No

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

10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 5

Refer to attached 208 print for an MOV to answer this question.

Relay K56A contacts R2/M2 are __(1)__ when valve is electrically opening and fail __(2)__ on a loss of power to the relay coil.

Attachment Provided:

	(1)	(2)
A.	open	open
B.	open	closed
C.	closed	open
D.	closed	closed

QUESTION RO 5

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	Generic	2.2.41
	Importance Rating	3.5	

K&A: Ability to obtain and interpret station electrical and mechanical drawings.

Generic

Explanation: **Answer B –** Relays are shown in their "shelf state" or deenergized on electrical prints. That is the "a" contacts are shown open and the "b" contacts are shown closed. Therefore, when this valve is being electrically opened, the R2/M2 pair of contacts must be open since they are in the CLOSE portion of the circuit.

A – Incorrect – Contacts R2/M2 are "b" contacts and fail closed on a loss of coil power.

C – Incorrect – Contacts R2/M2 must be open while the valve is electrically opening since they are in the CLOSE portion of the control circuit. Contacts R2/M2 are "b" contacts and fail closed on a loss of coil power.

D – Incorrect – Contacts R2/M2 must be open while the valve is electrically opening since they are in the CLOSE portion of the control circuit.

Technical Reference(s): 208-013-017 Rev. X and 208-001 Rev. K	Reference Attached: 208-013-017 and 208-001

Proposed references to be provided to applicants during examination: 208-013-017 (modified)

Learning Objective (As available): PRINTRDG_PY-Eng #22

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 6

Which of the following conditions requires the Control Room Operator to verify that a liquid Radwaste discharge has automatically terminated?

A. Discharge Tunnel Service Water low flow.

B. Emergency Service Water Pump B low flow.

C. HPCS ESW Pump Discharge low pressure.

D. Service Water Pump Discharge Headers low pressure.

QUESTION RO 6

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	Generic	2.3.11
	Importance Rating	3.8	

K&A: Ability to control radiation releases.

Generic

Explanation: **Answer A –** The Discharge Tunnel Service Water low flow is the only condition which causes the discharge to automatically terminate and is required to be verified.

B, C, D – plausible because these conditions may reduce dilution flow for the discharge, but the only condition which causes the discharge to automatically terminate and is required to be verified, is Discharge Tunnel Service Water low flow

Technical Reference(s): ARI-H13-P970-01 Rev. 27 & SDM-P41 Rev. 13	Reference Attached: ARI-H13-P970-01 p. 12 & SDM-P41 p. 20

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-Combined-P41-I.1

Question Source:	Bank #	Perry 2007-2 # RO-72
	Modified Bank #	
	New	
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 7

You are required to make a Drywell Entry at 22% reactor power.

Since this is a considered a Very High Radiation Area, you must obtain approval of the ____.

1. Radiation Protection Manager
2. Operations Unit Supervisor
3. Operations Shift Manager
4. Director of Site Operations

A. 1, 2, & 3

B. 2, 3, & 4

C. 1, 2, & 4

D. 1, 3, & 4

QUESTION RO 7

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	Generic	2.3.13
	Importance Rating	3.4	

K&A: 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.

Generic

Explanation: **Answer D –** RP Manager, Shift Manager, and Director Of Site Operations approvals are required for entrance to a Very High Radiation Area IAW NOP-OP-4101

A – Incorrect – Unit Supervisor approval is not required this entry. Plausible since the US is responsible to verify preparations for initial DW entry are complete.

B – Incorrect – Unit Supervisor approval is not required this entry. Plausible since the US is responsible to verify preparations for initial DW entry are complete.

C – Incorrect – Unit Supervisor approval is not required this entry. Plausible since the US is responsible to verify preparations for initial DW entry are complete.

Technical Reference(s): IOI-17 Rev. 23 and NOP-OP-4102 Rev. 18	Reference Attached: IOI-17 p. 3 and NOP-OP-4102 pp. 19-20

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3039-03-J

Question Source:	Bank #	Perry 2013 # RO-07
	Modified Bank #	
	New	
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 8

The plant scrammed from 100% power following an earthquake. The following plant conditions exist:

- Control Rod 30-31 at position 2
- Control Rod 18-31 at position 4
- Drywell Pressure 1.5 psig
- MSIVs closed on high Steam Tunnel Temperature
- Suppression Pool Temperature 94 degrees F
- No valid RPV Level indication

Which of the following choices includes all of Emergency Operating Procedures (EOPs) the Unit Supervisor should be implementing concurrently?

A. EOP-01, RPV Control;
EOP-01-4, RPV Flooding;
EOP-02 Primary Containment Control

B. EOP-01-5, ATWS RPV Control;
EOP-01-4, RPV Flooding;
EOP-03, Secondary Containment

C. EOP-01, RPV Control;
EOP-01-4, RPV Flooding;
EOP-02, Primary Containment Control;
EOP-03, Secondary Containment

D. EOP-01-4, RPV Flooding;
EOP-02, Primary Containment Control;
EOP-03, Secondary Containment

QUESTION RO 8

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	Generic	2.4.2
	Importance Rating	4.5	

K&A: Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.

Generic

Explanation: **Answer D –** The US will enter EOP-01 and immediately transition to EOP-01-5 and exit EOP-01 due to more than 1 control rod above notch 00. The US will also enter EOP-03 due to the high Steam Tunnel Temperature. EOP-01-5, directs the US to GO TO EOP-01-4 due to not having any valid RPV level indication, which also direct entry into EOP-02.

A – Incorrect – The US must enter EOP-01-5, exit EOP-01, and enter EOP-03. Plausible if the applicant does not believe an ATWS has occurred and EOP-03 entry is not yet appropriate.

B – Incorrect – The US must also enter EOP-02 as directed by EOP-01-4. Plausible if the applicant does not recall the required entry into EOP-02.

C – Incorrect – The US must enter EOP-01-5 and exit EOP-01. Plausible if the applicant does not believe an ATWS has occurred.

Technical Reference(s): EOP-01 Bases Rev. 9, EOP-01-4 Bases Rev. 0, EOP-01-5 Bases Rev. 0, EOP-03/04 Bases Rev. 0, & ARI-H13-P601-19-B4 Rev. 21	Reference Attached: EOP-01 Bases p. 7, EOP-01-4 Bases pp. 5-6, EOP-01-5 Bases p. 7, EOP-03/04 Bases p. 9, & ARI-H13-P601-19-B4 p. 33

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3402-03-B, OT-3402-13-B, OT-3402-10-B, and OT-3402-17-B

Question Source:	Bank # Modified Bank # New	Perry 2007-2 # RO-73
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 9

The EOPs contain a CAUTION regarding rapid injection.

Which of the following is a consequence of rapid injection of water into the RPV per EOP Bases?

A. Exceeding 100 °F/hour cooldown rate

B. Fuel damage under ATWS conditions

C. Thermal shock to the feedwater nozzles

D. Feedwater pump trips on high RPV level

QUESTION RO 9

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	Generic	2.4.20
	Importance Rating	3.8	

K&A: Knowledge of the operational implications of EOP warnings, Cautions, and notes.

Generic

Explanation: **Answer B –** The Caution "Rx Period" pertains to for large power excursions and subsequent fuel damage if injection flow is increased rapidly under ATWS conditions.

A – Incorrect – Plausible as EOP Bases state that 100 °F/hr. cooldown rate should not be exceeded if Rapid Depressurization (ED) or AED are not required.

C – Incorrect – Plausible as thermal shock to steam lines and steam driven equipment is a concern if water level rises above the MSL elevation.

D – Incorrect – Plausible as maintaining RPV level < Level 8 to avoid trips of systems (HPCS, RCIC, & FW) needed to maintain RPV.

Technical Reference(s): EOP-01 Chart Rev. I and EOP-01 Bases Rev. 9	Reference Attached: EOP-01 Chart and EOP-01 Bases p. 17

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3402-02-F

Question Source:	Bank #	Columbia 2015 # RO-61
	Modified Bank #	
	New	
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 10

The plant is operating at 100% power when a loss of off-site power occurs. The following plant conditions exist:

- The Division 1 and Division 2 Diesel Generators failed to start
- Bus EH13 is energized from the Division 3 Diesel Generator
- ONI-R10, Loss of AC Power, has been entered
- Limited NLO manpower is available

The ONI-R10 flowchart that should be used is __(1)__ .
And, specifically, regarding Electrical DC Power, the Time Critical or Time Sensitive Action that has priority is __(2)__ .

	(1)	(2)
A.	ONI-R10-1, Loss of Off-Site Power	Cross-tying of Unit1 and Unit 2 Batteries
B.	ONI-R10-1, Loss of Off-Site Power	Commence Shutdown of DC Lube Oil Pumps
C.	ONI-R10-2, Station Blackout	Cross-tying of Unit1 and Unit 2 Batteries
D.	ONI-R10-2, Station Blackout	Commence Shutdown of DC Lube Oil Pumps

QUESTION RO 10

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	Generic	2.4.22
	Importance Rating	3.6	

K&A: Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.

Generic

Explanation: **Answer C –** ONI-R10-2, Station Blackout is the correct flowchart to enter because HPCS is still available. Additionally, this flowchart will reduce station battery loads until AC Power is restored from either off-site, or from on-site resources. Cross tie of Unit Batteries is a Time Critical Operator Action that must be completed within 35 minutes.

A – Incorrect – ONI-R10-1, Loss of Off-Site Power is not correct with EH-11 and EH-12 deenergized. Plausible if the applicant does not understand the difference in the terms, loss of Off-Site power and Station Blackout. The priority is correct.

B – Incorrect - ONI-R10-1, Loss of AC Power in not correct with EH-11 and EH-12 deenergized. Plausible if the applicant does not understand the difference in the terms, loss of AC power and Station Blackout. The priority is not correct since shutdown of DC lube oil pumps is classified as 4 hour Time Sensitive Action, therefore the 35 minute TCOA should be performed first.

D – Incorrect – ONI-R10-2 is correct, but the second part is incorrect. See B above.

Technical Reference(s): ONI-R10 Rev. 16	Reference Attached: ONI-R10 pp. 8 and 81-83

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): x

Question Source:	Bank # Modified Bank # New	 Perry 2003 # SRO-90
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 11

The following conditions exist:

- Reactor is in MODE 4
- Residual Heat Removal (RHR) A loop is in Shutdown Cooling
- Residual Heat Removal (RHR) B pump is tagged out
- The Reactor Recirculation pumps are secured

Which of the following describes the significance of RPV water level lowering to 240 inches on the Shutdown Range level indication?

A. Coolant stratification in the RPV could occur if the RHR A pump tripped.

B. The RHR A pump would cavitate due to insufficient net positive suction head.

C. Fluid vortexes could develop by the fuel support pieces due to insufficient water level and may result in fuel damage.

D. This level is too low to provide the thermal driving head required for adequate bypass cooling flow to the nuclear instruments.

QUESTION RO 11

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295001	AK1.01
	Importance Rating	3.5	

K&A: Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Natural circulation

Partial or Complete Loss of Forced Core Flow Circulation

Explanation: **Answer A –** The minimum RPV level to promote natural circulation is 250 inches. Raising RPV water level to greater than or equal to 250 inches indicated on the Shutdown Range level indication provides a natural circulation path between inside and outside the core shroud. With RPV level < 250 inches, coolant thermal stratification will occur if the shutdown cooling forced circulation is lost for the given plant conditions.

B – Incorrect – RHR pumps will trip on Level 3 prior to cavitation resulting. Plausible if the applicant believes the RPV level required to provide net positive suction head for an RHR pump providing shutdown cooling flow is the same as the loss of natural circulation limit of 250 inches.

C – Incorrect – Vortices will not form with the given plant conditions. Plausible if the applicant believes there is a possibility of vortices developing near the fuel support pieces if RPV level is below the loss of natural circulation limit of 250 inches.

D – Incorrect – This is not a concern in Mode 4. Plausible if the applicant does not recognize that adequate cooling to the NIs is available in Mode 4 without core bypass.

Technical Reference(s): SDM-E12 Rev. 4	Reference Attached: SDM-E12 p.55

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3035-11(LP)-A.2

Question Source:	Bank # Modified Bank # New	Perry 2010 Audit # RO-01
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 12

The plant is operating at rated power.

All DC buses are being fed from their Normal Chargers.

MCC F1B08 Automatic Transfer Switch is locked on Normal Source due to unavailability of the Alternate Source.

If a loss of Load Center F-1-B occurs, what is the consequence as the affected DC Bus voltage degrades to zero?

A. DB-1-A, BOP Static Inverter will be lost.

B. Rx Recirculation Pumps will shift to Slow speed.

C. RFPT A Emergency Lube Oil Pump cannot be started.

D. Annunciator ANN PWR SUPPLY FAIL (H13-P680-07-E15) illuminates.

QUESTION RO 12

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295003	AK2.06
	Importance Rating	3.4	

K&A: Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF A.C. POWER and the following: D.C. electrical loads

Partial or Complete Loss of AC Power

Explanation: **Answer C –** > With MCC F1B08 transfer switch locked on the Normal Source, the Normal charger for DC Bus D-1-B is lost if Load Center F-1-B is lost. Since the Normal charger is lost to bus D-1-B, voltage will degrade and eventually RFPT A ELOP cannot be started.

A – Incorrect – Plausible if MCC F-1-D transfer switch was locked in Alternate and a loss of Load Center F-1-D occurred, as this load is powered from DC Bus D-1-A.

B – Incorrect – Plausible if MCC F-1-D transfer switch was locked in Alternate and a loss of Load Center F-1-D occurred, as this load is powered from DC Bus D-1-A.

D – Incorrect – Plausible if MCC F-1-D transfer switch was locked in Alternate and a loss of Load Center F-1-D occurred, as this load is powered from DC Bus D-1-A.

Technical Reference(s): PDB-H04 Rev. 5, PDB-H05 Rev. 5, 206-034 Rev. UUU, 206-052 Rev. DDD, & ARI-H13-P680-07 Rev. 34	Reference Attached:): PDB-H04 pp. 2 & 7, PDB-H05 p. 22, 206-034, 206-052, & ARI-H13-P680-07 p. 145

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-R42-34

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 13

The plant is operating at 100% power when annunciator H13-P870-0001-A5, BATTERY 1A DC SYSTEM TROUBLE, alarms.

The indication used to determine if a ground fault is indicated is/are __(1)__ .
And, it is necessary to quickly locate and isolate the ground because __(2)__ .

	(1)	(2)
A.	the ground indicating lamps on DC Bus D-1-A	grounds can be indicators or precursors to serious degradation of equipment
B.	Bus D-1-A DC Voltmeter on Panel P870	a ground on DC Bus D-1-A makes the Division 1 EDG inoperable due to the loss of control power
C.	the ground indicating lamps on DC Bus D-1-A	a ground on DC Bus D-1-A makes the Division 1 EDG inoperable due to the loss of control power
D.	Bus D-1-A DC Voltmeter on Panel P870	grounds can be indicators or precursors to serious degradation of equipment

QUESTION RO 13

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295004	AK3.02
	Importance Rating	2.9	

K&A: Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: Ground isolation/fault determination

Partial or Total Loss of DC Power

Explanation: **Answer A –** ARI-H13-P870-01-A5 says a possible cause of the alarm could be a Bus D-1-A ground. SOI-R42 (Sys A), Non Divisional DC System A Distribution, says that all control for this system are located at the local panels, and ATTACHMENT 1, Ground Check Operator Aid, provides instructions to the operator for performing ground detection on Bus D-1-A by placing the Ground Test Switch in GROUND TEST and noting that if either of the two indicating lamps is dim or out, a ground is indicated. The Battery Bus ground status instrumentation is only available locally on the DC Load Buses. Per SDM-R42 (DC Electrical Systems), the DC Power Systems operate ungrounded, and are provided with ground detection features to quickly identify and locate potentially degraded equipment.

B – Incorrect – Although 125 VDC voltage can be read on panel P870 and the voltage value may change, you cannot determine if there is a ground using this reading. Also, Div. 1 EDG control power is supplied by Div. 1 125 VDC. Plausible if the applicant believes there is also ground detection status available in the Control Room or that D-1-A supplies the Div. 1 EDG control power.

C – Incorrect – Although the ground detection status location is correct, the Div. 1 EDG control power is supplied by Div. 1 125 VDC. Plausible if the applicant believes that D-1-A supplies the Div. 1 EDG control power.

D – Incorrect - The ground detection status location is not correct. Plausible if the applicant believes there is also ground detection status available in the Control Room.

Technical Reference(s): ARI-H13-P870-01-A5, Rev. 18 & SOI-R42 (SYS A), Rev. 6	Reference Attached: ARI-H13-P870-01-A5, p. 13 & SOI-R42 (SYS A), p. 74

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-R42 (#33).

Question Source:	Bank #	Quad Cities 2014 # RO-03
	Modified Bank #	
	New	

Question History: Previous 2 NRC Exams? No

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

10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 14

The plant is operating at 80% rated power.

The A channel of the C85 Pressure Regulator is in TEST.

Then the B channel of the C85 Pressure Regulator starts failing causing the MAX COMB FL LMT IN CONT indicating light to illuminate.

Based on this information:

The Turbine Control Valves will be __(1)__ .

And in order to mitigate this event, the operator is required to __(2)__ .

	(1)	(2)
A.	opening	Scram the Rx and trip the main turbine
B.	closing	Scram the Rx and trip the main turbine
C.	opening	adjust MAX COMBINED FLOW LIMIT setpoint
D.	closing	adjust MAX COMBINED FLOW LIMIT setpoint

QUESTION RO 14

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295005	AA1.05
	Importance Rating	3.6	

K&A: Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP: Reactor/turbine pressure regulating system

Main Turbine Generator Trip

Explanation: **Answer A –** With one pressure regulator in TEST, a failure of the other pressure regulator in the open direction will cause the MAX COMB FL LMT IN CONT indicating light to illuminate and TCVs and TBVs to open resulting in lowering RPV pressure. With 2 pressure regulators failed, the operator is required to insert a scram and trip the main turbine IAW ONI-C85.

B – Incorrect – 1st part - Plausible as the TCVs would be closing if the B pressure regulator failed closed.

C – Incorrect – 2nd part - In this situation the BPVs would also open. The MCFL setpoint in this case could not be adjusted low enough to close the BPVs. Plausible since the MAX COMB FL LMT IN CONT indicating light is on. Adjustment of MCFL not appropriate due to reactivity control concerns and guidance in ONI-C85.

D – Incorrect – 1st part - Plausible as the TCVs would be closing if the B pressure regulator failed closed.
2nd part - In this situation (TCVs closing), the MCFL setpoint would not have any effect on the TCVs or BPVs. Plausible since the MAX COMB FL LMT IN CONT indicating light is on. Adjustment of MCFL not appropriate due to reactivity control concerns and guidance in ONI-C85.

Technical Reference(s): ONI-C85 Rev. 1	Reference Attached: ONI-C85 pp. 4-5

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3035-06(LP)-A.1

Question Source:	Bank #	
	Modified Bank #	
	New	x

Question History: Previous 2 NRC Exams? No

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

10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 15

The reactor was operating at rated power when a scram occurred.

The scram has not been reset.

'CHAN 1 DATA' and 'CHAN 2 DATA' lights are lit.

Control Rod 34-31 indication is alternating between green LED lit and blank numerical display.

Select the statement that is applicable to Control Rod 34-31:

A. Control Rod is full-in.

B. Control Rod is full-out.

C. Control Rod position is unknown.

D. All PIPs have failed for the current Control Rod position.

QUESTION RO 15

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295006	AA2.02
	Importance Rating	4.3*	

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

Scram

Explanation: **Answer A –** This indicates that one channel's reed switches (located within each control rod's stationary Position Indication Probe (PIP)) has failed for the full-in (LED) position indication. Only one channel's green full-in LED is needed to determine the control rod is fully inserted.

B – Incorrect – If the control rod was full-out, the RED LED would be alternating for the same type of reed switch failure. Plausible if the applicant confuses the definition of full-in with full-out LED display colors. (The red full-out or position 48 lights would be lit on the full core display.)

C – Incorrect – The control rod position may be determined by accessing the back panels in the control room. One will have a green (full-in) light lit and the other Division's green light will not be lit. Plausible if the applicant does not understand that there are two PIPs inside each control rod assembly for rod bottom indication and only one channel reed switch needs to be operable to determine the rod position.

D – Incorrect – If both reed switches have failed there is no position indication, i.e., no green full-in LED alternating between lit and unlit, and no green full in light lit in the back panels for either channel of the PIP. Plausible if the applicant does not recall that the channels are independent and that one operable channel will provide the alternating indications provided in the stem.

Technical Reference(s): SDM-C11(RC&IS) Rev. 10	Reference Attached:): SDM-C11(RC&IS) pp. 4-6

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-C11_RC&IS-1.16

Question Source:	Bank #	Clinton 2007 # RO-04
	Modified Bank #	
	New	

Question History: Previous 2 NRC Exams? No

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

10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 16

The control room was abandoned IAW ONI-C61, Evacuation Of The Control Room and control was established at the Remote Shutdown Panel, C61-P001 IAW IOI-11, Shutdown From Outside The Control Room.

If it becomes necessary, which of the following EOP related actions can be manually controlled from C61-P001?

A. HPCS Injection

B. Containment Spray

C. Suppression Pool Cooling

D. Anticipate Emergency Depressurization

QUESTION RO 16

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295016	2.4.8
	Importance Rating	3.8	

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

Control Room Abandonment

Explanation: **Answer C –** Only a limited number of EOP Actions can be accomplished from the Remote Shutdown Panel (RSD). Of the ones listed, Suppression Pool Cooling is the only action that can be manually controlled from the RSD.

A – Incorrect – Plausible since HPCS is the backup level control system to RCIC for RSD operations. However, HPCS is only used in auto mode (L2 to L8) and no manual control is available at the RSD

B – Incorrect – Plausible since control of the Containment Spray OTBD S/O valve E12-F028 is provided at the RSD. However, control of the INBD S/O valve, E12-F537 is not provided at the RSD. Additionally, the Containment Spray auto initiation function is bypassed when control is transferred to the RSD.

D – Incorrect – Plausible since 3 SRVs are controlled from the RSD. However AED is performed by manually controlling the Main Turbine BPVs. And there is no control provided of the BPVs at the RSD. Additionally, MSIVs are isolated when control is established at the RSD.

Technical Reference(s): IOI-11 Rev.38	Reference Attached: IOI-11 pp. 3 & 19

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-C61-E

Question Source:	Bank #	
	Modified Bank #	
	New	x

Question History: Previous 2 NRC Exams? No

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

10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 17

The plant is operating at rated power when annunciator RCIRC A/B TEMP HI, H13-P680-04-D8 alarmed.

(1) What procedure should be entered, first?

(2) What would be an appropriate operator action in response to this alarm while keeping the plant operating at power?

	(1)	(2)
A.	ONI-P43, Loss of Nuclear Closed Cooling	Isolate NCC flow to Containment and Drywell to reduce loads on the NCC system
B.	ONI-P43, Loss of Nuclear Closed Cooling	Shift Recirculation Pumps to SLOW Speed
C.	ARI-H13-P680-04, Recirc Flow Control	Isolate NCC flow to Containment and Drywell to reduce loads on the NCC system
D.	ARI-H13-P680-04, Recirc Flow Control	Shift Recirculation Pumps to SLOW Speed

QUESTION RO 17

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295018	AK1.01
	Importance Rating	3.5	

K&A: Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Effects on component/system operations

Partial or Complete Loss of CCW

Explanation: **Answer D –** Alarm Response Instruction, ARI-H13-P680-04, Recirc Flow Control, Rev. 26 states in D8 (RCIRC A/B TEMP HI) Section 4.3 "CONSIDER shifting the Recirculation Pumps to slow speed in accordance with IOI-3 and SOI-B33."

A – Incorrect – Plausible, the operator should enter the Alarm Response Instruction, ARI-H13-P680-004 since no ONI-P43 entry condition was provided. Also, plausible if the operator does not understand that isolating NCC flow to Containment and the Drywell would result in having to scram the reactor with NCC flow being stopped to the Drywell Coolers and the Reactor Recirculation Pumps, which would prevent continued operations at power. See ONI-P43

B – Incorrect – Plausible, since no ONI-P43 entry condition was provided; the operator should enter the Alarm Response Instruction, ARI-H13-P680-04, for alarm RCIRC A/B TEMP HI. The second part is correct; the operator should consider shifting recirculation pumps to slow speed to reduce loads on the NCC system.

C – Incorrect – Plausible, the operator should enter the Alarm Response Instruction, ARI-H13-P680-004, for alarm RCIRC A/B TEMP HI. For part (2) see A above.

Technical Reference(s): ARI-H13-P680-04 Rev. 26 and ONI-P43 Rev. 14	Reference Attached: ARI-H13-P680-04 p. 108 and ONI-P43 p. 3

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-Combined-P43 1.11

Question Source:	Bank #	
	Modified Bank #	Grand Gulf 2010 # RO-25
	New	
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: The phrasing of Part (1) may be considered SRO level. Adding a second part to the question stem does not change the intent or answer from the original question. Should consider this a Bank question, not significantly modified. Modified stem to fit Perry Plant design and modified answers to fit Perry procedures.

QUESTION RO 18

The plant is operating at rated power.

Then due to an electrical short, P52-F200, IA CNTMT ISOL VLV closes and cannot be reopened.

Due to a small Instrument Air leak on a J-header in containment, Instrument Air pressure is decaying in containment.

If Instrument Air pressure lowers to 0 psig in containment, what will be the positions of the following valves?

	Outlet Scram Valve C11-EP-127	SDV First Vent Valve C11-F010	CRD Flow Control Valve C11-F002A
A.	Open	Open	Open
B.	Closed	Open	Open
C.	Open	Closed	Closed
D.	Closed	Closed	Closed

QUESTION RO 18

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295019	AK2.01
	Importance Rating	3.8	

K&A: Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following: CRD hydraulics

Partial or Complete Loss of Instrument Air

Explanation: **Answer C –** On a loss of IA, the Scram Outlet (and Inlet) valves will go from closed to open; the SDV Vent (and Drain) valves go from open to closed; and the CRD Flow Control Valves will go from open to closed.

A – Incorrect – The SDV vent valves and the CRD FCVs fail closed on a loss of air.

B – Incorrect – The Scram outlet valves fail open and SDV vent valves and the CRD FCVs fail closed on a loss of air.

D – Incorrect – The Scram outlet valves fail open.

Technical Reference(s): ONI-P52 Rev. 18, 302-244 Rev. N, 302-871 Rev. FF, & 302-872 Rev. EE	Reference Attached: ONI-P52 p. 27, 302-244, 302-871, & 302-872

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-C11_CRDH-N

Question Source:	Bank # Modified Bank # New	 Columbia 2011 # RO-08
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 19

During RPV Head detensioning a transient occurred that resulted in the following conditions:

- RHR ‘A’ Pump tripped while in Shutdown Cooling Mode
- RPV level is now 240 inches and stable
- The plant entered ONI-E12-2, Loss of Decay Heat Removal

Why does ONI-E12-2 direct the use of an alternate decay heat removal system?

A. To ensure adequate mixing of the bulk coolant to avoid exceeding Recirculation Loop to Steam Dome differential temperature limits to protect primary system piping from thermal stresses.

B. To ensure that the thermal stress limits on the feedwater piping thermal sleeve are not exceeded.

C. Because decay heat removal must be maintained in order to prevent boiling in the reactor vessel.

D. Because excessive coolant temperature will result in damage to RWCU demineralizer resin.

QUESTION RO 19

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295021	AK3.05
	Importance Rating	3.6	

K&A: Knowledge of the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING: Establishing alternate heat removal flow paths

Loss of Shutdown Cooling

Explanation: **Answer C –** If an alternate decay heat removal method is not placed in service, saturation conditions (boiling) will eventually occur. From IOI-0012, Maintaining Cold Shutdown, if RPV water drops below 245" on Reactor Shutdown Range Level 1B21-R605, natural circulation flow will be inhibited. This could result in saturated conditions occurring locally in the core. And, IOI 0012, states in 4.1.2 – When reactor recirculation pumps are not operating then maintain RPV water level > 250 inches.

A – Incorrect – plausible if the applicant incorrectly assumes that thermal shock limitations need to be taken into consideration when reactor recirculation pumps are secured. Thermal shock limitations, here, are of concern during Recirc pump startup.

B – Incorrect – plausible sense Reactor Water Cleanup is operated in reduced feedwater temperature mode to avoid this thermal stress.

D – Incorrect – plausible because high temperatures can damage RWCU resin; however, other interlocks protect the resin from high temperature.

Technical Reference(s): IOI-0012 Rev. 17 & ONI-E12-2 Rev. 40	Reference Attached: IOI-0012 p. 4 & ONI-E12-2 p. 19

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3035-11(LP)-A.2

Question Source:	Bank #	River Bend 2008 # RO-09
	Modified Bank #	
	New	
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 20

The plant is in a refuel outage with movement of irradiated fuel in the FHB.

Fuel Handling Building Ventilation System is in operation as directed by SOI-F11 Section 7.33, Fuel Handling Building Alignment for Fuel Handling Operations.

Then an irradiated fuel bundle is dropped in the lower IFTS canal.

Shortly thereafter, a HIGH radiation alarm is received on the FHB Ventilation Exhaust GAS and IODINE modules.

EOP-3, Secondary Containment Control, is entered.

Which of the following describes the current status of the FHB Ventilation System based on these plant conditions?

A. Only two Exhaust Fans are running due to the HIGH alarm on the iodine channel.

B. Only two Exhaust Fans are running due to the HIGH alarm on the noble gas channel.

C. Two Exhaust Fans and one Supply Fan are running due to the HIGH alarm on the noble gas channel.

D. Two Exhaust Fans and one Supply Fan are running due to the HIGH alarm on the iodine channel.

QUESTION RO 20

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295023	AA1.01
	Importance Rating	3.3	

K&A: Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS: Secondary containment ventilation

Refueling Accidents

Explanation: **Answer B –** A HIGH on the FHB Gas channel will cause the running FHB supply fan to trip. SOI-F11 Section 7.33 directs FHB Ventilation to be in service IAW SOI-M40. Since the FHB ventilation was running IAW SOI-M40, 2 exhaust fans and one supply fan would be running initially.

A – Incorrect – A HIGH alarm on the iodine channel will only cause a FHB Evacuation Alarm.

C – Incorrect – This is the Normal mode of operation for the FHB ventilation system. The Supply fan would have tripped due to the High alarm on the Gas channel.

D – Incorrect – This is the Normal mode of operation for the FHB ventilation system. . The Supply fan would have tripped due to the High alarm on the Gas channel.

Technical Reference(s): ONI-D17 Rev. 20, SOI-F11 Rev. 22, SOI-M40 Rev. 12, & IOI-009 Rev. 43	Reference Attached: ONI-D17 p. 19, SOI-F11 p. 110, SOI-M40 pp. 5 & 16, & IOI-009 pp. 58 & 62

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-M40 (#7) & OT-COMBINED-D17-L.2

Question Source:	Bank # Modified Bank # New	Perry 2015 # RO37
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 21

The plant was operating at rated power when a DBA LOCA occurs.

Initially, the increasing pressure in the Drywell is mitigated by __(1)__ .

Once the steam in the Drywell has condensed, __(2)__ mitigate a negative pressure in the Drywell.

	(1)	(2)
A.	Backup Drywell Purge	Drywell Vacuum Breakers
B.	Backup Drywell Purge	Horizontal vents
C.	Horizontal vents	Drywell Vacuum Breakers
D.	Horizontal vents	Horizontal vents

QUESTION RO 21

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295024	EK2.14
	Importance Rating	3.9	

K&A: Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following: Containment pressure: Mark-III

High Drywell Pressure

Explanation: **Answer C –** Initially the increased pressure from a DBA LOCA will push down on the SP water level in the DW, uncovering the first row of vents in the Drywell weir wall, then the second row of vents and then third row, if necessary to mitigate the increase in DW pressure. As the steam, released from the Drywell into the Containment continues and as the steam in the Drywell condenses, forming a potential vacuum in the Drywell, eventually pressure in the Containment will exceed that of the DW. To prevent the inward pressure on the DW structure, the Drywell Vacuum Breakers will actuate to relieve the pressure differential.

A – Incorrect – 1st part - plausible if the applicant confuses Backup Drywell Purge with DW pressure leakage mitigation. Also, Drywell Vacuum Breakers is the correct answer for the second part of the question.

B – Incorrect – 1st part - plausible since Backup Drywell Purge will reduce DW pressure during a DBA LOCA. 2nd part - plausible since a negative pressure in the DW affects SP level; however, the ΔP required to open the vacuum breaker is much less then that required to uncover the horizontal vents.

D – Incorrect – 1st part Drywell Vacuum Breakers is a correct answer. 2nd part -plausible since a negative pressure in the drywell effects suppression pool level; however, the ΔP required to open the vacuum breaker is much less then that required to uncover the horizontal vents.

Technical Reference(s): SDM T23 Rev. 15	Reference Attached: SDM T23 p. 2

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-T23-B

Question Source:	Bank # Modified Bank # New	 x
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 22

The unit was operating at rated power when a transient resulted in a reactor scram.

During the transient RPV pressure peaked at 1120 psig.

Currently, main condenser vacuum is 21” HgA and degrading.

Reactor pressure is still lowering from the initial transient and one SRV remains open in AUTO.

Given the choices below, current reactor pressure is ____ psig (disregard setpoint tolerances)

A. 950

B. 940

C. 930

D. 920

QUESTION RO 22

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295025	EA2.01
	Importance Rating	4.3*	

K&A: Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Reactor pressure

High Reactor Pressure

Explanation: **Answer C** – Since RPV pressure exceeded 1083 psig, entry into EOP-01 is required. With condenser pressure at 21" HgA, the MSIVs are closed. Therefore, Bypass Valves are not being used in conjunction with the SRV for pressure control. Since RPV pressure is peaked at 1120 psig, 10 SRVs initially opened, LLS armed lowering the opening and closing set-points for 6 SRVs. Only one SRV has a LLS reclose value of 926 psig. All others are higher.

A – Incorrect – At 950 psig all 6 LLS SRVs would be open.

B – Incorrect – At 940 psig 2 LLS SRVs would be open.

D – Incorrect – At 920 psig all SRVs would be closed.

Technical Reference(s): ONI-B21-1 Rev.11 and ARI-H13-P601-19 Rev. 21	Reference Attached: ONI-B21-1 p.12 and ARI-H13-P601-19 p. 57

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-B21_N11-F

Question Source:	Bank #	Perry 2009 # RO-22
	Modified Bank #	
	New	
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 23

The plant is operating at rated power with the reactor well to steam dryer storage pool gate installed.

Which of the following exceeds a Technical Specification Limiting Condition For Operation.

A. Drywell-to-primary containment differential pressure is 1.9 psid

B. Primary containment average air temperature is 95 °F

C. Corrected suppression pool level is 18 feet, 3.5 inches

D. Suppression pool average temperature is 96 °F

QUESTION RO 23

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295026	2.2.42
	Importance Rating	3.9	

K&A: Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

Suppression Pool High Water Temperature

Explanation: **Answer D** – Tech Specs limit for Average Suppression Pool Temperature is ≤ 95 °F.

A – Incorrect – plausible in that the EOP entry for high Drywell Pressure is >1.68 psig, where 1.9 psid could be confused as exceeding that limit and thus higher than T.S. limits; however, the T.S. LCO for Drywell Pressure is, Drywell-to-Containment differential pressure shall be ≥-0.5 psid ≤2.0 psid.

B – Incorrect – plausible, in that the T.S. LCO for primary containment average air temperature shall be ≤95 °F. Individual temperatures in Containment may exceed 95 degrees, but the average must not exceed 95 °F.

C – Incorrect – plausible because one of the Suppression Pool Water Level T.S. LCOs is Corrected suppression pool water level shall be ≥18 feet 3.2 inches and ≤18 feet 6 inches when the reactor well to steam dryer storage pool gate is installed. The applicant would have to error in not remembering the lower limit which is 18' 3.2"

Technical Reference(s): TS 3.6.1.5 Rev. Amend. 171, 3.6.5.4 Rev. Amend. 171, 3.6.2.2 Rev. Amend. 174, and 3.6.2.1 Rev. Amend. 69	Reference Attached: TS 3.6.1.5 p. 3.6-21, 3.6.5.4 p. 3.6-69 3.6.2.2 p. 3.6-39, and 3.6.2.1 p. 3.6-36

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3037-10-1

Question Source:	Bank # Modified Bank # New	 x
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 24

A transient occurred that resulted in the following:

- Containment temperature is 158 °F and rising.
- EOP-1, RPV Control and EOP-2, Primary Containment Control were entered.
- The US has determined that Emergency Depressurization (ED) is required to be performed due to increasing Containment temperature.

What is EOP Bases for performing ED now?

A. To lower Containment temperature below the Technical Specification LCO limit.

B. To ensure the environmental qualification temperature limit of SRV solenoids is not exceeded.

C. To preserve the Pressure Suppression function of Containment in the event a loss of coolant accident.

D. To delay exceeding the environmental qualification temperature for safety related electrical equipment.

QUESTION RO 24

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295027	EK1.01
	Importance Rating	2.5	

K&A: Knowledge of the operational implications of the following concepts as they apply to HIGH CONTAINMENT TEMPERATURE (MARK III CONTAINMENT ONLY): Equipment environmental qualifications: Mark-III

High Containment Temperature (Mark III Containment Only)

Explanation: **Answer D** – The containment design temperature limit of 185 °F is based on not exceeding the environmental qualifications of safety related electrical equipment. Emergency depressurizing prior to reaching 185°F will maintain equipment operability (operation) for as long as possible.

A – Incorrect – Plausible since the EOP-2 action to spray Containment would in most cases lower temperature below the TS LCO limit. However, this is not the EOP Bases for ED.

B – Incorrect – Plausible as this is the Bases for the DW temperature limit for ADS SRVs.

C – Incorrect – Plausible since rising Containment temperature has an effect on SP temperature. However, the Pressure Suppression function is based on suppression pool temperature and level.

Technical Reference(s): EOP Bases Rev. 8 and EOP-2 Bases Rev. 6	Reference Attached: EOP Bases p. 91 and EOP-2 Bases p. 9

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3402-07-C

Question Source:	Bank # Modified Bank # New	 Perry 2017 # RO-24
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 25

Plant is operating at rated power.

Suppression Pool Level is 16.5 feet and lowering.

IAW EOP-02, Primary Containment Control Bases, if Suppression Pool level continues to lower, EOP-01, RPV Control is entered to _____.

A. ensure injection systems are available for reactor water level control

B. allow ADS to be inhibited to prevent uncontrolled depressurization of the RPV

C. allow reactor pressure to be reduced as necessary to control below the Heat Capacity Limit

D. ensure a reactor scram is inserted and the reactor is shutdown prior to emergency depressurization

QUESTION RO 25

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295030	EK3.06
	Importance Rating	3.6	

K&A: Knowledge of the reasons for the following responses as they apply to LOW SUPPRESSION POOL WATER LEVEL: Reactor SCRAM

Low Suppression Pool Water Level

Explanation: **Answer D** – Per EOP-02 Bases; entering EOP-01 ensures reactor shutdown and allows for proper coordination and transition to ED, use of bypass valves to anticipate ED, and in case of ATWS proper transition to EOP-01-5 to ensure no uncontrolled low pressure injection during ED action.

A – Incorrect - This is not the EOP-02 Bases reason for entry into EOP-1, RPV Control. Plausible in that EOP-1 does provide level control guidance.

B – Incorrect – plausible as this is the purpose of inhibiting ADS during an ATWS; however, ATWS conditions not provided.

C – Incorrect - This is not the EOP-02 Bases reason for entry into EOP-1, RPV Control. Plausible in that EOP-1 does provide reactor pressure control guidance and EOP-2 gives guidance on controlling pressure to maintain below the HCL. However, at 12.25 feet in the S.P. the top row of horizontal vents will start to become uncovered and therefore all heat capacity of the S.P. can be assumed to be lost.

Technical Reference(s): EOP-02 Bases Rev. 6	Reference Attached: EOP-02 Bases p. 39

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3402-05-C

Question Source:	Bank # Modified Bank # New	Perry 2013 Audit # RO-19
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 26

The plant was operating at rated power with HPCS tagged out for motor replacement.

Then a loss of off-site power occurred.

Current status is as follows:

- RCIC tripped on mechanical overspeed
- RPV level is 20 inches and lowering
- RPV pressure is 560 psig and lowering
- Div 1 Diesel Generator tripped for unknown reason
- Div 2 Diesel Generator failed to start
- Div 3 Diesel Generator is supplying bus EH13
- Attempts are being made to lineup Alternate Injections Subsystems
- All appropriate EOPs have been entered

Based on this information when must SRVs be opened to rapidly depressurize the RPV?

A. Before RPV level lowers to 16.5 inches

B. When RPV level cannot be maintained > -25 inches

C. When RPV level cannot be maintained > -45 inches

D. After RPV level lowers below -75 inches

QUESTION RO 26

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295031	EA1.07
	Importance Rating	3.7*	

K&A: Ability to operate and/or monitor the following as they apply to REACTOR LOW WATER LEVEL: Safety/relief valves

Reactor Low Water Level

Explanation: **Answer D** – IAW EOP-01-1, Alternate Level/Pressure Control, with no injection systems currently injecting, the crew waits until RPV level lowers to < -75 inches before Rapid Depressurization is performed.

A – Incorrect – Plausible since EOP-01, RPV Control directs the operator to ED if RPV level cannot be restored and maintained > 16.5 inches. However, this is based on having injection systems available.

B – Incorrect – Plausible since this is one of the ED criteria in an ATWS. However, the question did not state any control rods were not inserted.

C – Incorrect – Plausible since -45 inches is one of the definitions for Adequate Core Cooling – Spray Cooling. However, no spray cooling (LPCS HPCS) exists.

Technical Reference(s): EOP-01-1 Bases Rev. 0	Reference Attached: EOP-01-1 Bases pp. 37, 48-49, & 58

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3403-01(LP)

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 27

An ATWS is in progress with the following conditions:

- Reactor pressure at stable 900 psig
- Both SLC Pumps have been started, with SLC Tank level initially at 4780 gallons
- All actions taken to insert Control Rods are ineffective
- 25 minutes following the start of both SLC Pumps, the ‘A’ SLC Pump trips and cannot be restarted

Based upon the above conditions, if RPV pressure remains stable at 900 psig, what is the minimum time (rounded to the nearest minute) required to inject enough boron to maintain the reactor shut down under all conditions, IAW EOP-01-5, ATWS RPV Control?

A. 29 minutes

B. 53 minutes

C. 82 minutes

D. 106 minutes

QUESTION RO 27

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295037	EA2.03
	Importance Rating	4.3*	

K&A: Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: SBLC tank level

Scram Condition Present and Reactor Power Above APRM Downscale or Unknown

Explanation: **Answer C** – With the given conditions and SLC flow per pump of 43 gpm and EOP-01-5 statement to Monitor Boron Injection and to Verify SLC pumps trip when level reaches 200 gallons in the SBLC Tank, a calculation of the approximate minimum time to reach CSDBW gives the following minimum time.

86 gpm (2 pumps) X 25 minutes = 2150 gallons (2 pumps)

((4780 gal – 200 gal) – 2150 gal) / 43 gpm = (2430 gal) / (43 gpm) = 57 minutes (1 pump)

Total time = 25 minutes (2 pumps) + 57 minutes (1 pump) = 82 minutes

A – Incorrect – Plausible if applicant confuses HSBW value (<2450 gallons in tank per EOP-01-5) with shutdown under all conditions (injecting with two pumps for 25 minutes and one pump for 4 minutes).

B – Incorrect – Plausible if applicant correctly uses value for shutdown under all conditions but does not account for one pump tripping and uses two pumps injecting at normal flow rate of 43 gpm per pump.

D – Incorrect – Plausible if applicant correctly uses value for shutdown under all conditions but only uses one pump injecting with one pump at normal flow rate of 43 gpm.

Technical Reference(s): EOP-01-5 Bases Rev. 0 and SDM C41 Rev. 10	Reference Attached: EOP-01-5 Bases p.104 and SDM C41 p. 8

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED_C41-D

Question Source:	Bank #	
	Modified Bank #	Grand Gulf 2010 # RO-20
	New	
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 28

IAW EOP-03 Secondary Containment Control Bases, before exceeding Fuel Pool Cooling and Cleanup (FPCC) heat exchanger outlet temperature of __(1)__, the Control Room will perform/direct __(2)__ .

	(1)	(2)
A.	135 °F	removal FPCC demineralizers to prevent resin breakdown
B.	135 °F	opening Fuel Handling Building man doors to allow natural circulation
C.	150 °F	removal FPCC demineralizers to prevent resin breakdown
D.	150 °F	opening Fuel Handling Building man doors to allow natural circulation

QUESTION RO 28

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295038	2.4.18
	Importance Rating	3.3	

K&A: Knowledge of the specific bases for EOPs.

High Offsite Radioactivity Release Rate

Explanation: **Answer A –** IAW EOP-3 Bases, the FPCC demineralizers are removed from service prior to reaching 135 °F to prevent resin breakdown.

B – Incorrect – Opening fuel handling building doors is done at 200 °F in the pools.

C – Incorrect – This is the temperature for shutting the plant down.

D – Incorrect – This is the temperature for shutting the plant down. And opening fuel handling building doors is done at 200 °F in the pools.

Technical Reference(s): EOP-3 Bases Rev. 7	Reference Attached: EOP-3 pp. 53-54, 56, 58-59, & 61-62

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3402-17-D

Question Source:	Bank #	Perry 2019-1 # RO 67
	Modified Bank #	
	New	
Question History:	Previous 2 NRC Exams?	YES
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 29

Overall command responsibility for a fire in the MB100 Maintenance Building resides with __(1)__ .
Overall command responsibility for a fire in the Administration Building resides with __(2)__ .

	(1)	(2)
A.	Fire Brigade Leader	Fire Brigade Leader
B.	Fire Brigade Leader	Responding Off-Site Fire Department
C.	Responding Off-Site Fire Department	Fire Brigade Leader
D.	Responding Off-Site Fire Department	Responding Off-Site Fire Department

QUESTION RO 29

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	600000	AK1.02
	Importance Rating	2.9	

K&A: Knowledge of the operation applications of the following concepts as they apply to Plant Fire On Site: Fire Fighting

Plant Fire On Site

Explanation: **Answer B** – IAW ONI-P54, the Fire Brigade Leader is responsible for firefighting in the protected area. For fires in the owner controlled area the Responding Off-Site Fire Department is responsible for fires that do not affect plant safety or operability.

A – Incorrect – IAW ONI-P54, the Fire Brigade Leader is responsible for firefighting in the protected area. For fires in the owner controlled area the Responding Off-Site Fire Department is responsible for fires that do not affect plant safety or operability. It's plausible that an applicant might pick this one because the Admin Building is within a few feet of the protected area standoff fencing.

C and D - Incorrect – Fire Brigade Leader is responsible for firefighting inside the protected area; however, plausible in that the MB-100 building is a large industrial building that could become a large involved fire well beyond the capabilities of the fire brigade.

Technical Reference(s): ONI-P54 Rev. 27	Reference Attached: ONI-P54 pp. 27 and 44

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3035-05(LP)-A.9

Question Source:	Bank # Modified Bank # New	 Perry 2013 # RO-29
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 30

The Transmission System Operator informs the Control Room of a Degraded Grid condition and the following occurs:

1. ONI-S11, Degraded Grid, is entered
2. Grid voltage and generator voltage swings are observed

What is the nominal EH bus voltage at which the under-voltage relays will actuate <u>and</u> what is the basis for this protection?

A. 3000 Vac (~75% nominal), protection is provided to prevent damage to safety related power transformers.

B. 3800 Vac (~95% nominal), protection is provided to prevent damage to safety related power transformers.

C. 3800 Vac (~95% nominal), protection is provided to prevent jeopardizing the reliability of starting safety related motors.

D. 3000 Vac (~75% nominal), protection is provided to prevent jeopardizing the reliability of starting safety related motors

QUESTION RO 30

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	700000	AK2.01
	Importance Rating	3.1	

K&A: Knowledge of the interrelations between GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES and the following: Motors

Generator Voltage and Electric Grid Disturbances

Explanation: **Answer C** – The Degraded Voltage level of protection (95% voltage) is the highest level which the UV relays will actuate and is based on protecting large safety related motors when starting.

A – Incorrect – 1st part – Plausible as this is the Loss of Voltage protection level. However, the UV relays will actuate at 3800 Vac with a small time delay. 2nd part – plausible since the Loss of voltage protection is provided to protect supplied equipment. However, the safety related power transformers supply the Safety buses not vice versa.

B – Incorrect – 2nd part – plausible since the Loss of voltage protection is provided to protect supplied equipment. However, the safety related power transformers supply the Safety buses not vice versa.

D – Incorrect – 1st part – Plausible as this is the Loss of Voltage protection level. However, the UV relays will actuate at 3800 Vac with a small time delay.

Technical Reference(s): OT-COMBINED-R10 (LP) Rev. 4, ONI-S11 Rev. 14, ARI-H13-P877-01 Rev. 15, and NOP-OP-1003 Rev. 10	Reference Attached: OT-COMBINED-R10 (LP) Slide 134, ONI-S11 p. 3, ARI-H13-P877-01 p. 19, and NOP-OP-1003 p. 15

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-R10-1.49

Question Source:	Bank #	
	Modified Bank #	Perry 2010 # RO-30 – See Comments below
	New	
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: When administered in 2010, incorrect training material was used as basis for question. The material has since been corrected.

QUESTION RO 31

Which of the following describes the bases for maximizing Containment Cooling during the execution of EOP-02, Primary Containment Control?

To preclude exceeding the Containment _____.

A. average air temperature LCO limit

B. design temperature limit of 330 °F

C. environmental qualification temperature of 185 °F

D. environmental qualification temperature of 330 °F

QUESTION RO 31

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295011	AK3.01
	Importance Rating	3.6	

K&A: Knowledge of the reasons for the following responses as they apply to HIGH CONTAINMENT TEMPERATURE (MARK III CONTAINMENT ONLY): Increased containment cooling: Mark-III

High Containment Temperature (Mark III Containment only)

Explanation: **Answer C** – Containment equipment qualification temperature is 185 °F.

A. – Incorrect – Entry into EOP-02 is not required until the containment average air temperature is > 95 °F. LCO limit is ≤ to 95 °F

B. – Incorrect – The containment design temperature limit is, ≤ 185 °F. (330 °F is the Drywell design temperature limit).

D. – Incorrect – The environmental qualification temperature for safety-related electrical equipment in Containment is 185 °F. (330 °F is the environmental qualification temperature for safety-related electrical equipment in the Drywell).

Technical Reference(s): EOP-02 Bases Rev. 6, SDM-T23 Rev. 15, TS 3.6.1.5 Rev Amend. 171 & TS 3.6.5.5 Rev. Amend. 171	Reference Attached: EOP-02 Bases pp. 9 and 24, SDM-T23 p. 36, TS 3.6.1.5 p. 3.6-21, TS 3.6.5.5 p. 3.6-70

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3402-07-A

Question Source:	Bank #	
	Modified Bank #	
	New	x

Question History: Previous 2 NRC Exams? No

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

10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 32

The plant is operating at rated power.

The Drywell Cooling coils are lined up as follows:

- UP DW CLG COOLER SEL VLV, 1P43-F020A is open and 1P43-F020B is closed
- MID DW CLG COOLER SEL VLV, 1P43-F015A is open and 1P43-F015B is closed
- LW DW CLG COOLER SEL VLV, 1P43-F025A is closed and 1P43-F025B is open

Additionally:

- LW DW Clrs Temp Cont VLV, 1P43-F365 is closed
- One fan on each DW cooler is operating

Then the following alarms are received:

- DRYWELL AVERAGE TEMP A HI – H13-P601-20-F3
- DRYWELL AVERAGE TEMP B HI – H13-P601-17-F5

The Unit Supervisor directs the BOP Operator to operate all available Drywell cooling.

Subsequently, a loss of Instrument Air to the containment and drywell occurs.

How will the Drywell Ventilation system will respond to the loss of Instrument Air?

A. 1P43-F020A & 1P43-F015A, & 1P43-F025B indicate OPEN

B. 1P43-F020A, 1P43-F015A, & 1P43-F025A indicate OPEN

C. 1P43-F020B, 1P43-F015B, & 1P43-F025B indicate OPEN

D. P43-F365 fails open

QUESTION RO 32

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295012	AA1.01
	Importance Rating	3.5	

K&A: Ability to operate and/or monitor the following as they apply to HIGH DRYWELL TEMPERATURE: Drywell ventilation system

High Drywell Temperature

Explanation: **Answer C** – On a loss of IA, the DW cooler selector valves fail such that the B valves are open and the A valves are closed. This indication is observed on H13-P800.

A – Incorrect – This is equivalent to a fail-as-is indication. However, all DW cooling coil selector valves fail so that the B valves open.

B – Incorrect – This is equivalent to a fail to the A valves open. However, all DW cooling coil selector valves fail so that the B valves open.

D – Incorrect – The Lower DW TCV is an MOV and is not affected by a loss of IA. Plausible since many TCVs are air operated valves.

Technical Reference(s): ONI-P52 Rev. 18, SDM-M13 Rev. 6, ARI-H13-P601-020 Rev. 24, SOI-M13 Rev. 7, 208-174 Sh. 13 Rev. M, 302-613 Rev. CC	Reference Attached: ONI-P52 p. 33, SDM-M13 pp. 11-12, ARI-H13-P601-020 p. 81, SOI-M13 pp. 4-5, 208-174 Sh. 13, and 302-613

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-M13-(#21)

Question Source:	Bank #	
	Modified Bank #	Perry 2009 # RO-25
	New	
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 33

The plant was operating at rated power when a Loss of Feedwater Heating occurred.

The Reactor Engineer reports that Minimum Critical Power Ratio (MCPR) is 1.06.

This value of MCPR is _____.

A. within the Safety Limit; no operator actions are required

B. in violation of the Safety Limit; it is required to insert Cram Rods immediately

C. in violation of the Safety Limit; it is required to insert all insertable Control Rods immediately

D. in violation of the Safety Limit; it is required to insert all insertable Control Rods within two hours

QUESTION RO 33

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295014	AA2.05
	Importance Rating	4.2*	

K&A: Ability to determine and/or interpret the following as they apply to INADVERTENT REACTIVITY ADDITION: †Violation of safety limits

Inadvertent Reactivity Addition

Explanation: **Answer D** – For Cycle 18, with power at 100%, MCPR Safety Limit is > or = to 1.10 for two Recirc Loops operating. With MCPR below 1.10, it is required to insert all insertable Control Rods within 2 hours IAW TS 2.2, SL Violations.

A – Incorrect – This would be correct if MCPR were > or = 1.10

B – Incorrect – This would be correct if MEOD was exceeded. Not correct action for SL violation

C – Incorrect – The control rods need to be inserted within 2 hours, not immediately per TS 2.2

Technical Reference(s): TS 2.2 Rev. Amend. 188	Reference Attached: TS 2.2 Rev. p.2.0-1

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3037-03-1 & -6

Question Source:	Bank #	2013 Perry # RO-36
	Modified Bank #	
	New	
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 34

A plant startup is in progress with Rx power at the point of adding heat.

A manual Rx scram is inserted in response to a feedwater problem.

Annunciator FULL SCRAM, ARI-H13-P680-05-A10 alarms

The SCRAM VALVES pushbutton on P680 is backlit red.

Upon depressing the SCRAM VALVES pushbutton all control rods have green LEDs illuminated on the full core display except for control rod 30-19.

Based on the above information, the RO would conclude _____.

A. Control rod 30-19 is the only rod that scrammed.

B. Control rod 30-19 is the only rod that has not scrammed.

C. All scram valves are open since the SCRAM VALVES pushbutton is backlit red.

D. All scram valves are closed since the SCRAM VALVES pushbutton is backlit red.

QUESTION RO 34

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295015	2.4.46
	Importance Rating	4.2	

K&A: Ability to verify that the alarms are consistent with the plant conditions.

Incomplete Scram

Explanation: **Answer B –** IAW SOI-C11(RC&IS) Section 7.9.5, if SCRAM VALVES is backlit RED, it indicates that not all scram valves are in the same position (i.e. not all open or not all closed). All control rods with both scram valves open will be indicated by the green LED lit on the full core display. The lack of a green LED with the SCRAM VALVES pushbutton depressed indicates one or both of the scram valves are closed. Since the plant is at the POAH, a failure to open of either the inlet or outlet scram valve would not allow the control rod to insert. If Rx pressure was > 600 psig, Rx pressure would scram the control rod upon a failure of the inlet scram valve to open.

A – Incorrect – This is the opposite and is based on the misconception that a lack of lights has both scram valves open.

C – Incorrect – Red backlight means that not all scram valves are in the same position, it does not mean they are open.

D – Incorrect – Red backlight means that not all scram valves are in the same position, it does not mean they are closed.

Technical Reference(s): ARI-H13-P680-05-A10 Rev. 16, SDM-C11(CRDM Rev. 5, SDM-C71 Rev. 12, and SOI-C11(RC&IS) Rev. 32	Reference Attached: ARI-H13-P680-05-A10 p.23, SDM-C11(CRDM pp. 36-37 & 88, SDM-C71 p. 9, and SOI-C11(RC&IS) p. 36

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-C11_RC&IS-1.16

Question Source:	Bank #	
	Modified Bank #	Perry 2017 # RO-34
	New	
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 35

The reactor was at rated power when an un-isolable primary system rupture occurred, causing rising Offsite release rates.

Based on this information, entry into EOP-04, Radioactivity Release Control, is required when Offsite release rate exceeds the __(1)__ Emergency Action Level value.

And, entry into EOP-01-2, Emergency RPV Depressurization, is required before Offsite release rate exceeds the __(2)__ Emergency Action Level value to ensure protection of the general public.

	(1)	(2)
A.	Unusual Event	Site Area Emergency
B.	Alert	Site Area Emergency
C.	Alert	General Emergency
D.	Site Area Emergency	General Emergency

QUESTION RO 35

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295017	AK1.02
	Importance Rating	3.8*	

K&A: Knowledge of the operational implications of the following concepts as they apply to HIGH OFF-SITE RELEASE RATE: †Protection of the general public

Abnormal Offsite Release Rate

Explanation: **Answer C** – Entry into EOP-04 is required at the Alert level. Entry into EOP-01-02 is required before the General Emergency level.

A – Incorrect – Entry into EOP-04 is required at the Alert level. Entry into EOP-01-02 is required before the General Emergency level. Plausible, memory error

B – Incorrect – Entry into EOP-01 is required before the General Emergency level. Plausible as part (1) is correct and part (2) is a higher classification level.

D – Incorrect – Entry into EOP-04 is required at the Alert level. Plausible as part (2) is correct, although part (1) is not.

Technical Reference(s): EOP-03/04 Bases Rev 0	Reference Attached: EOP-03/04 Bases p. 53

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): x

Question Source:	Bank #	2010 Nine Mile Point Unit 1 # RO-59
	Modified Bank #	
	New	
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 36

The plant is operating at rated power with CRD pump B tagged out.

Annunciator CRD HCU LEVEL HI/PRESS LO, H13-P680-05-E8 alarms.

The NLO is dispatched to investigate and reports control rod 30-31 accumulator pressure is 1100 psig.

Then, CRD pump A trips on overcurrent.

Several minutes later multiple accumulator faults are received.

A manual Rx scram is inserted.

Based on this information:

A. Rx pressure alone will insert control rod 30-31.

B. Accumulator pressure alone will insert control rod 30-31.

C. Rx pressure will start to insert control rod 30-31 and accumulator pressure will assist to complete rod insertion.

D. Accumulator pressure will start to insert control rod 30-31 and reactor pressure will assist to complete rod insertion.

QUESTION RO 36

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295022	AK2.07
	Importance Rating	3.4	

K&A: Knowledge of the interrelations between LOSS OF CRD PUMPS and the following: Reactor pressure (SCRAM assist): Plant-Specific

Loss of Control Rod Drive Pumps

Explanation: **Answer D** – CRD accumulators are charged with N2 to reach an accumulator pressure of ~1785 psig. Also, during a scram when the accumulator pressure drops by ~600 psig. During a scram, without CRD pumps running accumulator pressure can drop below Rx pressure. If accumulator pressure drops below Rx pressure, the ball check valve repositions to allow Rx pressure to assist with the rod insertion. Since rod 30-31 accumulator pressure was 1100 psig prior to the scram, the Rx pressure will assist in rod insertion.

A – Incorrect – Plausible if accumulator pressure was less than Rx pressure (~1025 psig for the given condition).

B – Incorrect – Plausible if the accumulator was fully charged to ~1785 psig.

C – Incorrect – This is the opposite of what occurred for this control rod with given conditions.

Technical Reference(s): ARI-H13-P680-05-E8 Rev. 16, SOI-C11(HCU) Rev. 25, Lesson Plan C11(CRDM) Rev. 1, & SDM-C11(CRDH) Rev. 9	Reference Attached: ARI-H13-P680-05-E8 p. 103, SOI-C11(HCU) p. 36, Lesson Plan C11(CRDM) p. 10, & SDM-C11(CRDH) pp.8-9

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-C11_CRDM-C.1 & F

Question Source:	Bank #	Susquehanna 2007 # RO-62
	Modified Bank #	
	New	

Question History: Previous 2 NRC Exams? No

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

10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 37

The plant has experienced a LOCA and the following plant conditions exist:

- Reactor Level –50”
- Containment Hydrogen Concentration 6.5%
- Drywell Hydrogen Concentration 10%

Based on the above information, the primary hydrogen production mechanism is __(1)__ .

And if operating, the __(2)__ must be secured?

	(1)	(2)
A.	zirc-water reaction	Hydrogen Igniters
B.	zirc-water reaction	Hydrogen Recombiners
C.	steel oxidation reaction	Hydrogen Igniters
D.	steel oxidation reaction	Hydrogen Recombiners

QUESTION RO 37

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	500000	EK3.03
	Importance Rating	3.0	

K&A: Knowledge of the reasons for the following responses as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS: Operation of hydrogen and oxygen recombiners

High Containment Hydrogen Concentration

Explanation: **Answer B** – The major source of hydrogen generation during a LOCA is from the zirconium cladding reacting with water. SOI-M51, Combustible Gas Control System requires securing the Hydrogen Recombiners when hydrogen concentration reaches 6%.

A – Incorrect – 1st part – Plausible since other sources of hydrogen production contribute to the total hydrogen generated. 2nd part – Plausible since the H2 Igniters are not restarted of they lose power if H2 concentration is >6%.

C – Incorrect – 1st part – Plausible since other sources of hydrogen production contribute to the total hydrogen generated.

D – Incorrect – 2nd part – Plausible since the H2 Igniters are not restarted of they lose power if H2 concentration is >6%.

Technical Reference(s): SOI-M51/56 Rev. 29 and SDM-M51 Rev. 9	Reference Attached: SOI-M51/56 p. 4 and SDM-M51 pp. 1-2

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): x

Question Source:	Bank #	Perry 2007-2 # RO-57
	Modified Bank #	
	New	

Question History: Previous 2 NRC Exams? No

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

10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 38

The plant was operating at rated power with RCIC tagged out for oil replacement.

A transient occurred resulting in the following sequence of events:

- All ECCS pumps auto started
- HPCS pump tripped and cannot be restarted
- LPCS pump tripped and cannot be restarted
- Emergency Depressurization was performed
- RHR A, B, & C are injecting
- RPV level is -10” and stable
- Annunciator RHR B SUCTION PRESSURE LOW alarms

RHR B pump discharge flow and discharge pressure are observed to be lower than normal and fluctuating.

Which of the following actions is required?

A. Notify the US then stop RHR B pump.

B. Notify the RO ATC then override RHR B pump off.

C. Immediately override RHR B pump off then update the crew.

D. Maintain RHR B pump running as it is needed for adequate core cooling.

QUESTION RO 38

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	203000	K5.02
	Importance Rating	3.5	

K&A: Knowledge of the operational implications of the following concepts as they apply to RHR/LPCI: INJECTION MODE (PLANT SPECIFIC): †Core cooling methods

RHR/LPCI: Injection Mode

Explanation: **Answer D** – Per EOP-01 Bases, ACC is assured if RPV level can be maintained >-25" with injection. Since RPV level is stable at -10" with 3 RHR/LPCI injection systems injecting, ACC is assured. If one of the RHR/LPCI systems were to be removed, RPV level would lower and ACC would no longer be assured. Therefore, the RHR B pump should remain running. The Low Suction Pressure annunciator is an Entry Condition to ONI-G42, SP Suction Strainer Clogging, which directs the operator to shutdown the pump if not needed by the EOPs

A – Incorrect – Plausible since the US is notified of any ECCS equipment malfunction. However, the US concurrence must be obtained prior to stopping the pump.

B – Incorrect – Plausible since PAP-0205 requires notifying the ATC RO prior to overriding a safety system if the override is required by EOPs or other approved instructions.

C – Incorrect – Plausible since notification of the crew (crew update) is expected prior to stopping a safety system.

Technical Reference(s): ARI-H13-P601-17 Rev. 20, ONI-G42 Rev. 6, EOP-01 Bases Rev. 9, & PAP-0205 Rev. 23	Reference Attached: ARI-H13-P601-17 p. 71, ONI-G42 pp. 3 & 5, EOP-01 Bases pp. 48-49, & PAP-0205 p. 12

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3039-02-G

Question Source:	Bank #	
	Modified Bank #	Perry 2017 # RO-38
	New	
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 39

During cooldown of the plant for a refueling outage, the following conditions exist:

- All rods are inserted.
- RHR ‘A’ is in Shutdown Cooling
- The Low Flow Controller in MANUAL is maintaining RPV water level stable at 255 inches
- The Bypass Valve Opening Jack is maintaining RPV pressure stable at 15 psig

Then the mechanical coupling on ESW Pump ‘A’ fails.

With no operator action, RPV pressure will __(1)__ and RPV level will __(2)__ .

	(1)	(2)
A.	rise	rise
B.	rise	remain the same
C.	remain the same	remain the same
D.	remain the same	rise

QUESTION RO 39

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	205000	K6.08
	Importance Rating	3.5	

K&A: Knowledge of the effect that a loss or malfunction of the following will have on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE): RHR service water: Plant-Specific

Shutdown Cooling

Explanation: **Answer A** – A failure of ESW A pump results in a loss of cooling to the RHR heat exchangers and a loss of decay heat removal. Without operator action, RPV water heats up causing reactor pressure to rise and level will rise due to swell.

B – Incorrect – First part is correct, plausible if applicant is confused on thermodynamic principles.

C – Incorrect – plausible if the applicant does not understand the operation of the Bypass Valve Opening Jack. Plausible if applicant is confused on thermodynamic principles.

D – Incorrect - plausible if the applicant does not understand the operation of the Bypass Valve Opening Jack.

Technical Reference(s): IOI-0004 Rev 26 & GFE Components Chapter 7 Rev. 4	Reference Attached:): IOI-0004 pp. 26 & 30 & GFE Components Chapter 7 p. 61

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3303-7

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 40

Vortex formation at the Low Pressure Core Spray Pump Suction Strainer is prevented by maintaining Suppression Pool __(1)__ within an established limit. Per the EOP Bases, the damaging mechanism associated with vortex formation is __(2)__.

	(1)	(2)
A.	Level	LPCS Pump cavitation
B.	Level	air entrapment in the LPCS Pump
C.	Temperature	LPCS Pump cavitation
D.	Temperature	air entrapment in the LPCS Pump

QUESTION RO 40

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	209001	K1.02
	Importance Rating	3.4	

K&A: Knowledge of the physical connections and/or cause effect relationships between LOW PRESSURE CORE SPRAY SYSTEM and the following: Torus/suppression pool

Low-Pressure Core Spray

Explanation: **Answer B** – Suppression Pool Level is maintained above a minimum value to prevent vortex formation. Air entrapment in the LPCS Pump suction strainer may result.

A – Incorrect – 2nd part – Plausible since cavitation is the formation and collapse of vapor bubbles in the eye of the pump impeller.

C – Incorrect – 1st part – Plausible since SP temperature affects NPSH. The SP level limits are based on preventing both vortexing and loss of NPSH.

D – Incorrect – 1st part – Plausible since SP temperature affects NPSH. The SP level limits are based on preventing both vortexing and loss of NPSH. 2nd part – Plausible since cavitation is the formation and collapse of vapor bubbles in the eye of the pump impeller.

Technical Reference(s): EOP Bases Rev. 8	Reference Attached: EOP Bases pp. 78-79

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3402-05-C

Question Source:	Bank # Modified Bank # New	 x
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 41

Following a reactor scram, RPV level lowered to 10”.

Four minutes later a Loss Of Off-Site Power occurred.

Based on this information:
Emergency Diesel Generators __(1)__ on the Loss Of Off-Site Power.
And, when the EH buses are reenergized the Low Pressure Core Spray pump restarts __(2)__ .

	(1)	(2)
A.	trip and restart	five (5) seconds after the EH11 bus is re-energized
B.	continue to run	immediately upon re-energization of the EH11 bus
C.	trip and restart	five (5) seconds after the EH12 bus is re-energized
D.	continue to run	immediately upon re-energization of the EH12 bus

QUESTION RO 41

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	209001	A1.07
	Importance Rating	3.0	

K&A: Ability to predict and/or monitor changes in parameters associated with operating the LOW PRESSURE CORE SPRAY SYSTEM controls including: Emergency generator loading

Low-Pressure Core Spray

Explanation: **Answer B** – The EDGs continue to run even with a loss of off-site power and their respective output breakers will close when they sense a dead bus, i.e., breakers EH-1114 & EH-1115 open for Div. 1, EH11 Bus and breakers EH-1212 & EH-1213 open for Div. 2, EH12 Bus. The LPCS Pump breaker will NOT trip on under voltage, therefore the LPCS pump motor will immediately restart when voltage is restored to the EH11 bus.

A – Incorrect – plausible, if the applicant mistakenly believes that the Division 1 and Division 2 EDGs will trip, respectfully, on under voltage on EH11 and EH12, or will trip to load shed. Also, incorrect in that the LPCS Pump breaker will NOT trip on under voltage.

C – Incorrect – plausible, see A above for (1). Plausible, memory error, EH12 is the Division 2 safety bus and LPCS is powered from the Division 1 safety bus.

D – Incorrect – the EDGs will continue to run even with a complete loss of off-site power and their respective output breakers will close when they sense a dead bus, i.e., breakers EH-1114 & EH-1115 open for Div. 1, EH11 bus and breakers EH-1212 & EH-1213 open for Div. 2, EH12 bus. Plausible, memory error, EH12 is the Div. 2 safety bus and LPCS is powered from the Div. 1 safety bus

Technical Reference(s): SOI-R43 Rev. 49, SDM-R43 Rev. 14; SDM-E21 Rev. 1	Reference Attached: SOI-R43 p. 24, SDM-R43 pp. 19, 21, & 39; SDM-E21 p. 24

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-E21-F and OT-R43/48-F.10 & F.14

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 42

The plant was operating at rated power.

An inadvertent initiation of High Pressure Core Spray occurred due to failure of DW pressure trip units.

The HPCS pump was overridden OFF.

Subsequently, a loss of offsite power occurred coincident with a LOCA.

When power is restored to the Divisional buses by the diesel generators the HPCS Pump will ____.

A. <u>not</u> automatically restart

B. automatically restart immediately

C. automatically restart in 10 seconds

D. automatically restart in 27 seconds

QUESTION RO 42

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	209002	K2.01
	Importance Rating	3.2	

K&A: Knowledge of electrical power supplies to the following: Pump electrical power: BWR-5,6

HighPressure Core Spray

Explanation: **Answer B –** The override logic is powered from DC. However, when AC power is lost to Bus EH13, under-voltage relay R22-27X3A energizes and opens a contact that removes the "seal-in" for the override to OFF. This also deenergizes the K114 relay for the 10 second time delay upon reenergizing Bus EH13. This also deenergizes the K114A relay which closes a contact in the HPCS Pump start circuitry. When power is restored to EH13, the 1R22-27X3A closes energizing K114A & K114B. K114A is a TDO and K114B is an INST close. Therefore, the pump starts immediately.

A – Incorrect – Plausible as the HPCS pump would remain overridden OFF if a low RPV level signal were to be received with no LOOP.

C – Incorrect – Plausible as the HPCS pump will start after a 10-second time delay on a normal automatic start signal.

D – Incorrect – Plausible as this is the time that HPCS pump will deliver full flow to the RPV. This time includes the Div. 3 DG start time

Technical Reference(s): SDM-E22A Rev. 8, 208-066 Sh. 1 Rev. GG & Sh. 2 Rev. M, and 208-206 Sh. 49 Rev. HH	Reference Attached: SDM-E22A pp. 22-23, 208-066 Sh. 1 & Sh. 2

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-E22A (#33)

Question Source:	Bank #	
	Modified Bank #	Perry 2002 # RO-12
	New	
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 43

Plant is operating at RTP when the following sequence occurs:

- HPCS water leg pump trips
- HPCS WATER LEG PUMP DISCH PRESS LO alarm is received
- NLO reports that a HPCS Water Leg pump control power fuse was found blown
- NLO reports that the failed HPCS water leg pump control power fuse has been replaced

The low pressure condition will impact the HPCS system by __(1)__ and the operator should __(2)__ .

	(1)	(2)
A.	preventing automatic initiation of HPCS and injecting air into the RPV.	place the HPCS system in operation, IAW SOI-E22A, HIGH PRESSURE CORE SPRAY SYSTEM, using min. flow to maintain the discharge path full.
B.	preventing automatic initiation of HPCS and injecting air into the RPV.	start the HPCS Water Leg Pump and then fill and vent or vent the high point of the system as applicable.
C.	causing possible damage to the discharge piping if HPCS initiates.	place the HPCS system in operation, IAW SOI-E22A, HIGH PRESSURE CORE SPRAY SYSTEM, using min. flow to maintain the discharge path full.
D.	causing possible damage to the discharge piping if HPCS initiates.	start the HPCS Water Leg Pump and then fill and vent or vent the high point of the system as applicable.

QUESTION RO 43

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	209002	A2.02
	Importance Rating	3.6	

K&A: Ability to (a) predict the impacts of the following on the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Pump trips BWR-5,6:

High Pressure Core Spray

Explanation: **Answer D** – (1) to prevent causing possible damage to the discharge piping due to water hammer effect if HPCS initiates and (2) since the waterleg pump failed for one blown control power fuse which was replaced, the operator should try to start the waterleg pump, once. If it starts and runs properly then fill and vent or vent the high point of the system, which will then ensure the discharge piping is full of water, therefore protecting the discharge piping from water hammer.

A - Incorrect - (1) plausible if the applicant believes there is an interlock to prevent the automatic initiation of HPCS, which there is not. (2) plausible if the applicant believes that placing the system in service will remove any air in the line, which it will not. Only properly venting the system with the waterleg pump running will prevent the possibility of water hammer.

B – Incorrect – (1) wrong for same reason as A (1) above. (2) is a correct answer.

C - Incorrect – (1) is a correct answer (2) wrong for same reason as A (2) above.

Technical Reference(s): ARI-H13-P601-0016 Rev. 20, SOI-E22A Rev. 40, SVI-E22-T1183 Rev. 15, OAI-0201 Rev. 46, SDM-E22A Rev. 8	Reference Attached: ARI-H13-P601-0016 pp. 99-100, SOI-E22A p. 39, SVI-E22-T1183 p. 2, OAI-0201 p. 54, SDM-E22A p. 4

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-E22A-18

Question Source:	Bank #	
	Modified Bank #	River Bend 2007 # SRO 86
	New	
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 44

The plant was operating at rated power with Standby Liquid Control Pump A tagged out.

A transient occurred and the BOP operator initiated SLC B.

Which of the following was a <u>direct</u> result of initiating SLC B?

A. <u>only</u> 1G33-F001, RWCU SUCT FM CNTMT INBD ISOL valve closes

B. <u>only</u> 1G33-F004, RWCU SUCT FM CNTMT OTBD ISOL valve closes

C. both 1G33-F001 and 1G33-F004 valves close

D. all RWCU containment isolation valves close

QUESTION RO 44

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	211000	A3.06
	Importance Rating	4.0*	

K&A: Ability to monitor automatic operations of the STANDBY LIQUID CONTROL SYSTEM including: RWCU system isolation: Plant-Specific

Standby Liquid Control

Explanation: **Answer A** – Isolation of G33-F001 is initiated by taking the SLC B Keylock switch to ON.

B – Incorrect – Isolation of the G33-G004 valve is initiated from the SLC A Keylock switch.

C – Incorrect – Plausible since both the G33-F001 and G33-F004 valves close when both SLC systems are initiated.

D – Incorrect – Plausible since a high Δ Flow condition will isolate all RWCU containment isolation valves.

Technical Reference(s): LP OT-COMBINED-C41 Rev. 3	Reference Attached: LP OT-COMBINED-C41 p. 16

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-G33_G36-F.1 and OT-COMBINED-C41-N

Question Source:	Bank #	
	Modified Bank #	Riverbend 2010 # RO-33
	New	
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 45

An ATWS is in progress.

The Unit Supervisor has directed you to start Standby Liquid Control (SLC).

When SLC Pump 'B' switch is taken to ON, the Squib Valve fires, and which of the following sequence of events occur:

A. Reactor Water Cleanup isolates
1C41-F001B SLC PMP SUCT VALVE B starts to open
SLC B PUMP starts when 1C41-F001B indicates intermediate position

B. Reactor Water Cleanup isolates
1C41-F001B SLC PMP SUCT VALVE B starts to open
SLC B PUMP starts when 1C41-F001B is full open

C. 1C41-F001B SLC PMP SUCT VALVE B starts to open
SLC B PUMP starts when 1C41-F001B indicates intermediate position
Reactor Water Cleanup isolates after SLC B PUMP starts

D. 1C41-F001B SLC PMP SUCT VALVE B starts to open
SLC B PUMP starts when 1C41-F001B is full open
Reactor Water Cleanup isolates after SLC B PUMP starts

QUESTION RO 45

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	211000	(K4.08
	Importance Rating	4.2*	

K&A: Knowledge of STANDBY LIQUID CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: System initiation upon operation of SBLC control switch

Standby Liquid Control

Explanation: **Answer B** – The squib valve fires, G33-F001 will start to close and C41-F001B will start to open when the SLC pump control switch is taken to ON. The pump start is initiated when the control switch is in ON and the full open limit switch of the C41-F001B is closed.

A – Incorrect – SLC B pump will only start when either 1C41-F001B or 1C41-F031 (Test Tank Outlet Valve is fully open. Plausible if the applicant believes the pump will start once the suction valves starts to open.

C – Incorrect – the SLC pump will not start until either 1C41-F001B or 1C41-F031 is fully open, and the G33-F001 starts to close when the SLC B pump control switch is taken out of the OFF position. Plausible if the applicant believes that the pump will start once the suction valves starts to open and 1G33-F001 starts to close after the pump starts.

D – Incorrect – G33-F001 starts to close as soon as the SLC B pump control switch is taken out of the OFF position. Plausible if the applicant believes that 1G33-F001 starts to close after the pump starts.

Technical Reference(s): SDM-C41(SLC) Rev 10 & SOI-C41 Rev. 22	Reference Attached: SDM-C41(SLC) pp. 15 &16; SOI-C41 pp. 7 & 8

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-Combined-C41-G & OT-Combined-G33/G36-F.1

Question Source:	Bank # Modified Bank # New	Perry 2009 # RO-45
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 46

The plant is at rated power when GP3B RPS SCRAM SOL VLVS white light above one of the Manual Scram Pushbuttons extinguishes.

Which additional condition will result in control rod motion?

A. GP1A light extinguishes

B. GP2B light extinguishes

C. Power lost to RPS Bus A

D. Power lost to RPS Bus B

QUESTION RO 46

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	212000	K3.06
	Importance Rating	4.0	

K&A: Knowledge of the effect that a loss or malfunction of the REACTOR PROTECTION SYSTEM will have on following: Scram air header solenoid operated valves

Reactor Protection

Explanation: **Answer C** – The RPS scram solenoids are arranged in 4 groups with half of the scram solenoids powered from RPS A and half powered from RPS B. Both solenoids in a group must deenergize to reposition the Scram Pilot solenoid Valves which causes the associated rods to scram in. The GP3B RPS SCRAM SOL is powered from RPS B. Therefore, a loss of RPS A will deenergized the Scram Pilot Air Valves for approximately ¼ of the control rods.

A – Incorrect – Plausible as GP1A group is powered from RPS A. However, both solenoids in the same group must deenergize to cause rod notion.

B – Incorrect – Plausible if incorrect RPS supply for the given Channel is selected vs. Group

D – Incorrect – RPS B supplies power to GP3B group. This would result in a ½ scram signal and no rod motion.

Technical Reference(s): SDM-C71 Rev. 12	Reference Attached: SDM-C71 pp. 4 & 7-9

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-C71-1.4

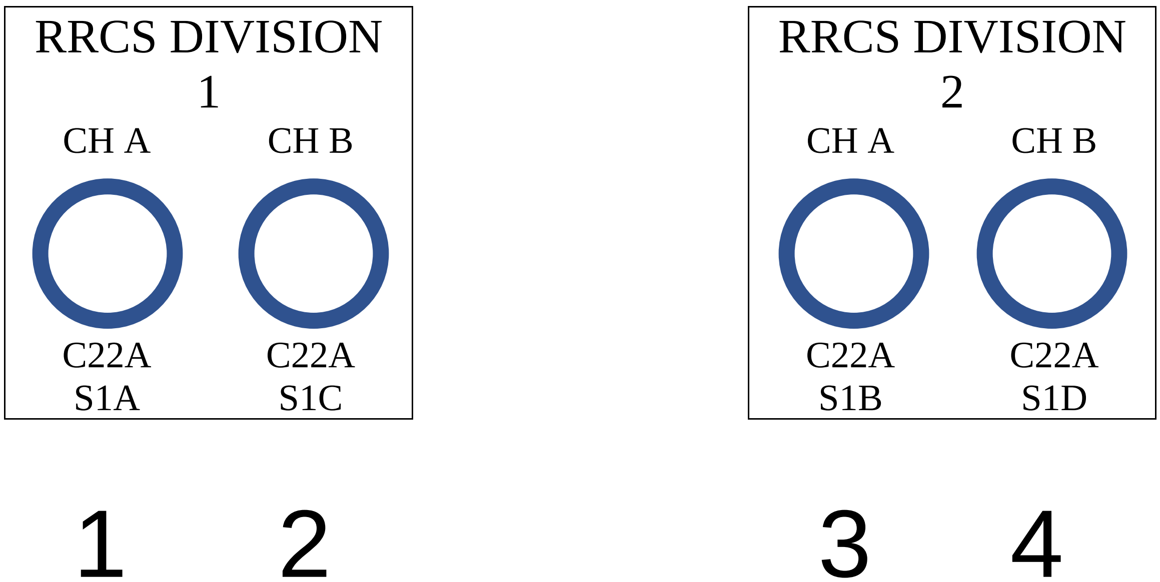
Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 47

You have been directed to initiate ARI.

Which of the following actions will initiate ARI?



A. Simultaneously depressing pushbuttons 1 and 2 or pushbuttons 3 and 4

B. Simultaneously depressing pushbuttons 1 and 3 or pushbuttons 2 and 4

C. Arming switches 1 and 2 and then depressing pushbuttons 1 and 2

D. Arming switches 1 and 3 and then depressing pushbuttons 1 and 3

QUESTION RO 47

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	212000	A4.16
	Importance Rating	4.4*	

K&A: Ability to manually operate and/or monitor in the control room: Manually activate anticipated transient without SCRAM circuitry/RRCS: Plant-Specific

Reactor Protection

Explanation: **Answer C** – ARI actuation is 2 out of 2 logic in any one division.

A – Incorrect – plausible if the applicant does not recall that they must turn the collars of the switches first. The switches must be turned clockwise from Disarm to Armed and then with both 1 and 2 depressed simultaneously to initiate ARI.

B – Incorrect – same as above, but distractor is plausible if the operator believes the logic is one button in each division, verses two buttons in one division.

D – Incorrect – push buttons are for two different divisions, but distractor is plausible if the operator believes the logic is one button in each division, verses two buttons in one division.

Technical Reference(s): SDM C22 Rev. 6	Reference Attached: SDM C22 pp. 8 & 9

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-Combined-C22-D

Question Source:	Bank #	Dresden 2017 # RO-39
	Modified Bank #	
	New	
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 48

The plant is operating at 50% rated thermal power when RPS Bus B EPA’s tripped open.

When the RPS bus is reenergized, which of the following is required?

A. Restart ERIS computer

B. Place IRM H on Range 3

C. Reset Div. 2 RCIC isolation logic

D. Reopen DW VAC RLF MOV ISOL VALVE, 1M16-F010B

QUESTION RO 48

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	215003	2.1.31
	Importance Rating	4.6	

K&A: Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.

Intermediate Range Monitor

Explanation: **Answer B** – RPS B supplies power to 4 IRMs including IRM H. When power is lost to the IRM power supply, the IRM fails down scale to Range 1. It will need to be re-ranged to Range 3 (in Mode 1) per SOI-C71

A – Incorrect – Plausible since the ERIS computer is supplied from an ATWS power panel.

C – Incorrect – Plausible as loss of an RPS bus will cause various isolations. However, the RCIC isolation logic is not affected by a loss of RPS.

D – Incorrect – Plausible since M16-F010B receives an isolation signal on loss of RPS B. However, it is normally closed and will automatically open on vacuum condition in the drywell.

Technical Reference(s): SOI-C71 Rev. 24, IOI-1 Rev. 54, and SDM-C51(IRM) Rev. 8	Reference Attached: SOI-C71 pp.56 & 114, IOI-1 p. 59, and SDM-C51(IRM) pp. 12-13

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-C51_(IRM)-1.7 & 1.14

Question Source:	Bank # Modified Bank # New	Perry 2010 # RO-46
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 49

Plant conditions are as follows:

- Reactor Mode Switch is in STARTUP/STANDBY
- All IRMs are on Range 3
- Source Range Monitor (SRM) A is reading 0.5 cps
- SRMs B and C are reading 8.3 x 10E4
- SRM D drawer Mode Switch is in STANDBY
- A rod block signal has been generated

Which of the following caused the rod block?

A. SRM Upscale

B. SRM Inoperable

C. SRM Downscale

D. SRM Detector Wrong Position

QUESTION RO 49

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	215004	K1.03
	Importance Rating	3.0	

K&A: Knowledge of the physical connections and/or cause effect relationships between SOURCE RANGE MONITOR (SRM) SYSTEM and the following: Rod control and information system: Plant-Specific

Source Range Monitor

Explanation: **Answer B** – A Rod Block signal will be generated due to an SRM INOP logic signal with the Reactor Mode Switch not in RUN and an SRM detector channel Mode Switch not in OPERATE.

A – Incorrect – An Upscale SRM signal does not come in until 10E5. Plausible if the applicant believes the SRM Upscale occurs above 10,000 cps. (SRM B & C > 10E4)

C – Incorrect – The SRM Downscale signal is bypassed when associated IRMs are on Range 3. Plausible if the applicant believes SRM A is generating an SRM Downscale signal and does not remember the IRM range limit.

D – Incorrect – Detector Wrong Position will not generate a Rod Block, with IRMs on or above Range 3. Plausible if the applicant does not remember the IRM range limit.

Technical Reference(s): SDM-C51(SRM) Rev. 8 & SDM-C11(RC&IS) Rev. 10	Reference Attached: SDM-C51(SRM) p. 10 & SDM-C11(RC&IS) pp. 55-56

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-Combined-C11_RC&IS-1.6 & OT-Combined-C51_SRM-1.6

Question Source:	Bank #	Perry 2015 # RO-47
	Modified Bank #	
	New	
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 50

The plant was operating at rated power with APRM A INOPERABLE and in BYPASS.

Subsequently, an electrical fault resulted in a loss of power to ATWS Dist Panel EV-1-B Div 2.

Based on the above information, ____ signal is generated.

A. a full scram

B. <u>only</u> a half scram

C. a Division 2 RRCS ARI

D. <u>only</u> a Recirc Flow Control Valve closure

QUESTION RO 50

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	215005	K2 02
	Importance Rating	2.6	

K&A: Knowledge of electrical power supplies to the following: APRM channels

Average Power Range Monitor/Local Power Range Monitor

Explanation: **Answer B** – When power is lost to panel EV-1-B, APRM's B, D, F, & H lose power which causes a ½ scram signal.

A – Incorrect – Plausible if APRM A were not bypassed. However, since APRM A is bypassed, no full scram signal will result.

C – Incorrect – Div 2 RRCS is not powered from EV-1-B. Plausible since the noun name of panel is ATWS.

D – Incorrect – Plausible since a FCV closure signal would be generated in addition to a ½ scram if power was lost to EV-1-A since EV-1-A supplies both APRM A & E.

Technical Reference(s): ARI-H13-P680-06 Rev 9, ELI-R14/15 Rev 14 , Dwg 206-65 Rev S	Reference Attached: ARI-H13-P680-06 p 39-40, ELI-R14/15 p 7, Dwg 206-065

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-C51AP_OPRM-1.8

Question Source:	Bank # Modified Bank # New	Perry 2015 # RO-48
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 51

There is sufficient steam pressure available to operate RCIC.

Which of the following situations would the loss of RCIC have the most adverse impact on the ability to adequately cool the reactor core?

A. Total Loss of AC Power

B. Steam line break in the drywell

C. Complete loss of feedwater coincident with MSIV closure when operating at rated power

D. Complete loss of feedwater coincident with MSIV closure following a Loss of Offsite Power

QUESTION RO 51

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	217000	K3.04
	Importance Rating	3.6	

K&A: Knowledge of the effect that a loss or malfunction of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) will have on following: Adequate core cooling

Reactor Core Isolation Cooling

Explanation: **Answer A** – In a TLAC (assuming sufficient steam pressure still available) only RCIC would be available for RPV makeup and adequate core cooling (i.e., independent of AC power needs).

B – Incorrect – plausible if the applicant does not have a good understanding of the purpose of RCIC and its operating characteristics which make it unique. A steam line break in the drywell is just a variety of an RCS leak. By itself, this situation suggests nothing in particular about the loss of RCIC with respect to adequate core cooling, especially with respect to the several other ECCS systems, as well as feedwater, that might well be available as backup.

C and D – Incorrect – plausible if the applicant does not have a good understanding of the purpose of RCIC and its operating characteristics which make it unique. RCIC's design purpose is to provide RPV inventory makeup and adequate core cooling in the case of a core isolation coincident with loss of feedwater with or without loss of offsite power. However, a loss of RCIC, by itself, in either of these situations is likely to be moot considering the availability of HPCS.

Technical Reference(s): TS 3.5.3 Bases Rev. 12 and ONI-R10 Rev. 16	Reference Attached: TS 3.5.3 Bases p. 3.5-21, and ONI-R10 p. 3

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-E51-4

Question Source:	Bank # Modified Bank # New	Grand Gulf 2010 # RO-29
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 52

The plant is operating in accordance with the EOPs with the following conditions:

- MSIVs are closed
- RCIC is injecting into the RPV
- SRVs are being cycled to control RPV pressure in band
- EOP-SPI 6.6, RCIC Injection And Pressure Control has not been performed
- RPV level 120" and rising
- Suppression Pool level 18.7' and rising
- Suppression Pool temperature 92 °F and rising

Based on these conditions, RCIC suction is on the __(1)__ .

The automatic response of the __(2)__ suction valve can be overridden using the control switch on P601.

	(1)	(2)
A.	Suppression Pool	Suppression Pool
B.	Suppression Pool	Condensate Storage Tank
C.	Condensate Storage Tank	Suppression Pool
D.	Condensate Storage Tank	Condensate Storage Tank

QUESTION RO 52

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	217000	K4.07
	Importance Rating	3.6	

K&A: Knowledge of REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) design feature(s) and/or interlocks which provide for the following: Alternate supplies of water

Reactor Core Isolation Cooling

Explanation: **Answer A** – With RPV level at 120 inches, a RCIC initiation signal is present. Additionally, with EOP-SPI 6.6 not performed yet and a SP level of 18.7', the SP suction valve will be open. Valve control logic allows only the SP suction valve to be overridden closed under these conditions.

B – Incorrect – 2nd part – Plausible since EOP-SPI 6.6 directs actions to shift RCIC suction to CST even with a high SP level. However, EOP-SPI 6.6 has not been performed.

C – Incorrect – 1st part – Plausible since RCIC is normally lined up to the CST and on a RCIC initiation, will align to the CST if SP suction valve is closed. However, with SP level at 18.7', the SP suction valve will have automatically opened causing the CST suction valve to close.

D – Incorrect – 1st part – Plausible since RCIC is normally lined up to the CST and on a RCIC initiation, will align to the CST if SP suction valve is closed. However, with SP level at 18.7', the SP suction valve will have automatically opened causing the CST suction valve to close. 2nd part – Plausible since EOP-SPI 6.6 directs actions to shift RCIC suction to CST even with a high SP level. However, EOP-SPI 6.6 has not been performed.

Technical Reference(s): SDM-E51 Rev. 15 and ARI-H13-P601-21 Rev. 16	Reference Attached: SDM-E51 pp. 27-28 and ARI-H13-P601-21 p. 89

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-E51 (#9)

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 53

The Plant was operating at rated power when a Loss of Offsite Power occurred.

The following condition now exist:

- The Reactor is shutdown
- All Emergency Diesel Generators started and tied to their respective Buses
- Reactor pressure is cycling with automatic relief valve actuation
- RCIC has isolated
- HPCS has tripped on overcurrent
- All Low Pressure ECCS pumps are in Standby
- Reactor water level is 185.0 inches and lowering at 10 inches/min
- Drywell pressure is 1.50 psig and rising at 0.25 psig/min

With no operator intervention, the Automatic Depressurization System (ADS) will automatically initiate in ____.

A. 2 minutes and 29 seconds

B. 7 minutes and 16 seconds

C. 16 minutes and 51 seconds

D. 18 minutes and 36 seconds

QUESTION RO 53

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	218000	K5.01
	Importance Rating	3.8	

K&A: Knowledge of the operational implications of the following concepts as they apply to AUTOMATIC DEPRESSURIZATION SYSTEM: ADS logic operation

Automatic Depressurization

Explanation: **Answer D –** With RPV level at 185” and lowering at 10”/min, Level 1 (16.5”) will be reached in 16 minutes and 51 seconds. When DW pressure exceeds 1.68 psig in less than one minute the low pressure ECCS pumps will automatically start. Therefore, when RPV level hits L1 the 105 second ADS timer will start and ADS will actuate in 18 minutes and 36 seconds.

A – Incorrect – Plausible since this is the time to hit Level 3 (177.7”) plus 105 seconds (ADS timer).

B – Incorrect – Plausible since this is the time to hit Level 2 (129.8”) plus 105 seconds (ADS timer).

C – Incorrect – Plausible since this is the time to hit Level 1 (16.5”) without the 105 second ADS timer.

Technical Reference(s): ARI-H13-P601-19 Rev. 21	Reference Attached: ARI-H13-P601-0019 p. 71

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-B21C-F and L.2

Question Source:	Bank #	Perry 2019-02 # RO-58
	Modified Bank #	
	New	
Question History:	Previous 2 NRC Exams?	Yes
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 54

The plant is operating in the EOPs.

EOP-SPI 2.5, Bypass of LOCA Interlocks has been directed.

Upon completion, EOP-SPI 2.5 will ____.

A. allow 1E12-F042C, LPCI C INJECTION VALVE to be throttled

B. prevent 1E22-F004, HPCS INJECTION VALVE closure on high RPV level

C. allow 1E12-F024A, RHR A TEST VALVE TO SUPP POOL to be throttled

D. prevent 1E12-F053B, SHUTDOWN COOLING TO FDW SHUTOFF valve closure on low RPV level

QUESTION RO 54

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	223002	K4.08
	Importance Rating	3.3	

K&A: Knowledge of PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF design feature(s) and/or interlocks which provide for the following: †Manual defeating of selected isolations during specified emergency conditions

Primary Containment Isolation/Nuclear Steam Supply Shutoff

Explanation: **Answer D** – EOP-SPI 2.5 defeats the Shutdown Cooling Isolation for the 1E12-F053B valve which allows RHR injection outside the shroud.

A – Incorrect – Plausible as RHR C Injection Valve can be throttled during the EOPs. However, EOP-SPI 6.3 performs this conversion.

B – Incorrect – Plausible as the HPCS Injection Valve L8 Shutoff is defeated in the EOPs. However, EOP-SPI 6.4 performs this function.

C – Incorrect – Plausible as this valve has the capability to be converted to a throttle valve by the use of a Keylock switch on the MCC bucket

Technical Reference(s): EOP-SPI 2.5 Rev. 1	Reference Attached: EOP-SPI 2.5 pp. 2-3

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3403-02b(SG)H

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 55

The plant was at rated power when Bus ED-1-A lost power.

Then, an MSIV isolation occurred.

Based on these conditions, operation of the individual SRV control switches on H13-P601 __(1)__ work and operation of the individual SRV control switches on H13-P631 __(2)__ work.

	(1)	(2)
A.	will	will
B.	will	will not
C.	will not	will
D.	will not	will not

QUESTION RO 55

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	239002	K6.04
	Importance Rating	3.0	

K&A: Knowledge of the effect that a loss or malfunction of the following will have on the RELIEF/SAFETY VALVES: D.C. power: Plant-Specific

Safety Relief Valves

Explanation: **Answer C** – Individual control switches for the SRVs are located on H13-P601 and H13-P631. The SRVs have 2 solenoids, 1 powered from Div. 1 (A solenoid) and 1 powered from Div. 2. (B solenoid). Bus ED-1-A supplies power to the 'A' solenoid for the SRVs. The control switches on P601 control the A solenoids. Therefore, operation from P631 will work and operation from P601 will not work.

A – Incorrect – see B. Operation from P601 will not work

B – Incorrect – plausible if the applicant confuses which Division Switches are on the two panels. Operation from P601 will not work.

D – Incorrect – see B, Operation from P631 will work.

Technical Reference(s): SDM-B21/ N11 Rev. 13 and PDB-H01 Rev. 2	Reference Attached: SDM-B21/ N11 pp. 8 & 19 and PDB-H01 p. 14

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-B21 N11-C

Question Source:	Bank # Modified Bank # New	 Perry 2019-1 # RO-54
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 56

The plant was operating at rated power with the Motor Feed Pump tagged out.

Following a Rx scram, the following conditions exist:

- A RFPT is on the Low Flow Controller in AUTO
- Directed level band is 192 to 200 inches
- RPV level is 165" and lowering slowly
- RPV pressure is 940 psig and lowering slowly with BPV #1 ~50% open

If the operator places the Low Flow Controller in MANUAL, the operator will be able control the Low Flow __(1)__ directly.

In order to restore RPV level the operator is allowed to __(2)__ .

	(1)	(2)
A.	Controller Flow Setpoint (SPT)	increase flow to the RPV up to 1500 gpm
B.	Controller Flow Setpoint (SPT)	raise RFPT discharge pressure up to 1300 psig
C.	Control Valve 1N27- F175 position	increase flow to the RPV up to 1500 gpm
D.	Control Valve 1N27- F175 position	raise RFPT discharge pressure up to 1300 psig

QUESTION RO 56

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	259002	A1.05
	Importance Rating	2.9	

K&A: Ability to predict and/or monitor changes in parameters associated with operating the REACTOR WATER LEVEL CONTROL SYSTEM controls including: FWRV/startup level control position: Plant-Specific

Reactor Water Level Control

Explanation: **Answer D** – With the LFC in AUTO, the output (OUT) of the LFC Level Input (R614_1P) is the input to the LFC SPT (controller flow setpoint) and the difference between actual (MEAS) flow and SPT creates a position demand signal for the LFC valve. However, when the LFC is placed in MAN, the output of the controller directly controls the position of the 1N27-F175 valve and can be adjusted the INC/DEC soft pushbuttons. Also, the operator is allowed to raise ΔP across the N27-F175 valve to 400 psig max. This is done by raising RFPT discharge pressure.

A – Incorrect – 1st part – Plausible since SPT is compared to RPV level to create a flow demand signal with the LFC in AUTO. 2nd part – The operator is allowed to increase flow through the 1N27-F175 valve to only 1400 gpm (F175 max capacity).

B – Incorrect – 1st part – Plausible since SPT is compared to RPV level to create a flow demand signal with the LFC in AUTO.

C – Incorrect – 2nd part – The operator is allowed to increase flow through the 1N27-F175 valve to only 1400 gpm F175 max capacity).

Technical Reference(s): SOI-C34 Rev. 36 and Lesson Plan C34 Rev. 7 and SDM-C34 Rev. 4	Reference Attached: SOI-C34 pp. 34 & 64 and Lesson Plan C34 slide 64 and SDM-C34 p. 14

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-N27 (#25) & OT-COMBINED-C34

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 57

The plant is operating at rated power with the following Initial plant conditions:

- AEGTS Train ‘A’ is in operation with Train ‘B’ in standby
- AEGTS Train ‘B’ electric heating coils are tagged out

Then an electrical failure results in the loss of power to Bus EH11.

With the loss of power to Bus EH11 AEGTS train ‘A’ __(1)__ .

The reactor operator should __(2)__ in response to this event.

	(1)	(2)
A.	has lost power and Train ‘B’ is operating	perform SOI-M15, Section 7.1, “Shifting Operating Trains” as directed by ARI for “ANNULUS EXH SYS TRAIN A OUT OF SERVICE” alarm
B.	has lost power and Train ‘B’ is operating	shutdown Train ‘A’ per SOI-M15, Section 6.2 “Shutdown One Train to Standby Readiness from Both Trains Operating”
C.	is operating, and Train ‘B’ has lost power	perform SOI-M15, Section 7.1, “Shifting Operating Trains” as directed by ARI for “ANNULUS EXH SYS TRAIN B OUT OF SERVICE” alarm
D.	is operating, and Train ‘B’ has lost power	shutdown Train ‘B’ per SOI-M15, Section 6.2 “Shutdown One Train to Standby Readiness from Both Trains Operating”

QUESTION RO 57

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	261000	A2.07
	Importance Rating	2.7*	

K&A: Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A.C. electrical failure

Standby Gas Treatment

Explanation: **Answer A** – Train ‘A’ Fan is off as power was lost to all Div. 1 AC supplied loads when bus EH11 lost power; however, Train ’B’ started on low flow in Train ‘A’ as designed. ARI-H13-P800-01-B2, ANNULUS EXH SYS TRAIN A OUT OF SERVICE directs the operator to shift operating trains from A to B IAW SOI-M15. This alarm would be received due to the loss of power to the A fan. This Section also maintains configuration control by placing the switches and independently verifying then in the proper configuration. “Shutdown One Train to Standby Readiness from Both Trains Operating” would not be appropriate since with no power to the “A” train, the desired outcome to be in Standby Readiness could not be achieved.

B – Incorrect – Although part (1) is correct, part (2) is incorrect for which proper procedure section is used to mitigate this specific event. This is plausible if the candidate considered the initial line up of “A” Train (operating) to still exist following the loss of power to EH11.

C – Incorrect – The loss of Bus EH11 results in a loss of power to Train ‘A’. Part (1) is plausible if the applicant confuses which train has actually loss power. Part (2) is correct for the proper procedure section used to mitigate this event but refers to the wrong train.

D – Incorrect –Not only is part (1) incorrect, part (2) is also incorrect but plausible as stated above.

Technical Reference(s): ARI-H13-P800-01-A1/B2 Rev. 7, SOI-M15 Rev. 13, ELI-R22 Rev. 9, and ELI-R24 Rev. 36	Reference Attached: ARI-H13-P800-01-A1/B2 pp. 5 & 15, SOI-M15 p. 13, ELI-R22 p. 7, and ELI-R24 pp. 26 & 27

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-M15-C.1 and F.1

Question Source:	Bank # Modified Bank # New	 x
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 58

Which of the following describes the RHR Pump start sequence on a Low Reactor Water level initiation signal concurrent with a Loss of Offsite Power?

After the respective Emergency Diesel Generator Output Breaker closes, ____.

A. each RHR pump starts immediately

B. each RHR pump starts after a 5 second time delay

C. RHR Pumps A and B start immediately and RHR Pump C starts after a 5 second time delay

D. RHR Pumps A and B start after a 5 second time delay and RHR Pump C starts immediately

QUESTION RO 58

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	262001	A3.04
	Importance Rating	3.4	

K&A: Ability to monitor automatic operations of the A.C. ELECTRICAL DISTRIBUTION including: Load sequencing

AC Electrical Distribution

Explanation: **Answer D –** On a LOCA concurrent with a LOOP, loads are sequenced onto the diesel generators to prevent overloading. When the DG breaker closes to reenergize the EH Bus, RHR C pump starts immediately, followed by RHR A & B pumps 5 seconds later.

A – Incorrect – Plausible since RHR C starts immediately. However, RHR A & B each has a 5 second time delay.

B – Incorrect – This is true for RHR A & B. However, RHR C pump does not have a time delay.

C – Incorrect – Plausible since this is opposite of the correct sequencing.

Technical Reference(s): SDM-E12 Rev 4	Reference Attached: SDM-OT-E12 p 37

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-E12-C & -F

Question Source:	Bank # Modified Bank # New	Perry 2015 # RO-27
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41 x 55.43	

Comments: Level of Difficulty = x

QUESTION RO 59

A fire in the Control Room forced all personnel to abandon the Control Room.

A reactor scram could not be initiated prior to evacuating the Control Room.

Which of the following describes the preferred method for initiating a reactor scram, including the bases for this method?

Cycle the specified __(1)__ because this will not result in __(2)__ .

	(1)	(2)
A.	ATWS UPS distribution panel breakers	an MSIV closure
B.	ATWS UPS distribution panel breakers	a loss of LPRMs/APRMs
C.	RPS power distribution panel breakers	an MSIV closure
D.	RPS power distribution panel breakers	a loss of LPRMs/APRMs

QUESTION RO 59

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	262002	2.1.20
	Importance Rating	4.6	

K&A: Ability to interpret and execute procedure steps.

Uninterruptable Power Supply (AC/DC)

Explanation: **Answer A** – From the NOTE on ONI-C61 page 5: “The next two steps outline options to scram the reactor from outside the control room (Scram From ATWS UPS and Scram From RPS Power Supply). Scram insertion via the ATWS UPS distribution panels is preferred since it does not result in a closure of the MSIVs.

B – Incorrect, although this is the preferred method of performing a Scram, this will cause the APRMs to lose power. Plausible if the applicant believes the APRMs have an independent power supply.

C – Incorrect; not only is this not the preferred breakers to open, it will also result in closing the MSIVs.

D – Incorrect; this is the 2nd method used to perform a Scram. Plausible is the applicant is confused on the procedurally preferred method.

Technical Reference(s): ONI-C61 Rev. 9 and SDM R14/15 Rev. 3	Reference Attached: ONI-C61 p. 5 and SDM R14/15 p. 25

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3035-13(LP)-A.2

Question Source:	Bank #	Perry 2010 # RO-56
	Modified Bank #	
	New	
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 60

The Unit 1 Division 1 battery voltage is reading 130 VDC locally and in the Control Room.

The Control Room Operator directs a PO to adjust the FLOAT potentiometer for the on-service battery charger to obtain 135 VDC in accordance with SOI-R42 (Div 1), Section 7.5.

Following the adjustment, with ED1A bus loading remaining within the capacity of the Normal Charger, the DIV 1 BATT AMPS meter on H13-P877 will indicate a value in the __(1)__ region with the ability to restore battery voltage to 135 VDC __(2)__ on the number of DC loads.

	(1)	(2)
A.	CHARGE	dependent
B.	CHARGE	independent
C.	DISCHARGE	dependent
D.	DISCHARGE	independent

QUESTION RO 60

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	263000	K1.02
	Importance Rating	3.2	

K&A: Knowledge of the physical connections and/or cause effect relationships between D.C. ELECTRICAL DISTRIBUTION and the following: Battery charger and battery

DC Electrical Distribution

Explanation: **Answer B –** When the Float voltage is increased, the battery ammeter will indicate in the Charge Region. The charger has the capacity to maintain the battery voltage constant with stable load.

A – Incorrect – 2nd part - The charging rate is determined by the difference in voltage between the battery terminal voltage and the charger output voltage. It is not affected by the loading on the bus provided that the bus loading remains within the capacity of the charger.

C – Incorrect – 1st part - Raising the setting on the FLOAT potentiometer will increase the charger output voltage. Since the battery was initially producing 130 VDC, this action will cause a charge indication, not a discharge indication. 2nd part - The charging rate is determined by the difference in voltage between the battery terminal voltage and the charger output voltage. It is not affected by the loading on the bus provided that the bus loading remains within the capacity of the charger.

D – Incorrect – 1st part - Raising the setting on the FLOAT potentiometer will increase the charger output voltage. Since the battery was initially producing 130 VDC, this action will cause a charge indication, not a discharge indication.

Technical Reference(s): SOI-R42 Rev 22, SDM-R42 Rev 10.	Reference Attached: SOI-R42 pp. 35 & 61, SDM-R42 pp. 10-11

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-R42 (#18 & #34)

Question Source:	Bank #	Perry 2015 # RO-60
	Modified Bank #	
	New	

Question History:	Previous 2 NRC Exams?	No

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

10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 61

Division 2 Emergency Diesel Generator is operating in parallel with the grid IAW SOI-R43, Section 7.2 with the following indications:

- Frequency – 60 HZ
- Real load – 100 KW
- Reactive load – 60 KVAR

It is desired to load the diesel to 2500 KW.

Which of the following actions will establish correct operating conditions per SOI-R43?

A. Using the Diesel Generator Governor adjust generator load to 2500 KW

B. Using the Diesel Generator Voltage Regulator adjust generator load to 2500 KW

C. Using the Diesel Generator Governor adjust reactive load to between 300 and 500 KVAR

D. Using the Diesel Generator Voltage Regulator adjust generator load to 1000 KW, hold for at least 5 minutes, then raise load to 2500 KW

QUESTION RO 61

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	264000	A4.05
	Importance Rating	3.6	

K&A: Ability to manually operate and/or monitor in the control room: Transfer of emergency generator (with load) to grid

Emergency Generators (Diesel/Jet) EDG

Explanation: **Answer A** – The Diesel Generator Governor is used to adjust real load (KW) and normally if rapid loading is not required, there is not a hold point until you reach 2500 KW.

B – Incorrect – plausible if the applicant confuses the function of the diesel voltage regulator with that of the diesel generator governor. The Diesel Voltage Regulator is used to adjust reactive load (KVAR), not real load.

C – Incorrect – again, plausible if the applicant confuses the function of the diesel voltage regulator with that of the diesel generator governor. The Diesel Generator Governor is used to adjust real load, and not reactive load.

D – Incorrect – see A and B. The Diesel Voltage Regulator is used to adjust reactive load, not real load and the requirement for a five minute hold point is at 2500 KW, before adding additional load.

Technical Reference(s): SOI-R43 Rev. 49	Reference Attached: SOI-R43 pp. 53-54

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-R43/48-G.4

Question Source:	Bank #	Nine Mile Unit 1 2009, # RO-51
	Modified Bank #	
	New	
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 62

The plant was operating at 75% rated thermal power with the following conditions:

- Transformer LH-2-C is tagged out for deluge testing.
- Loads supplied by LH-2-C were transferred to the Alternate source.

The transformer LH-2-B experienced a lockout

What is the consequence of this electrical transient?

A. Service Water Pump D, P41-C001D, <u>will</u> trip if running

B. Control Complex Chiller C, P47-B001C, <u>cannot</u> be started

C. Nuclear Closed Cooling Pump C, P43-C001C, <u>cannot</u> be started

D. Unit 2 Instrument Air Compressor, 2P52-C001, <u>will</u> trip if running

QUESTION RO 62

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

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

Instrument Air

Explanation: **Answer D –** A lockout on an LH transformer will cause the buses supplied from the transformer to transfer to the Alternate or Normal supply. U2 IAC is powered from Bus H22 which is normally powered from LH-2-C. But, since LH-2-C is tagged out, H22 is on its Alternate source - LH-2-B. The U2 IAC will trip if running.

A – Incorrect – Plausible since SWP C would trip.

B – Incorrect – Would be true of Lockout was on LH-2-A.

C – Incorrect – Would be true of Lockout was on LH-2-A.

Technical Reference(s): ARI-2H13-P870-01 Rev 10, ELI-R22 Rev 9, & Dwg. 256-016 Rev U	Reference Attached: ARI-2H13-P870-01 p 35, ELI-R22 pp.30-31, & Dwg. 256-016

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-P51_52 (#4)

Question Source:	Bank # Modified Bank # New	Perry 2017 # RO-62	
Question History:	Previous 2 NRC Exams?	Yes	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x	
10 CFR Part 55 Content:	55.41 55.43	x	

Comments: Level of Difficulty = x

QUESTION RO 63

The plant is operating at rated power when annunciator NCC SURGE TANK LEVEL HIGH alarms on H13-P970.

The following conditions are observed:

- NCC surge tank level is 67 inches and slowly rising.
- NCC HX OUTLET radiation monitor on H13-P906 is reading 68 cps, and slowly rising
- Pump Seal Water Disch temp on recorder B33-R601 is slowly rising

The most likely cause of the NCC SURGE TANK LEVEL HIGH alarm is due to a leak in the ____?

A. Radwaste evaporator

B. Fuel Pool Cooling Cleanup heat exchanger

C. Reactor Water Cleanup pump motor coolers

D. Reactor Recirculation pump heat exchangers

QUESTION RO 63

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

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

Component Cooling Water

Explanation: **Answer D** – Based on the increase in Pump Seal Water Disch temperature on recorder B33-R601 which monitors temperatures of Recirculation Pumps, and the slowly increasing NCC HX OUTLET radiation monitor indication; the most likely leak location is the reactor recirculation pump heat exchangers.

A, B, and C – Plausible because all of these components: are cooled by the NCC system, operate at pressures above those of the NCC system, and have the potential to cause an increase in the reading on the NCC HX OUTLET radiation monitor. However, a leak in any of these does not explain the increase in Pump Seal Water Disch temperature on recorder B33-R601.

Technical Reference(s): ARI-H13-P970-01-E1 Rev. 27	Reference Attached: ARI-H13-P970-01-E1 pp. 59-60

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-P43-02

Question Source:	Bank #	
	Modified Bank #	Columbia 2017 # RO-53
	New	

Question History:	Previous 2 NRC Exams?	No

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

10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 64

The plant is operating at rated power when power is lost to Bus ED-1-A.

What is the status of Reactor Recirculation Pumps?

A. B33-C001A and B33-C001B are OFF

B. B33-C001A and B33-C001B are running in SLOW speed

C. B33-C001A is OFF and B33-C001B is running in FAST speed

D. B33-C001A is running in SLOW speed and B33-C001B is running in FAST speed

QUESTION RO 64

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	202001	K2.01
	Importance Rating	3.2*	

K&A: Knowledge of electrical power supplies to the following: Recirculation pumps: Plant-Specific

Recirculation

Explanation: **Answer A –** DC Bus ED-1-A supplies control power and indication to the 3A & 3B Recirc Pump breakers. These breakers are in series with the 13.8 kV power supply and the 5A and 5B breakers and must be closed as a permissive to close the 5A/5B breakers (pumps in FAST). If Bus ED-1-A is lost, the 5A & 5B breakers see a loss of control power to the 3A & 3B breakers and will trip open with no downshift.

B – Incorrect – Plausible since a requirement to shift pumps to FAST speed is to have the respective 3A &3B breakers closed and the logic would see them as open with no power from ED-1-A.

C – Incorrect – Plausible since typically loads with a Functional Location designated with an "A" are associated with Division 1.

D – Incorrect – Plausible since a requirement to shift pumps to FAST speed is to have the respective 3A(3B) breaker closed and the logic would see them as open due to loss of power from ED-1-A. Also loads with a Functional Location designated with an "A" are typically associated with Division 1.

Technical Reference(s): ONI-R42-1 Rev. 7 and Lesson Plan B33 Rev. 3	Reference Attached: ONI-R42-1 p. 4 and Lesson Plan B33 p. 45

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-B33-L.2

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 65

The plant was at rated power with the Division 1 Emergency Diesel tagged out for maintenance when a Loss of Off-Site Power occurred

Current conditions are as follows:

- Power was just restored to Bus EH12.
- RPV water level is 180 inches and lowering at 10 inches per minute

An NSSSS (BOP) containment isolation will automatically occur __(1)__ .

Upon the automatic isolation, the RWCU __(2)__ PCIVs will close.

	(1)	(2)
A.	immediately	inboard
B.	immediately	outboard
C.	in 5 minutes	inboard
D.	in 5 minutes	outboard

QUESTION RO 65

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	204000	K6.07
	Importance Rating	3.5	

K&A: Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR WATER CLEANUP SYSTEM: PCIS/NSSSS

Reactor Water Cleanup

Explanation: **Answer A** – The BOP (NSSS) Isolation Relays are normally energized. They deenergize to cause isolation. Upon a LOOP, power to RPS Buses A & B is lost which de-energizes the BOP Isolation Relays to cause an isolation signal. Upon recovery of Div.2 power to the associated MOVs, the Isolation Valves will close. Division 1 (outboard) valves will not close since MOV power is not available, since the Div 1 D/G is tagged out.

B – Incorrect - Part 1 is correct, but Part 2 is incorrect since Division 1 valves do not have power and therefore the outboard valves will not close.

C – Incorrect - Plausible since a BOP & RWCU Isolation Signals are generated at L2 (130 inches). Additionally, the operator could incorrectly believe that the flywheels on the RPS Motor Generators will maintain power to the RPS Buses during the time that the Emergency Diesel Generators start and load. However, the flywheels will not support RPS power for more than 2-3 seconds before the RPS M/G Output Breaker will automatically trip. Part 2 is correct.

D – Incorrect - plausible, see C and B above.

Technical Reference(s): SDM B21(NS4) Rev. 7, SDM C71 Rev. 13, ONI-R10 Rev. 16, and ONI-C71-2 Rev. 9	Reference Attached: SDM B21(NS4) p. 53, 55-56, SDM C71 pp. 11-12, ONI-R10 p.6, ONI-C71-2 pp. 9 & 11

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-B21(NS4)-19

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 66

The following conditions exist:

- The plant is in a refueling outage with core alterations in progress.
- Troubleshooting is in progress on the Traversing In-core Probe (TIP) Drive Indexer Mechanism 'D' because it is stuck.

The TIP Indexer is then unstuck and the TIP detector is being inserted and withdrawn.

Which of the following hazards is created by this action?

A. High radiation exposure to personnel in Containment 599'

B. Damage to the TIP mechanism if fuel movement is in progress

C. High radiation exposure to personnel on the 360 Degree Platform

D. High airborne contamination level in Containment due to TIP purging

QUESTION RO 66

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	215001	A1.01
	Importance Rating	2.8	

K&A: Ability to predict and/or monitor changes in parameters associated with operating the TRAVERSING IN-CORE PROBE controls including: Radiation levels: (Not-BWR1)

Traversing In Core Probe

Explanation: **Answer A** – Per FTI-A001, operation of the TIP system may cause a very high radiation area in the vicinity of the TIP drive boxes, which are in containment on the 599' elevation.

B – Incorrect – The TIP probes are located within the LPRM tubes. No damage will occur due to fuel movement.

C – Incorrect – Personnel on the 360° platform are shielded by the upper pool.

D – Incorrect – The indexer enclosure is an air-tight unit. No air escapes into containment to cause high airborne contamination.

Technical Reference(s): FTI-A001 Rev 13, Survey Map – Containment 599', SDM C51 (TIP) Rev 7	Reference Attached: FTI-A001 pp. 3 & 6, Survey Map –Containment 599', SDM C51 (TIP) pp. 5 & 29

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-C51AP_OPRM-1.20

Question Source:	Bank #	Perry 2015 # RO-08
	Modified Bank #	
	New	
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 67

The plant is operating at rated power.

The status of Feedwater Control is as follows:

- DFWCS in Three Element Level Control
- RFPT A and B MANUAL/AUTO (M/A) STATIONS in AUTO
- Reactor Level "A" Transmitter (C34:N004A) failed low and is bypassed
- Reactor Level "B" Transmitter (C34:N004B) indicates 200 inches
- Reactor Level "C" Transmitter (C34:N004C) indicates 203 inches

Then the equalizing valve for the "B" Channel level detector develops seat leakage.

In response to this, reactor water level will initially __(1)__ .

If Reactor Level Transmitter "B" is later determined by the DFWCS to have a "Bad" output, the operator should respond by __(2)__ .

	(1)	(2)
A.	rise	bypassing C34:N004B and shifting to One Element Level Control
B.	lower	bypassing C34:N004B and shifting to One Element Level Control
C.	rise	using RFPT A and B M/A Stations in Manual to control level
D.	lower	using RFPT A and B M/A Stations in Manual to control level

QUESTION RO 67

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	216000	A2.01
	Importance Rating	2.9	

K&A: Ability to (a) predict the impacts of the following on the NUCLEAR BOILER INSTRUMENTATION; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Detector equalizing valve leaks

Nuclear Boiler Instrumentation

Explanation: **Answer D** – The leak will cause sensed ΔP to lower, which will cause the "indicated" level to rise. The DFWCS will respond to this "indicated" rising level by reducing RFPT speed which will cause actual RPV level to lower. When the difference between the 'B' & 'C' channels becomes large enough, DFWCS will 'throw out' 'B' channel. Upon the loss of both A & B (2 of 3) level transmitters, DFWCS will automatically shift RFPT A & B M/A Stations to Manual preventing Automatic control. The shift to Manual will occur even if the DFWCS is in One Element Control.

A – Incorrect – 1st part - The actual level will lower (See above); plausible if the applicant does not consider how the DFWLC system will try to mitigate the "indicated" level change. 2nd part – Plausible since bypassing the transmitter is required. However, Shift to 1E control is not appropriate. With 2 Reactor Level Transmitters not available, AUTO control of the RFPTs is not available.

B – Incorrect – Part (1) is correct. Part (2) is incorrect see A (2) above.

C – Incorrect – Part (1) is incorrect, see D above. Part (2) is correct.

Technical Reference(s): OT-3303-4 LP Rev. 4, SOI-C34 Rev. 36, and ARI-H13-P680-DFW Rev. 10	Reference Attached: OT-3303-4 LP pp. 35 & 136, SOI-C34 p. 6, and ARI-H13-P680-DFW pp. 22-23

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-C34-13

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 68

The Plant was operating at rated power with RHR A operating in Suppression Pool Cooling.

Then an automatic SCRAM occurred due to high Drywell pressure.

Three (3) minutes later; the BOP reports the following:

- All automatic actuations and isolations have been verified
- Div. 1 ECCS had to be manually initiated using the MANUAL INITIATION pushbutton, 1E21A-S9

Eight (8) minutes later, the following plant conditions exist:

- Reactor pressure is 350 psig and slowly lowering
- Reactor water level is 13 inches and stable
- Drywell pressure is 3 psig and slowly rising
- Containment pressure is 5 psig and slowly rising

Based on these conditions:

RHR Loop A is __(1)__ into the reactor vessel, and

RHR A TEST VALVE TO SUPR POOL, 1E12-F024A is __(2)__ .

	(1)	(2)
A.	not injecting	open
B.	not injecting	closed
C.	injecting	open
D.	injecting	closed

QUESTION RO 68

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	219000	A4.07
	Importance Rating	3.5	

K&A: Ability to manually operate and/or monitor in the control room: System Flows

RHR/LPCI: Torus/Suppression Pool Cooling Mode

Explanation: **Answer B** – RHR Loop 'A' is not injecting into the reactor vessel because shutoff head for the RHR pump is approximately 280 psig and RPV pressure is 350 psig. The RHR Test Valve to the Suppression Pool is closed, since a manual initiation using the LPCS & LPCI 'A' MANUAL INITIATION pushbutton, 1E21A-S9 will close it if it was open.

A – Incorrect – (1) is correct – see B above. (2) is incorrect but plausible since the position of RHR A Test Valve to the Suppression Pool is open when in Suppression Pool Cooling. Additionally, upon LPCI initiation, it takes several minutes for the RHR A Test Valve to the Suppression pool to reposition to the full closed position. Additionally, the RHR HX Outlet & Bypass valves remain open and cannot be closed for 10 minutes following a LOCA signal.

C – Incorrect – (1) plausible if the applicant does not recall the shutoff head pressure of the RHR pumps. (2) see 'A' above.

D – Incorrect – Parts (1) see B(2) above. Part 2 is correct

Technical Reference(s): SOI-E12 Rev. 77 and SDM-E12 Rev. 4	Reference Attached: SOI-E12 pp. 22-23 & 25 and SDM-E12 p. 45

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-Combined-E12-F

Question Source:	Bank #		Modified Bank #
	New	x	

Question History: Previous 2 NRC Exams? No

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

10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 69

The plant was operating at rated power when an earthquake occurred.

As of 1100, the following status exists:

- Suppression Pool Level is 18.0 feet lowering at 0.25 feet per minute.
- Drywell Pressure is 0.4 psig and rising at 0.1 psig per min
- RPV level is 140 inches and lowering 1 inch per min

Based on the above information, when will Suppression Pool Makeup Valves begin to automatically open?

A. 1105

B. 1110

C. 1113

D. 1140

QUESTION RO 69

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	223001	A3.01
	Importance Rating	3.4	

K&A: Ability to monitor automatic operations of the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES including: Suppression pool level

Primary Containment and Auxiliaries

Explanation: **Answer C** – Based on SP level lowering rate and DW pressure increase rate, conditions will be met at 1113 for the SPMU valves to automatically open and raise SP level. That is SP level <16.75' and DW pressure >1.68 psig.

A – Incorrect – This is the time that the SP level will reach 16.75" which is one permissive for the SPMU valves to automatically open. However, an RHR LOCA signal (L1 or 1.68 psig in DW) is also required for auto operation of the SPMU valves.

B – Incorrect – This is the time RPV level lowers to L2 which initiates a BOP LOCA signal (L2 or 1.68 psig in DW) not an RHR LOCA signal.

D – Incorrect – This is 30 minutes from the receipt of the BOP LOCA signal. Upon the receipt of an RHR LOCA signal the SPMU system starts a 30-minute timer to open the SPMU valves even if there is no low SP level condition.

Technical Reference(s): SOI-G43 Rev. 6, ARI-H13-P601-17 Rev. 20, and Lesson Plan T23 Rev. 5	Reference Attached: SOI-G43 p. 6, ARI-H13-P601-17-F1 p. 75, and Lesson Plan T23 pp. 52-53

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-G43-M.2

Question Source:	Bank # Modified Bank # New	 INL-36847
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 70

The plant is shutdown for a refuel outage with core alterations in progress.

A seismic event results in complete rupture of the common suction line of the Fuel Pool Cooling and Cleanup pumps.

Based on this information, EOP-3 Secondary Containment Control will <u>first</u> be entered based on Spent Fuel Pool __(1)__ .

To control this challenge, the Operators would __(2)__ .

	(1)	(2)
A.	level	makeup from Condensate Transfer System
B.	level	makeup from Fire Water System using hoses
C.	temperature	place RHR B in Fuel Pool Cooling Assist
D.	temperature	supply FPCC heat exchangers with ESW Cooling

QUESTION RO 70

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	233000	2.4.1
	Importance Rating	4.6	

K&A: Knowledge of EOP entry conditions and immediate action steps.

Fuel Pool Cooling/Cleanup

Explanation: **Answer C** – Spent Fuel Pool water level is controlled by the skimmers and is slightly above the skimmers with the FPCC pumps running. With a rupture on the FPCC pump suction line, water in the Spent Fuel Pool will only lower to the top of the skimmers. Since heat removal is lost, SPF temperature will rise. A CAUTION in EOP-3 warns of water expansion and pool overflow if heat removal is lost. EOP-3 is entered on a SFP temp of 127 °F or level 3” below normal. Therefore EOP-3 would first be entered on SFP temperature. Also, using RHR is a viable option for controlling SFP temperature in this situation

A – Incorrect – 1st part – SFP level will not lower to EOP-3 entry condition prior to SPF temperature exceeding 127 °F based on thermal expansion. 2nd part – Plausible since CTS is the normal M/U to FPCC via the surge tanks. However, CTS would not fill the surge tanks based on pipe break location.

B – Incorrect – 1st part – SFP level will not lower to EOP-3 entry condition prior to SPF temperature exceeding 127 °F based on thermal expansion. 2nd part – Plausible since Fire Water is a viable method of filling the pools if necessary.

D – Incorrect – 2nd part – Plausible since this is a method of cooling the pools. However, the FPCC pumps are needed to circulate water through the HXs to the pools.

Technical Reference(s): EOP-03/04 Bases Rev. 0, SDM-G41 Rev. 7, and Dwgs. 302-654 Rev. U & 302-655 Rev. CC	Reference Attached: EOP-03/04 Bases pp. 8, SDM-G41. 2, and Dwgs. 302-654 & 302-655

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-G41-R

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 71

The plant is operating at rated power.

Feedwater Heaters 5A/B receive drains from __(1)__ and Extraction Steam from __(2)__ ?

	(1)	(2)
A.	Second Stage Reheaters	7th stage of the HP Turbine
B.	Feedwater Heaters 6A/B	7th stage of the HP Turbine
C.	Second Stage Reheaters	7th stage of the LP Turbine
D.	Feedwater Heaters 6A/B	7th stage of the LP Turbine

QUESTION RO 71

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	245000	K1.05
	Importance Rating	2.7	

K&A: Knowledge of the physical connections and/or cause effect relationships between MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS and the following: Extraction steam system

Main Turbine Generator/Auxiliary

Explanation: **Answer** B – Feedwater Heaters 5A/B receive drains from the Feedwater Heaters 6A/B and Extraction Steam from the 7th stage of the HP Turbine.

A – Incorrect – plausible, in that (1) the 6A/B FWD HTRS receive drains from the second stage Reheaters. The Feedwater Heaters 5A/B do receive extraction steam from the 7th stage of the HP Turbine.

C – Incorrect – See A and from the HP Turbine 7th, not the LP Turbine

D – Incorrect – (1) is correct, see C above for (2)

Technical Reference(s): SDM-N36/25/26 Rev. 9	Reference Attached: SDM-N36/25/26 pp. 64,65, and 68

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-N36/25/26-B.1 and B.2

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 72

The plant is operating at rated power.

Offgas Post Treatment Radiation monitors 1D17-K601A and K601B alarmed on a High-High Radiation signal causing annunciator OG ISOL OG POST-TREAT PRCS RAD A/B 3XHI – H13-P604-0001-A5 to alarm.

Which of the following describes the effect on the plant due to this condition?

A. Offgas isolates causing a degradation of main condenser vacuum.

B. Offgas shifts into the Treat Mode allowing main condenser vacuum to remain stable.

C. Offgas shifts into the Bypass Mode causing a degradation of main condenser vacuum.

D. The Offgas Charcoal Adsorber Discharge valves open allowing main condenser vacuum to remain stable.

QUESTION RO 72

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	271000	K3.01
	Importance Rating	3.5	

K&A: Knowledge of the effect that a loss or malfunction of the OFFGAS SYSTEM will have on following: Condenser vacuum

Offgas

Explanation: **Answer A** – A HI-HI signal from both OG Pre-treat rad monitors causes the Offgas system to isolate. When OG isolates, the non-condensable gases leaking into the main and aux condensers cannot be removed and main condenser vacuum will degrade.

B – Incorrect – Plausible since OG is in Auto Mode at low power. However, OG should be shifted from Auto Mode to Treat Mode at 5% power.

C – Incorrect – Plausible since S45, (TREAT-AUTO-BYPASS) mode select switch controls OG flow. With this switch in AUTO OG will shift to TREAT, if necessary. However, OG will not auto shift into Bypass Mode

D – Incorrect – Plausible since the OG Discharge Isolation valves open on a single HI if the S45 MODE select switch is in AUTO. However, at rated power the MODE select switch would be in TREAT.

Technical Reference(s): ARI-H13-P604-001-A5 Rev. 7 and ARI-H13-P680-02-A1 Rev. 12	Reference Attached: ARI-H13-P604-001-A5 pp. 7 and ARI-H13-P680-02-A1 p. 3

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-N64-F

Question Source:	Bank #	Perry 2010 # RO-07
	Modified Bank #	
	New	

Question History: Previous 2 NRC Exams? No

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

10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 73

The plant was operating at rated power when annunciator RHR C PUMP ROOM SUMP LEVEL HIGH alarms.

The NI-RRA reports that there is approximately two feet of water in the RHR C Pump Room Sump with water leaking from the Division 2 RHR Water Leg Pump.

Based on this information:

An EOP-03, Secondary Containment Control, Maximum Safe Operating Condition __(2)__ been exceeded.

The RHR Loop C Pump Cubicle Sump Drain Valve, 1G61- F580 is normally __(2)__ .

	(1)	(2)
A.	has	open
B.	has	closed
C.	has not	open
D.	has not	closed

QUESTION RO 73

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	290001	K4.03
	Importance Rating	2.8	

K&A: Knowledge of SECONDARY CONTAINMENT design feature(s) and/or interlocks which provide for the following: Fluid leakage collection

Secondary Containment

Explanation: **Answer D** – The Max Safe Condition for RHR C Pump Room is the top of the room floor grading, which is at 574' elevation. The Sump Level Hi alarm comes in near the top of the room sump, which is at the 568' elevation. Therefore, the Max Safe has not been exceeded. However, the Maximum Normal Operating Water Level has been exceeded. Additionally, 1G61-F580 is normally closed to provide isolation of the RHR C Pump Room. This room isolation design is important in the event of a Suppression Pool leak into the RHR C Pump Room. If RHR C room was filled such that Suppression Pool level and RHR C room level were equalized, the Suppression Pool would still be able to perform its designed pressure suppression function.

A – Incorrect – 1st part - Plausible since receipt of the RHR C PUMP ROOM SUMP LEVEL HIGH alarm is an entry condition into EOP-03. 2nd part – Plausible since most specific area sump drains are aligned to drain to the larger building sumps.

B – Incorrect – 1st part - Plausible since receipt of the RHR C PUMP ROOM SUMP LEVEL HIGH alarm is an entry condition into EOP-03..

C – Incorrect – 2nd part – Plausible since most specific area sump drains are aligned to drain to the larger building sumps.

Technical Reference(s): ARI-H13-P601-18 Rev. 17, EOP-03/04 Bases Rev. 0, PYBP-POS-30 Rev. 6, RLI-G61(FDSS) Rev. 2, and DWG 302-740 Rev. FF	Reference Attached: ARI-H13-P601-18 pp. 41-43, EOP-03/04 Bases p. 9, PYBP-POS-30 p. 11, RLI-G61(FDSS) p. 9, and DWG 302-740

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-Combined-T23-C

Question Source:	Bank #	
	Modified Bank #	
	New	x

Question History: Previous 2 NRC Exams? No

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

10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 74

The plant is operating at rated power.

Control Complex Chill Water Chiller C, P47-B001C is in service.

A total loss of __(1)__ flow will cause control room temperature to rise.
Control room temperature is monitored from __(2)__ .

	(1)	(2)
A.	Nuclear Closed Cooling	H13-P904, Common HVAC Control Panel
B.	Nuclear Closed Cooling	H13-P800, Heating Ventilation And Air Conditioning Control Panel
C.	Emergency Closed Cooling	H13-P904, Common HVAC Control Panel
D.	Emergency Closed Cooling	H13-P800, Heating Ventilation And Air Conditioning Control Panel

QUESTION RO 74

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	290003	K5.03
	Importance Rating	2.6	

K&A: Knowledge of the operational implications of the following concepts as they apply to CONTROL ROOM HVAC: Temperature control

Control Room Ventilation

Explanation: **Answer A** – The P47C chiller can supply either P47 A or B cooling loops. This chiller receives cooling water from NCC. A total loss of NCC would cause the chiller to trip and cooling would be lost to the control room. Control Room temperatures are monitored from H13-P904 using Riley Modules.

B – Incorrect – 2nd part – Plausible since H13-P800 has Riley Modules used for temperature monitoring. However, these modules monitor temperatures of in-plant locations only.

C – Incorrect – 1st part – Plausible since ECC supplies cooling water to the P47 A and B chillers. However, P47C chiller is supplied from NCC.

D – Incorrect – 1st part – Plausible since ECC supplies cooling water to the P47 A and B chillers. However, P47C chiller is supplied from NCC. 2nd part – Plausible since H13-P800 has Riley Modules used for temperature monitoring. However, these modules monitor temperatures of in-plant locations only.

Technical Reference(s): ARI-H13-P904-01 Rev. 12, ARI-H13-P904-02 Rev. 13	Reference Attached: ARI-H13-P904-01 p. 53, ARI-H13-P904-02 p. 53

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-M25_26-E.1 & OT-COMBINED-P47-L.1

Question Source:	Bank # Modified Bank # New	 x
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 75

The plant was operating at 100% rated power when a failure of both Pressure Regulators occurred causing the following annunciators to alarm:

- RX PRESS HI. H13-P680-0007-D1
- RPS RX PRESS HI. H13-P680-0005-A8
- RRCS RX PRESS HI. H13-P680-0005-A2

Due to other equipment failures, the SRV's operated only on Spring Set Pressure.

Based on this information __(1)__ were/was exceeded.
To control RPV pressure, use the __(2)__ .

	(1)	(2)
A.	the Reactor Coolant System Pressure Safety Limit and the Reactor Steam Dome Pressure Technical Specification limit	SRVs
B.	the Reactor Coolant System Pressure Safety Limit and the Reactor Steam Dome Pressure Technical Specification limit	MAX COMBINED FLOW LIMIT
C.	only the Reactor Steam Dome Pressure Technical Specification limit	SRVs
D.	only the Reactor Steam Dome Pressure Technical Specification limit	MAX COMBINED FLOW LIMIT

QUESTION RO 75

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	290002	A2.02
	Importance Rating	3.6	

K&A: Ability to (a) predict the impacts of the following on the REACTOR VESSEL INTERNALS: and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: †Over pressurization transient

Reactor Vessel Internals

Explanation: **Answer C** – Only the TS limit for reactor steam dome pressure (1045 psig) will be exceeded based on annunciator for RRCS pressure high (1083 psig) and the SRVs opening on spring set pressure – safety mode (highest setpoint – 1190 psig). USAR Chapter 15 section 15.2.1.4.2 indicates the peak pressure for this analyzed event is 1180 psig which is below the Safety Limit value of < or = 1325 psig. The high pressure condition results if both pressure regulators failed closed, causing the turbine bypass valves and turbine control valves to fail closed, and requires either the turbine bypass jack (MSIVs open) or the SRVs (MSIVs closed) to be manually opened to control RPV pressure.

A – Incorrect – The max RPV pressure should be 1180 psig which is below the SL of 1325 psig. Plausible if applicant believes the peak pressure will exceed the Safety Limit value.

B – Incorrect – The max RPV pressure should be 1180 psig which is below the SL of 1325 psig and the MAX COMBINED FLOW LIMIT (MCFL) setpoint adjustment would control pressure. Plausible if applicant believes the MCFL adjustment would be effective and peak pressure will exceed the Safety Limit value.

D – Incorrect - the MAX COMBINED FLOW LIMIT (MCFL) setpoint adjustment would be ineffective. Plausible if applicant believes the MCFL adjustment would control pressure.

Technical Reference(s): TS 2.1.2 Rev. Amend 188, TS 3.4.12 Rev. Amend. 171, ONI-B21-1 Rev. 11, ONI-C85 Rev. 1, ARI-H13-P680-05 Rev 16, & USAR Chapt 15, 15.2.1.4.2 Rev 12	Reference Attached: TS 2.1.2 p 2.0-1, TS 3.4.12 p 3.4-32, ONI-C85 pp. 9 & 11, ARI-H13-P680-05 p. 7, & USAR Chapt 15 p 15.2-6

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): x

Question Source:	Bank #	Perry 2013 # RO-75
	Modified Bank #	
	New	
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION SRO 1

The plant was operating at rated power when a manual reactor scram was inserted due to an electrical transient.

SPDS and the Full Core Display have been lost.

The RO has performed only the following:

- Placed Mode Switch in SHUTDOWN
- Inserted SRMs
- Inserted & ranged IRMs

The following conditions now exist:

- MSIVs are closed
- Two SRVs are open
- RPV pressure is 940 psig and rising slowly
- RPV level is 120 inches and lowering slowly
- Suppression Pool temperature is 102 °F and rising
- IRMs are reading 25 to 30 on range 10

Based on the above information, which of the following procedures contain the specific steps to mitigate the consequences for the given conditions?

A. EOP-01, RPV Control

B. EOP-01-05, ATWS RPV Control

C. ONI-B21-1, SRV Inadvertent Opening/Stuck Open

D. EOP-SPI 2.3, Bypass MSIVs Low Level Interlocks

QUESTION SRO 1

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	Generic	2.1.45
	Importance Rating		4.3

K&A: Ability to identify and interpret diverse indications to validate the response of another indication.

Generic

Explanation: **Answer B –** EOP-01-05 chart directs Performance of Scram and ARI and initiation of SLC if Rx power is above 4%. With 2 SRVs open and pressure increasing, the Rx is still at ~10% power. This indication is backed up by IRM readings. Since the Suppression Pool is heating up, SLC must be imitated prior to reaching 100 °F in SP.

A – Incorrect – Plausible since RPV level is 120" and entry for EOP-01 is at 128". However, one of the first decisions in EOP-1 is to determine if APRMs are downscale. Since the APRM DNSC lights are out, this decision would be incorrect.

C – Incorrect – Plausible since two SRVs are open. However, they are responding as designed with MSIVs closed and Rx at ~10% power.

D – Incorrect – Plausible since EOP-SPI 2.3 is directed in an ATWS. However, it is given in the stem that the MSIVs are already shut. This would require performance of EOP-SPI 9.2, Opening MSIVs, which also bypasses the Low Level Interlocks..

Technical Reference(s): EOP-01-05 Chart Rev. A, and SDM-C51(IRM) Rev. 8	Reference Attached: EOP-01-05 Chart and SDM-C51(IRM) pp. 29-30

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3402-03-D.1

Question Source:	Bank #	
	Modified Bank #	Columbia 2015 # SRO-08(83)
	New	

Question History: Previous 2 NRC Exams? No

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

10 CFR Part 55 Content:	55.41	
	55.43	b(5)

SRO Justification: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]

- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps.

QUESTION SRO 2

Five minutes after a Rx power change, FMEOD is observed to be 1.001.
IAW IOI-3, FMEOD is required to be restored to __(1)__ by __(2)__ .

	(1)	(2)
A.	≤0.990	inserting control rods IAW ONI-C51
B.	≤0.990	lowering Rx Recirc flow IAW SOI-B33
C.	≥ 1.010	inserting control rods IAW ONI-C51
D.	≥ 1.010	lowering Rx Recirc flow IAW SOI-B33

QUESTION SRO 2

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	Generic	2.1.32
	Importance Rating		4.0

K&A: Ability to explain and apply system limits and precautions.

Generic

Explanation: **Answer A –** IOI-3 P&L requires FMEOD ≤ 0.990 (to provide a 1% margin to the left sloping side of the MEOD power limit). If FMEOD > 0.990, insert control rods to restore FMEOD ≤ 0.990 per Reactor Engineering direction. If FMEOD > 1.000, enter ONI-C51, Unplanned Change in Reactor Power or Reactivity. FMEOD is calculated (by the computer) comparing heat balance reactor and instantaneous reactor flow to a preprogrammed graph.

B – Incorrect – 1st part – Correct. 2nd part – Plausible since Rx power must be lowered to restore FMEOD within limit. However, using recirc flow to lower power will not appreciably change the value of FMEOD.

C – Incorrect – 1st part – Plausible since FMEOD is an inverse function of MEOD. 2nd part – Correct.

D – Incorrect – 1st part – Plausible since FMEOD is an inverse function of MEOD. 2nd part – Correct. 2nd part – Plausible since Rx power must be lowered to restore FMEOD within limit. However, using recirc flow to lower power will not appreciably change the value of FMEOD.

Technical Reference(s): IOI-3 Rev. 80 and ONI-C51 Flowchart Rev. N	Reference Attached: IOI-3 p. 6 and ONI-C51 Flowchart Rev. N

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3035-08(LP)-B.6

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

SRO Justification - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

- Knowledge of diagnostic steps and decision points in the emergency operating procedures (EOP) that involve transitions to event specific sub-procedures or emergency contingency procedures.

QUESTION SRO 3

It is night shift and the plant is at rated power.

As the Shift Manager, you discover that work on the C Hotwell Pump (HWP) breaker scheduled for release on your shift will cause Plant Risk to become Red.

Which of the following must be done?

A. Perform the HWP breaker work around the clock.

B. Prevent the scheduled work for the HWP pump breaker.

C. Post Protected Equipment signs prior to starting work on the HWP.

D. Verify the OCC is activated to provide support prior to releasing the HWP work.

QUESTION SRO 3

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	Generic	2.2.17
	Importance Rating		3.8

K&A: Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator.

Generic

Explanation: **Answer B –** IAW NOP-OP-1007, Red risk shall not be intentionally entered. Red Risk is entered from emergent conditions only.

A – Incorrect – Plausible since the work group will provide 24/7 coverage as required for Orange and Yellow Risk activities. However, Red Risk shall not be intentionally entered.

C – Incorrect – Plausible since Protected Equipment postings are required for all high Risk activities. However, Red Risk shall not be intentionally entered.

D – Incorrect – Plausible since it is a requirement to activate the OCC for Red Risk conditions. However, intentionally entering Red Risk is not allowed.

Technical Reference(s): NOP-OP-1007 Rev. 31	Reference Attached: NOP-OP-1007 pp. 18

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3039-02-D

Question Source:	Bank #	Clinton 2013 # SRO-20
	Modified Bank #	
	New	

Question History:	Previous 2 NRC Exams?	No

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

10 CFR Part 55 Content:	55.41	
	55.43	Plant Specific

SRO Justification for Plant Specific Exemption - knowledge/ability is “unique to the SRO position” SRO Task: Assess and classify the risk of an activity
SM Task: Terminate tests or evolutions that will or did place the plant in an unacceptable condition.

QUESTION SRO 4

Which of the following Limiting Conditions for Operation (LCOs) listed below provide guidance concerning a supported system LCO not being met because the support system LCO is not met?

A. LCO 3.0.4

B. LCO 3.0.5

C. LCO 3.0.6

D. LCO 3.0.7

QUESTION SRO 4

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	Generic	2.2.38
	Importance Rating		4.5

K&A: Knowledge of conditions and limitations in the facility license.

Generic

Explanation: **Answer C –** This one is for support system LCOs

A - Incorrect – plausible as these are specific LCOs with unique properties and an applicant may confuse these. This one is for mode changes and LCOs

B - Incorrect – plausible as these are specific LCOs with unique properties and an applicant may confuse these. This for equipment returned to service administratively for testing

D - Incorrect – plausible as these are specific LCOs with unique properties and an applicant may confuse these. This is for special operations LCOs

Technical Reference(s): TS Section 3.0 Rev. Amend. 182	Reference Attached: TS Section 3.0, pp. 3.0-2 and 3.0-3

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3037-04-8

Question Source:	Bank #	River Bend 2016 # SRO-96
	Modified Bank #	
	New	

Question History: Previous 2 NRC Exams? No

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

10 CFR Part 55 Content:	55.41	
	55.43	(b)2

SRO Justification - Facility operating limitations in the TS and their bases.

- Application of generic Limiting Condition for Operation (LCO) requirements (LCO 3.0.1 thru 3.0.7; SR 4.0.1 thru 4.0.4).

QUESTION SRO 5

A plant transient resulted in fuel damage and off-site release.

- You, as the Shift Manager declared a Site Area Emergency
- The TSC is in the process of being activated
- In this emergency situation you must waive a plant worker's Federal 10CFR20 TEDE dose limit in order to stop the release and protect the population within the Emergency Planning Zone.

Which of the following is the recommended maximum emergency TEDE dose you can authorize the plant worker to receive in accordance with HPI-B0003, Processing of Personnel Dosimetry?

A. 5000 mrem

B. 10,000 mrem

C. 15,000 mrem

D. 25,000 mrem

QUESTION SRO 5

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	Generic	2.3.14
	Importance Rating		3.8

K&A: Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Generic

Explanation: **Answer D –** IAW HPI-B-003, The Emergency Coordinator can waive the 10CFR20 limits and authorize a dose increase to 25 rem TEDE for protecting large populations. In this case the Shift Manager is acting as the Emergency Coordinator

A – Incorrect – Plausible since this is the 10CFR20 Limit and the limit for a Planned Special Exposure (PSE).

B – Incorrect – Plausible since this is the HPI-B-003 limit for protecting valuable property.

C – Incorrect – Plausible since this is the LDE (Lens Dose Equivalent) limit for emergency services.

Technical Reference(s): HPI-B-003 Rev. 28	Reference Attached: HPI-B-003 p. 4

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3039-03-1

Question Source:	Bank #	Perry 2002 # SRO-100
	Modified Bank #	
	New	

Question History: Previous 2 NRC Exams? No

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

10 CFR Part 55 Content:	55.41	
	55.43	b(4)

Comments: SRO Justification - Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

- It is the shift Managers responsibility to authorize exceeding 10CFR20 limits when acting as EC.

QUESTION SRO 6

Which of the following responsibilities does the Shift Manager retain after transfer of Emergency Coordinator duties during implementation of the Emergency Plan?

A. Termination of the emergency event.

B. Re-classification of the emergency event.

C. Transition to the Severe Accident Management Guidelines.

D. Approval of protective action recommendations for the general public.

QUESTION SRO 6

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	Generic	2.4.16
	Importance Rating		4.4

K&A: 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.

Generic

Explanation: **Answer C –** Transition into the SAMGs is directed by the EOPs and remains a responsibility of the Shift Manager.

A, B, D – plausible because the Shift Manager is the Emergency Coordinator (EC) until EC duties are transferred to TSC Ops Manager or EOF EC when their facility is operational. These are responsibilities of the Emergency Coordinator and would be transferred.

Technical Reference(s): EPI-A2 Rev. 24 & EOP-01-1 Rev. 0	Reference Attached: EPI-A2 Rev 24 pp. 2 and 4, and EOP-01-1 p.16

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): EPL-0804-01-1

Question Source:	Bank #	Perry 2003 Audit # SRO-79
	Modified Bank #	
	New	

Question History:	Previous 2 NRC Exams?	No

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

10 CFR Part 55 Content:	55.41	
	55.43	(b)5

SRO Justification - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

- Knowledge of diagnostic steps and decision points in the emergency operating procedures (EOP) that involve transitions to event specific subprocedures or emergency contingency procedures.

QUESTION SRO 7

Consider the following reports from the Security Shift Supervisor:

1. A security intrusion has been detected in the protected area and the intruders appear to be armed and moving in a hostile manner.
2. A loss of plant security control is imminent.
3. A credible security threat exists that is not likely to result in a direct challenge within the next 30 minutes.
4. A credible, imminent land based threat is occurring.

Which of the above reports would require the Command SRO to direct a plant shutdown by reactor scram in accordance with ONI-P56-2 Land Based Security Threat?

A. 1, 2, 3

B. 2, 3, 4

C. 1, 3, 4

D. 1, 2, 4

QUESTION SRO 7

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	Generic	2.4.28
	Importance Rating		4.1

K&A: Knowledge of procedures relating to a security event (non-safeguards information).

Generic

Explanation: **Answer D –** The communication of a Security Threat is given to the Shift Manager from the Security Shift Supervisor. Reports 1, 2, 4 are considered credible and imminent security threats which require a Plant Shutdown by Reactor Scram

A – Incorrect – Report 3 is considered a credible, though not imminent, security threat and does not require a Plant Shutdown by Reactor SCRAM. Reports 1 and 2 are considered credible, imminent, security threats which require a Plant Shutdown by Reactor Scram

B – Incorrect- Report 3 is considered a credible, though not imminent, security threat and does not require a Plant Shutdown by Reactor SCRAM. Reports 2 and 4 are considered credible, imminent, security threats which require a Plant Shutdown by Reactor Scram

C – Incorrect - Report 3 is considered a credible, though not imminent, security threat and does not require a Plant Shutdown by Reactor SCRAM. Reports 1 and 4 are considered credible, imminent, security threats which require a Plant Shutdown by Reactor Scram

Technical Reference(s): ONI-P56-2 Rev. 23	Reference Attached: ONI-P56-2 pp. 7 & 12

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3035-16(LP)-A.2

Question Source:	Bank #	Perry 2015 # SRO-06
	Modified Bank #	
	New	

Question History: Previous 2 NRC Exams? No

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

10 CFR Part 55 Content:	55.41	
	55.43	b(5)

SRO Justification - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]

- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps

QUESTION SRO 8

The plant was at 12% RTP with startup in progress when B Rx Recirculation Pump tripped.

All actions for Single Loop operation were completed and power ascension continued.

Repairs on the B Rx Recirculation Pump were completed.

B Rx Recirculation pump was restarted and loop flows matched on April 4th at 0800.

Reactor power exceeded 25% RTP on April 4th at 1400

The <u>latest</u> time that Surveillance Requirement 3.4.3.1 must be completed for B loop jet pumps is ____.

Reference Provided:

A. 1200 on April 4th

B. 1300 on April 4th

C. 1400 on April 5th

D. 1800 on April 5th

QUESTION SRO 8

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	295001	AA2.05
	Importance Rating		3.4

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

Partial or Complete Loss of Forced Core Flow Circulation

Explanation: **Answer C –** SR 3.4.3 is modified by 2 NOTES. #1 allows this SR to not be performed until 4 hours after the loop is in operation. #2 allows this SR to not be performed when RTP≤25%. Therefore, 1400 on April 5th is the latest this SR can be performed.

A – Incorrect – Plausible since this is 4-hours from starting B Recirc pump. However, thermal power is still <25% and NOTE 2 TS Bases SR 3.4.3.1 does not allow performance of this surveillance until >25% power.

B – Incorrect – Plausible since this is 4-hours + 25% allowance in SR 3.0.2 from starting B Recirc pump. However, SR 3.0.2 is based on frequency and not completion time. Additionally, NOTE 2 TS Bases SR 3.4.3.1 does not allow performance of this surveillance until >25% power.

D – Incorrect – Plausible since this is 24-hours + 25% - SR 3.0.2 application. However, since this is the initial performance the SR 3.0.2 allowance cannot be used since this the initial performance of the surveillance.

Technical Reference(s): TS 3.4.1 Rev. Amend. 171, TS 3.4.3.Bases Rev. 11, TS 3.0 Rev Amend. 182, and PDB-R002 Rev. 10	Reference Attached: TS 3.4.1 p. 3.4-8, TS 3.4.3.Bases p. B 3.4-17, TS 3.0 p. 3.0-4, and PDB-R002 p. 36

Proposed references to be provided to applicants during examination: TS 3.4.1 & PDB-R002 (partial & modified)

Learning Objective (As available): OT-3037-08

Question Source:	Bank #	
	Modified Bank #	LaSalle 2003 # SRO-108
	New	
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	
	55.43	b(2)

SRO Justification - Facility operating limitations in the TS and their bases.

- Application of Required Actions (Section 3) and Surveillance Requirements (SR) (Section 4) in accordance with rules of application requirements (Section 1).

QUESTION SRO 9

Thirty minutes ago the plant was operating at rated power when breaker 1CB6, BUS ED-1-C MAIN BREAKER tripped open due to a battery fault.

The following annunciators have alarmed on 1H13-P601:

- DIV 3 BATTERY DC SYSTEM TROUBLE
- DC BUS ED-1-C UNDERVOLTAGE

The current status of High Pressure Core Spray is __(1)__ .

If bus ED-1-C is energized from bus ED-2-C via the Unit 2 battery 2E22-S005 and Unit 2 Normal Charger EFD-2-C, then bus ED-1-C will be __(2)__ .

	(1)	(2)
A.	Inoperable	Operable
B.	Inoperable	Inoperable
C.	Operable	Inoperable
D.	Operable	Operable

QUESTION SRO 9

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	295004	2.1.27
	Importance Rating		4.0

K&A: Knowledge of system purpose and/or function.

Partial or Total Loss of DC Power

Explanation: **Answer B –** With CB-6 opening, Div. 3 DC Bus de-energizes. Per TS 3.8.7 Condition D, HPCS must be declared INOP immediately. Additionally, per T.S. 3.8.4 Bases, the Unit 2 Normal charger is not listed for OPERABILITY of Div 3 DC subsystem (not powered from a Class 1E source).

A – Incorrect – (2) With ED-1-C powered from the Unit 2 charger, the bus is INOP. Plausible, if the applicant believes that either the Unit 2 charger is diesel backed (i.e., a Class 1E source) or that because you have a backup source you meet the T.S. requirement.

C – Incorrect – When Div 3 DC becomes INOP; HPCS must be declared INOP immediately. Plausible since this may be confused with the 1-hour to verify RCIC/HPCS Operable with HPCS/RCIC INOP.

D – Incorrect – When Div 3 DC becomes INOP; HPCS must be declared INOP immediately. Plausible since this may be confused with the 1-hour to verify RCIC/HPCS Operable with HPCS/RCIC INOP. With ED-1-C powered from the Unit 2 charger, the bus is INOP. Again, plausible, if the applicant believes that either the Unit 2 charger is diesel backed (i.e., a Class 1E source) or that because you have a backup source you meet the T.S. requirement.

Technical Reference(s): ARI-H13-P601-16 Rev. 20, TS 3.8.4 Rev. Amend 69, TS 3.8.4 Bases Rev. 7, TS 3.8.7 Rev. Amend 171, SOI-R42(DIV3) Rev. 10, SDM R42 Rev. 10	Reference Attached: ARI-H13-P601-16 p. 78, TS 3.8.4 p. 3.8-24, TS 3.8.4 Bases p. B 3.8-53, TS 3.8.7 p. 3.8-37, SOI-R42(DIV3) p. 33, SDM R42 p32

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): x

Question Source:	Bank #	Perry 2015 # SRO-08
	Modified Bank #	
	New	

Question History: Previous 2 NRC Exams? No

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

10 CFR Part 55 Content:	55.41	
	55.43	b(2)

SRO Justification - . Facility operating limitations in the TS and their bases.

- Knowledge of TS bases that are required to analyze TS required actions and terminology.

QUESTION SRO 10

The plant was operating at rated power when a fire in panel P680 required immediate evacuation of the Control Room.

No actions were performed prior to evacuation.

Based on this information, __(1)__ will be used to shutdown the reactor.

Ten minutes after performing Rx shutdown actions, the following is observed at the Division 1 Remote Shutdown Panel:

- SPDS has been lost.
- MSIVs are closed
- SRV 1B21-F051D is open & SRV 1B21-F051C is cycling

Based on this information, the reactor __(2)__ shutdown.

	(1)	(2)
A.	ONI-C61, Evacuation Of The Control Room	is
B.	IOI-11, Shutdown From Outside The Control Room	is
C.	ONI-C61, Evacuation Of The Control Room	is not
D.	IOI-11, Shutdown From Outside The Control Room	is not

QUESTION SRO 10

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	295016	AA2.01
	Importance Rating		4.1*

K&A: Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT: Reactor power

Control Room Abandonment

Explanation: **Answer C** – 1st part – ONI-C61 contains the actions to shutdown the Rx if a scram was not able to be inserted prior to evacuating the Control Room. 2nd part - With 1 SRV open and 1 SRV cycling, the Rx is still producing power. One SRV open equates to ~ 5% Rx power. Since an additional SRV is cycling the Rx still has to be at power. Typically, 10 minutes after a scram with MSIVs closed, 1 SRV should be able to handle decay heat.

A – Incorrect – 2nd part - Plausible since the MSIVs are closed; SRVs would need to cycle to control Rx pressure. However, typically 10 minutes after a scram with MSIVs closed, cycling 1 SRV should be able to handle decay heat.

B – Incorrect – 1st part – Plausible since IOI-11 is used to stabilize the plant. However, it does not contain steps to shutdown the Rx.

D – Incorrect. – 1st part – Plausible since IOI-11 is used to stabilize the plant. However, it does not contain steps to shutdown the Rx. 2nd part - Plausible since the MSIVs are closed, SRVs would need to cycle to control Rx pressure. However, typically 10 minutes after a scram with MSIVs closed, 1 SRV should be sufficient to remove decay heat.

Technical Reference(s): PYBP-POS-30 Rev. 5, and ONI-C61 Rev. 9 and IOI-11 Rev. 38	Reference Attached: PYBP-POS-30 pp. 9, ONI-C61 pp. 5-6 and IOI-11 pp. 4 & 6

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3035-13(LP)-A.2

Question Source:	Bank #	
	Modified Bank #	Clinton 2007 # SRO-90
	New	

Question History:	Previous 2 NRC Exams?	No

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

10 CFR Part 55 Content:	55.41	
	55.43	b(5)

SRO Justification - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps.

QUESTION SRO 11

The plant is in a refuel outage with a fuel shuffle in progress when the following occurs:

- Bubbles are seen rising from a fuel bundle removed from location RX 07-12after it was bumped against the steam dam.
- The containment evacuation alarm activates.

(1) Based on these conditions, who should be evacuated?
(2) Using a FREE MOVE, where should the damaged fuel bundle be placed?

Attachment Provided

	(1)	(2)
A.	RP Tech	RX1 07-12
B.	RP Tech	RP1 F-1
C.	Spotter	RX1 07-12
D.	Spotter	RP1 F-1

QUESTION SRO 11

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	295023	2.1.25
	Importance Rating		4.2

K&A: Ability to interpret reference materials, such as graphs, curves, tables, etc.

Refueling Accidents

Explanation: **Answer B –** The RP tech is not considered required personnel to place the fuel in a safe location and must be evacuated during a fuel handling accident. For a FREE MOVE, the fuel bundle may not be returned to the reactor and must be placed in an authorized location outside the reactor.

A – Incorrect – plausible in that part one is correct. The second one is plausible because that is the location that the bundle was originally removed from.

C – Incorrect - plausible because the RP Tech is required during most radiological accidents and during normal refueling operations. The fuel bundle may not be returned to the reactor and must be placed in an authorized location outside the reactor.

D – Incorrect – plausible, see B and C.

Technical Reference(s): ONI-J11-2 Rev. 17 & FTI-D09 Rev. 19	Reference Attached: ONI-J11-2 p. 7 & FTI-D09 p. 12

Proposed references to be provided to applicants during examination: FTI-D09 Map (attached to question)

Learning Objective (As available): OT-3035-14(LP)-A.1, OT-3602-10 and 13

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	
	55.43	(b)7

SRO Justification - Fuel handling facilities and procedures.

- Refuel floor SRO responsibilities.

QUESTION SRO 12

The plant was operating at rated power when a loss of off-site power occurred.

The crew is operating in EOP-01, RPV Control and EOP-02, Primary Containment Control, and ONI-R10, Loss of AC Power.

The following conditions exist:

• RPV level	60” and lowering
• RPV pressure	600 psig and lowering
• Drywell temperature	165 °F and rising 1 °F/min
• Drywell pressure	7.3 psig and rising 0.1 psig/min
• Containment temperature	166 °F and rising 1 °F/min
• Containment pressure	7.4 psig and rising 0.1 psig/min

Which of the following actions has the highest priority?

A. Reenter EOP-01, RPV Control, & Anticipate Emergency Depressurization

B. Direct EOP-SPI 3.1, Containment Spray Operations, and spray containment

C. Direct EOP-SPI 7.1, Preparation For Containment Venting, and vent containment

D. Transition to EOP-01-2, Emergency RPV Depressurization, and rapidly depressurize the RPV.

QUESTION SRO 12

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	295027	EA2.02
	Importance Rating		3.7

K&A: Ability to determine and/or interpret the following as they apply to HIGH CONTAINMENT TEMPERATURE: Containment pressure

High Containment Temperature (Mark III Containment Only)

Explanation: **Answer B –** Based on the conditions listed, containment spray is required when Containment pressure reaches 8 psig, which will be in ~6 minutes, IAW EOP-02.

A – Incorrect – Plausible since AED is an option for rising containment temperature. However, AED is not available as the MSIVs closed on the LOOP.

C – Incorrect – Plausible as Containment pressure is rising and this action is required if adequate core cooling is threatened or off-site release rate reduction is required. However, neither condition was given in the stem.

D – Incorrect – Plausible since ED is required if Containment pressure cannot be maintained < 15 psig or Containment temperature cannot be maintained < 185 °F. However, Containment Spray is to be attempted first.

Technical Reference(s): EOP-02 Chart Rev. G	Reference Attached: EOP-02 Chart

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3402-9C

Question Source:	Bank #	
	Modified Bank #	
	New	x

Question History: Previous 2 NRC Exams? No

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

10 CFR Part 55 Content:	55.41	
	55.43	b(5)

SRO Justification - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

- Knowledge of diagnostic steps and decision points in the emergency operating procedures (EOP) that involve transitions to event specific sub-procedures or emergency contingency procedures.

QUESTION SRO 13

The plant was operating at 50% power with CRD B pump tagged out.

Then a transient occurred, resulting in the following conditions:

- A manual scram and ARI were initiated
- NO Control Rod movement occurred
- White SCRAM SOL VLS Lights above the Manual Scram Pushbuttons on P680 are all off
- The Scram Discharge Volume vent and drain valves indicate open on P680
- Bus EH11 locked out

Based on these conditions, which of the following EOP Actions to insert control rods has the highest priority?

A. EOP-SPI 1.2, Scram And ARI

B. EOP-SPI 1.3, Manual Rod Insertion

C. EOP-SPI 1.4, Venting Scram Air Header

D. EOP-SPI 1.5, Venting CRD Overpiston Volumes

QUESTION SRO 13

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	295037	2.4.35
	Importance Rating		4.0

K&A: Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.

Scram Condition Present and Reactor Power Above APRM Downscale or Unknown

Explanation: **Answer C –** Venting the scram air header is the only option with the given conditions. An in-field operator would perform this task from the containment building.

A – Incorrect – Plausible since this action would insert control rods in the event of a hydraulic ATWS as indicated by some inward rod motion. However, the stem states no rod motion occurred which would indicate an electrical ATWS.

B – Incorrect – Plausible since this method will insert control rods if a CRD pump and RC&IS were available. However, with CRD B pump tagged out and a lockout on bus EH11, no CRD pumps are available and RC&IS will not allow rod movement based on channel disagreement.

D – Incorrect – Plausible since this would be a viable option if the scram air header was depressurized. However, this method would only insert 1 rod at a time and the scram air header is not depressurized..

Technical Reference(s): EOP-SPI 1.4 Rev. 4	Reference Attached: EOP-SPI 1.4 pp. 1-5

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3403-07(LP)-A

Question Source:	Bank # Modified Bank # New	 x
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43	 (b)5

SRO Justification - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps.

QUESTION SRO 14

The plant was at rated power when FCMS received a fire alarm on the Unit 1 Main Transformer.

The Fire Brigade was toned out and reports that the B phase Main Transformer oil cooler has ruptured and is burning.

Which of the following will be used to mitigate this causality?

A. ONI-C71-1, Reactor Scram and ONI-P54, Fire – Attachment 3, Protected Area Fire Actions

B. SOI-S11, Power Transformers and ONI-P54, Fire – Attachment 3, Protected Area Fire Actions

C. SOI-S11, Power Transformers and ONI-P54, Fire – Attachment 5, Large Area Accelerant Fed Fire Plan

D. ONI-C71-1, Reactor Scram and ONI-P54, Fire – Attachment 5, Large Area Accelerant Fed Fire Plan

QUESTION SRO 14

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	600000	AA2.13
	Importance Rating		3.8

K&A: Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: Need for emergency plant shutdown

Plant Fire On Site

Explanation: **Answer A –** A fire in the oil cooler of the Main Transformer would require the Control Room to trip the Main Turbine in order to de-energize the transformer. A trip of the main turbine from rated power would cause a Rx scram and ONI-C71-1 would be entered and contains actions to stabilize the plant. Also, ONI-P54 Att-3 contains actions for firefighting in this situation.

B – Incorrect – Part-1 - Plausible since SOI-S11 contains actions to de-energize the Main Transformers and shuts down the cooling fans. However, this section is performed after the turbine is tripped.

C – Incorrect – Part-1 - Plausible since SOI-S11 contains actions to de-energize the Main Transformers and shuts down the cooling fans. However, this section is performed after the turbine is tripped. Part-2 – Plausible since ONI-P54 Att 5 is used for accelerant fed fires. However the intent of this section is for airline crash or tanker truck fire.

D – Incorrect – Part-2 – Plausible since ONI-P54 Att 5 is used for accelerant fed fires. However the intent of this section is for airline crash or tanker truck fire.

Technical Reference(s): ONI-P54 Rev. 27 and ONI-C71-1 Rev. 23	Reference Attached: ONI-P54 pp. 15, 27, 30-31, 33-34, 36, & 40 and ONI-C71-1 p. 3

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3035-05(LP)-A-2 & OT-3035-02(LP)-A-2

Question Source:	Bank #	
	Modified Bank #	
	New	x

Question History: Previous 2 NRC Exams? No

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

10 CFR Part 55 Content:	55.41	
	55.43	b(5)

SRO Justification - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps.

QUESTION SRO 15

(1) Which of the following is an EOP entry condition?

AND

(2) What is the associated Technical Specification Bases for the limit of the associated EOP entry condition?

	(1)	(2)
A.	Suppression Pool average temperature of 100 °F	To ensure primary containment pressure does not exceed 30.0 psig
B.	RPV water level of 120”	To maintain level above the top of active fuel
C.	Drywell pressure 2 psig	To ensure drywell pressure does not exceed 45.5 psig
D.	RPV pressure of 1068 psig	Increasing RPV pressure compresses steam voids and results in inserting positive reactivity which could challenge the integrity of fuel cladding and the RCPB

QUESTION SRO 15

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		2
	K/A#	295009	2.2.4
	Importance Rating		4.7

K&A: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Low Reactor Water Level

Explanation: **Answer B –** EOP-01 requires entry when RPV level is below 130 inches; so 120 inches would be an entry condition. The associated Technical Specification bases for the limit of the associated EOP entry condition is to maintain level above the top of active fuel

A – Incorrect – plausible in that 100 °F suppression pool temperature is greater than the 95 °F EOP-02 entry condition. The T.S. Bases though says it is to maintain the Primary Containment pressure below 15 psig

C – Incorrect – plausible in that the T.S. bases description is incorrect, 45.5 psig is the containment yield limit. Drywell pressure for EOP-01 and EOP-02 entry is > 1.68 psig.

D – Incorrect - plausible in that the T.S. bases description is correct. RPV pressure for EOP entry is > 1083 psig, which might be remembered incorrectly. Plausible because 1065 psig is the high pressure Scram setpoint.

Technical Reference(s): TS Bases: 3.6.2.1 Rev. 1, 3.3.4.2 Rev. 9, 3.3.1.1.7 Rev. 3; and 3.3.1.1.4 Rev. 3, EOP-01 Rev. 9, & EOP-02 Bases Rev. 6	Reference Attached: TS Bases 3.6.2.1 p. B 3.6-70; 3.3.4.2 p. B 3.3-82, 3.3.1.1.7 p. B 3.3-15; and 3.3.1.1.4 p. B3.3-12, EOP-01 p. 8, EOP-02 Bases p. 8

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3402-02-B, OT-3402-05-B, OT-3037-07

Question Source:	Bank # Modified Bank # New	 Oyster Creek 2007 #SRO-09
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41 55.43	 (b)2

SRO Justification - Facility operating limitations in the TS and their bases.

- Knowledge of TS bases that are required to analyze TS required actions and terminology.

QUESTION SRO 16

Plant startup is in progress following a refuel outage.

Then the following occurred in short succession:

- While performing a surveillance, a jumper was landed incorrectly causing relay E12-K110BX5 to actuate. This resulted in an inadvertent ½ isolation of NCC to the Drywell causing 1P43-F400, NCC DW RETURN INBD DW ISOL valve to close
- This was immediately reported to the Unit Supervisor and the jumper was removed
- Annunciator DRYWELL RECORDER TEMPERATURE HIGH/LOW, H13-P800-03-F4 alarmed

DRYWELL COOLING TEMPERATURE Recorder 1M13-R110 Points #5 & #11, REACTOR SKIRT (SOUTHEAST & NORTHWEST) are reading 122 °F and rising slowly.

Prior to reopening of 1P43-F400, reset or bypass of the Division __(1)__ isolation signal must be performed.
In order to restore cooling to the Drywell, the Unit Supervisor would direct __(2)__ .

	(1)	(2)
A.	1	IOI-0018, Emergency Operating Procedure and Isolation Restoration
B.	1	EOP-SPI 2.1, Bypass of NCC Isolation
C.	2	IOI-0018, Emergency Operating Procedure and Isolation Restoration
D.	2	EOP-SPI 2.1, Bypass of NCC Isolation

QUESTION SRO 16

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		2
	K/A#	295020	AA2.02
	Importance Rating		3.4

K&A: Ability to determine and/or interpret the following as they apply to INADVERTENT CONTAINMENT ISOLATION: Drywell/containment temperature

Inadvertent Containment Isolation

Explanation: **Answer C –** P43-F400 closes on a Div 2 isolation signal. IOI-0011 is the procedure that is used to reset and clear isolation signals and restore systems so they can perform their intended function.

A – Incorrect – 1st part – Plausible if the isolation involved P43-F355 or P43-F410, which are outboard isolation valves.

B – Incorrect – 1st part – Plausible if the isolation involved P43-F355 or P43-F410, which are outboard isolation valves. 2nd part – Plausible since this is the correct procedure to use if operating in the EOPs. The EOP can be entered prior to reaching an entry setpoint if all normal/off-normal actions are to be completed first. However, loss of DW cooling at this point in a S/U would allow plenty of time to complete normal isolation recovery actions.

D – Incorrect – 2nd part – Plausible since this is the correct procedure to use if operating in the EOPs. The EOP can be entered prior to reaching an entry setpoint if all normal/off-normal actions are to be completed first. However, loss of DW cooling at this point in a S/U would allow plenty of time to complete normal isolation recovery actions.

Technical Reference(s): IOI-018 Rev. 17, Dwgs 208-055 Sh. 3 Rev. JJ, Sh. 8 Rev. EE, Sh. 102 Rev. N, & 208-174 Sh. 12 Rev. N	Reference Attached: IOI-018 pp. 3, 90-91 Dwgs 208-055 Sh. 3, Sh. 8, Sh. 102, & 208-174 Sh. 12

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3402-16-I.1

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	
	55.43	b(5)

SRO Justification - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps.

QUESTION SRO 17

The plant has experienced a severe accident from rated power.

The following conditions exist:

- Hydrogen has been detected in the Drywell and in Containment
- The Unit Supervisor has determined that operation of the Combustible Gas Mixing Compressors is required

Based on the given conditions:
Actions to startup the Combustible Gas Mixing System IAW SOI-M51/56 are performed from the __(1)__ .

And;

IAW Tech Spec Bases, the Combustible Gas Mixing System is credited with reducing __(2)__ .

	(1)	(2)
A.	Control Room only	Containment hydrogen concentration
B.	Control Room and Intermediate Building	post LOCA Dose
C.	Control Room only	post LOCA Dose
D.	Control Room and Intermediate Building	Containment hydrogen concentration

QUESTION SRO 17

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		2
	K/A#	500000	2.1.30
	Importance Rating		4.0

K&A: Ability to locate and operate components, including local controls.

High Containment Hydrogen Concentration

Explanation: **Answer B –** The combustible gas mixing system is operated from both the control room and from panels along the west wall of the Intermediate Building – Elevation 620' and Tech Spec Bases, B 3.6.3.3 States, that post-LOCA dose is reduced with mixing compressor operation.

A – Incorrect – 1st part – Plausible if the applicant believes that the Combustible Gas Mixing System can only be operated from the control room. 2nd part – The Combustible Gas Mixing System takes suction from containment and discharges into the Drywell and are designed to ensure a uniformly mixed post-accident containment atmosphere. Plausible if the applicant believes this will reduce the H2 concentration in the Containment

C – Incorrect – see A and B above

D – Incorrect – see A above

Technical Reference(s): SOI-M51/M56 Rev. 29, TS Bases 3.6.3.3, and SDM-M51 Rev. 2	Reference Attached: SOI-M51/M56 pp.17 & 20, and SDM-M51 p. 16

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-Combined-M51/56-1.7

Question Source:	Bank #	
	Modified Bank #	Monticello 2013 # RO-27
	New	

Question History: Previous 2 NRC Exams? No

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

10 CFR Part 55 Content:	55.41	
	55.43	b(2)

SRO Justification - Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]

- Knowledge of TS bases that are required to analyze TS required actions and terminology.

QUESTION SRO 18

While operating at rated power, NCC was lost to the containment.

The crew inserted a manual Rx scram.

The following conditions now exist:

• All control rods inserted	
• Rx Pressure	850 psig – rising slowly
• Motor Feed Pump	tripped on overcurrent
• RPV level	120 inches – rising slowly
• RCIC injecting	700 gpm
• Drywell pressure	1.7 psig – rising slowly
• Drywell temperature	180 °F – rising slowly

Later the following occurred:

- RCIC tripped
- HPCS is manually started to maintain RPV level 150 to 219 inches

This event meets the conditions for ____ notifications to the NRC Operations Center.

Reference Provided:

A. 1 hour and 4 hour <u>only</u>

B. 1 hour and 8 hour <u>only</u>

C. 4 hour and 8 hour <u>only</u>

D. 1 hour, 4 hour, and 8 hour

QUESTION SRO 18

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A#	209002	2.4.30
	Importance Rating		4.1

K&A: Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.

High Pressure Core Spray

Explanation: **Answer C –** ECCS injection and RPS actuation require a 4-hour report and failure of HPCS to auto start is considered a failure of equipment that could prevent the fulfillment of a safety function (decay heat removal). HPCS is considered a single train system.

A – Incorrect – A 1-hour notification is not required. Plausible since DW pressure >1.68 psig is an E-plan entry if the pressure is due to RCS leakage which would require a 1-hour notification. In this case, the increase in DW pressure was due to the loss of DW cooling.

B – Incorrect – A 1-hour notification is not required. Plausible since DW pressure >1.68 psig is an E-plan entry if the pressure is due to RCS leakage which would require a 1-hour notification. In this case, the increase in DW pressure was due to the loss of DW cooling.

D – Incorrect – A 1-hour notification is not required. Plausible since DW pressure >1.68 psig is an E-plan entry if the pressure is due to RCS leakage which would require a 1-hour notification. In this case, the increase in DW pressure was due to the loss of DW cooling.

Technical Reference(s): NOP-OP-1015 Rev. 7	Reference Attached: NOP-OP-1015 pp. 33, 34, & 37

Proposed references to be provided to applicants during examination: NOP-OP-1015

Learning Objective (As available): OT-3039-01-A

Question Source:	Bank #	
	Modified Bank #	Perry 2013 # SRO-13
	New	

Question History:	Previous 2 NRC Exams?	No

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

10 CFR Part 55 Content:	55.41	
	55.43	b(5)

SRO Justification - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

- SRO is responsible for making NRC notifications. (SRO specific task 341-013-01-03 Report Safety Limit Violations and Reportable Occurrences)

QUESTION SRO 19

The plant is operating at rated power with the following conditions:

- Containment pressure is 0.1 psig
- Containment temperature is 74 °F.

Then an NLO reports that the MCC disconnect to the Standby Liquid Control Storage Tank Operating Heater, C41-D002 was found in the OFF position.

Based on this information, both SLC A and SLC B subsystems are __(1)__ .
The procedure used to restore the MCC disconnect to the ON position is __(2)__ .

	(1)	(2)
A.	INOPERABLE	ARI for SLC A/B OUT OF SERVICE
B.	INOPERABLE	ELI-R24, 480 VOLT MCC
C.	OPERABLE	ARI for SLC A/B OUT OF SERVICE
D.	OPERABLE	ELI-R24, 480 VOLT MCC

QUESTION SRO 19

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A#	211000	A2.05
	Importance Rating		3.4

K&A: 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: Loss of SBLC tank heaters.

Standby Liquid Control

Explanation: **Answer D –** Both SLC subsystems remain operable with the heaters off as long as containment temperature remains ≥70 °F (requires knowledge of surveillance requirements). The lineup ELI-R24 is the correct procedure for restoring the MCC for the heaters to the ON position.

A & B – Incorrect – 1st part – plausible because the applicant may believe that SLC is inoperable with no electrical supply to the SLC tank operating heaters. SLC remains operable since the given containment temperature is 74°F.
SR 3.1.7.2 only looks at tank temperature.

A & C – Incorrect – 2nd part – The ARI is plausible since loss of power to other SLC components are annunciated by alarms. The ARI directs the operator to check MCC disconnects for loss of power to pumps and valves, but loss of power to SLC tank heater is not an input to “SLC Out of Service” alarm.

Technical Reference(s)	Reference Attached
Technical Reference(s): TS 3.1.7 Rev. Amend. 171, SOI-C41 Rev 22, ELI-R24 Rev 36, PAP-0205 Rev 24, ARI-H13-P601-19 Rev. 21, & ARI-H13-P601-18 Rev. 17	Reference Attached: TS 3.1.7 pp. 3.1-20-21, SOI-C41 p. 5, ELI-R24 p. 47, PAP-0205 pp. 4, 15-16, ARI-H13-P601-19 p 64, & ARI-H13-P601-18 p 14

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-C41-K & -M

Question Source:	Bank #	Perry 2015 # SRO-18
	Modified Bank #	
	New	
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	
	55.43	b(2)

SRO Justification - Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]

- Application of Required Actions (Section 3) and Surveillance Requirements (SR) (Section 4) in accordance with rules of application requirements (Section 1).

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]

- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps.

QUESTION SRO 20

The plant was operating at rated power with HPCS tagged out for maintenance when Digital Feedwater Control was lost.

The following sequence then occurred:

- A manual scram was successfully inserted at 160” RPV level and RCIC was manually initiated.
- One minute later RCIC tripped.
- RPV level continued to lower
- 1G33-F001 and 1G33-F004 failed to isolate on a high Δ Flow isolation signal.
- 1G33-F004 was closed from P881
- MSIVs automatically isolated
- While controlling RPV pressure on SRVs, one SRV stuck partially open
- RPV level lowered to -15 inches when LPCS injection commenced

What is the highest Emergency Plan Classification required for this event?

Reference Provided: (only page 2 of 3)

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

QUESTION SRO 20

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A#	259002	2.4.41
	Importance Rating		4.6

K&A: Knowledge of the emergency action level thresholds and classifications.

Reactor Water Level Control

Explanation: **Answer A –**RPS failed to automatically scram the Rx at L3 but the subsequent manual initiation at 150 inches was successful. This results in a UE based on SU6.1. Lowering of RPV level to <TAF does not itself constitute an E-Plan entry.

B – Incorrect – Plausible if considers a failure to isolate RWCU resulted in an unisolable leak in RWCU (FPB Loss). However, manual isolation of G33-F004 was successful.

C – Incorrect – Plausible if considers RPV level < TAF not recoverable. (Potential FC LOSS & RCS LOSS). However, LPCS will restore level since no RCS leak is indicated and LPCS capacity exceeds makeup needed for a stuck open SRV.

D – Incorrect – Plausible if considers a failure to isolate RWCU resulted in an unisolable leak in RWCU (FPB Loss) and considers RPV level < TAF not recoverable. (Potential FC LOSS & RCS LOSS). However, LPCS will restore level since no RCS leak is indicated and LPCS capacity exceeds makeup needed for a stuck open SRV. Additionally, manual isolation of G33-F004 was successful.

Technical Reference(s): EAL Matrix Rev. 1/3/17 and PSI-19 Rev. 21	Reference Attached: EAL Matrix p. 2 and PSI-19 pp. 185-186

Proposed references to be provided to applicants during examination: EAL Matrix (only page 2)

Learning Objective (As available): EPL-0804-01-4

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	
	55.43	b(6)

SRO Justification - Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity.

- Evaluating core conditions and emergency classifications based on core conditions.

QUESTION SRO 21

The plant is operating at RTP, with the following conditions:

- Division 1 Standby Diesel Generator carrying Bus EH11
- CRD ‘B’ pump is tagged out
- NCC ‘C’ pump is tagged out

Then ESW ‘A’ LOW FLOW alarms are received and the ESW ‘A’ pump ammeter is pegged high.

Which of the following procedures contain actions to mitigate the consequences of this condition?

1. ONI-R22-1, Loss of an Essential And/Or A Stub 4.16KV Bus
2. SOI-R43, Division 1 and 2 Diesel Generator Systems
3. ONI-C71-2, Loss of One RPS Bus
4. SOI-P45/49, Emergency Service Water and Screen Wash Systems

A. 1 <u>only</u>

B. 1 <u>and</u> 2

C. 3 <u>and</u> 4 <u>only</u>

D. 2, 3, <u>and</u> 4

QUESTION SRO 21

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A#	262001	A2.10
	Importance Rating		3.4

K&A: 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: Exceeding current limitations.

AC Electrical Distribution

Explanation: **Answer B –** With the loss of ESW ‘A’ and no LOCA signal present the operator is required to perform an emergency shutdown of the diesel. This will result in the loss of Bus EH11. SOI-R43 contains the actions to perform an emergency shutdown and ONI-R22-1 contains actions for the loss of Bus EH11.

A – Incorrect – Plausible if the applicant believes that an emergency shutdown of the diesel generator is not required.

C – Incorrect – An RPS bus will not be lost, but an applicant may believe that since the RPS MGs are in the Divisional Switchgear rooms then ONI-C71-2 should also be entered. SOI-P45/49 would eventually be entered due to the loss of ESW ‘A’ pump and eventual system shutdown.

D – Incorrect – See C above. Plausible because SOI-R43 is correct, but ONI-C71-2 still does not apply.

Technical Reference(s): ONI-R22-1 Rev. 15, SOI-R43 Rev. 49, SDM C71 Rev. 13, PYBP- POS-0027 Rev 6	Reference Attached: ONI-R22-1 p. 5, SOI-R43 pp. 10 & 44, SDM C71 p. 68, PYBP- POS-0027 pp. 3 and 4

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3035-07(LP)-A.2

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	
	55.43	b(5)

SRO Justification - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps.

QUESTION SRO 22

The plant shutdown two days ago for refueling.

The following conditions exist:

- RPV head detensioning was just completed
- Reactor Coolant temperature is 90 °F
- RPV level is at the RPV flange
- Rx Recirculation Pumps are OFF
- RHR A loop is operating in Shutdown Cooling (SDC) Mode
- RHR B loop is in standby

Then the following occurs:

- ESW A Pump trips on overcurrent
- 1E12-F003B. RHR HX'S OUTLET VALVE has lost power

Based on this information, Rx water level __(1)__ and a minimum of __(2)__ alternate method(s) of decay heat removal must be verified.

	(1)	(2)
A.	must be raised	1
B.	must be raised	2
C.	may be maintained at current level	1
D.	may be maintained at current level	2

QUESTION SRO 22

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A#	400000	2.4.9
	Importance Rating		4.2

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

Component Cooling Water

Explanation: **Answer C –** Raising RPV level is required per ONI-E12-2 if no recirc pumps or RHR pumps are running and RPV level is <250”. With RPV water level at the flange (~360”), natural circulation will occur. TS 3.9.9 requires 2 RHR systems to be Operable and 1 in operation. Since RHR A requires ESW A running for cooling, it is considered Inoperable. However, TS 3.9.9 Bases allows RHR B Loop to be considered Operable since it can be manually aligned for SDC. Therefore only 1 alternate method must be verified.

A – Incorrect – 1st part - Plausible since raising RPV level is required per ONI-E12-2 if no recirc pumps are running and RPV level is <250”.

C – Incorrect – 1st part - Plausible since raising RPV level is required per ONI-E12-2 if no recirc pumps are running and RPV level is <250”. 2nd part - Plausible if the operator fails to recall that per TS Bases, SDC B loop is Operable if it can be manually aligned. Therefore, only 1 alternate method must be verified.

D – Incorrect – 2nd part - Plausible if the operator fails to recall that per TS Bases, SDC B loop is Operable if it can be manually aligned. Therefore, only 1 alternate method must be verified.

Technical Reference(s): TS 3.9.9 Rev. Amend 69 and TS 3.9.9 Bases Revs. 7 & 1, and ONI-E12-2 Rev. 40	Reference Attached: TS 3.9.9 p. 3.9-13 and TS 3.9.9 Bases pp. B 3.9-30 & 31, and ONI-E12-2 p. 11

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3037-13 and OT-3035-11(LP)-A.2

Question Source:	Bank #	
	Modified Bank #	Susquehanna 2015 # SRO-79
	New	

Question History:	Previous 2 NRC Exams?	No

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

10 CFR Part 55 Content:	55.41	x
	55.43	b(2)

Comments: SRO Justification - Facility operating limitations in the TS and their bases.

- Knowledge of TS bases that are required to analyze TS required actions and terminology.

QUESTION SRO 23

The plant was operating at rated power when a transient occurred.

RPV water level indicated -5” on validated SPDS prior to low pressure ECCS restoring RPV water level.

Notification to the __(1)__ is required.

And, IAW Technical Specifications __(2)__ permission to commence a reactor startup is required.

	(1)	(2)
A.	NRC Operations Center within 4 hours	NRC
B.	NRC Operations Center within 4 hours	Vice President - Nuclear
C.	Vice President - Nuclear within 24 hours	NRC
D.	Vice President - Nuclear within 24 hours	Vice President - Nuclear

QUESTION SRO 23

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		2
	K/A#	216000	2.2.25
	Importance Rating		4.2

K&A: Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

Nuclear Boiler Instrumentation

Explanation: **Answer A –** The stem stated that RPV level dropped less than the top of active fuel (0” indicated). Therefore, the RPV Level Safety Limit was violated. NOP-OP-1015 requires NRC Operations Center notification within 4 hours of a safety limit violation. Additionally, TS 2.2 Bases directs operation not to be resumed until authorized by the NRC.

B – Incorrect – (1) is correct, (2) is plausible since IOI-1 requires management permission to commence a startup.

C – Incorrect – (1) plausible since NOP-OP-1002 requires management notification. (2) is correct.

D – Incorrect – see B(2) and C(1) above

Technical Reference(s): TS 2.1.1.3 Rev. Amend 188, TS 2.2 Bases Rev. 7, NOP-OP-1015 Rev. 8	Reference Attached: TS 2.1.1.3 p. 2.0-1, TS 2.2 Bases p. B 2.0-5, , NOP-OP-1015 p. 33

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3037-03-1, OT-3037-03-4

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	
	55.43	(b)2

SRO Justification - Facility operating limitations in the TS and their bases.

- Knowledge of TS bases that are required to analyze TS required actions and terminology.

QUESTION SRO 24

The plant is operating at 97 % rated power with the Steam Bypass and Pressure Regulating System operating in Single Channel Mode on channel B.

If Pressure Regulator Channel B starts to slowly fail low, #4 Turbine Control Valve will __(1)__ more. The Unit Supervisor would implement __(2)__ to mitigate this situation.

	(1)	(2)
A.	open	ONI-C51, Unplanned Change In Reactor Power Or Reactivity
B.	open	ONI-C71-1, Reactor Scram
C.	close	ONI-C51, Unplanned Change In Reactor Power Or Reactivity
D.	close	ONI-C71, Reactor Scram

QUESTION SRO 24

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		2
	K/A#	241000	A2.16
	Importance Rating		3.4

K&A: 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: Low turbine inlet pressure (loss of pressure signal)

Reactor/Turbine Pressure Regulating

Explanation: **Answer D** – The pressure regulating system in Single Channel Mode on channel B means that the "A" channel is out of service so no swap will occur upon failure of the "B" channel. At 97% power, the #4 TCV is ~40% open. So, if the pressure regulator fails low, the TCV will close more. When TCV #4 is fully closed, TCVs 1, 2, & 3 will start moving in the close direction. This will cause Rx pressure to increase which causes Rx power to also increase. If left unattended, the Rx would scram on either high pressure or high flux. Because the recovery from failure of both pressure regulators is not recoverable, ONI-C81, Pressure Regulator failure directs a Rx scram and entry into ONI-C71-1.

A – Incorrect – 1st part – This would be true if the PR was failing high. 2nd part – This ONI could be correct if the power increase was "unexplained". However, since it was given that the PR failed, the power increase would be due to the PR failure and not 'unexplained".

B – Incorrect – 1st part – This would be true if the PR was failing high.

C – Incorrect – . 2nd part – This ONI could be correct if the power increase was "unexplained". However, since it was given that the PR failed, the power increase would be due to the PR failure and not 'unexplained".

Technical Reference(s): ONI-C85 Rev. 1 and ONI-C71-1 Rev. 23	Reference Attached: ONI-C85 pp. 3-6 & 9 and ONI-C71-1 pp. 3 & 5-7

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): x

Question Source:	Bank #	
	Modified Bank #	
	New	x

Question History:	Previous 2 NRC Exams?	No

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

10 CFR Part 55 Content:	55.41	
	55.43	(b)5

SRO Justification - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations

- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps.

QUESTION SRO 25

The plant is operating at rated power when the following alarms are received:

- HTR 4 ISOL HOT SRG TK LEVEL HI, H13-P680-02-E1
- HOT SURGE TANK LEVEL HI, H13-P680-02-E2
- RFPT A HYD OIL PRESS LO, H13-P680-03-B3
- HP CNDR LEVEL HI/LO, H13-P680-02-B3

Based on the above annunciators, which procedure contains the highest priority action(s)?

A. ONI-C51, UNPLANNED CHANGE IN REACTOR POWER OR REACTIVITY

B. ONI-C34, FEEDWATER FLOW MALFUNCTION

C. ONI-N32, MAIN TURBINE TRIP

D. ONI-N36, LOSS OF FEEDWATER HEATING

QUESTION SRO 25

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		2
	K/A#	256000	2.4.45
	Importance Rating		4.1

K&A: Ability to prioritize and interpret the significance of each annunciator or alarm.

Condensate

Explanation: **Answer D –** With the plant at rated power, HTR 4 ISOL HOT SRG TK LEVEL HI will result in isolation of #4 FW Heater. When this heater isolates, FW temperature lowers and Rx power will increase. ONI-N36 directs the operator to lower power to ≤96% to prevent exceeding the licensed power limit.

A – Incorrect – Plausible since a RFPT will trip on low bearing oil pressure of 4 psig. However, RFPT A HYD OIL PRESS LO alarms on bearing header pressure of 150 psig.

B – Incorrect – ONI-C51 would be entered after Rx power is lowered IAW ONI-N36 and is directed by ONI-N36 Supplemental Actions. Additionally, ONI-N36 requires power reduction to ≤96% while ONI-C51 only requires the operator to maintain Power ≤ initial power level.

C – Incorrect – Plausible since HP CNDR LEVEL HI/LO directs the operator to consider main turbine shutdown.

Technical Reference(s): ARI-H13-P680-02 Rev. 12, ARI-H13-P680-03 Rev. 19, ONI-N36 Rev. 19, & ONI-C51 Chart Rev. N	Reference Attached: ARI-H13-P680-02 pp. 25, 49, 53, ARI-H13-P680-03 p. 31-23, ONI-N36 pp. 3 & 5, & ONI-C51 Chart (partial)

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3035-08(LP)-A.1

Question Source:	Bank #	
	Modified Bank #	Perry 2004 # SRO-92
	New	
Question History:	Previous 2 NRC Exams?	No
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	
	55.43	(b)5

SRO Justification - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations

- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps.

2021 NRC Exam Answer Key

Reactor Operator							SRO	
1	B	26	D	51	A		1	B
2	C	27	C	52	A		2	A
3	A	28	A	53	D		3	B
4	B	29	B	54	D		4	C
5	B	30	C	55	C		5	D
6	A	31	C	56	D		6	C
7	D	32	C	57	A		7	D
8	D	33	D	58	D		8	C
9	B	34	B	59	A		9	B
10	C	35	C	60	B		10	C
11	A	36	D	61	A		11	B
12	C	37	B	62	D		12	B
13	A	38	D	63	D		13	C
14	A	39	A	64	A		14	A
15	A	40	B	65	A		15	B
16	C	41	B	66	A		16	C
17	D	42	B	67	D		17	B
18	C	43	D	68	B		18	C
19	C	44	A	69	C		19	D
20	B	45	B	70	C		20	A
21	C	46	C	71	B		21	B
22	C	47	C	72	A		22	C
23	D	48	B	73	D		23	A
24	D	49	B	74	A		24	D
25	D	50	B	75	C		25	D

Perry 2021

Written Examination

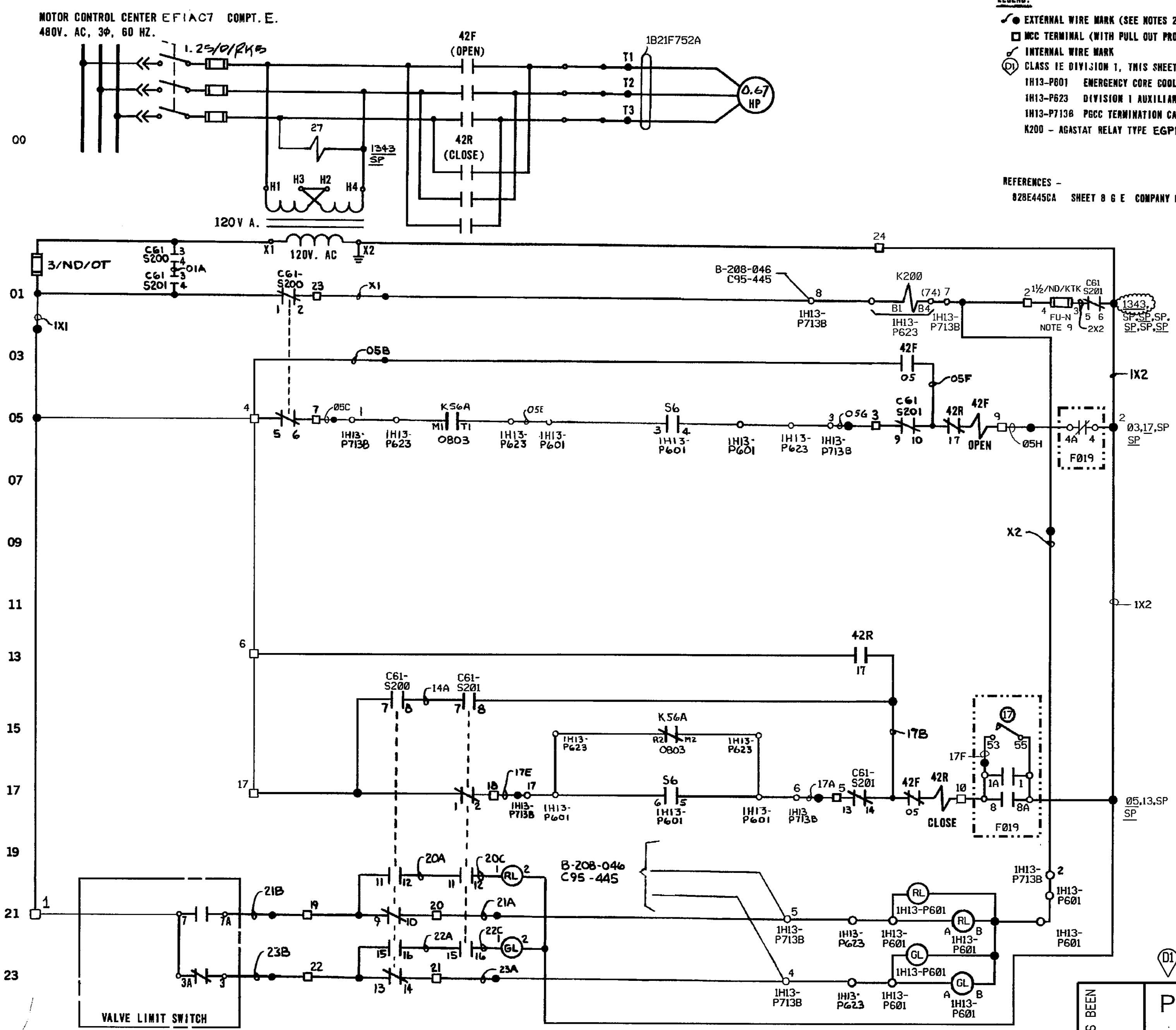
Reference Materials Provided

RO

References

RO QUESTION #5 Reference

MOTOR CONTROL CENTER EF1AC7 COMPT. E.
480V. AC, 3Φ, 60 HZ.



LEGEND.

- EXTERNAL WIRE MARK (SEE NOTES 2 & 3)
- MCC TERMINAL (WITH PULL OUT PROVISION)
- INTERNAL WIRE MARK
- (D1) CLASS IE DIVISION 1, THIS SHEET
- 1H13-P601 EMERGENCY CORE COOLING B B
- 1H13-P623 DIVISION I AUXILIARY RELAY PANEL
- 1H13-P713B PGCC TERMINATION CABINET TM014
- K200 - AGASTAT RELAY TYPE EGP1 *

REFERENCES -
828E445CA SHEET 8 G E COMPANY DRAWING

NOTES:

1. ALL DEVICES WITHOUT LOCATION MARKS ARE LOCATED IN M.C.C.
2. ALPHABETICAL WIRE MARKS ARE PREFIXED BY: SUB-SYSTEM NO., SHEET NO., & LINE NO. (EXAMPLE B21H1705A)
3. POTENTIAL WIRE MARKS ARE PREFIXED BY: SUB-SYSTEM NO. & SHEET NO. (EXAMPLE B21H17X1)
4. DEVICE DESIGNATIONS ARE PREFIXED BY: UNIT AND SYSTEM NO. (EXAMPLE 1B21-F019) EXCEPT AS NOTED, SEE NOTE 7
5. VALVE SHOWN IN FULL CLOSED POSITION.
6. FOR GRAPHIC STANDARDS REFER TO DRAWING B-208-001.
7. THE FOLLOWING DEVICES ARE PREFIXED BY 1B21h
   S6 - SWITCH
8. SEE B-208-039 SH. A05 FOR C61-S200, S201 SWITCH DEVELOPMENTS
9. FU-N: CONTROL ROOM ISOLATION FUSE BUSSMANN KTK-R AND GOULD - SHAWMUT ATM OR ATMR FUSES MAY ALSO BE USED HERE.

* FOR VENDOR CONFIGURATION SEE DWG B-208-005.

TYPE							
		LIMIT SW CONTACT DEVELOPMENT					
ROTOR	CONT	FULL OPEN	A	B	FULL CLOSED	CONTACT LOCATION	
1	1					16	
	2					SPARE	
	3					23	
	4					05	
2	5					SPARE	
	6					SPARE	
	7					21	
	8					17	
3	9					SPARE	N/A
	10					SPARE	N/A
	11					SPARE	N/A
	12					SPARE	N/A
4	13					SPARE	N/A
	14					SPARE	N/A
	15					SPARE	N/A
	16					SPARE	N/A

(17) CLOSING TORQUE SWITCH INTERRUPTS CONTROL CIRCUIT IF MECHANICAL OVERLOAD OCCURS DURING CLOSING CYCLE OR FULLY CLOSED VALVE.

(16) OPENING TORQUE SWITCH INTERRUPTS CONTROL CIRCUIT IF MECHANICAL OVERLOAD OCCURS DURING OPENING CYCLE OR FULLY OPENED VALVE. (SPARE)

CLOSED CONTACT
OPEN CONTACT

ROTORS 3 AND 4 CAN BE SET AT VALVE POSITION FULL OPEN, FULL CLOSED OR ANY POSITION IN BETWEEN AS INDICATED BY POINTS A AND B.
REF: PHILA. GEAR CORP. DWG. 15-477-2759-3 REV. A

S6/1H13-P601		CONTROL SWITCH			
(CAM 2) CONTACTS		NAMEPLATE (F.V.)			CONTACT LOCATION
OPERATOR		CLOSE	AUTO	OPEN	
TF 1	2	X			SPARE
TR 3	4			X	05
BF 5	6	X			17
BR 7	8			X	SPARE
OPER & BLOCK CR2940-UN203A		X - CLOSED CONTACT			
SPRING RETURN TO AUTO					

RECEIVED
FEB 2 3 2005

(D1) NUCLEAR SAFETY RELATED

THIS AS-BUILT DRAWING HAS BEEN REVISED TO INCORPORATE: MISCELLANEOUS AS-BUILT (MAB 05-021)

PERRY NUCLEAR POWER PLANT
10 CENTER RD., PERRY, OHIO 44081

ELECTRICAL			ELEMENTARY DIAGRAM	
NUCLEAR			SYSTEM	
MOTOR OPERATED VLV				
MADE	CHECKED	ENGINEER APPROVALS		DATE
LJD	[signature]	—	[signature]	2-27-05
SCALE NONE	B	208-00 -000		X
SYSTEM B21	SIZE	SYSTEM CRITICAL DRAWING		REV

SRO

References

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Jet Pumps

LCO 3.4.3 All jet pumps shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more jet pumps inoperable.	A.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	-------------------NOTES------------------- 1. Not required to be performed until 4 hours after associated recirculation loop is in operation. 2. Not required to be performed until 24 hours after > 25% RTP. --- Verify at least two of the following criteria (a, b, and c) are satisfied for each operating recirculation loop: a. Recirculation loop drive flow versus flow control valve position differs by ≤ 10% from established patterns.	In accordance with the Surveillance Frequency Control Program (continued)

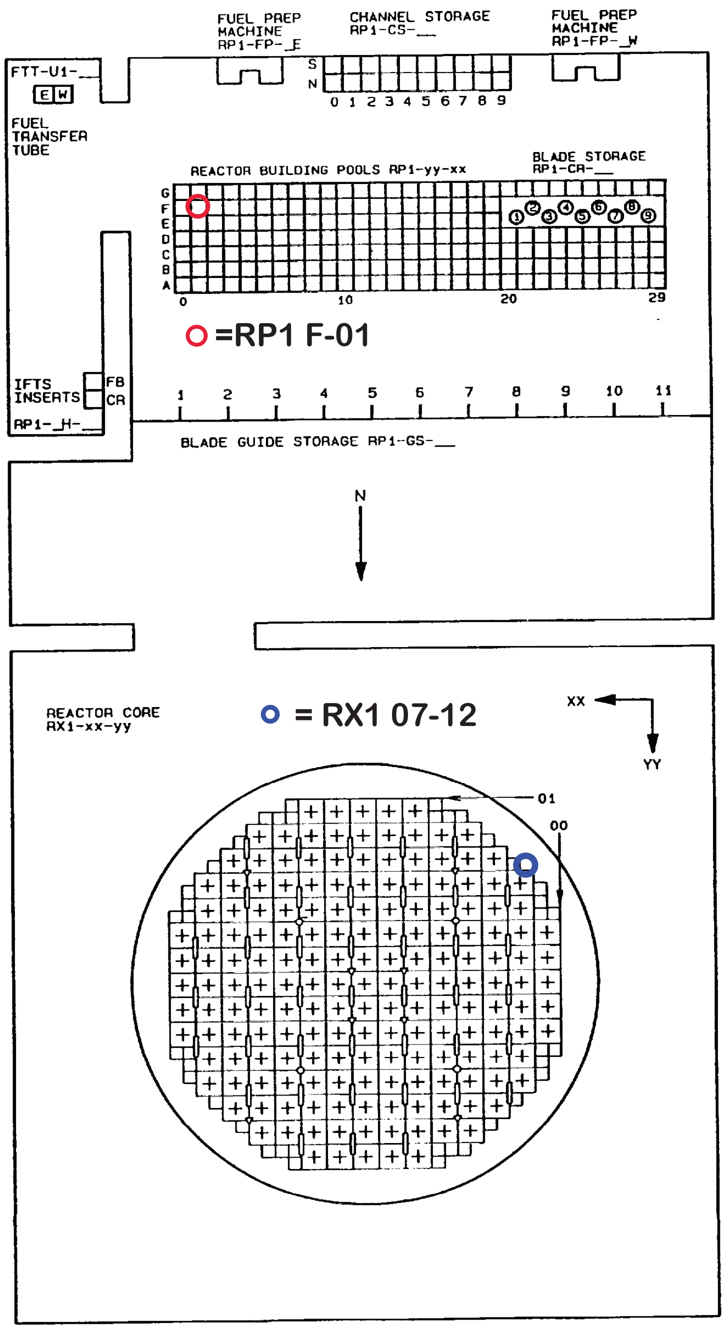
SRO QUESTION #8 Reference

Surveillance Requirement	Frequency	SVI	Notes	STRIDE
Deleted				
3.4.3.1	24 Hours	TSR		N/A
		SVI-B33T1160		N/A
		SVI-F41T3008		N/A
Deleted				

SRO QUESTION #11 Reference

ATTACHMENT 3: MAP OF STORAGE LOCATIONS

Page 2 of 5



CADD No. - PXFTI-2

SRO QUESTION #18 Reference

NUCLEAR OPERATING PROCEDURE	Procedure Number: NOP-OP-1015	
Title: Event Notifications	Use Category: General Skill Reference	
	Revision: 7	Page: 1 of 46

EVENT NOTIFICATIONS

Effective Date: 9-4-18

Approved: Scott Plymale / 8-23-18

Program Manager Date

SRO QUESTION #20 Reference

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

R — Abnormal Rad Levels / Rad Effluent

1 Rad Effluent

GENERAL EMERGENCY — Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem child thyroid CDE

RG1.1 [1 | 2 | 3 | 4 | 5 | DEF]

Reading on **any** Table R-1 effluent radiation monitor > column "GE" for ≥ 15 min. (Notes 1, 2, 3, 4)

RG1.2 [1 | 2 | 3 | 4 | 5 | DEF]

Dose assessment using actual meteorology indicates doses > 1,000 mrem TEDE or > 5000 mrem child thyroid CDE at or beyond the SITE BOUNDARY (Note 4)

RG1.3 [1 | 2 | 3 | 4 | 5 | DEF]

Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:
- Closed window dose rates > 1,000 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate child thyroid CDE > 5,000 mrem for 60 min. of inhalation

(Notes 1, 2)

SITE AREA EMERGENCY — Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem child thyroid CDE

RS1.1 [1 | 2 | 3 | 4 | 5 | DEF]

Reading on **any** Table R-1 effluent radiation monitor > column "SAE" for ≥ 15 min. (Notes 1, 2, 3, 4)

RS1.2 [1 | 2 | 3 | 4 | 5 | DEF]

Dose assessment using actual meteorology indicates doses > 100 mrem TEDE or > 500 mrem child thyroid CDE at or beyond the SITE BOUNDARY (Note 4)

RS1.3 [1 | 2 | 3 | 4 | 5 | DEF]

Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:
- Closed window dose rates > 100 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate child thyroid CDE > 500 mrem for 60 min. of inhalation

(Notes 1, 2)

ALERT — Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem child thyroid CDE

RA1.1 [1 | 2 | 3 | 4 | 5 | DEF]

Reading on **any** Table R-1 effluent radiation monitor > column "ALERT" for ≥ 15 min. (Notes 1, 2, 3, 4)

RA1.2 [1 | 2 | 3 | 4 | 5 | DEF]

Dose assessment using actual meteorology indicates doses > 10 mrem TEDE or > 50 mrem child thyroid CDE at or beyond the SITE BOUNDARY (Note 4)

RA1.3 [1 | 2 | 3 | 4 | 5 | DEF]

Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses > 10 mrem TEDE or > 50 mrem child thyroid CDE at or beyond the SITE BOUNDARY for 60 min. of exposure (Notes 1, 2)

RA1.4 [1 | 2 | 3 | 4 | 5 | DEF]

Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:
- Closed window dose rates > 10 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate child thyroid CDE > 50 mrem for 60 min. of inhalation

(Notes 1, 2)

UNUSUAL EVENT — Release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer

RU1.1 [1 | 2 | 3 | 4 | 5 | DEF]

Reading on **any** Table R-1 effluent radiation monitor > column "UE" for ≥ 60 min. (Notes 1, 2, 3)

RU1.2 [1 | 2 | 3 | 4 | 5 | DEF]

Sample analyses for a gaseous or liquid release indicates a concentration or release rate > 2 x ODCM limits for ≥ 60 min. (Notes 1, 2)

2 Irradiated Fuel Event

GENERAL EMERGENCY — Spent fuel pool level cannot be restored to at least the top of the spent fuel racks for 60 minutes or longer

RG2.1 [1 | 2 | 3 | 4 | 5 | DEF]

Spent fuel pool level cannot be restored to at least 3.5 ft for ≥ 60 min. (Note 1)

SITE AREA EMERGENCY — Spent fuel pool level at the top of the fuel racks

RS2.1 [1 | 2 | 3 | 4 | 5 | DEF]

Lowering of spent fuel pool level to 3.5 ft

ALERT — Significant lowering of water level above, or damage to, irradiated fuel

RA2.1 [1 | 2 | 3 | 4 | 5 | DEF]

Uncovery of irradiated fuel in the REFUELING PATHWAY

RA2.2 [1 | 2 | 3 | 4 | 5 | DEF]

Damage to irradiated fuel resulting in a release of radioactivity

AND

Any of the following radiation monitor indications:
- SPENT FUEL POOL D21-K332 (high alarm)
- UPPER POOL AREA 1D21-K083 (high alarm)
- FUEL PREP POOL D21-K322 (high alarm)
- FHB VENT EXH GAS D17-K716 (high alarm)
- CNTMT ATMOS GAS 1D17-K686 (high alarm)

RA2.3 [1 | 2 | 3 | 4 | 5 | DEF]

Lowering of spent fuel pool level to 23.5 ft

UNUSUAL EVENT — UNPLANNED loss of water level above irradiated fuel

RU2.1 [1 | 2 | 3 | 4 | 5 | DEF]

UNPLANNED water level drop above irradiated fuel in the REFUELING PATHWAY as indicated by **EITHER** of the following:
- Fuel Pool Water low level alarm
- FPCC Surge Tank low level alarm

AND

UNPLANNED rise in area radiation levels as indicated by **any** of the following radiation monitors:
- SPENT FUEL POOL D21-K332
- UPPER POOL AREA 1D21-K083
- FUEL PREP POOL D21-K322

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Unit 1 Plant Vent	1D17-K786 1D19-K300	---- 1.3E+00 µCi/cc	---- 1.3E-01 µCi/cc	---- 1.3E-02 µCi/cc	2 x High alarm ----
Gaseous	OG Vent Pipe	1D17-K836 1D19-K400	---- 4.7E+00 µCi/cc	---- 4.7E-01 µCi/cc	---- 4.7E-02 µCi/cc	2 x High alarm ----
Gaseous	TB/HB Vent	1D17-K856	7.7E+04 cpm	7.7E+03 cpm	7.7E+02 cpm	2 x High alarm
Gaseous	Unit 2 Plant Vent	2D17-K786 2D19-K300	---- 3.0E+00 µCi/cc	---- 3.0E-01 µCi/cc	---- 3.0E-02 µCi/cc	2 x High alarm ----
Liquid	Emergency Service Water Loop A	D17-K604	----	----	----	High alarm
Liquid	Emergency Service Water Loop B	D17-K605	----	----	----	High alarm

3 Area Radiation Levels

ALERT — Radiation levels that IMPEDE access to equipment necessary for normal plant operations, cooldown or shutdown

RA3.1 [1 | 2 | 3 | 4 | 5 | DEF]

Dose rate > 15 mR/hr in **EITHER** of the following areas:
- Control Room
- CAS (by survey)

RA3.2 [3 | 4 | 5]

An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to **any** Table R-2 rooms or areas (Note 5)

Table R-2 Safe Shutdown Rooms/Areas

Room/Area	Modes
AX 574' Elev. RHR B	3, 4, 5
AX 620' Elev. West Hallway	3, 4, 5
CC 620' Elev. Div. 1 AC	3, 4, 5
CC 620' Elev. Div. 2 AC	3, 4, 5

E — ISFSI

1 Confinement Boundary

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

ALERT: None

UNUSUAL EVENT — Damage to a loaded cask CONFINEMENT BOUNDARY

EU1.1 [ALL]

Damage to a loaded canister (MPC) CONFINEMENT BOUNDARY as indicated by measured dose rates on a loaded overpack > **EITHER**:
- 60 mrem/hr (gamma + neutron) on the top of the overpack
- 600 mrem/hr (gamma + neutron) on the side of the overpack, excluding inlet and outlet ducts

H — Hazards

1 Security

GENERAL EMERGENCY — HOSTILE ACTION resulting in loss of physical control of the facility

HG1.1 [1 | 2 | 3 | 4 | 5 | DEF]

A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor

AND EITHER of the following has occurred:

Any of the following safety functions cannot be controlled or maintained
- Reactivity
- RPV water level
- RCS heat removal

OR

Damage to spent fuel has occurred or is IMMINENT

SITE AREA EMERGENCY — HOSTILE ACTION within the PROTECTED AREA

HS1.1 [1 | 2 | 3 | 4 | 5 | DEF]

A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor

ALERT — HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes

HA1.1 [1 | 2 | 3 | 4 | 5 | DEF]

A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervisor

OR

A validated notification from NRC of an aircraft attack threat within 30 min. of the site

UNUSUAL EVENT — Confirmed SECURITY CONDITION or threat

HU1.1 [1 | 2 | 3 | 4 | 5 | DEF]

A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by the Security Shift Supervisor

OR

Notification of a credible security threat directed at the site

OR

A validated notification from the NRC providing information of an aircraft threat

2 Seismic Event

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

ALERT: None

[Refer to EAL CA6.1 or SA8.1 for escalation due to seismic event]

UNUSUAL EVENT — Seismic event greater than OBE levels

HU2.1 [1 | 2 | 3 | 4 | 5 | DEF]

Seismic event greater than Operating Basis Earthquake (OBE) as indicated by seismic instrumentation in the Control Room recording level greater than an OBE

Notes

1. The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.
2. If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.
3. If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is **no** longer VALID for classification purposes.
4. The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.
5. If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then **no** emergency classification is warranted.
6. If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, declaration of a General Emergency is **not** required.
7. This EAL does **not** apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.
8. A manual scram action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **not** include manually driving in control rods or implementation of boron injection strategies.
9. In the absence of reliable RCS temperature indication caused by the loss of decay heat removal capability, classification should be based on the RCS pressure increase criteria when the RCS is INTACT in Mode 4 or based on time to boil data when in Mode 5 or the RCS is **not** INTACT in Mode 4.

3 Natural or Tech. Hazard

SITE AREA EMERGENCY: None

ALERT: None

[Refer to EAL CA6.1 or SA8.1 for escalation due to natural or technological hazard]

UNUSUAL EVENT — Hazardous event

HU3.1 [1 | 2 | 3 | 4 | 5 | DEF]

A tornado strike within the PROTECTED AREA

HU3.2 [1 | 2 | 3 | 4 | 5 | DEF]

Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode

HU3.3 [1 | 2 | 3 | 4 | 5 | DEF]

Movement of personnel within the PROTECTED AREA is IMPEDED due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release)

HU3.4 [1 | 2 | 3 | 4 | 5 | DEF]

A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles (Note 7)

4 Fire

Table H-1 Fire Areas

- Control Complex (all elevations)
- Auxiliary Building (all elevations)
- Intermediate Building (all elevations)
- Fuel Handling Building (all elevations)
- Reactor Building (all elevations)
- Emergency Service Water Pump House (all elevations)
- Electrical Duct Chase Leading to ESW Building
- Diesel Generator Building (all areas except the Unit 2 Division 1, 2, and 3 DG Rooms)
- Steam Tunnel (all elevations)
- Diesel Generator Fuel Oil Storage Area
- Condensate Storage Tank
- Intake/Discharge Structure

ALERT: None

[Refer to EAL CA6.1 or SA8.1 for escalation due to fire]

UNUSUAL EVENT — FIRE potentially degrading the level of safety of the plant

HU4.1 [1 | 2 | 3 | 4 | 5 | DEF]

A FIRE is **not** extinguished within 15 min. of **any** of the following FIRE detection indications (Note 1):
- Report from the field (i.e., visual observation)
- Receipt of multiple (more than 1) fire alarms or indications
- Field verification of a single fire alarm

AND

The FIRE is located within **any** Table H-1 area

HU4.2 [1 | 2 | 3 | 4 | 5 | DEF]

Receipt of a single fire alarm (i.e., no other indications of a FIRE)

AND

The fire alarm is indicating a FIRE within **any** Table H-1 area

AND

The existence of a FIRE is **not** verified within 30 min. of alarm receipt (Note 1)

HU4.3 [1 | 2 | 3 | 4 | 5 | DEF]

A FIRE within the plant PROTECTED AREA **not** extinguished within 60 min. of the initial report, alarm or indication (Note 1)

HU4.4 [1 | 2 | 3 | 4 | 5 | DEF]

A FIRE within the plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish

5 Hazardous Gases

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

ALERT — Gaseous release IMPEDING access to equipment necessary for normal plant operations, cooldown or shutdown

HA5.1 [3 | 4 | 5]

Release of a toxic, corrosive, asphyxiant or flammable gas into **any** Table H-2 rooms or areas

AND

Entry into the room or area is prohibited or IMPEDED (Note 5)

Table H-2 Safe Shutdown Rooms/Areas

Room/Area	Modes
AX 574' Elev. RHR B	3, 4, 5
AX 620' Elev. West Hallway	3, 4, 5
CC 620' Elev. Div. 1 AC	3, 4, 5
CC 620' Elev. Div. 2 AC	3, 4, 5

6 Control Room Evacuation

GENERAL EMERGENCY: None

SITE AREA EMERGENCY — Inability to control a key safety function from outside the Control Room

HS6.1 [1 | 2 | 3 | 4 | 5 | DEF]

An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panels

AND

Control of **any** of the following key safety functions is **not** reestablished within 15 min. (Note 1):
- Reactivity
- RPV water level
- RCS heat removal

ALERT — Control Room evacuation resulting in transfer of plant control to alternate locations

HA6.1 [1 | 2 | 3 | 4 | 5 | DEF]

An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panels

UNUSUAL EVENT: None

7 Judgment

GENERAL EMERGENCY — Other conditions existing that in the judgment of the Emergency Coordinator warrant declaration of General Emergency

HG7.1 [1 | 2 | 3 | 4 | 5 | DEF]

Other conditions exist which in the judgment of the Emergency Coordinator 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 — Other conditions existing that in the judgment of the Emergency Coordinator warrant declaration of Site Area Emergency

HS7.1 [1 | 2 | 3 | 4 | 5 | DEF]

Other conditions exist which in the judgment of the Emergency Coordinator 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 — Other conditions existing that in the judgment of the Emergency Coordinator warrant declaration of an Alert

HA7.1 [1 | 2 | 3 | 4 | 5 | DEF]

Other conditions exist which, in the judgment of the Emergency Coordinator, indicate that events are in progress or have occurred which involve an 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

UNUSUAL EVENT — Other conditions existing that in the judgment of the Emergency Coordinator warrant declaration of a Unusual Event

HU7.1 [1 | 2 | 3 | 4 | 5 | DEF]

Other conditions exist which in the judgment of the Emergency Coordinator 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

Modes: 1 Power Operations | 2 Startup | 3 Hot Shutdown | 4 Cold Shutdown | 5 Refueling | DEF Defueled

SRO QUESTION #20 Reference

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

S System Malfunct.

1 Loss of Essential AC Power

General Emergency

Prolonged loss of all offsite and all onsite AC power to essential buses

SG1.1 1 2 3

Loss of **all** offsite and **all** onsite AC power capability to essential buses EH-11 and EH-12
AND EITHER:
Restoration of essential bus EH-11 or EH-12 in < 4 hours is **not** likely (Note 1)
OR
RPV water level **cannot** be restored and maintained > -25 in.

Loss of **all** essential AC and vital DC power sources for 15 minutes or longer

SG1.2 1 2 3

Loss of **all** offsite and **all** onsite AC power capability to essential buses EH-11 and EH-12 for ≥ 15 min.
AND
Indicated voltage < 116 VDC on ED-1-A and < 112 VDC on ED-1-B for ≥ 15 min. (Note 1)

Site Area Emergency

Loss of **all** offsite and **all** onsite AC power to essential buses for 15 minutes or longer

SS1.1 1 2 3

Loss of **all** offsite and **all** onsite AC power capability to essential buses EH-11 and EH-12 for ≥ 15 min. (Note 1)

Alert

Loss of **all but one** AC power source to essential buses for 15 minutes or longer

SA1.1 1 2 3

AC power capability, Table S-6, to essential buses EH-11 and EH-12 reduced to a single power source for ≥ 15 min. (Note 1)
AND
Any additional single power source failure will result in loss of **all** AC power to essential buses EH-11 and EH-12

Unusual Event

Loss of **all** offsite AC power capability to essential buses for 15 minutes or longer

SU1.1 1 2 3

Loss of **all** offsite AC power capability, Table S-6, to essential buses EH-11 and EH-12 for ≥ 15 min. (Note 1)

Table S-6 AC Power Sources

Offsite:
- Unit 1 Startup Transformer
- Unit 2 Startup Transformer
- Auxiliary Transformer Backfeed (**only** if already aligned)

Onsite:
- DG 1 (Division 1)
- DG 2 (Division 2)

2 Loss of Vital DC Power

General Emergency: see SG1.2 above

Site Area Emergency

Loss of **all** vital DC power for 15 minutes or longer

SS2.1 1 2 3

Indicated voltage on ED-1-A < 116 VDC and ED-1-B < 112 VDC for ≥ 15 min. (Note 1)

Alert: None

Unusual Event: None

3 Loss of Control Room Indications

General Emergency: None

Table S-1 Safety System Parameters
- Reactor power
- RPV water level
- RPV pressure
- Containment pressure
- Suppression Pool water level
- Suppression Pool temperature

Alert

UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress

SA3.1 1 2 3

An UNPLANNED event results in the inability to monitor one or more Table S-1 parameters from within the Control Room for ≥ 15 min. (Note 1)
AND
Any Significant transient is in progress, Table S-2

Unusual Event

UNPLANNED loss of Control Room indications for 15 minutes or longer

SU3.1 1 2 3

An UNPLANNED event results in the inability to monitor one or more Table S-1 parameters from within the Control Room for ≥ 15 min. (Note 1)

4 RCS Activity

Notes

1 The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

8 A manual scram action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **not** include manually driving in control rods or implementation of boron injection strategies.

Table S-2 Significant Transients
- Reactor scram
- Runback > 25% full electrical load
- Electrical load rejection > 25% electrical load
- ECCS injection
- Thermal power oscillations > 10%

Alert: None

Unusual Event

Reactor coolant activity greater than Technical Specification allowable limits

SU4.1 1 2 3

Offgas Pretreatment radiation monitor 1D17-K612 high alarm

SU4.2 1 2 3

Coolant activity > 0.2 µCi/gm Dose Equivalent I-131 for > 48 hours
OR
Coolant activity > 4.0 µCi/gm Dose Equivalent I-131 instantaneous

5 RCS Leakage

General Emergency: None

Site Area Emergency: None

Alert: None

Unusual Event

RCS leakage for 15 minutes or longer

SU5.1 1 2 3

RCS unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min. (Note 1)
OR
RCS identified leakage > 25 gpm for ≥ 15 min. (Note 1)
OR
Leakage from the RCS to a location outside containment > 25 gpm for ≥ 15 min. (Note 1)

6 RPS Failure

General Emergency: None

Site Area Emergency

Inability to shut down the reactor causing a challenge to RPV water level or RCS heat removal

SS6.1 1 2

An automatic or manual scram fails to shut down the reactor
AND
All actions to shut down the reactor are **not** successful as indicated by reactor power ≥ 4%
AND EITHER:
RPV water level **cannot** be restored and maintained > -25 in. or **cannot** be determined
OR
HCL exceeded (EOP Figures)

Alert

Automatic or manual scram fails to shut down the reactor, and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor

SA6.1 1 2

An automatic or manual scram fails to shut down the reactor
AND
Manual scram actions taken at the reactor control console (Manual PBs, Mode Switch, ARI) are **not** successful in shutting down the reactor as indicated by reactor power ≥ 4% (Note 8)

Unusual Event

Automatic or manual scram fails to shut down the reactor

SU6.1 1 2

An automatic scram did **not** shut down the reactor after **any** RPS setpoint is exceeded
AND
A subsequent automatic scram or manual scram action taken at the reactor control console (Manual PBs, Mode Switch, ARI) is successful in shutting down the reactor as indicated by reactor power < 4% (APRM downscale) (Note 8)

SU6.2 1 2

A manual scram did **not** shut down the reactor after **any** manual scram action was initiated
AND
A subsequent automatic scram or manual scram action taken at the reactor control console (Manual PBs, Mode Switch, ARI) is successful in shutting down the reactor as indicated by reactor power < 4% (APRM downscale) (Note 8)

7 Loss of Comm.

Table S-3 Communication Methods

System	Onsite	ORO	NRC
Plant Public Address System	X		
Plant Radio System Channels 1, 2 and 3	X		
State and County Notification Circuit (5-way)		X	
Control Room private (259-) lines		X	X
Private Branch Exchange, Service Building ("5000") Switch		X	X
Private Branch Exchange, Warehouse Building ("6000") Switch		X	X
Company Off-Premise Exchange		X	X
Commercial Telephone Systems	X	X	X
Emergency Telecommunications System (ETS)			X

Site Area Emergency: None

Alert: None

Unusual Event

Loss of **all** onsite or offsite communications capabilities

SU7.1 1 2 3

Loss of **all** Table S-3 onsite communication methods
OR
Loss of **all** Table S-3 ORO communication methods
OR
Loss of **all** Table S-3 NRC communication methods

8 Hazardous Event Affecting Safety Systems

Table S-4 Hazardous Events
- Seismic event
- Internal or external FLOODING event
- High winds
- Tornado strike
- FIRE
- EXPLOSION
- Other events with similar hazard characteristics as determined by the Shift Manager

Alert

Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode

SA8.1 1 2 3

The occurrence of **any** Table S-4 hazardous event
AND EITHER:
Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode
OR
The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode

Unusual Event: None

F Fission Product Barrier Degradation

FG1.1 1 2 3

Loss of **any** two barriers
AND
Loss or potential loss of the third barrier (Table F-1)

FS1.1 1 2 3

Loss or potential loss of **any** two barriers (Table F-1)

FA1.1 1 2 3

Any loss or **any** potential loss of **EITHER** Fuel Clad or RCS barrier (Table F-1)

Unusual Event: None

Table F-1 Fission Product Barrier Threshold Matrix

	FC - Fuel Clad Barrier		RCS - Reactor Coolant System Barrier		CNTMT - Containment Barrier	
	Loss	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
A RPV Water Level	1. SAMG entry is required	1. RPV level **cannot** be restored and maintained > 0 in. (TAF) or **cannot** be determined.	1. RPV level **cannot** be restored and maintained > 0 in. (TAF) or **cannot** be determined.	None	None	1. SAMG entry is required
B RCS Leak Rate	None	None	1. UNISOLABLE break in **any** of the following: • Main Steam Line • RCIC Steam Line • RWCU • Feedwater 2. Emergency Depressurization is required	1. UNISOLABLE primary system leakage that results in exceeding **EITHER**: One or more EOP-03 radiation entry conditions that can be read in the control room are met **OR** One or more EOP-03 area temperature entry conditions are met	1. UNISOLABLE primary system leakage that results in exceeding **EITHER**: One or more EOP-03 MAX SAFE area radiation conditions that can be read from the control room are reached **OR** One or more EOP-03 MAX SAFE area temperatures are reached	None
C CNTMT Conditions	None	None	1. Drywell pressure > 1.68 psig due to RCS leakage	None	1. UNPLANNED rapid drop in containment pressure following containment pressure rise 2. Containment pressure response **not** consistent with LOCA conditions	1. Containment pressure > 15 psig 2. Drywell or containment hydrogen concentration > 4% 3. HCL exceeded (EOP Figures)
D CNTMT Rad / RCS Activity	1. Drywell radiation > 400 R/hr **OR** Containment radiation > 600 R/hr 2. Primary coolant activity > 300 µCi/gm I-131 Dose Equivalent	None	1. Drywell radiation > 40 R/hr **OR** Containment radiation > 60 R/hr	None	None	1. Drywell radiation > 4,000 R/hr **OR** Containment radiation > 6,000 R/hr
E CNTMT Integrity or Bypass	None	None	None	None	1. UNISOLABLE direct downstream pathway to the environment exists after Containment isolation signal 2. Intentional Containment venting per EOPs	None
F EC Judgment	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the Fuel Clad barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the Fuel Clad barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the RCS barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the RCS barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the Containment barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the Containment barrier

Modes:

1	2	3	4	5	DEF
Power Operations	Startup	Hot Shutdown	Cold Shutdown	Refueling	Defueled

The perry Plant

HOT CONDITIONS (RCS > 200°F)

SRO QUESTION #20 Reference

C Cold SD/ Refueling System Malfunct.

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

1 RPV Level

GENERAL EMERGENCY

Loss of RPV inventory affecting fuel clad integrity with Containment challenged

CG1.1 | 4 | 5 |

RPV level < 0 in. (TAF) for ≥ 30 min. (Note 1)

AND

Any of the following indications of containment challenge:

- CONTAINMENT CLOSURE **not** established (Note 6)
- Drywell or containment hydrogen concentration > 4%
- UNPLANNED rise in Containment pressure
- Exceeding one or more Secondary Containment Control MAX SAFE area radiation levels that can be read in the control room (EOP-03)

CG1.2 | 4 | 5 |

RPV level **cannot** be monitored for ≥ 30 min. (Note 1)

AND

Core uncovery is indicated by **EITHER** of the following:

- UNPLANNED increase in **any** Table C-1 sump or pool levels of sufficient magnitude to indicate core uncovery
- UPPER POOL AREA 1D21-K083 high alarm

AND

Any of the following indications of containment challenge:

- CONTAINMENT CLOSURE **not** established (Note 6)
- Drywell or containment hydrogen concentration > 4%
- UNPLANNED rise in Containment pressure
- Exceeding one or more Secondary Containment Control MAX SAFE area radiation levels that can be read in the control room (EOP-03)

SITE AREA EMERGENCY

Loss of RPV inventory affecting core decay heat removal capability

CS1.1 | 4 | 5 |

CONTAINMENT CLOSURE **not** established

AND

RPV level < 16.5 in. (Level 1)

CS1.2 | 4 | 5 |

CONTAINMENT CLOSURE established

AND

RPV level < 0 in. (TAF)

CS1.3 | 4 | 5 |

RPV level **cannot** be monitored for ≥ 30 min. (Note 1)

AND

Core uncovery is indicated by **EITHER** of the following:

- UNPLANNED increase in **any** Table C-1 sump or pool levels of sufficient magnitude to indicate core uncovery
- UPPER POOL AREA 1D21-K083 high alarm

ALERT

Loss of RPV inventory

CA1.1 | 4 | 5 |

Loss of RPV inventory as indicated by RPV level < 130 in. (Level 2)

CA1.2 | 4 | 5 |

RPV level **cannot** be monitored for ≥ 15 min. (Note 1)

AND

UNPLANNED increase in **any** Table C-1 sump or pool levels due to a loss of RPV inventory

Table C-1 Sumps/Pool

- Drywell equipment drain sump
- Drywell floor drain sump
- CNTMT equipment drain sump
- CNTMT floor drain sump
- Suppression Pool
- RHR A, B, C, HPCS, LPCS, RCIC cubicle drain sumps
- Auxiliary Building floor drain sump
- IB / FHB floor drain sump
- Visual observation

UNUSUAL EVENT

UNPLANNED loss of RPV inventory for 15 minutes or longer

CU1.1 | 4 | 5 |

UNPLANNED loss of reactor coolant results in RPV level less than a required lower limit for ≥ 15 min. (Note 1)

CU1.2 | 4 | 5 |

RPV level **cannot** be monitored

AND

UNPLANNED increase in **any** Table C-1 sump or pool levels due to a loss of RPV inventory

2 Loss of Essential AC Power

GENERAL EMERGENCY

None

ALERT

Loss of **all** offsite power and **all** onsite AC power to essential buses for 15 minutes or longer

CA2.1 | 4 | 5 | DEF |

Loss of **all** offsite and **all** onsite AC power capability to essential buses EH-11 and EH-12 for ≥ 15 min. (Note 1)

UNUSUAL EVENT

Loss of **all** but one AC power source to essential buses for 15 minutes or longer

CU2.1 | 4 | 5 | DEF |

AC power capability, Table C-2, to essential buses EH-11 and EH-12 reduced to a single power source for ≥ 15 min. (Note 1)

AND

Any additional single power source failure will result in loss of **all** AC power to essential buses EH-11 and EH-12

Table C-2 AC Power Sources

Offsite:

- Unit 1 Startup Transformer
- Unit 2 Startup Transformer
- Auxiliary Transformer backfeed (**only** if already aligned)

Onsite:

- DG 1 (Division 1)
- DG 2 (Division 2)

3 RCS Temp.

GENERAL EMERGENCY

None

Table C-3 RCS Heat-up Duration Thresholds

RCS Status	CONTAINMENT CLOSURE Status	Heat-up Duration
RCS INTACT	N/A	60 min. *
RCS **Not** INTACT	established	20 min. *
	not established	0 min.

* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is **not** applicable

ALERT

Inability to maintain plant in cold shutdown

CA3.1 | 4 | 5 |

UNPLANNED increase in RCS temperature to > 200°F for > Table C-3 duration (Notes 1, 9)

OR

UNPLANNED RPV pressure increase > 10 psig

UNUSUAL EVENT

UNPLANNED increase in RCS temperature

CU3.1 | 4 | 5 |

UNPLANNED increase in RCS temperature to > 200°F

CU3.2 | 4 | 5 |

Loss of **all** RCS temperature and RPV water level indication for ≥ 15 min. (Note 1)

4 Loss of Vital DC Power

GENERAL EMERGENCY

None

ALERT

None

UNUSUAL EVENT

Loss of vital DC power for 15 minutes or longer

CU4.1 | 4 | 5 |

Indicated voltage on required vital DC buses ED-1-A < 116 VDC and ED-1-B < 112 VDC for ≥ 15 min. (Note 1)

5 Loss of Comm.

Table C-4 Communication Methods

System	Onsite	ORO	NRC
Plant Public Address System	X		
Plant Radio System Channels 1, 2 and 3	X		
State and County Notification Circuit (5-way)		X	
Control Room private (259-) lines		X	X
Private Branch Exchange, Service Building ("5000") Switch		X	X
Private Branch Exchange, Warehouse Building ("6000") Switch		X	X
Company Off-Premise Exchange		X	X
Commercial Telephone Systems	X	X	X
Emergency Telecommunications System (ETS)			X

SITE AREA EMERGENCY

None

ALERT

None

UNUSUAL EVENT

Loss of **all** onsite or offsite communications capabilities

CU5.1 | 4 | 5 | DEF |

Loss of **all** Table C-4 onsite communication methods

OR

Loss of **all** Table C-4 ORO communication methods

OR

Loss of **all** Table C-4 NRC communication methods

6 Hazardous Events Affecting Safety Systems

Table C-5 Hazardous Events

- Seismic event
- Internal or external FLOODING event
- High winds
- Tornado strike
- FIRE
- EXPLOSION
- Other events with similar hazard characteristics as determined by the Shift Manager

ALERT

Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode

CA6.1 | 4 | 5 |

The occurrence of **any** Table C-5 hazardous event

AND EITHER:

Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode

OR

The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode

UNUSUAL EVENT

None

Notes

1 The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

6 If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, declaration of a General Emergency is **not** required.

9 In the absence of reliable RCS temperature indication caused by the loss of decay heat removal capability, classification should be based on the RCS pressure increase criteria when the RCS is INTACT in Mode 4 or based on time to boil data when in Mode 5 or the RCS is **not** INTACT in Mode 4.

Modes:

1	2	3	4	5	DEF
Power Operations	Startup	Hot Shutdown	Cold Shutdown	Refueling	Defueled

The perry Plant

COLD CONDITIONS

(RCS ≤ 200°F)