GGNS LOT 5/2017 NRC INITIAL LICENSED OPERATOR WRITTEN EXAMINATION

RO EXAM

ANSWER KEY

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

Examination Outline Cross-Reference	Level	RO
295001 Partial or Complete Loss of Forced Core Flow Circulation 2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	Tier #	1
	Group #	1
	K/A #	**295001 – G2.1.7**
	Rating	4.4

Question 1

The plant is operating with control rods at the target pattern and reactor core flow at 70 mlbm/hr.

Which of the following requires immediate entry into the Scram ONEP?

A. Jet pump #5 and #6 indicate a Rams head failure.

B. 'A' Recirc Pump has tripped and APRM indications are fluctuating between 46% and 58% power.

C. ‘B’ Reactor Feedwater pump trips and both Recirc Flow control valves perform a runback

D. ‘A’ Recirc Flow Control valve begins to drift close and is stopped with ‘A’ Recirc loop flow at 23,000 gpm and ‘B’ Recirc loop flow is 28,000, at 45% core flow.

Answer: B
Explanation: With the plant at target pattern and flow only at 70 mlbm/hr, a reduction in core flow by a large amount would cause a high power low flow condition. Per ONEP 05-1-02-III-3. Reduction in Recirculation System Flow Rate, If any of the following conditions exist then immediately place the reactor mode switch to SHUTDOWN, Thermal hydraulic instability symptoms being observed on neutron instrumentation, 6.9.2 states Oscillations of APRM readings with peak-to-peak swings of greater than 10% rated power. ‘A’ is wrong –but plausible due to a Rams head failure would not lower core flow enough to cause entry into an undesired region of the P/F map and ONEP Jet Pump Anomalies, 05-1-02-III-6 does not require an immediate reactor scram. ‘B’ is correct ‘C’ is wrong – but plausible due to a FCV runback would not lower core flow enough to cause entry

into an undesired region of the P/F map

'D' is wrong – but plausible 45% core flow is outside an undesired region of the P/F map. Tech Specs would require entry on flow mis match but not a scram.

Technical References:

ONEP 05-1-02-III-3. Reduction in Recirculation System Flow Rate
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References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-B3300 Objective: 41

Question Source:	Bank #10 Not used	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
295003 Partial or Complete Loss of A.C. Power AA2. Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : AA2.01 Cause of partial or complete loss of A.C. power	**Tier #**	1
	Group #	1
	K/A #	**295003 – AA2.01**
	Rating	3.4

Question 2

The plant is operating at 100% power.

The following electrical buses lost power:

- 12HE
- 13AD
- 15AA
- 18AG
- 19UD

Which of the following describes the reason for the loss of power to these buses?

A. Loss of ST-11

B. Loss of ST-21

C. Loss of Bus 22R

D. Loss of Bus 12R

Answer: A

Explanation:

Per E001 Drawing, ST-11 (Service Transformer) feeds power to Bus 11R which feeds power to the following:

- ESF-11 transformer which feeds bus 15AA
- BOP-13 transformer which feeds bus 18AG
- Bus 12R which feeds BOP 11A transformer which feeds bus 13AD and BOP 11B transformer which feeds bus 12HE and BOP 14 transformer which feeds bus 19UD

A is correct

B is wrong – this is the other incoming service transformer that feeds the other 6.9kv and 4160 v buses. Plausible if student confuses the two service transformers normal feeds.

C is wrong – Bus 22R is a 34.5kv bus that feeds BOP transformer 24 only. Plausible if the student doesn't remember the normal electrical lineup and major 34.5Kv buses.

D is wrong – Bus 12R is a 34.5kv bus that feeds power to BOP transformers 11A and 11B but not bus 18AG and 15AA. Plausible if the student doesn't remember the normal electrical lineup and major 34.5Kv buses.

Technical References:

E0001, Main One Line Diagram
04-1-01-R21-12, Step 3.2
04-1-01-R21-13, Step 3.2
04-1-01-R21-15, Step 3.3
04-1-01-R21-18, Step 3.2

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-R20/21 Obj. 3

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	3
10CFR Part 55 Content:	55.43(b)(4)	

Examination Outline Cross-Reference	Level	RO
295004 Partial or Complete Loss of D.C. Power AK2. Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF D.C. POWER and the following: AK2.01 Battery charger	**Tier #**	1
	Group #	1
	K/A #	**295004 – AK2.01**
	Rating	3.1

Question 3

The plant is operating at rated power.

Power from Div 3 battery charger 1C4 is suddenly lost when the feeder breaker to the 11DC bus trips open.

Which of the following identifies the initial response of the Div 3 DC bus voltage as indicated on control room panel 1H13-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	2015 NRC Q #2
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	2
10CFR Part 55 Content:	55.41(b)(5)	

Examination Outline Cross-Reference	Level	RO
295005 Main Turbine Generator Trip AA1. 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 4

Which of the following describes the status of RPS when a spurious Main Turbine Trip signal is generated and only ‘A’ and ‘B’ Main Turbine Control valves close?

A. No Scram signal present

B. ½ scram signal on Division 1 RPS

C. ½ scram signal on Division 2 RPS

D. Full Scram

Answer: D
Explanation: Turbine Control valve logic is A valve to A RPS, B to B and so on, therefore with A and B valves closed a trip signal should have been generated to RPS A and RPS B causing a full scram. All distractors are plausible if the student confuses this logic with Main Stop valves or MSIVs. A is wrong – a full scram will be generated B is wrong – This would be true if it were the Stop Valves C is wrong – This would be true if it were the MSIV steam lines D is correct
Technical References: GLP-OPS-C7100, GFIG-OPS-C7100 E1173
References to be provided to applicants during exam:

NONE		
Learning Objective: Document learning objective if possible. GLP-OPS-C7100		
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)(7)	

Examination Outline Cross-Reference	Level	RO
295006 SCRAM AK3. Knowledge of the reasons for the following responses as they apply to SCRAM : AK3.04 Reactor water level setpoint setdown: Plant-Specific.	**Tier #**	1
	Group #	1
	K/A #	**295006 – AK3.04**
	Rating	3.1

Question 5

Which of the following, when initiated, assists with overcoming the effects of RPV level shrink and swell after a reactor scram?

A. Lo-Lo Set actuation.

B. Reactor Recirc Pump Auto Transfer to Slow Speed.

C. RPV water level control Setpoint Setdown.

D. Reactor Feedpump Turbine Trip on Level 9.

Answer: C

Explanation:

Per GLP-OPS-C3400 ; page 35 of 57, Following a Reactor scram, the Setpoint Setdown feature automatically operates to overcome the effects of shrink and swell on RPV water level control.

A is wrong – Lo-Lo Set is a SRV actuation the helps control reactor pressure, plausible if the student confuses lo-lo set with setpoint setdown and this is also a parameter that is control post scram like level.

B is wrong – This helps control reactor level be securing the recirc pumps if water level is lowering.

C is correct

D is wrong – This helps prevent water induction into the steam lines.

Technical References:

GLP-OPS-C3400

References to be provided to applicants during exam:

None		
Learning Objective: Document learning objective if possible. GLP-OPS-C3400 Obj. 12		
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)(5)	

Examination Outline Cross-Reference	Level	RO
295018 Partial or Complete Loss of Component Cooling Water AK1. Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : AK1.01 Effects on component/system operations	Tier #	1
	Group #	1
	K/A #	**295018 – AK1.01**
	Rating	3.5

Question 6

The plant is operating at rated thermal power one month away from the next refueling outage.

A transient occurs on the CCW Pumps.

Currently only one (1) CCW pump is running.

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

Which of the following describes the components that still have CCW flowing through them?

A. Reactor Recirculation Pumps and Control Rod Drive Pump coolers

B. Reactor Recirculation Pumps and RWCU Non Regen Heat Exchangers

C. Fuel Pool Cleaning and Cleanup Heat Exchangers and RWCU Non Regen Heat Exchangers

D. RWCU pump coolers and RWCU Heat Exchangers

Answer: A
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. A is correct – Recirc and CRD cannot be isolated from the control room only local manual. B is wrong – RWCU should have been isolated by the ONEP, plausible if the student forgets

that this system is manually isolated per the ONEP.

C is wrong – Fuel pool and RWCU should have already been isolated, plausible if the student forgets that these systems are manually isolated per the ONEP.

D is wrong – RWCU heat exchangers should have already been isolated by the ONEP, RWCU pump coolers should have cooling water, plausible if the student forgets that this system is manually isolated per the ONEP.

Technical References:

GLP-OPS-P4200
05-1-02-V-1, Loss of CCW ONEP

References to be provided to applicants during exam:

None

Learning Objective: Document learning objective if possible.

GLP-OPS-P4200 Obj 10

Question Source:	Bank #638	2015 Audit
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(5)	

Examination Outline Cross-Reference	Level	RO
295019 Partial or Complete Loss of Instrument Air AK3. Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR : AK3.02 Standby air compressor operation.	Tier #	1
	Group #	1
	K/A #	**295019 – AK3.02**
	Rating	3.5

Question 7

The plant is operating at rated power with the following Plant Air Compressor (PAC) configuration:

- PAC ‘A’ in Standby
- PAC ‘B’ shutdown (idle)
- PAC ‘C’ running

A lockout occurs on ST-21.

<u>No</u> <u>operator</u> action has yet been taken to re-energize the buses affected by the lockout.

30 seconds has elapsed since the lockout.

Which of the following describes the status of the Plant Air System compressors?

What operator action is required?

A. No Compressors are running
Manually start PAC ‘A’

B. Standby compressor, PAC ‘A’ auto started
Verify proper operation of PAC ‘A’

C. PAC ‘C’ continues to run.
Manually start PAC ‘A’

D. PAC ‘C’ continues to run and Standby compressor, PAC ‘A’ auto started
Manually shutdown PAC ‘A’

Answer: A
Explanation:

Loss of ST-21 takes away Bus 16AB (Removing power to PAC ‘A’) and Bus 11HD (taking away the MCC 11B31 control power needed to continue operation of the running PAC ‘C’). Thus, we’re left with no PACs running. Without the 11B31 control power, PAC ‘C’ cannot be manually started either. Of course, the Div 2 DG has automatically re-powered Bus 16AB, as well as restored the needed control power for PAC ‘A’ (i.e., MCC 16B42 is restored via LSS re-sequencing). So, operators can manually start PAC ‘A’.

All distracters are plausible because they each represent another potential system response/required operator action should the Applicant fail to recall the described power distribution and impact of a BUV on Bus 16AB.

Technical References:

04-1-01-P51-1, Plant Air SOI
04-1-01-R21-1, Load Shedding & Sequencing System SOI
GLP-OPS-P5100 page 33 of 39

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-P5100, Objective 8

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)(5)	

Examination Outline Cross-Reference	Level	RO
295021 Loss of Shutdown Cooling AA1. Ability to operate and/or monitor the following as they apply to LOSS OF SHUTDOWN COOLING: AA1.05 Reactor recirculation	**Tier #**	1
	Group #	1
	K/A #	**295021 – AA1.05**
	Rating	3.0

Question 8

The plant is in MODE 4 with the following:

- RHR ‘B’ operating in Shutdown Cooling
- Both Reactor Recirc Pumps operating in SLOW speed
- One RWCU Pump operating

RHR Pump ‘B’ trips (motor thermal overload) and cannot be re-started.

Which of the following is the preferred method for operators to monitor reactor coolant temperature?

A. RWCU Regenerative HX Inlet temperature

B. RWCU Non-Regenerative HX Inlet temperature

C. Reactor Recirc Loop ‘A’ or ‘B’ Suction temperature

D. RHR ‘B’ HX Inlet temperature

Answer: C
Explanation: See RHR Shutdown Cooling (SDC) SOI (04-1-01-E12-2), P/L 3.8.16. Regardless of the status RHR SDC (operating or not), if a Reactor Recirc Pump is running the preferred method for monitoring reactor coolant temperature is use the Loop Suction Temperature computer point(s) on PDS. A is wrong - plausibility comes from Section 4.2.2c(19)(c), on page 40, of the SOI.

B is wrong - is plausible to the Applicant who recalls Section 4.2.2c(19)(c) but who cannot remember the HX arrangement in the RWCU flowpath.

C is correct

D is wrong - plausibility comes from P/L 3.8.16.a that directs us to use the RHR HX Inlet temperature if no Reactor Recirc Pump is running.

Technical References:

04-1-01-E12-2, Shutdown Cooling SOI

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-E1200 Obj 14.1

Question Source:	Bank #70,	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2012 NRC Q #49
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	2
10CFR Part 55 Content:	55.43(b)(2)	

Examination Outline Cross-Reference	Level	RO
295023 Refueling Accidents AA2. Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS: AA2.01 Area radiation levels	Tier #	1
	Group #	1
	K/A #	**295023 – AA2.01**
	Rating	3.6

Question 9

A refueling outage is in progress.

Fuel is being moved by both the Refuel and Fuel Handling Platforms.

The following alarm is received on 1H13-P844 panel:

- AUXILIARY BLDG FUEL HDLG AREA RADIATION HIGH

NO other alarms are received.

Which of the following describes the condition that could have caused this alarm?

A. a fuel assembly being inadvertently raised too high at the refueling platform

B. a fuel assembly being inadvertently raised too high at the fuel handling platform

C. drop of a spent fuel assembly from the refueling platform causing a release of gases from the assembly.

D. drop of a spent fuel assembly from the fuel handling platform causing a release of gases from the assembly.

Answer: B
Explanation: Raising an irradiated fuel assembly too high could result in reduced shielding with a subsequent rise in area radiation levels. Answers C and D are wrong – but plausible because if release of gaseous activity from a damaged fuel assembly is the cause of the area radiation alarm, this would be accompanied by gaseous process radiation alarms such as Fuel pool sweep ventilation rad levels and/or

Containment vent radiation.

Answers A and C are wrong – but plausible because the refueling platform is in the containment building.

Technical References: GLP-RF-F1101, Fuel handling/Refueling Platform ARI 04-1-02-1H13-P844-1A-A4, AUXILIARY BLDG FUEL HDLG AREA RADIATION HIGH		
References to be provided to applicants during exam: NONE		
Learning Objective: Document learning objective if possible. GLP-OPS-D1721, Objective 19		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #1051	X
	New	
Question History:	Last NRC Exam	2014 NRC Q #9
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(7, 10 &13))	

Examination Outline Cross-Reference	Level	RO
295024 High Drywell Pressure EA2. Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE: EA2.10 Containment temperature: Mark-III	**Tier #**	1
	Group #	1
	K/A #	**295024 – EA2.10**
	Rating	3.7

Question 10

The plant is operating at 100% power.

An Instrument Air line has ruptured inside the Drywell.

The Reactor scrams on High Drywell Pressure.

Instrument Air has failed to isolate.

Which of the following describes the effect this will have on Containment pressure and Temperature?

A. Containment pressure will rise
Containment temperature will remain the same

B. Containment pressure will rise
Containment temperature will rise

C. Containment pressure will remain the same
Containment temperature will rise

D. Containment pressure will remain the same
Containment temperature will remain the same

Answer: B
Explanation: With the given information the Containment pressure will definitely rise due to drywell pressure being relieved through the drywell vents. At a drywell pressure of 1.23 psig the Plant chilled water system will isolate to the containment,

therefore the containment coolers will no longer have cooling water and containment temperature will rise.

A is wrong – due to containment temperature will rise. Plausible if the student forgets coolers losing cooling water.

B is correct

C is wrong – containment pressure will rise due to flow through the horizontal vents, plausible if the student forgets the connection between the drywell and containment.

D is wrong - containment pressure will rise due to flow through the horizontal vents and containment coolers will no longer have cooling water and containment temperature will rise. Plausible if the student forgets coolers losing cooling water.

Technical References:

GLP-OPS-M4100
GLP-OPS-M4101
GLP-OPS-P7100

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-M4101 Objective 7 & 8

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)(8)	

Examination Outline Cross-Reference	Level	RO
295025 High Reactor Pressure 2.2.39 Knowledge of less than or equal to one hour Technical Specification action statements for systems.	**Tier #**	1
	Group #	1
	K/A #	**295025 – G2.2.39**
	Rating	3.9

Question 11

The plant is operating at 95% power.

A Main Turbine pressure control system malfunction occurs, causing reactor pressure to rise to 1048 psig.

Which of the following is required by Tech Specs and the time limit required?

A. Restore reactor steam dome pressure to less than 1045 psig within 1 hour.

B. Restore reactor steam dome pressure to less than 935 psig within 1 hour.

C. Restore reactor steam dome pressure to less than 1045 psig within 15 minutes.

D. Restore reactor steam dome pressure to less than 935 psig within 15 minutes.

Answer: C
Explanation: With the given information, reactor pressure has exceeded tech spec 3.4.12, "Reactor steam dome pressure shall be ≤ 1045 psig." Condition A states that "Restore reactor steam dome pressure to within limit within 15 minutes." A is wrong – the time limit should be 15 minutes but plausible due to numerous other tech specs are limited to 1 hour. B is wrong – the pressure is incorrect but plausible due to this is the pressure which the Main Turbine pressure set is adjusted to for 100% power operations and the time limit should be 15 minutes but plausible due to numerous other tech specs are limited to 1 hour. C is correct D is wrong - the pressure is incorrect but plausible due to this is the pressure which the Main Turbine pressure set is adjusted to for 100% power operations.

Technical References: Tech Specs 3.4.12		
References to be provided to applicants during exam: NONE		
Learning Objective: Document learning objective if possible. GLP-OPS-TS01		
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)(7)	

Examination Outline Cross-Reference	Level	RO
295026 Suppression Pool High Water Temperature 2.2.42 Ability to recognize system parameters that are entry-level conditions for Technical Specifications.	Tier #	1
	Group #	1
	K/A #	**295026 – G2.2.42**
	Rating	3.9

Question 12

With the plant at rated conditions, which of the following is the lowest suppression pool temperature that requires a Tech Spec entry?

A. 70°F

B. 95°F

C. 105°F

D. 110°F

Answer: B
Explanation: Per Tech Spec 3.6.2.1, Suppression Pool Average Temperature, LCO – Suppression Pool average temperature shall be ≤ 95°F when Thermal power is >1% RTP and no testing that adds heat to the suppression pool is being performed. A is wrong – but plausible, 70°F is an administrative limit on water temperature that is being supplied to the reactor vessel to protect vessel internals. B is correct C is wrong – but plausible, Per Tech Spec 3.6.2.1, Suppression Pool average temperature shall be ≤ 105°F when Thermal power is >1% RTP and testing that adds heat to the suppression pool is being performed. D is wrong – but plausible, Per Tech Spec 3.6.2.1, Suppression Pool average temperature shall be ≤ 110°F when Thermal power is ≤ 1% RTP.
Technical References: Tech Specs 3.6.2.1

References to be provided to applicants during exam: NONE		
Learning Objective: Document learning objective if possible. GLP-OPS-TS01		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
295027 High Containment Temperature (Mark III Containment Only) EK1. 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 #**	1
	Group #	1
	K/A #	**295027 – EK1.02**
	Rating	3.0

Question 13

A Containment steam line break is in progress with the following:

Wide Range level is -10”
Fuel Zone level is oscillating between -100” and -25”
Upset Range level is 5”
Shutdown Range level is 10”
RPV pressure is 0 psig
Drywell temperature (166 ft) = 198°F; (139 ft) = 180°F
CTMT temperature (166 ft) = 235°F; (139 ft) = 210°F

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

A. Wide Range only

B. Wide Range and Fuel Zone, only

C. Upset Range, and Shutdown Range only

D. Wide Range, Upset Range, and Shutdown Range

Answer: A
Explanation: See EP-1 CAUTION 1. The DW and CTMT temperatures in the stem fall in the “POSSIBLE BOILING” region of the RPVST curve (Figure 2); therefore, there are possible boiling concerns. With the indication of level notching this makes the Fuel Zone Range instrument completely unvalid and **NOT usable** per Caution 1.1. This makes answer B wrong 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. This makes answer A correct. The indicated level for Upset Range (5”) is below its limit (159”) AND the stem’s given DW temperature at the 166 ft elevation

(198°F) is above the associated limit (195°F); therefore, Upset Range is **not usable**. This makes answers C and D wrong

The indicated level for Shutdown Range (10”) is below its limit (139”) AND caution 1 says at >66°F at DW El. 166 ft , therefore Shutdown Range is **not useable**. This also makes answers C and D wrong

All distracters are wrong but are plausible based on the Applicant’s need to apply Caution 1 as already described.

Technical References:

EP-1, Caution 1
02-S-01-40, EP Technical Bases

References to be provided to applicants during exam:

EP-1, Caution 1

Learning Objective: Document learning objective if possible.

GLP-OPS-EP01

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #364	X
	New	
Question History:	Last NRC Exam	2015 NRC Q #13
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(5)	

Examination Outline Cross-Reference	Level	RO
295028 High Drywell Temperature EK3. Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE : EK3.04 Increased drywell cooling	**Tier #**	1
	Group #	1
	K/A #	**295028 – EK3.04**
	Rating	3.6

Question 14

What is the bases for operating all drywell cooling when Drywell Temperature exceeds the EP-3 entry condition?

A. To reduce and maintain Drywell temperature less than the Drywell temperature Tech Spec LCO where no further operator action need be taken.

B. To maintain Drywell temperature less than the Drywell design temperature limit where an Emergency Depressurization is required.

C. To maintain Drywell temperature less than the ADS qualification temperatures where a Reactor scram is required.

D. To reduce and maintain Drywell temperature less than the Drywell temperature Tech Spec LCO where the ADS/SRVs are declared INOP.

Answer: A
Explanation: The Drywell temp. EP-3 entry condition is 135°F, 02-S-01-40, EP Technical Bases, states “As long as drywell temperature remains below 135°F, the higher of the drywell temperature LCO and the maximum normal operating drywell temp., no further operator action need be taken in the DW temp. branch. A is correct B is wrong – by starting all available drywell cooling could be to prevent exceeding the design temperature limit (330°F), only a scram is required not an ED. C is wrong – same as B above by maxing drywell cooling you should maintain below the ADS qualification temperature (355°F), an ED is required if this setpoint cannot be maintained. Not just a reactor scram. D is wrong – the ADS qualification temp is 355°F not 135°F

All distractors are plausible due to all are setpoints listed are in the Drywell temperature leg of EP-3.		
Technical References: 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-EP003		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
295030 Low Suppression Pool Water Level EA1. Ability to operate and/or monitor the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: EA1.04 Suppression pool make-up system: Mark-III	Tier #	1
	Group #	1
	K/A #	**295030 – EZ1.04**
	Rating	4.0

Question 15

A LOCA has occurred.

- Reactor water level is being maintained -30” to +50”
- Lowest indicated level was -130”
- Drywell pressure is 1.5 psig

Which of the following will cause the Suppression Pool Makeup System to automatically initiate?

A. 29 minutes from reaching 1.39 psig Drywell pressure

B. Suppression pool level at EP-3 entry condition.

C. Reactor water level lowering to below -41.6” again.

D. Suppression pool level at 17.5 ft.

Answer: D
Explanation: A is wrong – but plausible because the SPMU system will auto initiate 29 minutes from reaching 1.23 psig drywell pressure not 1.39 psig B is wrong – but plausible because EP-3 entry is a required from memory setpoint of 18.34 ft. . C is wrong – but plausible because until reset the -41.6” signal remains sealed in, therefore it is not required to reach this level setpoint again. D is correct.
Technical References:

04-1-01-E30-1 GLP-OPS-E30		
References to be provided to applicants during exam: NONE		
Learning Objective: Document learning objective if possible. GLP-OPS-E300, Obj 7.1		
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)(7)	

Examination Outline Cross-Reference	Level	RO
295031 Reactor Low Water Level EK2. Knowledge of the interrelations between REACTOR LOW WATER LEVEL and the following: EK2.13 ARI/RPT/ATWS: Plant-Specific	**Tier #**	1
	Group #	1
	K/A #	**295031 – EK2.13**
	Rating	4.1

Question 16

The reactor was operating at rated power when the reactor scrammed due to instrument failures.

Which of the following indications confirm that reactor water level is below +11.4 inches and above -41.6 inches?

A. On P680 panel - Reactor Recirc pumps A and B running in slow speed.
On P601 panel - ADS Confirmatory Level annunciator.

B. On P680 panel - Reactor Recirc pumps A and B tripped to OFF.
On P601 panel - ADS Confirmatory Level annunciator.

C. On P680 panel - Reactor Recirc pumps A and B tripped to OFF.
On P601 panel – ADS Timers have initiated.

D. On P680 panel - Reactor Recirc pumps A and B running in slow speed.
On P601 panel – ADS Timers have initiated.

Answer: A
Explanation: On P601 panel the ADS confirmatory level is <+11.4” reactor water level. Recirc pumps will also shift to slow speed at this level on cavitation interlocks. At -41.6”, the Recirc pumps will trip off due to ATWS/ARI initiation. This means that in order to confirm level between11.4” and -41.6” Recirc pumps must be in slow speed and the ADS Confirmatory Level annunciator must be in alarm. ADS timers will initiate only when an ECCS/LSS LOCA signal is received (1.39 psig in DW or -150.3” reactor level 1). A is correct B is wrong – with the recirc pumps being tripped off then level must have reached -41.6”

which is the ATWS/ARI setpoint, plausible if the student can't differ between a downshift and a trip of recirc pumps.

C is wrong - with the recirc pumps being tripped off then level must have reached -41.6" which is the ATWS/ARI setpoint and the ADS timers will initiate when only when an ECCS/LSS LOCA signal is received (1.39 psig in DW or -150.3" reactor level 1), plausible if the student can't differ between a downshift and a trip of recirc pumps or differ between ADS conformation level and timers running setpoints

D is wrong - the ADS timers will initiate when only when an ECCS/LSS LOCA signal is received (1.39 psig in DW or -150.3" reactor level 1), plausible if the student can't differ between ADS conformation level and timers running setpoints

Technical References:

04-1-02-1H13-P680-2A-C15
04-1-02-1H13-P601-18A-A2
04-1-02-1H13-P601-18A-C2

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-E2202, Objective 15.0

Question Source:	Bank #728	2015 AUDIT EXAM
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2013 NRC (Q17)
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown EA2. Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : EA2.06 Reactor pressure	**Tier #**	1
	Group #	1
	K/A #	**295037 – EA2.06**
	Rating	4.0

Question 17

An ATWS has occurred.

Reactor water level is being maintained by the ATC at -70” to -130” Wide Range using Feedwater.

Reactor pressure is being maintained 800 psig to 1060 psig using Main Turbine Bypass valves.

The MSIVs close.

Which of the following describes further control of reactor pressure to maintain reactor water level band?

A. Pressure band of 800 psig to 1060 psig using SRVs

B. Pressure band of 450 psig to 600 psig using SRVs

C. Pressure band of 800 psig to 1060 psig using Main Turbine Bypass valves.

D. Pressure band of 450 psig to 600 psig using Main Turbine Bypass valves.

Answer: B
Explanation: When the MSIVs close the Main Turbine Bypass valves can no longer be used, the SRVs must be used, a pressure band of 450 to 600 psig must be called for to be able to maintain reactor water level due to loss of steam to the feedpumps and must feed using the condensate booster pumps. A is wrong – pressure band is too high to allow feedwater to continue to feed the RPV

B is correct

C is wrong - pressure band is too high to allow feedwater to continue to feed the RPV and bypass valves are no longer available.

D is wrong - bypass valves are no longer available.

Technical References:

02-S-01-43, Transient Mitigation Strategy

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-EP02

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)(5)	

Examination Outline Cross-Reference	Level	RO
295038 High Off-Site Release Rate 2.1.27 Knowledge of system purpose and/or function.	Tier #	1
	Group #	1
	K/A #	**295038 – G2.1.27**
	Rating	3.9

Question 18

An event causing high offsite release rates from the Turbine Building Ventilation exhaust stack is in progress.

Which of the following describes operation of the Turbine Building Ventilation Accident Range Monitor (AXM) as the Turbine Building Ventilation exhaust effluent radiation level rises?

A. The AXM will not automatically start but can only be manually started coincident with any radiation alarm from SPING.

B. The AXM should automatically begin operating when the Turbine Building Ventilation Noble Gas (GE) Monitor reaches its HIGH alarm setpoint.

C. The AXM will not automatically start but can only be manually started if the iodine subsystem of SPING is in operation.

D. The AXM should automatically begin operating when the Turbine Building Ventilation SPING Mid Range Noble Gas Monitor reaches its alarm setpoint.

Answer: D
Explanation: The TBV AXM can be started by three normal means: if the TBV SPING Mid Range Noble Gas Monitor reaches its alarm setpoint, as stated in correct answer D, or if the SPING is placed in FLUSH mode, or manually from the control terminal. The SPING and GE monitors are separate subsystems. Distracters B ans C are plausible since the TBV Noble Gas (GE) Monitors monitor the same ductwork as does the SPING and are required by TS/TRM and provide high effluent radiation alarms on 1H13-P601. Distracter A is plausible since the AXM can be started manually, but it is wrong because it states that it will not automatically start.

Technical References: 08-S-04-220 step 3.3 04-1-01-D17-1 step 4.8.1		
References to be provided to applicants during exam: NONE		
Learning Objective: Document learning objective if possible. GLP-OPS-D1721 Obj. 16		
Question Source:	Bank # 874	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2007 Q #19
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.43(b)(11)	

Examination Outline Cross-Reference	Level	RO
600000 Plant Fire On Site AK3 Knowledge of the reasons for the following responses as they apply to PLANT FIRE ON SITE: AK3.04 Actions contained in the abnormal procedure for plant fire on site	Tier #	1
	Group #	1
	K/A #	**600000 AK3.04**
	Rating	2.8

Question 19

A fire has started in Div 2 Diesel Generator Room.

The fire brigade is responding.

Per 05-S-02-V-1, Response To Fires, which of the following DG Room Outside Air Fans is/are required to be started?

A. Div 3, only

B. Div 1, only

C. Div 2 and Div 3, only

D. Div 1 and Div 3, only

Answer: D
Explanation: Per 05-S-02-V-1, Response To Fires step 3.4.1, 3.4.1 **<u>IF</u>** the fire is reported to be in a Diesel Generator Bay(s), **<u>THEN</u> START** the ventilation system in the unaffected Diesel Generator Bay(s). A and B are wrong – the above step says to start the ventilation in the unaffected DG bays. Plausible if the student doesn't remember that both fans should be started. C is wrong – Plausible if the student forgets that you do not start the ventilation in the room with the fire, i.e. Div 2 DG. D is correct
Technical References: 05-S-02-V-1, Response To Fires

References to be provided to applicants during exam: NONE		
Learning Objective: Document learning objective if possible. GLP-OPS-PROC		
Question Source:	Bank #63	2015 Audit
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
700000 Generator Voltage and Electric Grid Disturbances AA2. Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: AA2.04 VARs outside capability curve	Tier #	1
	Group #	1
	K/A #	**700000 AA2.04**
	Rating	3.6

Question 20

Which of the following is a main generator configuration that poses an operational **RISK** to either grid stability and/or protection of the generator itself?

A. Carrying 800 MWe while operating at a 0.9 pf.

B. Initial sync with minimum MWe and with the generator carrying 0 MVAR.

C. Carrying 1000 MWe and +50 MVARs with a steady-state hydrogen gas pressure of 72 psig.

D. Carrying 600 MWe and -100 MVARs with a single hydrogen cooler section out-of-service.

Answer: A
Explanation: The figure is the GGNS generator capacity curve found in its Integrated Operating Procedure 03-1-01-2, Power Operation. The suggested 800 MWe at 0.9 pf operating point clearly shows an MVAR load in excess of +297.5 MVARs. This violates the IOI-2 Precaution/Limitation 2.4 “emergency operation” limit. In fact, the plotted point is actually about 380 MVARs which exceeds +272 MVARs, the value at which the generator’s “Reverse Power Relay” may not recognize a reverse power condition (as described in this same P/L). B is wrong. This would be normal indications during a Generator startup so no RISK is involved. C is wrong. This is the more difficult distracter because GGNS’s generator capacity does not provide the typical “spider curve” family of curves (i.e., several curves for differing hydrogen gas pressures). This is because GGNS always attempts to maintain 75 psig (see the operator actions for alarm response instruction 04-1-02-1H22-P148-2A-B4, “H2 PRESSURE LOW”). However, that ARI shows that not until a pressure of 69 psig would we consider unloading the generator for a possible generator shutdown.

D is wrong. This choices suggests a MWe-versus-MVAR operating point that yields a total reactive load (MVA) that is still well below the “90% of rated MVA” that we can operate at with a single hydrogen cooler section out-of-service. Plausibility of choice is based on the Applicant being able to determine that this operating point must be well within the limits for a single cooler OOS.		
Technical References: 03-1-01-2, Power Operations 04-1-02-1H22-P148-2A-B4, H2 PRESSURE LOW alarm response instruction		
References to be provided to applicants during exam: 03-1-01-2, Power Operations, REACTIVE CAPABILITY CURVE, FIGURE 2		
Learning Objective: Document learning objective if possible. GLP-OPS-N4151 Obj 14		
Question Source:	Bank #363	2013 AUDIT X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2011 NRC Q #57
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
295007 High Reactor Pressure AK1. Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE : AK1.01 Pump shutoff head	Tier #	1
	Group #	2
	K/A #	**295007 AK1.01**
	Rating	2.9

Question 21

An ATWS is in progress.

The CRS has lowered the pressure band to 450 psig to 600 psig.

Which of the following describes the system(s) that can be used to maintain reactor water level?

A. HPCS and/or LPCS

B. Condensate pumps only

C. Condensate and Condensate Booster pumps

D. RHR A and/or B through the Shutdown cooling injection E12-F053A(B).

Answer: C
Explanation: During an ATWS with a lower pressure band it is recommend to maintain level with a system that injects outside the shroud. A is wrong – EP-2A instructs to override HPCS and LPCS, even though each could inject due to shut off head being over 1100 psig for HPCS and approximately 500 psig for LPCS. B is wrong – Even though these are listed as preferred injection system during an ATWS, Condensate pumps alone DO NOT have enough discharge head to inject at 450 psig reactor pressure. C is correct – Condensate pumps along with Booster pumps produce approximately 650 psig of head. D is wrong – Even though these are listed as preferred injection systems during an ATWS,

RHR pumps only produce about 230 psig discharge head.		
Technical References: SFD-1053 GLP-OPS-E1200, N1900, N2100 041-1-01-E12-1, N21-1 02-S-01-43, Transient Mitigation Strategy		
References to be provided to applicants during exam: NONE		
Learning Objective: Document learning objective if possible. GLP-OPS-EP02		
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)(7)	

Examination Outline Cross-Reference	Level	RO
295008 High Reactor Water Level 2.1.20 Ability to interpret and execute procedure steps.	**Tier #**	1
	Group #	2
	K/A #	**295008 G2.1.20**
	Rating	4.6

Question 22

A feedwater control malfunction has occurred, causing reactor water level to rise to offscale high on Wide and Narrow Ranges.

A reactor scram occurred and both feed pumps tripped.

Which of the following is required **in order** to restart a feed pump for reactor level control?

A. (1) Restore reactor water level below 58”
(2) Manually Reset Level 9 trip
(3) Reset Feed pump Turbine trip
(4) Restart Feed pump

B. (1) Reset Feed pump Turbine trip
(2) Restore reactor water level below 58”
(3) Manually Reset Level 9 trip
(4) Restart Feed pump

C. (1) Restore reactor water level below 58”
(2) Reset Feed pump Turbine trip
(3) Restart Feed pump

D. (1) Reset Feed pump Turbine trip
(2) Restore reactor water level below 58”
(3) Restart Feed pump

Answer: C
Explanation: If a RFPT trips on level 9 the first requirement is to restore water level below 58”. Second the level 9 trip will auto reset when all three narrow range trip units reduce to <58”, therefore no reset is required, only restoring below 58”. Then the RFPT trip logic is required to be reset prior to starting a feed pump. A and B are wrong – Level 9 trip no longer needs to be manually reset. Removed manual reset when digital feedwater system control was incorporated.

C is correct

D is wrong – the feed pump cannot be reset until reactor water level is restored below the level 9 trip.

Technical References:

04-1-01-N21-1, Attachment VI, Re-establish Feed Water Flow
04-1-02-1H13-P680

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-N2100

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)(10)	

Examination Outline Cross-Reference	Level	RO
295015 Incomplete SCRAM AA2. Ability to determine and/or interpret the following as they apply to INCOMPLETE SCRAM : AA2.01 Reactor power	Tier #	1
	Group #	2
	K/A #	**295015 AA2.01**
	Rating	4.1

Question 23

An ATWS has occurred.

Standby Liquid Control was initiated.

All APRMs indicate 2% power and falling.

Which of the following can be used to accurately determine current reactor power?

A. IRMs with detectors withdrawn

B. SRMs with detectors withdrawn

C. IRMs with detectors inserted

D. SRMs with detectors inserted

Answer: C
Explanation: During an ATWS determining reactor power and monitoring reactor power is required. After the APRMs are offscale low the IRMs are the next neutron monitoring system to monitor reactor power. For an accurate determination of power the detectors must be inserted. A is wrong – IRMs must be inserted for an accurate indication of power. B is wrong – at 2% reactor power the SRMs would read high and not be a good indication of power until lower on the IRM scale. And the detectors should be inserted C is correct D is wrong - at 2% reactor power the SRMs would read high and not be a good indication of power until lower on the IRM scale
Technical References:

05-1-02-I-1, Reactor Scram ONEP		
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	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(6)	

Examination Outline Cross-Reference	Level	RO
295020 Inadvertent Containment Isolation AK3. Knowledge of the reasons for the following responses as they apply to INADVERTENT CONTAINMENT ISOLATION: AK3.01 Reactor SCRAM	Tier #	1
	Group #	2
	K/A #	**295020 AK3.01**
	Rating	3.8

Question 24

The plant is at rated conditions.

ESF 11 transformer trips.

All Automatic actions have occurred.

Without any operator action which of the following describes what would be first to cause a reactor scram?

A. Manual Scram from Drywell pressure rising due to loss of Drywell Chilled water on loss of power.

B. Manual Scram from Containment temperature due to a loss of Plant Chilled water on loss of power.

C. Inboard MSIV closure due to loss of Instrument Air to Drywell from loss of control power to P53-F007, INSTR AIR SPLY HDR TO DRWL.

D. Manual Scram for more than one Control Rod Drift due to loss of Instrument Air to Containment from loss of solenoid power to P53-F001, INSTR AIR SPLY HDR TO CTMT.

Answer: D
Explanation: When ESF 11 trips a loss of ESF bus 15AA occurs, all auto actions are the D/G restoring power to 15AA. Without any operator action the P53-F001, INSTR AIR SPLY HDR TO CTMT will close on loss of solenoid power. If not reopened Scram air header will fall causing multiple rod drifts and/or scrams. Per 05-1-02-IV-1, Control Rod Drive Malfunctions more than one control rod drifting place the mode switch to Shutdown. A is wrong – Drywell chilled water pumps and chillers are not powered from 15AA, however the Division 1 Drywell isolation valves are powered from 15AA. However, the valves are AC powered and will fail as is which is open. A loss of the ‘A’ Drywell coolers will occur. This will cause a rise in drywell

parameters but will take a very long time to occur.

B is wrong – Plant chilled water isolation to containment will isolate close. But, due to the design and size of the Containment, it will take a very long time before an action is required.

C is wrong – The Inboard MSIVs will lose air due to closure of the P53-F001 and close but it will take approximately 10 to 15 minutes due to air accumulators on the inboard MSIVs. The P53-F007 is a Division 2 power MOV that will fail as is OPEN.

D is correct – Rod drifts will occur approximately 2 to 3 minutes after the loss of air.

Technical References:

ONEP 05-1-02-IV-1, Control Rod Drive Malfunctions
ONEP 05-1-02-V-9, Loss of Instrument Air.

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

Examination Outline Cross-Reference	Level	RO
295022 Loss of CRD Pumps AK2. Knowledge of the interrelations between LOSS OF CRD PUMPS and the following: AK2.03 Accumulator pressures	Tier #	1
	Group #	2
	K/A #	**295022 K2.03**
	Rating	3.4

Question 25

The plant is operating at rated power late in core life. No control rods are fully inserted.

CRD pump 'A' is operating and CRD pump 'B' is tagged out.

Time 0815 - 'A' CRD pump trips.

Operations and Electrical Maintenance are trying to restore pump

Time 0820 - First HCU TROUBLE alarm occurs on low pressure

Time 0825 - Second HCU TROUBLE alarm occurs on low pressure

Time 0835 - Five more HCUs go into Trouble on low pressure

Per Tech Specs and 05-1-02-IV-1, Control Rod/Drive Malfunctions ONEP, which of the following describes the latest time at which the Mode Switch should be placed to SHUTDOWN?

A. 0835.

B. 0840.

C. 0845.

D. 0855.

Answer: C
Explanation: Per Tech spec 3.1.5 and 05-1-02-IV-1, Control Rod/Drive Malfunctions ONEP step 3.1.2, "When two or more control rod accumulators are declared INOP with RPV steam dome pressure ≥ 600 psig, then perform the following:

If CRD charging water header pressure cannot be restored to ≥ 1520 psig within 20 minutes and any inoperable control rod accumulator with a withdrawn control rod, then place the reactor mode switch to SHUTDOWN. C is the correct answer. 20 minutes from the second accumulator fault.		
Technical References: Tech spec 3.1.5 and 05-1-02-IV-1, Control Rod/Drive Malfunctions ONEP		
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	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(5)	

Examination Outline Cross-Reference	Level	RO
295029 High Suppression Pool Water Level 2.4.2 Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.	**Tier #**	1
	Group #	2
	K/A #	**295029 G2.4.2**
	Rating	4.5

Question 26

A high Suppression Pool level has reached EP-3 entry condition.

What is the setpoint for the entry condition and what else uses the same setpoint?

A. 18.81 ft.
Tech Specs High Suppression Pool level limit

B. 23.75 ft.
Tech Specs High Suppression Pool level limit

C. 18.81 ft.
EP-3 for Reactor shutdown

D. 23.75 ft.
EP-3 for Reactor shutdown

Answer: A
Explanation: EP-3 entry condition for Suppression pool level is 18.81 ft which is also Tech Spec entry for 3.6.2.2 LCO. A is correct B is wrong – plausible due to 23.75 is discussed in EP-3 to be the high level if SPMU has initiated. However, 18.81 is the Tech Spec high level limit. C is wrong – EP-3 requires a reactor shutdown by entering EP-2 is Before SP level rises to 24.6 ft. D is wrong - plausible due to 23.75 is discussed in EP-3 to be the high level if SPMU has initiated. However, 18.81 is the Tech Spec high level limit and EP-3 requires a reactor shutdown by entering EP-2 is Before SP level rises to 24.6 ft.
Technical References:

Tech spec 3.6.2.2 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-EP03		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(9)	

Examination Outline Cross-Reference	Level	RO
295034 Secondary Containment Ventilation High Radiation EA1. Ability to operate and/or monitor the following as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION : EA1.02 Process radiation monitoring system	Tier #	1
	Group #	2
	K/A #	**295034 EA1.02**
	Rating	3.9

Question 27

Which of the following radiation monitors will initiate automatic actions that will affect Secondary Containment ventilation systems?

A. Main Steam Line

B. Fuel Handling Area

C. Standby Gas Treatment

D. Containment Vent

Answer: B
Explanation: Fuel Handling Area Exhaust Radiation monitors detect secondary containment vent radiation levels and initiate an isolation of Secondary Containment Ventilation and initiate of Standby Gas Treatment. Main Steam Line Radiation Monitors go to trip Mechanical Vacuum Pumps when operating and isolate the B33-F019 and F020, Standby Gas Treatment is normally not in operation and is for indication only no actions result Containment Vent Radiation monitor is for isolating Containment Vent Exhaust and has no interaction with Standby Gas or Secondary Containment Ventilation.
Technical References: **Technical References:** 04-1-02-1H13-P601 19A-B9, C9 GFIG-OPS-D1721

References to be provided to applicants during exam: NONE		
Learning Objective: Document learning objective if possible. GLP-OPS-T4200, objective 7 GLP-OPS-T4800, objective 7.4		
Question Source:	Bank # 591	2010 Audit Q #2
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(11)	

Examination Outline Cross-Reference	Level	RO
203000 RHR/LPCI: Injection Mode (Plant Specific) K2. Knowledge of electrical power supplies to the following: K2.01 Pumps	Tier #	2
	Group #	1
	K/A #	**203000 K2.01**
	Rating	3.5

Question 28

What is the power supply for RHR pump ‘C’?

A. 14AE

B. 15AA

C. 16AB

D. 17AC

Answer: C
Explanation: The power supply for the RHR ‘C’ pump is bus 16AB A is wrong – but plausible if candidate confuses BOP power with ESF power. 14AE is a BOP 4.16kv bus B is wrong – 15AA powers RHR A and LPCS pumps C is correct D is wrong – 17AC powers HPCS pump.
Technical References: 04-1-01-R21-16
References to be provided to applicants during exam: NONE

Learning Objective: Document learning objective if possible. GLP-OPS-E1200,		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
203000 RHR/LPCI: Injection Mode (Plant Specific) A2. Ability to (a) predict the impacts of the following on the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.09 Inadequate system flow	**Tier #**	2
	Group #	1
	K/A #	**203000 A2.09**
	Rating	3.3

Question 29

Surveillance 06-OP-1E12-Q-0023, LPCI/RHR SUBSYSTEM A QUARTERLY FUNCTIONAL TEST, is being performed.

The RO starts RHR A pump, E12-F064A, RHR A MIN FLOW VALVE auto closes for no apparent reason and remains closed (will not open).

No other valves were manipulated by the RO.

Which of the following describes the impact on RHR pump ‘A’?

Which mode of RHR should be declared INOP?

A. Pump will overheat without minimum flow path
LPCI and Containment Spray

B. Pump will trip on low flow
Suppression Pool Cooling and Shutdown cooling

C. Pump will overheat without minimum flow path
Suppression Pool Cooling and Shutdown cooling

D. Pump will trip on low flow
LPCI and Containment Spray

Answer: A
Explanation:

Any pump without a min flow path will overheat

The RHR pumps and all ECCS pump do not have a low flow trip.

Per precaution and limitation 3.2.5 “If the minimum flow valves on RHR A, B, OR C will not perform their intended function, declare the associated RHR loop inoperable. (For LPCI and CTMT Spray modes only)

A is correct.

Technical References:

04-1-01-E12-1

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-E1200,

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)(7)	

Examination Outline Cross-Reference	Level	RO
205000 Shutdown Cooling System (RHR Shutdown Cooling Mode) K3. Knowledge of the effect that a loss or malfunction of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) will have on following: K3.04 Recirculation loop temperatures	Tier #	2
	Group #	1
	K/A #	**205000 K3.04**
	Rating	3.7

Question 30

The plant has entered mode 4.

- Time after shutdown is 48 hours
- Current RPV coolant temperature is 145°F
- Current RPV water level is 90” shutdown range
- Both Reactor Recirc pumps are running on Slow speed.
- ‘B’ RHR system is in Shutdown Cooling.

RHR pump ‘B’ trips on overcurrent.

Which of the following indicates the RPV coolant temperature indication that should be monitored?

AND

What is the time to 200°F?

A. ‘A’ Recirc loop temperature only
24 minutes

B. ‘A’ and/or ‘B’ Recirc loop temperatures
24 minutes

C. ‘A’ Recirc loop temperature only
36 minutes

D. ‘A’ and/or ‘B’ Recirc loop temperatures
36 minutes

Answer: B

Explanation:

Even though Shutdown cooling gets is suction from ‘B’ Recirc suction loop only, with both recirc loops are still running both suction temperature indications are accurate.

Using Figure 8 with 2 days shutdown and using the 150°F line, cause on day 2 of the outage no fuel moves have occurred and with current temp at 145°F conservative action is to move up to the 150°F line. Figure 8 is used due to level being at 90”. The bottom of the Main Steam lines are at 101”, at 90” is about 11” below the main steam lines.

A is wrong – Just because the RHR B pump tripped this doesn’t remove the recirc suction loop temperature indicator from being used as long as the ‘B’ recirc pump is still running

B is correct

C is wrong - Just because the RHR B pump tripped this doesn’t remove the recirc suction loop temperature indicator from being used as long as the ‘B’ recirc pump is operating and operating and 36 minutes is at 120°F line

D is wrong - 36 minutes is at 120°F line

Technical References:

04-1-01-E12-2
05-1-02-III-1, Inadequate Decay Heat Removal ONEP

References to be provided to applicants during exam:

05-1-02-III-1, Inadequate Decay Heat Removal ONEP, Attachment 1 only (all graphs)

Learning Objective: Document learning objective if possible.

GLP-OPS-E1200,

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)(3)	

Examination Outline Cross-Reference	Level	RO
209001 Low Pressure Core Spray System K6. Knowledge of the effect that a loss or malfunction of the following will have on the LOW PRESSURE CORE SPRAY SYSTEM : K6.01 A.C. power	Tier #	2
	Group #	1
	K/A #	**209001 K6.01**
	Rating	3.4

Question 31

An ATWS has occurred.

Reactor Pressure is currently 860 psig and slowly rising. Pressure band is 800-1060 psig

All ECCS systems have been overridden per EP-2A.

2 minutes later, 15AA experiences a loss of power; Div 1 Diesel Generator starts and assumes the load of 15AA.

(1) What is the status of Low Pressure Core Spray system?

(2) What is the required action (if any)?

A. (1) LPCS pump running, Injection valve open
   (2) Stop LPCS pump and close Injection valve

B. (1) LPCS pump stopped, Injection valve closed
   (2) Place handswitch for injection valve to CLOSE and Pump to STOP

C. (1) LPCS pump stopped, Injection valve open
   (2) Close Injection valve only

D. (1) LPCS pump running, Injection valve closed
   (2) Place handswitch for injection valve to CLOSE and Pump to STOP

Answer: D
Explanation: During an ATWS all Low Pressure ECCS system pumps and injection valves are overridden. When a loss of power occurs, the override for the pump and the injection valve is lost. The RO must re-override the pump and injection valve.

A is wrong – yes the pump will be running but pressure is too high for the injection valve to open, must be <476 psig reactor pressure. Injection valve will stay closed. Also (2) part is wrong cause injection valve does not have to be closed

B is wrong – LPCS pump will restart due to initiation signal still present. (2) part is correct

C is wrong - LPCS pump will restart due to initiation signal still present and pressure is too high for the injection valve to open, must be <476 psig reactor pressure. (2) part is wrong

D is correct. – Even though the injection valve remains closed the override is lost and the valve Handswitch must be placed in the CLOSE position to regain the override.

Technical References:

04-1-01-E21-1
GLP-OPS-E2100
02-S-01-43

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-E2100, Obj 12.2

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #649	2013 Audit
	New	
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(8)	

Examination Outline Cross-Reference	Level	RO
209002 High Pressure Core Spray System (HPCS) K5. Knowledge of the operational implications of the following concepts as they apply to HIGH PRESSURE CORE SPRAY SYSTEM (HPCS): K5.02 Heat removal (transfer) mechanism: BWR-5,6	Tier #	2
	Group #	1
	K/A #	**209002 K5.02**
	Rating	2.6

Question 32

Which of the following assures heat removal from the reactor is sufficient to prevent failure of the fuel clad when using High Pressure Core Spray system?

A. Flow rate of 550 gpm and Reactor Water level is -220"

B. Flow rate of 7115 gpm and Reactor Water level is -215"

C. Flow rate of 7400 gpm and Reactor Water level is -225"

D. Flow rate of 5000 gpm and Reactor Water level is -205"

Answer: B
Explanation: Adequate core cooling is defined as HPCS OR LPCS flow above 7000 gpm and Reactor water level above -217" (spray cooling) A is wrong – flow rate and level is below min required. Plausible due to 550 gpm is the design flow for HPCS at high pressure. B is correct C is wrong – flow rate is good however, water level is too low. Plausible due to 7400 is rated flow for RHR pumps. D is wrong – Flow rate is too low, level is not high enough for Minimum Zero Injection RPV Water Level of -204"
Technical References: 04-1-01-E22-1 GLP-OPS-E2200 02-S-01-43

References to be provided to applicants during exam: NONE		
Learning Objective: Document learning objective if possible. GLP-OPS-EP01, Obj 4a		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(8)	

Examination Outline Cross-Reference	Level	RO
211000 Standby Liquid Control System 2.4.6 Knowledge of EOP mitigation strategies.	**Tier #**	2
	Group #	1
	K/A #	**211000 G2.4.6**
	Rating	3.7

Question 33

An ATWS has occurred.

Standby Liquid Control (SLC) has been initiated.

Which of the following describes when the SLC pump should be secured?

A. First pump is stopped when SLC tank level indicates 2000 gallons
Second pump is stopped when tank level indicates 0 gallons

B. Both pumps are stopped when all control rods are inserted to position 04 or beyond or Cold Shutdown Boron Weight is reached

C. Both pumps are stopped only when SLC tank level indicates 0 gallons

D. Both pumps are stopped when SLC tank level indicates 2000 gallons

Answer: A
Explanation: Per EP-2A, step Q-5 “WHEN SLC tank level drops to 2000 gal”, step Q-6 “Are both SLC pumps running?”, step Q-7 “Trip one SLC pump.” Step Q-8 “WHEN SLC tank level drops to 0 gal.” Step Q-9 “Trip the running SLC pump.” A is correct. B is wrong – per EP-2A states in step 2 “IF All control rods are inserted to or beyond position 02. THEN Terminate boron injection.” C and D are wrong – see description above.
Technical References: 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-EP02		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
212000 Reactor Protection System K1. Knowledge of the physical connections and/or cause effect relationships between REACTOR PROTECTION SYSTEM and the following: K1.03 Recirculation system	Tier #	2
	Group #	1
	K/A #	**212000 K1.03**
	Rating	3.7

Question 34

The plant is operating at rated conditions with no APRMs bypassed.

Recirculation Flow elbow transmitters B33-N014A and B33-N024A both fail downscale.

Which of the following action will directly occur to RC&IS and/or RPS?

A. A half scram on RPS 'A' only

B. A half scram on RPS 'B' only

C. A rod block and no RPS actuation

D. A rod block and a full reactor scram

Answer: C
Explanation: One APRM channel can give a rod block. Two of the four APRM channel tripping is required to cause a full reactor scram. The only method to get a half scram is to lose electrical power to the respective APRM Voter. The B33-N014A (RR Loop A flow) and B33-N024A (RR Loop B flow) input into APRM Channel 1. When B33-N014A and N024A fail downscale, then the flow input for the STP trip is 0. The STP rod block is calculated by 0.64Wd + 56.8 and the trip is 0.64Wd + 59.8. Wd is the RR loop flow signal. With Wd at "0", the rod block and scram setpoint for STP on Channel 1 is 54.1% and 59.1% respectively. Reactor power is 100% and therefore a STP trip will occur on Channel 1, a rod block and only one vote will be sent to the APRM voter system, therefore no RPS actuation will occur. 'A' is wrong. The only method to get a half scram through APRMs is by UPS power loss to the channel chassis which de-energizes the respective 'voter'. 'B' is wrong. The only method to get a half scram through APRMs is by UPS power loss to the channel chassis which de-energizes the respective 'voter' 'C' is correct. A rod block will occur due to the flow elements going downscale and being below the setpoint, a vote is sent to the APRM RPS scram signal but it requires 2 votes to cause a full scram. 'D' is wrong . A rod block will occur due to the flow elements going downscale and being below the

setpoint, a vote is sent to the APRM RPS scram signal but it requires 2 votes to cause a full scram.
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Technical References:

GLP-OPS-C51-4

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-C7100, Objective 11.4, 23, GLP-OPS-B3300, Objective 36.12
GLP-OPS-C51-4

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	2012 Audit #2 Q #27
	New	
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(7) & (10)	

Examination Outline Cross-Reference	Level	RO
212000 Reactor Protection System A1. Ability to predict and/or monitor changes in parameters associated with operating the REACTOR PROTECTION SYSTEM controls including: A1.01 RPS motor-generator output voltage	**Tier #**	2
	Group #	1
	K/A #	**212000 A1.01**
	Rating	2.8

Question 35

Reactor Protection System (RPS) is in normal operation.

The ‘A’ Motor / Generator output voltage to falls to 112v AC.

A. One EPA breaker will trip and a half scram on RPS ‘A’

B. Two EPA breakers will trip and a half scram on RPS ‘A’

C. One EPA breaker will trip and no RPS actuation

D. Two EPA breakers will trip and no RPS actuation

Answer: B
Explanation: The RPS system has two EPA breakers after the M/G set in series. Both will trip on <115v AC. Therefore if the M/G output is < 115 v AC then both EPA breakers will trip. This will cause a ½ scram on the Division 1 side only. A is wrong – with the EPA breakers being in series both will trip not one, Plausible if the student doesn’t remember that there are two EPA breakers and with the M/G set being the cause and both are down steam of the M/G set, both will be affected. B is correct C is wrong - with the EPA breakers being in series both will trip not one and a Div 1 RPS actuation will occur causing a Div 1 ½ scram, PLAUSIBLE if the student doesn’t remember the normal electrical lineup for the RPS system which is the M/G sets are the normal feed. D is wrong – a Div 1 RPS actuation will occur causing a Div 1 ½ scram. PLAUSIBLE if the student doesn’t remember the normal electrical lineup for the RPS system which is the M/G sets are the normal feed
Technical References:

GLP-OPS-C7100		
References to be provided to applicants during exam: NONE		
Learning Objective: Document learning objective if possible. GLP-OPS-C7100, Objective 6.1, 6.3, 8		
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)(8)	

Examination Outline Cross-Reference	Level	RO
215003 Intermediate Range Monitor (IRM) System K5. Knowledge of the operational implications of the following concepts as they apply to INTERMEDIATE RANGE MONITOR (IRM) SYSTEM : K5.03 Changing detector position	Tier #	2
	Group #	1
	K/A #	**215003 K5.03**
	Rating	3.0

Question 36

A reactor scram has occurred.

The ATC is performing the subsequent actions of 05-1-02-I-1, Reactor Scram ONEP.

The scram has been reset.

Which of the following describes how the IRMs should be inserted and why?

A. All IRMs can be inserted simultaneously
To ensure that Reactor power is being monitored

B. Only one IRM detector at a time.
Prevent inadvertent scram signal

C. Only one division of IRM detectors at a time
Prevent inadvertent scram signal

D. Two detectors at a time, one in each division
To ensure that Reactor power is being monitored

Answer: C
Explanation: 05-1-02-I-1, Reactor Scram ONEP states, "Insert IRMs as follows: If Reactor scram signal has not been reset then insert all IRMs. If Reactor scram signal has been reset then insert one RPS division of IRMs, when the first RPS division of IRMs are fully inserted then insert the second RPS division of IRMs.' 04-1-01-C51-1 P&L 3.7 and 4.2.2 CAUTION states: "Due to possible converter failures which May cause a half scram." A is wrong – The only time we can insert all IRMs is if the scram has not been reset.

B is wrong – The scram ONEP allows for one division to be inserted.

C is correct

D is wrong – Two detectors in each division could cause a full scram.

Technical References:

05-1-02-I-1
04-1-01-C51-1

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-C5100,
GLP-OPS-IOI

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)(6)	

Examination Outline Cross-Reference	Level	RO
215004 Source Range Monitor (SRM) System K4. Knowledge of SOURCE RANGE MONITOR (SRM) SYSTEM design feature(s) and/or interlocks which provide for the following: K4.01 Rod withdrawal blocks	Tier #	2
	Group #	1
	K/A #	**215004 K4.01**
	Rating	3.7

Question 37

A normal reactor startup is in progress with the reactor subcritical.

Control rods are being withdrawn to achieve critically.

SRM 'E' spikes to 2.5×10^5 cps.

Which of the following describes the plant response?

A. SRM Upscale alarm only

B. SRM Upscale alarm
Control Rod Withdrawal Block

C. SRM Upscale alarm
Control Rod Withdrawal Block
Half scram

D. SRM Upscale alarm
Control Rod Withdrawal Block
Full scram

Answer: B

Explanation:

A spike of any SRM during startup (IRM < range 8) of 2.5×10^5 is above the setpoint of 1×10^5 Control rod block.

A is wrong, yes you would receive the upscale alarm however a rod block would also be received.

B is correct

C is wrong, a half scram would not occur due to shorting links are installed. Plausible due to being above the Upscale trip setpoint of 2 x $10^{5,}$

D is wrong, a full scram would not occur due to shorting links are installed. Plausible due to being above the Upscale trip setpoint of 2 x $10^{5,}$.

Technical References:

GLP-OPS-C5101

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-C5101, Objective 7.3

Question Source:	Bank # 742	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2013 Q #38
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(5)	

Examination Outline Cross-Reference	Level	RO
215005 Average Power Range Monitor/Local Power Range Monitor System K3. Knowledge of the effect that a loss or malfunction of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM will have on following: K3.01 RPS	Tier #	2
	Group #	1
	K/A #	**215005 K3.01**
	Rating	4.0

Question 38

Which of the following will cause a Division 1 half scram only?

A. Loss of 1Y87 power to APRM Channel 1 cabinet

B. Less than 21 LPRMs feeding Channel 1 APRM

C. Mode Switch in RUN with APRM channel 4 indicating 118%

D. Mode Switch in STARTUP with APRM channels 2 and 3 indicating 19%

Answer: A
Explanation: The UPS system supplies regulated 120VAC power to the particular APRM channel and Voter. Each channel and voter has individual UPS power supply. 1Y87 feeds Channel 1. A loss of 120VAC UPS will result in a half scram condition for the particular RPS due to the Voter de-energizing. A is correct B is wrong – this will cause a Control Rod Block not a scram signal, plausible due to this will cause an INOP condition but no RPS actuation. C is wrong - this will cause nothing with RPS but alarms on APRM channel 4, This will cause a high flux trip of channel 4 and a trip of channel 4 voter but no RPS actuation due to logic is 2 of 4 voters tripped. D is wrong – this will cause a full scram due to 2 of 4 channels exceed the scram setpoint of 18% and send a signal to the voters.
Technical References:

GLP-OPS-C5104		
References to be provided to applicants during exam: NONE		
Learning Objective: Document learning objective if possible. GLP-OPS-C5104, Objective 11.2		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(5)	

Examination Outline Cross-Reference	Level	RO
217000 Reactor Core Isolation Cooling System (RCIC) K4. Knowledge of REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) design feature(s) and/or interlocks which provide for the following: K4.02 Prevent over filling reactor vessel	Tier #	2
	Group #	1
	K/A #	**217000 K4.02**
	Rating	3.3

Question 39

A loss of feedwater occurs.

RCIC is manually initiated.

Which of the following describes the action that will prevent over filling the reactor vessel?

A. At 53.5” Wide Range reactor water level, the E51-C002 (RCIC TURB / THROT VLV) will automatically trip close.

B. At 53.5” Narrow Range reactor water level, the E51-C002 (RCIC TURB / THROT VLV) will automatically trip close.

C. At 53.5” Wide Range reactor water level, the E51-F045 (RCIC STM SPLY TO RCIC TURB) will automatically close.

D. At 53.5” Narrow Range reactor water level the E51-F045 (RCIC STM SPLY TO RCIC TURB) will automatically close.

Answer: D
Explanation: The E51-F045 will auto close when reactor water level reaches 53.5” Narrow Range. This will in turn cause injection valve F013 to auto close from the position of the F045. A is wrong – The only action above vessel level 0” that wide range will do is auto close the E22-F004 HPCS injection valve. And, the Turbine Trip Throttle will not close. B is wrong - the Turbine Trip Throttle will not close C is wrong - The only action above vessel level 0” that wide range will do is auto close the E22-F004 HPCS injection valve D is correct.

Technical References:

GLP-OPS-E5100
GLP-OPS-B2101 page 15

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-E5100, Objective 8.9

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)(8)	

Examination Outline Cross-Reference	Level	RO
217000 Reactor Core Isolation Cooling System (RCIC) A3. Ability to monitor automatic operations of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) including: A3.04 System flow	Tier #	2
	Group #	1
	K/A #	**217000 A3.04**
	Rating	3.6

Question 40

A LOCA has occurred.

RCIC automatically initiates and is currently injecting.

RCIC discharge flow transmitter fails and RCIC Discharge Flow indicator E51-R606 begins to slowly lower.

Which of the following describes when the RCIC Min Flow valve, E51-F019 will auto OPEN?

A. When flow rate indicates <175 gpm.

B. When flow rate indicates <125 gpm.

C. When flow rate indicates <95 gpm.

D. When Steam Supply Valve E51-F045 is closed.

Answer: C
Explanation: The E51-F019 will auto open when RCIC flow indicates <95 gpm and discharge pressure is > 125 psig. The Open signal for the min flow valve and the discharge flow indication comes from the same flow transmitter. If the transmitter had failed immediately to zero then the F019 would have opened immediately. With a slow failure of the flow transmitter, evident by the indication also slowly lowering, the F019 will open when the indicator reaches <95 gpm. The pump is still in operation and injecting so discharge pressure can be assumed to be >125 psig. A is wrong – This is the flow at which the F019 will auto close on a system start. B is wrong – The is actually the discharge pressure required >125 psig for F019 to auto open.

C is correct

D is wrong. – E51-F045 is not in the logic for the F019, however the Turbine Trip valve is, any time the trip/throttle valve is closed the F019 gets a closed signal not an open signal.

Technical References:

GLP-OPS-E5100
GFIG-OPS-E5100

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-E5100, Objective 8.4

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)(7)	

Examination Outline Cross-Reference	Level	RO
218000 Automatic Depressurization System K2. 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 41

The Div 2 ADS logic is powered from…

A. Power Panel 16P61

B. RPS bus 'B'

C. Inverter 1Y88

D. Distribution panel 1DB1

Answer: D

Explanation:

ADS logic is DC powered from the Div 2 DC subsystem (11DB) via its distribution panel 1DB1

A is wrong - This is one of the 120 VAC power panels fed from the Div 2 vital bus 16AA via an MCC. Plausible for the same reason as choice 'B'

B is wrong - RPS Bus 'B' supplies 120 VAC, not 125 DC. Plausible to the Applicant who cannot recall that ADS logic is DC powered.

C is wrong - This is the Div 2 inverter which supplies 120 VAC, not 125 DC. Plausible for the same reason as choice 'B'.

D is correct.

Technical References:

GLP-OPS-E2202
04-1-01-L11-1

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible. GLP-OPS-E2202 Obj 19.3		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
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 Primary Containment Isolation System/Nuclear Steam Supply Shut-Off A2. Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.04 Process radiation monitoring system failures	**Tier #**	2
	Group #	1
	K/A #	**223002 A2.04**
	Rating	2.9

Question 42

The plant is at rated power.

A false high Main Steam Line (MSL) Radiation signal is received on all channels.

(1) Which of the following describes that action that will occur?

The condition has been corrected, and you have been directed to restore the isolation

(2) What must be done <u>before</u> the valves are re-opened?

A. (1) Group 10 isolation signal

(2) Main Steam Line Radiation Monitors must be manually reset at each monitor to prevent subsequent re-isolation when the affected valves reach full open.

B. (1) Group 7 isolation signal

(2) Plant Chemistry must be notified to isolate certain manual valves to prevent damage to low pressure piping

C. (1) Group 7 isolation signal

(2) Main Steam Line Radiation Monitors must be manually reset at each monitor to prevent subsequent re-isolation when the affected valves reach full open.

D. (1) Group 10 isolation signal

(2) Plant Chemistry must be notified to isolate certain manual valves to prevent damage to low pressure piping

Answer: D

Explanation:

Group 10 includes reactor sample line drywell isolation valves B33-F019 and B33-F020 which supply the reactor water sample station inside containment. Low pressure sample lines could be damaged by the pressure spike which occurs when the automatic isolation MOVs are opened at rated pressure. Therefore, Chemistry must be notified to close certain manual sample isolation valves that are under Chemistry's control, not Operations' control.

A is wrong - is plausible since the MSL Radiation circuit caused the event, however the MSL Radiation Monitors have no manual reset feature. The logic involved is reset using NSSSS isolation reset pushbuttons on H13-P601. Group 10 valves would not open until NSSSS was reset, and NSSSS could not be reset unless sufficient MSL Radiation Monitors were below the isolation setpoint..

B is wrong – Group 7 is Containment Ventilation isolation which has a containment vent high radiation siganal, not MSL radiation signal.

C is wrong - Group 7 is Containment Ventilation isolation which has a containment vent high radiation siganal, not MSL radiation signal. is plausible since the MSL Radiation circuit caused the event, however the MSL Radiation Monitors have no manual reset feature. The logic involved is reset using NSSSS isolation reset pushbuttons on H13-P601. Group 10 valves would not open until NSSSS was reset, and NSSSS could not be reset unless sufficient MSL Radiation Monitors were below the isolation setpoint.

Technical References:

04-1-01-B33-1 step 5.3
05-1-02-III-5,

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-B3300 Obj. 38

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #928	X
	New	
Question History:	Last NRC Exam	2007 NRC (Q42)
		2007 Exam is not on record in the bank.
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
239002 Relief/Safety Valves	Tier #	2
	Group #	1
A3. Ability to monitor automatic operations of the RELIEF/SAFETY VALVES including:	K/A #	**239002 A3.01**
	Rating	3.8
A3.01 SRV operation after ADS actuation		

Question 43

A LOCA has occurred.

ADS has automatically initiated.

Reactor pressure is lowering

All other plant parameters have not changed since the auto initiation of ADS.

Which of the following will close all ADS valves and maintain them in a closed position?

A. Depress and release the ADS A LOGIC RESET and ADS B LOGIC RESET pushbuttons

B. Stop all Low Pressure ECCS pumps.

C. Place the ADS A MANUAL INHIBIT and ADS B MANUAL INHIBIT switch to INHIBIT

D. Depress and release the ADS A HI DRWL PRESS RESET and ADS B HI DRWL PRESS RESET pushbuttons.

Answer: B
Explanation: Once ADS has auto initiated the only way to close the valves is to stop the logic. Depressing the logic reset pushbuttons will reset the logic however the valves will reopen after a 105 sec time delay if all parameters remain the same. The only way to close the valves and ensure no auto reopening from the answers listed is to stop all Low Pressure ECCS pumps, this breaks the logic just prior to the open signal, the valves will close and remain close. A is wrong - as mentioned above the valves will close but will reopen after 105 seconds. B is correct C is wrong – has no effect on the valves if already open. If answer C is used with answer A

then the valves would remain closed, but, neither by themselves will close and remain close the ADS valves.

D is wrong – after the Hi Drywell pressure signal, pressure must be below the setpoint before this will work, Drywell pressure has a seal in signal.

Technical References:

GLP-OPS-E2202
GFIG-OPS-E2202

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-E2202 Obj. 12, 15,

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)(7)	

Examination Outline Cross-Reference	Level	RO
259002 Reactor Water Level Control System A4. Ability to manually operate and/or monitor in the control room: A4.02 All individual component controllers in the automatic mode	Tier #	2
	Group #	1
	K/A #	**259002 A4.02**
	Rating	3.7

Question 44

The plant is operating at rated power when a total loss of feedwater causes a reactor scram on low reactor water level.

Assuming reactor water level never rises above Level 3, what will be the setpoint on the FW MASTER LEVEL Controller 30 seconds after the scram?

A. 54.0 inches

B. 12.4 inches

C. 11.4 inches

D. 4.0 inches

Answer: D

Explanation:

The Low level (Level 3) scram setpoint is 11.4”. DFCS will initiate Setpoint Setdown also at 11.4”. Once initiated, the FW MASTER LEVEL Controller changes to 54.0” for either 10 seconds or until 12.4” is reached, at which time the setpoint changes to 4.0”.

A is wrong – this would be correct if the time was within 10 seconds.

B is wrong – this is the reset level as discussed above.

C is wrong – this is the setpoint for Setdown to initiate.

D is correct

Technical References:

GLP-OPS-C3401

References to be provided to applicants during exam: NONE		
Learning Objective: Document learning objective if possible. GLP-OPS-C3401, Objective 12		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #298	X
	New	
Question History:	Last NRC Exam	2011 NRC EXAM Q3
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	2
10CFR Part 55 Content:	55.41(b)(4)	

Examination Outline Cross-Reference	Level	RO
259002 Reactor Water Level Control System	Tier #	2
	Group #	1
Ability to explain and apply system limits and precautions.	K/A #	**259002 G2.1.32**
	Rating	3.8

Question 45

In Speed Auto the RFPT governor speed setpoint raises at a rate of _________(1)________ and the RAISE pushbutton will extinguish if __________(2)_________.

A. (1) 10 rpm/sec for 1 second and then 120 rpm/sec thereafter
   (2) the RFPT reaches the High Speed Setpoint

B. (1) 0 to 100% in 15 seconds
   (2) the RFPT reaches the High Speed Setpoint

C. (1) 10 rpm/sec for 1 second and then 120 rpm/sec thereafter
   (2) the RFPT reaches critical speeds

D. (1) 0 to 100% in 15 seconds
   (2) the RFPT reaches critical speeds

Answer: A

Explanation:

Per 04-1-01-N21-1 P&L 3.14 - 3.16 states that the Speed Auto ramp rate is 10 rpm/sec for 1 second and then 120 rpm/sec thereafter (0-100% in 15 seconds is the rate for MANUAL operation).

In Speed Auto, RAISE pb will extinguish if it reaches the High Speed Setpoint (critical speeds are 3100-3200 and 3279-3379 rpm well below the High Speed Setpoint).

A is correct

B is wrong – this is the rate for the controls while in MANUAL

C is wrong – the RAISE pb will extinguish if it reaches the High Speed Setpoint

D is wrong - this is the rate for the controls while in MANUAL, the RAISE pb will extinguish if it reaches the High Speed Setpoint

Technical References: 04-1-01-N21-1, Feedwater SOI		
References to be provided to applicants during exam: NONE		
Learning Objective: Document learning objective if possible. GLP-OPS-C3401, Objective 10.4		
Question Source:	Bank # 618	2013 AUDIT Q #45
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(4)	

Examination Outline Cross-Reference	Level	RO
261000 Standby Gas Treatment System K6 Knowledge of the effect that a loss or malfunction of the following will have on the STANDBY GAS TREATMENT SYSTEM : K6.01 A.C. electrical distribution	**Tier #**	2
	Group #	1
	K/A #	**261000 K6.01**
	Rating	2.9

Question 46

A LOCA is in progress with the following:

- RCIC automatically initiated
- Reactor water level is now 0 inches
- Drywell pressure is 1.12 psig
- All radiation monitors are indicating below their trip setpoints
- 'A' SGTS is operating
- 'B' SGTS has been placed in the STANDBY mode with its Mode Select Switch

Then, bus 15AA receives a BUV signal.

Without operator action, which of the following describes the resulting status of SGTS, 2 minutes after the BUV signal was received?

A. Neither train is operating

B. Both trains are operating

C. Only the 'A' train is operating

D. Only the 'B' train is operating

Answer: B
Explanation: Train A trips when the bus 15AA BUV is received, causing Train B to auto-start (on low flow from Train A). After LSS, 'B' is still running and 'A' has restarted because a LOCA signal still exists and Train A is still in AUTO mode. A is wrong – As stated above both trains will be operating B is correct

C is wrong – As stated above both trains will be operating D is wrong - As stated above both trains will be operating		
Technical References: 04-1-02-1H13-P870 2A-F2		
References to be provided to applicants during exam: NONE		
Learning Objective: Document learning objective if possible. GLP-OPS-T4801 Objective: 9.1		
Question Source:	Bank # 389	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(8)	

Examination Outline Cross-Reference	Level	RO
262001 A.C. Electrical Distribution A1. Ability to predict and/or monitor changes in parameters associated with operating the A.C. ELECTRICAL DISTRIBUTION controls including: A1.03 Bus voltage	**Tier #**	2
	Group #	1
	K/A #	**262001 A1.03**
	Rating	2.9

Question 47

A grid disturbance has occurred.

Bus 15AA is being powered from Division 1 Diesel Generator.

The grid disturbance has been corrected and the CRS has directed to parallel an offsite source to Division 1 Diesel Generator.

Which of the following describes the control that is used to adjust Diesel Generator voltage?

AND

What is the desired voltage on the 15AA bus prior to paralleling?

A. DG 11 AUTO VR SETPT CONT
   50 volts below INCOMING volts

B. DG 11 GOV MAN CONT
   50 volts below INCOMING volts

C. DG 11 AUTO VR SETPT CONT
   50 volts above INCOMING volts

D. DG 11 GOV MAN CONT
   50 volts above INCOMING volts

Answer: C
Explanation: Per 04-1-01-P75-1, Standby Diesel Generator System SOI, step 4.3.2 d – ADJUST Standby Diesel Generator 11 RUNNING VOLTS about 50 volts above INCOMING VOLTS DIV 1 with AUTO VOLT SETPOINT CONT DG 11 handswitch.

A is wrong – As stated above it should be 50 volts above incoming not below

B is wrong – This switch is used to adjust speed, frequency and load not voltage. As stated above it should be 50 volts above incoming not below

C is correct – ADJUST Standby Diesel Generator 11 RUNNING VOLTS about 50 volts above INCOMING VOLTS DIV 1 with AUTO VOLT SETPOINT CONT DG 11 handswitch.

D is wrong – This switch is used to adjust speed, frequency and load not voltage.

Technical References:

04-1-01-P75-1 step 4.3.2 d

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-P7501 Objective: 17

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)(7)	

Examination Outline Cross-Reference	Level	RO
262002 Uninterruptable Power Supply (A.C/D.C.) K1. Knowledge of the physical connections and/or cause effect relationships between UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) and the following: K1.14 Main steam line radiation monitors	**Tier #**	2
	Group #	1
	K/A #	**262002 K1.14**
	Rating	2.8

Question 48

Which of the following describes the power supply for the ‘A’ Main Steam Line Radiation Monitor?

A. RPS ‘A’

B. 11DD 125 volts DC

C. BOP 1Y80 Inverter

D. ESF 1Y87 Inverter

Answer: D
Explanation: Per 04-1-01-D21-1, Attachment III Panel 1Y89 breaker 07 supplies power to the ‘A’ Main Steam Line Monitor. 1Y89 distribution panel is powered from 1Y87 ESF Static Inverter. A is wrong – Only supplies power to ‘A’ RPS B is wrong – Plausible due to it being the battery supply for BOP inverter .1Y79 C is wrong – 1Y80 is a Static inverter but it’s a BOP inverter not an ESF inverter. D is correct –1Y89 panel is powered from 1Y87 Inverter.
Technical References: 04-1-01-D17-1 Attachment III, 04-1-01-L62-01, Attachment II
References to be provided to applicants during exam: NONE

Learning Objective: Document learning objective if possible. GLP-OPS-D1721 Objective: 9		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(8)	

Examination Outline Cross-Reference	Level	RO
263000 D.C. Electrical Distribution K3. Knowledge of the effect that a loss or malfunction of the D.C. ELECTRICAL DISTRIBUTION will have on following: K3.03 Systems with D.C. components	**Tier #**	2
	Group #	1
	K/A #	**263000 K3.03**
	Rating	3.4

Question 49

The 11DK breaker that powers the Alternate Rod Insertion system has tripped open (breaker problem) and will not re-close.

How is the Alternate Rod Insertion system capability impacted and why?

A. Cannot function to depressurize the scram air header because all 3 vents paths are disabled.

B. Can still function to depressurize the scram air header because only one Channel is disabled.

C. Can still function to depressurize the scram air header because only one vent path is disabled.

D. Cannot function to depressurize the scram air header because 6 of the ARI valve solenoids are disabled.

Answer: A
Explanation: Bus 11DE powers the logic for ARI Channel 1 and the solenoids for ARI valves F162B, F162D, and F164A. Bus 11DK powers the logic for ARI Channel 2 and the solenoids for ARI valves F162A, F162C, and F160. The normally-closed vent paths are thru: F162A & B; F162C & D; F164A & F160. Given these 4 answer choices, the only one that fits is choice 'C'. In other words without 11DK power, we cannot energize the solenoids for one of the ARI valves that we need to open in each of the 3 possible vent paths; i.e., none of the vent paths can be opened and so the header cannot be depressurized by ARI. Note, these are single-solenoid, not dual-solenoid valves. All distracters are wrong (but their plausibility is based on the Applicant's need to firmly understand both the DC power interrelationship within the ARI system and the ARI vent path arrangement).

Technical References: GLP-OPS-C111A, CRD system lesson plan GFIG-OPS-C111A, CRD system lesson plan figures…Figures 9 & 10 for ARI arrangement		
References to be provided to applicants during exam: NONE		
Learning Objective: Document learning objective if possible. GLP-OPS-C111A, Objective 5.5		
Question Source:	Bank # 377	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2011 NRC EXAM Q #68
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
264000 Emergency Generators (Diesel/Jet) A2. Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS (DIESEL/JET) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.03 Operating unloaded, lightly loaded, and highly loaded	Tier #	2
	Group #	1
	K/A #	**264000 A2.03**
	Rating	3.4

Question 50

A spurious initiation of HPCS has occurred.

30 minutes later I&C has found the problem and the HPCS system is reset.

1.) Which of the following describes the impact on Division 3, HPCS Diesel Generator running unloaded?

AND

2.) What are the requirements of loading the Diesel Generator?

A. (1.) The exhaust system may experience an excessive oil residue buildup.
(2.) The Generator must be loaded for at least 4 hours with at least 50% load.

B. (1.) The exhaust system may experience an excessive oil residue buildup.
(2.) The Generator must be loaded for at least 30 minutes with at least 50% load.

C. (1.) The exhaust gas ΔT from each engine may exceed 111°C of each other.
(2.) The Generator must be loaded for at least 30 minutes with at least 50% load.

D. (1.) The exhaust gas ΔT from each engine may exceed 111°C of each other.
(2.) The Generator must be loaded for at least 4 hours with at least 50% load.

Answer: B

Explanation:

Per 04-1-01-P81-1, High Pressure Core Spray Diesel Generator, step 3.1:

3.1 To prevent excessive oil residue buildup within the exhaust system (souping) which can become a source of ignition at STARTUP, the following limits apply:

3.1.1 **WHENEVER** a non-surveillance start occurs that is **NOT** terminated within two minutes, **THEN** the generator Must be loaded for at least 30 minutes with at least 50% load.

3.1.2 The diesel engine May run a maximum of four hours at a no-load condition, **THEN** the generator Must be loaded for at least 30 minutes with at least 50% load.

A is wrong – per the above procedure if its run for >2 minutes then load it for at least 30 min., the 4 hours is used in 3.1.2 above the max time the engine can be run at no load.

B is correct

C is wrong – precaution 3.3 is for anytime not just running unloaded.

D is wrong - precaution 3.3 is for anytime not just running unloaded, per the above procedure if its run for >2 minutes then load it for at least 30 min., the 4 hours is used in 3.1.2 above the max time the engine can be run at no load.

Technical References:

04-1-01-P81-1, Div III D/G SOI

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-P8100, Objective 19.1

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

Examination Outline Cross-Reference	Level	RO
264000 Emergency Generators (Diesel/Jet) 2.1.30 Ability to locate and operate components, including local controls.	**Tier #**	2
	Group #	1
	K/A #	**264000 G2.1.30**
	Rating	4.4

Question 51

Per the SOI, operators have to place DG 11 in MAINTENANCE mode using the local pushbutton and the DG 11 MAINT PERM pushbutton at control room panel H13-P864.

Which of the following describes which local panel is the DG 11 MAINTENANCE mode select pushbutton located?

AND

Which of the following describes which pushbutton MUST be released first for Diesel to remain in MAINTENANCE mode?

A. (1.) H22-P400
(2.) Control Room H13-P864 pushbutton

B. (1.) H22-P113
(2.) Control Room H13-P864 pushbutton

C. (1.) H22-P400.
(2.) Local MAINTENANCE mode select pushbutton

D. (1.) H22-P113
(2.) Local MAINTENANCE mode select pushbutton

Answer: C
Explanation: Per 04-1-01-P75-1, Standby Diesel Generator System, Attachment VI.: **PLACE** Standby Diesel Generator 11 [12] in MAINTENANCE mode by **SIMULTANEOUSLY PRESSING** Remote 1H13-P864 **AND** Local 1H22-P400 [1H22-P401] MAINTENANCE MODE SELECT pushbuttons. LOCAL pushbutton Must be **RELEASED** first for Diesel to remain in MAINTENANCE mode A is wrong – per the above procedure local pushbutton must be released first.

B is wrong – per the above procedure the local panel is the P400, plausible because the P113 is the generator local panel and per the above procedure local pushbutton must be released first.

C is correct

D is wrong - per the above procedure the local panel is the P400, plausible because the P113 is the generator local panel

Technical References:

04-1-01-P75-1, Standby Diesel Generator System

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-P7500, Objective 12

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

Examination Outline Cross-Reference	Level	RO
300000 Instrument Air System (IAS) A4. Ability to manually operate and / or monitor in the control room: A4.01 Pressure gauges	Tier #	2
	Group #	1
	K/A #	**300000 A4.01**
	Rating	2.6

Question 52

The plant is operating at rated power with **<u>no</u>** Instrument Air System problems.

Which of the following identifies the control room panel where operators can read INSTRUMENT AIR SUPPLY HEADER PRESSURE, <u>and</u> identifies what that indicator normally reads?

A. Panel H13-P854; normally reads approximately 135 psig

B. Panel H13-P870; normally reads approximately 110 psig

C. Panel H13-P854; normally reads approximately 110 psig

D. Panel H13-P870; normally reads approximately 135 psig

Answer: B
Explanation: There is only one Instrument Air Supply Header Pressure indicator in the control room; it is located on H13-P870 and normally reads about 110 psig. See 02-S-01-31, Control Room Rounds two sheets (one for Instrument Air at H13-P870 page 15 of 39, the other for Service Air at H13-P854 page 12 of 39). Panel H13-P854 is plausible because that is where operators control two of the Plant Air Compressors (PAC 'B' and 'C') and is the location of the indicator for Service Air Header Pressure. 135 psig is plausible because this is the "unload" setpoint for the in-service ("LEAD") PAC. A is wrong – per the above procedure indication is on H13-P870, and normal header pressure is 110 psig. B is correct C is wrong - per the above procedure indication is on H13-P870 D is wrong - normal header pressure is 110psig.

Technical References: 02-S-01-31, Control Room Rounds GLP-OPS-P5100, Plant Air System lesson		
References to be provided to applicants during exam: NONE		
Learning Objective: Document learning objective if possible. GLP-OPS-P5100, Objective 33		
Question Source:	Bank # 95	2015 AUDIT
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2012 NRC EXAM Q #3
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(8)	

Examination Outline Cross-Reference	Level	RO
400000 Component Cooling Water System (CCWS) K4. Knowledge of CCWS design feature(s) and or interlocks which provide for the following: K4.01 Automatic start of standby pump	Tier #	2
	Group #	1
	K/A #	**400000 K4.01**
	Rating	2.6

Question 53

Which of the following indicates the header pressure when the Standby CCW pump will automatically start?

A. 45 psig

B. 80 psig

C. 100 psig

D. 110 psig

Answer: C
Explanation: Per 04-1-02-1H13-P870-5A-C2 and 8A-E1, the standby CCW pump (any one of the three) will automatically start when system discharge pressure is less than 100 psig. A is wrong – This is the pressure that the Standby TBCW pumps will auto start. B is wrong – This pressure will initiate the CCW DISCH HDR PRESS LO-LO alarm. C is correct D is wrong - normal header pressure is 110 psig.
Technical References: GLP-OPS-P4200, CCW System lesson Plan 04-1-02-1H13-P870-5A-C2 and 8A-E1
References to be provided to applicants during exam: NONE

Learning Objective: Document learning objective if possible. GLP-OPS-P4200, Objective 7.1		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(4)	

Examination Outline Cross-Reference	Level	RO
201001 Control Rod Drive Hydraulic System A1. Ability to predict and/or monitor changes in parameters associated with operating the CONTROL ROD DRIVE HYDRAULIC SYSTEM controls including: A1.03 CRD system flow	**Tier #**	2
	Group #	2
	K/A #	**201001 A1.03**
	Rating	2.9

Question 54

The plant is operating at rated power.

CRD pump ‘A’ is operating.

A reactor scram occurs.

Which of the following describes the total CRD system Cooling Water Header flow rate?

A. 0 gpm

B. 5 gpm

C. 60 gpm

D. 165 gpm

Answer: B
Explanation: Normal system flow rate is 60 gpm, which is all going through the Cooling water header during normal operations with no control rod movement. During a reactor scram flow is diverted to the charging water header through a restricting orifice to limit flow to 165 gpm (to prevent pump runout), the charging water header is downstream of the flow element but, upstream of the flow control valve. When a scram occurs the flow element senses high flow and tells the FCV to close due to being above the 60 gpm setpoint. The FCV is mechanically blocked to continue to pass 5 gpm even with the valve indicating full closed. A is wrong – Plausible if the student believes that the FCV will fully close. B is correct C is wrong – This is normal system flow rate.

D is wrong – this is the charging water header flow rate not cooling water header.		
Technical References: GLP-OPS-C111A, Control Rod Drive Hydraulic system		
References to be provided to applicants during exam: NONE		
Learning Objective: Document learning objective if possible. <u>GLP-OPS-C111A, Objective 7.3</u>		
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)(6 & 14)	

Examination Outline Cross-Reference	Level	RO
201003 Control Rod and Drive Mechanism K5. Knowledge of the operational implications of the following concepts as they apply to CONTROL ROD AND DRIVE MECHANISM : K5.08 How control rods affect shutdown margin	Tier #	2
	Group #	2
	K/A #	**201003 K5.08**
	Rating	3.1

Question 55

A plant Startup is in progress.

As reactor operators withdraw control rods to achieve criticality Shutdown Margin will ______________.

The Tech Spec limit for Shutdown Margin when the highest worth control rod analytically determined is ______________.

A. Raise
$\geq 0.28\% \Delta k/k$

B. Lower
$\geq 0.28\% \Delta k/k$

C. Raise
$\geq 0.38\% \Delta k/k$

D. Lower
$\geq 0.38\% \Delta k/k$

Answer: D
Explanation: Any time control rods are withdrawn the shutdown margin will lower. Per Tech Specs 3.1.1 states that "SDM shall be: $\geq 0.38\% \Delta k/k$, with the highest worth control rod analytically determined." A is wrong – SDM will lower and 0.28 is when the highest worth rod is determined by test. B is wrong - 0.28 is when the highest worth rod is determined by test C is wrong – SDM will lower.

D is correct		
Technical References: GLP-OPS-TS001, Tech Specs Tech Specs 3.1.1		
References to be provided to applicants during exam: NONE		
Learning Objective: Document learning objective if possible. GLP-OPS-TS001, Objective 4.13		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(1)	

Examination Outline Cross-Reference	Level	RO
201005 Rod Control and Information System (RCIS) K3. Knowledge of the effect that a loss or malfunction of the ROD CONTROL AND INFORMATION SYSTEM (RCIS) will have on following: K3.02 Reactor startup: BWR-6	Tier #	2
	Group #	2
	K/A #	**201005 K3.02**
	Rating	3.5

Question 56

A plant Startup is in progress.

Control rods are being withdrawn to achieve criticality.

Which of the following will allow the startup to continue **without** bypassing a Control Rod within the RCIS system?

A. Withdrawing a control rod and a reed switch on one channel fails to close.

B. Withdrawing a control rod past its banked position.

C. Loss of one RACS channel

D. Loss of RGDS

Answer: A
Explanation: If one reed switch fails then the operator can enter substitute position per C11-2 SOI and continue the startup. A is correct B is wrong – this will cause an Insert and Withdraw block, the only way to continue is to bypass the control rods position within RACS channel 1 & 2 and move the rod to its intended position. C is wrong – loss of one channel of RACS will cause a channel disagree and a control rod block, it must be working prior to continuing with startup. D is wrong - loss of one channel of RGDS will cause a control rod block, it must be working prior to continuing with startup

Technical References: GLP-OPS-C1102 04-1-01-C11-2		
References to be provided to applicants during exam: NONE		
Learning Objective: Document learning objective if possible. GLP-OPS-C1102, Objective 13		
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)(6)	

Examination Outline Cross-Reference	Level	RO
202001 Recirculation System A2. Ability to (a) predict the impacts of the following on the RECIRCULATION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.08 Recirculation flow mismatch: Plant-Specific	Tier #	2
	Group #	2
	K/A #	**202001 A2.08**
	Rating	3.1

Question 57

The plant at 85% rated power when the 'A' Recirc Flow Control Valve malfunctions and begins to close.

The ATC operator arms and depresses the HPU 'A' SHUTDN pushbutton. The following conditions exist following the transient:

- TOT JP FLO 1B33-UR-R613 is 80.5 MLBM/HR
- RECIRC PMP 'A' DRIVING FLOW 1C51-FR-R614 is 28.4 KGPM
- RECIRC PMP 'B' DRIVING FLOW 1C51-FR-R614 is 33.8 KGPM

What is the impact of this event?

What operator action is required to mitigate this event?

A. Tech Specs entry is required
   Balance loop flows to within 4460 gpm

B. NO Tech Specs entry required
   Balance loop flows to within 4460 gpm.

C. Tech Specs entry is required
   Balance loop flows to within 2230 gpm

D. NO Tech Specs entry required
   Balance loop flows to within 2230 gpm

Answer: C
Explanation:

Tech Specs 3.4.1 is entered due to flows not being balanced, when ≥ 70% core flow loop flows must be within 5% of each other and when < 70% core flow, loop flows must be with 10% of each other.

The student has to calculate % core flow, which is approximately 72%. Tech Specs says that when ≥ 70% core flow loop flows must be within 5% of each other and when < 70% core flow, loop flows must be with 10% of each other OR as said in the Reduction in Recirc System Flow Rate ONEP, “At greater than 78.7 mlbm/hr core flow, BALANCE loop flows to within 2230 gpm." This is within the value that will allow the plant to exit TS 3.4.1 (entered due to flow mismatch).

A is wrong – loop flows must be within 5% when ≥ 70% core flow, this is 10%

B is wrong – Tech Spec entry is required, and loop flows must be within 5% when ≥ 70% core flow, this is 10%.

C is correct

D is wrong - Tech Spec entry is required

Technical References:

TS 3.4.1
05-1-02-III-3
04-1-01-B33-1 P&L 3.2.4

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-B3300 Obj 48

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #46	X**
	New	
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(10)	

** Not used on any NRC or Audit exam.

Examination Outline Cross-Reference	Level	RO
202002 Recirculation Flow Control System K2. Knowledge of electrical power supplies to the following: K2.02 Hydraulic power unit: Plant-Specific	**Tier #**	2
	Group #	2
	K/A #	**202002 K2.02**
	Rating	2.6

Question 58

Which of the following describes the power supply for the Reactor Recirc HPU Oil Pump ‘C002A’ on the ‘A’ skid?

A. 15BA1

B. 11DF

C. 11B51

D. 15B42

Answer: C
Explanation: Per 04-1-01-B33-1, Att. III, power supply for Reactor Recirc HPU Oil Pump ‘C002A’ is 11B51. A is wrong – This is a ESF LCC not a MCC, plausible if the student believes that the FCV HPU is ESF powered B is wrong – This is a 250VDC supply, Main Turbine Emergency Oil pump is powered from this supply along with a number of emergency oil pumps, plausible if the student believes that the HPU oil pump is a DC powered oil pump. C is correct D is wrong – This is an ESF power supply and Recirc HPU is BOP Powered, and this powers the ‘A” RPS M/G set, plausible for the stated reasons.
Technical References: 04-1-01-B33-1 Att III page 3 of 5
References to be provided to applicants during exam: NONE

Learning Objective: Document learning objective if possible. GLP-OPS-B3300 Obj 21		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(4)	

Examination Outline Cross-Reference	Level	RO
204000 Reactor Water Cleanup System A3. Ability to monitor automatic operations of the REACTOR WATER CLEANUP SYSTEM including: A3.03 Response to system isolations	Tier #	2
	Group #	2
	K/A #	**204000 A3.03**
	Rating	3.6

Question 59

The plant is operating at 100% power.

A problem has occurred in the CCW system causing RWCU non-regenerative heat exchanger outlet temperature to rise.

Which of the following states the response to the RWCU system if non-regenerative heat exchanger outlet temperature continues to rise to >140°F?

A. Full Group 8 isolation

B. G33-F004 isolates only

C. Division 1 Group 8 isolation only

D. G33-F001 and G33-F251 isolate only

Answer: B
Explanation: Per 04-1-01-G33-1, 3.1, system isolation at 140°F. 04-1-02-1H13-P680-11A-C6, Auto Actions: RWCU valve F004 closes. A is wrong – 140°F will only close the G33-F004, and not a group 8 isolation signal. B is correct C is wrong - 140°F is not a group 8 isolation signal D is wrong – These two valves will close on a SLC ‘B’ initiation.
Technical References: 04-1-01-G33-1 3.1 04-1-02-1H13-P680-11A-C6, Auto Actions: RWCU valve F004 closes.

References to be provided to applicants during exam: NONE		
Learning Objective: Document learning objective if possible. GLP-OPS-G3300 Obj 8		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
216000 Nuclear Boiler Instrumentation 2.1.20 Ability to interpret and execute procedure steps	Tier #	2
	Group #	2
	K/A #	**216000 G2.1.20**
	Rating	4.6

Question 60

The plant is experiencing a LOCA.

No injection systems are available

PDS/SPDS computers are inoperable.

The following conditions are present:

- Reactor pressure is 610 psig and slowly lowering
- Indicated Fuel Zone level is -190 inches and lowering

Which of the following states the Fuel Zone indication that would correspond to the CRS directing an Emergency Depressurization at -204 inches?

A. -228 inches

B. -232 inches

C. -237 inches

D. -241 inches

Answer: B
Explanation: Fuel Zone level indication on P601 panel is NOT density compensated, therefore an operator must be able to use the nomograph to determine actual reactor level. Using the given level indication the operator must know that conservative bias requires them to use the 600 psig line for the most conservative indication. Per 02-S-01-43, Transient Mitigation Strategy, using figure 1 determines that the 600 psig line is to be used and it intersects the -204" line at -232". A is wrong – correct if using the 450 psig pressure line.

B is correct

C is wrong – correct if using the 800 psig pressure line.

D is wrong – correct if using the 1060 psig pressure line.

Technical References:

02-S-01-43, Transient Mitigation Strategy, using figure 1

References to be provided to applicants during exam:

02-S-01-43, Transient Mitigation Strategy, Figure 1

Learning Objective: Document learning objective if possible.

GLP-OPS-PROC

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

Examination Outline Cross-Reference	Level	RO
223001 Primary Containment System and Auxiliaries K1. Knowledge of the physical connections and/or cause effect relationships between PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES and the following: K1.14 RCIC: Plant-Specific	Tier #	2
	Group #	2
	K/A #	**223001 K1.14**
	Rating	3.3

Question 61

The plant is experiencing a LOCA.

Containment pressure is rising.

How will this rise in Containment pressure affect the RCIC system?

A. RCIC Turbine may trip on high exhaust pressure

B. RCIC suppression pool suction swap will be less conservative

C. RCIC Turbine exhaust vacuum breakers will not automatically open against a positive pressure in the Containment.

D. RCIC pump minimum flow valve will open sooner due to the effects on the RCIC pump discharge pressure.

Answer: A
Explanation: Per 05–S-01-EP-1 CAUTION 3, Elevated CTMT pressure may trip the RCIC turbine on high exhaust pressure. A is correct. B is wrong – Suppression pool level will be affected but will be more conservative by swapping prior to actual level reaching the setpoint due to positive pressure in the containment on the level transmitter. C is wrong – RCIC vacuum breakers do discharge into the air space inside containment above the suppression pool level but a positive pressure will not affect their operation. D is wrong – RCIC pump does discharge through the min flow back to the suppression pool but an elevated containment pressure will have no effect on RCIC pump discharge pressure.

Technical References: 05–S-01-EP-1 CAUTION 3		
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	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(9)	

Examination Outline Cross-Reference	Level	RO
226001 RHR/LPCI: Containment Spray System Mode K4. Knowledge of RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE design feature(s) and/or interlocks which provide for the following: K4.01 Testability of all operable components	Tier #	2
	Group #	2
	K/A #	**226001 K4.01**
	Rating	2.6

Question 62

The plant is operating at 100% power.

RHR 'A' system is in STANDBY.

The Crew is currently performing 06-OP-1E12-Q-0005, LPCI/RHR SUBSYSTEM A MOV FUNCTIONAL TEST.

Which of the following RHR valves is interlocked with and must be closed to test the E12-F028A, CTMT SPR A SPARGER INL VLV?

A. E12-F004A, RHR PMP A SUCT FM SUPP POOL

B. E12-F042A, RHR A INJ SHUTOFF VLV

C. E12-F024A, RHR A TEST RTN TO SUPP POOL

D. E12-F027A, RHR A SYS SHUTOFF VLV

Answer: D
Explanation: The F027A must be closed prior to the F028A being stroked A is wrong – This valve is interlocked with the F024A and F006A. B is wrong – This valve is interlocked with reactor pressure. C is wrong – This valve is interlocked with the F004A and F006A. D is correct.
Technical References:

GLP-OPS-E1200		
References to be provided to applicants during exam: NONE		
Learning Objective: Document learning objective if possible. GLP-OPS-E1200, OBJ 8.11		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
239001 Main and Reheat Steam System A4. Ability to manually operate and/or monitor in the control room: A4.01 MSIV's .	Tier #	2
	Group #	2
	K/A #	**239001 A4.01**
	Rating	4.2

Question 63

The plant has scrammed with a full group 1 isolation.

All MSIVs are closed.

Current conditions:

- Reactor water level is +5 inches Narrow Range
- Condenser vacuum is 15” Hg Vac
- Main Steam Line Tunnel temperature is normal.

The CRS directs you to reset the group 1 isolation and open ALL MSIVs.

To ensure a Group 1 logic reset, which of the following describes the handswitches that are **required** to be manipulated prior to the NSSSS reset pushbuttons being depressed?

1. NSSSS Div 1, 2, 3, 4 CNDSR LO VAC BYP to BYPASS
2. MSL A, B, C and D DRWL INBD ISOL handswitches to CLOSE
3. MSL A, B, C and D CTMT OTBD ISOL handswitches to CLOSE
4. MAIN STEAM LINE TRIP LOGIC A, B, C, D TEST to TEST

A. 1 and 4 only

B. 2 and 3 only

C. ALL

D. 1, 2, and 3

Answer: B
Explanation:

Per 04-1-01-M71-1, Containment and Drywell Instrumentation and Control, section 5.6,

- Place the NSSSS div 1-4 CNDSR LO VAC BYP switches in BYPASS IF main condenser vacuum is less than 9” Hg.
- Place MSL A,B,C and D DRWL INBD AND CTMT OTBD, handswitches to CLOSED.

The low vacuum bypass switches are not required due to being > 9” Hg. Therefore only the MSIV handswitches are required

A is wrong – #1 Not required due to being >9”Hg and #4 has nothing to do with group 1 isolation but is used during a RPS MSIV closure test.

B is correct.

C is wrong – #1 Not required due to being >9”Hg and #4 has nothing to do with group 1 isolation but is used during a RPS MSIV closure test.

D is wrong – #1 Not required due to being >9”Hg

Technical References:

04-1-01-M71-1, Containment and Drywell Instrumentation and Control, section 5.6

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-B1300, Obj 10
GLP-OPS-M7100, Obj 9.3

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)(10)	

Examination Outline Cross-Reference	Level	RO
259001 Reactor Feedwater System K6. Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR FEEDWATER SYSTEM : K6.02 Condensate system	Tier #	2
	Group #	2
	K/A #	**259001 K6.02**
	Rating	3.3

Question 64

The plant is operating at 100% reactor power.

Condensate Booster pump ‘C’ trips on overcurrent.

Reactor Feedwater Pump A and B suction pressure lowers to <157 psig.

If these parameters stay constant, which of the following describes the effect on the Reactor Feedwater system?

A. ‘A’ RFPT will trip immediately and ‘B’ will trip in 15 seconds

B. ‘A’ RFPT will trip in 45 seconds

C. Both RFPTs will trip in 15 seconds

D. ‘A’ RFPT will trip in 15 seconds

Answer: D
Explanation: Per 04-1-02-1H13-P680- 2A-A2 and A12, RFPT A / B TRIP, RFP suction pressure low <157 psig with 15 second time delay for the ‘A’ pump and 45 second time delay for the ‘B’ pump. After the ‘A’ pump trips a Recirc flow control valve runback will occur causing an immediate reduction in power and causing suction pressure for the ‘B’ pump to rise above the trip setpoint and continue operation. A is wrong – A will not trip immediately must have a 15 second TD and ‘B’ will trip with a 45 second TD B is wrong – ‘A’ will trip with a 15 second TD and ‘B’ will trip with a 45 second TD C is wrong – ‘A’ will trip with a 15 second TD and ‘B’ will trip with a 45 second TD D is correct.

Technical References: 04-1-02-1H13-P680- 2A-A2 and A12, RFPT A / B TRIP		
References to be provided to applicants during exam: NONE		
Learning Objective: Document learning objective if possible. GLP-OPS-N2100, Obj 12		
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)(4)	

Examination Outline Cross-Reference	Level	RO
286000 Fire Protection System 2.1.30 Ability to locate and operate components, including local controls.	Tier #	2
	Group #	2
	K/A #	**286000 G2.1.30**
	Rating	4.4

Question 65

A fire has been reported at Transformer BOP-14.

The CRO manually starts the Motor Driven Fire Pump from H13-P862.

Then an Auto Deluge operates causing fire water header pressure to drop to 115 psig for 3 seconds and stabilize at 120 psig.

Fire Brigade Leader reports fire is out and the deluge is isolated.

Which of the following describes the requirements to place the Fire Water Pumps back in Standby?

A. Manually shutdown the Diesel Driven pump ‘A’ and the Motor-driven pump.

B. Verify all pumps automatically shut down after header pressure is stable at > 130 psig.

C. Manually shutdown the Diesel Driven pumps ‘A’ and ‘B’, the Motor-driven pump will automatically stop after 7 minutes.

D. Manually shutdown the Diesel Driven pumps ‘A’ and the Motor-driven pump will automatically stop after 7 minutes.

Answer: A
Explanation: If the motor driven pump is started from the control room it must be manually stopped, anytime a diesel driven pump starts it must be manually stopped. A is correct B is wrong. If the motor driven pump is started from the control room it must be manually stopped, anytime a diesel driven pump starts it must be manually stopped.

C is wrong. If the motor driven pump is started from the control room it must be manually stopped, and the Diesel Driven ‘B’ pump did NOT receive a start signal. ‘B’ pump auto starts at 117 psig + 5 seconds

D is wrong. If the motor driven pump is started from the control room it must be manually stopped, anytime a diesel driven pump starts it must be manually stopped.

Technical References:

04-S-01-P64-1, 4.2 & 4.4
GLP-OPS-P6400

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-P6400, Objective 5.0

Question Source:	Bank # 244,	2015 Audit
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(4)	

Examination Outline Cross-Reference	Level	RO
2.1.4 Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, “no-solo” operation, maintenance of active license status, 10CFR55, etc.	**Tier #**	3
	Group #	
	K/A #	**G2.1.4**
	Rating	3.3

Question 66

Which of the following would constitute a change in license status such that notification of the NRC would be required regarding a Licensed Operator?

A. A Licensed operator is accused of committing a felony offense by local authorities.

B. A Licensed operator is arrested in another state on a charge of DUI while on his seven day off period.

C. A Licensed operator has a broken wrist on his writing hand, and was prescribed pain medication for only 2 weeks.

D. A Licensed operator has an operation to install a heart pacemaker.

Answer: D
Explanation: 10CFR55.25 If, during the term of the license, the licensee develops a permanent physical or mental condition that causes the licensee to fail to meet the requirements of § 55.21 of this part, the facility licensee shall notify the Commission, within 30 days of learning of the diagnosis, in accordance with § 50.74(c). For conditions for which a conditional license (as described in § 55.33(b) of this part) is requested, the facility licensee shall provide medical certification on Form NRC 396 to the Commission (as described in § 55.23 of this part). A is wrong. 10CFR55.53g states: The licensee shall notify the Commission within 30 days about a conviction for a felony, conviction not arrest. B is wrong. 10CFR55.53j states. (j) The licensee shall not consume or ingest alcoholic beverages within the protected area of power reactors, within the protected area not on his 7 day off. C is wrong. A broken wrist does not prevent an operator from performing his license duties and prescription drugs must be prescribed for >30 days. D is correct

Technical References: 10CFR55		
References to be provided to applicants during exam: NONE		
Learning Objective: Document learning objective if possible. GLP-OPS-CFR01, Objective 30		
Question Source:	Bank # GGNS-OPS-00055	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	Tier #	3
	Group #	
	K/A #	**G2.1.7**
	Rating	4.4

Question 67

The plant is operating at 65% power.

A retest is being performed on ‘A’ Main Turbine Stop valve.

When the ‘A’ Main Turbine Stop valve closes the pressure control system malfunctions and causes 30 psig oscillations in reactor pressure and reactor power is oscillating between 58% to 72%.

Which of the following is the action that should be taken?

A. Immediately reduce core flow to 70 Mlbm/hr.

B. Insert control rods to stabilize power.

C. Place the Reactor Mode Switch to SHUTDOWN.

D. Immediately attempt to reopen the ‘A’ Main Stop valve.

Answer: C

Explanation:

05-1-02-V-21, Reactor Pressure Control Malfunctions, states the following: “IF APRM readings are OBSERVED to be OSCILLATING with sustained peak-to-peak swings of greater than 10% rated power. THEN IMMEDIATELY PLACE the Reactor mode Switch in the SHUTDOWN position.

This is plant site OE

A is wrong - Not an action to be performed with this type of failure.

B is wrong - Not an action to be performed with this type of failure.

C is correct

D is wrong - Not an action to be performed with this type of failure

Technical References: 05-1-02-V-21, Reactor Pressure Control Malfunctions		
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	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
2.2.1 Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.	Tier #	3
	Group #	
	K/A #	**G2.2.1**
	Rating	4.5

Question 68

The plant is performing a plant startup after a refueling outage.

Which of the following is performed just prior to placing the Reactor Mode Switch to STARTUP, due to the effects on reactivity?

A. Open MSIVs.

B. Secure RHR Shutdown Cooling.

C. Realign the Reactor Head vent valves.

D. Align RWCU system for blowdown operations.

Answer: B
Explanation: 03-1-01-1, Cold Shutdown to Generator Carrying Minimum Load Reactor Startup / Hot Shutdown, States in step 29 of Attachment VI page 6 of 13 to Verify Shutdown cooling mode secured. A is wrong – This could have an effect on reactivity only at pressure and could be performed prior to or after mode switch is placed in STARTUP. B is correct C is wrong – This is done long before mode switch is placed in STARTUP, and it has no effect on reactivity during a cold startup due to no steam D is wrong – This is done to control level but usually is done prior to startup but not used until actual heat is being produced by the reactor.
Technical References: 03-1-01-1, Cold Shutdown to Generator Carrying Minimum Load

References to be provided to applicants during exam: NONE		
Learning Objective: Document learning objective if possible. GLP-OPS-IOI01		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
2.2.13 Knowledge of tagging and clearance procedures.	Tier #	3
	Group #	
	K/A #	**G2.2.13**
	Rating	4.1

Question 69

Which of the following complies with EN-OP-102 when using Motor Operated Valves (MOVs) as a tagout boundary?

A. Control Switch - Danger tagged CLOSED
MOV Breaker - Danger tagged OPEN
MOV handwheel - Danger Tagged CLOSED

B. Control Switch - Danger Tagged CLOSED
MOV Breaker - Test & Maintenance tagged CLOSED
MOV handwheel - Danger Tagged CLOSED

C. Control Switch - Danger Tagged CLOSED
MOV Breaker - Danger Tagged OPEN
MOV handwheel - Test & Maintenance tagged CLOSED

D. Control Switch - Danger tagged CLOSED
MOV Breaker - Danger tagged CLOSED
MOV handwheel - Danger Tagged CLOSED

Answer: A

Explanation:

EN-OP-102, Attachment 9.2, step 6.2.1,

6.2.1 MOVs Used in Tagout Boundaries.

- Control Switch - Will be tagged to state the required position of the valve for the isolation. Removal or a Temporary Lift of the tag must be performed if a change of the tagged valve position is required.
- Breaker - Danger Tagged.
- Handwheel operator - Danger Tags hung on MOVs for isolation should only be hung on the handwheel. If a change of a tagged MOV's position is required, then Removal or Temporary Lift of the Danger Tag must be performed.

A is correct.

B is wrong – Breakers are to be tagged open not closed and use danger tags not T&M

C is wrong – use danger tags not T&M

D is wrong – Breakers are to be tagged open not closed.

Technical References:

EN-OP-102, Attachment 9.2, step 6.2.1,

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

FLP-OPS-ESOMS Objective: 3

Question Source:	Bank # 674	2013 Audit X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
2.2.21 Knowledge of pre- and post-maintenance operability requirements.	Tier #	3
	Group #	
	K/A #	**G2.2.21**
	Rating	2.9

Question 70

Which of the following describes the actions to be taken whenever any ECCS pump breaker has been racked out and is now racked back in to ensure breaker operability?

A. Perform electrical lineup checksheet per SOI, Attachment III, ONLY

B. Perform monthly surveillance on system.

C. Perform quarterly surveillance on pump only.

D. The associated pump must be started.

Answer: D

Explanation:

Per 04-1-01-E12, E21, E22-1, E12 SOI, E21 SOI and E22 SOI the pump must be started to ensure breaker operability.

A is wrong – Not required but plausible due to the pump breaker is listed on Att. III

B is wrong – Not required if pump breaker is only racked out, the monthly only verifies valve / system lineup.

C is wrong – Not required if pump breaker is only racked out

D is correct

Technical References:

04-1-01-E12, E21, E22-1

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-E1200, E2100, E2200 Precaution and limitations.		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
2.2.35 Ability to determine Technical Specification Mode of Operation.	Tier #	3
	Group #	
	K/A #	**G2.2.35**
	Rating	3.6

Question 71

Average reactor coolant temperature is 220°F.

Which of the following is/are the possible MODE(s) of operation that the plant could be operating in at this temperature?

A. 2

B. 3

C. 2 or 3

D. 1, 2, or 3

Answer: C
Explanation: See TS Table 1.1-1, Modes of Operation. By definition, the plant would be in MODE 3 if the Mode Switch were in Shutdown and temperature were >200 F. Per IOI-1, during a plant pressurization, the plant remains in MODE 2 (MS in Startup) while the plant heats up and pressurizes to near rated pressure, after which the MS is transferred to RUN (MODE 1). By the time MODE 1 is reached, reactor coolant temperature is well above 200 F. 'C' is correct All distracters are plausible because they each represent a possible MODE and the need for the Applicant to firmly grasp/apply MODE definitions in order to eliminate them as potential answers. 'D' is wrong because Mode 1 must be verified at >850 psig which is well above 220°F
Technical References: Tech Spec Table 1.1-1, MODES

References to be provided to applicants during exam: NONE		
Learning Objective: Document learning objective if possible. GLP-OPS-TS001, Objective 5.		
Question Source:	Bank # 355	2015 AUDIT
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2011 NRC
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions.	**Tier #**	3
	Group #	
	K/A #	**G2.3.4**
	Rating	3.2

Question 72

Which of the following Federal Occupational Dose limits would be exceeded if you received 11 Rem/Year in that area?

A. Skin of any Extremity

B. Whole-body

C. Lens of the eye

D. Internal organs

Answer: B

Explanation:

All distractors are plausible due to all being limits found in 10CFR20.

A is wrong, the limit for the skin of the whole body or skin of any extremity is 50 Rem/Year

B is correct, the limit for whole body is 5 Rem/year

C is wrong, the limit for Lens of the eye is 15 Rem/Year

D is wrong, the limit for internal organs is 50 Rem/Year

Technical References:

10CFR20

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

ENS Generic RadWorker Training, Objective 4.1

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #258	2015 Audit
	New	
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(12)	

Examination Outline Cross-Reference	Level	RO
2.3.12 Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.	**Tier #**	3
	Group #	
	K/A #	**G2.3.12**
	Rating	3.2

Question 73

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. Radiation Protection Manager

B. Plant Operations General Manager

C. Radiation Protection Supervisor

D. Operations Manager

Answer: A
Explanation: See EN-RP-101, Section 5.5[10], 6th bullet. Per the above reference, the RPM's approval is required. 'A' is correct Distracters are all as plausible, to the SRO 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, Section 5.5[10], 6th bullet
References to be provided to applicants during exam: NONE
Learning Objective: Document learning objective if possible.

GLP-OPS-PROC Obj 50		
Question Source:	Bank # 156	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2015 Q #72
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(12)	

Examination Outline Cross-Reference	Level	RO
2.4.3 Ability to identify post-accident instrumentation.	**Tier #**	3
	Group #	
	K/A #	**G2.4.3**
	Rating	3.7

Question 74

Which of the following indications is considered to be “Post Accident” instrumentation in the Main Control Room?

A. Reactor Vessel Water Level – H13-P680 Wide Range

B. Reactor Vessel Pressure – H13-P680 Wide Range

C. Standby Liquid Control Tank Level – H13-P601

D. Containment Pressure – H13-P870 Narrow Range

Answer: D
Explanation: Per Tech Spec 3.3.3.1 Post Accident Monitoring Instrumentation and 06-OP-1M71-M-0003 Post Accident Monitoring Instrumentation Channel Check, Containment Pressure is required using both wide and Narrow ranges. A is wrong – Reactor vessel water level wide range instrumentation is required but only on the P601 panel not the P680. B is wrong - Reactor vessel pressure wide range instrumentation is required but only on the P601 panel not the P680. C is wrong – SLC tank level is listed in tech specs but not listed as a post-accident indication. D is correct – Containment pressure requires both Narrow range and Wide Range.
Technical References: Per Tech Spec 3.3.3.1 Post Accident Monitoring Instrumentation 06-OP-1M71-M-0003 Post Accident Monitoring Instrumentation Channel Check
References to be provided to applicants during exam: NONE

Learning Objective: Document learning objective if possible. GLP-OPS-TS001		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
2.4.20 Knowledge of the operational implications of EOP warnings, cautions, and notes.	Tier #	3
	Group #	
	K/A #	**G2.4.20**
	Rating	3.8

Question 75

Per 05-S-01-EP-1, Emergency/Severe Accident Procedure Support Documents, Caution number 2 states, “Suppression Pool temperature must be determined by alternate means when SP level is below _________ ft.”

A. 10.5 ft.

B. 14.25 ft.

C. 14.5 ft.

D. 17.5 Ft.

Answer: B
Explanation: Per 05-S-01-EP-1, Emergency/Severe Accident Procedure Support Documents, Cautions number 2 states, “Low SP level may uncover SP temperature elements. SP temperature must be determined by alternate means when SP level is below 14.25 ft.” A is wrong – but plausible because this is the required SP level before an emergency depressurization can be performed. B is correct. C is wrong – but plausible because this is the level in which an ED is required and also the minimum level for NPSH for ECCS pumps D is wrong – but plausible because this is the level that will automatically initiate Supp Pool Makeup System if a LOCA is occurring.
Technical References: 05-S-01-EP-1, Emergency/Severe Accident Procedure Support Documents
References to be provided to applicants during exam:

NONE		
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	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.41(b)(10)	

GGNS LOT 2017 NRC INITIAL LICENSED OPERATOR WRITTEN EXAMINATION
OPEN-REFERENCES TABLE OF CONTENTS
RO EXAM

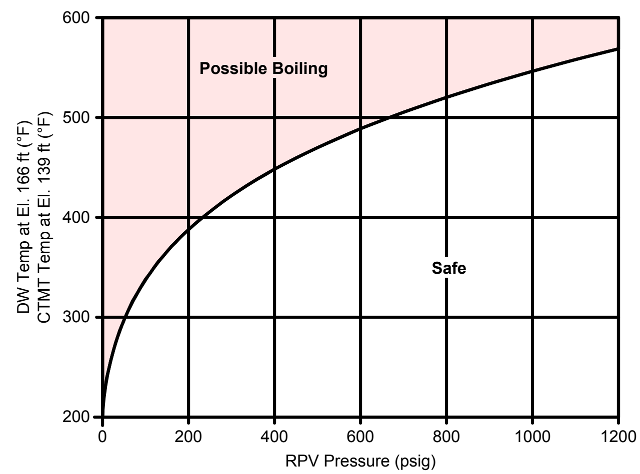
TAB 1

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)

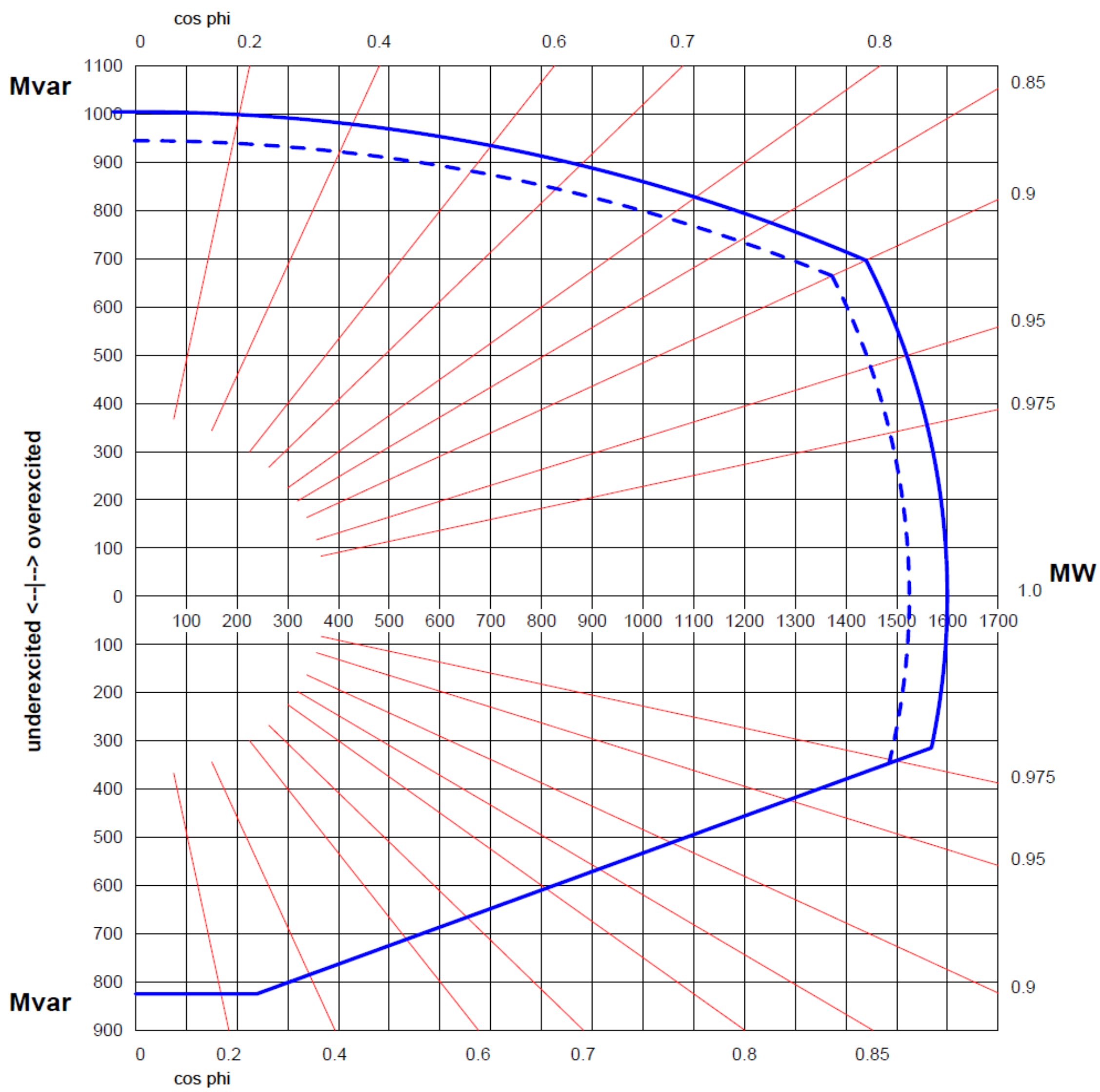


2. Do <u>not</u> use any RPV level instrument that exceeds <u>both</u> limits in the following table:

Level Instrument	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

TAB 2

Generator Type		**THFF 180/76-18**			
		uprate (——)		**prev. (- - -)**	
Apparent Power	S	1600.00	MVA	1525.00	MVA
Armature Voltage	U	22.00	kV		
Armature Current	I	41.989	kA	40.021	kA
Frequency	f	60.0	Hz		
Power Factor	P.F.	0.900		0.900	
H2-Pressure (gauge)	pe	5.170	bar	4.140	bar
Cold Gas Temperature	Tk	40.0	°C	40.0	°C



TAB 3

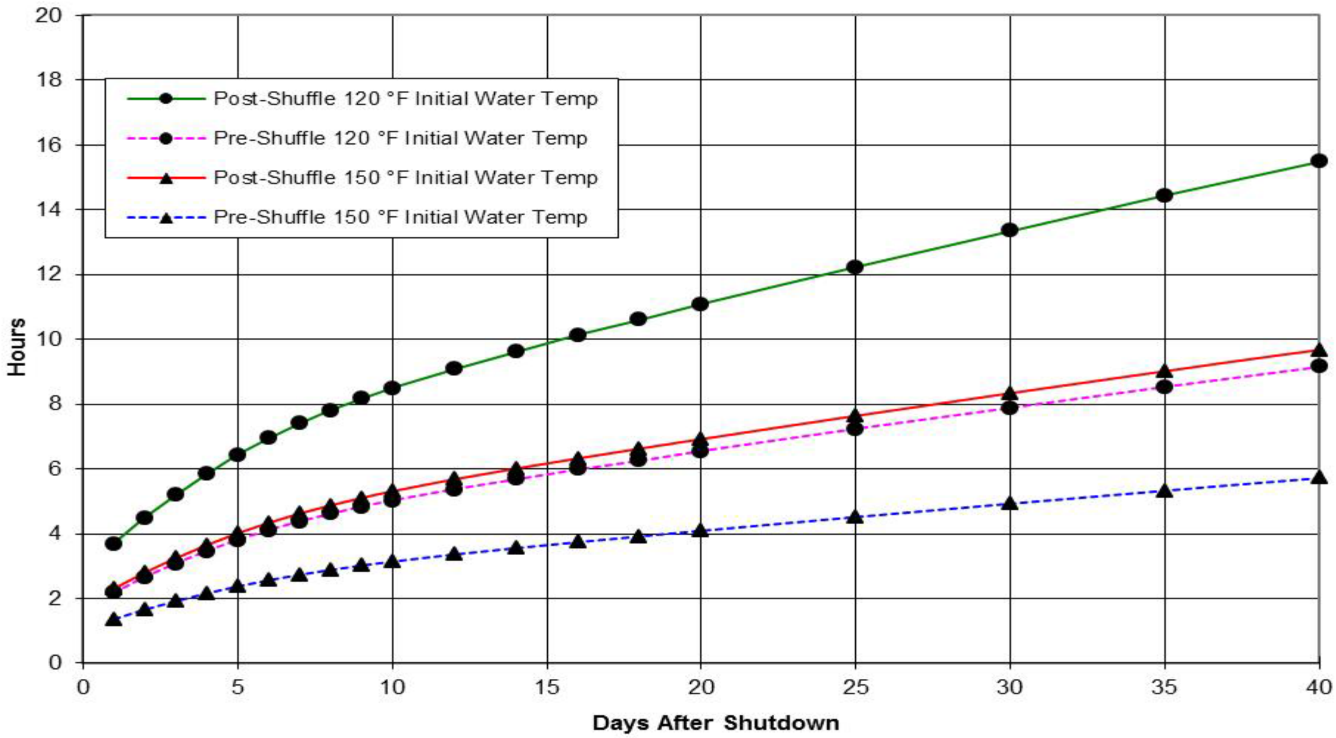


Figure 2: Time for Reactor Cavity to Reach 200°F for Initial High Water

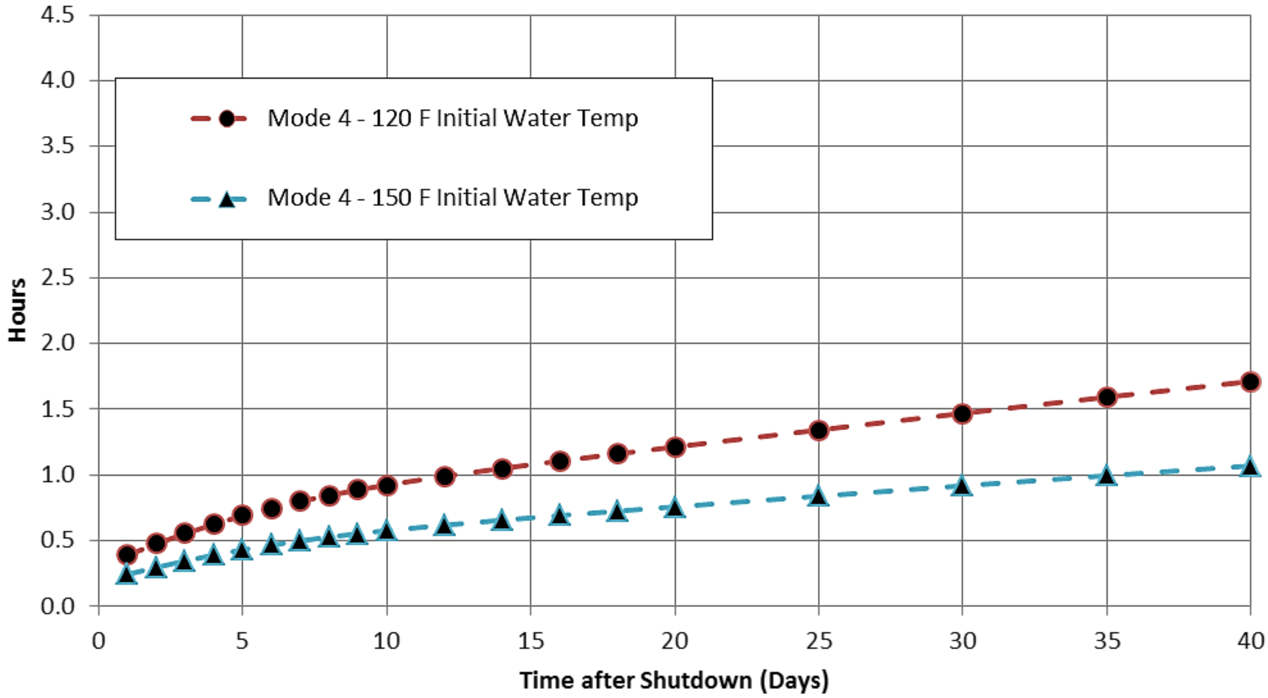


Figure 3: Time for Reactor Vessel to reach 200°F from Normal Water level 32", Mode 4, zero to forty days (*Assumes Cycle 21 EOC decay heat load*)

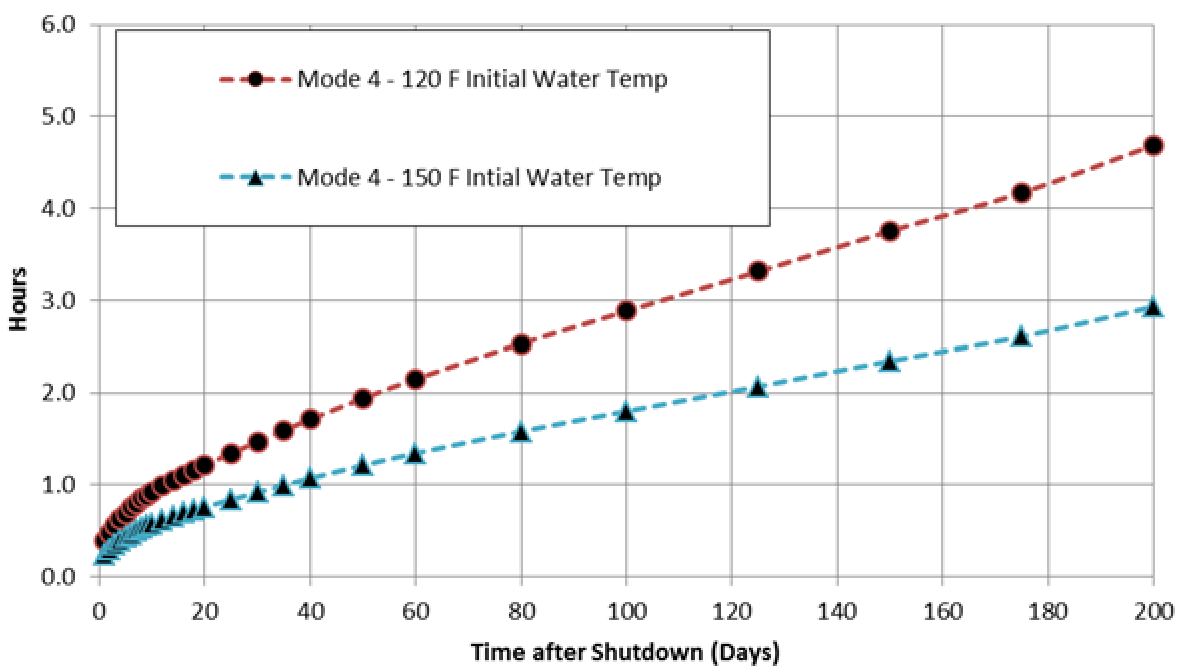


Figure 4: Time for Reactor Vessel to reach 200°F from Normal Water level 32", Mode 4, zero to two hundred days (*Assumes Cycle 21 EOC decay heat load*)

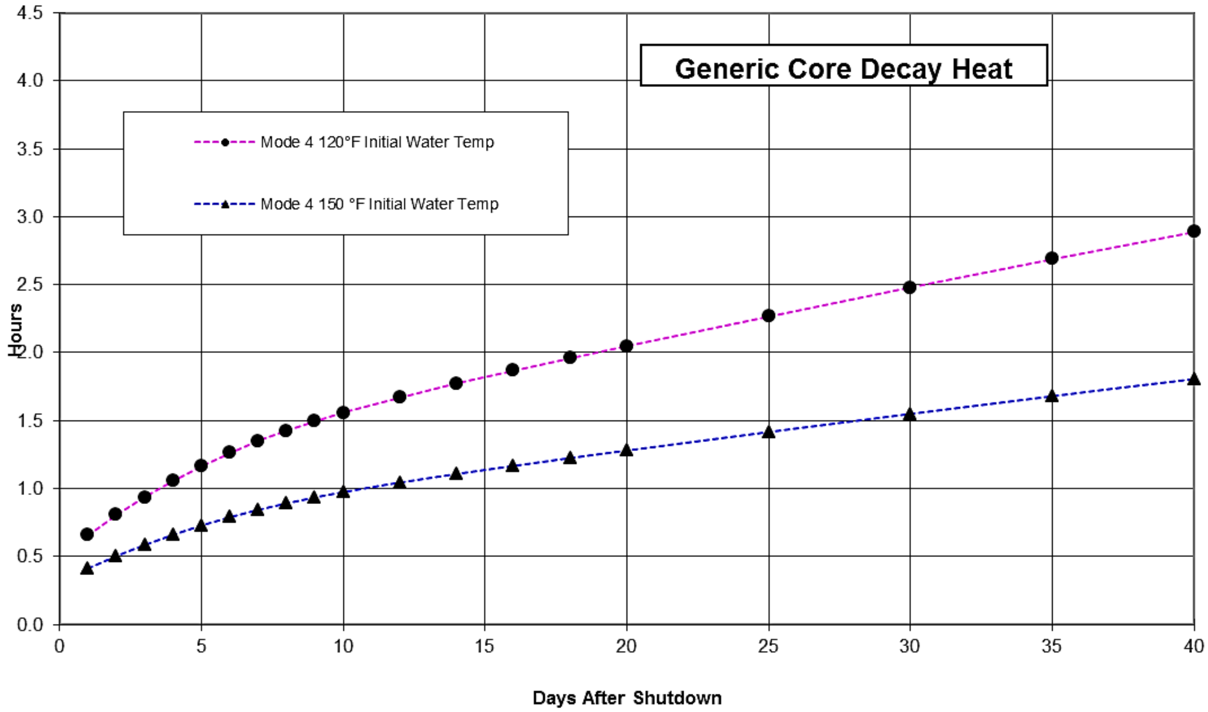


Figure 5: Time for Reactor Vessel to Reach 200°F for Initial Water Level 24" Below Vessel Flange, Mode 4, zero to forty days.
(*Assumes Cycle 21 EOC decay heat load*)

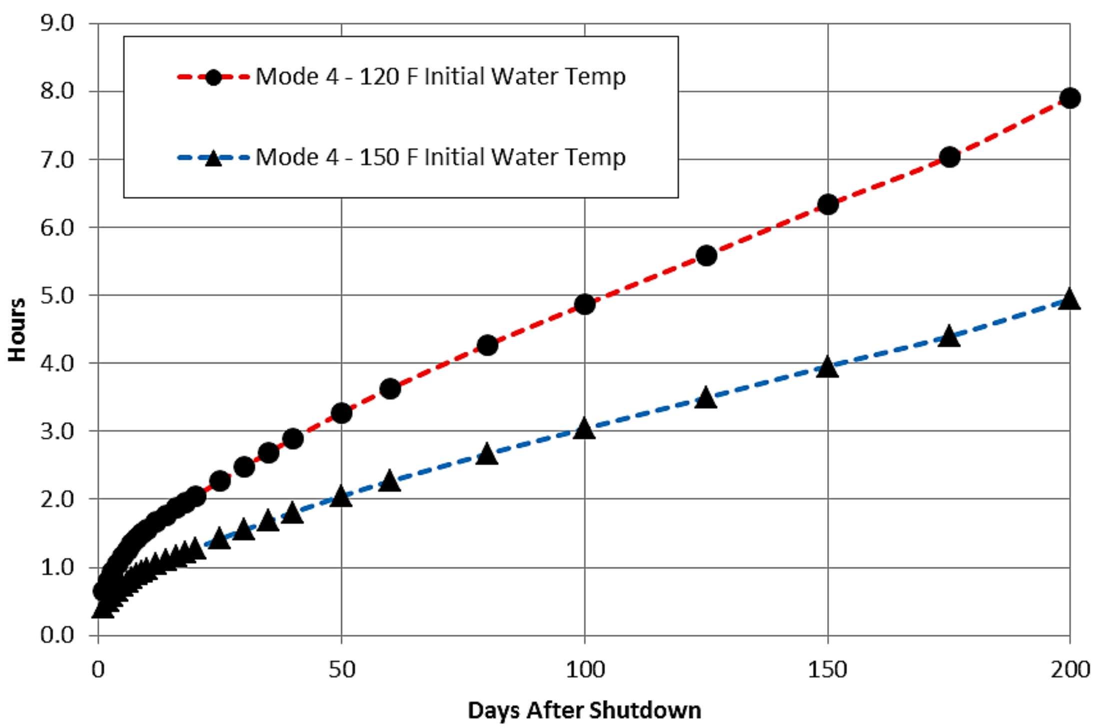


Figure 6: Time for Reactor Vessel to Reach 200°F for Initial Water Level 24" Below Vessel Flange, Mode 4, zero to two hundred days
(*Assumes Cycle 21 EOC decay heat load*)

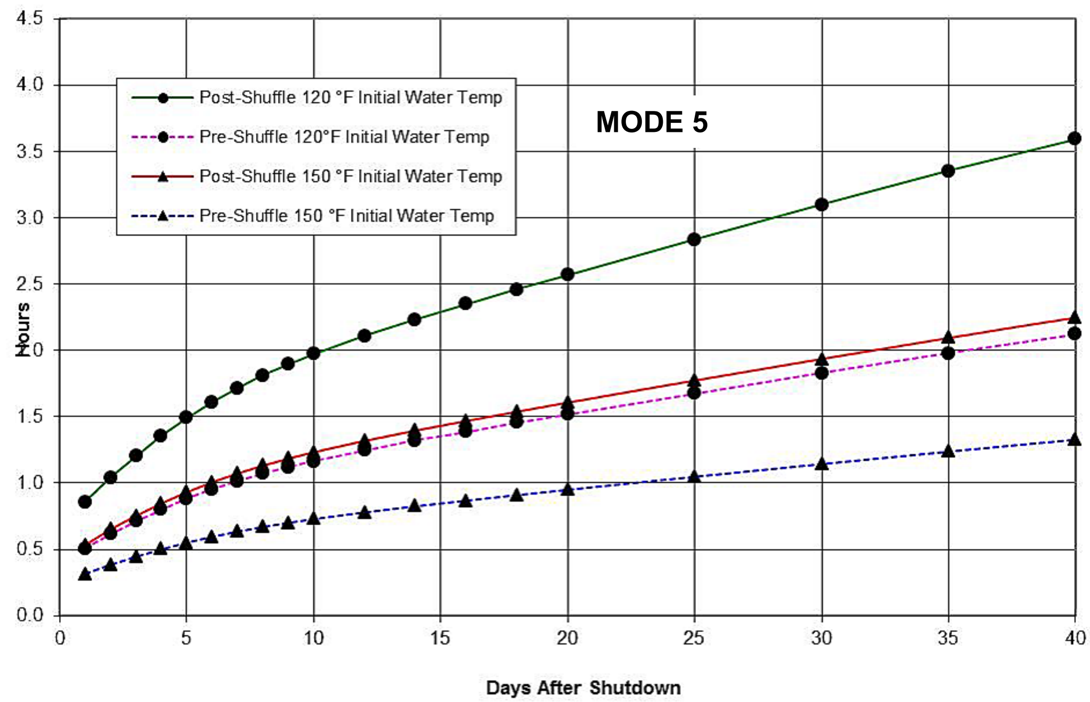


Figure 7: Time for Reactor Vessel to Reach 200°F for Initial Water Level 24" Below Vessel Flange, Mode 5, zero to forty days

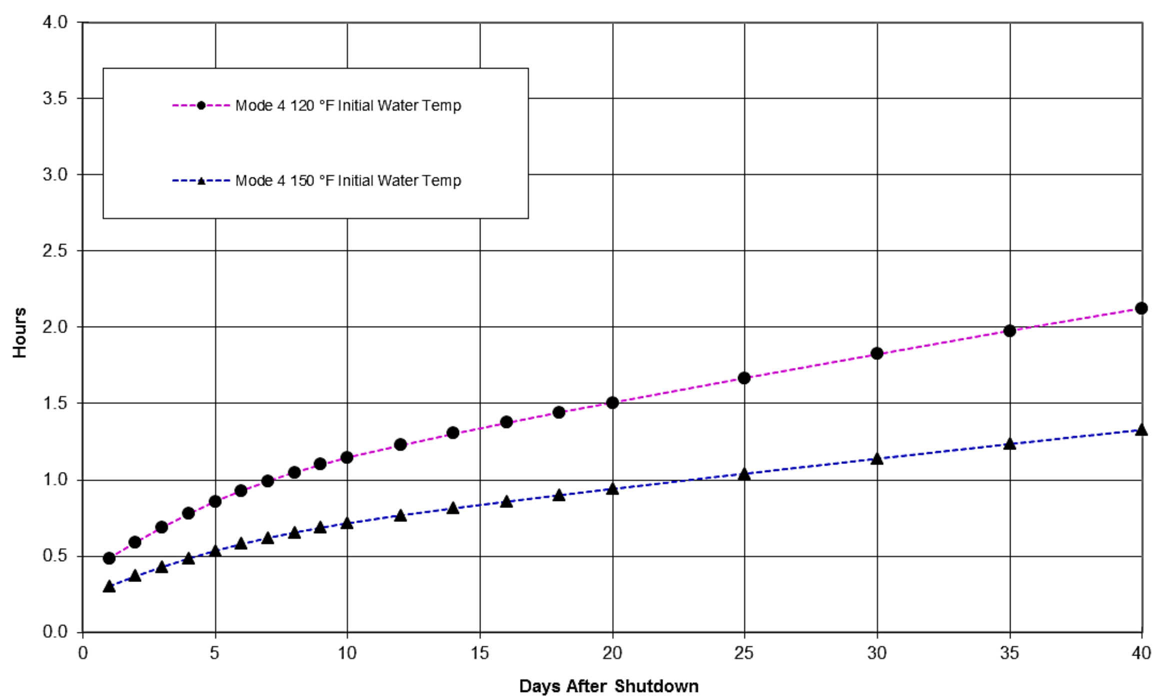


Figure 8: Time for Reactor Vessel to Reach 200°F for Initial Water Level 12" Below Main Steam Line, Mode 4, zero to forty days
(*Assumes Cycle 21 EOC decay heat load*)

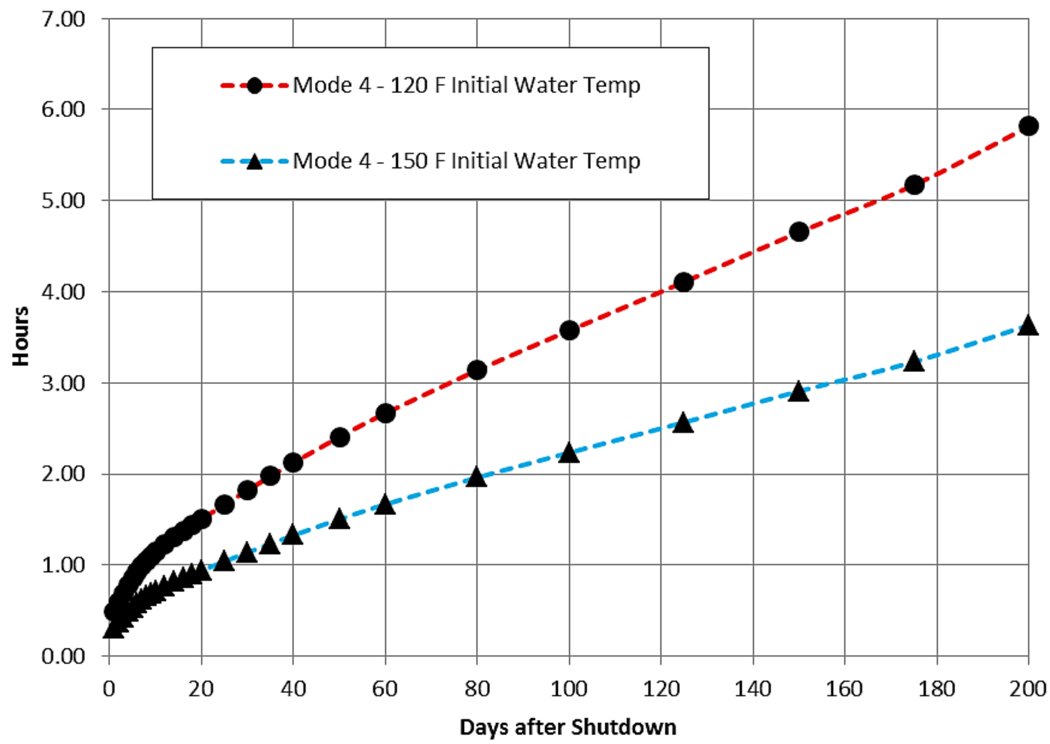


Figure 9: Time for Reactor Vessel to Reach 200°F for Initial Water Level 12" Below Main Steam Line, Mode 4, zero to two hundred days
(*Assumes Cycle 21 EOC decay heat load*)

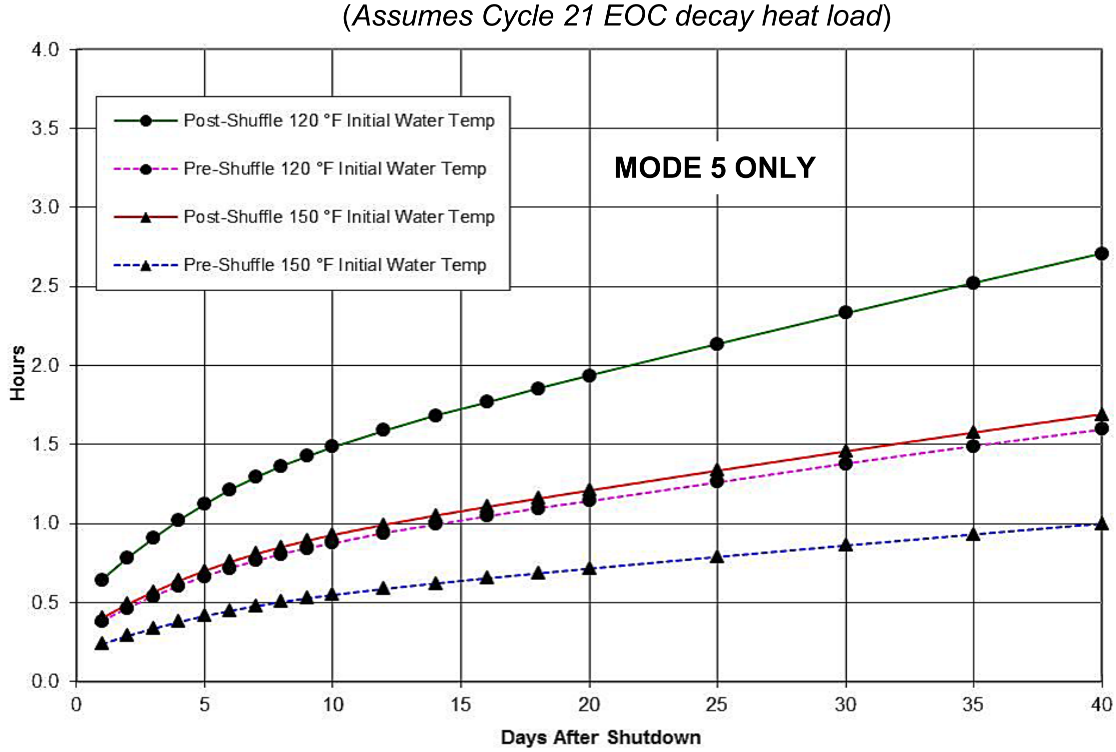


Figure 10: Time for Reactor Vessel to Reach 200°F for Initial Water Level 12" Below Main Steam Line, Mode 5, zero to forty days

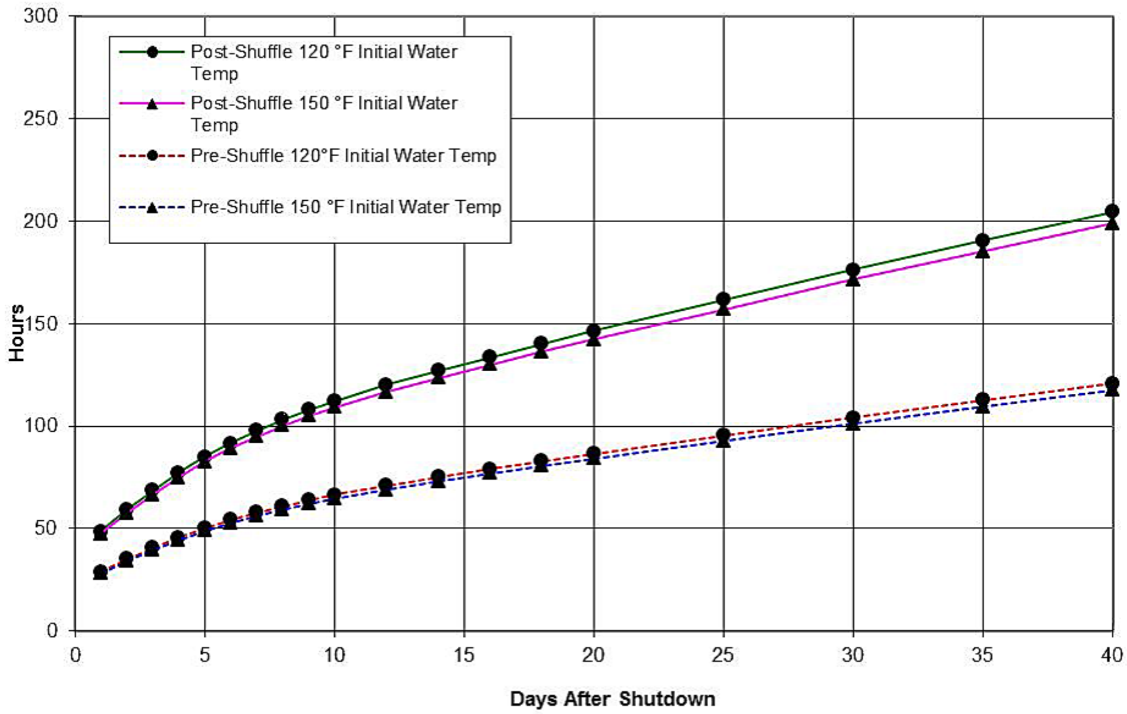


Figure 11: Time to TAF from High Water Level, zero to forty days

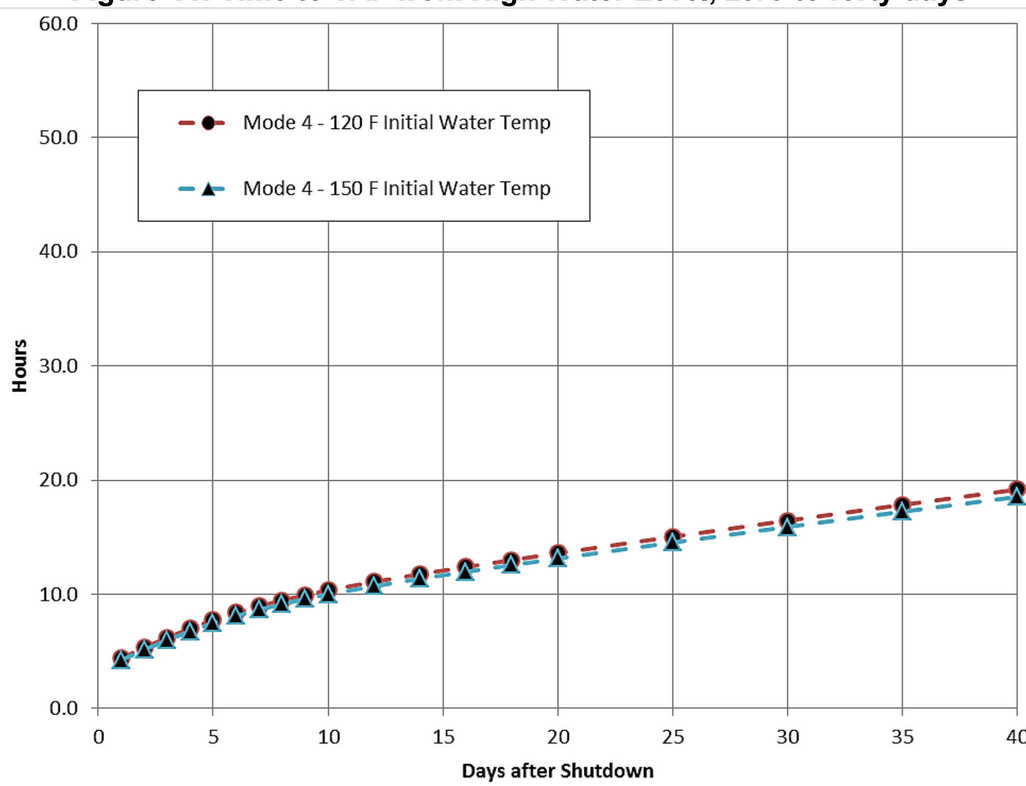


Figure 12: Time to TAF from 32" Normal Water Level, Mode 4, zero to forty days
(*Assumes Cycle 21 EOC decay heat load*)

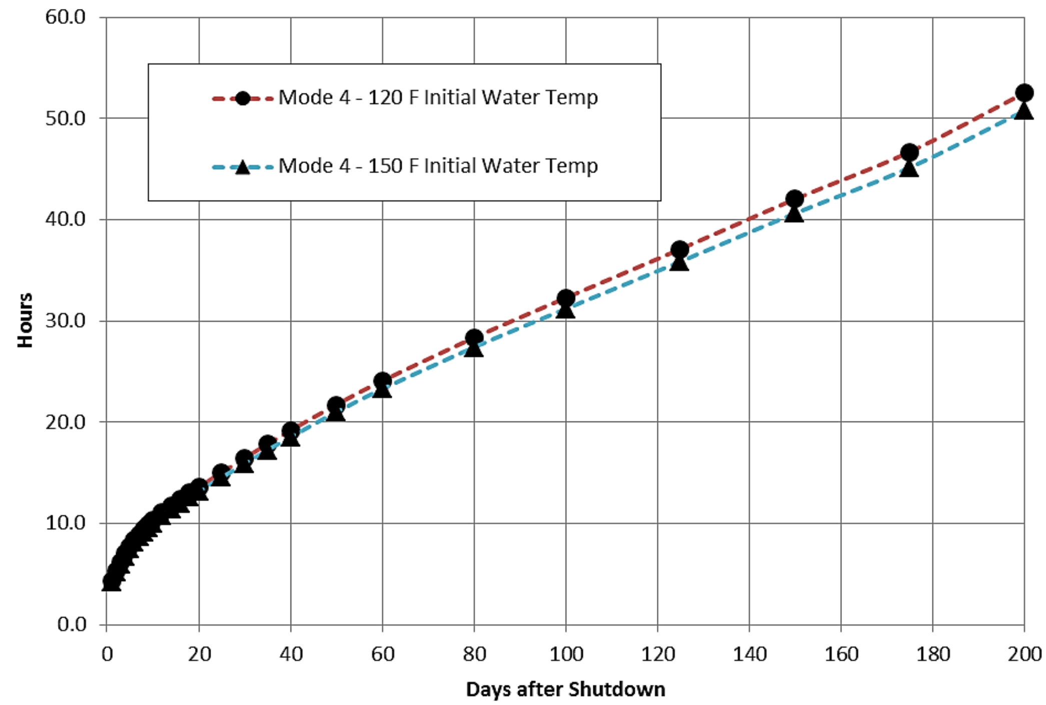


Figure 13: Time to TAF from 32" Normal Water Level, Mode 4, zero to two hundred days (*Assumes Cycle 21 EOC decay heat load*)

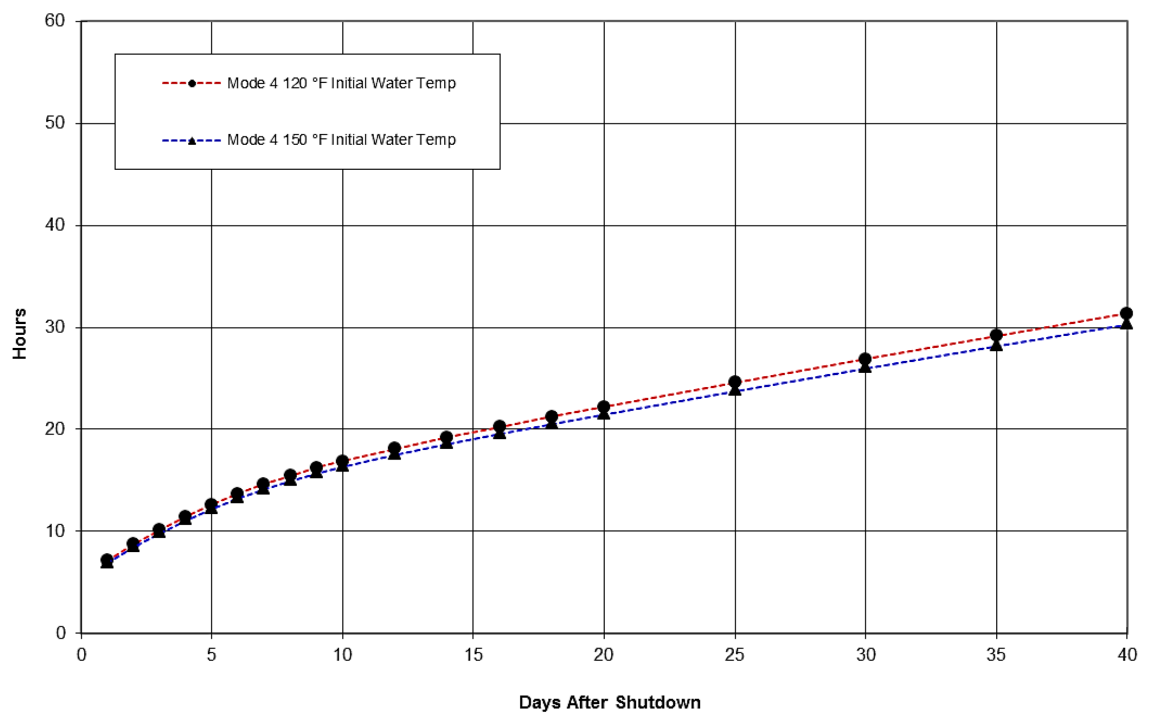


Figure 14: Time to TAF from 24" Below Reactor Vessel Flange, Mode 4, zero to forty days (*Assumes Cycle 21 EOC decay heat load*)

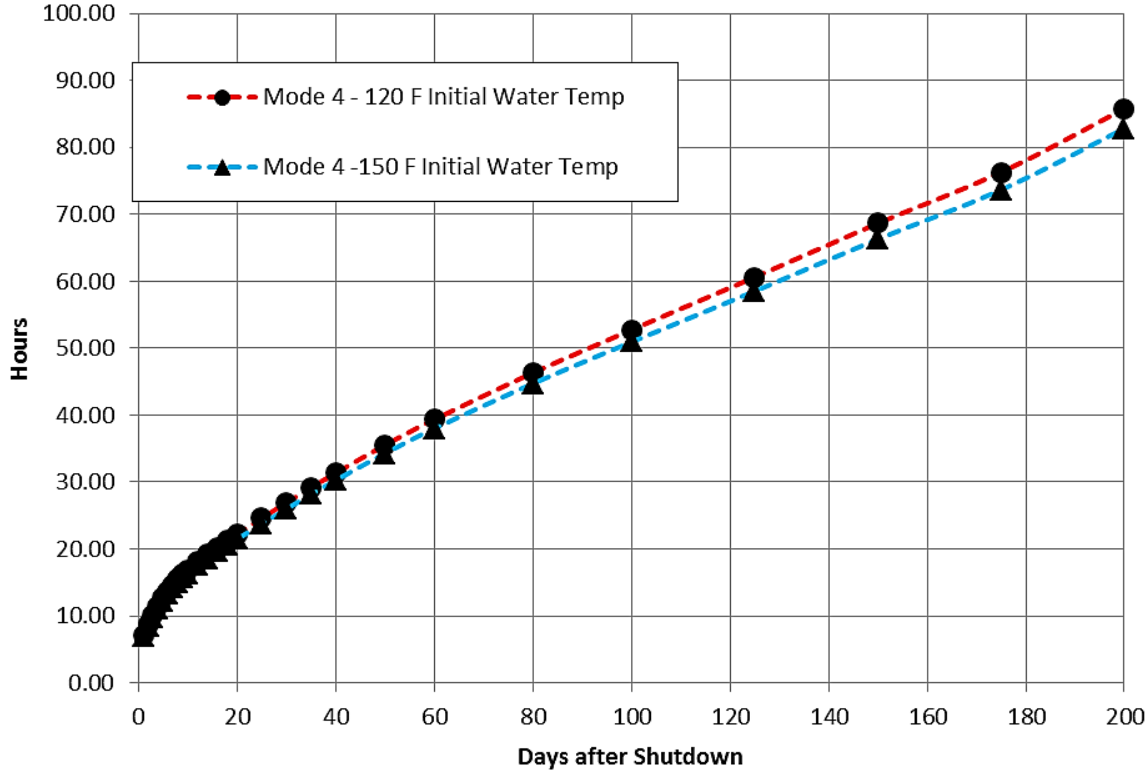


Figure 15: Time to TAF from 24” Below Reactor Vessel Flange, Mode 4, zero to two hundred days (*Assumes Cycle 21 EOC decay heat load*)

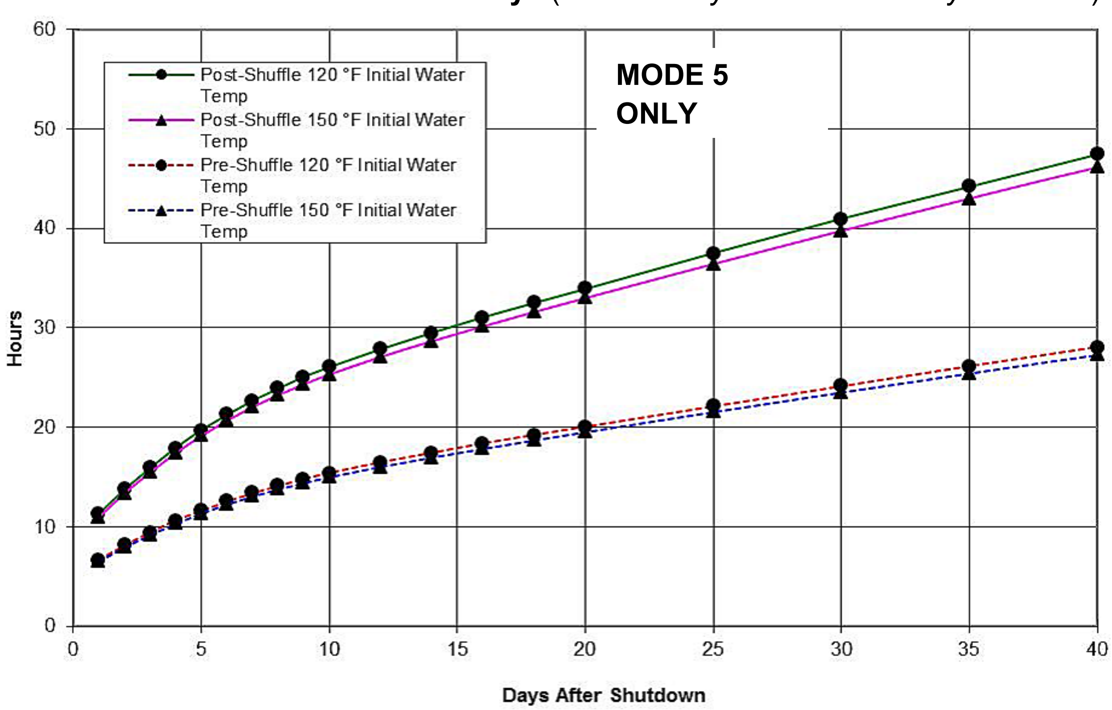


Figure 16: Time to TAF from 24” Below Reactor Vessel Flange, Mode 5, zero to forty days

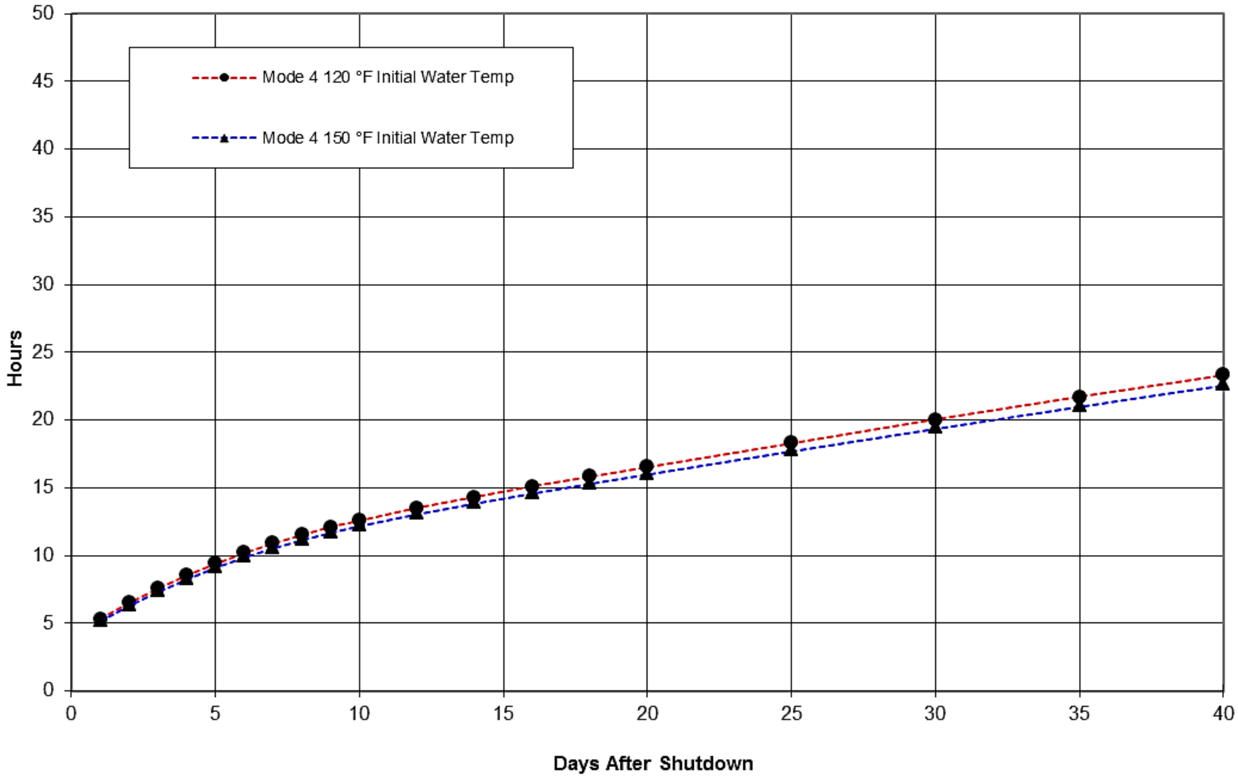


Figure 17: Time to TAF from 12" Below Main Steam Line, Mode 4, zero to forty days
(*Assumes Cycle 21 EOC decay heat load*)

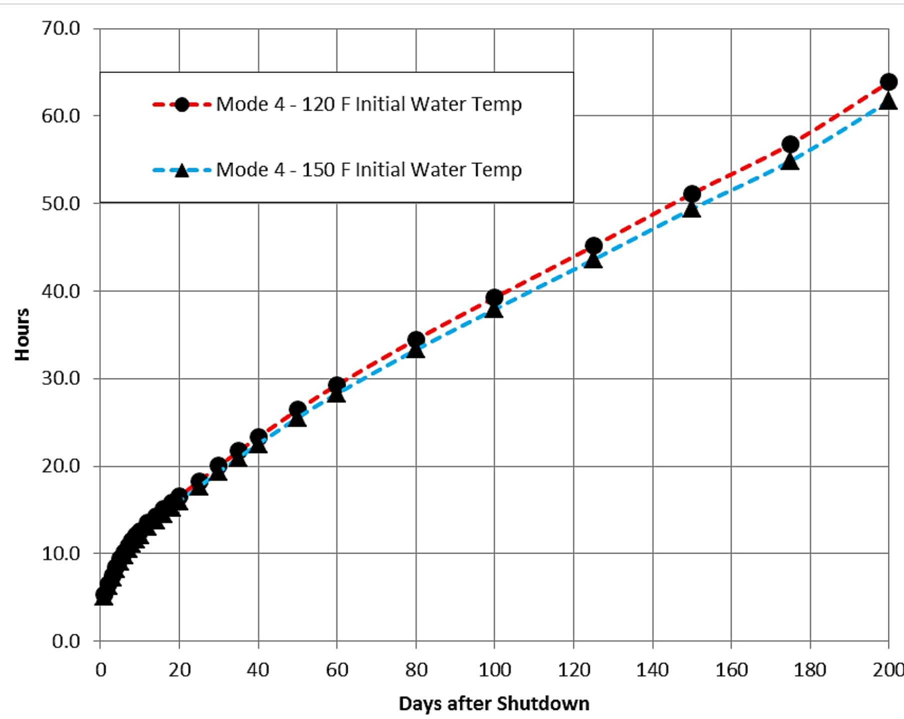


Figure 18: Time to TAF from 12" Below Main Steam Line, Mode 4, zero to two hundred days
(*Assumes Cycle 21 EOC decay heat load*)

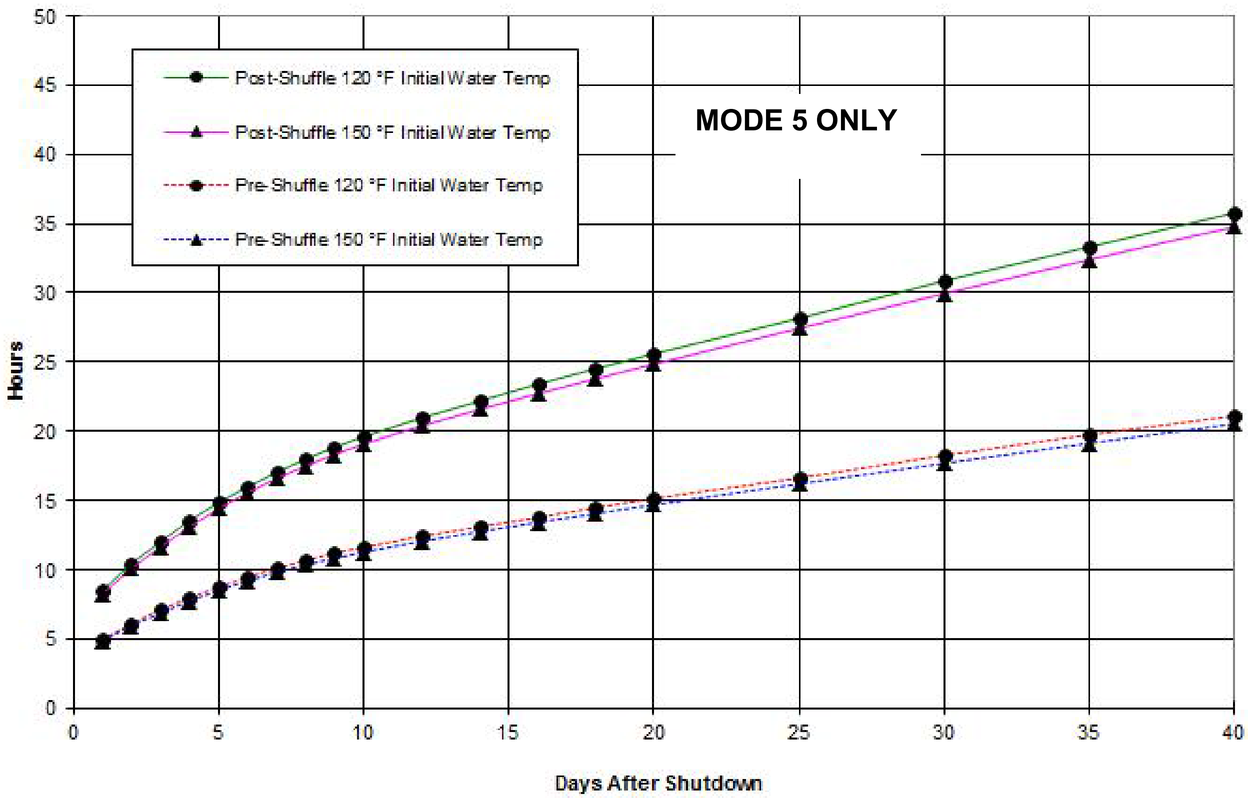
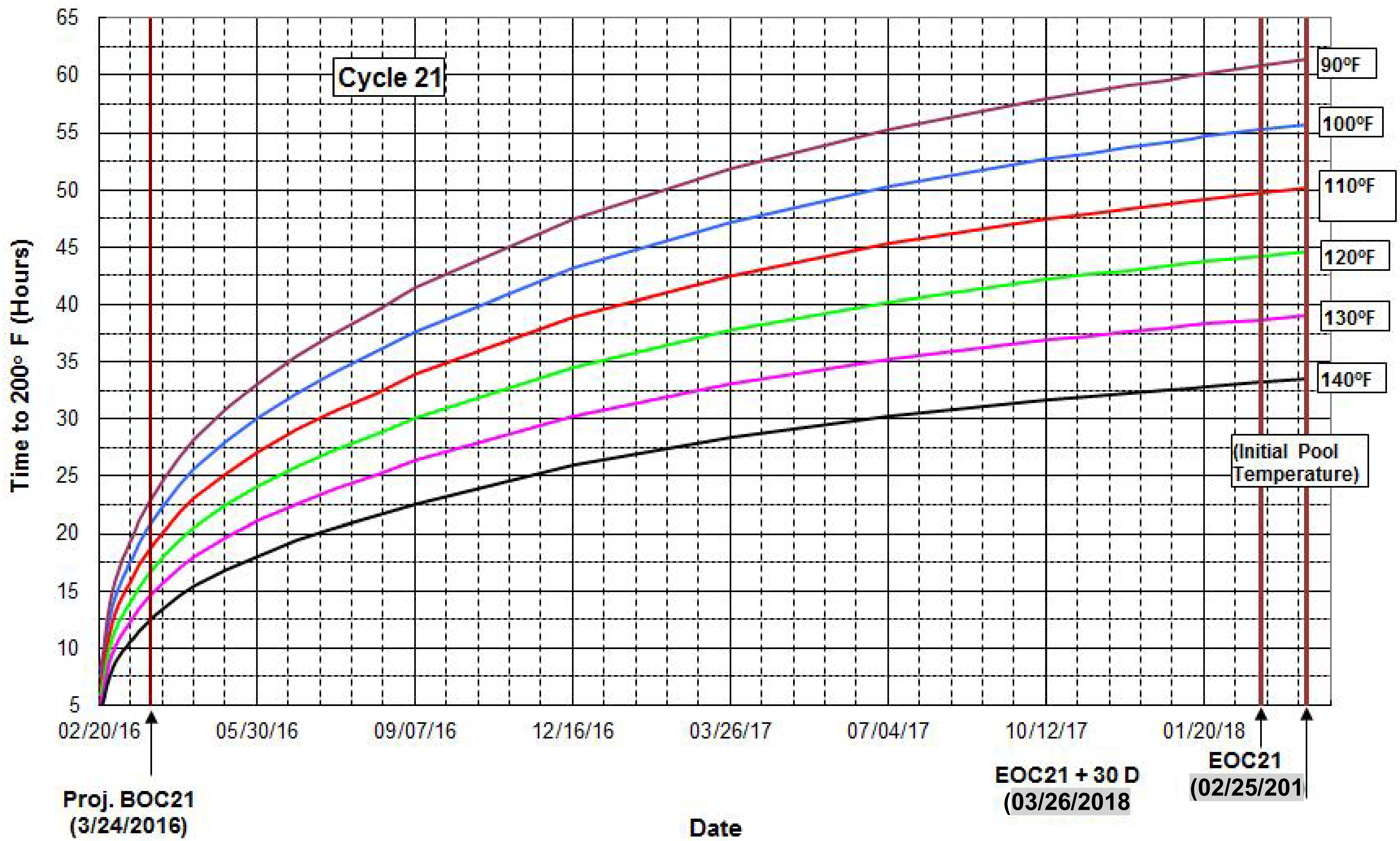


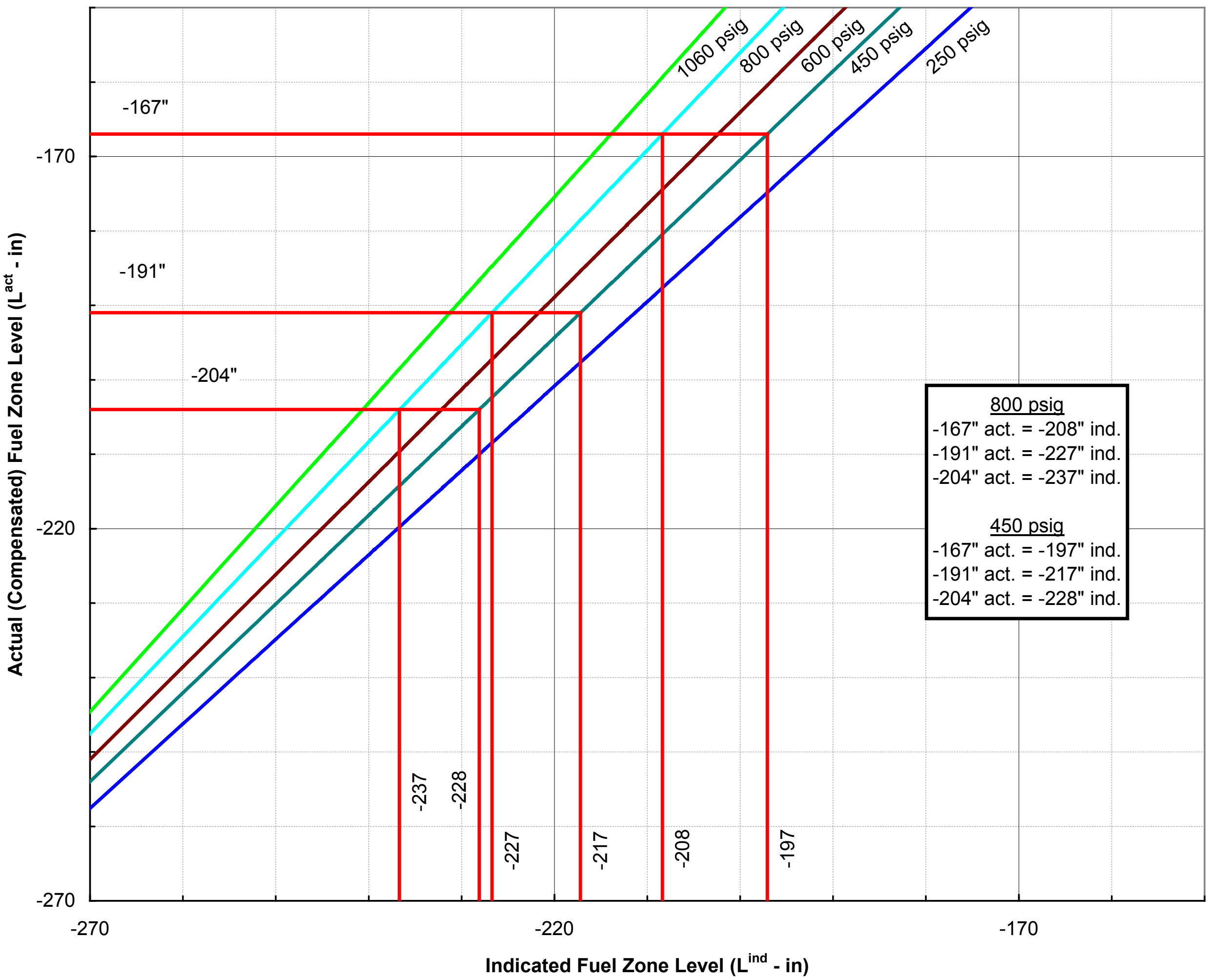
Figure 19: Time to TAF from 12" Below Main Steam Line, Mode 5, zero to forty days



Cycle 21
90ºF
100ºF
110ºF
120ºF
130ºF
140ºF
(Initial Pool Temperature)
Time to 200° F (Hours)
65
60
55
50
45
40
35
30
25
20
15
10
5
02/20/16
05/30/16
09/07/16
12/16/16
03/26/17
07/04/17
10/12/17
01/20/18
Proj. BOC21 (3/24/2016)
EOC21 + 30 D (03/26/2018
EOC21 (02/25/201
Date

TAB 4

Figure 1
Density Compensated Fuel Zone Nomograph



GGNS LOT 5/2017 NRC INITIAL LICENSED OPERATOR WRITTEN EXAMINATION

SRO EXAM

ANSWER KEY

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

Examination Outline Cross-Reference	Level	SRO
295003 - Partial or Complete Loss of AC Power **AA2.02 - Ability to determine and interpret the following as they apply to Partial or Complete Loss of AC Power:** **Reactor power / pressure / and level**	**Tier #**	1
	Group #	1
	K/A #	**295003 – AA2.02**
	Rating	4.3

Question 76

A LOCA is in progress.

RCIC has isolated on room temperature.

Neither Diesel-Driven Fire Pump will start.

Reactor water level is lowering at 20” per minute.

Which of the following would require entry into STEAM COOLING leg of EP-2?

A. Station Blackout

B. Loss of Offsite Power and ALL onsite AC power

C. Loss of all high and low pressure ECCS

D. Only SLC and CRD are injecting

Answer: B
Explanation: With RCIC isolated and both fire pumps unavailable anything that would cause a loss of all other ways to feed the reactor would require an entry into steam cooling. A station blackout, described as a loss of offsite power and Div 1 and 2 D/Gs onsite power, Table 1 and Table 2 systems (including Firewater) per EP-2 are unavailable. However, HPCS should still be available. A loss of all offsite and onsite power would leave nothing available to inject into the core, both fire pumps are gone and with level lowering at 20”/min rate the FSG pumps would not be lined up in time. Therefore, Step L-8, Any Injection Subsystem lined up with a pump running? Would be NO. Step L-10, Any injection source lined up with a pump running (Table 1 and 2 systems) would be NO. Exit to steam cooling, at STM-4 When level drops to -204” Enter Level at 3 and Emergency Depressurize.

B is correct: Loss of all power (On and OFF) would render all Table 1 and 2 systems unavailable.

A is wrong. Station Blackout by definition is loss of offsite power and loss of Div 1 and 2 D/Gs, Div 3 would be considered still available with HPCS. Plausible is the student forgets the definition of “Station Blackout”

C is wrong. Table 2 systems are still available, plausible if the student forgets about table 2 systems are available and could be used.

D is wrong. CRD is a table 1 system and SLC is a table 2 system therefore, not allowed to wait to -204” FZ. Plausible if the student remembers the low flow volume of the two systems stated and doesn’t realize that is enough to satisfy the EOPs to call for an immediate ED.

Technical References:

EP-2
02-S-01-40, EP Technical Bases, Attachment IV, pages 21, 35, 43

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-EP02, Objective 22

Question Source:	Bank #283	2015 Audit
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.43(b)(5)	

Examination Outline Cross-Reference	Level	SRO
295023 - Refueling Accidents **2.4.4 - Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.**	**Tier #**	1
	Group #	1
	K/A #	**295023 – G2.4.4**
	Rating	4.7

Question 77

A spent fuel handling accident has occurred within the spent fuel pool causing damage to a fuel bundle.

As a result:

- Fuel Handling Area Ventilation Exhaust radiation levels have risen to 4.5 mR/hr and stabilized there
- Fuel Pool Sweep Ventilation Exhaust radiation levels have risen to 35 mR/hr and stabilized there

The Crew enters ONEP, HIGH RADIATION DURING FUEL HANDLING, suspends all refuel floor activities and evacuates affected areas.

Which of the following actions that should be taken by the CRS?

A. Start SGTS train
   Declare an Unusual Event

B. Verify SGTS initiation
   Declare an Alert

C. Enter EP-4
   Verify SGTS initiation
   Declare an Unusual Event

D. Enter EP-4
   Verify SGTS initiation
   Declare an Alert

Answer: D
Explanation:

The Fuel Pool Sweep Vent Exhaust rad level of 35 mR/hr and Fuel Handling Area of 4.5 is above the EP-4 entry condition of 30 mR/hr and 3.6 mR/hr (shown on Table SC-1 of the EP). Therefore, an EP-4 entry is required. Step 1 indicates that if FHA or FPS rad level are above the setpoints then verify SBGT initiation. However, once the EP has been entered, there is no other specific EP-4 actions required related to these vent exhaust rad levels. This is because Table SC-1 does not acknowledge "max safe" values for either of these two exhaust rad level conditions. Additionally, EP-4 has been entered solely because of a spent fuel handling accident (i.e., a situation of a unisolable primary system discharging outside the primary containment does not exist). The only "action" for CRS to do at this point is to monitor the Table SC-1area rad monitors for any rising trends. ONEP, HIGH RADIATION DURING FUEL HANDLING, is also entered and executed. Suspend all refueling floor activities on 208' ele. except as necessary to place fuel/components in a safe position, and evacuate affected areas. AA2, 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. EALs states, (2) A VALID alarm on any or the following radiation monitors due to damage to irradiated fuel or loss of water level. FH Area Vent (P601-19A-C11) This alarm is FH AREA VENT RAD HI-HI/INOP which its setpoint is 3.6 mR/hr. The SRO candidate should recognize that this alarm should already be in due to being above the setpoint A is wrong. SBGT should have auto started and an Alert is required. Plausible is the student forgets the auto start of the SBGT system and if they go to the wrong EAL B is wrong. EP-4 should be entered. Plausible is the student forgets that these two alarms are EP-4 entry conditions from the chart and not a listed entry condition. C is wrong. an Alert is required. D is correct.
Technical References: EP-4, Secondary Containment Control flowchart 02-S-01-40, EP Technical Bases, Attachment VII, pages 3 and 5 of 21 05-1-02-II-8, High Radiation During Fuel Handling ONEP 10-S-01-1 EAL Flow Charts
References to be provided to applicants during exam: EAL Flow Charts
Learning Objective: Document learning objective if possible. GLP-OPS-EP4 Obj 3

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:	55.43(b)(4)	

Examination Outline Cross-Reference	Level	SRO
295024 High Drywell Pressure **EA2 Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE:** **EA2.09 Containment pressure: Mark-III**	**Tier #**	1
	Group #	1
	K/A #	**295024 – EA2.09**
	Rating	4.1

Question 78

A LOCA has occurred.

When the crew placed the mode switch to shutdown an ATWS/LOCA occurred.

- Drywell pressure is 8.5 psig
- Containment pressure is 7.0 psig and rising
- Containment temperature is 145°F.
- Suppression pool level is at the Tech Spec high level limit.

Which of the following describes the actions required by the CRS?

A. Vent the Containment
   Emergency Depressurize immediately.

B. Initiate Containment Spray
   Emergency Depressurize when Containment pressure cannot be restored and maintained below 7.5 psig.

C. Vent the Containment
   Emergency Depressurize when Containment pressure cannot be restored and maintained below 7.5 psig.

D. Initiate Containment Spray
   Emergency Depressurize immediately

Answer: B
Explanation:

With the given information the SRO should determine that Containment venting is only allowed in the EOPs if Containment pressure reduction is required to Restore and maintain adequate core cooling OR Reduce total offsite dose. The next option is to initiate containment spray but it must be within the CSIPL curve. A CTMT temp of 145°F and CTMT pressure of 7.0 psig is in the safe to initiate zone. If CTMT pressure were to continue to rise to approximately 7.5 psig the unsafe zone of the PSP curve would be entered, however, EP-3 step PCP6 states IF CTMT pressure cannot be restored and maintained in the safe zone of PSP THEN Emergency Depressurize. Which means you can exceed the line on the graph if actions are being taken to restore back below.

A is wrong - as stated above venting the containment is not allowed at this time and even if the line is exceeded on the PSP curve ED is not required if it can be restored back below. Plausible if the student is confused on when the containment can be vented and per the EP Bases, the PSP curve can be exceeded if measures are being taken to restore back below the line.

B is correct

C is wrong - as stated above venting the containment is not allowed at this time. Plausible if the student is confused on when the containment can be vented

D is wrong - even if the line is exceeded on the PSP curve ED is not required if it can be restored back below. Plausibe if the student thinks they must ED when the PSP line is exceeded, per the EP Bases, the PSP curve can be exceeded if measures are being taken to restore back below the line.

Technical References:

EP-3
02-S-01-40 EP Technical Bases, Attachment VI pages 22, 23, and 25 of 37

References to be provided to applicants during exam:

PSP and CSIPL curves

Learning Objective: Document learning objective if possible.

GLP-OPS-EP03

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:	55.43(b)(5)	

Examination Outline Cross-Reference	Level	SRO
295025 High Reactor Pressure **EA2. Ability to determine and/or interpret the following as they apply to** HIGH **REACTOR PRESSURE:** **EA2.02 Reactor power**	**Tier #**	1
	Group #	1
	K/A #	**295025 – EA2.02**
	Rating	4.2

Question 79

An ATWS has occurred.

Current conditions are:

- Reactor power is 40%
- Reactor pressure is 1045 psig and steady
- Two SRVs are currently open
- Suppression pool level is at the Tech Spec low level limit
- Suppression pool temperature is 150°F and rising.
- RHR ‘A’ and RHR ‘B’ systems have just been placed in Suppression Pool Cooling

Which of the following describes the next action the CRS will direct?

A. Immediately Emergency Depressurize the Reactor due to exceeding HCTL is unavoidable

B. Anticipate an Emergency Depressurization and fully open the Main Turbine Bypass valves.

C. Initiate Suppression Pool Makeup System.

D. Emergency Depressurize the Reactor when HCTL is exceeded.

Answer: C
Explanation: With the plant maintaining current conditions, reactor power exceeds the capacity of the bypass valves therefore SRVs will continue to be opened to maintain pressure. With SRVs open Suppression pool temp will rise. Prior to exceeding HCTL step SPT-5 of EP-3 directs to initiate Suppression pool makeup system (SPMU) per step SPT-6.

The SRO must first know the required steps prior to ED on HCTL which is to initiate SPMU then based on the HCTL curve and knowing that the limit can be exceeded if there is the ability to restore.

SRO must know EP steps without reference.

C is correct

All other distractors are plausible, they all are different paths that could be taken if the student forgets the step in the EOPs and that HCTL can be exceeded if it can be restored to the safe zone.

A is wrong: Emergency Depress is required only if RPV pressure cannot be restored and maintained in the Safe Zone of the HCTL. Since the RHR systems have just been started in Suppression Pool cooling the SRO should wait and determine if the limit can be restored.

B is wrong - Reactor power is too high, the Bypass valves are already fully open and procedures do not allow to anticipate an ED while in EP-2A ATWS.

D is wrong - Emergency Depress is required only if RPV pressure cannot be restored and maintained in the Safe Zone of the HCTL.

Technical References:

EP-3
02-S-01-40, EP Technical Bases, Attachment V, pages 41, 44, and 45

References to be provided to applicants during exam:

HCTL curve

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	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.43(b)(5)	

Examination Outline Cross-Reference	Level	SRO
295031 Reactor Low Water Level 2.4.6 Knowledge of EOP mitigation strategies.	Tier #	1
	Group #	1
	K/A #	**295031 – G2.4.6**
	Rating	4.7

Question 80

A LOCA has occurred.

All high pressure RPV injection has been lost.

Reactor water level is -25” wide range and lowering.

Which of the following describes the NEXT step that should be performed?

A. Verify Suppression Pool level is >10.5 ft.

B. Wait until level reaches -204” Fuel Zone and Emergency Depressurize

C. Commence lining up as many Alternate Injection Systems as possible.

D. Inhibit ADS

Answer: D
Explanation: With a LOCA in progress and all High pressure injection lost, the SRO should recognize that reactor water level will continue to lower, so moving to Alternate level control leg should be done as soon as possible. EOPs does not give an actual level to move to Alt. Lvl Cont leg. **All distractors are plausible due to they are all possible paths that could be taken if the student forget the first step when entering the Alternate level control leg of EP-2.** A is wrong – This would be the first step if entering the Emergency Deprss leg of EP-2 B is wrong – This would be the step if entering the Steam Cooling leg of EP-2 C is wrong – This is in the Alt Level Control leg and after “Inhibit ADS” step, this is done only if less that 2 injection subsystems can be lined up.

D is correct. - This is the first step after entering the Alternate Level Control leg of EP-2		
Technical References: EP-2 02-S-01-40 Att. IV page 15 of 53.		
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	
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 2.4.21 Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.	**Tier #**	1
	Group #	1
	K/A #	**295038 – G2.4.21**
	Rating	4.6

Question 81

A main steam line has ruptured in the Aux Building Main Steam Tunnel (MST).

MST Temperature has exceeded the Group 1 isolation signal.

Both MSIVs on the 'A' Main Steam Line has failed to isolate

Dose assessment has reported that General Emergency radiation levels will be exceeded in approximately 5 minutes.

The CRS should…

A. Immediately exit EP-2 Pressure control and enter Emergency Depressurization.

B. When a General Emergency is declared on high radiation, exit EP-2 Pressure control and enter Emergency Depressurization.

C. When the second radiation level reaches the max safe value from Table SC-1, exit EP-2 Pressure control and enter Emergency Depressurization.

D. Stay in EP-2 and operate all available coolers and sump pumps to restore and maintain Table SC-1 parameters below their max safe values while shutting down the reactor (IOI-03-1-01-2)

Answer: A
Explanation: EP-4 step 11 states "If A discharge cannot be isolated AND Declaration of a General Emergency is expected due to radioactivity release dose calculation. THEN; enter EP-2, Emergency Depressurization is required: Exit EP-2 Enter Emergency Depressurization.

A is correct

B is wrong – The SRO does not have to wait for the GE limit to be exceeded due to the EP-4 Step 11 saying Declaration of a GE is expected…Plausible if the student forgets the bases of the step stating that a GE is expected.

C is wrong – When the GE limit is expected to be exceeded then monitoring the other areas are not necessary. Plausible due to this is a possible path if the student forgets about the immediate ED if GE limits are expected to be exceeded.

D is wrong - When the GE limit is expected to be exceeded then monitoring the other areas are not necessary. Plausible due to this is a possible path if the student forgets about the immediate ED if GE limits are expected to be exceeded.

Technical References:

EP-4, Secondary Containment Control flowchart
02-S-01-40, Att. VII 15 & 15 of 21

References to be provided to applicants during exam:

None

Learning Objective: Document learning objective if possible.

GLP-OPS-EP04 Obj 3 & 6

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:	55.43(b)(5)	

Examination Outline Cross-Reference	Level	SRO
700000 Generator Voltage and Electric Grid Disturbances 2.2.22 Knowledge of limiting conditions for operations and safety limits.	Tier #	1
	Group #	1
	K/A #	**700000 – G2.2.22**
	Rating	4.7

Question 82

Division 1 Diesel Generator has been declared INOP.

An operator is currently performing 06-OP-1R20-W-0001, PLANT AC AND DC ELECRICAL POWER DISTRIBUTION, and reports the following:

- Recorded Voltage for Baxter Wilson and Franklin is 490kV
- Recorded Frequency for Baxter Wilson and Franklin is 58Hz
- Recorded Voltage for 115kV Line Port Gibson is 112.33kV

If the given information stays the same, what time should the plant be in Mode 3?

A. 13 hours

B. 24 hours

C. 36 hours

D. 14.5 days

Answer: B
Explanation: The given information results in Baxter Wilson and Franklin line being declared INOP. Along with Division 1 D/G INOP would require entry into LCO 3.8.1 condition D with one D/G and one required Offsite feeder inop. With a restore either within 12 hours. Therefore after 12 hours you would enter condition G which is be in mode 3 in 12 hours, which makes a total time of 24 hours. A is wrong – If the student believes that all offsite feeders are inop then they would enter condition H which is enter 3.0.3 immediately, be in mode 3 in 13 hours B is correct

C is wrong – If the student forgets that Div 1 D/G is also inop and only enters condition C which is restore within 24 hours and enter condition G which is be in mode 3 in 12 hours, which makes a total time of 36 hours.

D is wrong – If the student believes that all offsite circuits are operable and Div 1 D/G is the only inop equipment then Condition B would be entered with a 14 day LCO to enter condition G which is be in mode 3 in 12 hours, which makes a total time of 14.5 days.

Technical References:

Tech Specs 3.8.1
06-OP-1R20-W-0001

References to be provided to applicants during exam:

Tech Specs 3.8.1
06-OP-1R20-W-0001 Attachment 1 page 2 of 12

Learning Objective: Document learning objective if possible.

GLP-OPS-TS01

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:	55.43(b)(2)	

Examination Outline Cross-Reference	Level	SRO
295013 High Suppression Pool Water Temperature 2.2.40 Ability to apply Technical Specifications for a system.	**Tier #**	1
	Group #	2
	K/A #	**295013 – G2.2.40**
	Rating	4.7

Question 83

RCIC is currently operating for a surveillance.

RCIC system has been INOP and this surveillance is required for return to operability.

Current Suppression Pool temperature is 97°F.

RHR A and B pumps have been started in Suppression Pool Cooling.

Which of the following describes the temperature at which the next action is required to be performed per Tech Specs and what is the Bases for this action?

A. 105°F secure testing which adds heat to the suppression pool
To provide margin to prevent required plant shutdown

B. 105°F secure testing which adds heat to the suppression pool
To ensure the suppression pool temperature does not exceed design limits

C. 110°F place the mode switch to Shutdown
To ensure the suppression pool temperature does not exceed design limits.

D. 110°F place the mode switch to Shutdown
To provide margin to prevent required plant shutdown

Answer: A
Explanation: Per Tech Specs a suppression pool temperature is allowed to be <105°F when performing testing which adds heat to the suppression pool. Once 105°F is reached Tech Specs requires testing to be stopped immediately. The Bases is to ensure a margin to required plant shutdown at 110°F. This allows the ability to cool down the suppression pool before 110°F is reached. "This requirement

ensures that the plant has testing flexibility, and was selected to provide margin below the 110°F limit at which reactor shutdown is required." A is correct B is wrong – the bases listed is for 110°F limit. Plausible if the student forgets the bases for the limits. C & D is wrong - 110°F is not the next action required 105°F is. Plausible if the student forgets that RCIC is required to be shutdown at 105°F.
Technical References: Tech Spec 3.6.2.1 and B3.6.2.1
References to be provided to applicants during exam: NONE
Learning Objective: Document learning objective if possible. GLP-OPS-TS001

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	2
10CFR Part 55 Content:	55.43(b)(2)	

Examination Outline Cross-Reference	Level	SRO
295017 High Off-Site Release Rate AA2. Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE AA2.01 Off-site release rate: Plant-Specific	Tier #	1
	Group #	2
	K/A #	**295017 – AA2.01**
	Rating	4.2

Question 84

An event has occurred that has caused an Off-site release.

Conditions are:

- Computer point D173001 and D173002 is in alarm
- SBGT A is indicating 30 Ci/sec and has been for 20minutes.
- Dose assessment Team indicates readings of the following:
    - 115 mR TEDE at Survey Point ML (0.2 miles)
    - 110 mR TEDE at Survey Point LK (0.3 miles)
    - 105 mR TEDE at Survey Point LK (0.5 miles)
    - 90 mR TEDE at Survey Point FG (0.6 miles)

Which of the following is the Emergency Declaration?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Answer: C
Explanation: With the two computer points in alarm and indicating 30 Ci/sec for 20 minutes meets the criteria for an Alert per EAL AA1(1), Valid reading on the radiation monitors in Table R1 “ALERT” > the reading

shown for ≥15 minutes. However, the Dose assessment team reports 115 mr at point ML (0.2) and 105 mr at point LK (0.5), the student will be required to know the definition of Site Boundary. Per 10-S-01-12, Radiological Assessment and Protective Action Recommendations, SB – Site Boundary – for emergency dose calculations; the site boundary is fixed at 696 meters (0.43 miles) from the center of the reactor. With indication of 105 mr at survey point LK which is 0.5 miles from the center of the reactor, this exceeds the threshold for a Site Area Emergency AS1 (2), Dose assessment using actual meteorology indicates doses >100 mR TEDE at or beyond the site boundary. C is correct all other distractors are plausible due to being emergency classifications.
Technical References: 10-S-01-1 EAL Flow Charts Two Mile Offsite Monitoring Team Map
References to be provided to applicants during exam: EAL Flow Charts Two Mile Offsite Monitoring Team Map
Learning Objective: Document learning objective if possible. GLP-EP-EPTS6

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:	55.43(b)(5)	

Examination Outline Cross-Reference	Level	SRO
295034 Secondary Containment Ventilation High Radiation EA2. Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION: EA2.02 Cause of high radiation levels	Tier #	1
	Group #	2
	K/A #	**295034 – EA2.02**
	Rating	4.2

Question 85

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

- DRWL AIRB RAD HI and HI HI, P870-3A-C1 and C2
- OG PRE-TREAT RAD HI, P601-19A-E7
- MSL RAD HI, P601-19A-D4

All of the indications for the above listed instruments are rising.

What is the source of the radiation levels?

AND

If MSL A/ MSL D HI HI alarm P601-18A-C4 were to alarm what would be the SRO's next action?

A. Resin intrusion
   Remain in EP-4, Manually SCRAM Reactor and Close all MSIVs and Drains

B. Resin intrusion
   Exit EP-4, Enter EP-2 and perform an Emergency Depressurization

C. Fuel failure
   Exit EP-4, Enter EP-2 and perform an Emergency Depressurization

D. Fuel failure
   Remain in EP-4, Manually SCRAM Reactor and Close all MSIVs and Drains

Answer: D

Explanation:

With the given information the SRO should determine that the radiation source is fuel failure. Drywell radiation would be the first to respond then Offgas pretreat then MSL rad monitor as evident by performing a fuel failure in the simulator.

EP-4 should be entered when MSL RAD HI, P601-19A-D4 alarm is received. Then, Per EP-4 retention step “If MSL radiation level is more than 3 times full power background THEN close the MSIVs and MSL drain valves. (3 times full power background is defined as alarms P601-18A(19A)-C4, MSL A(B) / MSL D(C) RAD Hi Hi). The CRS should direct to shutdown the reactor prior to closing MSIVs, due to MSIV closure will cause a scram.

A is wrong – resin intrusion would first be seen in RWCU and condensate conductivity, then Offgas pre treatment then MSL rad monitors, not drywell fission product monitor.

B is wrong – resin intrusion would first be seen in RWCU and condensate conductivity, then Offgas pre treatment then MSL rad monitors, not drywell fission product monitor. EP-4 is not Exited and EP-2 is not entered, only reactor shutdown prior to closing MSIVs.

C is wrong – EP-4 is not Exited and EP-2 is not entered, only reactor shutdown prior to closing MSIVs.

D is correct.

Technical References:

ARI Pre-Treatment Rad monitor 04-1-02-1H13-P601-19A-E7 & C8
ARI Drywell fission product monitor 04-1-02-1H13-P870-3A-C1 & C2
ARI MSL rad monitor 04-1-02-1H13-P601-19A-D4

02-S-01-40 EP Technical Bases, Attachment VII page 5 of 21

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-EP04

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:	55.43(b)(5)	

Examination Outline Cross-Reference	Level	SRO
209002 High Pressure Core Spray System 2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation	Tier #	2
	Group #	1
	K/A #	**209002 – G2.1.23**
	Rating	4.4

Question 86

An ATWS is in progress being controlled with EP-2A with the following parameters:

- RPV water level is being maintained with Condensate/Feedwater in band of -70" to -130" Wide Range.
- RPV pressure is being maintained with bypass valves in band of 800 psig to 1060 psig.
- E30-F002B is tagged out for Maintenance
- Bus 15AA has tripped on lockout.
- RCIC has isolated on High Room Temperature.
- All actions have been completed in EP-2A to maintain level and pressure bands.
- Attachments have been called for but not installed

Suppression pool level is 17.1 ft.

Which of the following procedures should the SRO transition to in order to raise suppression pool water level?

A. Normal Suppression Pool Makeup, SOI 04-1-01-P11-2

B. HPCS, SOI 04-1-01-E22-1

C. RCIC, SOI 04-1-01-E51-1

D. SPMU, SOI 04-1-01-E30-1

Answer: B
Explanation: With the given information, the SRO has entered EP-3 SP LEVEL leg step SPL-6 which states "Maintain SP level above 14.5 ft. using one or more of the following systems"

All answers given are from the list in the EOP-3. The student must determine which one is available, therefore all distractors are plausible.

A is wrong - due to water level is being maintain at -70” to -130”, <-41.6” which will isolate the flowpath and this will not work.

B is correct – HPCS should be overridden per EP-2A, however the suction valves can be aligned for gravity drain.

C is wrong – RCIC has an isolation signal which will not allow the system to be started and the SOI requires the pump/turbine to be started on min flow.

D is wrong – due to the E30-F002B is one valve in series for SPMU and will render Division 2 SPMU unable to perform it intended function. With a Bus 15AA lockout, power will be lost to the Division 1 SPMU valves and it also will be unable to perform its intended function.

Technical References:

EP-3
Normal Suppression Pool Makeup, SOI 04-1-01-P11-2
HPCS, SOI 04-1-01-E22-1
RCIC, SOI 04-1-01-E51-1
SPMU, SOI 04-1-01-E30-1
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-EP03

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:	55.43(b)(5)	

Examination Outline Cross-Reference	Level	SRO
211000 Standby Liquid Control System A2. Ability to (a) predict the impacts of the following on the STANDBY LIQUID CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.04 Inadequate system flow	**Tier #**	2
	Group #	1
	K/A #	**211000 – A2.04**
	Rating	3.4

Question 87

An ATWS has occurred.

Reactor power is 25%.

The SRO calls for Standby Liquid Control (SLC) system initiation.

The operator reports the squib valves did not “FIRE”.

Which procedure should the SRO use to mitigate?

A. 05-S-01-EP-1, Emergency/Severe Accident Procedures Support Documents, Attachment 28

B. 04-1-01-C41-1, Standby Liquid Control System Operating Instruction Attachment 6

C. 04-1-01-C41-1, Standby Liquid Control System Operating Instruction, Section 6.1

D. 05-S-01-EP-1, Emergency/Severe Accident Procedures Support Documents, Attachment 27

Answer: A
Explanation: With the given conditions the SLC pumps should be running due to the squib valves are on the discharge of the pumps, the Suction valves would prevent a pump start, but the pumps do not have a flow path and will have Inadequate system flow to the vessel. The pumps will be running but on minimum flow.

Per EP-2A "If Boron cannot be injected using SLC THEN Inject boron using RCIC or HPCS." Using Attachment 28 which is found in procedure 05-S-01-EP-1

A is correct

B is wrong – plausible because Attachment VI for Verification of SLC Injection is located in 04-1-01-C41-1

C is wrong - plausible because Section 6.1 of 04-1-01-C41-1 is for Adding Water to RPV from SLC Test Tank.

D is wrong – plausible because this attachment is used in a non ATWS situation to inject SLC test tank into the reactor for level control.

Technical References:

EP-2A
05-S-01-EP-1
04-1-01-C41-1

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-EP02

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:	55.43(b)(5)	

Examination Outline Cross-Reference	Level	SRO
217000 Reactor Core Isolation Cooling System (RCIC) A2. Ability to (a) predict the impacts of the following on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.15 Steam line break	Tier #	2
	Group #	1
	K/A #	**217000 – A2.15**
	Rating	3.8

Question 88

The plant is operating at 100% power.

The following alarms are received:

- P601-21A-G3, RCIC EQUIP AREA TEMP HI/INOP
- P601-21A-H2, RCIC PIPE/EQUIP AMBIENT TEMP HI
- P601-21A-H3, RCIC EQUIP AREA DT HI

NO RCIC valves have changed position.

Manual action did not change any valve position.

Current temperatures:

- RCIC Room Temp is 204°F and rising.
- MSL Pipe Tunnel Temp is 155°F and rising.
- RHR A & B Equip Area Temp is 90°F and stable.
- RWCU Pump Room 1 & 2 Temp is 115°F and stable.

Which of the following describes the effect on the RCIC system?

AND

What should be the SRO’s next action?

A. RCIC Turbine trip ONLY
Shutdown the reactor using IOI - 2

B. RCIC Turbine trip ONLY
Place the Reactor Mode Switch to SHUTDOWN

C. Full Group 4 Isolation Signal and RCIC Turbine trip
Shutdown the reactor using IOI - 2

D. Full Group 4 Isolation Signal and RCIC Turbine trip
   Place the Reactor Mode Switch to SHUTDOWN

Answer: D

Explanation:

With the given information alarm G3 exceeds the RCIC room isolation setpoint, this isolation will cause a complete group 4 isolation signal which includes the RCIC pump suction from the Suppression Pool.

With the statement of NO RCIC valves have changed state and RCIC room temperature is still rising the SRO should assume that there is a steam line break and RCIC system has failed to isolate.

Alarm G3 is also an entry condition for EP-4. The SRO would enter step 6, "IF a system that cannot be isolated from the RPV is discharging outside the CTMT, THEN go to 22."

Step 8 (22) states "BEFORE Any area temperature, radiation level, or water level reaches its max safe value, Enter EP-2." RCIC max safe value is 212°F.

Step 1 of EP-2 states "Verify the Rx Mode Switch in SHUTDOWN.

A is wrong –The RCIC turbine will trip but on an isolation signal so an isolation signal will also be present and an IOI – 2 shutdown is only done if the system has isolated and another area has high temp, water level or rad level. Plausible if the student does not think an isolation signal has been reached and a Turbine trip is the only action and If the student believes that an isolation has occurred and chooses the wrong path in the EP-4. The given temps of the other rooms are not even close to their max safe value that would cause an IOI Shutdown.

B is wrong – The RCIC turbine will trip but on an isolation signal so an isolation signal will also be present. Plausible if the student does not think an isolation signal has been reached and a Turbine trip is the only action.

C is wrong – an IOI – 2 shutdown is only done if the system has isolated and another area has high temp, water level or rad level, Plausible if the student believes that an isolation has occurred and chooses the wrong path in the EP-4.

D is correct

Technical References:

EP-4 02-S-01-40, EP Technical Bases, Attachment VII, page 12 of 21
GLP-OPS-E5100
04-1-01-E51-1
EP-2
04-1-02-1H13-P601

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-EP04
GLP-OPS-E5100
P601-21A-G3, RCIC EQUIP AREA TEMP HI/INOP
P601-21A-H2, RCIC PIPE/EQUIP AMBIENT TEMP HI
P601-21A-H3, RCIC EQUIP AREA DT HI

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:	55.43(b)(5)

Examination Outline Cross-Reference	Level	SRO
223002 Primary Containment Isolation System/Nuclear Steam Supply Shut-Off 2.2.12 Knowledge of surveillance procedures.	Tier #	2
	Group #	1
	K/A #	**223002– G2.2.12**
	Rating	4.1

Question 89

While performing 06-OP-1G33-Q-0001, Reactor Water Cleanup System Valve Operability, the following valve stroke times were recorded:

- G33-F028, RWCU BLWDN CTMT INBD ISOL 35.2 seconds
- G33-F034, RWCU BLWDN CTMT OTBD ISOL 27.5 seconds

Which of the following describes the result on RWCU and Tech Spec action required?

A. RWCU system must be shutdown
Isolate the affected penetration flow path within 4 hours

B. RWCU system can continue to operate
Isolate the affected penetration flow path within 1 hour

C. RWCU system must be shutdown
Isolate the affected penetration flow path within 1 hour

D. RWCU system can continue to operate
Isolate the affected penetration flow path within 4 hours

Answer: D
Explanation: The G33-F028 is required to be declared INOP due to exceeding the TS Maximum isolation time, however, the G33-F034 exceeds the IST Acceptable Stroke Time but less than the limiting value of full stroke closed therefore the valve currently operable but will be tested again in 4 hours. Also the system can remain operating due to the valve in question Is BLOWDOWN isolation valves, not required for power operations. A is wrong – RWCU system can remain operating B is wrong – The TS action is Isolate the affected penetration within 4 hours.

C is wrong - RWCU system can remain operating and The TS action is Isolate the affected penetration within 4 hours

D is correct

Technical References:

Tech Spec 3.6.1.3
06-OP-1G33-Q-0001

References to be provided to applicants during exam:

06-OP-1G33-Q-0001, Attachment 1 page 4 of 7

Learning Objective: Document learning objective if possible.

GLP-OPS-TS001

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:	55.43(b)(2)	

Examination Outline Cross-Reference	Level	SRO
264000 Emergency Generators (Diesel/Jet) 2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.	Tier #	2
	Group #	1
	K/A #	**264000 – G2.2.25**
	Rating	4.2

Question 90

In the event of the Division 3 Diesel Generator becomes INOP, Tech Specs will allow up to 17 days completion time.

Which of the following are additional contingencies that are to be in place during the duration of the extended AOT for the HPCS Diesel Generator?

1. Weather conditions will be evaluated.
2. Condition of the offsite power supply and switchyard will be evaluated.
3. Only one offsite power source can have any elective maintenance performed,
4. RCIC can be taken out of service but for only 12 hours during the AOT.
5. Division 1 and 2 DGs will not be taken out of service for planned maintenance during the AOT.

A. 2, 3, 5 only

B. 1, 2, 5 only

C. 1, 2, 3 only

D. 2, 4, 5 only

Answer: B

Explanation:

TS 3.5.1 Bases for B4 action for HPCS can be extended to 17 days for a repair outage if the following contingencies are in place:

1. Weather conditions will be evaluated prior to entering an extended DG allowed outage .
2. The condition of the offsite power supply and switchyard will be evaluated.
3. No elective maintenance will be scheduled within the switchyard that would challenge offsite power availability during the extended Division 3 DG allowed outage limits.
4. Operating crews will be briefed on the DG work plan

5. The RCIC high pressure injection system and the Division 1 and 2 DGs will not be taken out of service for planned maintenance while the Division 3 DG is out of service during the extended allowed outage time.

A is wrong – Number 3 is incorrect due to the bases states NO maintenance will be scheduled within the switchyard that would challenge offsite power availability

B is correct

C is wrong - Number 3 is incorrect due to the bases states NO maintenance will be scheduled within the switchyard that would challenge offsite power availability

D is wrong - Number 4 is incorrect due to the bases states The RCIC system will not be taken OOS for planned maintenance.

Technical References:

3.8.1 Tech Specs and Bases

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-TS01

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:	55.43(b)(2)	

Examination Outline Cross-Reference	Level	SRO
290003 Control Room HVAC	Tier #	2
	Group #	2
2.2.38 Knowledge of conditions and limitations in the facility license.	K/A #	290003 – G2.2.38
	Rating	4.5

Question 91

In Modes 1, 2 and 3:

Which of the following is the maximum required Control Room Envelope unfiltered air inleakage per SR 3.7.3.4?

A. 2000 scfm which is maintained by restricting the maximum opening of the Control Room Envelope boundary to less than 205 square inches.

B. 2000 scfm which is maintained by restricting the maximum opening of the Control Room Envelope boundary to less than 20 square inches.

C. 4000 scfm which is maintained by restricting the maximum opening of the Control Room Envelope boundary to less than 20 square inches.

D. 4000 scfm which is maintained by restricting the maximum opening of the Control Room Envelope boundary to less than 205 square inches.

Answer: B
Explanation: SR 3.7.3.4 states perform surveillance in accordance with the Control Room Envelope Habitability Program. 5.5.13 Control Room Envelope Habitability Program states in C, Requirements for determining the unfiltered air inleakage in accordance with Regulatory guide 1.197. Reg Guide 1.197 states in 2.5 Inleakage Test Acceptance Criteria, the acceptance criterion for the test should be the licensing basis amount less the amount for ingress and egress. Per Facility Operating License 2.C.(38) Control Room Leak Rate, Control room leak rate not to exceed 2000 cfm. Per 01-S-07-37, step 6.1.4a, opening will not exceed 20 square inches. A is wrong – but, plausible due to the 205 square inches is for modes 4 and 5

B is correct

C is wrong – but, plausible due to this is the normal 100% flow of one Control Room Standby Fresh air unit.

D is wrong - but, plausible due to this is the normal 100% flow of one Control Room Standby Fresh air unit and the 205 square inches is for modes 4 and 5

Technical References:

Tech Specs – 3.7.3
Tech Specs - 5.5.13 - Control Room Envelope Habitability Program
NPF-29 Facility Operating License – 2.C.(38) Control Room Leak Rate
01-S-07-37, Control of Work for Penetrations, Painting, Snubbers, Insulation and Control Room Envelope Breaches.

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-PROC Objective G5

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.43(b)(1)	

Examination Outline Cross-Reference	Level	SRO
201005 Rod Control and Information System A2. Ability to (a) predict the impacts of the following on the ROD CONTROL AND INFORMATION SYSTEM (RCIS) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.13 Rod drift: BWR-6	**Tier #**	2
	Group #	2
	K/A #	**201005 – A2.13**
	Rating	3.8

Question 92

Reactor STARTUP is in progress with power at 20%

Control rod that was at position 00 begins to drift out.

The following alarms and indications were received:

- CONT ROD WITHDRAWAL BLOCK, P680-4A2-C5
- CONT ROD DRIFT, P680-4A2-E4

RC&IS status lights illuminated:

- ROD DRIFT
- INSERT BLOCK
- WITHDRAW BLOCK

ATC operator performs the ONEP immediate operator actions but the control rod continues to drift out.

Which of the following describes the procedure actions that should be directed by the CRS to allow the control rod to be inserted?

A. 05-1-02-IV-1, CONTROL ROD/DRIVE MALFUNCTIONS, section 3.2, closing valve 102 and 103

B. 05-1-02-IV-1, CONTROL ROD/DRIVE MALFUNCTIONS, section 3.5, raise CRD Drive water differential pressure.

C. 04-1-01-C11-2, Rod Control and Information System, section 5.1, Rod Position Bypass and control rod movement.

D. 04-1-01-C11-2, Rod Control and Information System, section 6.1, RC&IS Rod Drive Bypass.

Answer: C

Explanation:

With the power level at 20% the Rod Pattern Controller is in effect. Any rod out of pattern an Insert and Withdrawal block will occur. With a drifting control rod the immediate operator actions of 05-1-02-IV-1, CONTROL ROD/DRIVE MALFUNCTIONS, says to apply a continuous INSERT signal, this is not possible to insert the rod due to the INSERT block. The SRO should realize that the rod must be bypassed in RACS to have the ability to insert.

All distractors are plausible due to possible paths from the CRD Malfunctions ONEP depending upon the failure.

A is wrong – this step is only done after the control rod is fully inserted

B is wrong – this is performed if control rod is not moveable

C is correct

D is wrong – by bypassing a control rod in this manner would be using RGDS and the rod cannot be moved at all.

Technical References:

05-1-02-IV-1, CONTROL ROD/DRIVE MALFUNCTIONS
04-1-01-C11-2, Rod Control and Information System

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-C1102

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:	55.43(b)(5)	

Examination Outline Cross-Reference	Level	SRO
234000 Fuel Handling Equipment K3. Knowledge of the effect that a loss or malfunction of the FUEL HANDLING EQUIPMENT will have on following: K3.04 core modifications/alterations	**Tier #**	2
	Group #	2
	K/A #	**234000 – K3.04**
	Rating	3.8

Question 93

A Refuel Outage is in progress.

Fuel movement is ongoing.

A new fuel bundle is being moved into the core when the Refuel Floor SRO notices the location is already occupied.

The Refueling crew returned the bundle to its original location.

After reviewing the movement plan, two misloading errors were discovered.

Which of the following describes the actions of the Refuel Floor SRO?

Who must approve resuming fuel movement?

A. Halt any further fuel movement
RE Supervisor

B. Continue fuel movement in adjacent quadrants only
Control Room Supervisor

C. Halt any further fuel movement
Control Room Supervisor

D. Continue fuel movement in adjacent quadrants only
RE Supervisor

Answer: A

Explanation:

Per 17-S-02-300, 6.1.1a(6), Placement of fuel shall not result in a core configuration in which the Tech Specs required shutdown margin (SDM) is not maintained. The following considerations apply to the SDM assessment:

Substep
(h) If a misloading error is discovered, further movement must be halted until a SDM analysis is done and recovery plan developed. RE Supervisor or designee approval is required to resume.

A is correct

B is wrong – per above procedure movement must be halted and RE Supervisor must approve

C is wrong - RE Supervisor must approve

D is wrong - per above procedure movement must be halted

Technical References:

17-S-02-300, Fuel and Core Component Movement Control, 6.1.1a(6)(h)

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-RF-F1105

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.43(b)(6)	
	55.43(b)(7)	

Examination Outline Cross-Reference	Level	SRO
2.1.5 Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.	Tier #	G
	Group #	
	K/A #	**2.1.5**
	Rating	3.9

Question 94

Per Tech Spec 5.2.2, shift crew may be one less than the minimum requirement of 10CFR50.54 for a period of two hours…

A. for any required crew member as long as action is taken to notify the NRC within 2 hours.

B. only if the missing crew member is a non-licensed operator and action is taken to notify the NRC within 2 hours.

C. only if the missing crew member is a licensed operator and immediate action is taken to restore the crew compliment.

D. for any required crew member as long as immediate action is taken to restore the crew compliment.

Answer: D
Explanation: Per the administrative requirements of TS 5.2.2, shift crew may be one less than any required crew member by 10CFR50.54(m)(2)(i). This section only addresses all operators (licensed and non-licensed) and does not specify any subtitle such as Shift Manager. The allowance is for two hours if immediate action is taken to restore crew compliment. Answer A is wrong but plausible if the applicant mistakes the reportability requirement for not meeting a license condition as being applicable if crew composition is restored within the 2 hour limit. Answers B & C are wrong but plausible since TRM Table 7.2.2-1 lists non-licensed operators as a requirement for minimum crew manning. EN-OP-115 attachment 9.3 section 1 “Shift Manning” places additional limitations in that the positions listed in TRM Table 7.2.2-1 including non-license operators must meet the two hour manning requirement along with the immediate action to restore. D is correct

Technical References: Tech Specs 5.2		
References to be provided to applicants during exam: NONE		
Learning Objective: Document learning objective if possible. GLP-OPS-TS01		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.43(b)(2)	

Examination Outline Cross-Reference	Level	SRO
2.1.34 Knowledge of primary and secondary plant chemistry limits.	Tier #	G
	Group #	
	K/A #	**2.1.34**
	Rating	3.5

Question 95

The plant is operating at rated thermal power.

- Feedwater and Condensate conductivity is trending up but do not exceed any operational limits.
- Reactor Water Conductivity is 1.5 umho/cm.
- 1H13-P680-11A-C3, RWCU FLTR DMIN EFL CNDCT HI/LO alarm is also received.

The CRS will…

A. (1) Enter Condensate/Reactor Water High Conductivity ONEP and TRM 6.4.1
(2) Restore reactor chemistry to within limits in 72 hours

B. (1) Enter Condensate/Reactor Water High Conductivity ONEP and TRM 6.4.1
(2) Manually scram the reactor

C. (1) Enter TRM 6.4.1, Only
(2) Restore reactor chemistry to within limits in 72 hours

D. (1) Enter Condensate/Reactor Water High Conductivity ONEP, Only
(2) Manually scram the reactor

Answer: A
Explanation: TRM 6.4.1 is entered in Mode 1 with Conductivity > 1.0 umho/cm. The Condensate/Reactor Water High Conductivity ONEP directs actions with as little as 0.3 umho/cm so it also requires entry. The TRM action in this case is to restore within 72 hours. The ONEP requires a scram when conductivity exceeds 5 umho/cm. A is correct B is wrong – only required when conductivity exceeds 5 umho/cm. Plausible if the student

forgets when a reactor shutdown is required per Tech Specs

C is wrong – The given alarm is a symptom for the ONEP and should be entered. Plausible is the student forgets that the ONEP should be entered.

D is wrong – Tech Specs should be entered when conductivity is >1.0 and a shutdown is required when 5 umho/cm. Plausible if the student forgets when a reactor shutdown is required per Tech Specs and the given alarm is a symptom for the ONEP and should be entered.

Technical References:

TRM 6.4.1, Chemistry
05-1-02-V-12, Condensate/Reactor Water High Conductivity ONEP

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-TS0V1 Obj 4

Question Source:	Bank #617	2014 AUDIT
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2013 NRC Q #99
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.43(b)(2)	

Examination Outline Cross-Reference	Level	SRO
2.2.5 Knowledge of the process for making design or operating changes to the facility.	Tier #	G
	Group #	
	K/A #	**2.2.5**
	Rating	3.2

Question 96

An emergency temporary modification can be implemented in the event of an imminent threat to the safety of personnel or facilities 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 Engineering 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 measure, then the Shift Manager will ensure that a CR is issued to track Comp measures.

A. 2, 3 and 4 only

B. 2, 4 and 5 only

C. 1, 3 and 5 only

D. 1, 2 and 3 only

Answer: C
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 of personnel or facilities:

(a) The Shift Manager, with the concurrence of the Engineering Director, or designee, and a completed 50.59 screening prior to implementation, 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 Tempor Modification shall follow the applicable steps of this procedure.

(c) **<u>IF</u>** the Temporary Modification is also a compensatory measure (operational), **<u>THEN</u>** 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 Engineering Manager is not notified but the Systems & Components Manger is and the CR is to be initiated by Engineering not Maintenance, and #4 is incorrect.

B is wrong #2 is wrong due to the Engineering Manager is not notified but the Systems & Components Manger is and the CR is to be initiated by Engineering not Maintenance, and #4 is incorrect.

C is correct

D is wrong - #2 is wrong due to the Engineering Manager is not notified but the Systems & Components Manger is and the CR is to be initiated by Engineering not Maintenance,, and #5 is also required.

Technical References:

EN-DC-136, Temporary Modifications

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-PROC,Obj. 40.0

Question Source:	Bank #1237,	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2015 Q #97
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.43(b)(3)	

Examination Outline Cross-Reference	Level	SRO
2.3.6 Ability to approve release permits.	Tier #	G
	Group #	
	K/A #	**2.3.6**
	Rating	3.8

Question 97

Per 01-S-08-11, Radioactive Discharge Controls, who is required to sign the **"Release Authorization"** block of a Batch Liquid Radioactive Release Permit?

A. Shift Manager and Operations Supervision

B. Operations Supervision, only

C. Shift Manager and Radiochemist

D. Shift Manager, only

Answer: D
Explanation: 01-S-08-11 "Release Authorization", Part 4, signature is only the Shift Manager. The responsibilities section states that the SM is responsible for all releases during his shift. A is wrong – But plausible due to Operations Supervision signs part 1 and at the end of part 5, Release. B is wrong - But plausible due to Operations Supervision signs part 1 and at the end of part 5, Release C is wrong – But plausible due to the Radiochemist signs part 2 and part 5 D is correct
Technical References: 01-S-08-11
References to be provided to applicants during exam: NONE

Learning Objective: Document learning objective if possible. GLP-OPS-PROC Obj 51.3		
Question Source:	Bank #628	2013 Audit
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.43(b)(5)	

Examination Outline Cross-Reference	Level	SRO
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 #**	G
	Group #	
	K/A #	**2.3.13**
	Rating	3.8

Question 98

A Site Area Emergency has just been declared.

The SRO acting as Emergency Director in the main control room must ensure emergency dosimetry issue requirements are met for the Control Room crew if:

A. radiation levels in the Control Room exceed the Main Control Room area radiation monitor (ARM) alarm setpoint.

B. an evacuation of the Auxiliary Building is ordered.

C. offsite radiation levels are expected to exceed General Emergency levels.

D. the designated Radiation Protection technician assigned to the Control Room is not present.

Answer: A

Explanation:

Correct answer is from 10-S-01-17, Emergency Personnel Exposure Control, step 6.2.2. "The area radiation monitors or habitability surveys in the Control Room Envelope provide monitoring that may be used to determine habitability. Individual issue of emergency dosimetry is not required unless:

a. As specified in Step 6.2.1.

b. Radiation levels exceed ARM setpoint.

A is correct

B, C and D are wrong but all distracters are plausible since they represent conditions with potentially elevated plant radiation and/or contamination levels. However, they are all incorrect, because the only other condition, in addition to that stated in the correct answer, that requires the Shift Supervisor/Manager to ensure proper emergency dosimetry issuance is when personnel are leaving the control room envelope, per 10-S-01-17 step 6.2.1.

Technical References:

10-S-01-17 step 6.2.2

References to be provided to applicants during exam: NONE		
Learning Objective: Document learning objective if possible. GLP-EP-EPTS6 Obj. 8		
Question Source:	Bank #999	2015 AUDIT
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.43(b)(5)	

Examination Outline Cross-Reference	Level	SRO
2.4.6 Knowledge of EOP mitigation strategies.	Tier #	G
	Group #	
	K/A #	**2.4.6**
	Rating	4.7

Question 99

All Condensate pumps have tripped and are unrecoverable.

An ATWS has occurred.

The CRS has called for the following attachments (but are not installed at this point):

- 18
- 19
- 20
- 12

Reactor water level is currently -180” and lowering.

Per CRS direction, operators open 8 ADS Valves.

When the Minimum Steam Cooling Pressure is reached, the CRS should…

A. wait no longer than 2 more minutes for the installation of attachment 12 before ordering injection with any available Table L-3, L-4 or L2 system.

B. wait as long as necessary for the installation of attachment 12 then inject via a B21-F053 valve.

C. order operators to inject with Low Pressure ECCS systems immediately.

D. order operators to inject with HPCS immediately.

Answer: A
Explanation: Per 02-S-01-43, Transient Mitigation Strategy, step 6.2.5, During EP-2A Emergency Depressurization with Reactor Pressure < MSCP, IF Attachment 12 is being installed AND

Will be completed within approximately 2 minutes, it is acceptable to wait AND INJECT with RHR outside of shroud. This WILL help minimize power spikes, OTHERWISE Step L-14 Should be answered “NO” AND INJECT per Step L-15. (Other ECCS systems inside shroud)

A is correct

B, C, and D are wrong but plausible due to these are systems that can be used to inject into the reactor, however, during an ATWS all other systems that inject outside the shroud should be used first. Including a wait time to install jumpers to allow injection outside the shroud. Plausible if the student does not understand the bases for the wait time before injection into the shroud.

Technical References:

EP-2A, 02-S-01-43, 6.2.5

References to be provided to applicants during exam:

NONE

Learning Objective: Document learning objective if possible.

GLP-OPS-EP02A, Objective 8

Question Source:	Bank #281	2013 Audit
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
	LOD	3
10CFR Part 55 Content:	55.43(b)(5)	

Examination Outline Cross-Reference	Level	SRO
2.4.40 Knowledge of SRO responsibilities in emergency plan implementation.	**Tier #**	G
	Group #	
	K/A #	**2.4.40**
	Rating	4.5

Question 100

The plant is operating at rated power.

A plant event occurs.

The event meets the threshold for an ALERT level of the EALs but is rapidly concluded (i.e., event no longer exists) before the CRS/SM actually classify the event.

Per 10-S-01-1, Activation of the Emergency Plan, which of the following describes the classification and notification requirements?

A. Classify the event but notify <u>only</u> the NRC.

B. Do <u>not</u> classify the event and notify <u>only</u> the NRC.

C. Classify the event and make <u>all</u> of the normal notifications to offsite agencies, including the NRC.

D. Do <u>not</u> classify the event and do <u>not</u> make any notifications to offsite agencies, including the NRC.

Answer: C
Explanation: See 10-S-01-1, Section 6.1.6.a(1), 1st bullet, where we are directed to both classify the rapidly concluded event and make all of the normal offsite agency notifications, including the NRC. A is wrong. It is plausible to the SRO Applicant who cannot recall the Section 6.1.6 criteria. B is wrong. Its plausibility comes directly from Section 6.1.6.b(1). C is correct D is wrong. It is plausible to the SRO Applicant who cannot recall the Section 6.1.6 criteria.

Technical References: 10-S-01-1, Activation of the Emergency Plan		
References to be provided to applicants during exam: NONE		
Learning Objective: Document learning objective if possible. GLP-EP-EPTS6 Obj 8		
Question Source:	Bank # 31	2013 Audit
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2012 NRC Q #99
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.43(b)(5)	

GGNS LOT 2017 NRC INITIAL LICENSED OPERATOR WRITTEN EXAMINATION
OPEN-REFERENCES TABLE OF CONTENTS
SRO EXAM

Additional References Provided Outside of This Binder:

- EAL Flow Chart

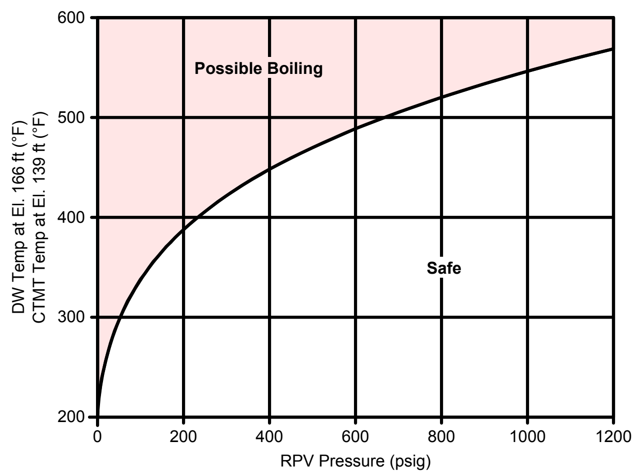
TAB 1

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)

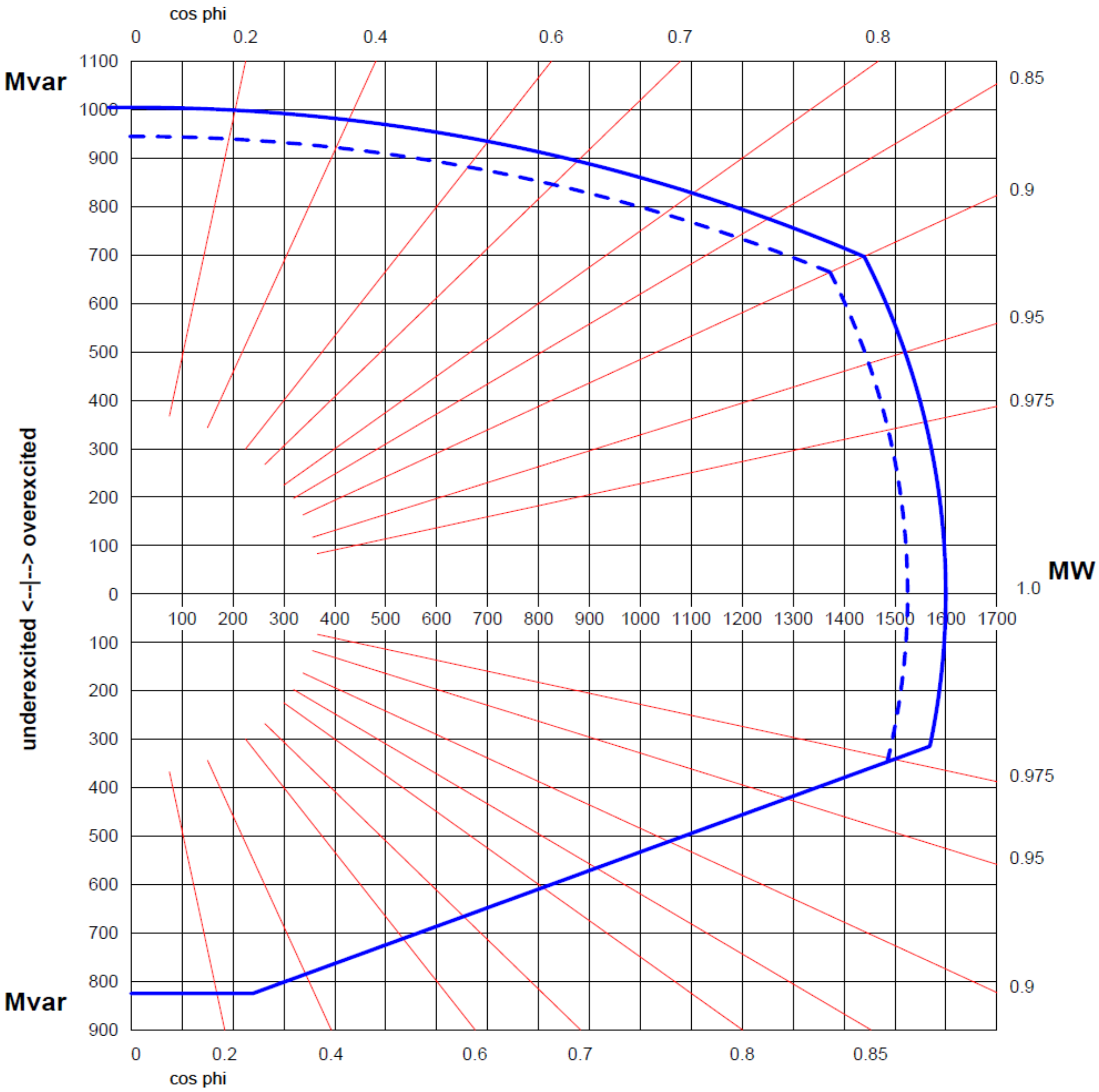


2. Do <u>not</u> use any RPV level instrument that exceeds <u>both</u> limits in the following table:

Level Instrument	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

TAB 2

Generator Type	THFF 180/76-18				
		uprate		prev.	
Apparent Power	S	1600.00	MVA	1525.00	MVA
Armature Voltage	U	22.00	kV		
Armature Current	I	41.989	kA	40.021	kA
Frequency	f	60.0	Hz		
Power Factor	P.F.	0.900		0.900	
H2-Pressure (gauge)	pe	5.170	bar	4.140	bar
Cold Gas Temperature	Tk	40.0	°C	40.0	°C



TAB 3

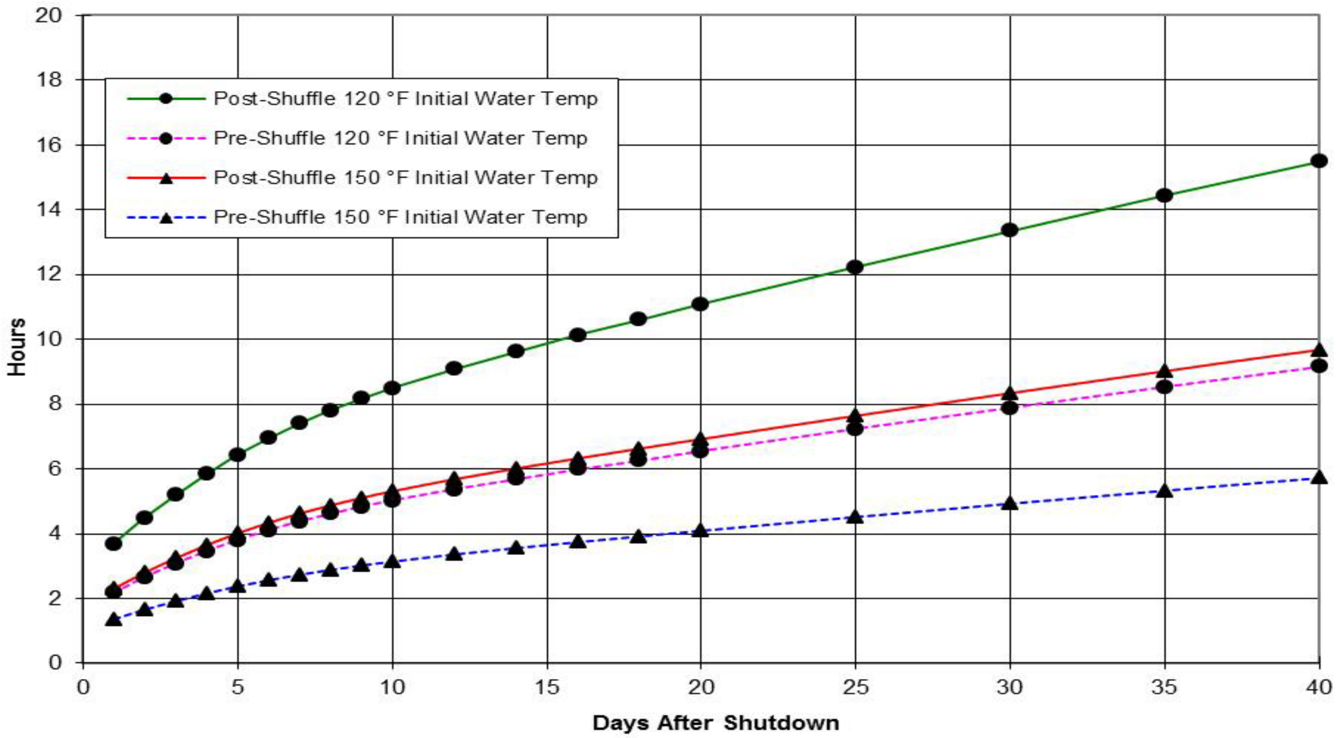


Figure 2: Time for Reactor Cavity to Reach 200°F for Initial High Water

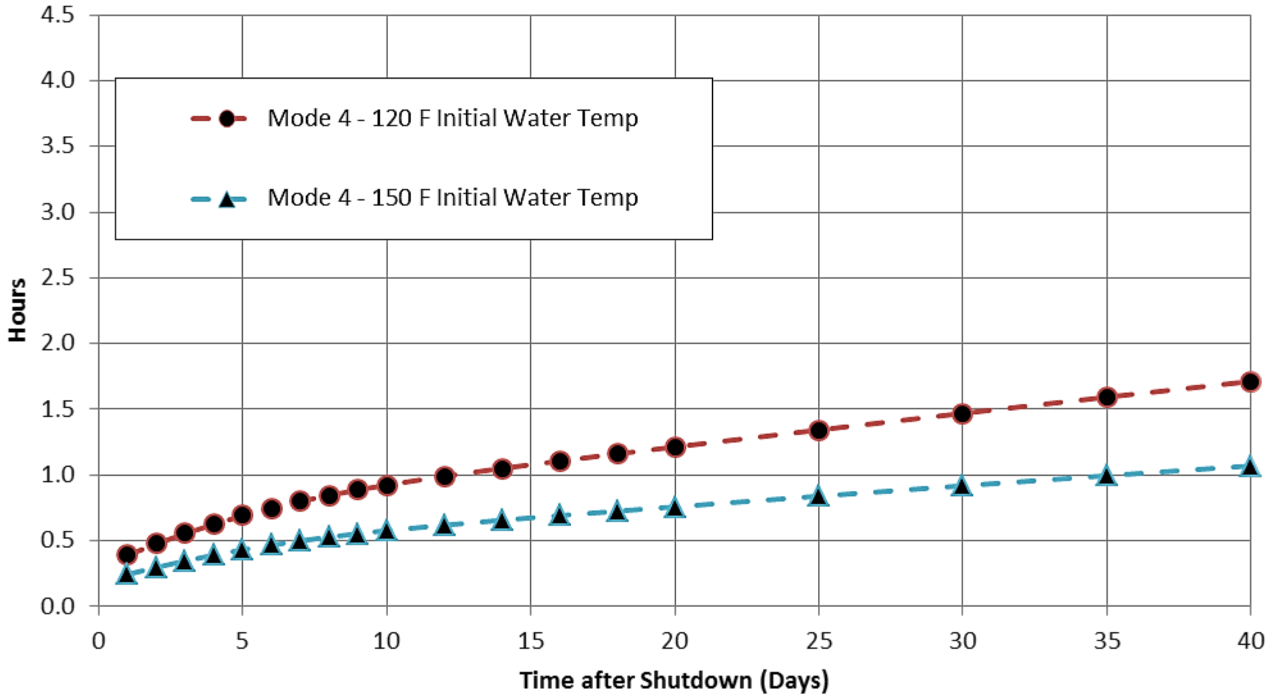


Figure 3: Time for Reactor Vessel to reach 200°F from Normal Water level 32", Mode 4, zero to forty days (*Assumes Cycle 21 EOC decay heat load*)

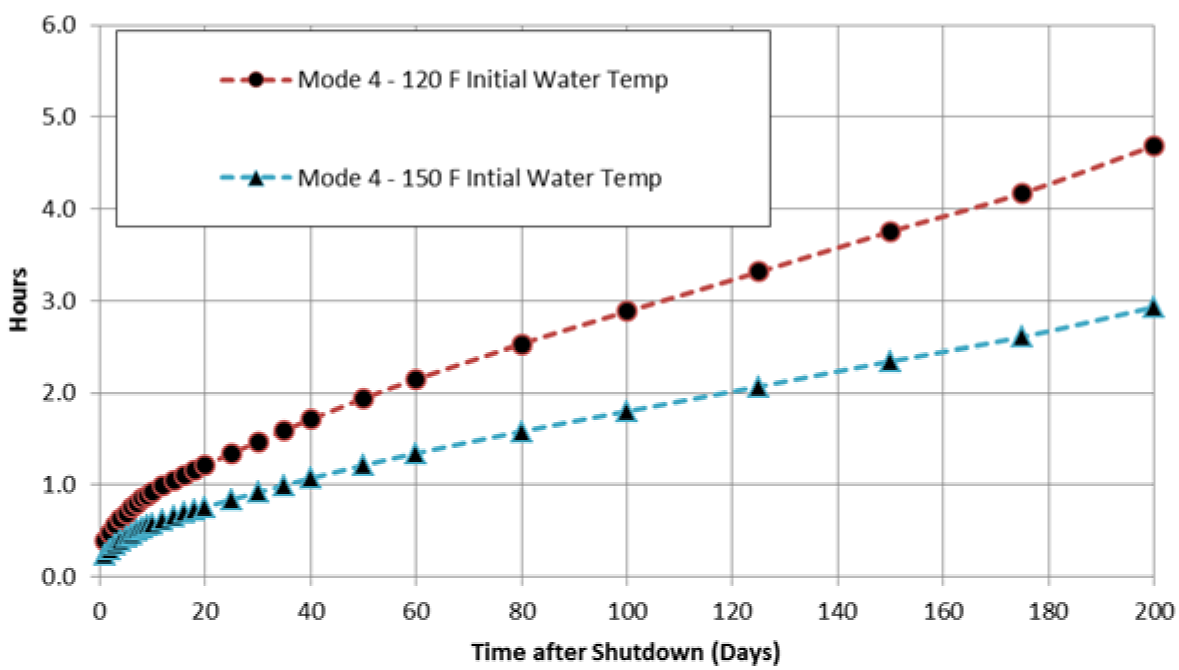


Figure 4: Time for Reactor Vessel to reach 200°F from Normal Water level 32", Mode 4, zero to two hundred days (*Assumes Cycle 21 EOC decay heat load*)

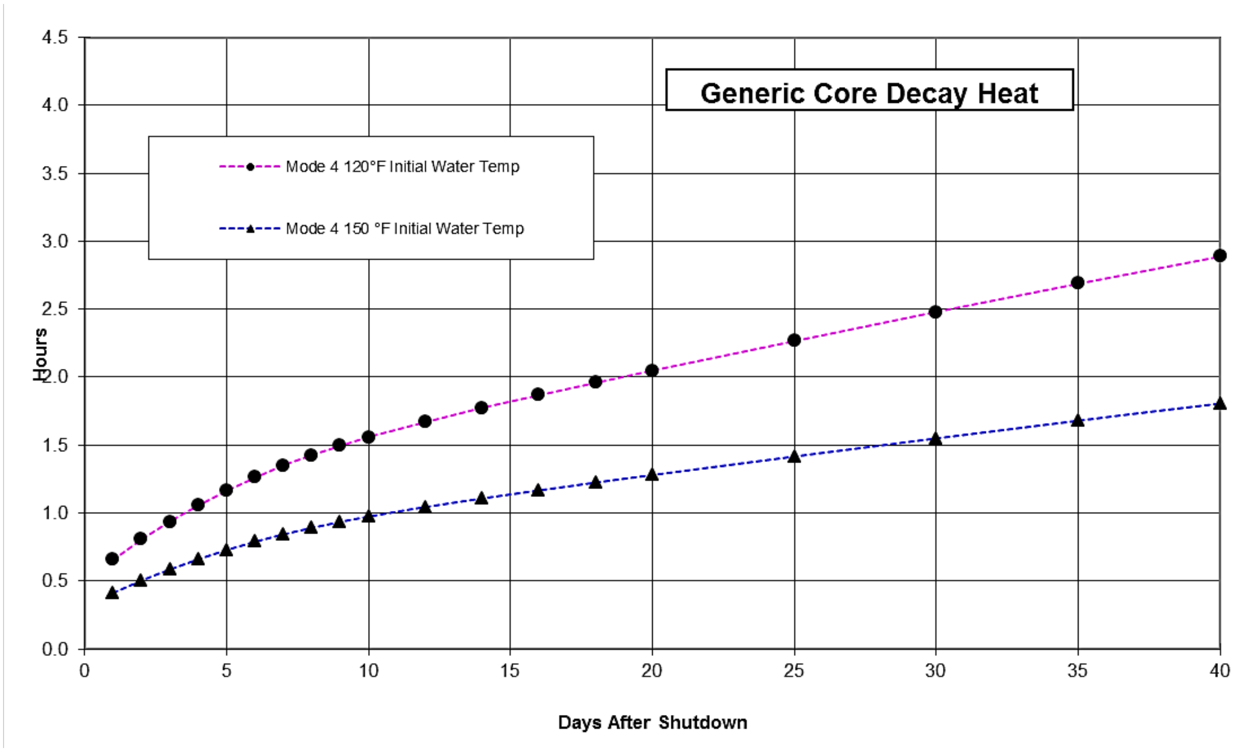


Figure 5: Time for Reactor Vessel to Reach 200°F for Initial Water Level 24" Below Vessel Flange, Mode 4, zero to forty days.
(*Assumes Cycle 21 EOC decay heat load*)

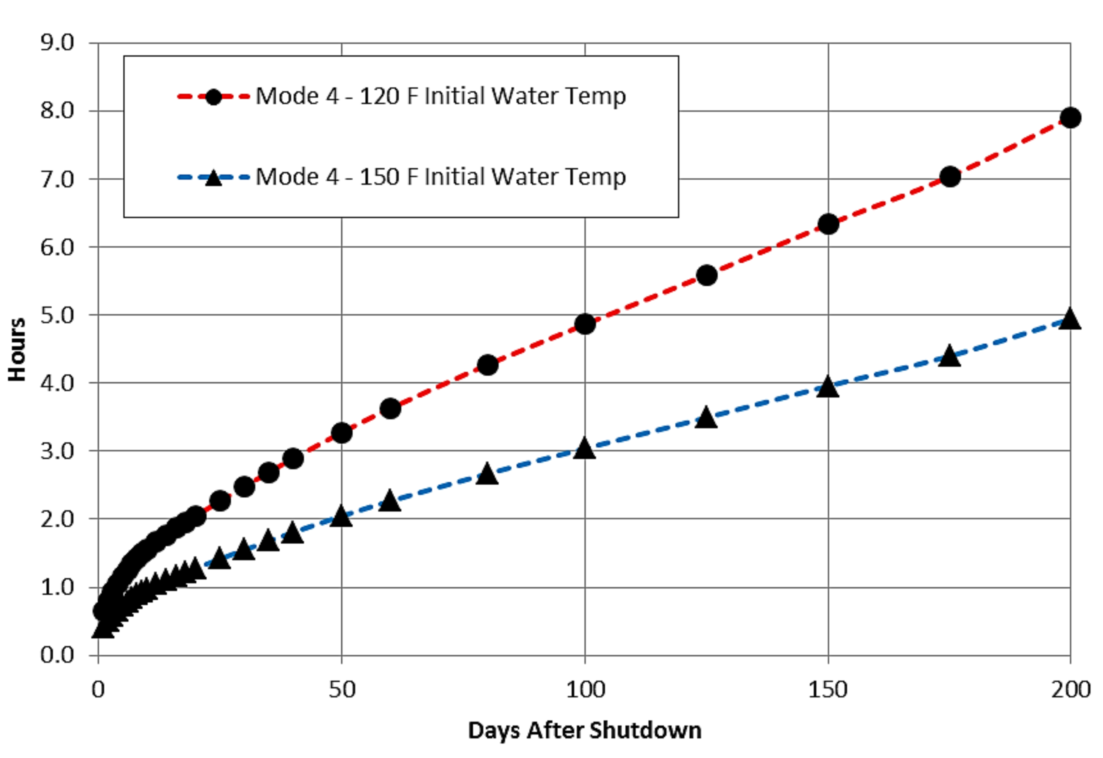


Figure 6: Time for Reactor Vessel to Reach 200°F for Initial Water Level 24" Below Vessel Flange, Mode 4, zero to two hundred days
(*Assumes Cycle 21 EOC decay heat load*)

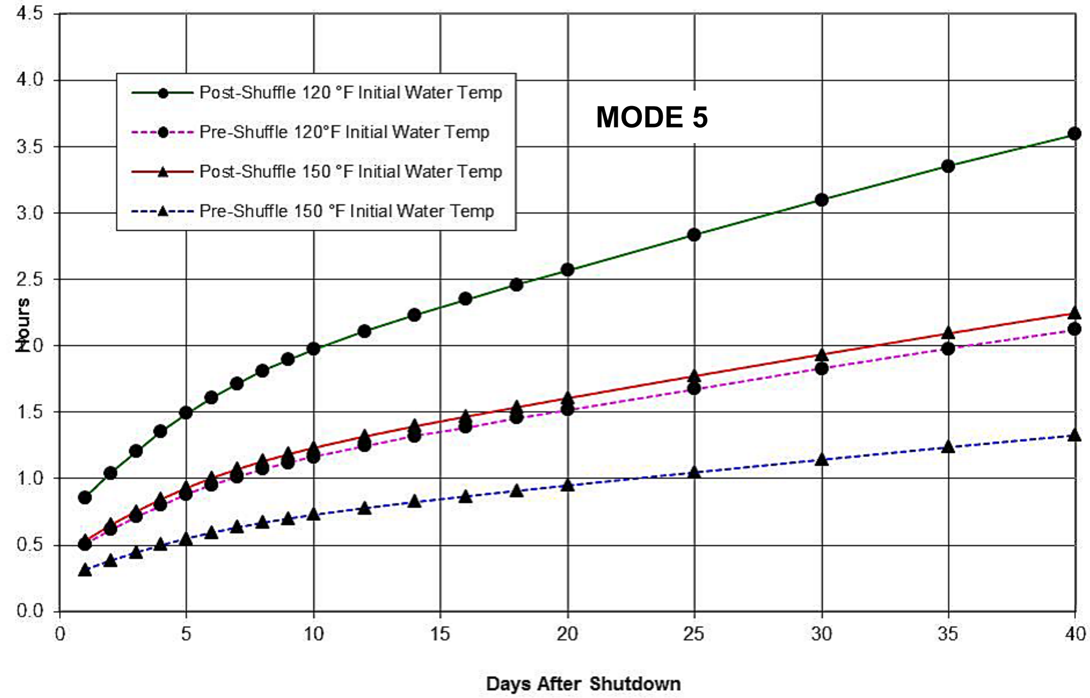


Figure 7: Time for Reactor Vessel to Reach 200°F for Initial Water Level 24" Below Vessel Flange, Mode 5, zero to forty days

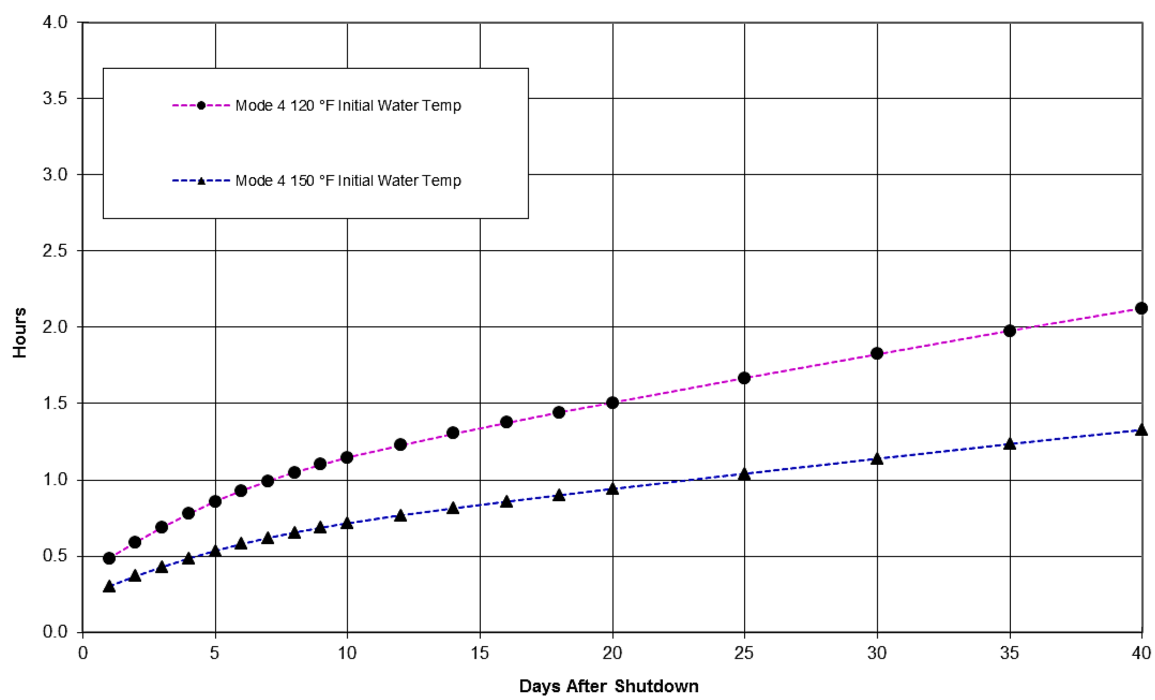


Figure 8: Time for Reactor Vessel to Reach 200°F for Initial Water Level 12" Below Main Steam Line, Mode 4, zero to forty days
(*Assumes Cycle 21 EOC decay heat load*)

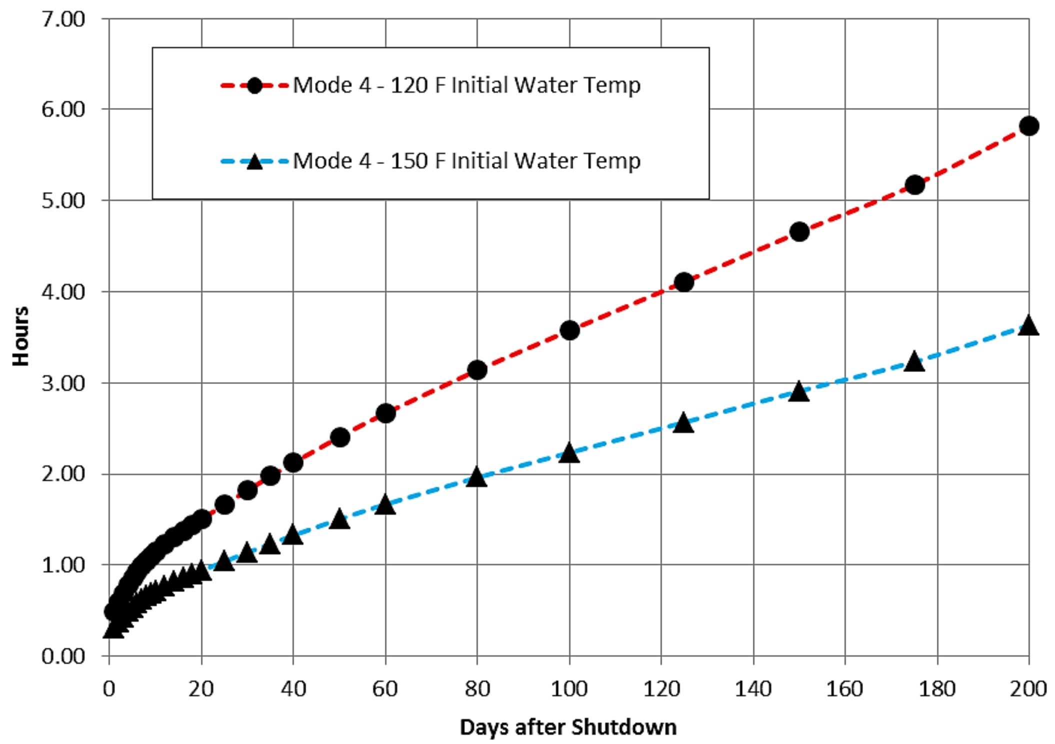


Figure 9: Time for Reactor Vessel to Reach 200°F for Initial Water Level 12" Below Main Steam Line, Mode 4, zero to two hundred days
(*Assumes Cycle 21 EOC decay heat load*)

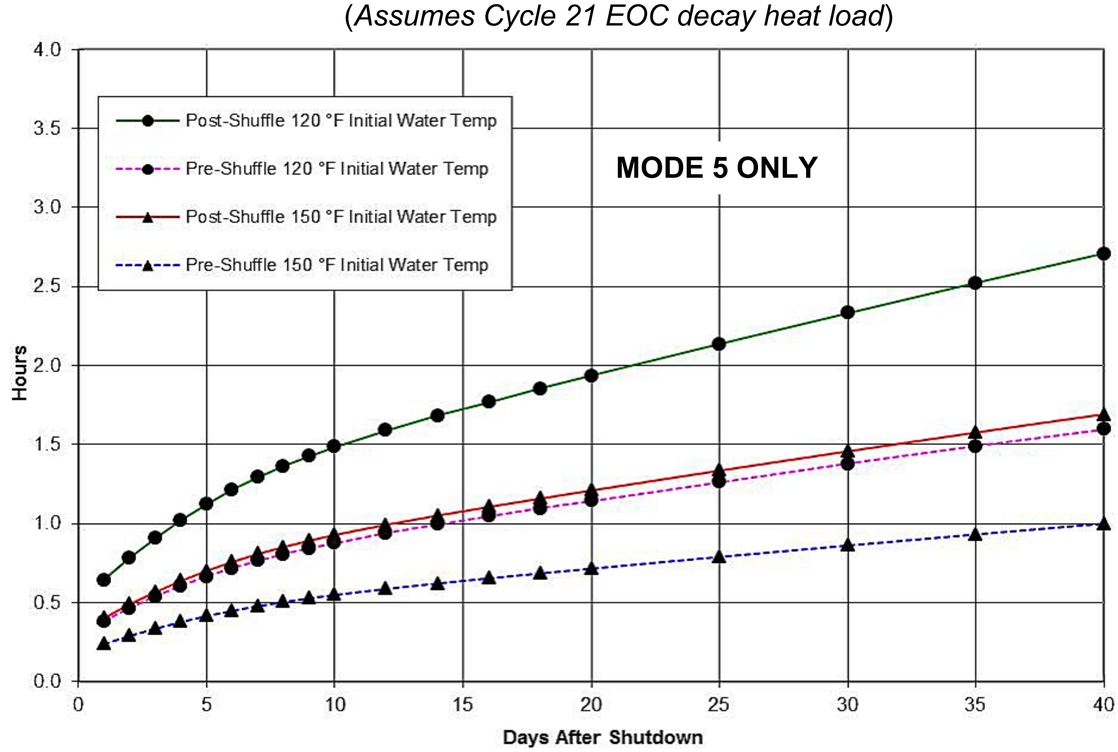


Figure 10: Time for Reactor Vessel to Reach 200°F for Initial Water Level 12" Below Main Steam Line, Mode 5, zero to forty days

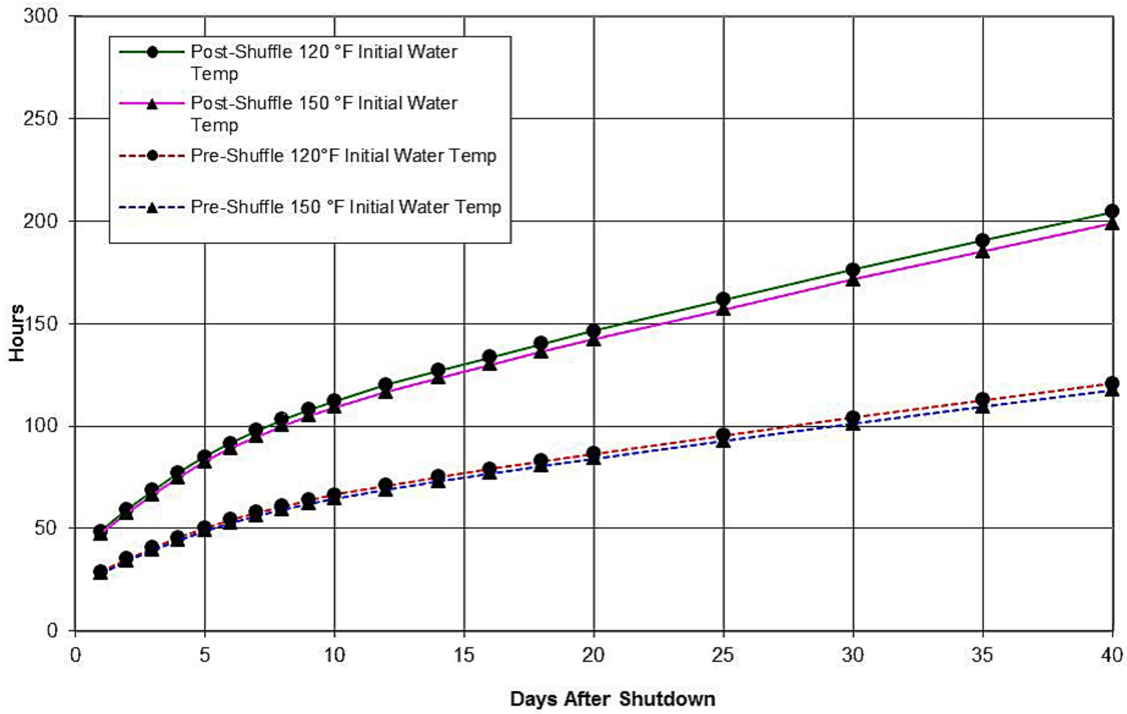


Figure 11: Time to TAF from High Water Level, zero to forty days

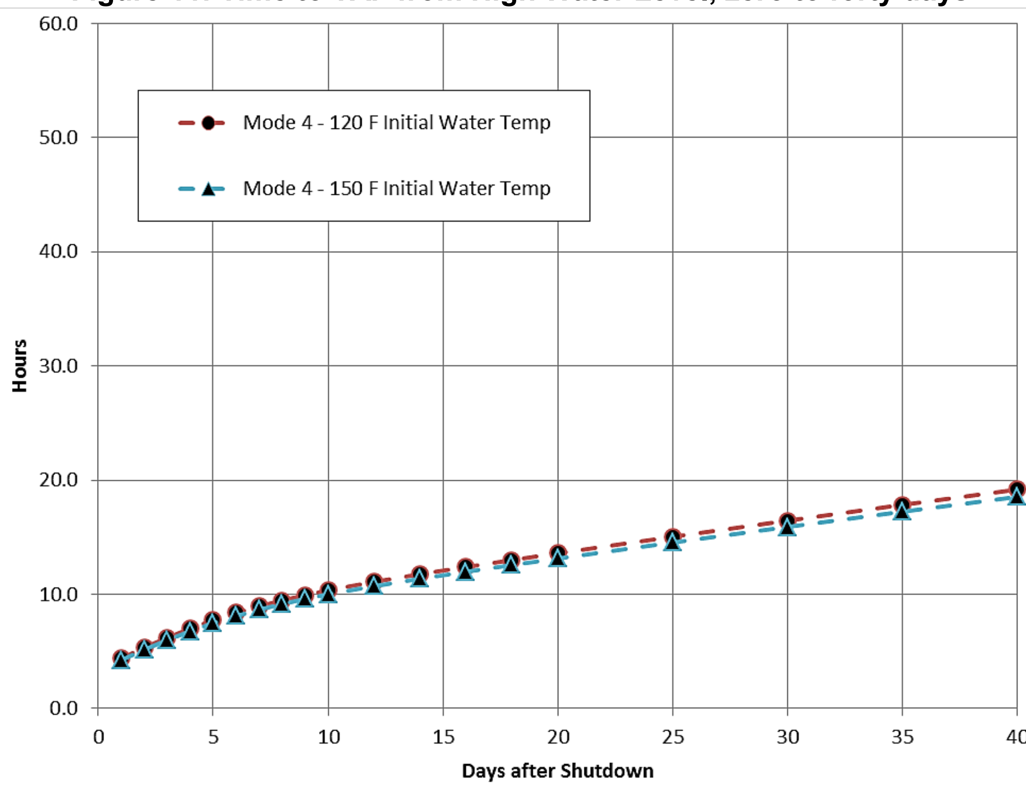


Figure 12: Time to TAF from 32" Normal Water Level, Mode 4, zero to forty days
(*Assumes Cycle 21 EOC decay heat load*)

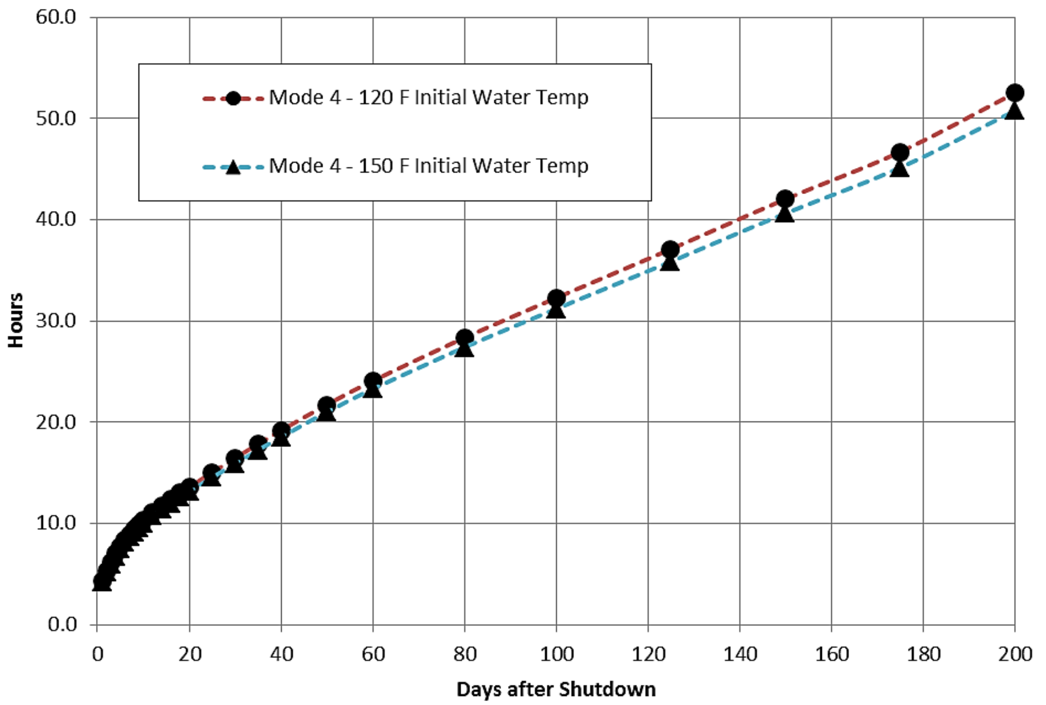


Figure 13: Time to TAF from 32" Normal Water Level, Mode 4, zero to two hundred days (*Assumes Cycle 21 EOC decay heat load*)

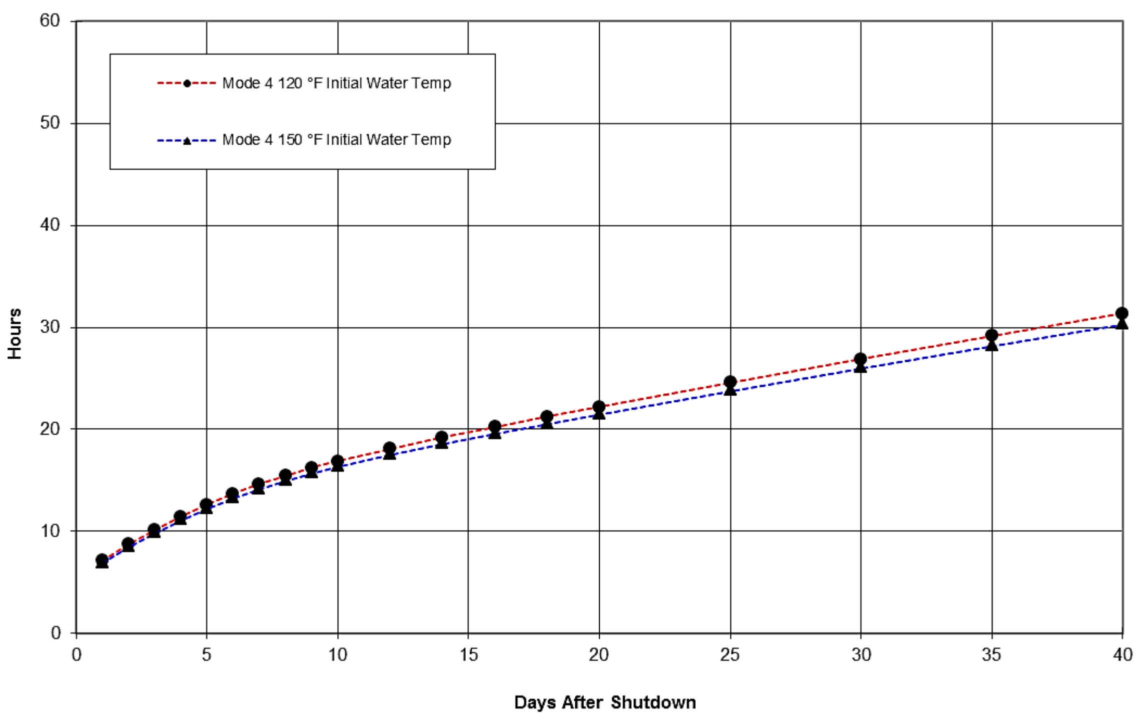


Figure 14: Time to TAF from 24" Below Reactor Vessel Flange, Mode 4, zero to forty days
(*Assumes Cycle 21 EOC decay heat load*)

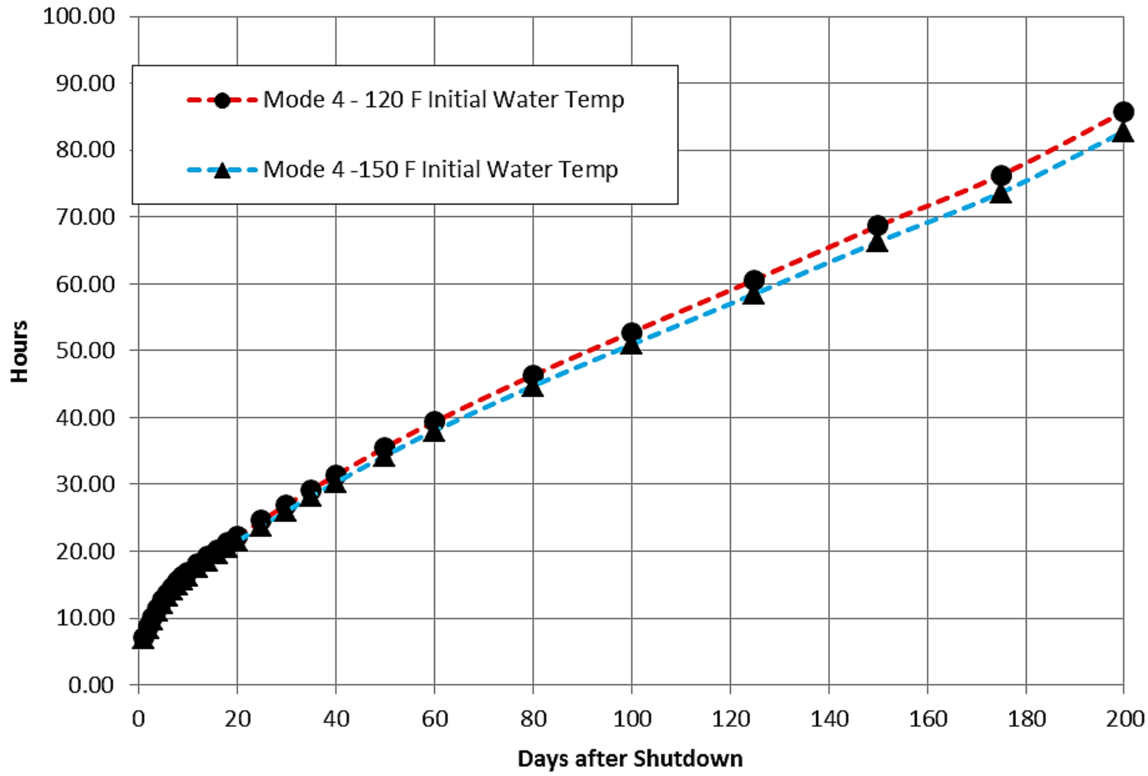


Figure 15: Time to TAF from 24" Below Reactor Vessel Flange, Mode 4, zero to two hundred days (*Assumes Cycle 21 EOC decay heat load*)

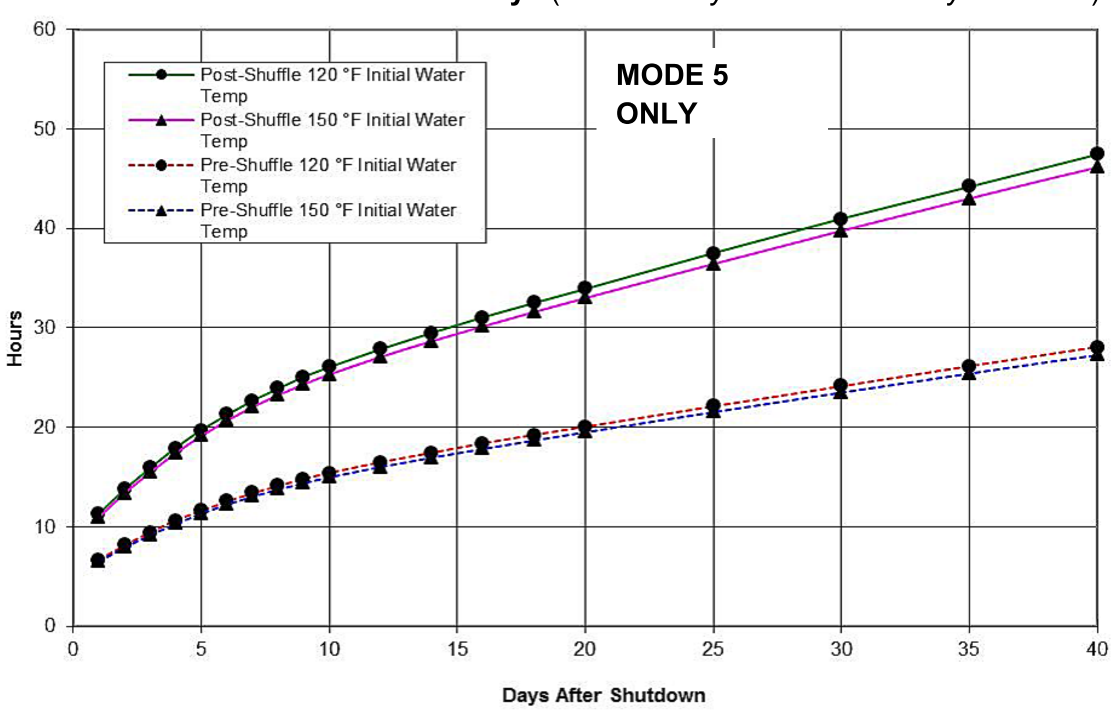


Figure 16: Time to TAF from 24" Below Reactor Vessel Flange, Mode 5, zero to forty days

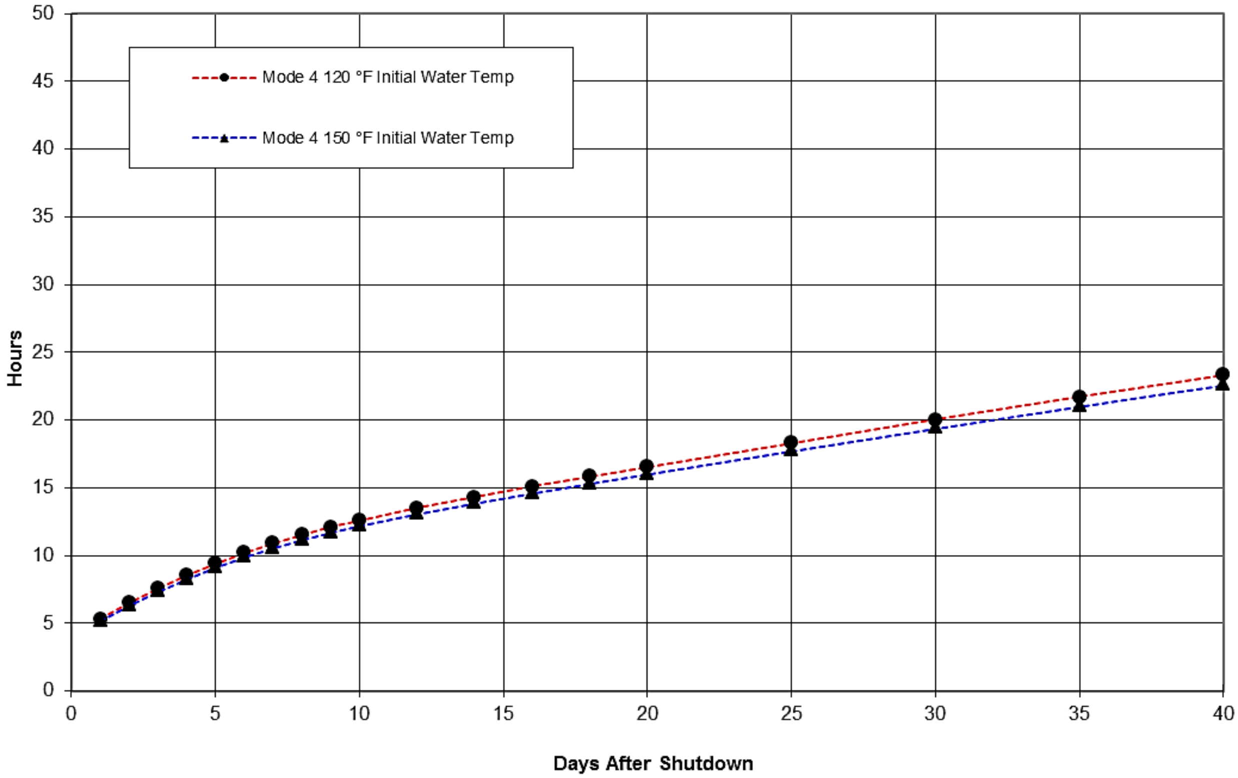


Figure 17: Time to TAF from 12" Below Main Steam Line, Mode 4, zero to forty days
(*Assumes Cycle 21 EOC decay heat load*)

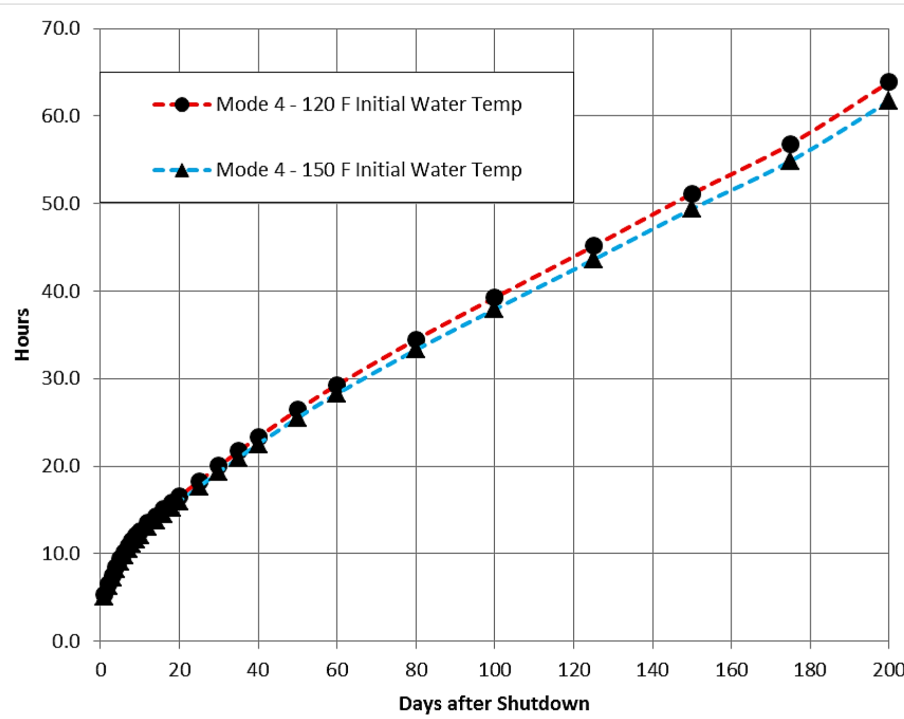


Figure 18: Time to TAF from 12" Below Main Steam Line, Mode 4, zero to two hundred days
(*Assumes Cycle 21 EOC decay heat load*)

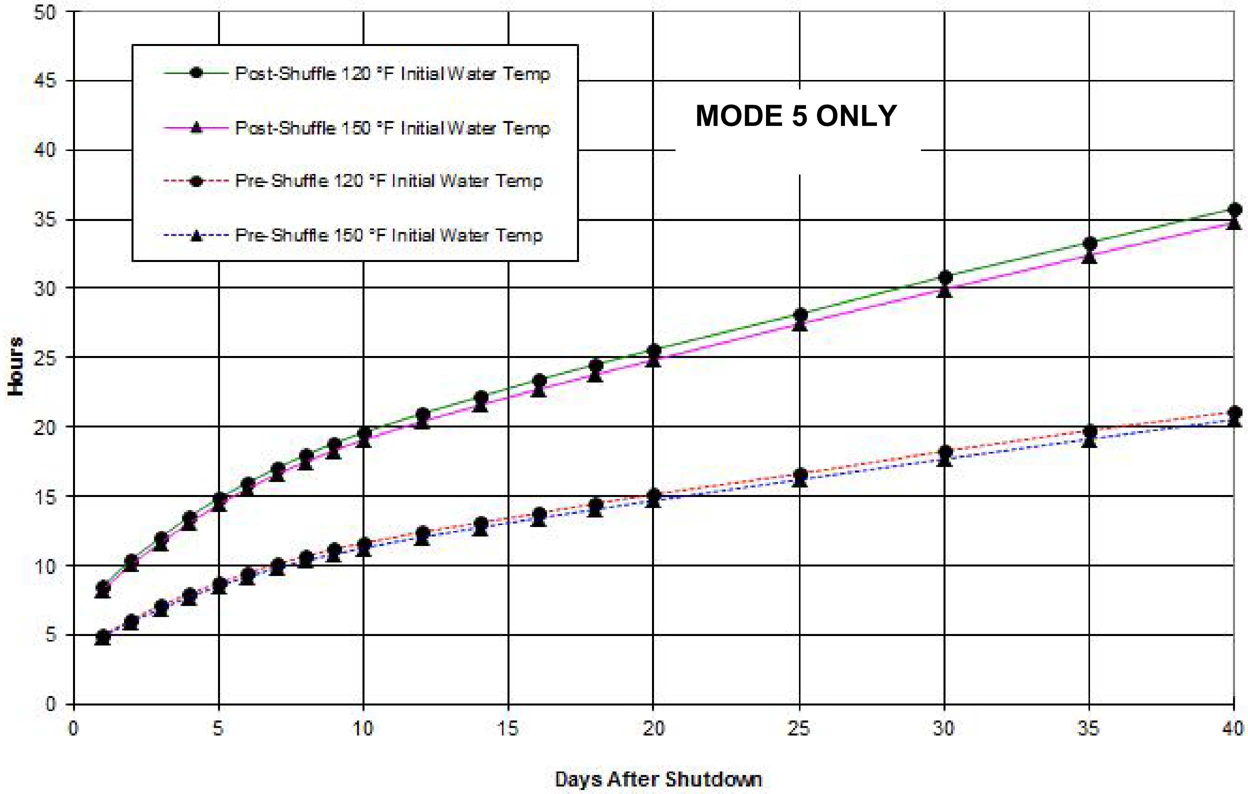
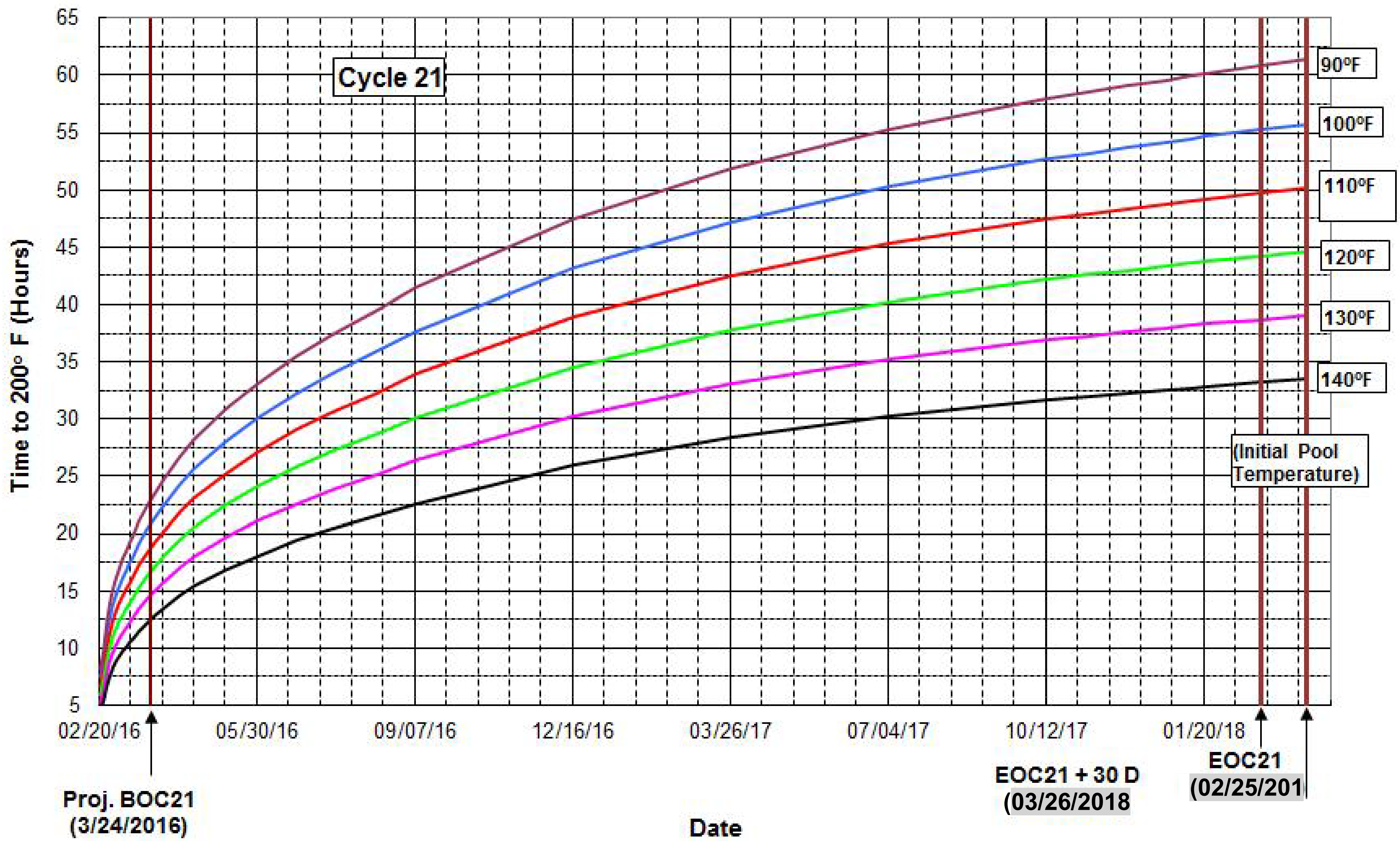


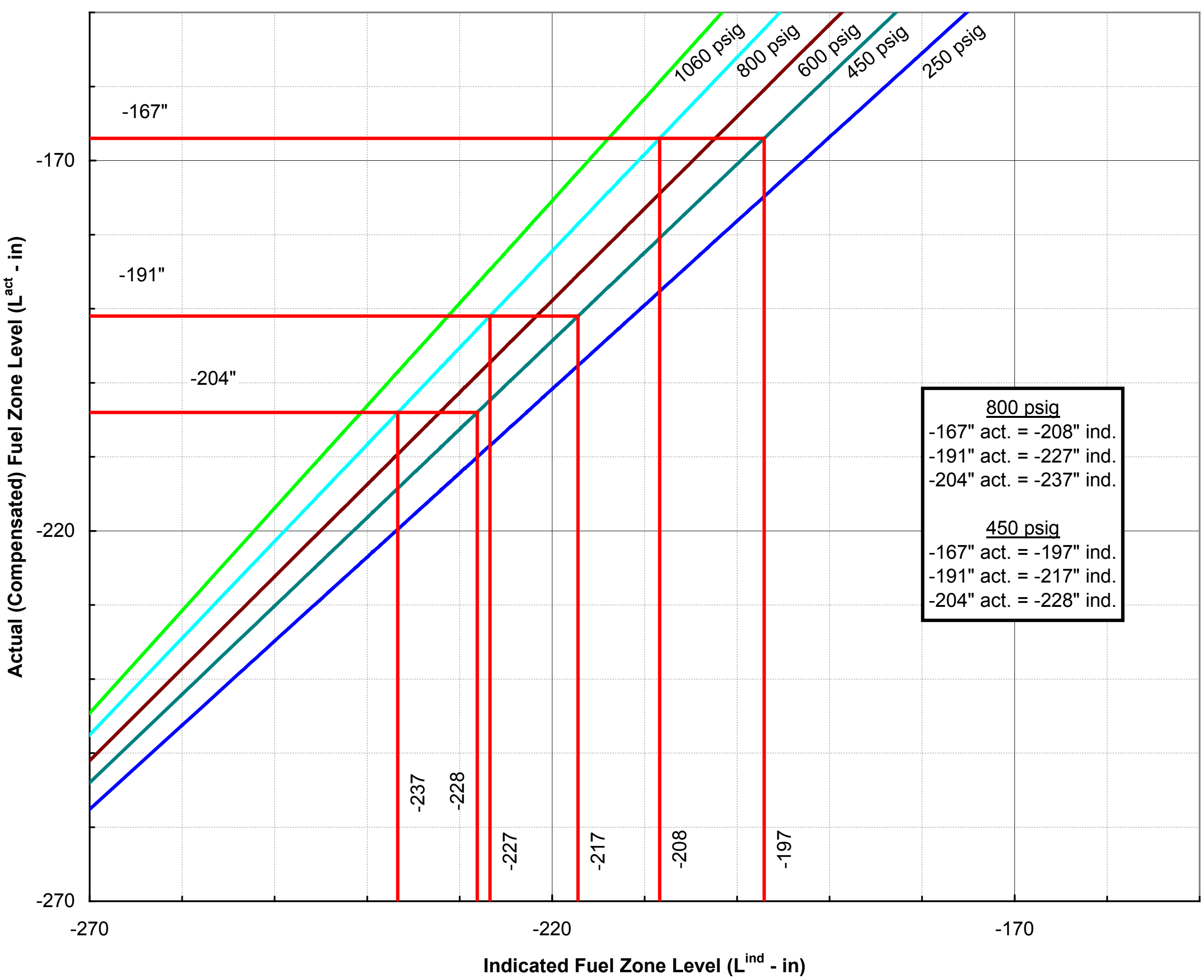
Figure 19: Time to TAF from 12" Below Main Steam Line, Mode 5, zero to forty days



Cycle 21
90ºF
100ºF
110ºF
120ºF
130ºF
140ºF
(Initial Pool Temperature)
Time to 200° F (Hours)
65
60
55
50
45
40
35
30
25
20
15
10
5
02/20/16
05/30/16
09/07/16
12/16/16
03/26/17
07/04/17
10/12/17
01/20/18
Proj. BOC21 (3/24/2016)
EOC21 + 30 D (03/26/2018
EOC21 (02/25/201
Date

TAB 4

Figure 1
Density Compensated Fuel Zone Nomograph



TAB 5

Fig 3

Containment Spray Initiation Pressure Limit (CSIPL)

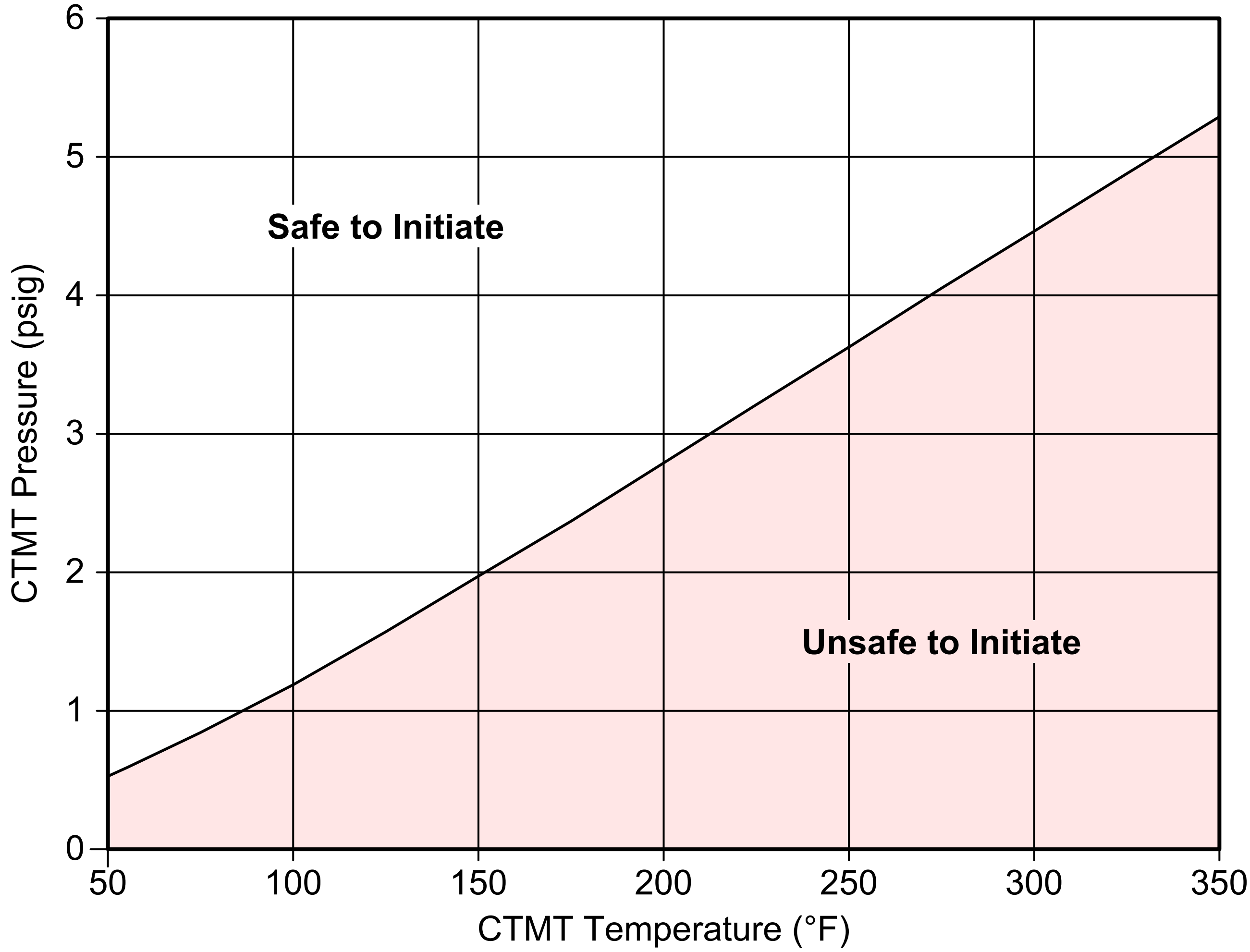
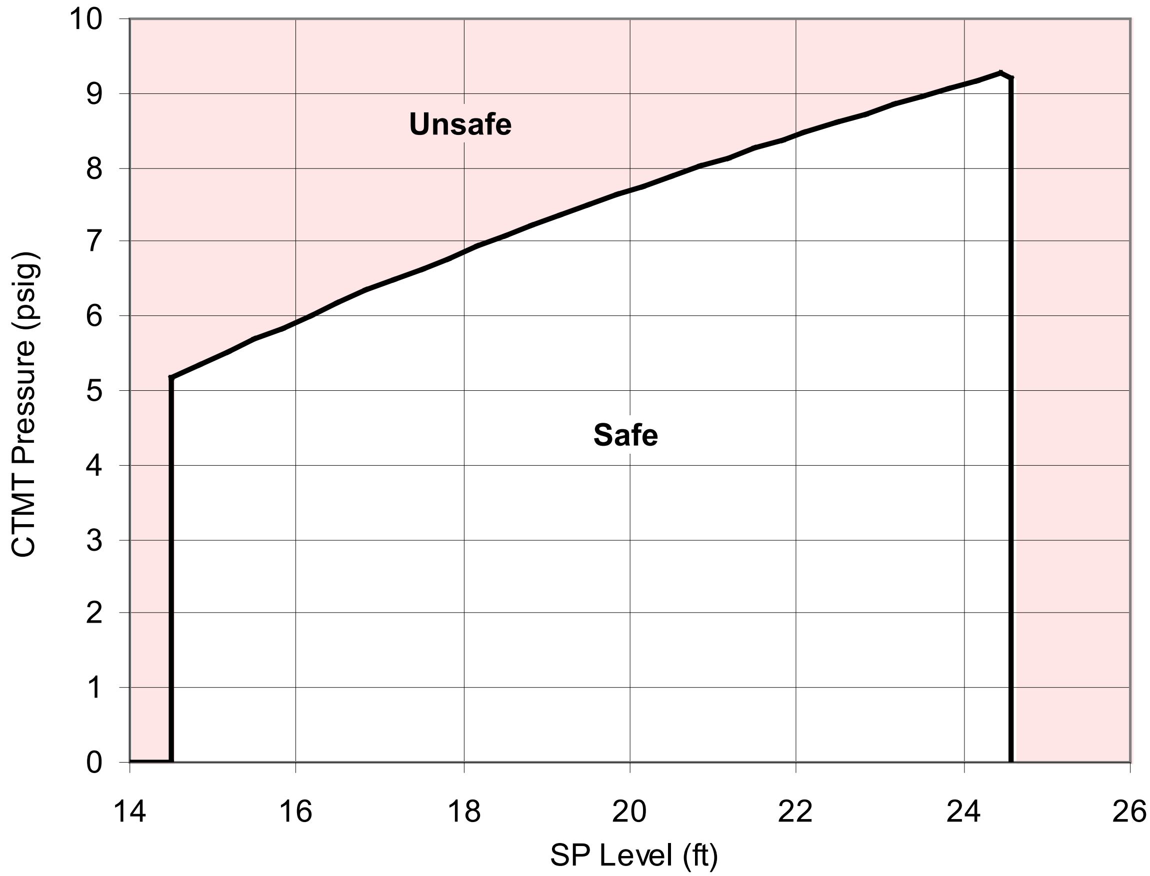
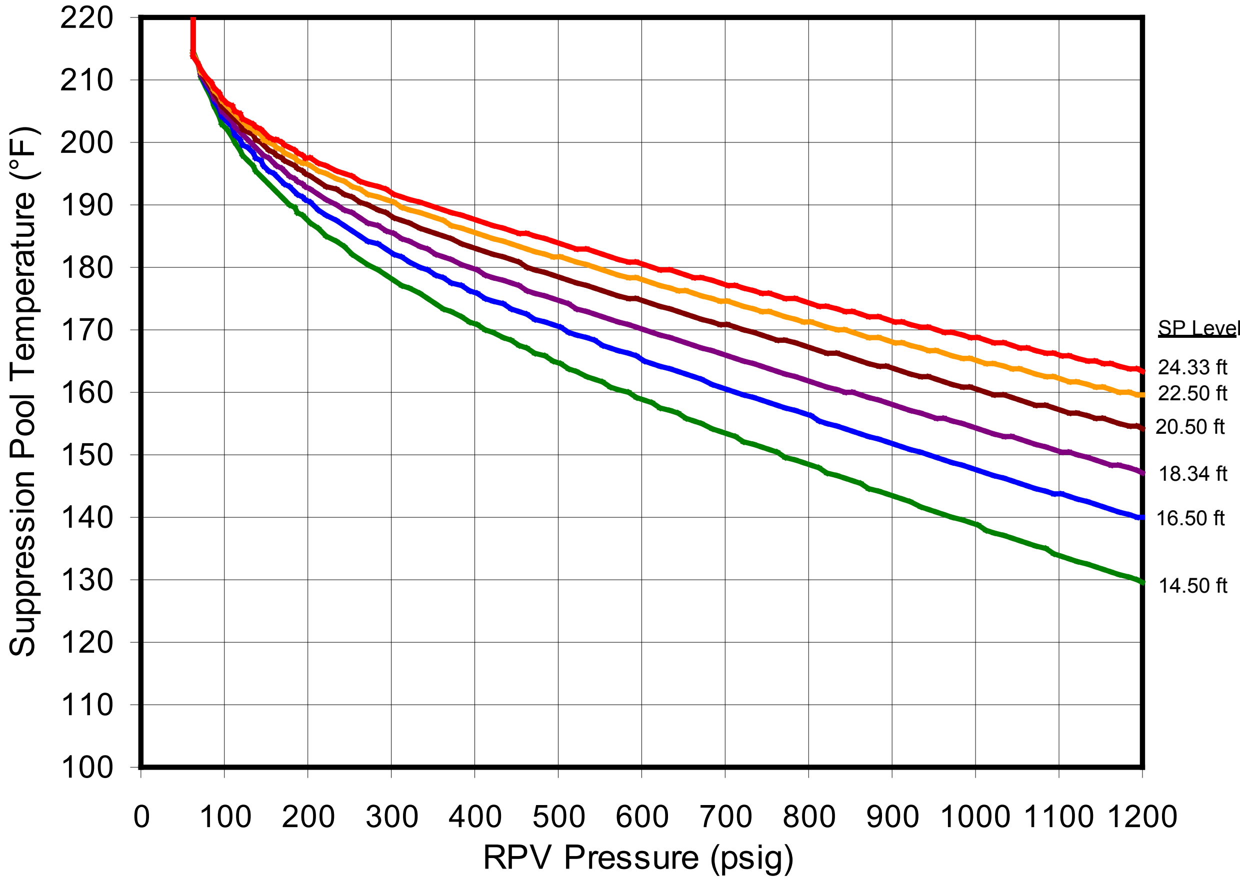


Fig 4
Pressure Suppression Pressure (PSP)



TAB 6

Fig 1
Heat Capacity Temperature Limit (HCTL)



TAB 7

DATA SHEET I
PLANT AC AND DC ELECTRICAL POWER DISTRIBUTION
WEEKLY LINEUP
SAFETY RELATED

(Step 5.1.2) Div 1, 2 & 3 Offsite Feeders

	OFFSITE FEEDER	ENERGIZED YES/NO	VOLTAGE INDICATOR (LOCATION)	RECORDED VOLTAGE (ACCEPTANCE CRITERIA)	FREQUENCY INDICATOR (LOCATION)	RECORDED FREQUENCY (ACCEPTANCE CRITERIA)	INITIALS
$	BAXTER WILSON	*	JACKSON DISPATCHER	___________kV (496-525kV)***	500 kV FREQ SR27-SR-R600 (H13-P807) or Pine Bluff Dispatcher	___________Hz (58.5-61.8Hz)	
$	FRANKLIN	*					
$	115kV LINE PORT GIBSON	*	** 152-1511 152-1611 152-1704	_______x 27.64 =________kV (120.75- 112.13) kV			

* To determine status of offsite feeders, **CONTACT** load dispatcher. **ENSURE** that the feeders are independently energized from the grid, such that the loss of one feeder would **<u>NOT</u>** result in the loss of another

** To determine voltage of the Port Gibson 115kV line, **RECORD** ESF 12 incoming voltage at Bus 15AA, 16AB **<u>OR</u>** 17AC placing the Sync switch for the designated breaker to ON. **MULTIPLY** this reading by 27.64 for equivalent feeder voltage. **RETURN** Sync switch to OFF after taking reading**.**

*** Allowable Value of minimum voltage is $\geq$491 kV for operability of Offsite Feeders. This value is based on analysis of the Class 1E ESF buses **<u>AND</u>** includes an allowance for instrument uncertainty associated with the voltage measurement in the switchyard. Extended operation beyond the normal continuous operating limits <u>Should</u> be evaluated **<u>AND</u>** caution <u>Should</u> be taken when starting large loads under these conditions.

TAB 8

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources—Operating

LCO 3.8.1 The following AC electrical power sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electric Power Distribution System;

b. Three diesel generators (DGs); and

c. Division 1 and Division 2 automatic load sequencers.

APPLICABILITY: MODES 1, 2, and 3.

---------------------------NOTE---------------------------
Division 3 AC electrical power sources are not required to be OPERABLE when High Pressure Core Spray System is inoperable.

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 AND Once per 8 hours thereafter
	AND		
			(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. OR B.3.2 Perform SR 3.8.1.2 for OPERABLE DG(s). AND B.4 Restore required DG to OPERABLE status.	24 hours 24 hours 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. AND C.2 Restore one required offsite circuit to OPERABLE status.	12 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s) 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 $\geq$ 3744 V and $\leq$ 4576 V and frequency $\geq$ 58.8 Hz and $\leq$ 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 9

OFFSITE MONITORING TEAM MAPS

TAB 10

DATA SHEET II
REACTOR WATER CLEANUP SYSTEM VALVE OPERABILITY
VALVE INSERVICE INSPECTION REQUIREMENTS
SAFETY RELATED

PARAMETER	EQUIPMENT MP&L NUMBERS			
Item Number	F028		F034	
Step Number	5.1.4		5.1.4	
Remote Open	I c		I c	
Time-Closed (sec)	$I d		$I d	
Remote Closed	I d		I d	
Min Acceptable Stroke Time-Closed (sec)	I	25.8	I	20.1
Max Acceptable Stroke Time-Closed (sec)	I	34.8	I	27.1
Limiting Value of Full Stroke Time-Closed (sec)	I	35.0	I	30.7
*IST Acceptance Criteria (A, E, **OR** U)	I		I	
Tech Spec (TRM) Limit-Closed (sec)	$	35.0	$	35.0
Tech Spec Acceptance Criteria (A, U)	$		$	
Data Acceptable Verified By				

* Acceptance Criteria:

A Acceptable

E Evaluation Required, **NOTIFY** Shift Supervision Immediately

U Unsatisfactory, **NOTIFY** Shift Supervision

Data Taken By: ______________________________ / __________
Signature Date