GGNS LOT 2015 NRC INITIAL LICENSED OPERATOR WRITTEN EXAMINATION

RO EXAM

ANSWER KEY

1	C		26	D		51	C
2	C		27	C		52	D
3	C		28	D		53	B
4	B		29	C		54	C
5	B		30	B		55	C
6	A		31	D		56	A
7	D		32	C		57	B
8	A		33	D		58	A
9	B		34	A		59	C
10	A		35	C		60	D
11	D		36	A		61	C
12	D		37	A		62	A
13	C		38	B		63	C
14	C		39	B		64	A
15	A		40	C		65	D
16	A		41	B		66	B
17	D		42	B		67	B
18	C		43	B		68	C
19	D		44	B		69	D
20	B		45	A		70	A
21	A		46	D		71	B
22	D		47	C		72	B
23	A		48	C		73	B
24	A		49	D		74	A
25	A		50	B		75	D

Examination Outline Cross-Reference	Level	RO
295003 Partial or Complete Loss of AC. Knowledge of the reasons for the following responses as they apply to a partial or complete loss of A.C. power: AK3.05 Reactor SCRAM	**Tier #**	1
	Group #	1
	K/A #	295003: AK3.05
	Rating	3.7

Question 1

The plant is operating at rated thermal power when the normal feeder breaker to Bus 15AA trips due to a ground fault on the bus itself.

Since Bus 15AA cannot be immediately re-energized, the Loss of AC Power ONEP directs operators to manually scram the reactor.

Which of the following describes a reason for scramming the reactor?

A. It is a conservative action based on the unavailability of Plant Air Compressor ‘A’ as being unacceptable to sustained plant operation.

B. Anticipates the automatic scram that will occur on high drywell pressure as drywell temperature rises due to the loss of two drywell chillers.

C. Anticipates the automatic scram or control rod drifts as a result of a loss of Instrument Air to Containment.

D. It is a conservative action to place the plant in a safe shutdown and cooled down condition before a station blackout event might occur.

Answer: C

Explanation:

P53-F001, Instrument Air Supply Header To CTMT, is an air-operated valve that fails closed on loss of power to its solenoid. This Div 1 isolation valve’s solenoid is powered from 15AA. Therefore, the 15AA loss fails F001 closed, cutting off Instrument Air to that header feeding CTMT. The inboard MSIVs and scram air header are loads on that header. MSIVs will begin to drift closed as air pressure lowers in their control units. An auto-scram will result from the MSIV closure. Control Rods will begin to drift in as scram air pressure lowers to the point that scram valves begin to open. Because this question is asking for just “a reason”, this single failure mechanism is enough to justify the correct answer. Pre-empting the auto-scram by inserting a manual scram conforms to the “Conservative Decision-Making” requirements of EN-OP-115, Conduct of Operations.

Answer A - is wrong for the reason already discussed. Also, PAC ‘A’ is powered from Bus 16AB, not 15AA.

Answer B - is wrong. This choice is alluding to the two (of 4 total) drywell chillers that are ESF bus powered (the other 2 being BOP bus powered). However, the 2 that are ESF powered are fed from bus 16AB, not from bus 15AA. Answer C is correct Answer D is wrong.
Technical References: **GLP-OPS-R21** **04-1-01-R21-15** **05-1-02-I-4, Loss of A/C Power ONEP**
References to be provided to applicants during exam: **None**
Learning Objective: **GLP-OPS-R21 Objective 4** **GLP-OPS-ONEP**

Question Source:	Bank #	466
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2010
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
295004 Partial or Total Loss of DC Power **Ability to determine and/or interpret the following as they apply to a partial or complete loss of D.C. power:** **AA2.03: Battery voltage**	**Tier #**	1
	Group #	1
	K/A #	295004: AA2.03
	Rating	2.8

Question 2

The plant is operating at rated power.

Power is suddenly lost to Div 3 battery charger 1C4 when its feeder breaker from MCC 17B01 trips open (breaker internal fault).

Which of the following identifies the initial response of the Div 3 DC bus voltage as indicated on control room panel P601?

A. Remains constant on the 11DC bus.

B. Remains constant on the 11DB bus.

C. Lowers by 5 to 10 volts on the 11DC bus.

D. Lowers by 5 to 10 volts on the 11DB bus.

Answer: C

Explanation:

Unlike the Div 1 and Div 2 batteries which have two load-sharing chargers always connected, the Div 3 battery only has one charger (normally the 1C4 charger) connected at a time. Therefore, when that charger loses its MCC AC power source, it de-energizes and is no longer able to float the bus at the normal 5-10 volts above battery bank terminal voltage. As such, the resulting P601 battery bus indication will drop by 5 to 10 volts (i.e., the bus will now be carried by the battery itself).

A is wrong. This choice represents the response if the same failure were to occur for one of the Div 1 or Div 2 battery chargers (i.e., where the load-sharing charger would continue to float the bus at the normal float voltage); its plausibility should speak for itself in this regard.

B is plausible if applicant misunderstands that the 1C4 battery charger is aligned to the 11DB bus which would allow battery bus voltage to remain constant if the 11DB battery charger was lost.

D is plausible if applicant understands the correct bus response but selects the wrong bus alignment.

Validation Results
A. One RO choice

Technical References: 04-1-01-L11-1, Plant DC SOI		
References to be provided to applicants during exam: **None.**		
Learning Objective: GLP-OPS-L1100, Objective 19		
Question Source:	Bank #	339
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2011 NRC
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	2
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
295005 Main Turbine Generator Trip **Ability to operate and/or monitor the following as they apply to main turbine generator trip:** **AA1.02: RPS**	**Tier #**	1
	Group #	1
	K/A #	295005: AA1.02
	Rating	3.6

Question 3

The plant is operating at 38% power.

The main turbine generator trips.

A malfunction with the Turbine Control Valves results in only the B & C Turbine Control Valves close. (Fluid Pressures <30 psig)

What is the expected status of RPS?

A. Division 1 half scram

B. Division 2 half scram

C. Full scram.

D. No scram signal.

Answer: C
Explanation: A. If student confuses with MSIV valve closure logic a division 1 half scram would be generated. B. if student confuses with NSSSS logic B&C combination do cause isolation. C. If reactor is operating >35.4 CTP, when the turbine trips and the B&C TSVs close the reactor will receive no scram signal. If the B&C TCVs close an automatic full scram signal will be generated. D. If student assumes the TCV logic is the same as the TSV, then no scram signal will be generated.
Technical References: GLP-OPS-C7100
References to be provided to applicants during exam: None
Learning Objective: GLP-OPS-C7100 Objective 9

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(6)	

Examination Outline Cross-Reference	Level	RO
295006 Scram **Knowledge of the interrelations between SCRAM and the following:** **AK2.06: Reactor Power**	**Tier #**	1
	Group #	1
	K/A #	295006 AK2.06
	Rating	4.2

Question 4

The plant is operating at rated power.

Recirculation flow is Wd = 40 KGPM

Which of the following MINIMUM VALUES will cause a reactor SCRAM on high reactor power if the coincidence is met (2 out of 4 voter)?

A. APRM upscale neutron flux at 115.3%.

B. APRM upscale neutron flux at 117.3%.

C. APRM upscale Thermal (STP) at 80.1%.

D. APRM upscale Thermal (STP) at 58.1%.

Answer: B
Explanation: APRM Upscale Neutron Flux (2 out 4 voter) max value at rated power is 117.3% when the reactor mode switch is in RUN. A. Plausible because the APRM Upscale Thermal (STP is clamped at 111%. B. Correct setpoint C. STP = 0.58Wd + 59.1 = 0.58(40)+59.1= 82.3% is the calculated setpoint for rated conditions.. D. STP = 0.58Wd + 37.4 = 0.58(40)+37.4= 60.6% is the calculated setpoint for single loop operations.
Technical References: GFIG-OPS-C7100, Table 1
References to be provided to applicants during exam: None
Learning Objective: GFIG-OPS-C7100 Objective 10

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(6)	

Examination Outline Cross-Reference	Level	RO
295016 Control Room Abandonment **For control room abandonment:** **G2.4.35: Knowledge of local auxiliary operator tasks during an emergency and the resultant operation effects.**	**Tier #**	1
	Group #	1
	K/A #	295016 G2.4.35
	Rating	3.8

Question 5

The control room has been abandoned due to a fire in panel P864.

Control of the plant has been established at the Remote Shutdown Panels (RSPs).

At local panel P152 (Area 25A, El. 111'), operators have placed the following switch in the ON position:

- Transfer Switch for Lockout Transfer Relay

When placing this switch in the ON position, it isolates what equipment from the control room?

A. Isolates ALL the Div 1 powered equipment.

B. Isolates ALL the equipment controlled from RSP P150.

C. Isolates BOTH Div 1 and 2 powered equipment.

D. Isolates BOTH equipment controlled from RSPs P150 and P151.

Answer: B
Explanation: A. Plausible because P150 contains division 1 equipment only. The transfer switch does not isolate ALL division one equipment from control room. B. Per the site electrical drawings, C61-HSS-M150 energizes lockout relays on H22-P152 which electrically isolate components operated at the H22-P150 (Division 1) Remote Shutdown Panel from the Main Control Room. Other selected Division 1 powered/controlled equipment have isolations provided on Alternate Shutdown Panels and the Diesel Control Panel. Not all Division 1 equipment is affected by Alternate Shutdown /Remote Shutdown Panels and NO Division 2 equipment is affected. C. Plausible because P150 and P151 are division one and division two equipment only. The transfer switch does not isolate ALL division one and two equipment from control room. D. Plausible because P150 is isolated but the P151 is not and is manned at the same time. Validation Two people selected A.

Technical References:
05-1-02-II-1, Shutdown from Remote Shutdown Panels ONEP sections 1.3; 1.10; Attachments III, IV, and XXI.
GLP-OPS-C6100

References to be provided to applicants during exam:
None

Learning Objective:
GLP-OPS-ONEP, objective 54; 55
GLP-OPS-C6100, objective 11

Question Source:	Bank #	579
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	None
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	3
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
295018 Partial or Total Loss of CCW **Ability to determine and/or interpret the following as they apply to partial or complete loss of component cooling water:** **AA2.01: Component temperatures**	**Tier #**	1
	Group #	1
	K/A #	295018 AA2.01
	Rating	3.3

Question 6

The plant is operating at 100% power.

The CRS has entered the Loss of CCW ONEP due to one pump running.

A reactor SCRAM and trip of both recirc pumps is required when which of the following temperatures cannot be maintained less than 135F?

A. Both CRD pump oil temperature

B. Both CRD pump stator temperature

C. Reactor Recirculation Pump seal water discharge temperature

D. Reactor Water Cleanup (RWCU) non-regenerative heat exchanger outlet temperature

Answer: A
Explanation: Per Loss of CCW ONEP when a complete loss is detected perform the immediate actions, complete loss is defined in the NOTE prior to Immediate actions that states to assume a complete loss when any of 4 items are reached. Two of the 4 have been reached, (1) Recirc Pump bearing temps cannot be maintained below their alarm setpoints and (2) CRD oil temp cannot be maintained less than 135. ‘A’ is correct ‘B’ is wrong this is plausible because it is another CRD pump temperature. ‘C’ is wrong this is plausible because recirc seal water temperature setpoint is 135F, but is not an entry into the Loss of CCW ONEP. The entry condition is based on high seal water CAVITY temperature of 180F. ‘D’ is wrong, but plausible because the RWCU heat exchanger ISOLATION is mentioned in the subsequent actions.
Technical References: GLP-OPS-P4200

GLP-OPS-B3300 05-1-02-V-1, Loss of Component Cooling Water ONEP		
References to be provided to applicants during exam: **None**		
Learning Objective: GLP-OPS-ONEP Objective 2		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	3
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
295019 Partial or Total Loss of Inst. Air Knowledge of the reasons for the following responses as they apply to partial or complete loss of instrument air: AK3.02: Standby air compressor operations	Tier #	1
	Group #	1
	K/A #	295019: AK3.02
	Rating	3.5

Question 7

The following Plant Air compressor (PAC) configuration currently exits:

PAC A in Standby
PAC B running
PAC C shutdown (idle)

An event occurred and all DGs have re-powered their buses.

Without operator action, 30 seconds after busses have been automatically restored no PACs are running and no cooling water is available.

What is the reason for the Total Loss of Air?

A. A 16AB Bus lockout occurred.

B. Only a LOP occurred.

C. Only a LOCA occurred.

D. A simultaneous LOP/LOCA occurred.

Answer: D
Explanation: A. Plausible because PAC A is powered from 16AB. Cooling water is not powered from 16AB. B. Plausible if PAC A and cooling water flow both lost power. C. LOCA shed trips and locks out PAC 'A'. LOCA interlocks closed the SSW 'B' valves to all PACs. D. The LOP/LOCA will shed PAC A and remove cooling water.
Technical References:

GLP-OPS-P5100		
References to be provided to applicants during exam: None		
Learning Objective: GLP-OPS-P5100 Obj 30, 31		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	None
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
295021 Loss of Shutdown Cooling **Knowledge of the interrelations between loss of shutdown cooling and the following:** **AK2.01: Reactor water temperature**	**Tier #**	1
	Group #	1
	K/A #	295021 AK2.01
	Rating	3.6

Question 8

A refueling outage is in progress with RHR Shutdown Cooling (SDC) in operation.

RHR SDC is lost and no alternate means of decay heat removal is available.

ONEP 05-1-02-III-1, Inadequate Decay Heat Removal directs to **CONTROL** Suppression Pool temperature to maintain RPV water temperature above ___(a)___ **AND** vessel cooldown rate to less than 100 degrees/hour.

Per IOI 3 (Plant Shutdown), reactor water temperature is required to be monitored once every ___(b)____ (30/60) minutes to verify cooldown rate and temperature and pressure are to the right of curve B of figure 1 of the PTLR per TS 3.4.11.

	(a)	(b)
A.	70	30
B.	70	60
C.	50	30
D.	50	60

Answer: A

Explanation:

ONEP 05-1-02-III-1, Inadequate Decay Heat Removal section 3.4.3f step 7 directs you to **CONTROL** Suppression Pool temperature to maintain RPV water temperature above 70 degrees **AND** vessel cooldown rate to less than 100 degrees/hour. Per IOI 3, Attachment II, Step 6.1.2 Check at least once per 30 minutes that cooldown rate is less than 90F/hr and Reactor Coolant temperature and pressure are to the right of curve B of Figure 1 of the PTLR per TS 3.4.11.

A. Correct
B. Incorrect but plausible as the first half is correct but the requirements for verification is 30 min not 60 min.
C. Incorrect but plausible as students may believe that 50 degrees is a low temp requirement for the vessel and the second half is correct

D. Incorrect but plausible as described in other distractors.		
Technical References: 05-1-02-III-1, Inadequate Decay Heat Removal ONEP 3-1-01-3, IOI 3 Plant Shutdown		
References to be provided to applicants during exam: **None**		
Learning Objective: GLP-OPS-ONEP, Objective 2		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(14)	

Examination Outline Cross-Reference	Level	RO
295023 Refueling Acc **Knowledge of the operational implications of the following concepts as they apply to refueling accidents:** **AK1.03: Inadvertent criticality**	**Tier #**	1
	Group #	1
	K/A #	295023 AK1.03
	Rating	3.7

Question 9

The bases of the refueling position one-rod out-interlock is to:

A. Prevent possible fuel damage caused by withdrawing a control rod from a fuel cell containing less than four fuel assembles.

B. Prevent the reactor from becoming critical during refueling operations.

C. Prevent the potential for draining the cavity / vessel during the performance of CRDM maintenance.

D. Prevent the movement of the refueling bridge over the reactor core with a control rod withdrawn.

Answer: B

Explanation:

A. Administrative controls are utilized to prevent the possibility of fuel damage in fuel cells containing less than four fuel assembles
B. Correct – Lesson Plan GLP-RF-F-1105 and GLP-OPS-MCD14 state that the refuel position one-rod-out interlock restricts the movement of control rods to reinforce unit procedures that prevent the reactor from becoming critical during refueling operations.
C. The one-rod-out interlock prevents withdrawal of more than one control rod and does not provide protection from an OPDRV during the performance of CRDM maintenance.
D. This is a Refuel Equipment Interlock and is governed under Technical Specification 3.9.1.

Technical References:
Technical Specifications 3.9.2 and Lesson Plan GLP-RF-F1105 and GLP-OPS-MCD14

References to be provided to applicants during exam: **None.**		
Learning Objective:		
Question Source:	Bank #	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	Riverbend 2014
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	4
10CFR Part 55 Content:	55.41(b)(13)(2)	

Examination Outline Cross-Reference	Level	RO
295024 High Drywell Pressure **Ability to determine and/or interpret the following as they apply to high drywell pressure:** **EA2.02: Drywell temperature**	**Tier #**	1
	Group #	1
	K/A #	295024 EA2.02
	Rating	3.9

Question 10

The plant is operating at 100% power.

A leak develops and drywell pressure steadies at 1.31 psig.

Without operator action, which of the following identifies an expected change due to the change in drywell pressure?

A. Drywell temperatures are rising.

B. LPCS surveillance using test return to suppression pool automatically secured.

C. RWCU flow secured.

D. RCIC exhaust vacuum breakers closed.

Answer: A
Explanation: A. A group 6 isolation takes Chilled Water (P72) away from the drywell. Without operator action to re-open these isolation valves 30 seconds after the isolation signal, DW temperatures will begin to rise. B. A group 5 isolation secures LPCS test line upon Drywell Pressure of 1.39 psig. C. A group 8 isolation secures RWCU flow upon reactor vessel water level -41.6”. D. A group 9 isolation closes RCIC exhaust vacuum breakers upon Drywell Pressure of 1.39 psig. Validation One person selected C.
Technical References: 05-1-02-III-5, Automatic Isolations ONEP
References to be provided to applicants during exam: None
Learning Objective: GLP-OPS-M7101 OBJ 30.1

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	3
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
295025 High Reactor Pressure **Knowledge of the interrelations between high reactor pressure and the following:** **EK2.09: Reactor Power**	**Tier #**	1
	Group #	1
	K/A #	295025 EK2.09
	Rating	3.9

Question 11

A reactor vessel steam dome high pressure scram signal should be the signal that **FIRST** generates a reactor scram on a(n):

A. turbine trip at full power

B. closure of all MSIVs at full power

C. single MSIV closure with reactor power at 75%

D. IPC fails causing Turbine control valves to close with reactor power at 75%

Answer: D		
Explanation: Interpreting reactor pressure involves determining if it is responding as expected. A is wrong because the reactor will scram on Turbine stop valve closure or low trip oil pressure. B is wrong because the reactor will scram on the main steam isolation valve closure. C is wrong because this should not result in a scram. D is correct. At less than RTP, the high pressure signal should precede the high power signal Validation One person selected B and one person selected C.		
Technical References: Mitigating of Core Damage (EPTS-2) Pressure Increase Events, GLP-OPS-MCD12 UFSAR section 5.2.2.2.3.1 Tech Spec Basis 3.3.1.1		
References to be provided to applicants during exam: None		
Learning Objective: GLP-OPS-MCD12 Obj 1		
Question Source:	Bank #	1054
(note changes; attach parent)	Modified Bank #	

	New	
Question History:	Last NRC Exam	2014
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(6)	

Examination Outline Cross-Reference	Level	RO
295026 Suppression Pool High Water Temp. **Ability to operate an/or monitor the following as they apply to suppression pool high water temperature:** **EA1.03: Temperature monitoring**	**Tier #**	1
	Group #	1
	K/A #	295026 EA1.03
	Rating	3.9

Question 12

The plant is operating at rated power with RCIC testing in progress.

Average Suppression Pool temperature has risen to 100°F.

Which of the following describes the required operator actions?

A. Place the Reactor Mode Switch to SHUTDOWN.
Enter EP-3 and place RHR in Suppression Pool Cooling.

B. Immediately secure RCIC.
Enter EP-3 and place RHR in Suppression Pool Cooling.

C. Place the Reactor Mode Switch to SHUTDOWN and enter EP-2.
Depressurize the reactor to <200 psig.

D. Enter EP-3 and place RHR is Suppression Pool Cooling.
Monitor Suppression Pool temperature.

Answer: D
Explanation: At > 1% power and testing which adds heat to the suppression pool suppression pool temperature is allowed to go to 105°F. At > 95°F the required actions are to enter EP-3 at 95°F and place suppression pool cooling in service and monitor suppression pool temp not to exceed 105°F. A is incorrect because Mode switch to shutdown is required at > 110°F. B is incorrect because RCIC is secured immediately at >105°F. C is incorrect because entry to EOP-2 is required at >110°F and depressurization is required at >120°F.
Technical References: Tech Spec 3.6.2.1, Supp Pool Average Water Temperature EP-3, PC Control

References to be provided to applicants during exam: **None.**		
Learning Objective: GLP-OPS-EP3TR obj 5		
Question Source:	Bank #	931
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2008
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
295027 High Containment Temperature **Knowledge of the operational implications of the following concepts as they apply to high containment temperature (Mark III containment only):** **EK1.02: Reactor water level measurement: Mark-III**	**Tier #** **Group #** **K/A #** **Rating**	1 1 295027 EK1.02 3.0

Question 13

A LOCA is in progress with the following:

- Wide Range level is -10”
- Fuel Zone level is -25”
- Upset Range level is 5”
- Shutdown Range level is 10”
- RPV pressure is 50 psig
- Drywell temperature (166 ft) = 220°F; (139 ft) = 190°F
- CTMT temperature (166 ft) = 155°F; (139 ft) = 150°F

Which of the reactor water level instruments **is/are** usable?

A. Fuel Zone Range, Wide Range, Upset Range, and Shutdown Range

B. Fuel Zone Range, Wide Range, and Upset Range, only

C. Fuel Zone Range and Wide Range, only

D. Fuel Zone Range, only

Answer: C
Explanation: See EP-1 CAUTION 1. The DW and CTMT temperatures in the stem fall within the “safe zone” of the RPVST curve (Figure 2); therefore, there are no possible boiling concerns. This makes the Fuel Zone Range instrument completely valid and usable per Caution 1.1. Per Caution 1.2, a Wide Range, Upset Range, or Shutdown Range instrument may not be used if BOTH 1) indicated level is below a certain limit AND 2) DW or CTMT temperature at a specified elevation is above a certain limit. The indicated level for Wide Range (-10”) is above the specified limit (-131”); therefore, Wide Range is usable. The indicated level for Upset Range (5”) is below its limit (159”) AND the stem’s given DW temperature at the 166 ft elevation (220°F) is above the associated limit (195°F); therefore, Upset Range is <u>not</u> usable. The indicated level for Shutdown Range (10”) is below its limit (139”) AND the stem’s given DW temperature at the 166 ft elevation (220°F) is above the associated limit (66°F); therefore, the Shutdown Range is <u>not</u> usable. Distracters A, B, and D are wrong but are plausible based on the Applicant’s need to apply Caution 1

as already described.		
Technical References: EP-1, CAUTION 1		
References to be provided to applicants during exam: EP-1 CAUTION 1		
Learning Objective: GLP-OPS-EP02, Objective 8		
Question Source:	Bank #	364
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2011
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	2
10CFR Part 55 Content:	55.41(b)(6)	

Examination Outline Cross-Reference	Level	RO
295028 High Drywell Temperature **For high drywell temperature:** **G2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.**	**Tier #**	1
	Group #	1
	K/A #	295028 G2.1.7
	Rating	4.4

Question 14

The following plant parameters exist:

RPV Level 12.4 inches (lowest level reached), rising
Drywell Temperature 137F, steady
Suppression Pool Level is 18.5 feet, steady
RHR room temperature 155F, steady (No associated alarms)

Based on given parameters, what Emergency Procedure(s) should the crew be entering?

A. EP-3, Containment Control, and EP-4, Auxiliary Building Control

B. EP-2, RPV Control, and EP-3, Containment Control.

C. EP-3, Containment Control, Only.

D. EP-2, RPV Control, Only.

Answer: C
Explanation: Only Drywell Temperature limit has been reached (135F) for EP entry conditions. A is wrong. EP-3 is entered for high Drywell Temperature. Plausible if RHR room temperature entry condition was met. B is wrong. EP-3 is entered for high Drywell Temperature. Plausible if RPV level entry condition was met. D is wrong. Plausible if RPV level or drywell pressure entry conditions were met.
Technical References: 05-S-01-EP–2, 05-S-01-EP–3, and 05-S-01-EP–4
References to be provided to applicants during exam: None

Learning Objective: GLP-EP–EPT19, Obj. 6.1		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	41(b)(10)	

Examination Outline Cross-Reference	Level	RO
295030 Low Suppression Pool Wtr Lvl **Ability to operate and/or monitor the following as they apply to low suppression pool water level:** **EA1.06: condensate storage and transfer (make-up to the suppression pool): Plant-specific**	**Tier #**	1
	Group #	1
	K/A #	295030 EA1.06
	Rating	3.4

Question 15

(a) What system can be used to makeup water to suppression pool from the CST?

(b) When making up to the Suppression Pool, level must be maintained below ___(b)___ to prevent the automatic suction swap.

	(a)	(b)
A.	RCIC	18 feet 9 3/4 inches
B.	RCIC	23 feet 9 3/4 inches
C.	LPCS	18 feet 9 3/4 inches
D.	LPCS	23 feet 9 3/4 inches

Answer: A

Explanation:

A. Correct. RCIC can be used to makeup water to the suppression pool by taking suction from the CST and using the test return to add water to the suppression pool. Per SOI P&L 3.7 maintain suppression pool below 19’9 ¾” to prevent automatic transfer of RCIC suction.

B. Plausible if student remembers that RCIC can makeup water to the suppression pool using the CST. Plausible because 23 feet 9 inches is the high suppression pool level in EP-3 requiring action to lower after SPMU has been initiated.

C & D. Plausible if student remembers that LPCS (HPCS can actually be used) can makeup water to the suppression pool using the CST.

Technical References:
04-1-01-E51-1, Reactor Core Isolation Cooling System, Section 6.4 and P&L 3.7

References to be provided to applicants during exam: **None**		
Learning Objective:		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	3
10CFR Part 55 Content:	55.41(b)(8)	

Examination Outline Cross-Reference	Level	RO
295031 Reactor Low Water Level **Knowledge of the reasons for the following responses as they apply to reactor low water level:** **EK3.02: Core coverage**	**Tier #**	1
	Group #	1
	K/A #	**295031: EK3.02**
	Rating	4.4

Question 16

Per EP-2 basis for Alternate Level Control, what is reason to restore and maintain RPV level above -160 in. with one or more Table 1 systems?

A. To re-establish core cooling by submergence.

B. To maintain above minimum steam cooling level.

C. To maintain above minimum zero injection RPV Water Level without RPV injection.

D. To maintain above minimum spray cooling level.

Answer: A		
Explanation: A. Correct. The EP-2 basis states that any or all of the listed systems may be used, as necessary to reestablish core cooling by submergence. B. Plausible because adequate core cooling is assured if level is above minimum Steam Cooling RPV Water level. (-191 inches) but this is not correct for the given stem conditions. C. Plausible because adequate core cooling is assured if level is above minimum zero injection RPV Water level (-204 inches) but this is not correct for the given stem conditions. D. Plaubile because adequate core cooling is assured if level is above spray cooling level (-217 inches) but this is not correct for the given stem conditions. B, C, & D are plausible because they are all definitions of adequate core cooling.		
Technical References: 02-S-01-40, Attachment IV, p. 19 of 51, Step L-6 02-S-01-43, Transient Mitigation Strategy, Section 5.1		
References to be provided to applicants during exam: None		
Learning Objective:		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	3
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown **Knowledge of the operational implications of the following concepts as they apply to SCRAM condition present and reactor power above APRM downscale or unknown:** **EK1.07: Shutdown margin**	**Tier #**	1
	Group #	1
	K/A #	295037 EK1.07
	Rating	3.4

Question 17

For which of the following conditions will shutdown margin be sufficient to allow EP-2A to be exited without obtaining concurrence from Reactor Engineering?

A. Cold Shutdown Boron weight has been injected with five control rods at position 48.

B. >50% of the control rods are at position 02 or beyond.

C. ONLY two peripheral control rods are at position 48.

D. ONLY one center core control rod is at position 48.

Answer: D		
Explanation: A. Distracter 1 is incorrect because reliance on boron to remain shutdown negates exiting EP-2A. B & C are incorrect because even as inconsequential as the rod densities appear, they would still have to be analyzed by reactor engineering. Per Tech Specs definitions, “SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: (c) All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. After a scram the crew can determine the reactor shutdown with only one control rod not full in. If more than one is withdrawn then a calculation must be performed.		
Technical References: 05-S-01-PSTG 02-S-01-40, EP Technical Bases Att IV. pg 5 of 51		
References to be provided to applicants during exam: **None.**		
Learning Objective: GLP-OPS-TS001, OBJ. 4.13		
Question Source:	Bank #	873

(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2007
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	4
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
295038 High Off-site Release Rate **Knowledge of the interrelations between high off-site release rate and the following:** **EK2.03: Plant ventilation systems**	**Tier #**	1
	Group #	1
	K/A #	295038 EK2.03
	Rating	3.6

Question 18

A High Off-site release is in progress and Standby Gas Treatment has been initated, The flow control vanes, T48-F500A(B), will control in their intermediate position when the Enclosure Building Differential pressure setpoint of ____________ is reached.

Assume that 60 seconds has elapsed since the initiation signal to start the Standby Gas Treatment System.

A. -.2 inches wc

B. -.25 inches wc

C. -.75 inches wc

D. -.88 inches wc

Answer: C		
Explanation: The flow control vanes modulate when -.75 inches wc is reached or 120 seconds has passed, whichever comes first; therefore Answer C is correct. Answers A, B, and D have setpoints for the system and therefore are plausible but not correct.		
Technical References: 04-1-01-T48-1 E-1257		
References to be provided to applicants during exam: **None.**		
Learning Objective: GLP-OPS-T4801 Objective: 8.4		
Question Source:	Bank #	390
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	N/A

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	3
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
600000 Plant Fire On Site **Ability to determine and interpret the following as they apply to plant fire on site:** **AA2.17: Systems that may be affected by the fire**	**Tier #**	1
	Group #	1
	K/A #	600000 AA2.17
	Rating	3.1

Question 19

A fire is in progress in room 1A221.

That room's fire suppression system has automatically gone into service.

Which of the following equipment/component is being directly damaged by the fire/fire suppression initiation?

A. The LPCI 'B' Injection Valve (E12-F042B) drive motor

B. The SSW Inlet Valve To DG12 Water Cooler (P41-F018B) drive motor

C. DG12 Outside Air Fan (X77-C001B) motor

D. Cabling for Battery Charger 1B5

Answer: D
Explanation: See GGNS Fire Pre-Plan, Volume 1, specifically pre-plan A-16, where we see that Total CO2 Flooding is the only automatic fire suppression for Zone 1A221. See procedure 10-S-03-2, Response To Fires, Attachment IV, page 131 of 211, for Zone 1A221, where we find the Safe Shutdown Equipment (or cabling for such) located in that zone. See the same procedure, Attachment III, page 1 of 2, specifically "General Note" #2, which explains the use of the 'C' and 'D' designators for the equipment in each Attachment IV fire zone. Only cabling for, not the 1B5 charger itself, is located in this zone ('C' designator), and this zone uses an automatic CO2 flooding system. A. Plausible because 1E12-F042C cabling is located in Zone 1A221 not the drive motor. B. Plausible because 1P41-F018B cabling is located in Zone 1A221 not the drive motor. C. Plausible because X77-C001B cabling is located in this zone, not the fan motor. D. Correct.
Technical References: GGNS Fire Pre-Plans, Volume 1 10-S-03-2, Response To Fires
References to be provided to applicants during exam: 10-S-03-2: cover page; Attachment III, both pages; Attachment IV, pages 125 through 135

Learning Objective: GLP-OPS-PROC Obj 58		
Question Source:	Bank #	148
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2012
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	2
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
700000 Generator Voltage and Electric Grid Disturbances **For generator voltage and electric grid disturbances:** **G2.4.45: Ability to prioritize and interpret the significance of each annunciator or alarm.**	**Tier #**	1
	Group #	1
	K/A #	700000: G2.4.45
	Rating	4.1

Question 20

The plant is operating at 100% power.

Following an electrical grid transient, the following annunciators are received on the P680:

9A-B1 TSE-STU CAB FAIL

9A-D9 GEN UNDERFREQ

9A-B9 GEN DC CONT PWR LOSS

9A-E10 MN XFMR PH-A TROUBLE

Which of the listed annunciators would require operators to prioritize to immediately scram the reactor if not already completed?

A. TSE-STU CAB FAIL

B. GEN UNDERFREQ

C. GEN DC CONT PWR LOSS

D. MN XFMR PH-A TROUBLE

Answer: B
Explanation: A. Plausible because a TSE fault condition will prevent the turbine generator speed or load reference signal from changing. Immediate operator action is only required to prevent possible turbine generator damage if starting up and at the critical speed range. The generator is not starting up at the critical speed range because the plant is at 100%.

B. Correct. If generator frequency decreases to 57 Hz the generator will trip, the turbine will trip, and the reactor will scram. If this automatic action has not occurred, the reactor operator should take manual control of the required automatic action.
C. Plausible because ARI possible causes include generator lockout relays. There are no listed automatic actions per the ARI.
D. Plausible because GFIG-OPS-N4151 states Phase B/C Generator Differential is a cause for the generator unit trip.

Validation
5 people selected C Gen Undervoltage 19.5 Hz

Technical References:
GFIG-OPS-N4151, Tables 1 and 2
05-1-02-I-4, Loss of AC Power ONEP, Section 3.4 Grid Instability
04-1-02-1H13-P680, Alarm Response Instruction
EN-OP-115 Section 5.2

References to be provided to applicants during exam:
None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	2
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
295009 Low Reactor Water Level **Knowledge of the operational implications of the following concepts as they apply to low reactor water level:** **AK1.05: Natural circulation**	**Tier #**	1
	Group #	2
	K/A #	295009 AK1.05
	Rating	3.3

Question 21

The plant is scrammed due to a trip of both reactor recirculation pumps.

Reactor water level is being maintained at 85" on Shutdown Range.

Which of the following will be the effect on natural circulation flow rate if reactor water level is lowered below the steam separators?

A. Flow rate will significantly decrease due to the loss of communication between the core and the annulus.

B. Flow rate will decrease initially and then increase to a new thermal equilibrium value slightly less than the original flow rate due to less water (head).

C. Flow rate will not be significantly affected because they decrease in water (head) will be offset by the increase in thermal driving head.

D. Flow rate will not be significantly affected because the thermal driving head is primarily dependent on the differential temperature between the core and the annulus.

Answer: A
Explanation: A. Correct. Per FSAR 4.4.3.6, the natural circulation achieved a lower vessel levels "are minimums, it should be noted that the flow rates would be the lowest flow achieved." Therefore, as water level is lowered the natural circulation flow rate will also lower and not return to the original value. This is especially true once reactor water level is below the steam separators at a level of 82". This is also the reason for actions in the SCRAM ONEP to raise reactor water level above 82" to allow for maximum natural circulation. . B, C, & D are plausible because they are different explanations for changes in driving head due to volume of water or temperature. Per a constant minimum level, as temperature increases flow increases per the table on page 4.4-15a. As level is lowered, flow will reduce.
Technical References:

FSAR 4.4.3.6 05-1-02-I-1 Reactor Scram ONEP, section 3.10.5		
References to be provided to applicants during exam: **None**		
Learning Objective: GLP-OPS-MCD01 Obj 3.2		
Question Source:	Bank #	604
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2013
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(5)	

Examination Outline Cross-Reference	Level	RO
295011 High Containment Temperature **For High Containment Temperature:** **G 2.4.1 Knowledge of EOP entry conditions and immediate action steps.**	**Tier #**	1
	Group #	2
	K/A #	**295011: G 2.4.1**
	Rating	4.6

Question 22

Which of the following contains an EP entry condition and associated action?

A. Suppression pool temperature 94F. Operate all available suppression pool cooling.

B. RPV pressure 1050 psig. Stabilize RPV pressure with one or more RPV pressure control systems.

C. Drywell pressure 1.2 psig. Vent primary containment to control pressure.

D. Containment temperature 98F. Operate all available containment coolers.

Answer: D		
Explanation: A. Suppression pool temperature entry condition is above 95F. Immediate action is to operate all SP cooling. B. RPV Pressure entry condition is 1064.7 psig. Correct associated action. C. Drywell pressure entry condition is above 1.23 psig. Immediate action is to maintain CTMT pressure below 1.23 psig with CTMT purge. D. Per EP-3, containment temperature entry condition is above 95F. Immediate action is to operate all available CTMT cooling.		
Technical References: EP-2 & 3		
References to be provided to applicants during exam: None		
Learning Objective:		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	

	Comprehensive/Analysis	X
	LOD	2
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
295014 Inadvertent Reactivity Addition **Ability to operate and/or monitor the following as they apply to inadvertent reactivity addition:** **AA1.05: Neutron monitoring**	**Tier #**	1
	Group #	2
	K/A #	295014: AA1.05
	Rating	3.9

Question 23

A reactor normal startup is in progress.

Reactor Pressure is 935 psig with Bypass Control Valves in Operation.

A feedwater malfunction results in inadvertent reactivity addition.

Operators noted the following IRM indications:

All indications are rising

IRM	Range	Indication
A	7	30
B	7	35
C	8	40
D	7	25
E	8	35
F	8	40
G	7	35
H	8	40

What is the reactor operator's next action?

A. Range UP on IRMs A, B and G

B. Range UP on IRMs C, E, F and H

C. Reset the half scram

D. Place the Mode Switch to SHUTDOWN

Answer: A

Explanation:

IRMs are on Range 7 & 8, Range 7 scale is 0 - 40 Range 8 scale is 0 - 125.

- the range UP push button indication will illuminate at 75% of scale
- the scram signal is 120/125 of scale or 38.5/40 of scale.

Therefore, the following indications are present.

- A, B and G IRMs have an UP indication

NO other indications are present due to not exceeding any scram setpoints.

Operator action should be to range IRMs A, B and G to range 8 ONLY.

A is correct

B is wrong - IRMs C, E, F and H are on range 8 which would put them mid scale and no actions required

C is wrong - but, plausible if the student confuses the ranges where range 8 scale would be 0 - 40 not 0-125.

D is wrong - but, plausible if the student confuses the ranges and setpoints where C E and F H would be above the scram setpoint then the first action would be perform immediate actions for the scram onep.

Technical References:
03-1-01-1, Cold Shutdown to Generator Carrying Minimum Load IOI, Attachment VI, Step 12 of 12
04-1-01-c51-1, Neutron Monitoring SOI, Page 11, Ranging IRMs

References to be provided to applicants during exam:
None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(1)(5)(7)	

Examination Outline Cross-Reference	Level	RO
295015 Incomplete SCRAM **Knowledge of the reasons for the following responses as they apply to incomplete SCRAM:** **AK3.01: Bypassing rod insertion blocks**	**Tier #**	1
	Group #	2
	K/A #	295015: AK3.01
	Rating	3.4

Question 24

The reason for inserting Attachment 20 (Defeat RC& IS control rod drive blocks) during an ATWS is to allow control rod insertion irrespective of __________.

A. Control rod pattern.

B. Mode Switch Position.

C. Reactor Power being above the Low Power Setpoint.

D. Any Nuclear Instrumentation Upscale alarm.

Answer: A		
Explanation: Attachment 20 is used to defeat RC&IS control rod drive blocks and allow control rod insertion irrespective of main turbine 1st stage pressure or control rod pattern. A is correct B is wrong - Mode switch position will insert a withdrawal block not an insert block C is wrong - with reactor power being above the LPSP removes the control rod pattern controller from control. Being below the LPSP is where the Rod Pattern controller is in effect and will insert a withdrawal and Insert block in RC&IS. D is wrong - this will insert a withdrawal block not an insert block.		
Technical References: 05-S-01-EP-1		
References to be provided to applicants during exam: None		
Learning Objective:		
Question Source:	Bank #	

(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
295034 Secondary Containment Ventilation High Radiation **Knowledge of the interrelations between secondary containment ventilation high radiation and the following:** **EK2.04: Secondary containment ventilation**	**Tier #**	1
	Group #	2
	K/A #	295034 EK2.04
	Rating	3.9

Question 25

Which of the following will cause an automatic isolation of Secondary Containment?

A. Fuel Handling Area Exhaust rad monitor 'A' at 4.0 mR/hr <u>with</u>
Fuel Handling Area Exhaust rad monitor 'D' INOP trip

B. Fuel Pool Sweep Exhaust rad monitor 'A' at 35 mR/hr <u>with</u>
Fuel Pool Sweep Exhaust rad monitor 'C' at 40 mR/hr

C. Fuel Handling Area Exhaust rad monitor 'B' at 5.0 mR/hr <u>with</u>
Fuel Handling Area Exhaust rad monitor 'D' at 3.0 mR/hr

D. Fuel Pool Sweep Exhaust rad monitor 'B' at 25 mR/hr <u>with</u>
Fuel Pool Sweep Exhaust rad monitor 'C' INOP trip

Answer: A

Explanation:
Whether it be the Fuel Handling Area or the Fuel Pool Sweep Exhaust Vents, the only rad monitor combination that satisfies the required logic is the channels 'A' + 'D' combination, or channels 'B' + 'C'.

The Fuel Handling Area channel 'A' rad monitor is above the trip setpoint of 3.6 mR/hr <u>and</u> the channel 'D' rad monitor is providing the other half of the required logic with a valid INOP trip making the answer correct.

B is wrong because it suggests a wrong channel combination.

C is wrong because it shows the wrong coincidence combination (B + D) and because the Fuel Handling Area 'D' rad monitor being below the trip setpoint (for Secondary CTMT isolation) of 3.6 mR/hr. It is plausible because it represents two possible channel combinations.

D is wrong because although a proper channel combination exists, FPS channel 'B' is below the 30 mR/hr setpoint.

Technical References:
ARI Windows 04-1-02-1H13-P601-19A-B9 & B10
GLP-OPS-T4801
Auto Isolations ONEP

References to be provided to applicants during exam: **None**		
Learning Objective: GLP-OPS-T4801, Objective 8.6		
Question Source:	Bank #	349
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2011
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(11)	

Examination Outline Cross-Reference	Level	RO
295035 Secondary Containment High Differential Pressure **Ability to determine and/or interpret the following as they apply to secondary containment high differential pressure:** **EA2.01: Secondary containment pressure: Plant-specific**	**Tier #**	1
	Group #	2
	K/A #	295035 EA2.01
	Rating	3.8

Question 26

Auxiliary Building Ventilation (T41) and Fuel Handling Area Ventilation (T42) are in service when, per design, the Fuel Handling Area Ventilation Pressure Control Damper T42-F021 partially throttles closed (from its previous position).

What has this change in F021's position done within the secondary containment ventilation system, and what condition caused this F021 response?

A. Less secondary containment air flow is being supplied to the T42 exhaust fans. It re-positioned because the secondary containment d/p became less negative (closer to 0).

B. Less outside air flow is being supplied to the T42 supply fans. It re-positioned because the secondary containment d/p became more negative (further from 0).

C. Less secondary containment air flow is being supplied to the T42 exhaust fans. It re-positioned because the secondary containment d/p became more negative (further from 0).

D. Less outside air flow is being supplied to the T42 supply fans. It re-positioned because the secondary containment d/p became less negative (closer to 0).

Answer: D
Explanation: See simplified Figure 1 from training material GFIG-OPS-T4200 (or P&ID M-1104A). Pressure Control Damper F021 is on the inlet side of the T42 supply fans. When it throttles in the closed direction less outside air flow is being supplied to the T42 supply fans. There is no such damper associated with the T42 exhaust fans. Therefore the condition now is that the T42 exhaust fans are still exhausting Fuel Handling Area air to the outside at the same capacity as before the F021 position change. Thus, with less outside air coming in and the same inside air going out, the result is that the d/p becomes greater (i.e., more negative). F021 responded to a condition where, for whatever reason, its controller sensed a d/p that had become less negative.

At GGNS all these distracters have been proven to readily discriminate for one or both of two reasons: 1) many examinees cannot recall where the F021 damper is located in the T42 system (supply versus exhaust); 2) many examinees struggle with the concept of "more negative" versus "less negative" d/p. Therefore, these choices are plausible.		
Technical References: GFIG-OPS-T4200, T42 System Figures P&ID M-1104A, Fuel Handling Area Ventilation System		
References to be provided to applicants during exam: **None**		
Learning Objective: GLP-OPS-T4200 Obj 4,8		
Question Source:	Bank #	114
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2012
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(8)	

Examination Outline Cross-Reference	Level	RO
295036 Secondary Containment High Sump/Area Water Level **Knowledge of the interrelations between secondary containment high sump area water level and the following:** **EK2.03: Radwaste**	**Tier #**	1
	Group #	2
	K/A #	295036: EK2.03
	Rating	2.8

Question 27

A Fire Water Protection Deluge valve has ruptured in the Auxiliary Building causing the Auxiliary Building Floor Drain Transfer Tank to fill quickly.

Where can the contents of the Auxiliary Building Floor Drain Transfer Tank be sent to?

A. Floor Drain Stabilizing Sump and Oil Separator

B. Equipment Drain Collector Tank in Radwaste

C. Floor Drain Collector Tank in Radwaste

D. Waste Surge Tank

Answer: C
Explanation: When the level in sump reaches the Hi level setpoint, the A sump pump starts and pumps the waste from the sump. The applicant must remember the sump pumps are normally in standby and start on the Hi level setpoint. The pumps also have a hi hi level setpoint that actuates additional alarms. A. Plausible because Turbine Building Floor Drains can also be aligned to the Floor Drain Stabilizing Sump and Oil Separator. B. Plausible because Auxiliary Building Equipment Drain Transfer Tank is sent to the Equipment Drain Collector Tank in Radwaste. C. Outlet flow from the Auxiliary Building Floor Drain Transfer Tank may be sent to the Floor Drain Collector Tank in Radwaste and Suppression Pool. D. Plausible because Turbine Building Floor Drains are normally aligned to Waste Surge Tank because of chemistry. Validation Two people selected D
Technical References: GLP-OPS-P4500

GFIG-OPS-P4500 Figure1, 2 & 6		
References to be provided to applicants during exam: **None**		
Learning Objective: GLP-OPS-P4500, 7.1		
Question Source:	Bank #	1022
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	None
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	2
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
203000 RHR/LPCI: Injection Mode **Ability to predict an/or monitor changes in parameters associated with operating the RHR/LPCI: Injection Mode (Plant Specific) controls including:** **A1.01: Reactor water level**	**Tier #**	2
	Group #	1
	K/A #	203000 A1.01
	Rating	4.2

Question 28

The plant is operating at rated power when a LOCA occurs in the Drywell.

The RPV leak rate is approximately 1000 gpm.

RHR 'A' auto-initiates and is the <u>only</u> available source of makeup.

At which of the following points in the event timeline should the operator expect to see reactor water level start to recover (i.e., level stops lowering and begins to rise)?

When…

A. the RHR A ACTUATED annunciator is received

B. the LPCS/LPCI A INJ VLV RPV PRESS LO annunciator is received

C. reactor pressure lowers to about 450 psig

D. reactor pressure lowers to about 250 psig

Answer: D
Explanation: Per the RHR SOI (04-1-01-E12-1), section 5.4.1.c, the RHR pump shutoff head is about 285 psig; therefore. A is incorrect because this annunciator is received very early in the event timeline when a LOCA signal is generated (1.39 psig in drywell or -150.3" reactor water level); see ARI P601-20A-B5. Plausible if applicant determines actuated alarm also means injecting. B is incorrect because this annunciator is received when reactor pressure lowers to 476 psig, signifying the opening permissive point for the LPCI injection valve (see ARI P601-21A-F7). Plausible if applicant determines that the valve is open, therefore water is injecting. C is plausible because RHR A injection shutoff valve (E12-F042 can be manually opened from the control room only if LPCI line pressure is <450 psig and containment spray initiation signal is not

present. (Table 2 E12 lesson plan)		
Technical References: 04-1-01-E12-1, RHR System SOI, Section 5.4.1 ARI P601-20A-B5, annunciator window ARI P601-21A-F7, annunciator window 04-1-01-N19-1, Condensate System SOI, Section 4.1.2		
References to be provided to applicants during exam: **None**		
Learning Objective: GLP-OPS-E1200 obj 20		
Question Source:	Bank #	852
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2008
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(8)	

Examination Outline Cross-Reference	Level	RO
205000 Shutdown Cooling **Knowledge of shutdown cooling system (RHR shutdown cooling mode) design feature(s) and/or interlocks which provide for the following:** **K4.03: Low reactor water level: Plant-specific**	**Tier #**	2
	Group #	1
	K/A #	205000: K4.03
	Rating	3.8

Question 29

Which isolation signal will close ALL of the following valves?

E12-F008 RHR SHUTDN CLG OTBD SUCT VLV
E12-F023 RHR B FLO TO HD SPR
E12-F037A RHR A TO CTMT POOL
E12-F053A RHR A SHUTDN CLG RTN TO FW

A. High Drywell pressure 1.23 psig

B. High Reactor Vessel Pressure 135 psig

C. Low Reactor Vessel Water Level 11.4 inches

D. High RCIC Room Ambient Temperature 185F

Answer: C		
Explanation: A. High Drywell Pressure will only isolate the E12-F037A. B. High reactor vessel pressure will isolate all valves except for E12-F037A. C. This is the correct answer per Automatic Isolations ONEP. D. This is an isolation signal for a Group 4 (Steam Supply to RHR and RCIC) isolation. Plausible if the candidate confuses a group 4 signal with a group 3 signal.		
Technical References: 05-1-02-III-5, Automatic Isolations ONEP		
References to be provided to applicants during exam: None		
Learning Objective:		
Question Source:	Bank #	

(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	2
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
209001 LPCS **Knowledge of the physical connections an/or cause-effect relationships between low pressure core spray system and the following:** **K1.02: Torus/suppression pool**	**Tier #**	2
	Group #	1
	K/A #	209001 K1.02
	Rating	3.4

Question 30

Per SOI-04-1-01-E21, LPCS, Limits and Precautions the Suppression pool level listed where the LPCS pump Should NOT be run is ≤__________ feet, except during an emergency. This limit and precaution is to ensure adequate NPSH for the LPCS pump.

A. 10.5

B. 14.5

C. 18.34

D. 18.81

Answer: B		
Explanation: Answer B is correct per the LPCS SOI (04-1-01-E21). Per P&L 3.13 To ensure adequate NPSH, the LPCS pump Should NOT be run if suppression pool level is ≤14.5 feet, except during an emergency. A is wrong. Per EP-2 (RPV Control), this is the lowest SP level at which an emergency depressurization may be performed via the ADS/SRVs; it is plausible for this reason. C&D is wrong. This is the EP-3 (Containment Control) entry for SP low level/high level and is plausible for this reason.		
Technical References: LPCS SOI (04-1-01-E21).		
References to be provided to applicants during exam: **None**		
Learning Objective:		
Question Source:	Bank #	315
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2011

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	3
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
209002 HPCS **Ability to monitor automatic operations of the high pressure core spray system (HPCS) including:** **A3.03: System pressure: BWR-5,6**	**Tier #**	2
	Group #	1
	K/A #	209002 A3.03
	Rating	3.6

Question 31

High Pressure Core Spray (HPCS) is being started for its quarterly surveillance test.

When the HPCS Pump is started, HPCS MIN FLO TO SUPP POOL valve E22-F012 would be expected to automatically open if:

A. System flow is 1300 gpm and discharge pressure is 125 psig.

B. System flow is 1200 gpm and discharge pressure is 125 psig.

C. System flow is 1300 gpm and discharge pressure is 135 psig.

D. System flow is 1200 gpm and discharge pressure is 135 psig.

Answer: D		
Explanation: D is correct, E22-F012 opens if discharge pressure is above 130 psig and flow is below 1206 gpm, as stated in the designated answer. A, B, and C are incorrect and plausible because values are close to actual values required to open the E22-F012.		
Technical References: GLP-OPS-E2201		
References to be provided to applicants during exam: **None**		
Learning Objective: GLP-OPS-E2201 Obj 9.5		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	

	LOD	2
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
211000 SLC **Knowledge of the effect that a loss or malfunction of the following will have on the standby liquid control system:** **K6.03: A.C. power**	**Tier #**	2
	Group #	1
	K/A #	211000 K6.03
	Rating	3.2

Question 32

The plant is at rated power.

06-OP-1C41-Q-0001, Standby Liquid Control Functional Test, is being performed with SLC A pump running, circulating the SLC Test Tank.

A Loss of Offsite Power and ATWS occur.

Without operator action, what is the status of SLC?

A. SLC A and B running and injecting to the storage tank

B. SLC A and B running and circulating to the test tank

C. SLC A running and circulating to the test tank

D. SLC A running and injecting to the storage tank

Answer: C
Explanation: A. SLC A and B pumps are powered from 15 and 16 Bus MCC's. Power is automatically restored via Emergency Diesel Generators. Pump suction valves F001A and B are interlocked with the test tank outlet valve F031 such that, if F031 is not fully closed, the pump suction valve will not open automatically on a pump start signal. Plausible as the applicant may recognize that SLC is required due to the ATWS and believe that both pumps would be running and injecting. B. SLC A and B pumps are powered from 15 and 16 Bus MCC's. Plausible if applicant recalls pump suction valves F001A and B are interlocked with the test tank outlet valve F031 such that, if F031 is not fully closed, the pump suction valve will not open automatically on a pump start signal. C. Correct, There is no standing start signal for SLC B so when power is restored it will not restart and the suction to the storage tank F001A is interlocked with FO31 such that it will not automatically open on a pump start signal. D. Plausible if applicant recalls that SLC A is running prior to loss of power and applicant recalls storage tank automatically opens upon initiation and test tank closes.

Technical References: GLP-OPS-C4100		
References to be provided to applicants during exam: **None**		
Learning Objective: GLP-OPS-C4100, Obj 10		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(6)	

Examination Outline Cross-Reference	Level	RO
212000 RPS **Knowledge of the operational implications of the following concepts as they apply to reactor protection system:** **K5.02: Specific logic arrangements**	**Tier #**	2
	Group #	1
	K/A #	212000 K5.02
	Rating	3.3

Question 33

The plant is operating at 45% power.

An RPS Trip System 'A' half-scram is sealed in due to an I&C surveillance.

Due to simultaneous faulty sensor inputs, RPS trip logics see Turbine Stop Valves (TSVs) 'A' and 'C' go closed (The TSVs did not actually close).

Which of the following describes the operational implications?

	Tripped RPS Channel(s)	Resulting Scram Signal
A.	A ONLY	Division 1 Half Scram
B.	A & B ONLY	Full Scram
C.	A & C ONLY	Division 1 Half Scram
D.	A & D ONLY	Full Scram

Answer: D

Explanation:
See GFIG-OPS-C7100, TSV Scram Logic Simplified. TSVs 'A' and 'C' will trip RPS Channel 'D' logic, producing a ½ scram on RPS Trip System 'B'. With RPS Trip System 'A' already tripped for the I&C surveillance, and with power >40% (enabling the TSV Closure Scram logic), a full scram results; this requires the ONEP scram actions and EP-2 entry as a result of hitting +11.4" water level on the resulting shrink, post-scram.

A is plausible if A & C TSVs cause RPS channel A to trip. Correct scram signal based on tripped channels.

B is plausible if A & C TSVs cause RPS channel B to trip. Correct scram signal based on tripped channels.

C is plausible if A & C TSVs cause RPS channel C to trip. Correct scram signal based on tripped channels.

Validation Three people selected C (half scram)		
Technical References: GLP-OPS-C7100		
References to be provided to applicants during exam: **None**		
Learning Objective:		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	970
	New	
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	2
10CFR Part 55 Content:	55.41(b)(6)	

Examination Outline Cross-Reference	Level	RO
215003 IRM **For IRM system:** **G2.1.32: Ability to explain and apply system limits and precautions.**	**Tier #**	2
	Group #	1
	K/A #	215003: G2.1.32
	Rating	3.8

Question 34

According to 04-1-01-C51-1, Neutron Monitoring, when driving IRM detectors for maintenance, troubleshooting or surveillance activities, drive only one detector or one division of detectors at a time.

What is the basis for this?

A. Reduce the potential for a full scram.

B. Reduce the possibility of detector cable entanglement.

C. Avoid potential overload of the detector motor power monitors.

D. Ensure Tech Spec operability requirements for IRMs are maintained during the activity.

Answer: A
Explanation: A. P&L 3.7 states potential for converter failure and resulting half scram exists when driving IRMs. One division should be driven at a time in case a half scram results from the division being driven. This is to avoid simultaneous half scrams in both RPS divisions. B. Plausible because there are several detector cables located undervessel with limited space. C. Plausible if the electrical circuit had power limitations within the capabilities of just one drive motor. D. Plausible if while moving the IRM detectors they were declared Inoperable per tech specs. Validation One person selected C
Technical References: 04-1-01-C51-1, Neutron Monitoring
References to be provided to applicants during exam: **None**
Learning Objective: GLP-OPS-C5102, Obj 12

Question Source:	Bank #	GLP-OPS-08183
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
215004 Source Range Monitor **Knowledge of the physical connections an/or cause-effect relationships between source range monitor (SRM) system and the following:** **K1.06: Reactor vessel**	**Tier #**	2
	Group #	1
	K/A #	215004: K1.06
	Rating	3.4

Question 35

During a reactor startup SRM detectors are:

A. maintained in the fixed location in core for the life of the core.

B. maintained in the fixed location in core until SRM/APRM overlaps are completed.

C. withdrawn from the core maintaining 10^2 to 10^5 cps.

D. withdrawn from the core when any single IRM is on range 1.

Answer: C		
Explanation: A. Plausible because this describes APRM operation. B. Plausible because this describes IRM operation. C. During a reactor startup, SRM detectors are gradually withdrawn from the core. This withdrawal causes the period meter to move in the negative direction. D. Plausible because SRMs are permitted to be withdrawn when both of the IRMs associated with the SRM are on range 3 or above.. Validation Three people selected D. gradually withdrawn from the core maintaining a constant indicated period.		
Technical References: 03-1-01-1, IOI-1 Cold Shutdown to Generator Carrying Minimum Load GLP-OPS-C5101		
References to be provided to applicants during exam: **None**		
Learning Objective: GLP-OPS-C5101 Obj 4		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	X

	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(6)	

Examination Outline Cross-Reference	Level	RO
215004 Source Range Monitor	**Tier #**	2
	Group #	1
Ability to monitor automatic operations of the source range monitor system including:	**K/A #**	215004: A3.04
	Rating	3.6
A3.04: Control rod block status		

Question 36

A reactor startup is in progress following a mid-cycle scram from rated power.

SRM B fails full upscale.

How does SRM Channel 'B' respond with respect to RC&IS and RPS?

A. Generates a rod block, only.

B. Generates a rod block and causes a half scram.

C. Causes a half scram, only.

D. Does not generate a rod block and does not cause a half scram.

Answer: A
Explanation: A. Stem conditions defining this reactor startup as following a mid-cycle scram ensures there is no doubt about the fact that all RPS shorting links are in fact installed. As such, when this trip de-energizes (i.e., its "normal" fail-safe mode of operation, which the Applicant is expected to recognize), the trip unit generates an "upscale" flux trip signal (i.e., above the 2x10E5 cps scram setpoint). However, with the RPS shorting links installed (an RPS system configuration), no half scram occurs. However, the rod block setpoint is at 1x10E5 cps; that setpoint has been exceeded. B. Plausible if the shorting links were removed. C. Plausible if the shorting links were removed. D. Plausible if the rod block setpoint was higher.
Technical References: GLP-OPS-C5101, SRM lesson plan GFIG-OPS-C5101
References to be provided to applicants during exam: **None**
Learning Objective: GLP-OPS-C5101, Objective 3.10

Question Source:	Bank #	316
(note changes; attach parent)	Modified Bank #	

	New	
Question History:	Last NRC Exam	2011
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	2
10CFR Part 55 Content:	55.41(b)(6)	

Examination Outline Cross-Reference	Level	RO
215005 APRM / LPRM **Ability to predict and/or monitor changes in parameters associated with operating the average power range monitor/local power range monitor system:** **A1.05: Lights and alarms**	**Tier #**	2
	Group #	1
	K/A #	215005: A1.06
	Rating	3.3

Question 37

Plant is operating at 100% power.

'B' Main Feed Pump trips.

The following annunciators are in alarm:
APRM CH 1 UPSC TRIP/OPRM TRIP/INOP
APRM CH 3 UPSC TRIP/OPRM TRIP/INOP

The following computer points are in alarm:
C51NC065 APRM 1 OPRM TRIP
C51NC067 APRM 3 OPRM TRIP

What operator action is required FIRST based on these indications?

A. Place the reactor mode switch in the shutdown position.

B. Verify half scram indications. Secure maintenance to prevent full scram.

C. Enter Tech Spec 3.3.1.1, Reactor Protection System (RPS) Instrumentation for two channels of APRMs INOP.

D. Verify channels 2 and 4 OPRMs are operable.

Answer: A
Explanation: A. When the B MFPT trips, the Recirc FCVs will runback and OPRM Enabled Region will be entered. If THI is detected then immediately place the reactor mode switch in the shutdown position. THI is detected by the two alarms and corresponding computer points in alarm. B. Plausible if applicant recalls RPS logic to Channels A and C will cause half scram on RPS A. C. Plausible because Table 3.3.1.1, 3 APRM channels must be operable. D. Plausible because step 3.4.1 Reduction Recirc ONEP, states to Verify that at least 3 OPRM

channels are not bypassed. Validation One person selected B.		
Technical References: 05-1-02-III-3, Reduction in Recirculation System Flow Rate ONEP TS 3.3.1.1		
References to be provided to applicants during exam: **None**		
Learning Objective:		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(6)	

Examination Outline Cross-Reference	Level	RO
217000 RCIC **Knowledge of the effect that a loss or malfunction of the reactor core isolation cooling system (RCIC) will have on the following:** **K3.03: Decay heat removal**	**Tier #**	2
	Group #	1
	K/A #	217000: K3.03
	Rating	3.5

Question 38

(1) What are the MINIMUM ESF busses lost during a station blackout?

(2) Which of the following system malfunctions during a station blackout will have the biggest impact on the plant's ability for decay heat removal?

	(1)	(2)
A.	ALL	RCIC
B.	15AA & 16AB ONLY	RCIC
C.	ALL	LPCS
D.	15AA & 16AB ONLY	LPCS

Answer: B
Explanation: A. Plausible if student confuses station blackout to include ALL ECCS buses lost. RCIC is the only system on the list that does not require AC power to operate. B. Per GGNS UFSAR, 5.4.6.1.3, The RCIC system could be operated if all AC power is lost. Per 05-1-02-I-4, Los of AC Power ONEP, Section 3.2 Loss of Div 1 AND Div 2 ESF Buses is a Station Blackout. This means 15AA and 16AB are not able to be re-energized. RCIC is the only system on this list that does not require AC power to operate. C. Plausible if student confuses station blackout to include ALL ECCS buses lost. LPCS is plausible because by definition of station blackout 17AC bus may still be available. HPCS is powered from 17AC and if student confuses LPCS with HPCS this answer may be selected. D. Per 05-1-02-I-4, Los of AC Power ONEP, Section 3.2 Loss of Div 1 AND Div 2 ESF Buses is a Station Blackout. This means 15AA and 16AB are not able to be re-energized. . LPCS is plausible because by definition of station blackout 17AC bus may still be available. HPCS is powered from 17AC and if student confuses LPCS with HPCS this answer may be selected. Validation One person selected A LPCS. This was previous revision of question that asked part 2 only.
Technical References: GGNS UFSAR, 5.4.6.1.3

05-1-02-I-4, Los of AC Power ONEP		
References to be provided to applicants during exam: None		
Learning Objective:		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	2
10CFR Part 55 Content:	55.41(b)(8)	

<table>
<tr><td>Examination Outline Cross-Reference</td><td>Level</td><td>RO</td></tr>
<tr><td rowspan="6">217000 RCIC

Knowledge of the operational implications of the following concepts as they apply to reactor core isolation cooling system (RCIC):

K5.02: Flow indication</td><td>Tier #</td><td>2</td></tr>
<tr><td>Group #</td><td>1</td></tr>
<tr><td>K/A #</td><td>217000: K5.02</td></tr>
<tr><td>Rating</td><td>3.1</td></tr>
<tr><td></td><td></td></tr>
</table>

Question 39

The plant has scrammed due to a loss of feedwater.

Control Room evacuation is required due to security threat.

HPCS is not available.

The Rover is directed to establish reactor water level control with RCIC at Remote Shutdown Panel P150.

Which of the following describes the Rover's operation of RCIC at P150?

A. The only way to prevent over-fill as RCIC makes up to the RPV is by the Rover closing E51-F045 RCIC Turbine Steam Supply at the panel.

B. The only way to determine RCIC flow at the panel is by the Rover nulling out the flow controller and reading the flow off the controller vertical tape.

C. The Rover will have to manually swap the RCIC suction over to the suppression pool if it becomes necessary.

D. The local building operator will have to keep the Rover aware of RCIC turbine speed; if a trip is necessary because of inadequate speed, closing the Trip/Throttle Valve is the only method available.

Answer: B
Explanation: P150 has no dedicated RCIC flow indicator (such as exists in the control room at P601). Only by nulling out the P150 RCIC flow controller can the Rover then read flow off the controller vertical tape. A & C are wrong. These two choices suggest facts that would be true <u>only if</u> operators had evacuated the control room due to a control room fire or security threat <u>and</u> Attachment III of the Remote Shutdown ONEP had already been performed (see Remote Shutdown ONEP, section 3.5.1 and Attachment V, page 1 of 2 (CAUTION). The CAUTION shows that when Att. III is performed RCIC automatic functions are disabled, including the Level 8 turbine shutdown (i.e., auto-closure of F045 and auto-swap of pump suction to supp pool).

D is wrong. Although it is true that the only way to trip RCIC at P150 is by closing the T/T Valve (i.e., the panel has no Trip pushbutton), there is in fact Turbine Speed indication at the panel. There is no need for the local operator to keep the Rover informed of speed. Validation One person selected A		
Technical References: 05-1-02-II-1, Remote Shutdown Panel ONEP		
References to be provided to applicants during exam: **None**		
Learning Objective: GLP-OPS-C6100, OBJ. 6		
Question Source:	Bank #	476
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2010
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	2
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
218000 ADS Knowledge of electrical power supplies to the following: K2.01: ADS logic	Tier #	2
	Group #	1
	K/A #	218000: K2.01
	Rating	3.1

Question 40

The Div 1 ADS logic is powered from…

A. RPS Bus 'A'

B. Inverter 1Y87

C. Distribution panel 1DA1

D. Power panel 15P61

Answer: C		
Explanation: A. RPS Bus 'A' supplies 120 VAC, not 125 DC. Plausible to the Applicant who cannot recall that ADS logic is DC powered. B. This is the Div 1 inverter which supplies 120 VAC, not 125 DC. Plausible for the same reason as choice 'A'. C. ADS logic is DC powered from the Div 1 DC subsystem (11DA) via its distribution panel 1DA1, breaker 72-11A23 (see E-1161-004 and -005, also E-1023). D. This is one of the 120 VAC power panels fed from the Div 1 vital bus 15AA via an MCC. Plausible for the same reason as choice 'A'. Validation One person selected D.		
Technical References: E-1023, One Line for 125 VDC Buses 11DA, 11DB & 11DC E-1161-004, ADS Power Distribution E-1161-005, ADS Relay Logics GLP-OPS-E2202, ADS LESSON PLAN		
References to be provided to applicants during exam: **None**		
Learning Objective: GLP-OPS-E2202 Obj 19.3		
Question Source:	Bank #	3
(note changes; attach parent)	Modified Bank #	

	New	
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
223002 PCIS/Nuclear Steam Supply Shutoff **Ability to manually operate and/or monitor in the control room:** **A4.01: Valve closures**	**Tier #**	2
	Group #	1
	K/A #	223002 A4.01
	Rating	3.6

Question 41

Depressing the CTMT-DRWL ISOL DIV 1(2) MAN INIT isolation pushbuttons on P870 will initiate Group _____ valve isolations.

A. 2

B. 7

C. 8

D. 10

Answer: B
Explanation: Only the Group 7 valves isolate when these pushbuttons are depressed. The list of the valves that close when these pushbuttons are depressed is found in the M71 system SOI, page 6, Section 5.1.2. Compare this list to the Containment Cooling Group 7 list found in the "Automatic Isolations" ONEP (05-1-02-III-5) to validate that only Group 7 valves will close. The plausibility of the Distracters is that each of these groups of NSSSS isolations contains penetrations/valves within the Drywell and/or Containment. Validation One person selected Group 8.
Technical References: CTMT and Drywell Instrumentation and Control System SOI, 04-1-01-M71-1 Automatic Isolations ONEP, 05-1-02-III-5
References to be provided to applicants during exam:
Learning Objective: GLP-OPS-M7101 Obj 8.2

Question Source:	Bank #	94
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2012

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
239002 SRVs **Knowledge of the effect that a loss or malfunction of the following will have on the relief/safety valves:** **K6.02: Air (Nitrogen) supply: Plant-Specific**	**Tier #**	2
	Group #	1
	K/A #	239002 K6.02
	Rating	3.4

Question 42

The plant has been in a Station Blackout for three hours.

All Safety Relief Valve (SRV) accumulator air pressures are zero psig.

What relief/safety mode is available?

A. ADS Relief mode only.

B. Safety mode only.

C. Safety and ADS Relief modes only.

D. Safety and Low-Low Set Relief modes only

Answer: B		
Explanation: B is correct. In safety mode, SRV operation is as described in answer B**.** No air pressure or DC is required. DC power is available. A and C are plausible since the ADS Relief logic would actuate and the SRV solenoids would energize, but with no air pressure for motive force, the SRV would not open. D is plausible since the Low-Low Set Relief logic would actuate and the SRV solenoids would energize, but with no air pressure for motive force, the SRV would not open.		
Technical References: GLP-OPS-E2202		
References to be provided to applicants during exam: **None**		
Learning Objective: GLP-OPS-E2202 Obj. 7		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(3)	

Examination Outline Cross-Reference	Level	RO
239002 SRVs **Ability to (a) predict the impacts of the following on the relief/safety valves; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** **A2.04: ADS actuation**	**Tier #**	2
	Group #	1
	K/A #	239002: A2.04
	Rating	4.1

Question 43

The plant is operating at 100% power.

An inadvertent Division 1ECCS initiation occurs due to failed downscale level transmitter.

Ten minutes later, a FW transient occurs and causes NR RWL to drop to 10 inches.

RWL is restored >12 inches 2 minutes after transient.

With no operator action, what is the status of ADS two minutes after the FW transient and what is the ***MINIMUM*** action required to mitigate the consequences?

A. ADS actuation has occurred.
The Division 1 ADS logic must be reset and the respective ADS MANUAL INHIBIT keylock switch placed in the INHIBIT position.

B. ADS actuation has occurred.
The Division 1 ADS logic must be reset.

C. ADS actuation has not completed time delays.
The Division 1 ADS logic must be reset and the respective ADS MANUAL INHIBIT keylock switch placed in the INHIBIT position.

D. ADS actuation has not completed time delays.
The Division 1 ADS logic must be reset.

Answer: B
Explanation: A. Actuation has occurred. Actuation requirements met are <-150.3 inches for 9.2 minutes, <+11.4

inches level confirmation, ADS in NORM, 105 sec TDE, and Division 1 ECCS pump running to raise discharge pressure above setpoint. Per GLP-OPS-E2202, If desired to close the ADS valves with all required signals present and without having to repeatedly reset the 105 second timer, depress the reset pushbuttons and place the respective ADS MANUAL INHIBIT keylock switches in the INHIBIT position. Both of these actions are not required to mitigate the spurious actuation. B. Actuation requirements are met. Per B21 SOI to reset an ADS initiation, Depress both A and B logic reset pushbuttons on P601. Check the valves closed and logic resets. C. Student may have confused the timer sequence and believed the ADS actuation sequence has not completed its time delays. The actions will mitigate actuation but are not the minimum actions. D. Student may have confused the timer sequence and believed the ADS actuation sequence has not completed its time delays. This is the minimum action required to minimize the actuation. Validation One person selected D.
Technical References: GLP-OPS-E2202 GFIG-OPS-E2202 04-1-01-B21-1, Section 5.2, Resetting an ADS initiation
References to be provided to applicants during exam: None.
Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(8)	

Examination Outline Cross-Reference	Level	RO
259002 Reactor Water Level Control **Ability to predict and/or monitor changes in parameters associated with operating the reactor water level control system controls including:** **A1.02: Reactor feedwater flow**	**Tier #**	2
	Group #	1
	K/A #	259002 A1.02
	Rating	3.6

Question 44

The reactor is operating at rated power.

The C34 Feedwater Flow transmitter fails full upscale to 12.75 mlbm/hr.

What effects will directly result from this condition?

A. The feedwater system will switch to MANUAL level control.

B. The feedwater system will disable three-element control.

C. An "estimated" feedwater flow signal replaces the failure transmitter with no change in control.

D. The feedwater system will automatically lower feedwater flow.

Answer: B
Explanation: "Hard-failures" of feedwater flow signals are responded to by immediately de-selecting and disabling three-element control. A. Plausible because if all four level channels hard fail the level control stations will automatically swap to MANUAL. B. Correct. C. Plausible because an estimated feedwater flow can replace the normal total feedwater flow signal and three-element control can be re-selected and returned to service. D. Plausible if the upscale feed flow signal is sensed and used as a good input to the level control system. High feed flow the system would attempt to reduce feed flow to anticipate a rise in level due to feed flow steam flow mismatch. Validation One person selected C.
Technical References: GLP-OPS-C3400, DFCS lesson
References to be provided to applicants during exam: **None**

Learning Objective: GLP-OPS-C3400 Obj 3		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	3
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
259002 Reactor Water Level Control **Ability to manually operate and/or monitor in the control room:** **A4.11: High level lockout reset controls: Plant-Specific**	**Tier #**	2
	Group #	1
	K/A #	259002 A4.11
	Rating	3.5

Question 45

Post-scram high level control problems have resulted in a trip of both RFPTs.

Currently at P680:

- RFPT A TRIP annunciator is sealed-in
- RX LVL 40"/32" HI/LO annunciator is sealed-in
- RFPT/MN TURB LVL 9 TRIP annunciator is sealed-in

To reset the trip on RFPT 'A' the operator must wait for the…

A. RFPT/MN TURB LVL 9 TRIP annunciator to clear, then depress the RFPT A TRIP RESET pushbutton.

B. RX LVL 40"/32" HI/LO annunciator to clear, then depress the RFPT A TRIP RESET pushbutton.

C. RX LVL 40"/32" HI/LO annunciator to clear, then depress the LEVEL A, B, C SELECTED/DISABLED pushbuttons.

D. RFPT/MN TURB LVL 9 TRIP annunciator to clear, then depress the LEVEL A, B, C SELECTED/DISABLED pushbuttons.

Answer: A
Explanation: The RFPT/MN TURB LVL 9 TRIP annunciator clearing is indicative of the Level 9 trip signal clearing, at which point the RFPT trip can be reset as soon as the operator depresses the RFPT A TRIP RESET pushbutton. B is wrong because it suggests that the operator must "Wait" until water level lowers to the point of clearing the RX LVL 40"/32" HI/LO annunciator before being able to reset the RFPT trip by depressing the RFPT A TRIP RESET pushbutton. This is not true. C & D are wrong because the Level A, B, C, Selected/Disabled pushbuttons are used to manually select the level instruments used by the Digital Feed Control System.
Technical References:

ARIs P680-2A-A2, 3A-A3, and 4A2-D1 GLP-OPS-N2100, page 21 04-1-01-N21-1		
References to be provided to applicants during exam: **None**		
Learning Objective: GLP-OPS-N2100, OBJ. 13.0		
Question Source:	Bank #	490
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2010
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	2
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
261000 SGTS **Knowledge of the effect that a loss or malfunction of the standby gas treatment system will have on the following:** **K3.01: Secondary containment and environment differential pressure**	**Tier #**	2
	Group #	1
	K/A #	261000 K3.01
	Rating	3.3

Question 46

The plant is operating at rated power when both RFPTs trip.

Both SGTS trains auto-initiate.

Reactor water level is being restored by HPCS and RCIC.

Operators place SGTS 'A' in STANDBY with its Mode Select Switch at P870 and secures SGTS 'A'.

A malfunction with SGTS 'B' causes Enclosure Building negative pressure to degrade to __________ setpoint which will auto-start SGTS 'A'?

A. -0.88" w.c

B. -0.75" w.c

C. -0.25" w.c

D. -0.2" w.c.

Answer: D
Explanation: When a SBGT system is placed in standby all initiation signals from reactor water level, hi drywell pressure and radiation are bypassed. The standby fan will only start on low Enclosure building recirc fan flow <8500scfm, low exhaust filter train flow <1250 scfm, or enclosure building low negative pressure -0.2" wc. A. Plausible because one of the requirements to open the Post-LOCA vacuum relief valves is Drywell pressure is ≤-0.88 psid less than Containment pressure. (GLP-OPS-E6100) B. Plausible because upon initiation SBGT system the electro-hydraulically actuated flow control vanes begin to modulate if enclosure Building pressure is drawn down to -0.75" wc. (GLP-OPS-T4800) C. Plausible because the SBGT system initial draw down is to -0.25" wc. (GLP-OPS-T4800) D. Correct
Technical References:

GLP-OPS-T4800 GLP-OPS-E6100		
References to be provided to applicants during exam: **None**		
Learning Objective: GLP-OPS-T4800, Objective 8.6 & 8.7		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	3
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
262001 AC Electrical Distribution For AC electrical distribution system: G2.1.19: Ability to use plant computer to evaluate system or component status.	Tier #	2
	Group #	1
	K/A #	
	Rating	3.9

Question 47

While monitoring N19 Condensate System on PDS, an electrical malfunction occurs.

Based on the provided PDS indications, which BOP transformer experienced an electrical malfunction?

A. 12A

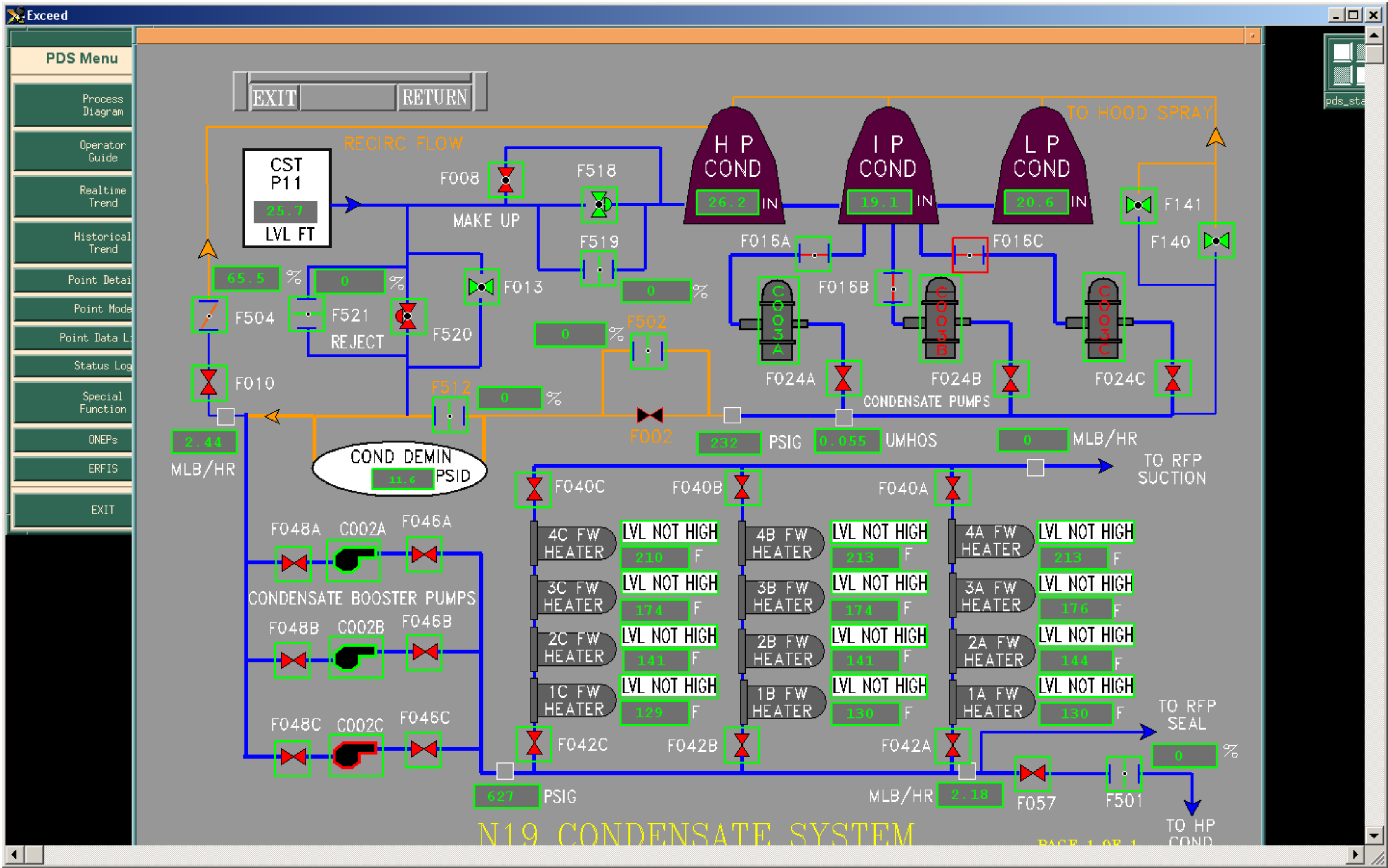
B. 12B

C. 11A

D. 11B

Answer: C		
Explanation: Based on the indications Condensate Pump B and C and Condensate Booster Pump C are running. These are powered from BOP Transformer 12B via 14AE. The loss of the other pumps indicates there was a loss of BOP Transformer 11A via 13AD. A. Plausible if pumps lost were powered from 14AE. B. Plausible if pumps lost were powered from 11HD. D. Plausible if pumps lost were powered from 12HE.		
Technical References: GLP-OPS-N1900 R2700 Figure 6		
References to be provided to applicants during exam: **PDS Display Printout**		
Learning Objective: GLP-OPS-N1900, Obj 10		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	2
10CFR Part 55 Content:	55.41(b)(7)	



Exceed
PDS Menu
Process Diagram
Operator Guide
Realtime Trend
Historical Trend
Point Detai
Point Mode
Point Data L
Status Log
Special Function
ONEPs
ERFIS
EXIT
RETURN
CST P11
25.7
LVL FT
RECIRC FLOW
H P COND
26.2 IN
I P COND
19.1 IN
L P COND
20.6 IN
TO HOOD SPRAY
F008
F518
MAKE UP
F519
F013
F504
F521
REJECT
F520
F502
F010
F512
COND DEMIN
11.6 PSID
F002
2.44 MLB/HR
232 PSIG
0.055 UMHOS
0 MLB/HR
TO RFP SUCTION
F016A
F016B
F016C
F024A
F024B
F024C
CONDENSATE PUMPS
F141
F140
F040C
F040B
F040A
F048A
C002A
F046A
CONDENSATE BOOSTER PUMPS
F048B
C002B
F046B
F048C
C002C
F046C
4C FW HEATER
LVL NOT HIGH
210 F
3C FW HEATER
174 F
2C FW HEATER
141 F
1C FW HEATER
129 F
4B FW HEATER
213 F
3B FW HEATER
174 F
2B FW HEATER
141 F
1B FW HEATER
130 F
4A FW HEATER
213 F
3A FW HEATER
176 F
2A FW HEATER
144 F
1A FW HEATER
130 F
F042C
F042B
F042A
627 PSIG
MLB/HR 2.18
TO RFP SEAL
F057
F501
TO HP COND
N19 CONDENSATE SYSTEM

Examination Outline Cross-Reference	Level	RO
262002 UPS (AC/DC) **Ability to (a) predict the impacts of the following on the uninterruptable power supply (AC/DC); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** **A2.01: Under voltage**	**Tier #**	2
	Group #	1
	K/A #	262002: A2.01
	Rating	2.6

Question 48

DC bus 11DB has been lost.

Which of the following describes the actions on the UPS inverter system?

Upon restoration of 11DB to the normal power supply, using 04-1-01-L62-1, Static Inverters SOI, what action should the operator perform to ensure return to normal operation?

A. Inverters 1Y88 and 1Y95 swap to alternate source.
Verify 1Y88 and 1Y95 automatically swap to Normal Supply

B. Inverters 1Y87 and 1Y96 swap to alternate source
Depress INVERTER TO LOAD pushbutton on both Inverters

C. Inverters 1Y88 and 1Y95 swap to alternate source
Depress INVERTER TO LOAD pushbutton on both Inverters

D. Inverters 1Y87 and 1Y96 swap to alternate source
Verify 1Y87 and 1Y96 automatically swap to Normal Supply

Answer: C

Explanation:

A loss of 11DB bus will cause a loss of Static inverters 1Y88 and 1Y95 on undervoltage and cause them to swap to alternate source.

When DC power is restored these inverters, being ESF, will not auto transfer back to normal supply. An operator is required to depress Inverter to load pushbutton per SOI 04-1-01-l62-1, Static Inverters.

A is wrong - these inverters (ESF) will not auto swap back to normal supply

B is wrong - 1Y87 and 1Y96 are powered from 11DA bus not 11DB.

C is correct. D is wrong - 1Y87 and 1Y96 are powered from 11DA bus not 11DB and these inverters (ESF) will not auto swap back to normal supply		
Technical References: **GLP-OPS-L62 Rev 13** **04-1-01-L62-1 Section 5.2.2**		
References to be provided to applicants during exam: **None**		
Learning Objective: **GLP-OPS-L62 Objective 4**		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
263000 DC Electrical Distribution **Knowledge of electrical power supplies to the following:** **K2.01: Major D.C. loads**	**Tier #**	2
	Group #	1
	K/A #	263000 K2.01
	Rating	3.1

Question 49

The plant is operating at rated conditions.

The following alarms are received:

- P680-9A-D11, GEN SEAL OIL TROUBLE
- P680-10A-C10, GEN H2 SEAL OIL PUMP C FAULT

Which of the following is the DC Electrical bus that was lost?

A. 11DB

B. 11DD

C. 11DE

D. 11DF

Answer: D
Explanation: See 04-1-01-1-L11 Attachment 1F shows the power supply for the DC SEAL OIL PUMP C MOTOR. A, B, and C are plausible because they are other divisional DC power sources. Validation One person selected C
Technical References: 04-1-01-L11-1, Attachment 1F, 125V DC BUS 11DF LOAD LIST
References to be provided to applicants during exam: **None**
Learning Objective: GLP-OPS-L1100 Obj 6.2, 8.3

Question Source:	Bank #	86
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	2
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
264000 EDGs	**Tier #**	2
	Group #	1
Ability to monitor automatic operations of the emergency generators (diesel/jet) including:	**K/A #**	264000: A3.06
	Rating	3.1
A3.06: Cooling water system operation		

Question 50

Operators have manually initiated Division 1 ECCS logic from P601 with the LPCS/RHR A MAN INIT arm and depress pushbutton.

What is the status of the diesel generator and SSW A?

A. Division 1 diesel generator is running and NOT supplying power to 15AA.
SSW A is automatically aligned for RHR A component start.

B. Division 1 diesel generator is running and NOT supplying power to 15AA.
SSW A is aligned for a LOCA start.

C. Division 1 diesel generator is running and supplying power to 15AA.
SSW A is automatically aligned for RHR A component start.

D. Division 1 diesel generator is running and supplying power to 15AA.
SSW A is aligned for a LOCA start.

Answer: B
Explanation: A. The RHR A manual initiate pushbuttons will simulate a division 1 ECCS LOCA signal. This will start the diesel generator automatically. There is no loss of power, so the diesel will not supply power to 15AA. When RHR A is manually started per SOI component by component a component start is possible; however, the pushbuttons generate a LOCA signal. B. The RHR A manual initiate pushbuttons will simulate a division 1 ECCS LOCA signal. This will start the diesel generator automatically. There is no loss of power, so the diesel will not supply power to 15AA. The RHR manual initiate pushbuttons generate a LOCA signal; therefore, SSW automatically aligns for a LOCA start. C. The RHR A manual initiate pushbuttons will simulate a division 1 ECCS LOCA signal. This will start the diesel generator automatically. The student may confuse this with a LOP/LOCA signal which will shed the bus and the diesel will automatically start and supply power to the bus. When RHR A is manually started per SOI component by component a component start is possible; however, the pushbuttons generate a LOCA signal. D. The RHR A manual initiate pushbuttons will simulate a division 1 ECCS LOCA signal. This will

start the diesel generator automatically. The student may confuse this with a LOP/LOCA signal which will shed the bus and the diesel will automatically start and supply power to the bus. The RHR manual initiate pushbuttons generate a LOCA signal; therefore, SSW automatically aligns for a LOCA start.		
Technical References: GLP-OPS-P4100 GLP-OPS-E1200 GLP-OPS-P7500		
References to be provided to applicants during exam: **None**		
Learning Objective:		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	2
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
300000 Instrument Air	**Tier #**	2
Knowledge of instrument air system design feature(s) and or interlocks which provide for the following:	**Group #**	1
	K/A #	300000 K4.02
	Rating	3.0
K4.02: Cross-over to other air systems		

Question 51

The INFI-90 Controller output for P53-PV-F503 (IA Pressure Reducing Station) has failed.

What describes the method of pressure control for the IA Pressure Reducing Station for this condition?

P53-PV-504 (IA Backup Pressure Regulator) will maintain Instrument Air pressure at ____(1)___ psig from ___(2)___.

	(1)	(2)
A.	100	service air system
B.	90	service air system.
C.	100	plant air compressors.
D.	90	plant air compressors.

Answer: C

Explanation:
If the INFI-90 Controller fails, the backup regulator PV-F504 will maintain pressure at 100 psig. Plant air compressors supply air to instrument air system

A. Plausible because the backup valve is set to maintain 100 psig. Service air system is plausible if student remembers simplified drawing because the service air pressure regulator is in parallel with the instrument air backup pressure regulator.

B. Plausible because the standby Plant Air Compressor will start if Service Air Pressure reaches it set point of 90 psig for 40 seconds. Service air system is plausible if student remembers simplified drawing because the service air pressure regulator is in parallel with the instrument air backup pressure regulator.

C. Correct

D. Plausible because the standby Plant Air Compressor will start if Service Air Pressure reaches it set point of 90 psig for 40 seconds. Plant air compressors do supply the normal and backup air supplies.

Technical References: **GLP-OPS-P5100** GFIG-OPS-P5100		
References to be provided to applicants during exam: **None**		
Learning Objective: GLP-OPS-P5100 Obj 18		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	120
	New	
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	3
10CFR Part 55 Content:	55.41(b)(4)	

Examination Outline Cross-Reference	Level	RO
400000 Component Cooling Water **Knowledge of the physical connections and/or cause-effect relationships between CCWS and the following:** **K1.02: Loads cooled by CCWS**	**Tier #**	2
	Group #	1
	K/A #	400000: K1.02
	Rating	3.2

Question 52

CCW Temperature Control Valve failed closed, stopping ___(a)___ the NORMAL cooling water flow though the CCW heat exchanger.

___(b)___ will automatically isolate first because of the increased CCW TEMPERATURES.

	(a)	(b)
A.	SSW	FPCCU Heat Exchangers
B.	SSW	RWCU system
C.	PSW	FPCCU Heat Exchangers
D.	PSW	RWCU system

Answer: D

Explanation:
Loss of CCW ONEP, Section 5.0 Automatic Actions states Fuel pool heat Exchangers isolate on low flow. RWCU system isolates on Demin Inlet high temperature.

A. Plausible because SSW is the backup cooling water supply to the CCW heat exchangers. The FPCCU Heat Exchangers will isolate quickly due to flow not temperature.
B. Plausible because SSW is the backup cooling water supply to the CCW heat exchangers. The RWCU system will isolate on demin inlet high temperature.
C. Plausible because PSW is the normal cooling water supply. The FPCCU Heat Exchangers will isolate quickly due to flow not temperature.
D. Correct answer. PSW is the normal cooling water supply. The RWCU system will isolate on demin inlet high temperature.

Technical References:
05-1-02-V-1, Loss of CCW ONEP

References to be provided to applicants during exam:
None

Learning Objective: GLP-OPS-ONEP, OBJ. 2.0

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
400000 Component Cooling Water **For the component cooling water system:** **G2.4.11 Knowledge of abnormal condition procedures.**	**Tier #**	2
	Group #	1
	K/A #	400000 G2.4.11
	Rating	4.0

Question 53

The plant is operating at rated conditions.

A transient occurs on the CCW Pumps.

Currently only one (1) CCW pump is running.

All subsequent actions from the Loss of CCW ONEP have been completed.

Which of the following, in addition to Reactor Recirculation Pumps, describes the components that still have CCW flowing through them?

A. RWCU Non Regen Heat Exchanger ONLY

B. Control Rod Drive Pump oil coolers ONLY

C. RWCU Non Regen Heat Exchanger and Control Rod Drive Pump oil coolers ONLY

D. RWCU Non Regen Heat Exchanger and Fuel Pool Cleaning and Cleanup heat Exchangers ONLY

Answer: B
Explanation: Per the ONEP If only one pump is running then isolate the CCW to Fuel Pool HT EX. and isolate CCW to RWCU non-regen Ht EX. This is RO knowledge because it covers the overall mitigating strategy of the ONEP. 'A' is wrong - RWCU would be isolated due to performing the actions of the onep. B is correct. 'C' is wrong - RWCU would be isolated due to performing the actions of the onep 'D' is wrong - RWCU would be isolated due to performing the actions of the onep and FPCCU would already be isolated by performing actions of the onep

Technical References: GLP-OPS-P4200 GLP-OPS-G3336 electrical drawings 05-1-02-V-1, Loss of CCW ONEP		
References to be provided to applicants during exam: None		
Learning Objective: GLP-OPS-P4200, Objective 10		
Question Source:	Bank #	2015 Biennial
(note changes; attach parent)	Modified Bank #	
	New	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	2
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
201001 CRD Hydraulic **Knowledge of the effect that a loss or malfunction of the control rod drive hydraulic system will have on the following:** **K3.02: Reactor water level**	**Tier #**	2
	Group #	2
	K/A #	201001 K3.02
	Rating	2.6

Question 54

Reactor is shutdown.

CRD Flow Control Valves fail full open.

What is the effect on reactor water level?

A. Reactor water level will remain the same because the pressure control valve will automatically adjust closed to maintain CRD cooling water flow.

B. Reactor water level will remain the same because the excess flow will be diverted to the B21 Reference Leg Purge line.

C. Reactor water level will rise due to rise in CRD cooling water flow.

D. Reactor water level will rise due to rise in CRD drive water flow.

Answer: C
Explanation: The CRD flow controller will fail upscale causing the flow control valves to fully open. As a result the CRD cooling water flow will rise. The pressure control valve will not automatically adjust to maintain downstream pressure for CRD cooling water and exhaust water. A. Plausible because the cooling water header flow will cause a change in reactor water level, however, the pressure control valve is a motor operated valve that must be manually adjusted. B. Plausible because the B21 Reference Leg Purge line comes off of the first header after the flow transmitter that failed. C. Correct see above D. Plausible because the Drive Water header is the first header located after the failed open Flow Control Valves. Validation One person selected D.
Technical References: GFIG-OPS-C1101, Figure 1

References to be provided to applicants during exam: **None**		
Learning Objective:		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	2
10CFR Part 55 Content:	55.41(b)(3)	

Examination Outline Cross-Reference	Level	RO
201003 Control Rod and Drive Mechanism **Ability to monitor automatic operations of the control rod and drive mechanism including:** **A3.01: Control rod position**	**Tier #**	2
	Group #	2
	K/A #	201003 A3.01
	Rating	3.7

Question 55

The reactor has just scrammed.

The scram has <u>not</u> been reset.

The ATC Operator is verifying that all control rods are fully inserted.

How could the operator determine that a control rod is stuck in the core at position 21?

__(1)__ RAW DATA mode and depress the ALL RODS pushbutton. The stuck control rod's position will indicate __(2)__ on the Rod Display Module.

A. (1) Select
(2) 22

B. (1) Select
(2) --

C. (1) Deselect
(2) 22

D. (1) Deselect
(2) --

Answer: C
Explanation: The immediate operator actions of the scram ONEP require the operator to verify all control rods are fully inserted. If the operator only depresses "All Rods" and a control rod was stuck at an odd reed switch position then he will only observe the "--" indication and could easily assume that rod is fully inserted. Per the RC&IS SOI, deselecting RAW DATA will cause the RC&IS Rod Display Module (RDM) display the last even rod position stored in the translator file (RIS cabinet). When RAW DATA is selected and a control rod is stuck at an odd reed switch position the RDM will display "--" (it also displays this for a scrammed rod that is fully inserted if the scram has not be reset as is the case for this question). Even though B represents a valid indication it will not allow the operator to determine that a control rod is stuck in the core since the control rods that have fully inserted will display the same indication, as

explained above. Validation Two people selected B		
Technical References: Scram ONEP 04-1-01-C11-2, RC&IS SOI		
References to be provided to applicants during exam: None		
Learning Objective:		
Question Source:	Bank #	1007
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	2
10CFR Part 55 Content:	55.41(b)(6)	

Examination Outline Cross-Reference	Level	RO
201005 RCIS **Ability to (a) predict the impacts of the following on the rod control and information system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** **A2.12: Rod uncoupled: BWR-6**	**Tier #**	2
	Group #	2
	K/A #	201005 A2.12
	Rating	3.7

Question 56

What is the impact of the CONT ROD OVERTRAVEL annunciator?
What procedure should be used to mitigate the consequences of the annunciator?

A. The control rod has become uncoupled.
Control Rod/Drive Malfunctions ONEP, 05-1-02-IV-1

B. The control rod has been inserted too far.
Control Rod/Drive Malfunctions ONEP, 05-1-02-IV-1

C. The control rod has become uncoupled.
Rod Control and Information SOI, 04-1-01-C11-2

D. The control rod has been inserted too far.
Rod Control and Information SOI, 04-1-01-C11-2

Answer: A
Explanation: A. Per 03-1-01-1, P&L 2.1.2 and 2.1.3, control rod coupling shall be verified by observing the absence of the control rod over-travel annunciator. If control rod uncoupling is observed then refer to Control Rod/Drive Malfunctions ONEP. B. Student may be confused with overtravel annunciator with inserting rods too far. The CRD ONEP is correct procedure. C. Per 03-1-01-1, P&L 2.1.2 and 2.1.3, control rod coupling shall be verified by observing the absence of the control rod over-travel annunciator. The SOI only describes the rod uncoupled alarm. It does not give any guidance on mitigation strategy. D. Student may be confused with overtravel annunciator with inserting rods too far. The SOI only describes the rod uncoupled alarm. It does not give any guidance on mitigation strategy. Validation Two people selected C
Technical References:

03-1-01-1, Cold Shutdown to Generator Carrying Minimum Load (IOI-1) 05-1-02-IV-1, Control Rod/Drive Malfunctions ONEP 04-1-01-C11-2		
References to be provided to applicants during exam: **None**		
Learning Objective:		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	2
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
202001 Recirculation **Ability to predict and/or monitor changes in parameters associated with operating the recirculation system controls including:** **A1.05: Reactor power**	**Tier #**	2
	Group #	2
	K/A #	202001 A1.05
	Rating	3.9

Question 57

The plant is operating at rated conditions at 100% Rod Line.

Due to ONEP actions, recirc flow is lowered to 70 Mlbm/hr. What is the final expected reactor power for this transient?

A. 65%

B. 74%

C. 77%

D. 83%

Answer: B		
Explanation: A. Plausible if student misreads axis on graph and swaps axis. 70% power correlates to 65% core. B. Correct. 100% Rod Line at 70 Mlbm/hr correlates to 74 % core power. C. Plausible because 77% core power correlates to 70 Mlbm/hr core flow at 105% rod line. D. Plausible because 83 % power correlates to 70 % core flow and 105% rod line. Validation One person selected C		
Technical References: 05-1-02-III-3, Reduction in Recirculation System Flow Rate		
References to be provided to applicants during exam: Power to flow map, 05-1-02-III-3, Reduction in Recirculation System Flow Rate		
Learning Objective: GLP-OPS-C5102 Obj 3.5, 7		
Question Source:	Bank #	653
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	N/A

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	2
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
204000 RWCU **For the reactor water cleanup system:** **G2.1.28: Knowledge of the purpose and function of major system components and controls.**	**Tier #**	2
	Group #	2
	K/A #	204000 G2.1.28
	Rating	4.1

Question 58

When the reactor pressure is 90 psig, RWCU must be operating in ___(a)___ mode to ___(b)___

	(a)	(b)
a.	Pre-pump	ensure sufficient NPSH for the RWCU pumps.
b.	Pre-pump	increase the life of the pump seals.
c.	Post-pump	ensure sufficient NPSH for the RWCU pumps.
d.	Post-pump	increase the life of the pump seals.

Answer: A		
Explanation: Per FLP-OPS-G3336, When reactor pressure is above 100 psig, the system is operated in the post-pump mode where reactor water is routed through the system heat exchangers and cooled prior to entering the pumps, thus increasing the life of the pump seals. When reactor pressure is below 100 psig, the systm is operated in the pre-pump mode to ensure sufficient NPSH for the RWCU Pumps. A. Correct B. Plausible because the correct mode but the purpose of post pump. C. Plausible because correct purpose but wrong mode. D. Plausible because the purpose of post pump mode is to increase the life of the pump seals.		
Technical References: GLP-OPS-G3336		
References to be provided to applicants during exam: **None**		
Learning Objective: GLP-OPS-G3336 Obj. 20		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	

	New	X
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
223001 Primary CTMT and Aux.	**Tier #**	2
	Group #	2
Knowledge of the operational implications of the following concepts as they apply to primary containment system and auxiliaries:	**K/A #**	223001 K5.08
	Rating	2.7
K5.08: Pressure measurement		

Question 59

Under which of the following conditions will the Normal Drywell Vacuum Relief Valves (E61-F007 and E61-F020) open?

A. 30 seconds after a LOCA if drywell to containment dp is less than +0.87 psid.

B. Drywell pressure is 0.77 psig and Containment pressure is 0.57 psig.

C. Drywell pressure is 1.12 psig, containment pressure is 1.32 psig, and reactor water level is -53".

D. Drywell pressure is 1.12 psig, containment pressure is 1.32 psig, and reactor water level is -162".

Answer: C
Explanation: Normal vacuum relief requires Drywell to containment dp to be <-.18 psid to open and there can be no LOCA (LSS) signal present. A is plausible because this is the permissive signal to initiate the CGCS, but initiation will isolate normal vacuum relief. B is plausible because it gives a dp of +.20 psid and applicants who have not mastered the subject will believe that this psid should have caused normal vacuum relief to actuate. D is plausible because is gives a dp of -.20 psid but at -162" in the reactor, Normal vacuum relief would be isolated. An applicant could falsely believe that the system is only designed for emergencies during a LOCA and choose this answer over the actual answer.
Technical References: GLP-OPS-E6100
References to be provided to applicants during exam: **None**
Learning Objective: GLP-OPS-E6100 Objective: 6.1

Question Source:	Bank #	329
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	2
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
226001 RHR/LPCI: CTMT Spray Mode **Knowledge of RHR/LPCI: containment spray system mode design feature(s) and/or interlocks which provide for the following:** **K4.11: Prevention of leakage to the environment through system heat exchanger**	**Tier #**	2
	Group #	2
	K/A #	226001 K4.11
	Rating	2.7

Question 60

LOCA has occurred and containment spray has been initiated.

RHR equipment room temperature reaches 170F due to system leakage.

Which group(s) isolation(s) occurred as a result?

A. Group 2 ONLY

B. Groups 2 and 3 ONLY

C. Groups 3 and 4 ONLY

D. Group 2, 3, and 4 ONLY

Answer: D		
Explanation: Per 05-1-02-III-5, Automatic Isolations ONEP, Groups 2, 3, ad 4 all isolate when RHR equipment room temperature reaches 165F. A, B, and C are all plausible because they are partial correct answers.		
Technical References: **05-1-02-III-5, Automatic Isolations ONEP**		
References to be provided to applicants during exam: **None**		
Learning Objective:		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	3
10CFR Part 55 Content:	55.41(b)(9)	

Examination Outline Cross-Reference	Level	RO
233000 Fuel Pool Cooling/Cleanup **Knowledge of the physical connections and/or cause-effect relationships between fuel pool cooling and clean-up and the following:** **K1.02: Residual heat removal system: Plant-Specific**	**Tier #**	2
	Group #	2
	K/A #	233000 K1.02
	Rating	2.6

Question 61

The plant is in Mode 4, preparing for a Refueling outage.

Both Fuel Pool Cooling and Cleanup Pumps have tripped and cannot be restored.

RHR A is started and operating in Fuel Pool Cooling Assist mode with the E12-F066A, RHR A FPC Assist Suction open and the E12-F004A, RHR A Suppression Pool Suction closed.

A valid LOCA signal is received.

ADHR trip/enable switch is in normal.

Which of the following describes the response of the RHR A subsystem after LSS sequencing?

A. RHR A pump will not auto start.

B. RHR A pump will auto restart on minimum flow.

C. RHR A pump will auto restart and inject Spent Fuel Pool water into the reactor through the LPCI injection flow path.

D. RHR A pump will restart in LPCI mode after the E12-F066A valve auto closes and the E12-F004A valve auto opens.

Answer: C
Explanation: Refer to RHR SOI 6.1.1.b Caution. A is incorrect because the RHR A pump will restart and inject. B is incorrect because the F042A will open, due to being in mode 4 and below 476 psig reactor

pressure, and provide a flowpath into the reactor. C. The RHR pump will restart after LSS sequence because the F066A is a suction start permissive for the pump, the F042A injection valve will open and pump the Spent Fuel Pool into the reactor. D is incorrect because the F066A and the F004A do not have any auto actions. Validation Three people selected A and one person selected D.		
Technical References: Refer to RHR SOI 6.1.1.b Caution. 04-1-01-E12-1, RHR System		
References to be provided to applicants during exam: **None**		
Learning Objective: GLP-OPS-E1200 obj 9.5		
Question Source:	Bank #	890
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2008
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
241000 Reactor/Turbine Pressure Regulator Ability to manually operate and/or monitor in the control room: A4.07: Main Stop/throttle Valves (operation)	Tier #	2
	Group #	2
	K/A #	241000 A4.07
	Rating	3.5

Question 62

Currently:

- Reactor power is 18%
- Main generator has just been sync to the grid with 120 MWe
- Bypass Control Valves are 7% open
- Load reference is currently OFF.

Feedwater Startup Flow Control Valve Fails OPEN.

Which of the following describes the change in Main Stop and Control Valve indication?

	STOP VALVES	CONTROL VALVES
A.	Remain AS-IS	Remain AS-IS
B.	Remain AS-IS	OPEN
C.	CLOSE	CLOSE
D.	Remain AS-IS	CLOSE

Answer: A
Explanation: During a plant startup with LOAD Reference OFF any power rise will cause reactor pressure to rise but the IPC is still controlling which will cause the Bypass control valves to open further. The Main Stop and Control valves will remain as is until LOAD Ref. is placed in service or the SPEED DEMAND is depressed. A is correct B is wrong - this would be true if Load Ref was on and the turbine was accepting all power from the

reactor, however with it being off the Bypass control valves will open and close to maintain pressure control. C. is wrong - Stop valves only close on a turbine trip, nothing indicated that a turbine trip has occurred, also the control valves would also not close since they are being controlled by the Speed Controller only D is wrong - Control valves would not close since they are being controlled by the Speed Controller only		
Technical References: GLP-OPS-N1136 GLP-OPS-N3201/02		
References to be provided to applicants during exam: None		
Learning Objective:		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #375	NRC EXAM 2011
	New	
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	2
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
245000 Main Turbine Gen./Aux. **Knowledge of the effect that a loss or malfunction of the following will have on the main turbine generator and auxiliary systems:** **K6.02: Reactor/turbine pressure control system: Plant-Specific**	**Tier #** **Group #** **K/A #** **Rating**	2 2 245000 K6.02 3.5

Question 63

The plant is operating at 75% power.

A malfunction causes 'A' Main Bypass Control valve to fully open.

What effect will this have on Turbine Control Valve (TCV) position and actual generator MW?

	TCV position	Actual MW
A.	Throttle open direction	Up
B.	Throttle open direction	Remains same
C.	Throttle close direction	Down
D.	Throttle close direction	Remains same

Answer: C
Explanation: A. Plausible if applicant believes the TCV position will do the same operation as main bypass control valves. More steam directed to the turbine would cause actual MW to go up.. B. Plausible if applicant believes the TCV position will do the same operation as main bypass control valves. Actual MW remains the same is plausible if the student believes the TCV position determines the MW produced. C. This is correct because when the Main Bypass Control valve goes fully open, the TCV will throttle closed to maintain pressure. A combination of the BCV opening and TCV throttling closed causes Actual MW to go down. D. When the Main Bypass Control valve goes fully open, the TCV will throttle closed to maintain pressure. The actual MW remaining the same is plausible if applicant bases actual MW produced to total steam demand.
Technical References:

GFIG-OPS-N3202		
References to be provided to applicants during exam: **None**		
Learning Objective:		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	2
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
271000 Offgas **Knowledge of the effect that a loss or malfunction of the offgas systems will have on the following:** **K3.01: Condenser vacuum**	**Tier #**	2
	Group #	2
	K/A #	271000 K3.01
	Rating	3.5

Question 64

The plant is operating at rated power with Offgas Train ‘A’ in service.

A major tube rupture occurs inside Offgas Condenser ‘A’.

Which of the following parameters **LOWERS** in response to this event?

A. Main condenser vacuum

B. Offgas Preheater ‘A’ Inlet Pressure

C. Offgas Condenser ‘A’ Level

D. Offgas Condenser ‘A’ Outlet Temperature

Answer: A

Explanation:
The reason condenser vacuum lowers is simply due to the “back-pressure” placed on the Offgas stream flow felt all the way back through the Offgas system to the Condenser Air Removal System (SJAEs) as a result of the flooded condenser. This is the mechanism for a main condenser degrading (lowering) vacuum.

TBCW flows through the Offgas Condenser tube-side. Thus, a tube rupture results in a flood-up (high level) on the condenser shell-side (at much lower Offgas system pressure).
For this reason, C is wrong.

B is wrong for the same reason. See ARI P845-1A-B1, which shows that Preheater Inlet Pressure goes high (not low) as a result of a flooded Offgas condenser.

D is wrong. This is not quite so obvious as the first two. See ARI P845-1A-C6. When the condenser floods, TBCW recirculation (i.e., heat transfer of BTUs away from the condenser) essentially stops; the “pool” of water in the shell-side becomes ineffective, causing a rise in the Offgas outlet temperature (i.e., less cooling of the Offgas stream through the condenser shell).

Technical References:
GFIG-OPS-N6465
M-1092A
ARIs P845-1A-A6, B1, C6

References to be provided to applicants during exam: **None**		
Learning Objective: GLP-OPS-N6465, OBJ. 25		
Question Source:	Bank #	427
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2010
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	2
10CFR Part 55 Content:	55.41(b)(5)	

Examination Outline Cross-Reference	Level	RO
286000 Fire Protection **Knowledge of electrical power supplies to the following;** **K2.02: Pumps**	**Tier #**	2
	Group #	2
	K/A #	286000 K2.02
	Rating	2.9

Question 65

The power supply to the motor driven fire pump is:

A. 15BA2

B. 16BB1

C. 11HD

D. 11BD3

Answer: D		
Explanation: A. Student may remember only one motor driven fire pump and could confuse to be ESF powered. Division 1 ESF LCC is 15BA2. B. Student may remember only one motor driven fire pump and could confuse to be ESF powered. Division 2 ESF LCC is 16BB1. 'A' PAC is also powered from Division 2 ESF. C. Student may remember the fire pump is BOP powered from 11HD bus. D. Motor driven fire pump is powered from BOP LCC 11BD3.		
Technical References: 04-1-01-R21-11, 15, and 16. 04-1-01-L11-1, Attachment IF		
References to be provided to applicants during exam: **None**		
Learning Objective:		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	

	LOD	2
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
1.Conduct of Operations	**Tier #**	3
	Group #	
2.1.3: Knowledge of shift or short-term relief turnover practices.	**K/A #**	2.1.3
	Rating	3.7

Question 66

The ACRO requests a relief at his position by the CRO for a bathroom break.

Both individuals were present at the beginning of shift brief.

Which of the following is required to be performed?

A. Turnover parameters, from previous shift, of special interest being monitored and required frequency.

B. Perform an Update Brief informing the shift crew of the change.

C. Turnover any compensatory actions in effect from previous shift.

D. Perform a Reactivity Brief.

Answer: B		
Explanation: Per 5.7 [2], EN-OP-115 requires the following 3 things: • Permission from the SM or CRS • Verbal turnover • Update brief to inform the crew. A is wrong. Per 5.2 [2], Only required if oncoming operator was not present during shift turnover C is wrong. Per 5.2 [2], Only required if oncoming operator was not present during shift turnover D is wrong. Only required if reactivity manipulation will be performed		
Technical References: EN-OP-115-3, 5.2 [2]		
References to be provided to applicants during exam: None.		
Learning Objective: GLP-OPS-PROC, Objective 5		
Question Source:	Bank #	254
(note changes; attach parent)	Modified Bank #	

	New	
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
1. Conduct of Operations **2.1.13: Knowledge of facility requirements for controlling vital/controlled access.**	**Tier #**	3
	Group #	
	K/A #	2.1.13
	Rating	2.5

Question 67

Per 01-S-06-4, Access and Conduct in the Control Room, the number of people inside 1H13-P680 panel horseshoe area should be limited to the At-The Controls Operator (ATC) and ___(1)_____ other people, unless otherwise authorized by the ___(2)_____.

	(1)	(2)
A.	Two	CRS
B.	Two	Shift Manager
C.	Three	CRS
D.	Three	Shift Manager

Answer: B		
Explanation: See 01-S-06-4 Step 6.3.6. The number of people inside 1H13-P680 panel horseshoe area should be limited to three (3), unless otherwise authorized by the shift manager. Three other people is plausible because the area should be limited to three total. ATC is plausible because the area discussed is the watch area assigned to the ATC.		
Technical References: 01-S-06-4		
References to be provided to applicants during exam: **None**		
Learning Objective: GLP–OPS–PROC, Obj. 11.2		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
1. Conduct of Operations	**Tier #**	3
	Group #	
2.1.14: Knowledge of criteria or conditions that require plant-wide announcements, such as pump starts, reactor trips, mode changes, etc	**K/A #**	2.1.14
	Rating	3.1

Question 68

Site announcements are required under which of the following condition(s):

A. Only EOOS risk changes out of green.

B. Only EOOS risk changes to red.

C. All EOOS risk condition color changes.

D. Only EOOS risk changes escalated up.

Answer: C		
Explanation: Per 02-S-01-41, 6.3.3 Site Announcements. The control room will make site wide announcements when risk conditions change colors.		
Technical References: 02-S-01-41		
References to be provided to applicants during exam: **None**		
Learning Objective: GLP-OPS-PROC, Obj. 31.4		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
2. Equipment Control	**Tier #**	3
	Group #	
2.2.7: Knowledge of the process for conducting special or infrequent tests.	**K/A #**	2.2.7
	Rating	2.9

Question 69

Per EN-OP-116, IPTEs, every IPTE must have a "Controlling Document" with which to actually perform the evolution.

Controlling Document types include Work Order Instructions, existing Plant Procedures, and…

A. Engineering Change Instructions.

B. Performance Engineering Instructions.

C. Special Process Instructions.

D. Special Test Instructions.

Answer: D
Explanation: Per EN-OP-116, section 3.0[7]. A is wrong; has strong face plausibility. B is wrong; this is a partial label for the series "17" procedures at GGNS. C is wrong; these are a subset of the series "07" procedures at GGNS.
Technical References: EN-OP-116, IPTEs
References to be provided to applicants during exam: **None**
Learning Objective: GLP-OPS-PROC, OBJ. 80.0

Question Source:	Bank #	412
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2010

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
2. Equipment Control **2.2.14: Knowledge of the process for controlling equipment configuration or status.**	**Tier #**	3
	Group #	
	K/A #	2.2.14
	Rating	3.9

Question 70

Which of the following is considered a temporary modification **required** to be controlled by EN-DC-136, Temporary Modifications?

A. Due to outage activities, a temporary feed is connected to MCC 11B12 to supply its loads.

B. Following an engineering evaluation and approval, a seismic class 1 support is temporarily removed to accommodate a maintenance activity.

C. Temporary scaffolding is installed over a reactor feedwater pump.

D. A circuit board is temporarily removed to support an electrical test.

Answer: A		
Explanation: See EN-DC-136, specifically Attachment 9.2, Part II, Exclusions, Sheet 1 of 4, Part I – SCREENING, and Part II – EXCLUSIONS. Answer is correct because its activity is not found on the list of Exclusions. B is wrong because Exclusion #9 applies to this activity. C is wrong because Exclusion #8 applies to this activity. D is wrong because Exclusion #11 applies to this activity.		
Technical References: EN-DC-136, Temporary Modifications		
References to be provided to applicants during exam: **None**		
Learning Objective: GLP-OPS-PROC Obj 40.5		
Question Source:	Bank #	155
(note changes; attach parent)	Modified Bank #	
	New	

Question History:	Last NRC Exam	2012
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	3
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
2. Equipment Control **2.2.35: Ability to determine Technical Specification Mode of Operation.**	**Tier #**	3
	Group #	
	K/A #	2.2.35
	Rating	3.6

Question 71

Which of the following changes in plant conditions (or system status) will necessarily involve a **MODE** change?

A. With the Mode Switch in REFUEL the reactor cavity is flooded up for Core Alterations; operators have now placed the Mode Switch in SHUTDOWN.

B. During a plant shutdown per IOI-3 the RHR Shutdown Cooling reactor pressure interlock has just cleared; 3 hours later average reactor coolant temperature reaches 199°F.

C. Following Core Alterations, RPV water level has been lowered to allow for removal of the main steam line plugs; a total loss of Shutdown Cooling results in average reactor coolant temperature rising above 200°F.

D. Operators have just completed the “Refuel Position One-Rod-Out Interlock” surveillance; that surveillance has failed and the interlock has been declared inoperable.

Answer: B
Explanation: The “RHR SDC reactor pressure interlock (i.e., 135 psig) has just cleared” indicates we’ve began SDC while in MODE 3. As soon as temperature drops below 200F, we’ve done a MODE change…to MODE 4. A is wrong. MODE 5 (clearly with the cavity flooded) simply moving the M/S to SHUTDOWN does not necessarily involve a MODE change. MODE 5 permits either switch position. C is wrong. Clearly we’re still in MODE 5 (RPV head removed for the plug work), where coolant temperature is irrelevant. D is wrong. See LCO 3.9.2. This surveillance can only be performed in MODE 5. If it fails we simply remain in MODE 5. Validation Three people selected C

Technical References: Tech Specs, Table 1.1-1		
References to be provided to applicants during exam: **None**		
Learning Objective: GLP-OPS-TS001, Objective 5		
Question Source:	Bank #	256
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
3. Radiation Control **2.3.13: 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.**	**Tier #**	3
	Group #	
	K/A #	2.3.13
	Rating	3.4

Question 72

Per EN-RP-101, Access Control For Radiologically Controlled Areas, who is **required** to give the **final** approval for personnel to enter a Locked High Radiation Area (LHRA) with general area dose rates greater than 1.5 Rem/hr in the actual work area?

A. Plant Operations General Manager

B. Radiation Protection Manager

C. Radiation Protection Supervisor

D. Operations Manager

Answer: B		
Explanation: See EN-RP-101, Section 5.5[10], 1st bullet at the top of page 18. Per the above reference, the RPM's approval is required. Distracters are all as plausible, to the Applicant who has never attended much to this procedure and who can only recall small amounts of information about LHRA access described in Administrative Controls Section 5.7 of GGNS Tech Specs.		
Technical References: EN-RP-101, Access Control For Radiologically Controlled Areas		
References to be provided to applicants during exam: None		
Learning Objective: GLP-OPS-PROC Obj 50		
Question Source:	Bank #	156
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2012

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	3
10CFR Part 55 Content:	55.41(b)(4)	

Examination Outline Cross-Reference	Level	RO
3. Radiation Control **2.3.15: Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc**	**Tier #**	3
	Group #	
	K/A #	2.3.15
	Rating	2.9

Question 73

The four high-range, detectors used to monitor post-accident area radiation levels within the Drywell and Containment utilize __________ detectors.

A. Geiger-Mueller tube

B. Ion chamber

C. Scintillation

D. Solid state alpha and beta scintillation

Answer: B		
Explanation: The four high range, ion chamber detectors are used to monitor post-accident area radiation levels within the drywell and containment. A. Plausible because the 50 general area monitor units use Geiger-Mueller tubes for detectors. B. Correct. C. The process liquid monitoring subsystem uses scintillation detectors. D. Ventilation release rad monitors use dsolid state alpha detectors and beta scintillation detectors.		
Technical References: GLP-OPS-D1721, Page 16 of 73		
References to be provided to applicants during exam: **None**		
Learning Objective: GLP-OPS-D1721 OBJ 5.1		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	

	LOD	2
10CFR Part 55 Content:	55.41(b)(11)	

Examination Outline Cross-Reference	Level	RO
4. Emergency Procedures / Plan	**Tier #**	3
	Group #	
2.4.12: Knowledge of general operating crew responsibilities during emergency operations.	**K/A #**	2.4.12
	Rating	4.0

Question 74

The Shift Manager has made the decision to evacuate the control room due to a fire.

Per the "Shutdown From the Remote Shutdown Panel" ONEP, who is permitted to respond and operate the Remote Shutdown Panels (RSP) and Alternate Shutdown Panels (ASP)?

	RSP	ASP
A.	Only licensed operators	NOBs
B.	Only licensed operators	Only licensed operators
C.	Licensed operators and NOBs	NOBs
D.	Licensed operators and NOBs	NOBs supervised by licensed operator

Answer: A		
Explanation: See the Remote Shutdown ONEP, section 3.2.1. Only licensed operators (ROs, CRS, FSS) man the RSPs. NOBs can man the ASPs. The Shift Manager reports to the OCC.		
Technical References: 05-1-02-II-1, Remote Shutdown ONEP		
References to be provided to applicants during exam: **None**		
Learning Objective: GLP-OPS-ONEP, Objective 1		
Question Source:	Bank #	206
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	N/A

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	3
10CFR Part 55 Content:		
	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
4. Emergency Procedures / Plan	**Tier #**	3
	Group #	
2.4.39: Knowledge of RO responsibilities in emergency plan implementation.	**K/A #**	2.4.39
	Rating	3.9

Question 75

A Site Area Emergency is declared at 1411.

As the communicator, regarding State/Local Agencies, which of the following complies with:

(1) the time limit requirements for the initial notification and
(2) the first follow-up notification?

A. (1) Initial is completed at 1430
(2) first-follow-up is completed at 1525.

B. (1) Initial is completed at 1426
(2) first-follow-up is completed at 1531.

C. (1) Initial is completed at 1425
(2) first-follow-up is completed at 1527.

D. (1) Initial is completed at 1424
(2) first-follow-up is completed at 1523.

Answer: D
Explanation: Per 10-S-01-6, sections 6.1.1.c and f, the initial notification is to be completed within 15 minutes of declaration (within 15 minutes of 1411) and the follow-up is to be completed within one hour of the initial. A is wrong; the initial is late. B is wrong; both the initial and follow-up are late. C is wrong; the follow-up is late. **NOTE –** At GGNS, the RO is a Control Room ERO Communicator, responsible for making the Initial and Follow-up Notifications tested in this question. As such he/she would also be responsible for ensuring such notifications are made within the required time limits. Therefore, this question is in fact a legitimate RO-level question for GGNS, and not an SRO-only one.
Technical References: 10-S-01-6, Notification of Offsite Agencies

References to be provided to applicants during exam: **None**		
Learning Objective: GLP-EP-EPTS6, OBJ. 3		
Question Source:	Bank #	510
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2010
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	2
10CFR Part 55 Content:	55.41(b)(10)	

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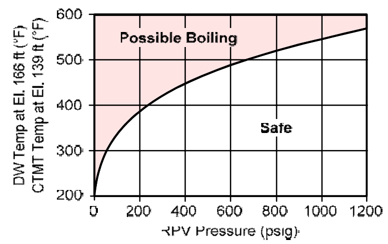
TAB 1

CAUTIONS

1 RPV level indications are affected by instrument run temperatures and RPV pressure:

1. If the temperature near any RPV level instrument leg is in the **Possible Boiling** zone of the **RPVST** (Fig 2), the instrument may be unreliable due to boiling in the run.

Fig 2: RPV Saturation Temperature (RPVST)



2. Do <u>not</u> use any RPV level instrument that exceeds <u>both</u> limits in the following table:

Level nstrument	Indicated Level		Reference Leg Temperature
Wide Range	< –131 in.	and	> 143°F at CTMT El. 139 ft.
Upset Range	< 159 in.	and	> 195°F at DW El. 166 ft.
Shutdown Range	< 139 in.	and	> 66°F at DW El. 166 ft.

2 Low SP level may uncover the SP temperature elements. SP temperature must be determined by alternate means when SP level is below 14.25 ft.

3 Elevated CTMT pressure may trip the RCIC turbine on high exhaust pressure.

4 Operation of RCIC with suction temperature above 225°F may result in equipment damage.

5 A rapid increase in injection into the RPV may induce a large power excursion resulting in substantial core damage.

6 High spent fuel pool temperature or low spent fuel pool water level may affect conditions within secondary containment

EP-1
EP/SAP CAUTIONS
GRAND GULF NUCLEAR STATION
Rev. Date 08/19/11

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TAB 2

PLANT OPERATIONS MANUAL

Volume 10	10-S-03-2
Section 03	Revision: 026
	Date: 6/10/14

REFERENCE USE

FIRE PROTECTION PROCEDURE

RESPONSE TO FIRES

SAFETY RELATED

Prepared: Randy E. Sande

Reviewed: Dennis W Clapley
Technical

Randy E. Sande
Fire Protection Coordinator

Approved: Thomas W. Thornton
Manager, Design & Program Engineering

List of Effective Pages:

Pages 1-16

Attachments I-V

CONTROLLED COPY # ______

List of TCNs Incorporated:

Revision	TCN	Revision	TCN
0	None	023	None
1	1	024	None
2-8	None	025	None
9	2	026	None
10	3		
11	None		
12	4,5		
13	6		
14-021	None		
022	7,8		

SAFE SHUTDOWN COMPONENTS BY FIRE ZONE

General Notes

1. The tables in Attachment IV provides a list of SAFE SHUTDOWN components and associated cables by fire zone, in order to evaluate the impact of a fire in a particular fire area or zone.

2. In each table, a 'D' beside the component will designate that the SAFE SHUTDOWN component or device is in the fire zone. A 'C' beside the device will designate that a circuit or cable for the device listed traverses through the fire zone.

   In addition, the component number in Attachment IV may include an suffix code on the component number to identify the specific condition that the device or cable provides in the Appendix R Compliance strategy or a Functional Service Code to identify cables and components whose function generates specific transients that may either be credited in the Safe Shutdown Analysis or identified for mitigative action. The component number suffix codes and Functional Service Codes are defined as follows:

   XXX.D1 - Represents the Division I portion of components which have more than one divisional control circuit (e.g., SRVs). The circuit analysis for these components considers the Division I control circuit only.

   XXX.D2 - Represents the Division II portion of components which have more than one divisional control circuit (e.g., SRVs). The circuit analysis for these components considers the Division II control circuit only.

   XXX.P - Represents the passive mode for valves or dampers which have both open and closed required safe shutdown positions. The circuit analysis for these valves considers only the passive condition for the component (i.e. spurious operation concern). The circuit analysis representing the active condition remains with the service code without the extension.

   XXX.R - Represents components which are used for the Remote Shutdown scenario and contain remote transfer contacts in their control circuits. The circuit analysis for these components considers the remote transfer switch only in the "EMERGENCY" position and the components as active safe shutdown components. These components are represented on SLDs as having a black triangle in the upper right corner of the component block.

- ☐ ADS.D1-AUTO - Represents the population of instruments and cables which contribute to the automatic initiation of the Division I Automatic Depressurization System.

- ☐ ADS.D2-AUTO - Represents the population of instruments and cables which contribute to the automatic initiation of the Division II Automatic Depressurization System.

- ☐ CS.D1-AUTO - Represents the population of instruments and cables which contribute to the automatic initiation of Division I Containment Spray System.

- ☐ CS.D2-AUTO - Represents the population of instruments and cables which contribute to the automatic initiation of Division II Containment Spray System.
- ☐ HPCS-AUTO - Represents the population of instruments and cables which contribute to the automatic initiation of the High Pressure Core Spray System.
- ☐ LPCI.D1-AUTO - Represents the population of instruments and cables which contribute to the automatic initiation of the Low Pressure Coolant Injection mode of Train A RHR and LPCS.

  LPCI.D1-MAN - Represents the population of cables which contribute to the manual initiation of the Low Pressure Coolant Injection mode of Train A RHR and LPCS.
- ☐ LPCI.D2-AUTO - Represents the population of instruments and cables which contribute to the automatic initiation of the Low Pressure Coolant Injection mode of Train B and Train C RHR.
- ☐ LPCI.D2-MAN - Represents the population of cables which contribute to the manual initiation of the Low Pressure Coolant Injection Mode of Train B and Train C RHR.
- ☐ RCIC-AUTO - Represents the population of instruments, components, and cables which contribute to the automatic initiation of the Reactor Core Isolation Cooling System.
- ☐ RHRALOS- Represents the population of cables which contribute to the loss of RHR Pump 1A suction trip.
- ☐ RHRBLOS- Represents the population of cables which contribute to the loss of RHR Pump 1B suction trip.
- ☐ SSW.D1-MAN - Represents the population of cables which contribute to the manual initiation of the Loop A Standby Service Water System.
- ☐ SSW.D2-MAN - Represents the population of cables which contribute to the manual initiation of the Loop B Standby Service Water System.

3. The Safe Shutdown Systems are those required to maintain water level in the RPV following an isolation/scram, depressurize the RPV by discharging steam to the suppression pool, cool the suppression pool, cool the RPV once it is depressurized, and the associated support systems to maintain these functions. This is accomplished in conjunction with a fire, with or without offsite power. The listed systems provide dual shutdown paths so that the reactor can be depressurized by the Main Steam Safety/Relief valves and water level maintained with low pressure ECCS. It also provides dual reactor cooling paths using the ALTERNATE SHUTDOWN mode of RHR.

4. Attachment IV will in no way imply or direct Shift Supervision to shutdown the plant. It only is for the purpose of analyzing the affects of a fire on components that could be required to shutdown the plant if that decision was made.

1E12-C002A	RHR PUMP 1-A	C
1E12-C002B	RHR PUMP 1-B	C
1E12-C003B	RHR PUMP 1-B JOCKEY PUMP	C
RHRALOS	RHR Pump A Loss Of Suction Trip	C
1E12-F064A.R	RHR PUMP A MINIMUM FLOW VALVE	C
1E12-F064A	RHR PUMP A MINIMUM FLOW VALVE	C
1E12-F006A	RHR PUMP A SHUTDOWN COOLING SUCT VALVE	C
1E12-F006A.R	RHR PUMP A SHUTDOWN COOLING SUCT VALVE	C
1E12-F004A.R	RHR PUMP A SUPP POOL SUCTION VALVE	C
1E12-F004A	RHR PUMP A SUPP POOL SUCTION VALVE	C
RHRBLOS	RHR Pump B Loss Of Suction Trip	C
1E12-F064B	RHR PUMP B MINIMUM FLOW VALVE	C
1E12-F006B	RHR PUMP B SHUTDOWN COOLING SUCT VALVE	C
1E12-F004B	RHR PUMP B SUPP POOL SUCTION VALVE	C
1T51-B003	RHR ROOM A COOLER FAN	C
1T51-B003.R	RHR ROOM A COOLER FAN	C
1T51-B004	RHR ROOM B COOLER FAN	C
1E12-F009	RHR SHUTDN CLG INBD SUCTION VALVE	C
1E12-F008	RHR SHUTDN CLG OUTBD SUCTION VALVE	C
1E12-F008.R	RHR SHUTDN CLG OUTBD SUCTION VALVE	C
1E12-FT-N052C	RHR SYSTEM C MINIMUM BYPASS FLOW TRANSMITTER	C
1E12-TR-R601	RHR TEMPERATURE RECORDER	D
1E12-TR-R601	RHR TEMPERATURE RECORDER	C
1B21-F002	RPV INBD VENT VLV	C
1B21-F001	RPV OTBD VENT VLV	C
1E12-PT-N058B	RPV Pressure Transmitter	C
1E12-PT-N058A	RPV Pressure Transmitter	C
1B33-F019	RW PROC SMPL DRWL INBD ISOL	C
1B33-F020	RW PROC SMPL DRWL OUTBD ISOL	C
1B33-F020.R	RW PROC SMPL DRWL OUTBD ISOL	C
1G33-F028	RWCU BLWDN CTMT INBD ISOL	C
1G33-F034.R	RWCU BLWDN CTMT OTBD ISOL	C
1G33-F034	RWCU BLWDN CTMT OTBD ISOL	C

1G33-F250	RWCU SPLY TO RWCU HXS	C
1G33-F251	RWCU SPLY TO RWCU HXS	C
1Z77-B001A.R	SFGD SWGR - BTRY RM AH UNIT A	C
2Z77-B001A	SFGD SWGR - BTRY RM AH UNIT A	C
1Z77-B001A	SFGD SWGR - BTRY RM AH UNIT A	C
2Z77-B001A.R	SFGD SWGR - BTRY RM AH UNIT A	C
2Z77-B001B	SFGD SWGR - BTRY RM AH UNIT B	C
1Z77-B001B	SFGD SWGR - BTRY RM AH UNIT B	C
2Z77-C001A.R	SFGD SWGR - BTRY RM EXH FAN A	C
1Z77-C001A.R	SFGD SWGR - BTRY RM EXH FAN A	C
2Z77-C001A	SFGD SWGR - BTRY RM EXH FAN A	C
1Z77-C001A	SFGD SWGR - BTRY RM EXH FAN A	C
2Z77-C001B	SFGD SWGR - BTRY RM EXH FAN B	C
1Z77-C001B	SFGD SWGR - BTRY RM EXH FAN B	C
1P60-F004	SPCU SPLY TO CNDS PC FLTRS	C
1P60-F003	SPCU SPLY TO CNDS PC FLTRS	C
2Z77-F003A.R	SPLY FAN INL DAMPER	C
2Z77-F003A	SPLY FAN INL DAMPER	C
1Z77-F003B	SPLY FAN INL DAMPER	C
2Z77-F003B	SPLY FAN INL DAMPER	C
1Z77-F003A	SPLY FAN INL DAMPER	C
1Z77-F003A.R	SPLY FAN INL DAMPER	C
2Z77-F035A.R	SPLY FAN OUTL DAMPER	C
2Z77-F035B	SPLY FAN OUTL DAMPER	C
1Z77-F035A	SPLY FAN OUTL DAMPER	C
1Z77-F035A.R	SPLY FAN OUTL DAMPER	C
1Z77-F035B	SPLY FAN OUTL DAMPER	C
2Z77-F035A	SPLY FAN OUTL DAMPER	C
1P42-F201A	SSW A RTN FM FPHX A	C
1P42-F200A	SSW A SPLY TO FPHX A	C
1P42-F200B	SSW B INL TO FPHX B	C
1P42-F201B	SSW B OUTL FM FPHX B	C
1P41-F007A	SSW BASIN A TRANSFER VLV	C

1P41-F007A.R	SSW BASIN A TRANSFER VLV	C
1P41-F007B	SSW BASIN B TRANSFER VLV	C
1P41-C003A.R	SSW COOLING TOWER FAN A	C
1P41-C003A	SSW COOLING TOWER FAN A	C
1P41-C003B.R	SSW COOLING TOWER FAN B	C
1P41-C003B	SSW COOLING TOWER FAN B	C
1P41-C003C	SSW COOLING TOWER FAN C	C
1P41-C003D	SSW COOLING TOWER FAN D	C
1P44-F054	SSW INBD SPLY TO DRWL CLRS/CCW HXS	C
SP41-F064A	SSW INL TO CR A/C A (LOOP A)	C
1P41-F014A.R	SSW INLET TO RHR HX A	C
1P41-F014A	SSW INLET TO RHR HX A	C
1P41-F014B	SSW INLET TO RHR HX B	C
1P41-F018A.R	SSW LOOP A INLET TO DG 11 WATER COOLER	C
1P41-F018A	SSW LOOP A INLET TO DG 11 WATER COOLER	C
1P41-F237.R	SSW LOOP A INLET TO ESF ROOM CLRS A	C
1P41-F237	SSW LOOP A INLET TO ESF ROOM CLRS A	C
1P41-F238.R	SSW LOOP A OUTLET FROM ESF ROOM CLRS A	C
1P41-F238	SSW LOOP A OUTLET FROM ESF ROOM CLRS A	C
1P41-F005A.R	SSW LOOP A RETURN TO COOLING TOWER A	C
1P41-F005A	SSW LOOP A RETURN TO COOLING TOWER A	C
1E12-F094	SSW LOOP B INBD INLET TO RHR B	C
SP41-F155B	SSW LOOP B INBD SUPPLY TO AIR COMPRS	C
1P41-F018B	SSW LOOP B INLET TO DG 12 WATER COOLER	C
1E12-F096	SSW LOOP B OTBD INLET TO RHR B	C
SP41-F081B	SSW LOOP B RETURN FROM CR HVAC & ESF ROOM CLRS B	C
1P41-F005B	SSW LOOP B RETURN TO COOLING TOWER B	C
SP41-F064B	SSW LOOP B SUPPLY TO CR HVAC & ESF ROOM CLRS B	C
SP41-F081A	SSW OUTL FM CR A/C A (LOOP A)	C
1P41-F068A.R	SSW OUTLET FROM RHR HX A	C
1P41-F068A	SSW OUTLET FROM RHR HX A	C
1P41-F068B	SSW OUTLET FROM RHR HX B	C
2Y47-TE-N005A	SSW PMPHS A AREA TEMPERATURE (DIV II)	C

2Y47-F001A	SSW PMPHS A EXH DMPR (DIV II)	C
2Y47-F002A	SSW PMPHS A INTK DMPR (DIV II)	C
1Y47-C001A	SSW PMPHS A O/A FAN	C
1Y47-C001A.R	SSW PMPHS A O/A FAN	C
2Y47-C001A	SSW PMPHS A O/A FAN (DIV II)	C
2Y47-TE-N013A	SSW PMPHS A OUTSIDE AIR TEMPERATURE (DIV II)	C
2Y47-F003A	SSW PMPHS A RECIRC DMPR (DIV II)	C
1Y47-TE-N005B	SSW PMPHS B AREA TEMPERATURE	C
1Y47-F001B	SSW PMPHS B EXH DMPR	C
1Y47-F002B	SSW PMPHS B INTK DMPR	C
1Y47-C001B	SSW PMPHS B O/A FAN	C
1Y47-TE-N013B	SSW PMPHS B OUTSIDE AIR TEMPERATURE	C
1Y47-F003B	SSW PMPHS B RECIRC DMPR	C
1P41-C001A	SSW PUMP A	C
1P41-C001A.R	SSW PUMP A	C
1P41-UR-R606A	SSW PUMP A DISCH PRESSURE RECORDER	D
1P41-UR-R606A	SSW PUMP A DISCH PRESSURE RECORDER	C
1P41-F001A	SSW PUMP A DISCH VLV	C
1P41-F001A.R	SSW PUMP A DISCH VLV	C
1P41-F016A	SSW PUMP A INBOARD BLOWDOWN VLV	C
1P41-F016A.R	SSW PUMP A INBOARD BLOWDOWN VLV	C
1P41-F006A.R	SSW PUMP A RECIRC VLV	C
1P41-F006A	SSW PUMP A RECIRC VLV	C
1P41-C001B	SSW PUMP B	C
1P41-PT-N009B	SSW PUMP B DISCH PRESSURE	C
1P41-UR-R606B	SSW PUMP B DISCH PRESSURE RECORDER	D
1P41-UR-R606B	SSW PUMP B DISCH PRESSURE RECORDER	C
1P41-F001B	SSW PUMP B DISCH VLV	C
1P41-F016B	SSW PUMP B INBOARD BLOWDOWN VLV	C
1P41-F006B	SSW PUMP B RECIRC VLV	C
1P75-E001A	STANDBY DIESEL GENERATOR ENGINE E001A	C
1P75-E001A.R	STANDBY DIESEL GENERATOR ENGINE E001A	C
1P75-E001B	STANDBY DIESEL GENERATOR ENGINE E001B	C

SSW.D1-MAN	Standby Service Water Loop A Manual Initiation	C
SSW.D2-MAN	Standby Service Water Loop B Manual Initiation	C
1P75-C002B	STBY DSL GEN 12 FUEL OIL TRANSFER PMP	C
1Z77-FT-N001	SWGR - BATT RM EXHAUST FAN A FLOW	C
1Z77-FT-N004	SWGR - BATT RM EXHAUST FAN B FLOW	C

Safe Shutdown Equipment for Fire Zone: 0C504

Component	Description	Type
15AA	4.16KV ESF BUS 15AA (1A5)	C
1C11-F110A	A Backup SCRAM Valve	C
1C11-F110B	B Backup SCRAM Valve	C
1C11-F110C	C Backup SCRAM Valve	C
RPSA.GRP1	Channel A/ Trip System A, SCRAM Pilot Valve Solenoid (GRP 1	C
RPSA.GRP2	Channel A/ Trip System A, SCRAM Pilot Valve Solenoid (GRP 2	C
RPSB.GRP1	Channel B/ Trip System B, SCRAM Pilot Valve Solenoid (GRP 1	C
RPSA.GRP3	Channel C/ Trip System A, SCRAM Pilot Valve Solenoid (GRP 3	C
RPSA.GRP4	Channel C/ Trip System A, SCRAM Pilot Valve Solenoid (GRP 4	C
RPSB.GRP3	Channel D/ Trip System B, SCRAM Pilot Valve Solenoid (GRP 3	C
RPSB.GRP4	Channel D/ Trip System B, SCRAM Pilot Valve Solenoid (GRP 4	C
SZ51-F017	CON ROOM AC UNIT ISO DAMPER	C
SZ51-F008	CONT RM FRESH AIR UNIT A INL/RECIRC VLV	C
SZ51-F007	CONT RM FRESH AIR UNIT A INL/RECIRC VLV	C
SZ51-F010	CONT RM NORM O/A INBD INL VLV	C
SZ51-F001	CONT RM PURGE OTBD EXH VLV	C
SZ51-F003	CONT RM UTILITY EXH FAN INBD INL VLV	C
SZ51-F005	CONT ROOM STANDBY FRESH AIR UNIT ISOL DAMPER	C
SZ51-B002A	CONTROL ROOM A/C COOLING UNIT	C
SZ51-D002A	CONTROL ROOM STANDBY FRESH AIR UNIT FAN	C
SZ51-TIT-N606A	CONTROL ROOM THERMOSTAT	D
SZ51-TIT-N606A	CONTROL ROOM THERMOSTAT	C
SZ51-TIT-N606B	CONTROL ROOM THERMOSTAT	C
1X77-C001A.R	DG 11 O/A FAN	C
1X77-C001A	DG 11 O/A FAN	C
LPCI.D1-AUTO	Division I LPCI Automatic Initiation Logic	C

TITLE: COMPONENT LOCATIONS SHEET NO: 55 OF 211

LPCI.D1-MAN	Division I LPCI Manual Initiation Logic	C
1T46-B001A.R	ESF SWGR RM CLR E EL 119	C
1T46-B001A	ESF SWGR RM CLR E EL 119	C
1T46-B003A	ESF SWGR RM CLR E EL 139	C
1T46-B004A	ESF SWGR RM CLR EL 166	C
1T46-B004A.R	ESF SWGR RM CLR EL 166	C
1T46-B002A	ESF SWGR RM CLR W EL 119	C
1T46-B002A.R	ESF SWGR RM CLR W EL 119	C
1T46-B005A	ESF SWGR RM CLR W EL 139	C
2Z77-F001A.R	EXH FAN INL DAMPER	C
2Z77-F001A	EXH FAN INL DAMPER	C
1Z77-F001A	EXH FAN INL DAMPER	C
1Z77-F001A.R	EXH FAN INL DAMPER	C
1Z77-F002A.R	EXH FAN OUTL DAMPER	C
2Z77-F002A.R	EXH FAN OUTL DAMPER	C
2Z77-F002A	EXH FAN OUTL DAMPER	C
1Z77-F002A	EXH FAN OUTL DAMPER	C
1Z77-F036A	HVAC EQUIP RM EXH	C
2Z77-F036A.R	HVAC EQUIP RM EXH	C
1Z77-F036A.R	HVAC EQUIP RM EXH	C
2Z77-F036A	HVAC EQUIP RM EXH	C
1Z77-F034A	HVAC EQUIP RM SPLY	C
2Z77-F034A.R	HVAC EQUIP RM SPLY	C
2Z77-F034A	HVAC EQUIP RM SPLY	C
1Z77-F034A.R	HVAC EQUIP RM SPLY	C
1E21-F005	LPCS Injection MOV	C
1E21-F011	LPCS Minimum Flow MOV	C
1E21-C001	LPCS Pump	C
1T51-B002	LPCS PUMP ROOM COOLER	C
1E21-F012	LPCS Test Return Line MOV To Supp Pool	C
1E21-F006	LPCS TESTABLE CHK VLV	C
1B21-F019	MS OUTBOARD DRAIN ISOLATION VALVE	C
1B21-F019.R	MS OUTBOARD DRAIN ISOLATION VALVE	C

TITLE: COMPONENT LOCATIONS SHEET NO: 56 OF 211

1E32-F001A	MSIV Leakage Inboard Valve	C
1E32-F001E	MSIV Leakage Inboard Valve	C
1E32-F001J	MSIV Leakage Inboard Valve	C
1E32-F001N	MSIV Leakage Inboard Valve	C
1B21-F051A.R	MSL A SAFETY RELIEF VALVE (ADS)	C
1B21-F051A.D1	MSL A SAFETY RELIEF VALVE (ADS)	C
1B21-F051A.P	MSL A SAFETY RELIEF VALVE (ADS) (SPURIOUS)	C
1B21-F047A.P	MSL A SAFETY RELIEF VALVE (ADS) (SPURIOUS)	C
1B21-F041E.P	MSL A SAFETY RELIEF VALVE (SPURIOUS)	C
1B21-F041A.P	MSL A SAFETY RELIEF VALVE (SPURIOUS)	C
1B21-F051F.R	MSL B SAFETY RELIEF VALVE	C
1B21-F051F.D1	MSL B SAFETY RELIEF VALVE	C
1B21-F051B.R	MSL B SAFETY RELIEF VALVE (ADS)	C
1B21-F051B.D1	MSL B SAFETY RELIEF VALVE (ADS)	C
1B21-F041F.P	MSL B SAFETY RELIEF VALVE (ADS) (SPURIOUS)	C
1B21-F051B.P	MSL B SAFETY RELIEF VALVE (ADS) (SPURIOUS)	C
1B21-F041K.P	MSL B SAFETY RELIEF VALVE (ADS) (SPURIOUS)	C
1B21-F041B.P	MSL B SAFETY RELIEF VALVE (SPURIOUS)	C
1B21-F051K.P	MSL B SAFETY RELIEF VALVE (SPURIOUS)	C
1B21-F051F.P	MSL B SAFETY RELIEF VALVE (SPURIOUS)	C
1B21-F047G.R	MSL C SAFETY RELIEF VALVE	C
1B21-F047G.D1	MSL C SAFETY RELIEF VALVE	C
1B21-F047L.P	MSL C SAFETY RELIEF VALVE (ADS) (SPURIOUS)	C
1B21-F051C.P	MSL C SAFETY RELIEF VALVE (ADS) (SPURIOUS)	C
1B21-F047C.P	MSL C SAFETY RELIEF VALVE (SPURIOUS)	C
1B21-F041C.P	MSL C SAFETY RELIEF VALVE (SPURIOUS)	C
1B21-F041G.P	MSL C SAFETY RELIEF VALVE (SPURIOUS)	C
1B21-F047G.P	MSL C SAFETY RELIEF VALVE (SPURIOUS)	C
1B21-F051D.R	MSL D SAFETY RELIEF VALVE	C
1B21-F047D.R	MSL D SAFETY RELIEF VALVE	C
1B21-F051D.D1	MSL D SAFETY RELIEF VALVE	C
1B21-F047D.D1	MSL D SAFETY RELIEF VALVE	C
1B21-F041D.P	MSL D SAFETY RELIEF VALVE (ADS) (SPURIOUS)	C

1B21-F047D.P	MSL D SAFETY RELIEF VALVE (SPURIOUS)	C
1B21-F051D.P	MSL D SAFETY RELIEF VALVE (SPURIOUS)	C
1B21-F047H.P	MSL D SAFETY RELIEF VALVE (SPURIOUS)	C
SZ51-F009	PLENUM ISOL DAMPER	C
1B21-UR-R623A	POST ACCIDENT MONITORING RECORDER – A SYSTEM	C
1M71-TR-R605A	POST LOCA SUP POOL TEMP 3-PEN RECORDER TR-R605A	C
1M71-TR-R605C	POST LOCA SUP POOL TEMP 3-PEN RECORDER TR-R605C	C
1M71-TE-N012A	POST LOCA SUP POOL TEMP DUAL THERMOCOUPLE TE-N012	C
1M71-TE-N012B	POST LOCA SUP POOL TEMP DUAL THERMOCOUPLE TE-N012	C
1M71-TE-N022A	POST LOCA SUP POOL TEMP DUAL THERMOCOUPLE TE-N022	C
1M71-TE-N022B	POST LOCA SUP POOL TEMP DUAL THERMOCOUPLE TE-N022	C
1M71-TE-N023A	POST LOCA SUP POOL TEMP DUAL THERMOCOUPLE TE-N023	C
1M71-TE-N023B	POST LOCA SUP POOL TEMP DUAL THERMOCOUPLE TE-N023	C
1M71-TE-N024A	POST LOCA SUP POOL TEMP DUAL THERMOCOUPLE TE-N024	C
1M71-TE-N024B	POST LOCA SUP POOL TEMP DUAL THERMOCOUPLE TE-N024	C
1M71-TE-N025A	POST LOCA SUP POOL TEMP DUAL THERMOCOUPLE TE-N025	C
1M71-TE-N025B	POST LOCA SUP POOL TEMP DUAL THERMOCOUPLE TE-N025	C
1M71-TE-N026A	POST LOCA SUP POOL TEMP DUAL THERMOCOUPLE TE-N026	C
1M71-TE-N026B	POST LOCA SUP POOL TEMP DUAL THERMOCOUPLE TE-N026	C
1P41-F241	PSW INBOARD OUTLET FROM ESF ROOM CLRS A	C
1P41-F241.R	PSW INBOARD OUTLET FROM ESF ROOM CLRS A	C
SP41-F066A	PSW INLET TO CR HVAC UNIT A	C
SP41-F074A	PSW OUTLET FROM CR HVAC UNIT A	C
1E51-F064	RCIC STM SPLY DRWL OTBD ISOL	C
1E51-F026	RCIC TURB STM LINE DRAIN (OUTBOARD)	C
1B21-LT-N091A	REACTOR LEVEL TRANSMITTER LT-N091A	C
1B21-PT-N062A	REACTOR PRESSURE TRANSMITTER PT-N062A	C
1P11-F040	REFUEL WTR XFER PMP SUCT FM RWST	C
1P11-F047	REFUEL WTR XFER PMP SUCT FM RWST	C
1E12-FT-N052A	RHR A FLOW INSTRUMENT	C
1E12-F066A	RHR A FUEL POOL CLG ASSIST SUCTION VLV	C
1E12-F042A.R	RHR A INJECTION SHUTOFF VALVE	C
1E12-F042A	RHR A INJECTION SHUTOFF VALVE	C

1E12-F053A.R	RHR A SHUTDOWN CLG RETURN TO FW	C
1E12-F053A	RHR A SHUTDOWN CLG RETURN TO FW	C
1E12-F027A	RHR A SYS SHUTOFF VALVE	C
1E12-F027A.R	RHR A SYS SHUTOFF VALVE	C
1E12-F024A.R	RHR A TEST RTN TO SUPP POOL	C
1E12-F024A	RHR A TEST RTN TO SUPP POOL	C
1E12-F037A.R	RHR A TO CTMT POOL	C
1E12-F037A	RHR A TO CTMT POOL	C
1E12-F040	RHR A TO RADWASTE OUTBOARD SHUTOFF VALVE	C
1E12-F040.R	RHR A TO RADWASTE OUTBOARD SHUTOFF VALVE	C
1E12-F028A.R	RHR CONTAINMENT SPRAY A SPARGER INLET VALVE	C
1E12-F028A	RHR CONTAINMENT SPRAY A SPARGER INLET VALVE	C
1E12-F026A	RHR HX A DISCH TO RCIC	C
1E12-F026A.R	RHR HX A DISCH TO RCIC	C
1E12-F011A	RHR HX A DISCH TO SUPP POOL	C
1E12-F011A.R	RHR HX A DISCH TO SUPP POOL	C
1E12-SV-F075A	RHR HX A OUTBOARD PROC SAMPLE VLV	C
1E12-SV-F075A.R	RHR HX A OUTBOARD PROC SAMPLE VLV	C
1E12-F073A.R	RHR HX A OUTBOARD VENT VALVE	C
1E12-F073A	RHR HX A OUTBOARD VENT VALVE	C
1E12-F048A	RHR HX A SHELL SIDE BYPASS VALVE	C
1E12-F048A.R	RHR HX A SHELL SIDE BYPASS VALVE	C
1E12-F047A	RHR HX A SHELL SIDE INLET VALVE	C
1E12-F047A.R	RHR HX A SHELL SIDE INLET VALVE	C
1E12-F003A	RHR HX A SHELL SIDE OUTLET VALVE	C
1E12-F003A.R	RHR HX A SHELL SIDE OUTLET VALVE	C
1E12-F290A	RHR JOCKEY PUMP A DISCH BLOCK VLV	C
1E12-C002A.R	RHR PUMP 1-A	C
1E12-C002A	RHR PUMP 1-A	C
RHRALOS	RHR Pump A Loss Of Suction Trip	C
1E12-F064A	RHR PUMP A MINIMUM FLOW VALVE	C
1E12-F064A.R	RHR PUMP A MINIMUM FLOW VALVE	C
1E12-F006A.R	RHR PUMP A SHUTDOWN COOLING SUCT VALVE	C

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1E12-F006A	RHR PUMP A SHUTDOWN COOLING SUCT VALVE	C
1E12-F004A.R	RHR PUMP A SUPP POOL SUCTION VALVE	C
1E12-F004A	RHR PUMP A SUPP POOL SUCTION VALVE	C
1T51-B003	RHR ROOM A COOLER FAN	C
1T51-B003.R	RHR ROOM A COOLER FAN	C
1E12-F008.R	RHR SHUTDN CLG OUTBD SUCTION VALVE	C
1E12-F008	RHR SHUTDN CLG OUTBD SUCTION VALVE	C
1E12-PT-N058A	RPV Pressure Transmitter	C
1B33-F020	RW PROC SMPL DRWL OUTBD ISOL	C
1B33-F020.R	RW PROC SMPL DRWL OUTBD ISOL	C
1G33-F034	RWCU BLWDN CTMT OTBD ISOL	C
1G33-F034.R	RWCU BLWDN CTMT OTBD ISOL	C
1G33-F250	RWCU SPLY TO RWCU HXS	C
1Z77-B001A.R	SFGD SWGR - BTRY RM AH UNIT A	C
2Z77-B001A.R	SFGD SWGR - BTRY RM AH UNIT A	C
1Z77-B001A	SFGD SWGR - BTRY RM AH UNIT A	C
2Z77-B001A	SFGD SWGR - BTRY RM AH UNIT A	C
2Z77-C001A.R	SFGD SWGR - BTRY RM EXH FAN A	C
1Z77-C001A	SFGD SWGR - BTRY RM EXH FAN A	C
2Z77-C001A	SFGD SWGR - BTRY RM EXH FAN A	C
1Z77-C001A.R	SFGD SWGR - BTRY RM EXH FAN A	C
1P60-F003	SPCU SPLY TO CNDS PC FLTRS	C
2Z77-F003A.R	SPLY FAN INL DAMPER	C
1Z77-F003A	SPLY FAN INL DAMPER	C
1Z77-F003A.R	SPLY FAN INL DAMPER	C
2Z77-F003A	SPLY FAN INL DAMPER	C
1Z77-F035A	SPLY FAN OUTL DAMPER	C
1Z77-F035A.R	SPLY FAN OUTL DAMPER	C
2Z77-F035A.R	SPLY FAN OUTL DAMPER	C
2Z77-F035A	SPLY FAN OUTL DAMPER	C
1P42-F201A	SSW A RTN FM FPHX A	C
1P42-F200A	SSW A SPLY TO FPHX A	C
1P41-F007A	SSW BASIN A TRANSFER VLV	C

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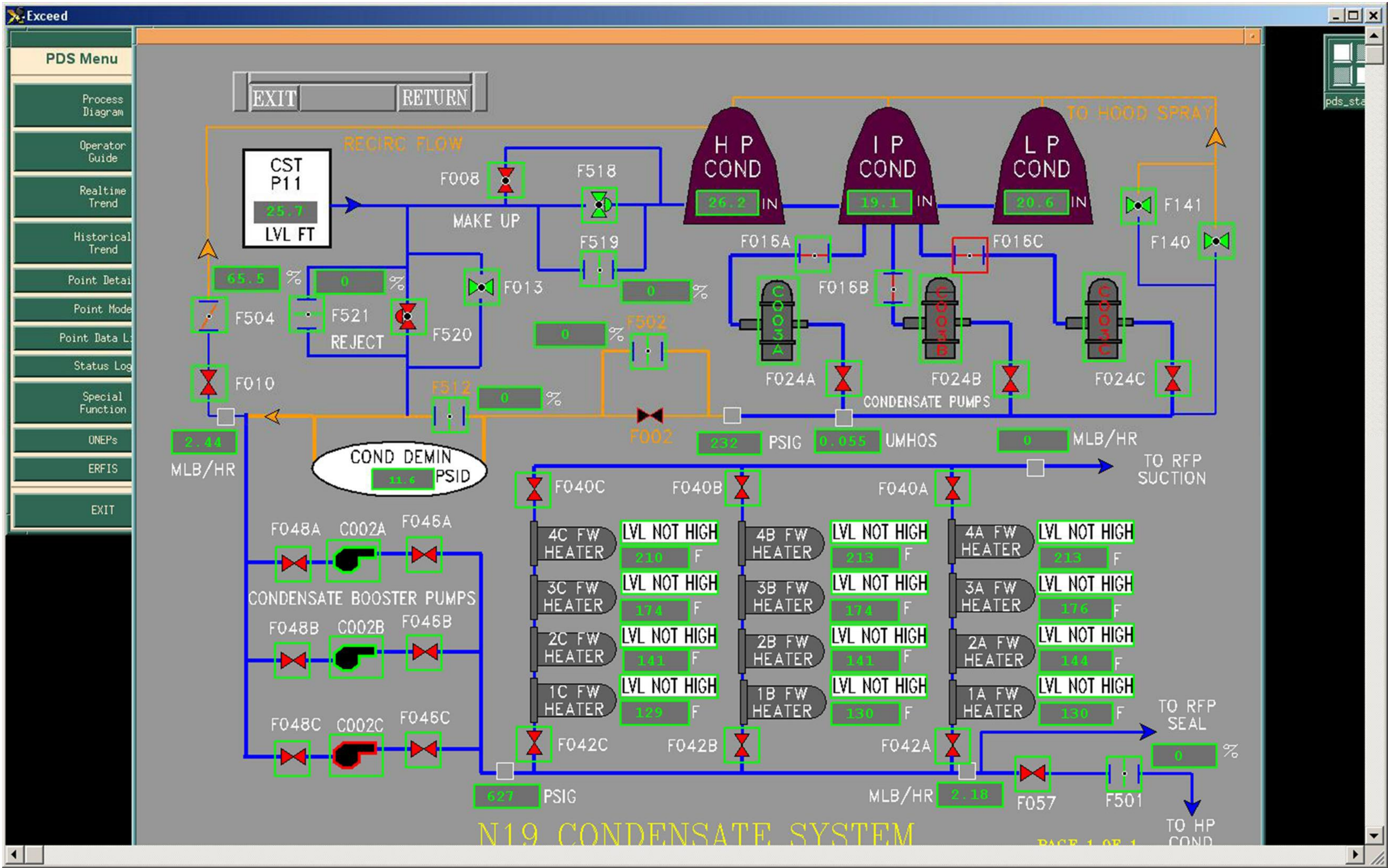
1P41-F007A.R	SSW BASIN A TRANSFER VLV	C
1P41-C003A	SSW COOLING TOWER FAN A	C
1P41-C003A.R	SSW COOLING TOWER FAN A	C
1P41-C003B	SSW COOLING TOWER FAN B	C
1P41-C003B.R	SSW COOLING TOWER FAN B	C
SP41-F064A	SSW INL TO CR A/C A (LOOP A)	C
1P41-F014A	SSW INLET TO RHR HX A	C
1P41-F014A.R	SSW INLET TO RHR HX A	C
1P41-F018A	SSW LOOP A INLET TO DG 11 WATER COOLER	C
1P41-F018A.R	SSW LOOP A INLET TO DG 11 WATER COOLER	C
1P41-F237	SSW LOOP A INLET TO ESF ROOM CLRS A	C
1P41-F237.R	SSW LOOP A INLET TO ESF ROOM CLRS A	C
1P41-F238	SSW LOOP A OUTLET FROM ESF ROOM CLRS A	C
1P41-F238.R	SSW LOOP A OUTLET FROM ESF ROOM CLRS A	C
1P41-F005A.R	SSW LOOP A RETURN TO COOLING TOWER A	C
1P41-F005A	SSW LOOP A RETURN TO COOLING TOWER A	C
SP41-F081A	SSW OUTL FM CR A/C A (LOOP A)	C
1P41-F068A.R	SSW OUTLET FROM RHR HX A	C
1P41-F068A	SSW OUTLET FROM RHR HX A	C
1Y47-C001A	SSW PMPHS A O/A FAN	C
1Y47-C001A.R	SSW PMPHS A O/A FAN	C
1P41-C001A.R	SSW PUMP A	C
1P41-C001A	SSW PUMP A	C
1P41-UR-R606A	SSW PUMP A DISCH PRESSURE RECORDER	C
1P41-F001A.R	SSW PUMP A DISCH VLV	C
1P41-F001A	SSW PUMP A DISCH VLV	C
1P41-F016A	SSW PUMP A INBOARD BLOWDOWN VLV	C
1P41-F016A.R	SSW PUMP A INBOARD BLOWDOWN VLV	C
1P41-F006A	SSW PUMP A RECIRC VLV	C
1P41-F006A.R	SSW PUMP A RECIRC VLV	C
1P75-E001A	STANDBY DIESEL GENERATOR ENGINE E001A	C
1P75-E001A.R	STANDBY DIESEL GENERATOR ENGINE E001A	C
SSW.D1-MAN	Standby Service Water Loop A Manual Initiation	C

GGNS LOT 2015 NRC INITIAL LICENSED OPERATOR WRITTEN EXAMINATION
OPEN-REFERENCES TABLE OF CONTENTS
RO EXAM

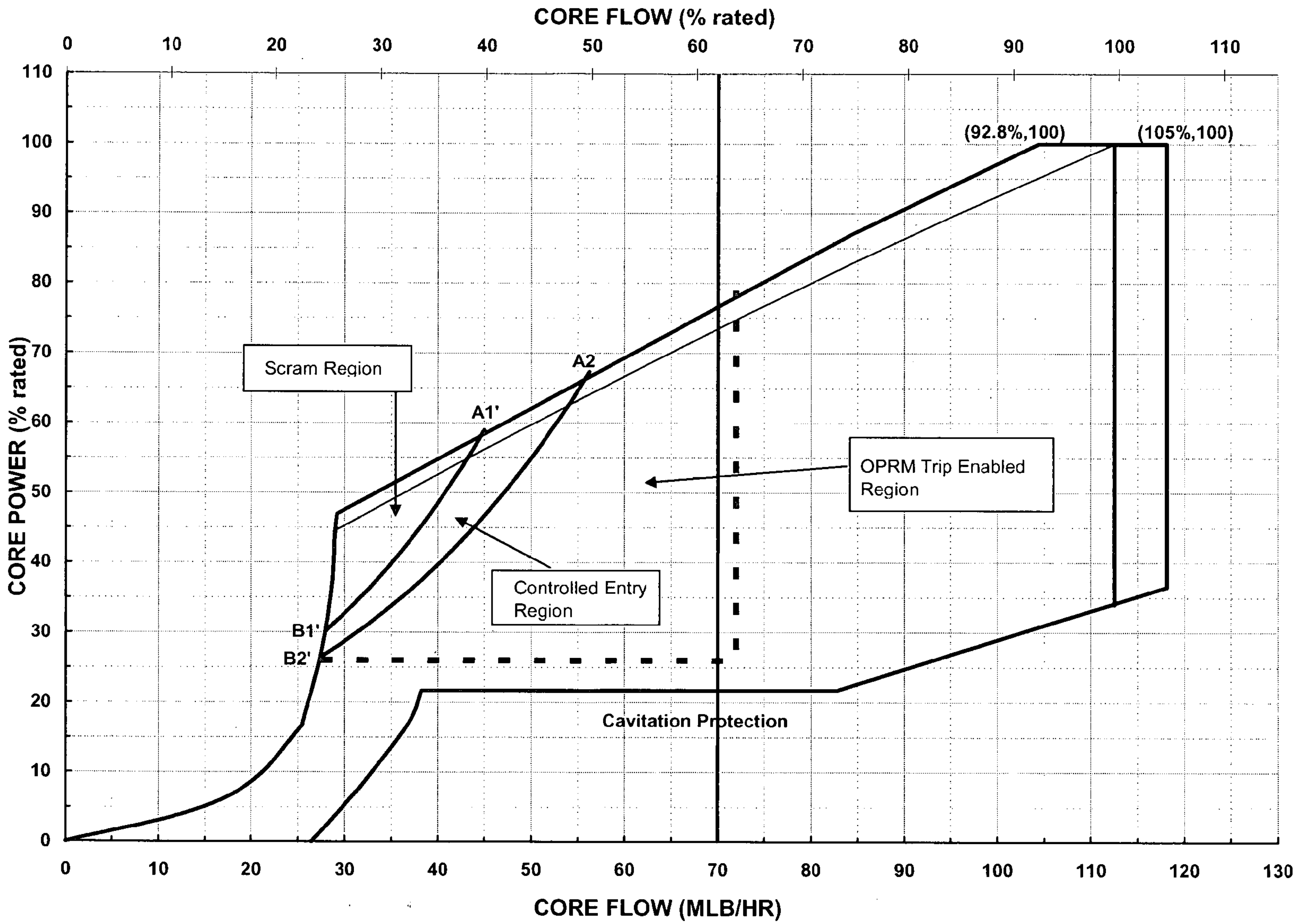
TAB 3


Exceed
PDS Menu
Process Diagram
Operator Guide
Realtime Trend
Historical Trend
EXIT
RETURN
RECIRC FLOW
CST P11
25.7
LVL FT
F008
F518
MAKE UP
F519
H P COND
26.2
IN
I P COND
19.1
L P COND
20.6
TO HOOD SPRAY
F141
F140
F016A
F016B
F016C
F504
65.5
F521
REJECT
F520
F013
F502
F002
F512
F010
2.44
MLB/HR
COND DEMIN
11.6
PSID
F024A
F024B
F024C
CONDENSATE PUMPS
232
PSIG
0.055
UMHOS
0
TO RFP SUCTION
F040C
F040B
F040A
4C FW HEATER
4B FW HEATER
4A FW HEATER
3C FW HEATER
3B FW HEATER
3A FW HEATER
2C FW HEATER
2B FW HEATER
2A FW HEATER
1C FW HEATER
1B FW HEATER
1A FW HEATER
LVL NOT HIGH
210
213
174
176
141
144
129
130
F
F048A
C002A
F046A
CONDENSATE BOOSTER PUMPS
F048B
C002B
F046B
F048C
C002C
F046C
F042C
F042B
F042A
627
2.18
F057
F501
TO RFP SEAL
TO HP COND
N19 CONDENSATE SYSTEM

TAB 4

Figure 1

GGNS C19 EPU Power/Flow Map



GGNS LOT 2015 NRC INITIAL LICENSED OPERATOR WRITTEN EXAMINATION

SRO EXAM

ANSWER KEY

76	C
77	D
78	B
79	C
80	A
81	C
82	B
83	A
84	B
85	C
86	D
87	C
88	B
89	D
90	B
91	A
92	D
93	B
94	C
95	B
96	A
97	D
98	A
99	B
100	C

Examination Outline Cross-Reference	Level	SRO
295001 Partial or Complete Loss of Forced Core Flow Circulation G2.4.8: Knowledge of how abnormal operating procedures are used in conjunction with EOPs.	**Tier #**	1
	Group #	1
	K/A #	**295001 – 2.4.8**
	Rating	4.5

Question 76

The plant is operating at 100% power.

Due to a completed RCIC surveillance, suppression pool water level is 18.87 ft.

EP-3 has been entered.

Reactor Feedwater pump 'A' trips on overspeed.

- Core flow 58.2 Mlbm/hr
- Reactor power 72 %

Which of the following actions should be directed NEXT by the SRO in order of priority?

A. 1. Lower Suppression Pool level using P11 SOI
   2. Insert control rods to restore power-flow conditions as soon as practical.

B. 1. Insert control rods to restore power-flow conditions as soon as practical.
   2. Lower Suppression Pool level using E51 SOI

C. 1. Insert control rods to restore power-flow conditions as soon as practical.
   2. Lower Suppression Pool level using P11 SOI

D. 1. Lower Suppression Pool level using E51 SOI
   2. Insert control rods to restore power-flow conditions as soon as practical

Answer: C
Explanation: With suppression pool water level at 18.87 ft. EOP 3 is entered at >18.81Ft. A loss of one Feedwater pump will cause a Reactor Recirc FCV runback which requires entry into the Reduction in

Recirculation System Flow Rate ONEP to ensure position on the Power to Flow map. Procedure requires use of Emergency Procedures and ONEPs concurrently. With the given parameters the plot on the power to flow map will be above the MEOD boundary. Currently EP-3 was entered due to Suppression Pool High level, however the Reduction in recirc ONEP is also entered in conduction with the EP. Being above the MEOD boundary is an unanalyzed condition therefore it will take priority over high suppression pool level. Subsequent actions of the ONEP give the correct direction to mitigate plant actions. Subsequent action are performed only at the direction of the SRO. A is wrong because, the steps are correct but listed in the wrong priority, Operation in the MEOD boundary takes priority. Plausible due to the SRO must make a priority decision on the next step for plant operation. B is wrong because Lowering Suppression pool level should be performed using normal plant systems (i.e. P11 system) not RCIC. Plausible due to first part is correct but should be lower on the priority list and RCIC (E51) should not be used to lower Supp. Pool level, normal plant systems should be used first. C is correct. D is wrong because lowering Suppression pool level should be performed using normal plant systems (i.e. P11 system) not RCIC, Plausible due to RCIC (E51) should not be used to lower Supp. Pool level, normal plant systems should be used first, second part is also a subsequent step but should be lower on the priority list		
SRO Only (see attached flow chart): The SRO should A. assess plant conditions and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed B. have the knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal and emergency procedures.		
Technical References: 05-1-02-III-3, Reduction in Recirculation System Flow Rate ONEP EP-3		
References to be provided to applicants during exam: Figure 1 of 05-1-02-III-3, Reduction in Recirc System Flow Rate ONEP		
Learning Objective: Document learning objective if possible. GLP-OPS-ONEP GLP-OPS-EP01		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	

	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(7) (10)	
	55.43(b)(5)	

Examination Outline Cross-Reference	Level	SRO
295016 Control Room Abandonment AA2-Ability to determine and/or interpret the following as they apply to control room abandonment: AA2.02 Reactor water level	Tier #	1
	Group #	1
	K/A #	**295016 - AA2.02**
	Rating	4.3

Question 77

The control room was abandoned due to a fire in progress in the control room.

In order to ensure adequate core cooling by the most preferred method (restoring reactor water level above top of active fuel), the CRS enters ONEP 05-1-02-II-1, Shutdown From the Remote Shutdown Panel.

What action does the CRS take to ensure adequate core cooling?

A. Complete Attachment XX, Control Room Fire Operator Actions (Div 1 DG & 15AA Actions) within a maximum of 12 minutes of the reactor scram

B. Dispatch a reactor operator to complete Attachment XX, Control Room Fire Operator Actions (Div 1 DG & 15AA Actions) within a maximum of 14.3 minutes of the reactor scram

C. Dispatch a reactor operator to complete Attachment XXI – Control Room Fire Operator Actions (RHR A Injection to Reactor) within a maximum of 12 minutes of the reactor scram

D. Complete Attachment XXI – Control Room Fire Operator Actions (RHR A Injection to Reactor) within a maximum of 14.3 minutes of the reactor scram

Answer: D
Explanation: The CRS will go to the RSP and monitor reactor water level there for restoration via Attachment XXI (once injection has been established with RHR and after the ED is completed). Both of these items are performed by the CRS (and FSS) at the RSP using Attachment XXI within a maximum of 14.3 minutes

of the reactor scram per the analysis discussed in this procedure. Therefore D is correct.

A. is wrong because the CRS does not perform this attachment.
B. Is wrong because the max time is incorrect. The CRS does dispatch an operator to complete this attachment, but its max time is 12 minutes, not 14.3 minutes
C. Is wrong because the CRS completes this attachment, not another operator, but the time is also wrong.

SRO Only (see attached flow chart):

The SRO should

1. assess plant conditions and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
2. have knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures.

Technical References:

05-1-02-II-1, Shutdown From the Remote Shutdown Panel, step 1.17

References to be provided to applicants during exam:

None

Learning Objective:

GLP-OPS-ONEP Obj 53
GLP-OPS-EP2 Obj 22

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	H
	LOD	3
10CFR Part 55 Content:	55.41(b)(7)(10)	
	55.43(b)(5)	

Examination Outline Cross-Reference	Level	SRO
295019 Partial or Total Loss of Inst. Air G2.4.11: Knowledge of abnormal condition procedures.	**Tier #**	1
	Group #	1
	K/A #	**295019 – 2.4.11**
	Rating	4.2

Question 78

If an inadvertent CTMT/AUX building isolation occurs due to a power loss which was subsequently restored, the CRS should direct the crew to perform which procedure **immediately** per Operations Philosophy 02-S-01-27?

A. ONEP 05-1-02-I-4, Loss of AC power

B. ONEP 05-1-02-V-9, Loss of Instrument Air

C. ONEP 05-1-02-III-5, Automatic isolations

D. ONEP 05-1-02-V-11, Loss of Plant Service Water

Answer: B
Explanation: Step 6.5.2 of OPs philosophy document 02-S-01-27_61, states the following: "if an inadvertent CTMT/Aux building isolation occurs due to a power loss, then immediately restore instrument air per the loss of IA ONEP 5-1-02-V-9 and other systems per the ONEP for Automatic isolations, 5-1-02-III-5. per 05-1-02-I-4_46 Loss of AC power ONEP, step 3.1.2, "if affected bus is re-energized, then recover systems as follows: a. Restore IA and plant SW per Att II of 05-1-02-III-5 Automatic Isolations then b. Restore other systems per 05-1-02-III-5" B is correct. A is wrong, but plausible because it is performed for loss of power, just not first C is wrong but plausible because it is performed for this event, just not first D is wrong but plausible because it could be performed since plant SW is lost, just not first
Technical References:

02-S-01-27_61, OPs Philosophy, step 6.5.2 05-1-02-V-9, Loss of Instrument Air 05-1-02-III-5, Automatic Isolations 05-1-02-I-4, Loss of AC Power 05-1-02-V-11, Loss of Plant Service Water
SRO Only (see attached flow chart): The SRO should A. assess plant conditions and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed.
References to be provided to applicants during exam: NONE
Learning Objective: Document learning objective if possible. GLP-OPS-ONEP

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	H
	LOD	3
10CFR Part 55 Content:	55.41(b)(7)(10)	
	55.43(b)(5)	

Examination Outline Cross-Reference	Level	SRO
295021 Loss of Shutdown Cooling	**Tier #**	1
	Group #	1
G2.2.38: Knowledge of conditions and limitations in the facility license.	**K/A #**	**295021 2.2.38**
	Rating	4.5

Question 79

What is the documented reason in the Technical Specification Bases that the Residual Heat Removal System is included in the Technical Specifications for Section 3.4.10, Residual Heat Removal (RHR) Shutdown Cooling System-Cold Shutdown, for a loss of shutdown cooling event?

A. It is required for mitigation of an accident in the safety analysis in the USAR

B. It is required to meet the GDCs in the Code of Federal Regulations

C. It is required for risk reduction against core damage

D. It is required to meet the applicable criterion within the NRC Policy Statement

Note: GDC is General Design Criteria
USAR is Updated Safety Analysis Report

Answer: C
Explanation: Per TSB for 3.4.10, page 55, in the applicable safety analysis, that RHR is required for this section because: "Decay heat removal by the RHR System in the shutdown cooling mode is not required for mitigation of any event or accident evaluated in the safety analyses. Loss of RHR shutdown cooling has been evaluated and found not to result in adverse consequences since adequate alternate decay heat removal methods remain available. Decay heat removal is, however, an important safety function that must be accomplished or core damage could result. Although the RHR Shutdown Cooling System does not meet a specific criterion of the NRC Policy Statement, it was identified in the NRC Policy Statement as a significant contributor to risk

reduction. Therefore, the RHR Shutdown Cooling System is retained as a Technical Specification.

C is correct

A is wrong – see explanation from TSB above.
B is wrong-see above explanation
D is wrong-see above explanation

SRO Only (see attached flow chart):

The SRO should

A. asses plant conditions and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
B. have knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures.

Technical References:
TSB for 3.4.10, Revision LDC 99050, page 55.

References to be provided to applicants during exam:

None

Learning Objective: Document learning objective if possible.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	H
	LOD	3
10CFR Part 55 Content:		
	55.43(b)(1)	

Examination Outline Cross-Reference	Level	SRO
295030 Low Suppression Pool Water Level G2.4.18: Knowledge of the specific bases for EOPs	**Tier #**	1
	Group #	1
	K/A #	**295030 – 2.4.18**
	Rating	4.0

Question 80

Suppression pool level is currently 14.7 ft. and lowering.

CRS is currently directing actions from EP-3.

All actions of EP-3 have been performed or attempted to be performed to raise Suppression Pool level.

(1) What is the next action that should be performed once EP-2 has been entered?

(2) What is the basis for this action?

A. (1) Place Mode Switch to shutdown and enter Emergency Depressurization
(2) Ensure that the RPV is <u>not</u> permitted to remain at pressure if pressure suppression capability is unavailable.

B. (1) Place Mode Switch to shutdown and Reduce pressure band to 400 psig to 650 psig.
(2) Ensure that the RPV is <u>not</u> permitted to remain at pressure if pressure suppression capability is unavailable.

C. (1) Place Mode Switch to shutdown and enter Emergency Depressurization
(2) Maintain adequate NPSH for ECCS systems and RCIC.

D. (1) Place Mode Switch to shutdown and Reduce pressure band to 400 psig to 650 psig.
(2) Maintain adequate NPSH for ECCS systems and RCIC.

Answer: A
Explanation: See EP Tech Bases, Attachment VI, pages 30-31. Step SPL-9's level of 14.5 feet is all about being just ~2 feet above the top of the horizontal vents. The concern is that should a LOCA occur, steam

discharged through the vents may not be adequately condensed. The language in the page 31 discussion is more direct…this is where we translate this to a situation where “pressure suppression capability is unavailable.”

A is correct
B is wrong - ED is required per EPs if Supp Pool cannot be maintained above 14.5 ft.
C is wrong - RCIC has a precaution not to operate RCIC at <14.6ft supp pool level for NPSH concerns, however, this is not the basis for ED on low level.
D is wrong - ED is required per EPs if Supp Pool cannot be maintained above 14.5 ft., and RCIC has a precaution not to operate RCIC at <14.6ft supp pool level for NPSH concerns, however, this is not the basis for ED on low level

SRO Only (see attached flow chart):

The SRO should:
- assess plant conditions and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- have knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures.
- knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures.

Technical References:

EP-2, RPV Control
EP-3, Containment Control
02-S-01-40, EP Technical Bases, Rev 6

References to be provided to applicants during exam:

None.

Learning Objective: Document learning objective if possible.

GLP-OPS-EP3, Objective 7

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.43(b)(5)	

Examination Outline Cross-Reference	Level	SRO
295038 High Off-site Release Rate **Ability to determine and/or interpret the following as they apply to high off-site release rate:** AA2.04: Source of off-site release	**Tier #**	1
	Group #	1
	K/A #	**295038 AA2.04**
	Rating	4.3

Question 81

An ATWS is in progress with reactor water level band is being maintained per EP-2A at -70” to -130” wide range.

A Group 1 isolation signal has been received on Main Steam Tunnel Temperature.

B21-F022A, MSL “A” DRWL INBD ISOL, valve has failed to close.

Dose Assessor has completed offsite dose calculations and reports site boundary dose indicates a release in progress.

Which of the following indicates the source of the offsite release?

A. Standby Gas Treatment effluent

B. Containment Vent Exhaust effluent

C. Auxiliary Building Roof blowout panels (unmonitored)

D. Turbine Building Roof (unmonitored)

Answer: C
Explanation: With Main steam tunnel (MST) high temperature and group 1 isolation the evidence of a steam line break in the Main steam tunnel. With one inboard MSIV failing to close causes a continuous supply through the break. A break in the MST will cause the safety blowout panels to open causing an unmonitored release to the Auxiliary Building roof. A is wrong - the SBGT system limits the Total Effective Dose Equivalent (TEDE) to within guidelines of 10CFR50.67 at the site boundary and the low population zone, therefore the release could not be coming from SBGT.

B is wrong - the containment vent exhaust will auto isolate at -41.6”, and with reactor water level being maintained at -70 to -130 this system has already isolated.

C is correct.

D is wrong - the outboard MSIVs have all closed therefore no steam is not entering the turbine building, plausible due to the turbine building exhaust has a system that has unmonitored release point from the Mechanical vacuum pumps.

SRO Only (see attached flow chart):

SRO only task

The SRO should:

- assess plant conditions and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- have knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures.
- have knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures.

Technical References:

EP-2A
10-S-01-1
10-S-01-12

References to be provided to applicants during exam:

None

Learning Objective: Document learning objective if possible.

GLP-OPS-EPTS6
GLP-OPS-D1721 OBJ., 12.2

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(7)	
	55.43(b)(4)	

Examination Outline Cross-Reference	Level	SRO
700000 Generator Voltage and Electric Grid Disturbances / 6 **Ability to determine and/or interpret the following as they apply to generator voltage and electric grid disturbances:** AA2.06: Generator frequency limitations	Tier #	1
	Group #	1
	K/A #	**295037 AA2.06**
	Rating	3.5

Question 82

The plant is operating at rated power when grid instabilities cause the following:

Day 1, 0000 hours	Baxter Wilson Line voltage reaches 529 KV and remains there Franklin Line voltage reaches 515 KV and remains there
Day 1, 0100 hours	Port Gibson Line voltage reaches 118 KV and remains there
Day 1, 0200 hours	Main Generator frequency matches grid frequency at 62 Hz
Day 1, 0300 hours	Port Gibson Line voltage reaches 124 KV and remains there

The Plant AC/DC Weekly Lineup (06-OP-1R20-W-0001) has been performed.

Per Tech Specs, if none of the above conditions change, the plant must be in MODE 3 no later than ______________.

A. Day 2 at 1400 hours

B. Day 2 at 1500 hours

C. Day 4 at 1400 hours

D. Day 4 at 1500 hours

Answer: B

Explanation:
See 06-OP-1R20-W-0001, Attachment I, page 2.

At Day 1, 0000 hours, Baxter Wilson line volts goes out of spec high (limit is 525 KV). We now have only two offsite circuits OPERABLE (Franklin and Port Gibson), but only two are required (Tech Spec LCO 3.8.1; therefore, no Tech Spec action yet applies.

At Day 1, 0100 hours, nothing has changed because the Port Gibson line volts are still in spec (limit is 120.75).

At Day 1, 0200 hours, With main generator frequency matching grid frequency Franklin line frequency goes out of spec high (limit is 61.9 Hz). We now have one of two remaining required offsite circuits inoperable. Per Tech Spec Action 3.8.1.A.2, we have 72 hours to restore one to OPERABLE; otherwise, be in MODE 3 within 12 hours thereafter (per Tech Spec Action 3.8.1.G.2). This would be no later than 84 hours after Day 1, 0200 hours; i.e., Day 4, 1400 hours.

At Day 1, 0300 hours, we now have zero offsite circuits OPERABLE because Port Gibson volts has gone out of spec high. Per Tech Spec Action 3.8.1.C.2, we have 24 hours to restore one to OPERABLE; otherwise, be in MODE 3 within 12 hours thereafter (per Tech Spec 3.8.1.G.2). This would be no later than 36 hours after Day 1, 0300 hours; i.e., Day 2, 1500 hours.

SRO Only (see attached flow chart):

Requires SRO to determine grid frequency controls generator frequency.

The SRO should:

- apply required actions and surveillance requirements in accordance with the rules of application requirements.
- have knowledge of TS bases that is required to analyze TS required actions and terminology.

Technical References:

06-OP-1R20-W-0001, Plant AC/DC Power Distribution Weekly Lineup
Tech Spec 3.8.1, AC Sources - Operating

References to be provided to applicants during exam:
Tech Spec 3.8.1, AC Sources - Operating, remove the LCO section

Learning Objective:
GLP-OPS-TS001, Objective 39

Question Source:	Bank #273	2014 NRC Q. 82
(note changes; attach parent)	Modified Bank #	X
	New	
Question History:	Last NRC Exam	**Yes Q.82**
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(7)	
	55.43(b)(2)	

Examination Outline Cross-Reference	Level	SRO
295011 High Containment Temperature (Mark III Containment Only) **Ability to determine and/or interpret the following as they apply to high Containment temperature:** AA2.02 - Containment Pressure	**Tier #**	1
	Group #	2
	K/A #	**295011 - AA 2.02**
	Rating	4.0

Question 83

A LOCA and ATWS has occurred.

5 minutes later:

- Reactor Power is 4% and lowering
- Standby Liquid Control is injecting
- RHR ‘A’ and ‘B’ systems are in Suppression Pool Cooling
- Reactor Water Level is -180” Compensated Fuel Zone, in band of -167” to -191”
- Suppression Pool level is 18.5 feet and rising
- Suppression Pool temperature is 120°F
- Containment pressure is 8.0 psig and rising
- Containment temperature is 155°F
- Drywell temperature is 215°F

The CRS should **immediately** direct operators to:

A. Initiate Containment Spray

B. Emergency Depressurize

C. Terminate and prevent all injection and lower level band

D. Initiate Containment Purge using attachment 14

Answer: A
Explanation: The SRO is required to interpret Containment Temp vs. Containment pressure to determine the

next step to mitigate the event. Per EP3, Containment Spray should have been initiated prior to entering the unsafe zone of PSP, However the step in the EPs states If CTMT pressure cannot be restored and maintained in the Safe zone of the PSP then Emergency Depressurization is required. First the SRO should verify that he is within the SAFE TO INITIATE area of the CSIPL (Containment Spray Initiation Pressure Limit). With a CTMT pressure of 8 psig and CTMT temperature of 155°F is well within the Safe To Initiate zone. With the statement of “restore and maintain” in the PSP limit, it can be exceeded only if you can restore it within limits. Both RHR systems are available for Containment Spray due to they are currently in Suppression Pool Cooling.

A is correct

B is wrong - An Emergency Depressurization is not required due to the step states to “restore and maintain” which means that the SRO should perform CTMT spray first before an Emergency Depressurization. Plausible if the student does not remember that the step states to restore and maintain.

C is wrong - This step has already been performed due to the level band given is already at the lowered band due to CTMT, Drywell, and reactor parameters from EP-2A. Plausible due to all criteria is met for the lowered band but if student does not recognize that the band has already been lowered.

D is wrong - Containment Purge using Attachment 14 is used after entry into EP-3 to control Containment pressure below 1.23 psig only. The given parameters show the containment pressure is well above the 1.23 psig and other strategies should be used. Plausible due to this step is performed after entry into EP-3.

SRO Only (see attached flow chart):

The SRO should:

- assess plant conditions and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- have knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures.
- have knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures.

Technical References:

EP2 & EP3

References to be provided to applicants during exam:

EP-1 figures Containment Spray Initiation Pressure Limit (CSIPL) and Pressure Suppression Pressure (PSP)

Learning Objective: Document learning objective if possible.

GLP-OPS-EP001

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(7)	
	55.43(b)(4)	

Examination Outline Cross-Reference	Level	SRO
295022 Loss of CRD Pumps **Ability to determine and/or interpret the following as they apply to loss of CRD pumps:** AA2.03: CRD mechanism temperatures	**Tier #**	1
	Group #	2
	K/A #	**295022 – AA2.03**
	Rating	3.2

Question 84

The plant is in Mode 2 startup at 950 psig.

The operating CRD pump trips.

A delay in starting the standby CRD pump causes the following alarm.

- CRD HYD TEMP HI, P680-4A2-A4

Local operator reports 4 CRDMs at ≥ 450°F and steady.

Reactor and System Engineering have been notified and reports the following control rods will have time added to their scram times:

Revised time to notch 13 @ 950 psig

- 32-41 1.45
- 40-29 1.51
- 44-33 1.41
- 52-37 1.41

Which of the following would require entry into Tech Spec 3.1.4, Control Rod Scram Times?

What is the Bases for this action?

A. 5 more Control rods are determined to be slow

Limit the potential amount of reactivity addition that could occur in the event of a control rod drop accident.

B. Control Rod 56-37 is determined to be slow

To ensure that local scram reactivity rates are maintained within acceptable limits.

C. 5 more Control rods are determined to be slow

To ensure that local scram reactivity rates are maintained within acceptable limits

D. Control Rod 56-37 is determined to be slow

Limit the potential amount of reactivity addition that could occur in the event of a control rod drop accident

Answer: B
Explanation: All 4 control rods are above the required scram time per tech spec 3.1.4, table 3.1.4-1 is required to have a time of <1.40 seconds. the LCO states No more than 14 operable control rods shall be slow in accordance with the table, and No operable control rod that is slow shall occupy a location adjacent to another Operable control rod that is slow. Since there is only 4 that are now considered to be slow it does not meet the 14 criteria, however control rods 56-37 and 52-37 are adjacent and that meets the b. statement of the LCO. The TS Bases for 3.1.4 states that “To ensure that local scram reactivity rates are maintained (continued) within acceptable limits, no "slow" control rod may occupy a location adjacent to another "slow" control rod.” A is wrong - 3.1.4 requires more than 14 slow control rods not 5 and the bases is from the Control Rod Pattern control. Plausible due to 9 or more control rods out of sequence, the control rod pattern significantly deviates from the prescribed sequence. Also, 9 or more control rods scram would require placing the mode switch to shutdown. If the student confuses the total number of slow control rods vs. the total number of control rods out of sequence or scrammed. B is correct C is wrong - 3.1.4 requires more than 14 slow control rods not 5 D is wrong - the bases is from the Control Rod Pattern control.
SRO Only (see attached flow chart): The SRO should: • apply required actions and surveillance requirements in accordance with the rules of application requiredments. • have knowledge of TS bases that is required to analyze TS required actions and terminology.
Technical References: 04-1-02-1H13-P680-4A2-A4, Alarm Response Instruction Tech Specs 3.1.4
References to be provided to applicants during exam: **NONE**
Learning Objective:

GLP-OPS-TS001		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(6)…	
	55.43(b)(2)	

Examination Outline Cross-Reference	**Level**	SRO
295029 High Suppression Pool Wtr Lvl G2.4.6: Knowledge of EOP mitigation strategies	**Tier #**	1
	Group #	2
	K/A #	**295029 2.4.6**
	Rating	4.7

Question 85

The unit was operating at full power when a loss of coolant accident occurred.

The CRS entered EP-2, the reactor was scrammed and all control rods inserted completely.

The Reactor Mode switch was been placed in SHUTDOWN but suppression pool water level is continuing to rise.

Suppression pool makeup and drains were placed out of service earlier in the day for maintenance.

RCIC and RHR A and B are out of service.

With Suppression Pool water level at 19.0 feet and rising, the CRS should _____?

A. Immediately Emergency Depressurize from EP-2 and concurrently enter EP-3

B. Enter EP-3 and Emergency Depressurize from EP-3

C. Enter EP-3 and use HPCS SOI 04-1-01-E22 to lower level

D. Enter EP-3 and use RHR ‘C’ SOI 04-1-01-E12 to lower level.

Answer: C
Explanation: EP-3 is entered at 18.81’ supp. pool level, EP-3 directs to lower suppression pool level by using one of the following systems. • normal Suppression pool drains • HPCS • RCIC • RHR All but HPCS is out of service therefore HPCS is the only way to reduce level in the SP. A is wrong - Depressurization is not required until SP level cannot be restored and maintained below 24.6ft. B is wrong - Emergency Depressurization is not required and if so would be from EP-2 not EP-3 C is correct - see explanation above D is wrong - RHR ‘C’ could be used to lower SP level if operating in Emergency Supp. Pool cleanup mode, this cannot be done due to LOCA signal. E12 SOI only describes using RHR A which is not available.
SRO Only (see attached flow chart): The SRO should: • assess plant conditions and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed • have knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps. • have knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures.
Technical References: EP5 05-S-01-EP-1, Emergency/Severe Accident Procedure Support Documents 02-S-01-40, EP Technical Bases
References to be provided to applicants during exam: **NONE**
Learning Objective: Document learning objective if possible.

GLP-OPS-EP01		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(10)…	
	55.43(b)(5)	

Examination Outline Cross-Reference	Level	SRO
209001 LPCS **Ability to (a) predict the impacts of the following on the low pressure core spray system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** A2.02: Valve closures	**Tier #**	2
	Group #	1
	K/A #	**209001 – A2.02**
	Rating	3.2

Question 86

The plant is operating at 100% power.

Division 2 Emergency Diesel Generator has been INOP for 24 hours due to a failed surveillance.

Tech Spec 3.8.1 Condition B was entered and associated requirements are being performed.

The BOP operator notices the E21-F011, LPCS MIN FLO TO SUPP POOL, valve going closed.

Investigation determines that the LPCS discharge flow transmitter has failed upscale.

Based on the above information LPCS is required to be declared inoperable _________ because ____________.

A. within 4 hours, LPCS pump overheating may occur on pump start at high RPV pressure

B. within 4 hours, due to RHR 'B' and 'C' also being inoperable

C. immediately, due to RHR 'B' and 'C' also being inoperable

D. immediately, LPCS pump overheating may occur on pump start at high RPV pressure

Answer: D
Explanation:

With only the DG INOP no other required features are required to be declared INOP as long as the redundant features are operable.

With the min flow valve failing, SOI 04-1-01-E21-1 step 3.15 states "**If** the min flow valve Will **NOT** perform its intended function the LPCS system is declared inop." per Tech Spec Bases this would cause LPCS to be declared INOP immediately. LPCS is a redundant required feature.

Per Tech Spec 3.8.1 Condition B.2 states "Declare required feature(s), supported by the inoperable DG, inoperable when the redundant required feature(s) are inoperable." The completion time states "4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)",

When LPCS is declared INOP within 4 hours RHR 'B' and 'C', which are required features supported by the inop DG must be declared inop.

A is wrong – because the LPCS system is required to be declared inoperable immediately. Plausible because the LPCS is required to be declared inoperable for the reason given

B is wrong - because the LPCS system is required to be declared inoperable immediately. Plausible because both RHR "B' and 'C' will also become inoperable in 4 hours.

C is wrong - due to RHR 'B' and 'C' being inoperable is not the reason for immediately declaring the LPCS system inoperable. Plausible because LPCS is declared immediately

D is correct

SRO Only (see attached flow chart):

The SRO should:

- apply required actions and surveillance requirements in accordance with the rules of application requirements.
- have knowledge of TS bases that is required to analyze TS required actions and terminology.
- apply generic LCO requirements.

Technical References:

Tech Specs 3.5.1, 3.0.3, 3.8.1
Tech Spec Bases 3.5.1

References to be provided to applicants during exam:

None

Learning Objective:

GLP-OPS-EP01

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)…	
	55.43(b)(2)	

Examination Outline Cross-Reference	Level	SRO
212000 RPS G2.2.22: Knowledge of limiting conditions for operations and safety limits	Tier #	2
	Group #	1
	K/A #	**212000 2.2.22**
	Rating	4.7

Question 87

The plant is at 100% power.

APRM Channel 1 is INOP and bypassed due to failing a surveillance.

Inverter 1Y96 trips.

(1) Which of the following describes the required Tech Spec LCO completion time?

(2) Which other RPS scram actuation is based on maintaining MCPR below Safety Limit setpoint?

A. (1) 6 hours

(2) Reactor Water Level High, Level 8

B. (1) 12 hours

(2) Reactor Water Level Low, Level 3

C. (1) 12 hours

(2) Reactor Water Level High, Level 8

D. (1) 6 hours

(2) Reactor Water Level Low, Level 3

Answer: C
Explanation: With APRM Ch 1 inop 3 are left. 1Y96 inverter powers the panel for APRM Ch. 3. Now only 2

APRM channels are left. 3.3.1.1 should be entered in condition A. When power is lost to the APRM panel the voter will de-energize and cause a half scram on RPS. Channels 1 and 3 feed Division 1 RPS and channels 2 and 4 feed Division 2 RPS. Therefore a Division 1 half scram will occur immediately.

Per Tech Spec Bases the APRM Fixed Neutron and Flow Biased Scram setpoint is based on not exceeding MCPR SL limit.

Reactor Water level hi, level 8 is also based on not exceeding MCPR SL limit

A is wrong - If student uses condition listed in table 3.3.1.1-1 this would be the required action and time.

B is wrong - A division 1 half scram will occur

C is correct

D is wrong - If student uses condition listed in table 3.3.1.1-1 this would be the required action and time. A division 1 half scram will occur

SRO Only (see attached flow chart):

The SRO should:
- apply required actions and surveillance requirements in accordance with the rules of application requirements.
- have knowledge of TS bases that is required to analyze TS required actions and terminology.

Technical References:

Tech Specs 3.3.1.1
04-1-01-C51-1

References to be provided to applicants during exam:

Tech Specs 3.3.1.1

Learning Objective:

GLP-OPS-C5101 - OBJECTIVE 10, 14

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	2

10CFR Part 55 Content:	55.41(b)(7)…
	55.43(b)(1)

Examination Outline Cross-Reference	Level	SRO
239002 Relief/Safety Valves **Ability to (a) predict the impacts of the following on the Relief/Safety Valves; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** A2.03: Stuck Open SRV	**Tier #**	2
	Group #	1
	K/A #	**259002 – A2.04**
	Rating	3.1

Question 88

The plant is operating at rated power when the following occurs:

- ADS / SRV LEAK. P601-18A-G-2, and SRV/ADS VLV OPEN/DISCH LINE PRESS HI, P601-19A-A5 annunciators are received
- Generator megawatt output has suddenly lowered by approximately 75 MWe
- SRV B21-F041D has a red indication on P601 handswitch
- Suppression Pool average water temperature is currently 98°F and rising
- Suppression Pool Water Level is 18.71ft. and rising.
- Handswitches on P601 and P631 for SRV B21-F041D are taken to the CLOSE position
- I&C have been notified to pull associated fuses.

What is the <u>next</u> required action to correct, control, or mitigate the impacts of this event on the plant?

A. Place the reactor Mode Switch in SHUTDOWN per EP-2

B. Place Both loops of RHR in Suppression Pool Cooling using the RHR E12 SOI Per EP-3

C. Place only one loop of RHR in Suppression Pool Cooling using the RHR E12 SOI Per EP-3

D. Place the reactor Mode Switch in SHUTDOWN and Emergency Depressurize Per EP-2

Answer: B

Explanation:
With the given information the student can determine that a stuck open SRV exsist.

All actions have been taken per the ARI to close the SRV. The next action should be enter EP-3 (>95°F Suppression Pool Temp.) and start all available Suppression Pool cooling (not needed for adequate core cooling). The SRO should direct both loop of suppression pool cooling to be started per EP-3. Before SP temp reaches 110°F the SRO will enter EP-2 which requires a Mode Switch to SHUTDOWN. This is based on Tech Spec 3.6.2.1 and bases.

A is wrong - This would only happen is Suppression pool temp cannot be maintained below 110°F. Plausible if the SRO believes that all actions have been taken to close the SRVs. Fuses have not been pulled yet.

B is correct - Per EP-3 Both loop are required.

C is wrong - Per EP-3 Both loop are required, the only time one loop would be correct is if the other loop is required for adequate core cooling. Plausible, if the SRO believes that only one loop is needed and/or required.

D is wrong - The EP-2 mode switch to SHUTDOWN is only required if Suppression pool temp cannot be maintained below 110°F. The Emergency Depress. would only be required if HCTL is exceeded. Suppression Pool Temp would have to be >150°F .

SRO Only (see attached flow chart):

The SRO should:

- assess plant conditions and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- have knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures.

Technical References:

04-1-02-1H13-P601-18A-G-2, ADS / SRV LEAK.
04-1-02-1H13-P601-19A-A5, SRV/ADS VLV OPEN/DISCH LINE PRESS HI
EP-3
EP-2

References to be provided to applicants during exam:

None

Learning Objective: Document learning objective if possible.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3

10CFR Part 55 Content:	55.41(b)(3) & (5) & (7) & (10)…
	55.43(b)(5)

Examination Outline Cross-Reference	Level	SRO
262002 UPS (AC/DC) 2.4.32 Knowledge of operator response to loss of all annunciators.	**Tier #**	2
	Group #	1
	K/A #	**262002 2.4.32**
	Rating	4.0

Question 89

The plant is operating at rated conditions when inverter malfunctions occur.

The following annunciators are in alarm:

- P680-4A1-E-7, ANNUN LOGIC PNL P851 PWR FAIL
- P680-4A2-E-1, ANNUN LOGIC PNL P630 PWR FAIL
- P870-4A-H-3, ANNUN LOGIC PNL P852 PWR FAIL

Plant conditions are stable and all other plant indications are normal.

After 20 minutes of receiving the alarms, what emergency declaration should be made?

A. None

B. Site Area Emergency

C. Alert

D. Unusual Event

Answer: D
Explanation: D is correct. The three annunciator panel power fail alarms indicate a loss of annunciator power. SU6 should be declared when an unplanned loss of >75% of the following for ≥15 minutes: Control Room safety system annunciation OR Control Room safety system indication. All plant indications are normal, but control room safety system annunciation has been lost. There is no significant transient in progress and compensatory indications are available and indicating normal. A is plausible if student does not recognize the given alarms indicate the loss of annunciation. If there is no loss of annunciation and plant conditions are stable, no declaration is required. General Emergency is not plausible because the loss of annunciation alone will not lead to a general emergency. B and C are plausible because they are the other possible emergency declaration options. (SS6 and

SA6)		
SRO Only (see attached flow chart): The SRO should: • assess plant conditions and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed		
Technical References: 10-S-01-1, Activation of the Emergency Plan 04-1-02-1H13-P870 04-1-02-1H13-P680		
References to be provided to applicants during exam: **10-S-01-1 (Activation of the Emergency Plan), EPP 01-02 (Flowchart) Page 1 and 2**		
Learning Objective:		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.43(b)(5)	

Examination Outline Cross-Reference	Level	SRO
261000 Standby Gas Treatment System **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:** A2.05: Fan Trips	**Tier #**	2
	Group #	1
	K/A #	**261000 – A2.05**
	Rating	3.1

Question 90

The plant is in day 10 of an outage.

Fuel movement is in progress in the Auxiliary Building.

Fuel movement has stopped in the Containment due to an OPDRV (Operation with a Potential for Draining the Reactor Vessel) in progress.

Standby Gas Treatment system 'A' receives an inadvertent initiation signal and the SGTS FLTR TR A EXH FAN trips.

(1) Which of the following describes how many Standby Gas Treatment Subsystems are required per Tech Specs.

(2) One of the Tech Spec Bases for this limit is to mitigate the radiological consequences of a _______________.

A. (1) One Subsystem

(2) Loss of Coolant Accident

B. (1) Two Subsystems

(2) Loss of Coolant Accident

C. (1) Two Subsystems

(2) Fuel handling accident involving irradiated fuel.

D. (1) One Subsystem

(2) Fuel handling accident involving irradiated fuel

Answer: B
Explanation: Per Tech Spec 3.6.4.3, Both SBGT subsystems are required to be operable. Due to an OPDRV in progress. Per Tech Spec B3.6.4.3 "The design basis for the SGT System is to mitigate the consequences of a loss of coolant accident and fuel handling accidents. Due to radioactive decay, the SGT System is required to be OPERABLE to mitigate only those fuel handling accidents involving the handling of recently irradiated fuel. A is wrong - Both systems are required B is correct -. C is wrong - The SGT System is required to be OPERABLE to mitigate only those fuel handling accidents involving the handling of recently irradiated fuel as defined as fuel that has occupied part of a critical reactor core within the previous 24 hours. At Day 10 of the outage the fuel is not recently irradiated fuel. D is wrong - Both systems are required
SRO Only (see attached flow chart): The SRO should: • assess plant conditions and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed • have knowledge of administrative procedures that specify hierarchy, implementation, an/or coordination of plant normal, abnormal, and emergency procedures..
Technical References: Tech Specs 3.6.4.3 and Bases 3.6.4.3
References to be provided to applicants during exam: **None.**
Learning Objective: GLP-OPS-TS001

Question Source:	Bank #	

(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(7)…	
	55.43(b)(2)	

Examination Outline Cross-Reference	Level	SRO
202002 Recirculation Flow Control **Ability to (a) predict the impacts of the following on the recirculation flow control system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** A2.09: Recirculation Flow mismatch	Tier #	2
	Group #	2
	K/A #	**202002 - A2.09**
	Rating	3.1

Question 91

The plant is operating at 100% power.

Plant conditions require core flow to be immediately reduced to 70 Mlbm/hr.

After the ATC reaches 70 Mlbm/hr, he reports the following conditions:

- Reactor Recirculation ‘A’ Driving Flow 29 kgpm
- Reactor Recirculation ‘B’ Driving Flow 23.5 kgpm
- Total Jet Pump Flow 70 mlbm/hr

(1) Which of the following would be the action directed by the SRO?

(2) Which of the following describes the time limit required to perform the action?

A. (1) Balance loop flows to within 4460 gpm maintaining 70 Mlbm/hr core flow.
   (2) Within 2 hours.

B. (1) Balance loop flows to within 2230 gpm maintaining 70 Mlbm/hr core flow.
   (2) Within 30 minutes

C. (1) Balance loop flows to within 4460 gpm maintaining 70 Mlbm/hr core flow.
   (2) Within 30 minutes.

D. (1) Balance loop flows to within 2230 gpm maintaining 70 Mlbm/hr core flow.
   (2) Within 2 hours.

Answer: A

Explanation:

With the given indication, the student should recognize that loop flows are outside the required mismatch, at < 78.7 Mlbm/hr (70%) core flow loop flows should be within 4460 gpm (10%). Per 05-1-02-III-3, Reduction in Recirculation System Flow Rate ONEP **subsequent actions 3.7**, “At less than 78.7 Mlbm/hr core flow, BALANCE loop flows to within 4460 gpm.

Core flow should be maintain at <70 Mlbm/hr due to the immediate action.

Tech Specs 3.4.1 requires matched flows and if out of limits then condition A would be entered and flows must be matched within 2 hours or one recirc loop will be shutdown.

A is correct

B is wrong - the 2230 gpm flow rate given is if above 70% core flow or 78.7 Mlbm/hr and the time limit is for tech spec TR3.4.1 Single loop operation to maintain <44,600 gpm in operating loop, if above that limit then restore within 30 minutes. Plausible if student confuses the 70% limit or the tech spec flow requirements for dual loop and single loop. .

C is wrong - the time limit is for tech spec TR3.4.1 Single loop operation to maintain <44,600 gpm in operating loop, if above that limit then restore within 30 minutes. Plausible if student confuses the tech spec flow requirements for dual loop and single loop. .

D is wrong - the 2230 gpm flow rate given is if above 70% core flow or 78.7 Mlbm/hr, Plausible if student confuses the 70% limit.

SRO Only (see attached flow chart):

The SRO should:

- assess plant conditions and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- have knowledge of administrative procedures that specify hierarchy, implementation, an/or coordination of plant normal, abnormal, and emergency procedures..
- have knowledge of TS bases that is required to analyze TS required actions and terminology

Technical References:

ONEP 05-1-02-III-6, Jet Pump Anomalies
ONEP 05-1-02-III-3, Reduction in Recirculation System Flow Rate ONEP
Tech Spec 3.4.1

References to be provided to applicants during exam:

NONE

Learning Objective:

GLP-OPS-ONEP,
GLP-OPS-B3300,

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(7)…	
	55.43(b)(5)	

.Examination Outline Cross-Reference	Level	SRO
234000 Fuel Handling Equipment	Tier #	2
	Group #	2
Ability to (a) predict the impacts of the following on the Fuel Handling Equipment; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	**K/A #**	**234000– A2.01**
	Rating	3.6
A2.01: Interlock Failure		

Question 92

The Refuel Platform is lowering a fuel bundle into the reactor vessel.

The Safety Travel interlock fails in an activated state.

(1) What is the impact of this condition on the Refuel Platform?

(2) What action is required for continued operation?

A. (1) Only Bridge movement is prevented. Trolley and Main Hoist movement may continue.

(2) Obtain Refuel Floor SRO permission and use the "Travel Override" button.

B. (1) Only Bridge movement is prevented. Trolley and Main Hoist movement may continue.

(2) Obtain Refuel Floor SRO permission and use the "Hoist Override" button.

C. (1) All Bridge, Trolley, and Main Hoist movements are prevented.

(2) Obtain Refuel Floor SRO permission and use the "Hoist Override" button.

D. (1) All Bridge, Trolley, and Main Hoist movements are prevented.

(2) Obtain Refuel Floor SRO permission and use the "Travel Override" button.

Answer: D
Explanation: 04-1-01-F11-1 P&L 3.19 states that when the grapple is loaded the Refuel Floor SRO or Shift Manager's permission is required to operate the Travel Override pushbutton.

04-1-01-F11-1 Att IX requires the Shift Manager's approval to bypass the Safety Travel interlock.

This question requires knowledge of the refuel floor SRO responsibilities.

A is wrong - All Bridge, Trolley, and Main Hoist movements are prevented

B is wrong - All Bridge, Trolley, and Main Hoist movements are prevented and the "Travel Override" button must be used.

C is wrong - the "Travel Override" button must be used.

D is correct.

SRO Only (see attached flow chart):

The SRO should:

- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal and emergency procedures.

Technical References:

04-1-01-F11-1, Refueling Platform
GLP-RF-F1101

References to be provided to applicants during exam:

None

Learning Objective:

GLP-RF-F1101 OBJ. 15.3

Question Source:	Bank	#1014 not used on an NRC exam
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(7)…	
	55.43(b)(7)	

Examination Outline Cross-Reference	Level	SRO
259001 Reactor Feedwater G2.2.38: Knowledge of conditions and limitations in the facility license.	Tier #	2
	Group #	2
	K/A #	**259001 2.2.38**
	Rating	4.5

Question 93

The plant is at rated power when the following occurs:

- P680 alarm FW HTR 6B LVL HI is received.
- ATC reports approximately 1.0 Mlbm/hr difference in Reactor Feedwater pump suction flow and Total Feedwater system flow.
- Reactor Power is indicating 4410 MWT
- Feedwater temperature had lowered by 5°F

Which of the following procedures will the SRO direct the crew actions from?

A. ARI 04-1-02-1H13-P680-2A-A10, FW HTR 6B LVL HI

B. IOI 03-1-02-2, Power Operations.

C. ONEP 05-1-02-V-5, Loss of Feedwater Heating.

D. SOI 04-1-01-N23-1, Heater Vents and Drains N23/Extraction Steam N36

Answer: B
Explanation: FW HTR 6B is a high pressure heater, a tube leak would cause a Hi level condition and cause a difference in flow on the feedwater system. With this lead the Feedwater control system will compensate and provide more flow to the vessel. This increase in flow and reduced efficiency of the HP feedwater heater causes a reduction in feedwater temperature which in turn causes a rise in Reactor Power. GGNS license states not to exceed 4408 MWT. With the indication of 4410 the SRO should direct the crew to lower reactor power per IOI-2 to within the maximum power limit. A is wrong - This ARI is good for the feedwater heater hi level, however reactor power must be reduced and this ARI does not give that direction. Plausible due to if the student does not recognize exceeding the power limit the ARI would give direction to monitor and enter the SOI B is correct - 03-1-01-2, step 2.27 C is wrong - The ONEP has not reached an entry condition yet a HI-HI level is required prior to entering the ONEP. Plausible due to if the student does not recognize that only the Hi level has

occurred not the HI Hi level.

D is wrong - System operating instruction gives guidance to isolate the heater which is a possibility after reactor power is reduced. Plausible due to this procedure is referenced in the ARI mentioned in answer A.

SRO Only (see attached flow chart):

The SRO should:

- assess plant conditions and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- have knowledge of administrative procedures that specify hierarchy, implementation, an/or coordination of plant normal, abnormal, and emergency procedures..

Technical References:

03-1-01-2, step 2.27
ONEP 05-1-02-V-5, Loss of Feedwater Heating

References to be provided to applicants during exam:

None

Learning Objective:

GLP-OPS-ONEP
GLP-OPS-IOI02

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	3
10CFR Part 55 Content:	55.41(b)(7) & (10)…	
	55.43(b(1)	

Examination Outline Cross-Reference	Level	SRO
Conduct of Operations	**Tier #**	3
	Group #	
Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc.	**K/A #**	**2.1.4**
	Rating	3.3

Question 94

An on-shift SRO, qualified as an STA also, has been on short term disability.

Total time away from work was 5 months (April thru August)

He has completed his STA proficiency.

He is scheduled to work the following hours:

Sunday 1	Monday 2	Tuesday 3	Wednesday 4	Thursday 5	Friday 6	Saturday 7
12 hrs parallel as CRS	12 hrs parallel as CRS	12 hrs as STA/FSS			12 hrs as STA/FSS	12 hrs as STA/FSS
Sunday 8	Monday 9	Tuesday 10	Wednesday 11	Thursday 12	Friday 13	Saturday 14
12 hrs as STA/FSS				12 hrs parallel as CRS	12 hrs parallel as CRS	12 hrs parallel as CRS

Which of the following describes when the SRO has completed the watch standing proficiency for a CRS?

A. Completion on Sunday the 1st

B. During the shift on Friday the 6th

C. During the shift on Friday the 13th

D. Completion on Saturday the 14th

Answer: C

Explanation:

Per 02-S-01-39, Maintaining Watchstanding Proficiency,6.1.1c “SROs who fail to meet watchstanding per quarter and are designated to maintain proficiency by parallel watchstanding will be returned to/maintained in active status as follows:

Licensee shall complete 40 hours (8 hours for Refueling SRO only) of parallel watch.”

A is wrong - This would be correct if it were for the Refuel SRO per step 6.1.1c (1)

B is wrong - This completes >than 40 hours, however the NOTE prior to step 6.1.1b states “An SRO may not take credit for watchstanding proficiency when filling the position of STA or FSS
.
C is correct

D is wrong - This would be correct for maintaining watchstanding proficiency per step 6.1.1 “The license holder must stand seven complete 8-hour shifts or 5 complete 12-hour shifts per quarter to maintain active status.

SRO Only (see attached flow chart):

The SRO should:

- have knowledge of administrative procedures that specify hierarchy, implementation, an/or coordination of plant normal, abnormal, and emergency procedures..

Technical References:

02-S-01-39, Maintaining Watchstanding Proficiency section 6.1 and Attachment 1

References to be provided to applicants during exam:

None.

Learning Objective: Document learning objective if possible.

GLP-OPS-PROC Obj. 6

Question Source:	Bank	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(10)…	
	55.43(b)(2)	

Examination Outline Cross-Reference	Level	SRO
Conduct of Operations Knowledge of procedures, guidelines, or limitations associated with reactivity management.	**Tier #**	3
	Group #	
	K/A #	**2.1.37**
	Rating	4.6

Question 95

Power is to be reduced from 100% to 65% for a control rod sequence exchange.

Who is responsible for reviewing the Reactivity Maneuver Plan and ensuring that the control rod pull sheets are highlighted to emphasize areas of concern?

A. Reactor Engineering

B. Reactivity Management SRO

C. ON-shift Control Room Supervisor

D. ON-shift Operations Shift Manager

Answer: B

Explanation:
02-S-01-27, Operations Philosophy, classifies the described power change as a Type 3 power maneuver per step 6.8.1a(3), which requires staffing an additional SRO, the Reactivity Management SRO (RMSRO). Step 6.8.1b(2) states the RMSRO is responsible for reviewing the Reactivity Maneuver Plan and ensuring that the control rod pull sheets are highlighted to emphasize areas of concern. Step 6.8.2d states the RMSRO is responsible for instructing ROs on the execution of the specific pull sheets.

A is wrong - but, is plausible since Reactor Engineers develop the control rod pull sheets and are involved in during the sequence exchange.

C is wrong - but is plausible since this is an on shift SRO who does not normally have the control room command function and might be considered capable of assuming dedicated reactivity management duties.

D is wrong - but is plausible because Ops Management is present in the control room for management oversight during power reductions for sequence exchanges.

SRO Only (see attached flow chart):

The SRO should:

- have knowledge of administrative procedures that specify hierarchy, implementation, an/or

coordination of plant normal, abnormal, and emergency procedures..		
Technical References: 02-S-01-27, Operations Philosophy		
References to be provided to applicants during exam: **None.**		
Learning Objective: GLP-OPS-PROC Obj. 4.10		
Question Source:	Bank #710	2013 NRC EXAM Q. 95
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2013 NRC EXAM Q. 95
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(7)…	
	55.43(b)(6)	

Examination Outline Cross-Reference	Level	SRO
Equipment Control **Knowledge of the process for making changes to procedures**	**Tier #**	3
	Group #	
	K/A #	**2.2.6**
	Rating	3.6

Question 96

The process for making changes to procedures using the Temporary Change Notice, requires an On-Shift SRO review to ensure ______________.

A. the change(s) do not adversely impact current plant operating conditions.

B. if an Emergency Preparedness Review is required.

C. change meets all QA-related aspects prior to implementation.

D. a qualified PADs Preparer is assigned to perform the Process Applicability Determination.

Answer: A
Explanation: From 01-S-02-9, Procedure Change Process, Rev 001, Section 5.20.5 2, "Ensure the change(s) do not adversely impact current plant operating conditions." A is correct B is wrong - This is performed by the Technical Reviewer 5.20.2 C is wrong - This is performed by the PAD Preparer 5.20.3 D is wrong - This is performed by the Supervision 5.20.6
SRO Only (see attached flow chart): The SRO should: • have knowledge of administrative procedures that specify hierarchy, implementation, an/or coordination of plant normal, abnormal, and emergency procedures..

Technical References: 01-S-02-9, Procedure Change Process, section 5.20		
References to be provided to applicants during exam: **None.**		
Learning Objective: Document learning objective if possible. GLP-OPS-PROC, Objective 11.2		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NO
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(10)…	
	55.43(b) (3)	

Examination Outline Cross-Reference	Level	SRO
Equipment Control	Tier #	3
	Group #	
Knowledge of the process for controlling temporary design changes.	K/A #	**2.2.11**
	Rating	3.3

Question 97

An emergency temporary modification can be implemented in the event of an imminent threat to the safety or reliability of the plant due to an unforeseen plant event:

Which of the following is required?

1. The Shift Manager, with the concurrence of the Engineering Director, or designee, may direct the installation or removal of a Temporary Modification to the plant without approved controlling documentation, as long as the Temporary Modification does not adversely affect nuclear safety.
2. As soon as conditions permit, the Operations Manager and the Systems & Components Manager or their designee shall be verbally notified of the modification and a Condition Report shall be initiated by Maintenance.
3. A Temporary Modification or a permanent Engineering Change shall be completed within 7 calendar days after installation.
4. The Responsible Engineer should coordinate with other Departments (i.e., the Systems & Components Engineer, Operations, Maintenance, Training, Planner and Installer) to ensure they are cognizant of the change and have provided appropriate input.
5. If the Temp Mod is a Comp action, a separate CR will be written by the Shift Manager to track the comp measure.

A. 1, 2 and 3 only.

B. 2, 3 and 4 only

C. 2, 4 and 5 only

D. 1, 3 and 5 only

Answer: D

Explanation:

EN-DC-136, Temporary Modifications, contains this requirement in section 5.3 .

5.3 EMERGENCY TEMPORARY MODIFICATION IMPLEMENTATION

[1] In the event of an imminent threat to the safety or reliability of the plant due to an unforeseen plant event:

(a) The Shift Manager, with the concurrence of the Engineering Director, or designee, may direct the installation or removal of a Temporary Modification to the plant on an “emergency” basis without approved controlling documentation, as long as the Temporary Modification does not adversely affect nuclear safety.

(b) As soon as conditions permit, the Operations Manager and the Systems & Components Manager or their designee shall be verbally notified of the “emergency” modification and a Condition Report shall be initiated by Engineering. The CR issued shall be used to track the installation of the Emergency Temporary Modification. Following installation, removal of the Emergency Temporary Modification shall follow the applicable steps of this procedure.

(c) **IF** the Temporary Modification is also a compensatory measure (operational), **THEN** the Shift Manager will ensure that a Condition Report is issued to track the compensatory measure. This is a separate CR from step 5.3.(1)(b).

(d) A Temporary Modification or a permanent Engineering Change shall be completed within 7 calendar days after installation.

A is wrong - #2 is wrong due to the CR is to be initiated by Engineering not Maintenance, and #5 is also required.

B is wrong - #2 is wrong due to the CR is to be initiated by Engineering not Maintenance, and #4 is incorrect.

C is wrong - #2 is wrong due to the CR is to be initiated by Engineering not Maintenance and #4 is incorrect.

D is correct

SRO Only (see attached flow chart):

The SRO should:

- have knowledge of administrative procedures that specify hierarchy, implementation, an/or coordination of plant normal, abnormal, and emergency procedures..

Technical References:

EN-DC-136, Temporary Modifications

References to be provided to applicants during exam:

None.

Learning Objective:

GLP-OPS-PROC, Objective 40.4

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	

	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	3
10CFR Part 55 Content:	55.41(b)(10)	
	55.43(b)(3)	

Examination Outline Cross-Reference	Level	SRO
Radiation Control	**Tier #**	3
	Group #	
Ability to control radiation releases.	**K/A #**	**2.3.11**
	Rating	4.3

Question 98

Consider the following processes:

1. Batch release of liquid effluents during normal plant operations
2. Continuous release of liquid radioactive waste
3. Discharge of solid radioactive waste
4. Continuous discharge of gaseous effluents below normal radiological limits during normal plant operations

Which of the above processes are controlled using a Discharge Permit per 01-S-08-11, Radioactive Discharge Controls?

A. 1 only

B. 1 and 2 only

C. 2 and 3 only

D. 4 only

Answer: A
Explanation: The SRO/Shift Manager duties is to review and sign the Batch Release paperwork prior to discharge. This task is only performed by a SRO license holder. See 01-S-08-11. 6.2 Solid waste must not be discharged 6.3 Continuous discharges do not require a discharge permit. 6.4.1 Batch Gaseous Releases, If a condition exists in which the Chemistry manager determines that a batch of gaseous release would have a radiological impact on the environment then a batch permit will be devised.

6.4.2 A batch liquid discharge permit should be completed for all batch liquid releases and will be processed.

A is correct

B is wrong - per Section 6.3, continuous discharges do not require a permit.

C is wrong - per Section 6.3, continuous discharges do not require a permit and per 6.2 Solid waste must not be discharged

D is wrong - per Section 6.3, continuous discharges do not require a discharge permit and 6.4.1 Batch Gaseous Releases, If a condition exists in which the Chemistry manager determines that a batch of gaseous release would have a radiological impact on the environment then a batch permit will be devised.

SRO Only (see attached flow chart):

The SRO should:

- have knowledge of administrative procedures that specify hierarchy, implementation, an/or coordination of plant normal, abnormal, and emergency procedures..

Technical References:

01-S-08-11, Radioactive Discharge Controls

References to be provided to applicants during exam:

None.

Learning Objective: Document learning objective if possible.

GLP-OPS-PROC, OBJ 51.0

Question Source:	Bank # NRC Bank 195	2008 NRC EXAM Q. 96
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(13)	
	55.43(b)(4)	

Examination Outline Cross-Reference	Level	SRO
2.4.44 Knowledge of emergency plan protective action recommendations.	Tier #	3
	Group #	
	K/A #	**2.4.44**
	Rating	4.1

Question 99

A General Emergency has been declared.

The ERO determines that an Ad-Hoc PAR needs to be issued.

By definition, this Ad-Hoc PAR will recommend evacuating all sectors out to _____ and evacuating downwind sectors out to ______.

A. 2 miles; a minimum of 10 miles

B. 2 miles; greater than 10 miles

C. 5 miles; greater than 10 miles

D. 5 miles; a minimum of 10 miles

Answer: B
Explanation: 10-S-01-12, sections 6.2.1and 6.2.2. By definition, an Ad-Hoc PAR incorporates the Extended PAR (as shown on the Table of 6.2.1, page 11) and will evacuate to the necessary distance (>10 miles) all downwind sectors. A is wrong - 10 mile sectors is only the extended PAR not the Ad-HOC. Ad-HOC goes >10 miles. B is correct C is wrong - All sectors are only evacuated in a 2 mile radius not 5 mile and 10 mile sectors is only the extended PAR not the Ad-HOC. Ad-HOC goes >10 miles. D is wrong - All sectors are only evacuated in a 2 mile radius not 5 mile and 10 mile sectors is only the extended PAR not the Ad-HOC. Ad-HOC goes >10 miles.
Technical References:

10-S-01-1
10-S-01-12

SRO Only (see attached flow chart):

The SRO should:

- have knowledge of administrative procedures that specify hierarchy, implementation, an/or coordination of plant normal, abnormal, and emergency procedures..

References to be provided to applicants during exam:

None.

Learning Objective:

GLP-EP-EPTS6, Objective 4

Question Source:	Bank # NRC Bank 182	2012 Audit Exam Q. 85
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	3
10CFR Part 55 Content:	55.41(b)(7)	
	55.43(b)(5)	

Examination Outline Cross-Reference	Level	SRO
2.4.42 Knowledge of emergency response facilities.	Tier #	3
	Group #	
	K/A #	**2.4.42**
	Rating	3.8

Question 100

Per 10-S-01-1 (Activation of the Emergency Plan), there is only one type of emergency where the Emergency Director is directed to **not activate** any of the onsite or offsite Emergency Response Facilities (even though an ALERT or higher EAL may have been declared).

What is that type of emergency?

A. Tornado resulting in Visible Damage within the Protected Area

B. Validated notification of an airborne attack threat

C. Armed attack against the plant

D. Validation that an Operating Basis Earthquake has caused major damage to plant vital structures

Answer: C
Explanation: See 10-S-01-1, section 6.1.8. This section prescribes the ED's responsibilities unique to an "armed attack against the plant". Section 6.1.8.f specifically directs the ED to "not activate…any of the ERFs". A and B are wrong - but are plausible because they each represent types of emergencies that might indicate the need to potentially shelter personnel (i.e., be concerned for safety). C is correct D is wrong - is the strongest distracter in that it may lead to a conclusion of an armed attack on the plant as well. However, per the EALs (i.e., HA1a), an "airborne attack threat" is simply

an airliner less than 30 minutes away from the plant.

SRO Only (see attached flow chart):

The SRO should:

- have knowledge of administrative procedures that specify hierarchy, implementation, an/or coordination of plant normal, abnormal, and emergency procedures..

Technical References:

10-S-01-1, Activation of the Emergency Plan
Emergency Action Levels (Flowcharts)

References to be provided to applicants during exam:

NONE

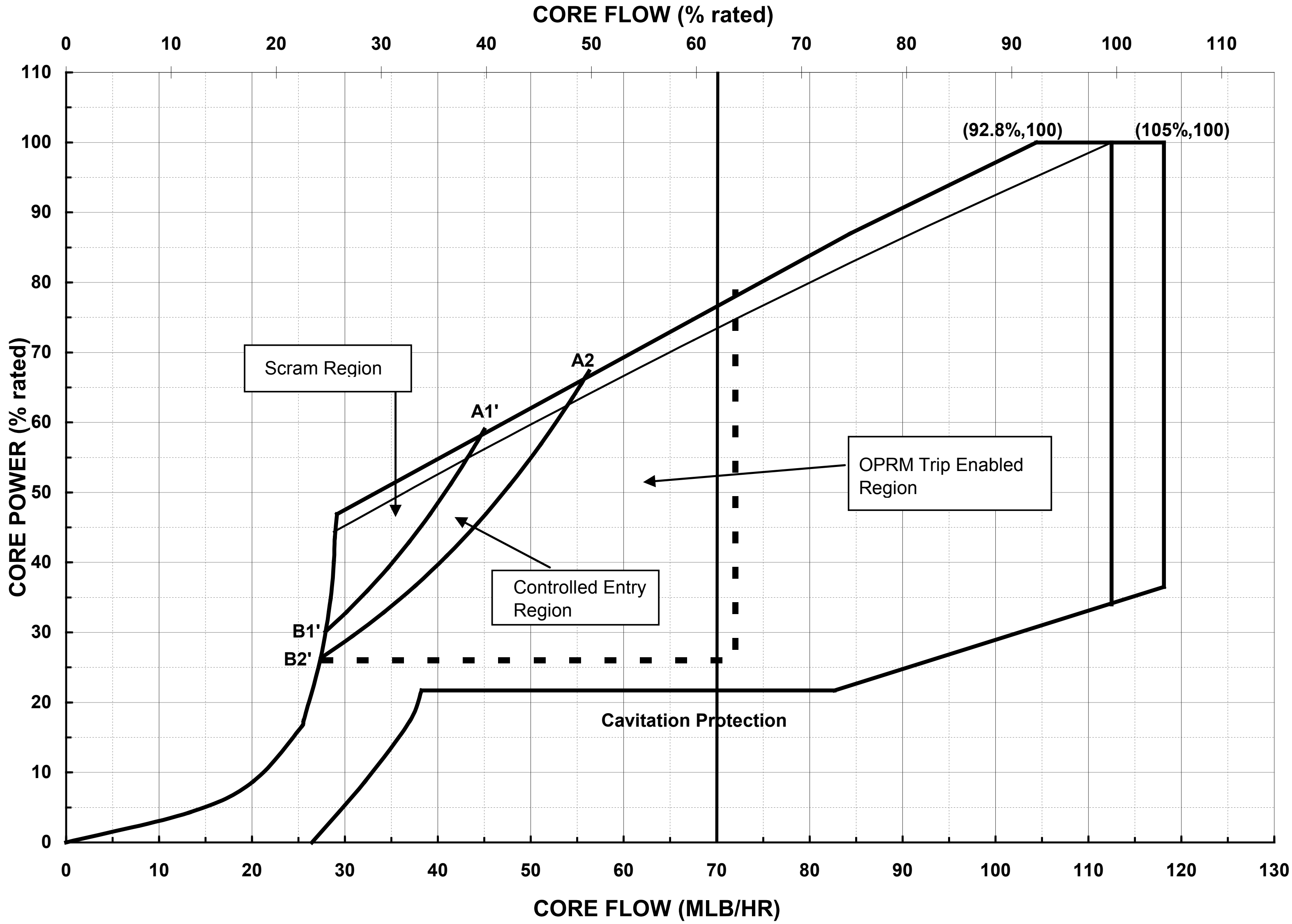
Learning Objective: Document learning objective if possible.

GLP-EP-EPTS 6 OBJ. 7

Question Source:	Bank # NRC Bank 404	2011 NRC Q. 96
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(10)…	
	55.43(b)(5)	

GGNS LOT 2015 NRC INITIAL LICENSED OPERATOR WRITTEN EXAMINATION
OPEN-REFERENCES TABLE OF CONTENTS
SRO EXAM

TAB 1



CORE FLOW (% rated)
0
10
20
30
40
50
60
70
80
90
100
110
CORE POWER (% rated)
0
10
20
30
40
50
60
70
80
90
100
110
(92.8%,100)
(105%,100)
Scram Region
A2
A1'
OPRM Trip Enabled Region
Controlled Entry Region
B1'
B2'
Cavitation Protection
0
10
20
30
40
50
60
70
80
90
100
110
120
130
CORE FLOW (MLB/HR)

TAB 2

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources—Operating

LCO 3.8.1

APPLICABILITY:

ACTIONS

----------------------------------NOTE----------------------------------
LCO 3.0.4.b is not applicable to DGs.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required offsite circuit inoperable for reasons other than Condition F.	A.1	Perform SR 3.8.1.1 for OPERABLE required offsite circuit.	1 hour <u>AND</u> Once per 8 hours thereafter
	<u>AND</u>		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	Restore required offsite circuit to OPERABLE status.	72 hours AND 24 hours from discovery of two divisions with no offsite power AND 17 days from discovery of failure to meet LCO
B. One required DG inoperable for reasons other than Condition F.	B.1	Perform SR 3.8.1.1 for OPERABLE required offsite circuit(s).	1 hour AND Once per 8 hours thereafter
	AND		
	B.2	Declare required feature(s), supported by the inoperable DG, inoperable when the redundant required feature(s) are inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	AND		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.3.1	Determine OPERABLE DG(s) are not inoperable due to common cause failure.	24 hours
	OR		
	B.3.2	Perform SR 3.8.1.2 for OPERABLE DG(s).	24 hours
	AND		
	B.4	Restore required DG to OPERABLE status.	72 hours from discovery of an inoperable Division 3 DG AND 14 days AND 17 days from discovery of failure to meet LCO
C. Two required offsite circuits inoperable.	C.1	Declare required feature(s) inoperable when the redundant required feature(s) are inoperable.	12 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s)
	AND		
	C.2	Restore one required offsite circuit to OPERABLE status.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One required offsite circuit inoperable for reasons other than Condition F. <u>AND</u> One required DG inoperable for reasons other than Condition F.	-----------NOTE-------------- Enter applicable Conditions and Required Actions of LCO 3.8.7, "Distribution Systems—Operating," when any required division is de-energized as a result of Condition D. ----------------------------- D.1 Restore required offsite circuit to OPERABLE status. <u>OR</u> D.2 Restore required DG to OPERABLE status.	 12 hours 12 hours
E. Two required DGs inoperable.	E.1 Restore one required DG to OPERABLE status.	2 hours <u>OR</u> 24 hours if Division 3 DG is inoperable
F. One automatic load sequencer inoperable.	F.1 Restore automatic load sequencer to OPERABLE status.	24 hours
G. Required Action and associated Completion Time of Condition A, B, C, D, E, or F not met.	G.1 --------NOTE--------- LCO 3.0.4.a is not applicable when entering MODE 3. --------------------- Be in MODE 3.	 12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. Three or more required AC sources inoperable.	H.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each required offsite circuit.	7 days
SR 3.8.1.2	-------------------NOTES------------------- 1. Performance of SR 3.8.1.21 satisfies this SR. 2. All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. 3. A modified DG start involving idling and gradual acceleration to synchronous speed may be used for this SR as recommended by the manufacturer. When modified start procedures are not used, the time, voltage, and frequency tolerances of SR 3.8.1.21 must be met. --- Verify each DG starts from standby conditions and achieves steady state voltage ≥ 3744 V and ≤ 4576 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.3	---NOTES--- 1. DG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load range do not invalidate this test. 3. This Surveillance shall be conducted on only one DG at a time. 4. This SR shall be preceded by, and immediately follow, without shutdown, a successful performance of SR 3.8.1.2 or SR 3.8.1.21. --- Verify each DG operates for ≥ 60 minutes at a load ≥ 5450 kW and ≤ 5740 kW for DG 11 and DG 12, and ≥ 3300 kW for DG 13.	31 days
SR 3.8.1.4	Verify each DG day tank contains ≥ 220 gal of fuel oil.	31 days
SR 3.8.1.5	Check for and remove accumulated water from each day tank.	31 days
SR 3.8.1.6	Verify the fuel oil transfer system operates to automatically transfer fuel oil from the storage tank to the day tank.	31 days
SR 3.8.1.7	Verify the load shedding and sequencing panels respond within design criteria.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.1.8	-----------------NOTE---------------------- This Surveillance shall not be performed in MODE 1 and 2. However, credit may be taken for unplanned events that satisfy this SR. --- Verify manual transfer of unit power supply from the normal offsite circuit to required alternate offsite circuit.	24 months
SR 3.8.1.9	-----------------NOTES--------------------- 1. Credit may be taken for unplanned events that satisfy this SR. 2. If performed with the DG synchronized with offsite power, it shall be performed at a power factor ≤ 0.9 for DG 11 and DG 13 and ≤ 0.89 for DG 12. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. --- Verify each DG rejects a load greater than or equal to its associated single largest post accident load and engine speed is maintained less than nominal plus 75% of the difference between nominal speed and the overspeed setpoint or 15% above nominal, whichever is lower.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.10	------------------NOTE------------------ 1. Credit may be taken for unplanned events that satisfy this SR. 2. *If performed with the DG synchronized* with offsite power, it shall be performed at a power factor ≤ 0.9 for DG 11 and DG 13 ≤ 0.89 for DG 12. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. -- Verify each DG does not trip and voltage is maintained ≤ 5000 V during and following a load rejection of a load ≥ 5450 kW and ≤ 5740 kW for DG 11 and DG 12 and ≥ 3300 kW for DG 13..	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.1.11 -------------------NOTES------------------ 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1, 2, or 3 (Not Applicable to DG 13). However, credit may be taken for unplanned events that satisfy this SR. -- Verify on an actual or simulated loss of offsite power signal: a. De-energization of emergency buses; b. Load shedding from emergency buses for Divisions 1 and 2; and c. DG auto-starts from standby condition and: 1. energizes permanently connected loads in ≤ 10 seconds, 2. energizes auto-connected shutdown loads, 3. maintains steady state voltage ≥ 3744 V and ≤ 4576 V, 4. maintains steady state frequency ≥ 58.8 Hz and ≤ 61.2 Hz, and 5. supplies permanently connected and auto-connected shutdown loads for ≥ 5 minutes.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.12	-------------------NOTES------------------- 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1, or 2 (Not Applicable to DG 13). However, credit may be taken for unplanned events that satisfy this SR. --- Verify on an actual or simulated Emergency Core Cooling System (ECCS) initiation signal each DG auto-starts from standby condition and: a. In ≤ 10 seconds after auto-start and during tests, achieve voltage ≥ 3744 V and frequency ≥ 58.8 Hz; b. Achieves steady state voltage ≥ 3744 V and ≤ 4576 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz; c. Operates for ≥ 5 minutes; and d. Emergency loads are auto-connected to the offsite power system.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.13	------------------NOTE-------------------- Credit may be taken for unplanned events that satisfy this SR. -- Verify each DG's non-critical automatic trips are bypassed on an actual or simulated ECCS initiation signal.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.1.14	-------------------NOTES------------------- 1. Momentary transients outside the load and power factor ranges do not invalidate this test. 2. Credit may be taken for unplanned events that satisfy this SR. 3. If performed with the DG synchronized with offsite power, it shall be performed at a power factor ≤ 0.9 for DG 11 and DG 13 and ≤ 0.89 for DG 12. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. --- Verify each DG operates for ≥ 24 hours: a. For DG 11 and DG 12 loaded ≥ 5450 kW and ≤ 5740 kW; and b. For DG 13: 1. For ≥ 2 hours loaded ≥ 3630 kW, and 2. For the remaining hours of the test loaded ≥ 3300 kW.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.15	------------------NOTES------------------ 1. This Surveillance shall be performed within 5 minutes of shutting down the DG after the DG has operated ≥ 1 hour or until operating temperatures stabilized loaded ≥ 5450 kW and ≤ 5740 kW for DG 11 and DG 12, and ≥ 3300 kW for DG 13. Momentary transients outside of the load range do not invalidate this test. 2. All DG starts may be preceded by an engine prelube period. --- Verify each DG starts and achieves: a. in ≤ 10 seconds, voltage ≥ 3744 V and frequency ≥ 58.8 Hz; and b. steady state voltage ≥ 3744 V and ≤ 4576 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.16	-------------------NOTE------------------- This Surveillance shall not be performed in MODE 1, 2, or 3 (Not Applicable to DG 13). However, credit may be taken for unplanned events that satisfy this SR. -- Verify each DG: a. Synchronizes with offsite power source while loaded with emergency loads upon a simulated restoration of offsite power; b. Transfers loads to offsite power source; and c. Returns to ready-to-load operation.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.1.17	-------------------NOTE------------------- Credit may be taken for unplanned events that satisfy this SR. -- Verify, with a DG operating in test mode and connected to its bus, an actual or simulated ECCS initiation signal overrides the test mode by: a. Returning DG to ready-to-load operation; and b. Automatically energizing the emergency loads from offsite power.	24 months
SR 3.8.1.18	-------------------NOTE------------------- This Surveillance shall not be performed in MODE 1, 2, or 3. However, credit may be taken for unplanned events that satisfy this SR. -- Verify interval between each sequenced load block is within ± 10% of design interval for each automatic load sequencer.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.19	-------------------NOTES------------------ 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1, 2, or 3 (Not Applicable to DG 13). However, credit may be taken for unplanned events that satisfy this SR. --- Verify, on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ECCS initiation signal: a. De-energization of emergency buses; b. Load shedding from emergency buses for Divisions 1 and 2; and c. DG auto-starts from standby condition and: 1. energizes permanently connected loads in ≤ 10 seconds, 2. energizes auto-connected emergency loads, 3. achieves steady state voltage ≥ 3744 V and ≤ 4576 V, 4. achieves steady state frequency ≥ 58.8 Hz and ≤ 61.2 Hz, and 5. supplies permanently connected and auto-connected emergency loads for ≥ 5 minutes.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.20	------------------NOTE------------------ All DG starts may be preceded by an engine prelube period. -- Verify, when started simultaneously from standby condition, each DG achieves: a. in ≤ 10 seconds, voltage ≥ 3744 V and frequency ≥ 58.8 Hz; and b. steady state voltage ≥ 3744 V and ≤ 4576 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	10 years
SR 3.8.1.21	------------------NOTE------------------ All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. -- Verify each DG starts from standby conditions and achieves: a. in ≤ 10 seconds, voltage ≥ 3744 V and frequency ≥ 58.8 Hz; and b. steady state voltage ≥ 3744 V and ≤ 4576 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	184 days

Table 3.8.1-1

Deleted

--NOTE--
The following surveillance requirements apply to LCO 3.8.1. Failure to meet these surveillance requirements requires entry into LCO 3.8.1.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR TR3.8.1.1	Verify the diesel generator is aligned to provide standby power to the associated emergency busses.	31 days
SR TR3.8.1.2	-----------------------NOTE----------------------- Inspections that require a retest that cannot be performed on-line, shall not be performed in MODE 1, or 2. -- Subject the diesels to an inspection, commensurate for nuclear standby service, that takes into consideration the following factors: the manufacturer's recommendations, diesel owners group recommendations, engine run time, calendar time, and the GGNS comprehensive maintenance inspection program.	Inspection frequencies for the various inspections are identified in the approved maintenance program.
SR TR3.8.1.3	-----------------------NOTE----------------------- This Surveillance shall not be performed in MODE 1 or 2. "(Not applicable to DG 13)" -- Verify that the auto-connected loads to each diesel generator do not exceed 5740 kW for diesel generators 11 and 12 and 3300 kW for diesel generator 13.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR TR3.8.1.4	-----------------------NOTES----------------------- 1. This Surveillance shall not be performed in MODE 1, 2 or 3. 2. All DG starts may be preceded by an engine prelube period. --- Verify, when started simultaneously from standby condition, each DG achieves: a. in ≤ 10 seconds, voltage ≥ 3744 V and frequency ≥ 58.8 Hz; and b. steady state voltage ≥ 3744 V and ≤ 4576 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	After any modifications which could affect DG interdependence
SR TR3.8.1.5	Perform a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code in accordance with ASME Code Section XI, Article IWD-5000.	10 years
SR TR3.8.1.6	Verify each DG automatic critical protective functions trip the DG (Reference: GNRO-2005/00056, GNRI-2006/00006). The critical protective functions are Engine Overspeed and Generator Differential Current (Reference: UFSAR 8.3.1.1.4.1.f(2) and 8.3.1.2.1.b.5.(g)).	24 months

TAB 3

Fig 3

Containment Spray Initiation Pressure Limit (CSIPL)

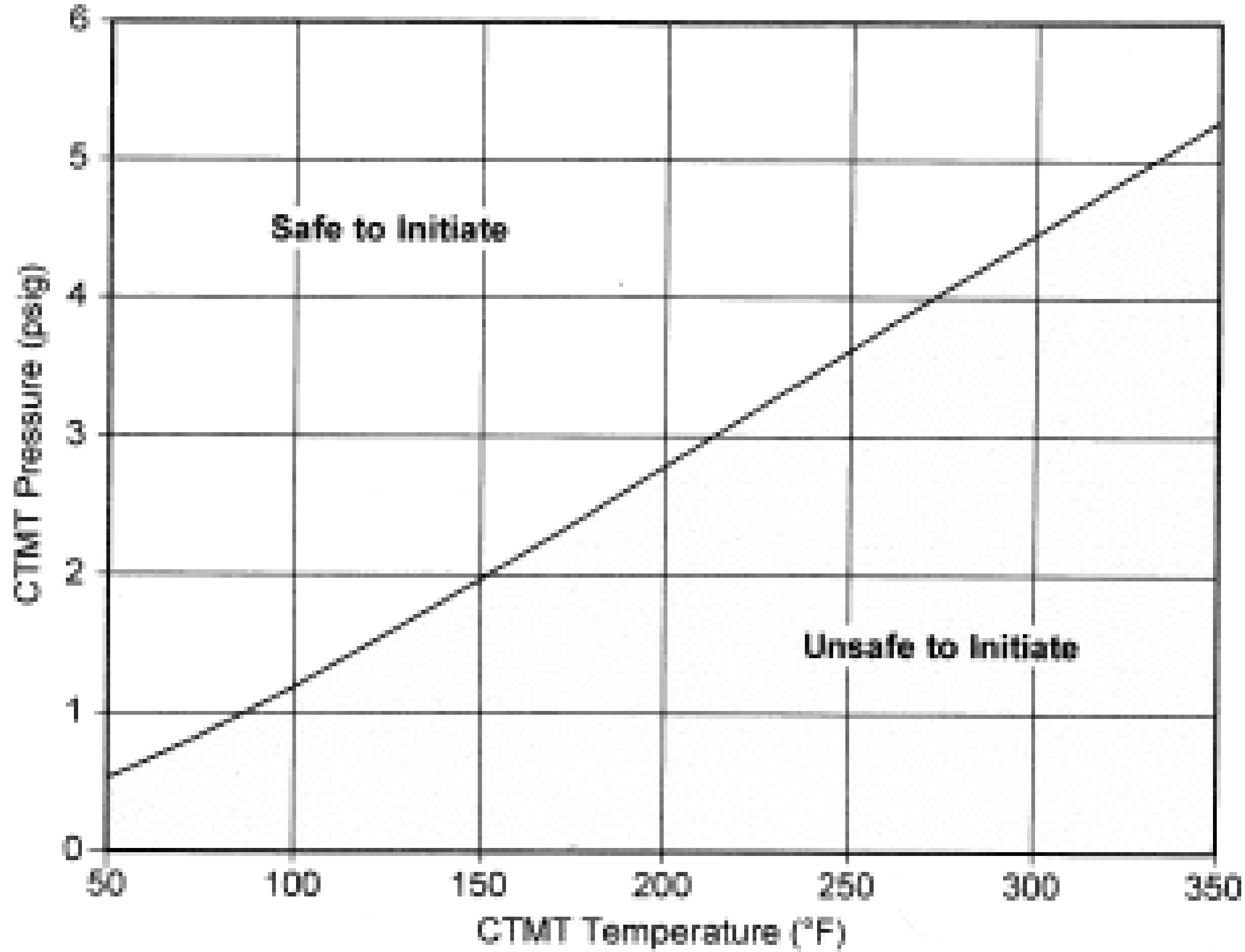
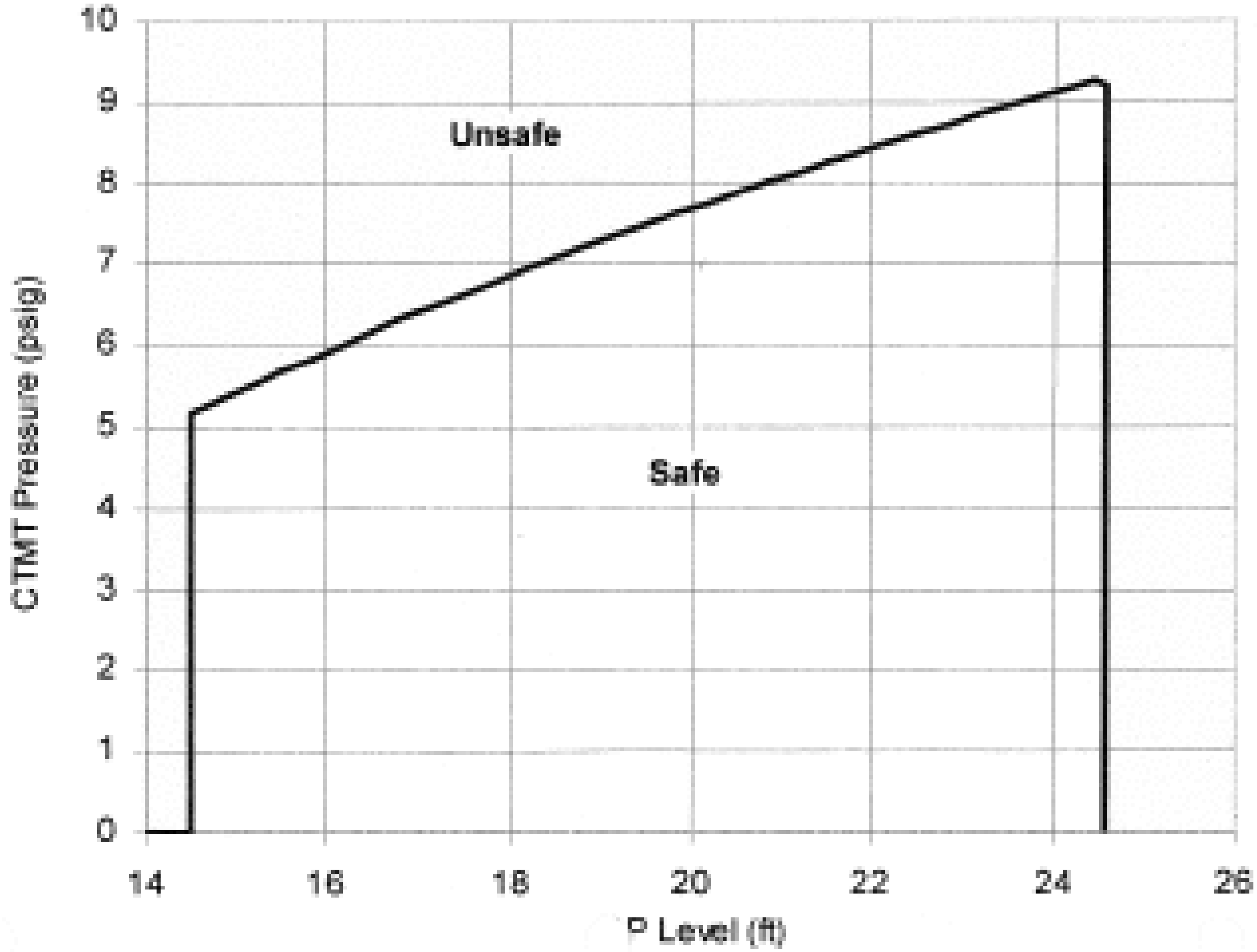


Fig 4

Pressure Suppression Pressure (PSP)



TAB 4

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1

APPLICABILITY:

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip. OR ---------Note-------- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f. --------------------- A.2 Place associated trip system in trip.	12 hours 12 hours
---------NOTE--------- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f. ---------------------- B. One or more Functions with one or more required channels inoperable in both trip systems.	B.1 Place channel in one trip system in trip. OR B.2 Place one trip system in trip.	6 hours 6 hours

(continued)

Table 3.3.1.1-1 (page 4 of 4)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
9. Turbine Stop Valve Closure,Trip Oil Pressure -Low	≥ 35.4% RTP	4	E	SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15	≥ 37 psig
10. Turbine Control Valve Fast Closure, Trip Oil Pressure -Low	≥ 35.4% RTP	2	E	SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15	≥ 42 psig
11. Reactor Mode Switch - Shutdown Position	1,2	2	H	SR 3.3.1.1.11 SR 3.3.1.1.13	NA
	5(a)	2	I	SR 3.3.1.1.11 SR 3.3.1.1.13	NA
12. Manual Scram	1,2	2	H	SR 3.3.1.1.4 SR 3.3.1.1.13	NA
	5(a)	2	I	SR 3.3.1.1.4 SR 3.3.1.1.13	NA

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

Table 3.3.1.1-1 (page 3 of 4)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. Reactor Vessel Steam Dome Pressure - High	1,2	2	H	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 1079.7 psig
4. Reactor Vessel Water Level - Low, Level 3	1,2	2	H	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.15	≥ 10.8 inches
5. Reactor Vessel Water Level - High, Level 8	≥ 21.8% RTP	2	F	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 54.1 inches
6. Main Steam Isolation Valve - Closure	1	8	G	SR 3.3.1.1.8 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 7% closed
7. Drywell Pressure - High	1,2	2	H	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13	≤ 1.43 psig
8. Scram Discharge Volume Water Level - High					
a. Transmitter/Trip Unit	1,2	2	H	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13	≤ 63% of full scale
	5[(a)]	2	I	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13	≤ 63% of full scale
b. Float Switch	1,2	2	H	SR 3.3.1.1.8 SR 3.3.1.1.12 SR 3.3.1.1.13	≤ 65 inches
	5[(a)]	2	I	SR 3.3.1.1.8 SR 3.3.1.1.12 SR 3.3.1.1.13	≤ 65 inches

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

Table 3.3.1.1-1 (page 2 of 4)
Reactor Protection System Instrumentation

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) Two-Loop Operation 0.58W + 59.1% RTP and ≤ 113% RTP
Single-Loop Operation 0.58W + 37.4% RTP

(c) Each channel provides inputs to both trip systems.

(d) If the as-found channel setpoint is outside its pre-defined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(e) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures to confirm channel performance. The NTSP and the methodologies used to determine the as-found and as-left tolerances are specified in the Technical Requirements Manual.

(f) The setpoint for the OPRM Upscale Period-Based Detection algorithm is specified in the COLR.

Table 3.3.1.1-1 (page 1 of 4)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux - High	2	3	H	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.12 SR 3.3.1.1.13	≤ 122/125 divisions of full scale
	5(a)	3	I	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.12 SR 3.3.1.1.13	≤ 122/125 divisions of full scale
b. Inop	2	3	H	SR 3.3.1.1.3 SR 3.3.1.1.13	NA
	5(a)	3	I	SR 3.3.1.1.4 SR 3.3.1.1.13	NA
2. Average Power Range Monitors					
a. Neutron Flux - High, Setdown	2	3(c)	H	SR 3.3.1.1.7 SR 3.3.1.1.10(d)(e) SR 3.3.1.1.19 SR 3.3.1.1.20	≤ 20% RTP
b. Fixed Neutron Flux - High	1	3(c)	G	SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.10(d)(e) SR 3.3.1.1.19 SR 3.3.1.1.20	≤ 119.3% RTP
c. Inop	1,2	3(c)	H	SR 3.3.1.1.20	NA
d. Flow Biased Simulated Thermal Power - High	1	3(c)	G	SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.10(d)(e) SR 3.3.1.1.17 SR 3.3.1.1.19 SR 3.3.1.1.20	(b)
e. 2-Out-Of-4 Voter	1,2	2	H	SR 3.3.1.1.19 SR 3.3.1.1.20 SR 3.3.1.1.21 SR 3.3.1.1.22	NA
f. OPRM Upscale	≥ 21%	3(c)	J	SR 3.3.1.1.7 SR 3.3.1.1.10(d)(e) SR 3.3.1.1.19 SR 3.3.1.1.20 SR 3.3.1.1.23	(f)

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.20	------------------NOTE------------------ 1. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 2. For Functions 2.a, 2.b, and 2.c, the APRM/OPRM channels and the 2-Out-Of-4 Voter channels are included in the CHANNEL FUNCTIONAL TEST. 3. For Functions 2.d and 2.f, the APRM/OPRM channels and the 2-Out-Of-4 Voter channels plus the flow input function, excluding the flow transmitters, are included in the CHANNEL FUNCTIONAL TEST. -- Perform CHANNEL FUNCTIONAL TEST.	184 days
SR 3.3.1.1.21	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.1.22	-----------------NOTE-------------------- For Function 2.e, "n" equals 8 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. Testing APRM and OPRM outputs shall alternate. -- Verify the RPS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS
SR 3.3.1.1.23	Verify OPRM is not bypassed when APRM Simulated Thermal Power is greater than or equal to 26% RTP and recirculation drive flow is less than 60% of rated recirculation drive flow.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.15	-------------------NOTES------------------- 1. Neutron detectors are excluded. 2. For Functions 3, 4, and 5 in Table 3.3.1.1-1, the channel sensors may be excluded. 3. For Function 6, "n" equals 4 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. --- Verify the RPS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS
SR 3.3.1.1.16	Deleted	
SR 3.3.1.1.17	Perform APRM recirculation flow transmitter calibration.	24 months
SR 3.3.1.1.18	Deleted	
SR 3.3.1.1.19	Perform CHANNEL CHECK.	24 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.11	Perform CHANNEL FUNCTIONAL TEST.	24 months
SR 3.3.1.1.12	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. For IRMs, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.1.13	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.1.14	Verify Turbine Stop Valve Closure, Trip Oil Pressure - Low and Turbine Control Valve Fast Closure Trip Oil Pressure - Low Functions are not bypassed when THERMAL POWER is ≥ 35.4% RTP.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.10	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 3. For Function 2.d, APRM recirculation flow transmitters are excluded. --- Perform CHANNEL CALIBRATION.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.5	Deleted	
SR 3.3.1.1.6	Deleted	
SR 3.3.1.1.7	Calibrate the local power range monitors.	2000 MWD/T average core exposure
SR 3.3.1.1.8	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.9	Calibrate the trip units.	92 days

(continued)

SURVEILLANCE REQUIREMENTS

------------------------------------NOTES------------------------------------

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.1.2	-------------------NOTE------------------- Not required to be performed until 12 hours after THERMAL POWER ≥ 21.8% RTP. -- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power ≤ 2% RTP while operating at ≥ 21.8% RTP.	7 days
SR 3.3.1.1.3	-------------------NOTE------------------- Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -- Perform CHANNEL FUNCTIONAL TEST.	7 days
SR 3.3.1.1.4	Perform CHANNEL FUNCTIONAL TEST.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	I.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately
J. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	J.1	Initiate alternate method to detect and suppress thermal hydraulic instability oscillations.	12 hours
	<u>AND</u>		
	J.2	------- NOTE -------- LCO 3.0.4 is not applicable. --------------------- Restore required channels to OPERABLE.	120 days
K. Required Action and associated Completion Time of Condition J not met.	K.1	Reduce THERMAL POWER to < 21% RTP.	4 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more Functions with RPS trip capability not maintained.	C.1	Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1	Reduce THERMAL POWER to < 35.4% RTP.	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1	Reduce THERMAL POWER to < 21.8% RTP.	4 hours
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1	Be in MODE 2.	6 hours
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1	Be in MODE 3.	12 hours

(continued)

EMERGENCY CLASSIFICATION FLOWCHARTS: Modes 1 through 3

ABNORMAL RAD LEVELS/ RADIOLOGICAL EFFLUENT

Plant Modes: 1 Power Operations, 2 Startup, 3 Hot Shutdown, 4 Cold Shutdown, 5 Refueling, D Defueled

Radiological Effluents

GENERAL EMERGENCY

AG1 (Modes 1 2 3 4 5 D)

Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity > 1000 mR TEDE or 5000 mR Thyroid CDE for the actual or projected duration of the release using actual meteorology.

Emergency Action Level(s): (1 or 2 or 3)

NOTE: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, the classification should be based on EAL #2 instead of EAL #1. Do not delay declaration awaiting dose assessment results.

1. **VALID readings on the radiation monitors in Table R1 "GENERAL EMERGENCY" > the reading shown for ≥15 minutes.**

OR

2. **Dose assessment using actual meteorology indicates doses > 1000 mR TEDE or > 5000 mR thyroid CDE at or beyond the site boundary.**

OR

3. **Field survey results indicate closed window dose rates > 1000 mR/hr expected to continue for ≥ ~~60~~ minutes; or analyses of field survey samples indicate thyroid CDE of > 5000 mR for one hour of inhalation, at or beyond site boundary.**

SITE AREA EMERGENCY

AS1 (Modes 1 2 3 4 5 D)

Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity > 100 mR TEDE or 500 mR Thyroid CDE for the actual or projected duration of the release.

Emergency Action Level(s): (1 or 2 or 3)

NOTE: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, the classification should be based on EAL #2 instead of EAL #1. Do not delay declaration awaiting dose assessment results.

1. VALID readings on the radiation monitors in Table R1 "SITE AREA EMERGENCY" > the reading shown for ≥ 15 minutes.

OR

2. Dose assessment using actual meteorology indicates doses > 100 mR TEDE or > 500 mR thyroid CDE at or beyond the site boundary.

OR

3. Field survey results indicate closed window dose rates > 100 mR/hr expected to continue for ≥ 60 minutes; or analyses of field survey samples indicate thyroid CDE of > 500 mR for one hour of inhalation, at or beyond the site boundary.

ALERT

AA1 (Modes 1 2 3 4 5 D)

Any release of gaseous or liquid radioactivity to the environment > 200 times the ODCM limit ≥ 15 minutes

Emergency Action Level(s): (1 or 2 or 3)

NOTE: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

1. VALID reading on the radiation monitors in Table R1 "ALERT" > the reading shown for ≥ 15 minutes.

OR

2. a. For liquid release SD17K606 Radwaste Effluent Radiation Monitor:
   EITHER
   VALID reading > 400 times the Hi-Hi alarm setpoint established by a current radioactivity discharge permit for ≥ 15 minutes.
   OR
   VALID reading ≥ 1E6 cpm for ≥ 15 minutes.
   OR
   b. VALID reading on any effluent monitor > 400 times the Hi-Hi alarm setpoint established by a current radioactivity discharge permit for ≥ 15 minutes..

OR

3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates, >200 times the ODCM limit for ≥ 15 minutes.

UNUSUAL EVENT

AU1 (Modes 1 2 3 4 5 D)

Any release of gaseous or liquid radioactivity to the environment >2 times the ODCM limit for ≥ 60 minutes.

Emergency Action Level(s): (1 or 2 or 3)

NOTE: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

1. VALID readings on the radiation monitors in Table R1 "UNUSUAL EVENT" > the reading shown for ≥ 60 minutes.

OR

2. VALID reading on any effluent monitor > 4 times the Hi-Hi alarm setpoint established by a current radioactivity discharge permit for ≥ 60 minutes.

OR

3. Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates >2 times the ODCM limit for ≥ 60 minutes.

Table R1: EAL THRESHOLD

Method	GENERAL EMERGENCY		SITE AREA EMERGENCY		ALERT		UNUSUAL EVENT	
	Computer Point	Threshold	Computer Point	Threshold	Computer Point	Threshold	Computer Point	Threshold
Release Point Total: OG/Radwaste Vent, FHA Vent, CTMT Vent, Turb Bldg Vent, SBGT A/B	D173004 In Alarm	3.37E+02 Ci/sec	D173003 In Alarm	3.37E+01 Ci/sec	D173002 In Alarm	3.73E+00 Ci/sec	D173001 In Alarm	3.73E-02 Ci/sec

Abnormal Rad Levels

ALERT

AA2 (Modes 1 2 3 4 5 D)

Damage to irradiated fuel or loss of water level that has resulted or will result in the uncovering of irradiated fuel outside the reactor vessel

Emergency Action Level(s): (1 or 2)

1. A water level drop in the Upper Ctmt Pools, Aux Bldg Fuel Pools or Fuel Transfer Canal that will result in irradiated fuel becoming uncovered.

OR

2. A VALID alarm on any of the following radiation monitors due to damage to irradiated fuel or loss of water level:
   Ctmt Vent **(P601-19A-G9)**
   FH Area Vent **(P601-19A-C11)**
   Ctmt 209 Airlock **(P844-1A-A1)**
   Ctmt Fuel Hdlg Area **(P844-1A-A3)**
   Aux Bldg Fuel Hdlg Area **(P844-1A-A4)**

AA3 (Modes 1 2 3 4 5 D)

Rise in radiation levels within the facility that impedes operation of systems required to maintain plant safety functions

Emergency Action Level(s):

1. Dose rate > 15 mR/hr in any of the following areas requiring continous occupancy to maintain plant safety functions:
   - Control Room Envelope

UNUSUAL EVENT

AU2 (Modes 1 2 3 4 5 D)

Unexpected rise in plant radiation

Emergency Action Level(s): (1 or 2)

1 a. VALID indication of uncontrolled water level drop in Upper Ctmt Pools or Aux. Bldg. Fuel Pools or the Fuel Transfer Canal with all irradiated fuel assemblies remaining covered by water

AND

b. Unplanned VALID Area Radiation Monitor reading rises on any of the following:

Ctmt 209 Airlock.................. (1D21K630)
Ctmt Fuel Hdlg Area............ (1D21K626)
Aux Bldg Fuel Hdlg Area...... (1D21K622)

OR

(NOTE: For Control Room envelope review EAL AA3)

2. Unplanned VALID Area Radiation Monitor readings rise by a factor of 1000, or full scale, over normal levels (highest reading in the past 24 hours excluding the current peak value)

FISSION PRODUCT BARRIER DEGRADATION

Plant Modes: 1 Power Operations, 2 Startup, 3 Hot Shutdown

FPB Loss/ Potential Loss

GENERAL EMERGENCY

FG1 (Modes 1 2 3)

Loss of ANY Two Barriers AND Loss or Potential Loss of Third Barrier

Emergency Action Level(s):

1. Loss of ANY Two Barriers AND Loss or Potential Loss of Third Barrier

SITE AREA EMERGENCY

FS1 (Modes 1 2 3)

Loss or Potential Loss of ANY Two Barriers

Emergency Action Level(s):

1. Loss or Potential Loss of ANY Two Barriers

ALERT

FA1 (Modes 1 2 3)

ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCS

Emergency Action Level(s):

1. ANY Loss or ANY Potential Loss of EITHER Fuel Clad or RCS.

UNUSUAL EVENT

FU1 (Modes 1 2 3)

ANY Loss or ANY Potential Loss of Primary Containment.

Emergency Action Level(s):

1. ANY Loss or ANY Potential Loss of Primary Containment.

Fission Product Barrier Matrix

Fuel Clad			Reactor Coolant System			Primary Containment		
Parameter	Loss	Potential Loss	Parameter	Loss	Potential Loss	Parameter	Loss	Potential Loss
FC1 Primary Coolant Activity	Coolant activity > 300 μCi/gm dose eq. I[131]	None	RC1 Drywell Pressure	Drywell pressure > 1.39 psig with indications of a reactor coolant leak in the drywell	None	PC1 Primary Containment Pressure	1. Rapid unexplained loss of pressure following initial pressure rise. OR 2. Pressure response not consistent with LOCA conditions.	1. Primary Containment pressure > 15 psig and rising. OR 2. a. DW H_2 concentration >9% OR b. Ctmt H_2 concentration in HDOL Unsafe Zone. OR 3. RPV pressure and suppression pool temperature cannot be maintained below the HCTL.
FC2 RPV Water Level	RPV water level cannot be restored and maintained above -191"	RPV water level cannot be restored and maintained above -167" or cannot be determined.	RC2 RPV water level	RPV water level cannot be restored and maintained above -167 in. or cannot be determined.	None	PC2 RPV water level	None	Primary Containment flooding required by: · SAPs
FC3 Drywell Radiation Monitoring	Drywell Radiation Monitor reading >3000 R/hr	None	RC3 RCS Leak Rate	1. a. UNISOLABLE MSL break as indicated by the failure of both MSIVs in any one line to close AND High MSL flow (P601-19A-E1) annunciator OR High Steam Tunnel Temperature (P601-19A-E3) annunciator OR Direct report of steam release. OR b. Indication of an UNISOLABLE HPCI, feedwater, RWCU or RCIC break. OR 2. Emergency RPV depressurization is required.	1. RCS leakage > 50 gpm inside the drywell. OR 2. UNISOLABLE RCS leakage outside Primary Containment as indicated by any Area Temperature or Area Radiation level > Alert Value in Table F1.	PC3 Primary Containment Isolation Failure or Bypass	1. a. Failure of all valves in any one line to close. AND b. Direct downstream pathway to the environment exists after Primary Containment Isolation signal. OR 2. Intentional venting per EOPs or SAPs. OR 3. UNISOLABLE RCS leakage outside Primary Containment as indicated by any Area Temperature or Area Radiation level > SAE / GE Value in Table F1	None
FC4 ED Judgment	Any condition in the opinion of the Emergency Director that indicates a loss of the Fuel Clad Barrier	Any condition in the opinion of the Emergency Director that indicates a potential loss of the Fuel Clad Barrier.	RC4 Drywell Radiation Monitoring	Drywell Radiation monitor reading > 100 R/hr with indications of a leak in the drywell	None	PC4 Significant Radioactive Inventory in Primary Containment	None	Containment radiation monitor reading > 10,000 R/hr
			RC5 ED Judgment	Any condition in the opinion of the Emergency Director that indicates a loss of the RCS Barrier	Any condition in the opinion of the Emergency Director that indicates a potential loss of the RCS Barrier	PC5 ED Judgment	Any condition in the opinion of the Emergency Director that indicates a loss of the Primary Containment Barrier	Any condition in the opinion of the Emergency Director that indicates a potential loss of the Primary Containment Barrier

Table F1

Area Temperature			Area Radiation Level		
Parameter	Alert Limit	SAE/GE Limit	Parameter	Alert Limit	SAE/GE Limit
MSL Pipe Tunnel Temp	185°F (P601 - 19A/18A - A3/A4)	250°F (E31-N604A, B, C, D, E, F)			
RHR-A Equipment Area Temp	165°F (P601 - 20A - B1)	225°F (E31-N608A, N610A)	RHR Room A Radiation	10^2 mR/hr (P844-1A-D4)	8 X 10^4 mR/hr
RHR-B Equipment Area Temp	165°F (P601 - 20A - B1)	225°F (E31-N608B, N610B)	RHR Room B Radiation	10^2 mR/hr (P844-1A-D4)	8 X 10^4 mR/hr
RCIC Equipment Area Temp	185°F (P601 - 21A - G3)	212°F (E31-N602A/B)	RHR HX A Hatch Radiation	10^2 mR/hr (P844-1A-C4)	8 X 10^4 mR/hr
RWCU Pump Rm 1 Temperature	170°F (P680 - 11A - A1)	NA	RHR HX B Hatch Radiation	10^2 mR/hr (P844-1A-C4)	8 X 10^4 mR/hr
RWCU Pump Rm 2 Temperature	170°F (P680 - 11A - A2)	NA	RCIC Room Radiation	10^2 mR/hr (P844-1A-D4)	8 X 10^4 mR/hr

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

Plant Modes: 1 Power Operations, 2 Startup, 3 Hot Shutdown, 4 Cold Shutdown, 5 Refueling, D Defueled

Security

GENERAL EMERGENCY

HG1 (Modes 1 2 3 4 5 D)

HOSTILE ACTION resulting in loss of physical control of the facility.

Emergency Action Level(s): (1 or 2)

1. A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions.

OR

2. A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and IMMINENT fuel damage is likely for a freshly off-loaded reactor core in pool.

SITE AREA EMERGENCY

HS1 (Modes 1 2 3 4 5 D)

HOSTILE ACTION within the PROTECTED AREA

Emergency Action Level(s):

1. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the GGNS security shift supervision.

ALERT

HA1 (Modes 1 2 3 4 5 D)

HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat.

Emergency Action Level(s): (1 or 2)

1. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the GGNS security shift supervision.

OR

2. A validated notification from NRC of an airliner attack threat within 30 minutes of the site.

UNUSUAL EVENT

HU1 (Modes 1 2 3 4 5 D)

Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant.

Emergency Action Level(s): (1 or 2 or 3)

1. A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by the GGNS security shift supervision.

OR

2. A credible site specific security threat notification.

OR

3. A validated notification from NRC providing information of an aircraft threat.

Discretionary

HG2 (Modes 1 2 3 4 5 D)

Other Conditions exist which in the judgment of the Emergency Director warrant declaration of a GENERAL EMERGENCY.

Emergency Action Level(s):

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

HS2 (Modes 1 2 3 4 5 D)

Other conditions exist which in the judgment of the Emergency Director warrant declaration of a SITE AREA EMERGENCY.

Emergency Action Level(s):

1. Other conditions exist which in the judgement of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to 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 EPS Protective Action Guideline exposure levels beyond the site boundary.

HA2 (Modes 1 2 3 4 5 D)

Other conditions exist which in the judgment of the Emergency Director warrant declaration of an ALERT.

Emergency Action Level(s):

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve 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

HU2 (Modes 1 2 3 4 5 D)

Other conditions exist which in the judgment of the Emergency Director warrant declaration of an UNUSUAL EVENT.

Emergency Action Level(s):

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

Main Control Room Evacuation

HS3 (Modes 1 2 3 4 5 D)

Control Room evacuation has been Initiated and plant control cannot be established.

Emergency Action Level(s):

1. a. Control Room evacuation has been initiated
   AND
   b. Control of the plant cannot be established in accordance with 05-1-02-II-1, Shutdown from the Remote Shutdown Panel, within 15 minutes.

HA3 (Modes 1 2 3 4 5 D)

Control Room evacuation has been initiated

Emergency Action Level(s):

1. 05-1-02-II-1, Shutdown from the Remote Shutdown Panel requires Control Room evacuation.

Fire

HA4 (Modes 1 2 3 4 5 D)

FIRE or EXPLOSION affecting the operability of plant safety systems required to establish or maintain safe shutdown.

Emergency Action Level(s):

1. FIRE or EXPLOSION resulting in VISIBLE DAMAGE to any of the structures or areas in Table H2 containing safety systems or components or Control Room indication of degraded performance of those safety systems.

HU4 (Modes 1 2 3 4 5 D)

FIRE within the PROTECTED AREA not extinguished within 15 minutes of detection or EXPLOSION within the PROTECTED AREA.

Emergency Action Level(s):

Note: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the duration has exceeded, or will likely exceed, the applicable time.

1. Fire in any Table H-2 structure or area NOT extinguished:
   a. Within 15 minutes of Control Room notification
   OR
   b. Within 15 minutes of verification of a Control Room FIRE alarm.

OR

2. EXPLOSION within the PROTECTED AREA.

Toxic or Flammable Gas

HA5 (Modes 1 2 3 4 5 D)

Access to a VITAL AREA is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of operable equipment required to maintain safe operations or safely shutdown the reactor.

Emergency Action Level(s):

Note: If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.

1. Access to the Control Room while in any operating mode or the Reactor Auxiliary Building while in modes 3, 4, or 5 ONLY is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shutdown the reactor.

HU5 (Modes 1 2 3 4 5 D)

Release of toxic, corrosive, asphyxiant or flammable gases deemed detrimental to NORMAL PLANT OPERATIONS.

Emergency Action Level(s): (1 or 2)

1. Toxic, corrosive, asphyxiant or flammable gases in amounts that have or could adversely affect NORMAL PLANT OPERATIONS.

OR

2. Report by Local, County/Parish or State Officials for evacuation or sheltering of site personnel based on an offsite event.

Natural or Destructive Phenomena

Table H1

Auxiliary Building Area Parameters

AREA	Max Safe Operating Value
RHR Room A	93 FT. 6 IN. (P870-2A-E1)
RHR Room B	93 FT. 6 IN. (P870-10A-G1)
RHR Room C	93 FT. 6 IN. (P870-10A-G2)
RCIC Room	93 FT. 6 IN. (P870-2A-A1)
LPCS Room	93 FT. 6 IN. (P870-2A-F1)
HPCS Room	93 FT. 6 IN. (P870-5A-H1)

Table H2

Structures Containing Functions or Systems Required for Safe Shutdown

UNIT I CONTAINMENT
UNIT I AUXILIARY BUILDING
CONTROL BUILDING
UNIT I TURBINE BUILDING
DIESEL GENERATOR ROOMS
SSW PUMP & VALVE ROOMS

HA6 (Modes 1 2 3 4 5 D)

Natural or destructive phenomena affecting VITAL AREAS.

Emergency Action Levels: (1 or 2 or 3 or 4 or 5 or 6)

1. a. Seismic event > Operating Basis Earthquake (OBE) as indicated by:
   - Receipt of **EITHER** of the following indications on SH13P856:
     - Containment Operating Basis Earthquake (P856-1A-A3)
     OR
     - Drywell Operating Basis Earthquake (P856-1A-A5)
   AND
   b. Earthquake confirmed by any of the following:
     - Earthquake felt in plant
     - National Earthquake Center
     - Control Room indication of degraded performance of systems required for the safe shutdown of the plant.

OR

2. Tornado striking resulting in VISIBLE DAMAGE to any of the Table H2 structures or areas containing safety systems or components or Control Room indication of degraded performance of those safety systems.

OR

3. Internal flooding in any Table H1 areas resulting in an electrical shock hazard that precludes access to operate or monitor safety equipment or Control Room indication of degraded performance of those safety systems.

OR

4. Turbine failure-generated PROJECTILES resulting in VISIBLE DAMAGE to or penetration of any of the Table H2 structures or areas containing safety systems or components or Control Room indication of degraded performance of those safety systems.

OR

5. Vehicle crash resulting in VISIBLE DAMAGE to any of the Table H2 structures or areas containing safety systems or components or Control Room indication of degraded performance of those safety systems.

OR

6. Severe weather with indication of sustained high winds ≥ 74 mph within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the Table H2 structures or areas containing safety systems or components or Control Room indication of degraded performance of those safety systems.

HU6 (Modes 1 2 3 4 5 D)

Natural or destructive phenomena affecting the PROTECTED AREA.

Emergency Action Level(s): (1 or 2 or 3 or 4 or 5)

1. Seismic event identified by any 2 of the following:
   - Seismic event confirmed by activated seismic switches as indicated by activation of the Seismic Monitoring System: Strong Motion Accelerometer System Activation (P856-1A-A1)
   - Earthquake felt in plant
   - National Earthquake Center

OR

2. Tornado striking within PROTECTED AREA boundary.

OR

3. Internal flooding that has the potential to affect safety related equipment required by Technical Specifications for the current operating mode in any Table H1 area.

OR

4. Turbine failure resulting in casing penetration or damage to turbine or generator seals.

OR

5. Severe weather with indication of sustained high winds ≥ 74 mph within PROTECTED AREA boundary.

SYSTEM MALFUNCTIONS

Plant Modes: 1 Power Operations, 2 Startup, 3 Hot Shutdown

Loss of AC Power

GENERAL EMERGENCY

SG1 (Modes 1 2 3)

Prolonged loss of all offsite and all onsite AC power to Div I, II & III ESF busses

Emergency Action Level(s):

1. a. Loss of all offsite and all onsite AC power to Div I, II, & III ESF busses.
   AND
   b. Either of the following:
   - Restoration of at least one emergency bus in < 4 hours is not likely.
   OR
   - RPV level can not be maintained > -167 in.

SITE AREA EMERGENCY

SS1 (Modes 1 2 3)

Loss of all offsite and all onsite AC power to Div. I, II & III ESF busses for ≥ 15 minutes.

Emergency Action Level(s):

Note: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

1. Loss of all offsite and all onsite AC power to Div I, II & III ESF busses for ≥ 15 minutes.

ALERT

SA1 (Modes 1 2 3)

AC power capability to Div. I & II ESF busses reduced to a single power source for ≥ 15 minutes such that any additional single failure would result in station blackout.

Emergency Action Level(s):

Note: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

1. a. AC power capability to Div I & II ESF busses reduced to a single power source for ≥ 15 minutes.
   AND
   b. Any additional single power source failure will result in station blackout.

UNUSUAL EVENT

SU1 (Modes 1 2 3)

Loss of all offsite AC power to Div. I & II ESF busses for ≥ 15 minutes

Emergency Action Level(s):

Note: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

1. Loss of all offsite AC power to Div I & II ESF busses for ≥ 15 minutes.

Failure of Reactor Protection System

SG3 (Modes 1 2)

Automatic scram and all manual actions fail to shutdown the reactor and indication of an extreme challenge to the ability to cool the core exists.

Emergency Action Level(s):

1. a. An automatic scram failed to shutdown the reactor
   AND
   b. All manual actions do not shutdown the reactor as indicated by reactor power ≥ 4%
   AND
   c. Either of the following exist or have occurred due to continued power generation:
   - Core cooling is extremely challenged as indicated by RPV level can not be restored and maintained > -191 in.
   OR
   - Heat removal is extremely challenged as indicated by RPV pressure and Suppression Pool temperature cannot be maintained in the EOP Heat Capacity Temperature Limit (HCTL) Safe Zone.

SS3 (Modes 1 2)

Automatic scram fails to shutdown the reactor and the manual actions taken from the reactor control console are not successful in shutting down the reactor.

Emergency Action Level(s):

1. a. An automatic scram failed to shutdown the reactor
   AND
   b. Manual actions taken at the reactor control console do not shutdown the reactor as indicated by reactor power ≥ 4%.

SA3 (Modes 1 2)

Automatic scram fails to shutdown the reactor and the manual actions taken from the reactor control console are successful in shutting down the reactor.

Emergency Action Level(s):

1. a. An automatic scram failed to shutdown the reactor as indicated by reactor power ≥ 4%
   AND
   b. Manual actions taken at the reactor control console successfully shutdown the reactor as indicated by reactor power <4%.

Loss of DC Power

SS4 (Modes 1 2 3)

Loss of all vital DC power for ≥ 15 minutes.

Emergency Action Level(s):

Note: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

1. < 105 VDC on all vital DC busses for ≥ 15 minutes.

Loss of Annunciation / Indication

SS6 (Modes 1 2 3)

Inability to monitor a SIGNIFICANT TRANSIENT in progress.

Emergency Action Level(s):

Note: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

1. a. UNPLANNED loss of > approximately 75% of the following for ≥ 15 minutes:
   - Control Room safety system annunciation.
   OR
   - Control Room safety system indication.
   AND
   b. A SIGNIFICANT TRANSIENT is in progress.
   AND
   c. Compensatory indications are unavailable.

SA6 (Modes 1 2 3)

UNPLANNED loss of safety system annunciation or indication in the Control Room with either (1) a SIGNIFICANT TRANSIENT in progress, or (2) compensatory indicators unavailable.

Emergency Action Level(s):

Note: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

1. a. UNPLANNED loss of > approximately 75% of the following for ≥ 15 minutes:
   - Control Room safety system annunciation.
   OR
   - Control Room safety system indication.
   AND
   b. Either of the following:
   - A SIGNIFICANT TRANSIENT is in progress.
   OR
   - Compensatory indications are unavailable.

SU6 (Modes 1 2 3)

UNPLANNED loss of safety system annunciation or indication in the Control Room for ≥ 15 minutes.

Emergency Action Level(s):

Note: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

1. UNPLANNED loss of > approximately 75% of the following for ≥ 15 minutes:
   a. Control Room safety system annunciation.
   OR
   b. Control Room safety system indication.

Reactor Coolant System Leakage

SU7 (Modes 1 2 3)

RCS leakage

Emergency Action Level(s): (1 or 2)

1. Unidentified or pressure boundary leakage >10 gpm.

OR

2. Identified leakage > 35 gpm.

Loss of Communication

Table S1

Onsite Communications Methods

Plant Radio System
Plant Paging System
Sound Powered Phones
In plant Telephones

Table S2

Offsite Communications Methods

All telephone lines (commercial & fiber optic)
Satellite telephone
NRC phones (ENS, HPN, MCL, RSCL, PMCL)

SU8 (Modes 1 2 3)

Loss of all onsite and offsite communications capabilities.

Emergency Action Level(s): (1 or 2)

1. Loss of all Table S1 onsite communications methods affecting the ability to perform routine operations.

OR

2. Loss of all Table S2 offsite communications methods affecting the ability to perform offsite notifications.

Cladding Degradation

Table S3

Offgas Pre-treatment Rad Monitor

Offgas Flow (cfm)	Radiation Monitor Limit (mR/hr)
0 - 65	1400
66 - 130	700
131 - 200	460
201 - 300	310
301 - 400	230

SU9 (Modes 1 2 3)

Fuel clad degradation

Emergency Action Level(s): (1 or 2)

1. Offgas Pre-Treatment radiation monitor (D17R604 or 1D17K612) reading > the Table S3 Limit for the actual indicated Offgas flow indicating fuel clad degradation > T.S. allowable limits.

OR

2. Reactor coolant sample activity value indicating fuel clad degradation > T.S. allowable limits
   - > 4.0 μCi/gm dose equivalent I-131.
   OR
   - > 0.2 μCi/gm dose equivalent I-131 for > 48 hours.

Inadvertent Criticality

SU10 (Mode 3)

Inadvertent criticality

Emergency Action Level(s):

1. UNPLANNED sustained positive period observed on nuclear instrumentation.

Tech Spec Time Limit Exceeded

SU11 (Modes 1 2 3)

Inability to reach required operating mode within Technical Specification limits.

Emergency Action Level(s):

1. Plant is not brought to required operating mode within Technical Specifications LCO Action Statement Time

EMERGENCY CLASSIFICATION FLOWCHARTS: Modes 4 through De-Fuel

ABNORMAL RAD LEVELS/ RADIOLOGICAL EFFLUENT

Plant Modes: 1 Power Operations, 2 Startup, 3 Hot Shutdown, 4 Cold Shutdown, 5 Refueling, D Defueled

Radiological Effluents

GENERAL EMERGENCY

AG1 (1 2 3 4 5 D)

Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity > 1000 mR TEDE or 5000 mR Thyroid CDE for the actual or projected duration of the release using actual meteorology.

Emergency Action Level(s): (1 or 2 or 3)

NOTE: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, the classification should be based on EAL #2 instead of EAL #1. Do not delay declaration awaiting dose assessment results.

1. **VALID readings on the radiation monitors in Table R1 "GENERAL EMERGENCY" > the reading shown for ≥15 minutes.**

OR

2. **Dose assessment using actual meteorology indicates doses > 1000 mR TEDE or > 5000 mR thyroid CDE at or beyond the site boundary.**

OR

3. **Field survey results indicate closed window dose rates > 1000 mR/hr expected to continue for ≥ 60 minutes; or analyses of field survey samples indicate thyroid CDE of > 5000 mR for one hour of inhalation, at or beyond site boundary.**

SITE AREA EMERGENCY

AS1 (1 2 3 4 5 D)

Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity > 100 mR TEDE or 500 mR Thyroid CDE for the actual or projected duration of the release.

Emergency Action Level(s): (1 or 2 or 3)

NOTE: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, the classification should be based on EAL #2 instead of EAL #1. Do not delay declaration awaiting dose assessment results.

1. VALID readings on the radiation monitors in Table R1 "SITE AREA EMERGENCY" > the reading shown for ≥ 15 minutes.

OR

2. Dose assessment using actual meteorology indicates doses > 100 mR TEDE or > 500 mR thyroid CDE at or beyond the site boundary.

OR

3. Field survey results indicate closed window dose rates > 100 mR/hr expected to continue for ≥ 60 minutes; or analyses of field survey samples indicate thyroid CDE of > 500 mR for one hour of inhalation, at or beyond the site boundary.

ALERT

AA1 (1 2 3 4 5 D)

Any release of gaseous or liquid radioactivity to the environment > 200 times the ODCM limit ≥ 15 minutes

Emergency Action Level(s): (1 or 2 or 3)

NOTE: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

1. VALID reading on the radiation monitors in Table R1 "ALERT" > the reading shown for ≥ 15 minutes.

OR

2. a. For liquid release SD17K606 Radwaste Effluent Radiation Monitor:
   EITHER
   VALID reading > 400 times the Hi-Hi alarm setpoint established by a current radioactivity discharge permit for ≥ 15 minutes.
   OR
   VALID reading ≥ 1E6 cpm for ≥ 15 minutes.
   OR
   b. VALID reading on any effluent monitor > 400 times the Hi-Hi alarm setpoint established by a current radioactivity discharge permit for ≥ 15 minutes.

OR

3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates, >200 times the ODCM limit for ≥ 15 minutes.

UNUSUAL EVENT

AU1 (1 2 3 4 5 D)

Any release of gaseous or liquid radioactivity to the environment >2 times the ODCM limit for ≥ 60 minutes.

Emergency Action Level(s): (1 or 2 or 3)

NOTE: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

1. VALID readings on the radiation monitors in Table R1 "UNUSUAL EVENT" > the reading shown for ≥ 60 minutes.

OR

2. VALID reading on any effluent monitor > 4 times the Hi-Hi alarm setpoint established by a current radioactivity discharge permit for ≥ 60 minutes.

OR

3. Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates >2 times the ODCM limit for ≥ 60 minutes.

Table R1: EAL THRESHOLD

Method	GENERAL EMERGENCY		SITE AREA EMERGENCY		ALERT		UNUSUAL EVENT	
Release Point Total:	Computer Point	Threshold	Computer Point	Threshold	Computer Point	Threshold	Computer Point	Threshold
OG/Radwaste Vent FHA Vent CTMT Vent Turb Bldg Vent SBGT A/B	D173004 In Alarm	3.37E+02 Ci/sec	D173003 In Alarm	3.37E+01 Ci/sec	D173002 In Alarm	3.73E+00 Ci/sec	D173001 In Alarm	3.73E-02 Ci/sec

Abnormal Rad Levels

ALERT

AA2 (1 2 3 4 5 D)

Damage to irradiated fuel or loss of water level that has resulted or will result in the uncovering of irradiated fuel outside the reactor vessel

Emergency Action Level(s): (1 or 2)

1. A water level drop in the Upper Ctmt Pools, Aux Bldg Fuel Pools or Fuel Transfer Canal that will result in irradiated fuel becoming uncovered.

OR

2. A VALID alarm on any of the following radiation monitors due to damage to irradiated fuel or loss of water level:

   Ctmt Vent **(P601-19A-G9)**
   FH Area Vent **(P601-19A-C11)**
   Ctmt 209 Airlock **(P844-1A-A1)**
   Ctmt Fuel Hdlg Area **(P844-1A-A3)**
   Aux Bldg Fuel Hdlg Area **(P844-1A-A4)**

AA3 (1 2 3 4 5 D)

Rise in radiation levels within the facility that impedes operation of systems required to maintain plant safety functions

Emergency Action Level(s):

1. Dose rate > 15 mR/hr in any of the following areas requiring continous occupancy to maintain plant safety functions:
   - Control Room Envelope

UNUSUAL EVENT

AU2 (1 2 3 4 5 D)

Unexpected rise in plant radiation

Emergency Action Level(s): (1 or 2)

1 a. VALID indication of uncontrolled water level drop in Upper Ctmt Pools or Aux. Bldg. Fuel Pools or the Fuel Transfer Canal with all irradiated fuel assemblies remaining covered by water

AND

b. Unplanned VALID Area Radiation Monitor reading rises on any of the following:

   Ctmt 209 Airlock.................. (1D21K630)
   Ctmt Fuel Hdlg Area........... (1D21K626)
   Aux Bldg Fuel Hdlg Area...... (1D21K622)

OR

(NOTE: For Control Room envelope review EAL AA3)

2. Unplanned VALID Area Radiation Monitor readings rise by a factor of 1000, or full scale, over normal levels (highest reading in the past 24 hours excluding the current peak value)

Cask Damage

Plant Modes: 1 Power Operations, 2 Startup, 3 Hot Shutdown, 4 Cold Shutdown, 5 Refueling, D Defueled

GENERAL EMERGENCY | SITE AREA EMERGENCY | ALERT | UNUSUAL EVENT

UNUSUAL EVENT

E-HU1 (1 2 3 4 5 D)

Damage to a loaded cask CONFINEMENT BOUNDARY.

Emergency Action Level(s):

1. Damage to a loaded cask CONFINEMENT BOUNDARY.

HAZARDS AND OTHER MALFUNCTIONS

Plant Modes: 1 Power Operations, 2 Startup, 3 Hot Shutdown, 4 Cold Shutdown, 5 Refueling, D Defueled

Security

GENERAL EMERGENCY

HG1 (1 2 3 4 5 D)

HOSTILE ACTION resulting in loss of physical control of the facility.

Emergency Action Level(s): (1 or 2)

1. A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions.

OR

2. A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and IMMINENT fuel damage is likely for a freshly off-loaded reactor core in pool.

SITE AREA EMERGENCY

HS1 (1 2 3 4 5 D)

HOSTILE ACTION within the PROTECTED AREA

Emergency Action Level(s):

1. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the GGNS security shift supervision.

ALERT

HA1 (1 2 3 4 5 D)

HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat.

Emergency Action Level(s): (1 or 2)

1. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the GGNS security shift supervision.

OR

2. A validated notification from NRC of an airliner attack threat within 30 minutes of the site.

UNUSUAL EVENT

HU1 (1 2 3 4 5 D)

Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant.

Emergency Action Level(s): (1 or 2 or 3)

1. A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by the GGNS security shift supervision.

OR

2. A credible site specific security threat notification.

OR

3. A validated notification from NRC providing information of an aircraft threat.

Discretionary

GENERAL EMERGENCY

HG2 (1 2 3 4 5 D)

Other Conditions exist which in the judgment of the Emergency Director warrant declaration of a GENERAL EMERGENCY.

Emergency Action Level(s):

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

SITE AREA EMERGENCY

HS2 (1 2 3 4 5 D)

Other conditions exist which in the judgment of the Emergency Director warrant declaration of a SITE AREA EMERGENCY.

Emergency Action Level(s):

1. Other conditions exist which in the judgement of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to 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 EPS Protective Action Guideline exposure levels beyond the site boundary.

ALERT

HA2 (1 2 3 4 5 D)

Other conditions exist which in the judgment of the Emergency Director warrant declaration of an ALERT.

Emergency Action Level(s):

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve 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

HU2 (1 2 3 4 5 D)

Other conditions exist which in the judgment of the Emergency Director warrant declaration of an UNUSUAL EVENT.

Emergency Action Level(s):

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

Main Control Room Evacuation

SITE AREA EMERGENCY

HS3 (1 2 3 4 5 D)

Control Room evacuation has been Initiated and plant control cannot be established.

Emergency Action Level(s):

1. a. Control Room evacuation has been initiated
   AND
   b. Control of the plant cannot be established in accordance with 05-1-02-II-1, Shutdown from the Remote Shutdown Panel, within 15 minutes.

ALERT

HA3 (1 2 3 4 5 D)

Control Room evacuation has been initiated

Emergency Action Level(s):

1. 05-1-02-II-1, Shutdown from the Remote Shutdown Panel requires Control Room evacuation.

Fire

ALERT

HA4 (1 2 3 4 5 D)

FIRE or EXPLOSION affecting the operability of plant safety systems required to establish or maintain safe shutdown.

Emergency Action Level(s):

1. FIRE or EXPLOSION resulting in VISIBLE DAMAGE to any of the structures or areas in Table H2 containing safety systems or components or Control Room indication of degraded performance of those safety systems.

UNUSUAL EVENT

HU4 (1 2 3 4 5 D)

FIRE within the PROTECTED AREA not extinguished within 15 minutes of detection or EXPLOSION within the PROTECTED AREA.

Emergency Action Level(s):

***Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the duration has exceeded, or will likely exceed, the applicable time.*

1. FIRE in any table H-2 structure not extinguished:

   a. Within 15 minutes of Control Room notification

   OR

   b. Within 15 minutes of verification of a Control Room FIRE alarm

OR

2. EXPLOSION within the PROTECTED AREA.

Toxic or Flammable Gases

ALERT

HA5 (1 2 3 4 5 D)

Access to a VITAL AREA is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of operable equipment required to maintain safe operations or safely shutdown the reactor.

Emergency Action Level(s):

***Note:** If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.*

1. Access to the Control Room while in any operating mode or the Reactor Auxiliary Building while in modes 3, 4, or 5 ONLY is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shutdown the reactor.

UNUSUAL EVENT

HU5 (1 2 3 4 5 D)

Release of toxic, corrosive, asphyxiant or flammable gases deemed detrimental to NORMAL PLANT OPERATIONS.

Emergency Action Level(s): (1 or 2)

1. Toxic, corrosive, asphyxiant or flammable gases in amounts that have or could adversely affect NORMAL PLANT OPERATIONS.

OR

2. Report by Local, County/Parish or State Officials for evacuation or sheltering of site personnel based on an offsite event.

Natural or Destructive Phenomena

Table H1

Auxiliary Building Area Parameters

AREA	Max Safe Operating Value
RHR Room A	93 FT. 6 IN. (P870-2A-E1)
RHR Room B	93 FT. 6 IN. (P870-10A-G1)
RHR Room C	93 FT. 6 IN. (P870-10A-G2)
RCIC Room	93 FT. 6 IN. (P870-2A-A1)
LPCS Room	93 FT. 6 IN. (P870-2A-F1)
HPCS Room	93 FT. 6 IN. (P870-5A-H1)

Table H2

Structures Containing Functions or Systems Required for Safe Shutdown

UNIT I CONTAINMENT
UNIT I AUXILIARY BUILDING
CONTROL BUILDING
UNIT I TURBINE BUILDING
DIESEL GENERATOR ROOMS
SSW PUMP & VALVE ROOMS

ALERT

HA6 (1 2 3 4 5 D)

Natural or destructive phenomena affecting VITAL AREAS.

Emergency Action Levels: (1 or 2 or 3 or 4 or 5 or 6)

1. a. Seismic event > Operating Basis Earthquake (OBE) as indicated by:
   - Receipt of EITHER of the following indications on SH13P856:
     - Containment Operating Basis Earthquake (P856-1A-A3)
       OR
     - Drywell Operating Basis Earthquake (P856-1A-A5)

   AND

   b. Earthquake confirmed by any of the following:
   - Earthquake felt in plant
   - National Earthquake Center
   - Control Room indication of degraded performance of systems required for the safe shutdown of the plant.

OR

2. Tornado striking resulting in VISIBLE DAMAGE to any of the Table H2 structures or areas containing safety systems or components or Control Room indication of degraded performance of those safety systems.

OR

3. Internal flooding in any Table H1 areas resulting in an electrical shock hazard that precludes access to operate or monitor safety equipment or Control Room indication of degraded performance of those safety systems.

OR

4. Turbine failure-generated PROJECTILES resulting in VISIBLE DAMAGE to or penetration of any of the Table H2 structures or areas containing safety systems or components or Control Room indication of degraded performance of those safety systems.

OR

5. Vehicle crash resulting in VISIBLE DAMAGE to any of the Table H2 structures or areas containing safety systems or components or Control Room indication of degraded performance of those safety systems.

OR

6. Severe weather with indication of sustained high winds ≥ 74 mph within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the Table H2 structures or areas containing safety systems or components or Control Room indication of degraded performance of those safety systems.

UNUSUAL EVENT

HU6 (1 2 3 4 5 D)

Natural or destructive phenomena affecting the PROTECTED AREA.

Emergency Action Level(s): (1 or 2 or 3 or 4 or 5)

1. Seismic event identified by any 2 of the following:
   - Seismic event confirmed by activated seismic switches as indicated by activation of the Seismic Monitoring System: Strong Motion Accelerometer System Activation (P856-1A-A1)
   - Earthquake felt in plant
   - National Earthquake Center

OR

2. Tornado striking within PROTECTED AREA boundary.

OR

3. Internal flooding that has the potential to affect safety related equipment required by Technical Specifications for the current operating mode in any Table H1 area.

OR

4. Turbine failure resulting in casing penetration or damage to turbine or generator seals.

OR

5. Severe weather with indication of sustained high winds ≥ 74 mph within PROTECTED AREA boundary.

COLD SHUTDOWN / REFUELING

Plant Modes: 4 Cold Shutdown, 5 Refueling, D Defueled

Cold Shutdown/Refueling Loss of RCS/RPV Inventory

GENERAL EMERGENCY

CG1 (4 5)

Loss of RCS inventory affecting fuel clad integrity with containment challenged.

Emergency Action Level(s): (1 or 2)

***Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. a. RPV level < -167 in. (TAF) for ≥ 30 minutes
   AND
   b. Any containment challenge indication in Table C1

OR

2. a. RCS level cannot be monitored with core uncovery indicated by any of the following for ≥ 30 minutes
   - Containment High Range Radiation Monitor reading > 100 R/hr.
   - Erratic Source Range Monitor indication.
   - Unexplained rise in floor or equipment sump level, Suppression Pool level, vessel make-up rate or observation of leakage or inventory loss.

   AND

   b. Any containment challenge indication in Table C1.

Table C1: Containment Challenge Indications

- **CONTAINMENT CLOSURE not established**
- **> 2.9% hydrogen concentration inside containment**
- **UNPLANNED rise in containment pressure**
- **Secondary Containment area radiation monitor reading above the value below:**

AREA	Max Safe Operating Value
RHR Room A	8 x 10^4 mR/hr
RHR Room B	8 x 10^4 mR/hr
RHR HX A Hatch	8 x 10^4 mR/hr
RHR HX B Hatch	8 x 10^4 mR/hr
RCIC Room	8 x 10^4 mR/hr
MSL Rad Monitor	8 x 10^4 mR/hr
SGTS Fltr. Tr.	8 x 10^2 mR/hr

SITE AREA EMERGENCY

CS1 (4 5)

Loss of RCS/RPV inventory affecting core decay heat removal capability.

Emergency Action Level(s): (1 or 2 or 3)

***Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. With CONTAINMENT CLOSURE not established, RPV level < -47.6 in.

OR

2. With CONTAINMENT CLOSURE established, RPV level < -167 in. (TAF)

OR

3. RCS level cannot be monitored for ≥ 30 minutes with a loss of RCS inventory as indicated by any of the following:
   - Containment High Range Radiation Monitor reading > 100 R/hr.
   - Erratic Source Range Monitor indication.
   - Unexplained rise in floor or equipment sump level, Suppression Pool level, vessel make-up rate or observation of leakage or inventory loss.

ALERT

CA1 (4 5)

Loss of RCS/RPV inventory

Emergency Action Level(s): (1 or 2)

***Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. Loss of RCS inventory as indicated by RPV level < -41.6 in. (Level 2)

OR

2. RCS level cannot be monitored for ≥ 15 minutes with a loss of RCS inventory as indicated by an unexplained rise in floor or equipment sump level, Suppression Pool level, vessel make-up rate or observation of leakage or inventory loss.

UNUSUAL EVENT

CU1 (4)

RCS leakage.

Emergency Action Level(s):

***Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. RCS leakage results in the inability to maintain or restore RPV level > +11.4in. (Level 3) for ≥ 15 minutes.

CU2 (5)

UNPLANNED loss of RCS/RPV inventory.

Emergency Action Level(s): (1 or 2)

***Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. UNPLANNED RCS level drop as indicated by either of the following:
   a. RCS water level drop below the RPV flange for ≥ 15 minutes when the RCS level band is established above the RPV flange.
   OR
   b. RCS water level drop below the RPV level band for ≥ 15 minutes when the RCS level band is established below the RPV flange.

OR

2. RCS level cannot be monitored with a loss of RCS inventory as indicated by an unexplained rise in floor or equipment sump level, Suppression Pool level, vessel make-up rate or observation of leakage or inventory loss.

Loss of Decay Heat Removal

Table C2
RCS Reheat Duration Thresholds

RCS	Containment Closure	Duration
Intact	N/A	60 minutes*
Not Intact	Established	20 minutes*
	Not Established	0 minutes*

* If an RCS heat removal system is in operation within this time frame and RCS temperature is reduced, then the EAL is not applicable.

ALERT

CA3 (4 5)

Inability to maintain plant in Cold Shutdown.

Emergency Action Level(s): (1 or 2)

1. An UNPLANNED event results in RCS temperature >200 °F > the specified duration in Table C2.

OR

2. An UNPLANNED event results in RCS pressure rise > 10 psig due to a loss of RCS cooling.

UNUSUAL EVENT

CU3 (4 5)

UNPLANNED loss of decay heat removal capability with irradiated fuel in the RPV.

Emergency Action Level(s): (1 or 2)

***Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. An UNPLANNED event results in RCS temperature exceeding 200°F.

OR

2. Loss of all RCS temperature and RPV level indication for ≥ 15 minutes.

Loss of AC Power

ALERT

CA5 (4 5 D)

Loss of all Offsite and Loss of All Onsite AC power to Div. I & II ESF busses

Emergency Action Level(s):

1. a. Loss of power to all of the following transformers:
   - ESF-11
   - ESF-21
   - ESF-12

   AND

   b. Failure of both Div. I and Div. II Diesel Generators to supply power to emergency busses.

   AND

   c. Failure to restore power to at least one emergency bus within 15 minutes from the time of loss of both offsite and onsite AC power

UNUSUAL EVENT

CU5 (4 5)

AC power capability to Div I and II ESF busses reduced to a single power source for ≥ 15 minutes such that any additional single failure would result in Station Blackout.

Emergency Action Level(s):

***Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. a. AC power capability to Div I and II ESF busses reduced to a single power source for ≥ 15 minutes.
   AND
   b. Any additional single power source failure will result in station blackout.

Loss of DC Power

CU6 (4 5)

Loss of required DC power for ≥ 15 minutes.

Emergency Action Level(s):

***Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. < 105 VDC on required vital DC busses for ≥ 15 minutes.

Inadvertent Criticality

CU7 (4 5)

Inadvertent criticality

Emergency Action Level(s):

1. UNPLANNED sustained positive period observed on nuclear instrumentation.

Loss of Communication

Table C3

Onsite Communications Methods

Plant Radio System
Plant Paging System
Sound Powered Phones
In-plant Telephones

Table C4

Offsite Communications Methods

All telephone lines (commercial & fiber optic)
Satellite telephone
NRC phones (ENS, HPN, MCL, RSCL and PMCL)

CU8 (4 5 D)

Loss of all onsite or offsite communications capabilities

Emergency Action Level(s): (1 or 2)

1. Loss of all onsite communications methods affecting the ability to perform routine operations (See **Table C3**).

OR

2. Loss of all offsite communications methods affecting the ability to perform offsite notifications (See **Table C4**)