QUESTION RO 1

The Plant is operating at 95% rated thermal power. Feedwater heater 6B level switch, 1N25-N0263B, failed, causing 6B Feedwater heater to isolate.

This will cause Reactor Power to __(1)__. The following instructions will be used to mitigate the consequences of any reactivity changes that have occurred __(2)__.

	__(1)__	__(2)__
A.	rise	ONI-N36, Loss Of Feedwater Heating and FTI-B002 Control Rod Movements
B.	rise	ONI-C51, Unplanned Changes in Reactor Power or Reactivity and SOI-B33 Reactor Recirculation System
C.	lower	ONI-N36, Loss Of Feedwater Heating and FTI-B002 Control Rod Movements
D.	lower	ONI-C51, Unplanned Changes in Reactor Power or Reactivity and SOI-B33 Reactor Recirculation System

QUESTION RO 1

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	2.1.43	
	Importance Rating	4.1	

K&A: Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc.

Generic

Explanation: **Answer B** – FW Heater 6B isolation will cause a rise in Rx power. ONI-C51 is entered due to the rise in Rx power and directs lowering Rx power to less than the initial power level per SOI-B33.

A & C – incorrect – No control rod movement (cram rods) is required for a FW heater isolation.

C & D – incorrect – Rx power will rise due to a loss of FW heating

Technical Reference(s): ONI-C51 Flow Chart rev I	Reference Attached: ONI-C51 Flow Chart (partial)

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3035-08(LP) A2

Question Source:	Bank #	
	Modified Bank #	
	New	x

Question History: Previous NRC Exam:

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

10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 2

The following conditions exist:

- The plant has been shutdown for a refuel outage
- Fuel shuffle is in progress
- RHR Shutdown Cooling has been removed from service per the outage schedule for maintenance

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

A. Monitor reactor coolant temperature

B. Authorize commencement of fuel movements

C. Verify required refueling surveillances are current

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

QUESTION RO 2

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	2.1.44	
	Importance Rating	3.9	

K&A: Knowledge of RO duties in the control room during fuel handling such as responding to alarms from the fuel handling area, communication with the fuel storage facility, systems operated from the control room in support of fueling operations, and supporting instrumentation.

Generic

Explanation: **Answer A** – IOI-9, Refueling, requires moderator temperature to be maintained greater than 68 F whether fuel is irradiated or new. Tech Spec Rounds (TS 3.9.8 C.2) monitors for this condition when no RHR loop is in operation.

B and D – incorrect – this is the responsibility of the Unit Supervisor.

C – incorrect – this is the responsibility of the Refueling Supervisor and the Fuel Handling Supervisor

Technical Reference(s): TS 3.9.8 C.2 & IOI-9 rev 26	Reference Attached: TS 3.9.8 C.2 p 3.9.11 & IOI-9 p 7

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3035-12(LP)-E

Question Source:	Bank #	
	Modified Bank #	Perry 2009
	New	
Question History:	Previous NRC Exam:	Perry 2009
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 3

The following sequence of events occurred in rapid succession:

1. The plant was operating at 100% rated thermal power
2. An MSIV isolation occurred
3. Reactor pressure peaked at 1115 psig
4. The reactor scrammed on high drywell pressure
5. Safety Relief Valves (SRV's) cycled to relieve pressure during this event
6. No other scram signals have been actuated

Based on these conditions, RPS __(1)__ OPERABLE and the SRV's __(2)__ OPERABLE.

	__(1)__	__(2)__
A.	is	are <u>not</u>
B.	is	are
C.	is <u>not</u>	are <u>not</u>
D.	is <u>not</u>	are

QUESTION RO 3

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	2.2.37	
	Importance Rating	3.6	

K&A: Ability to determine operability and/or availability of safety related equipment.

Generic

Explanation: **Answer D –** RPS is inop due to no scram on High Rx Pressure and no scram on MSIV closure. SRV's are operable due to opening when RPV pressure > 1103 psig.

A & B – incorrect – RPS is inoperable (two functions)

A & C – incorrect – SRV's required to open on Relief at 1103 ± 15 psig

Technical Reference(s): TS 3.3.1.1 & 3.3.6.4 and ARI-H13-P680-A8 rev 12, & ARI-H13-P680-A6 rev 12	Reference Attached: TS 3.3.1.1 pp. 3.3-1&8 3.3.6.4 pp. 3.3-68-69 and ARI-H13-P680-A8 p12, & ARI-H13-P680-A6 p 15

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-C71-P.E & OT-COMBINED-B21_N11-H

Question Source:	Bank # Modified Bank # New	Peach Bottom 2008
Question History:	Previous NRC Exam:	Peach Bottom 2008
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 4

The following conditions exist:

- A plant startup is in progress
- Reactor power is above the Low Power Setpoint
- SVI-C11-T1022, Rod Pattern Control System – Rod Withdrawal Limiter is in progress
- The Rod Withdrawal Limiter failed to inhibit withdrawal at the 4-notch limit

Which one of the following describes the required actions to be taken?

A. Immediately suspend control rod withdrawal.

B. Within one (1) hour take action to insert all control rods.

C. Immediately suspend control rod movement except by scram only.

D. Within one (1) hour take action to be in MODE 2 within 7 hours.

QUESTION RO 4

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#		2.2.39
	Importance Rating	3.9	

K&A: Knowledge of less than or equal to one hour Technical Specification action statements for systems.

Generic

Explanation: **Answer A –** per TS 3.3.2.1 Condition A, this is the correct action.

B – incorrect – this is the required action for more than 2 SRM's INOP in modes 3 or 4 - this is not a required action for an INOP RWL

C – incorrect – this is the required action for an INOP rod pattern controller

D – incorrect – this is the required action when the 1-hour action has not been met for TS 3.3.4.2 & 3.3.6.1 - not a required action for INOP RWL

Technical Reference(s): TS 3.3.2.1 & svo-C11-T1022 rev 11	Reference Attached: TS 3.3.2.1 p 3.3-15 & SVI-C11-T1022 p 2

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-C11_RCIS-J.5

Question Source:	Bank #	INL-0971
	Modified Bank #	
	New	

Question History: Previous NRC Exam:

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

10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 5

The plant was operating at rated power when a scram was inserted.

An Anticipated Transient Without a Scram (ATWS) occurred resulting in the following conditions:

- Injection Systems: Terminated and Prevented IAW EOP 1A Level Power Control
- SRVs: 8 ADS valves are open IAW EOP 04-2 Emergency Depressurization
- Reactor Pressure: 200 psig and lowering
- Reactor Level: -20" and lowering
- Reactor Power: 2% and lowering

Which of the following identifies if a Technical Specification Safety Limit value has been exceeded <u>and</u> if adequate core cooling currently exists?

A Technical Specification Safety Limit value __(1)__ been exceeded and IAW EOP Bases, adequate core cooling __(2)__ exist.

	__(1)__	__(2)__
A.	has	does
B.	has	does not
C.	has not	does
D.	has not	does not

QUESTION RO 5

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#		2.2.42
	Importance Rating	3.9	

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

Generic

Explanation: **Answer A** – The Reactor Vessel Water Level SL (2.1.1.3) is applicable in all Modes. RPV water level must be maintained greater than or equal to zero inches. Greater than 0" is also considered Adequate Core Cooling. EOP's direct lowering RPV water level, but the SL is still violated. Per EOP Bases, Adequate Core Cooling is also achieved by maintaining RPV pressure > Minimum Steam Cooling pressure. With 8 SRVs open, this would be any pressure higher than 140 psig. The Safety Limit is violated but Adequate Core Cooling currently exists.

B & D – incorrect – Adequate Core Cooling currently exists

C & D – incorrect – the Reactor Vessel Water Level SL has been exceeded

Technical Reference(s): Tech Spec 2.1, TS 2.1.1.3 Bases rev 4 & EOP-1A Bases rev 1	Reference Attached: Tech Spec 2.1 p 2.0-1, TS 2.1.1.3 Bases p B 2.0-4 & EOP-1A Bases pp. 55-56

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3037-08 B

Question Source:	Bank # Modified Bank # New	Hatch 2009
Question History:	Previous NRC Exam:	Hatch 2009
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 6

As the Field Supervisor, you must assign non-licensed operators to perform a surveillance test on the Fuel pool Cooling Cleanup system (FPCC). Per PAP-0114, Radiation Protection Program, you must maintain dose ALARA.

The following conditions exist for a job to be performed.

- The general area radiation levels are 10 mrem/hr in the room
- The hot spot in the heat exchanger room is a pipe elbow that has a radiation level of 100 mrem/hr
- The job will be performed near the hot spot area

Additional information:

- <u>All</u> 4 cases below have the same transition time to and from destinations
- Dose rate for all shielding placement and removal is at 100 mrem/hr
- Dose rate for the hot spot with shielding in place is 10 mrem/hr

Which method complies with ALARA procedural requirements for performance of the surveillance?

A. The job is performed using 2 operators for 3 hrs each on the job at the hot spot.

B. The job is performed using 3 operators for 1 hr each on the job at the hot spot and a fourth operator reading instructions in the general room area for 1 hr.

C. The job is performed using 2 operators for 2 hrs each on the job at the hot spot and a third operator reading instructions in the general room area for 2 hrs.

D. The placement and removal of lead shielding on the hot spot is performed by 2 Radiation Protection personnel in 1.5 hours. The job is performed after the lead shielding is in place using 2 operators for 3 hrs each on the job.

QUESTION RO 6

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

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

Generic

Explanation: **Answer B** – > The job is performed by 3 operators for 1 hr each on the job at the hot spot and a fourth operator reading instructions in the general room area for 1hr. (3 operators x 100 mrem/hr x 1 hr) + (1 operator x mrem/hr x 1 hr) = 310 mrem

A – incorrect - The job is performed using 2 operators for 3 hrs each on the job at the hot spot - (2 operators x 100 mrem/hr x 3 hrs) = 600 mrem

C – incorrect - The job is performed by 2 operators for 2 hrs each on the job at the hot spot and a third Operator reading instructions in the general room area for 2 hrs - (2 operators x 100 mrem/hr x 2 hr) + (1 operators X 10 mrem/hr) = 420 mrem

D – incorrect - Two RP personnel hang and remove lead shielding on the hot spot in 1.5 hours - The job is performed after the lead shielding is in place using 2 operators for 3 hrs each on the job. (2 RP techs x 100 mrem/hr x 1.5hrs) + (2 operators x 10 mrem/hr x 3 hr) = 360 mrem

Technical Reference(s): PAP-0114 rev 15	Reference Attached: PAP-0114 p 15

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): None

Question Source:	Bank # Modified Bank # New	Hope Creek 2009
Question History:	Previous NRC Exam:	Hope Creek 2009
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 x 55.43	

Comments: Level of Difficulty = x

QUESTION RO 7

The plant is operating at 100% power.

Offgas Post Treatment Radiation monitors 1D17-K601A and K601B have alarmed on a High-High Radiation signal (OG ISOL OG POST-TREAT PRCS RAD A/B 3XHI – H13-P604-0001-A5).

Which one of the following describes the effect on Offgas and the Main Condenser?

A. Only the Offgas charcoal adsorber's inlet valves will isolate causing a loss of main condenser vacuum.

B. Offgas will shift into the Bypass Mode of operation causing a loss of main condenser vacuum.

C. Offgas will shift to Treat Mode allowing main condenser vacuum to remain constant.

D. Offgas System will isolate causing a loss of main condenser vacuum.

QUESTION RO 7

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#		2.3.15
	Importance Rating	2.9	

K&A: Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Generic

Explanation: **Answer D** – a Hi-Hi signal from both rad monitors causes the Offgas system to isolate. When OG isolates, main condenser vacuum will degrade.

A – incorrect – OG Discharge Isolation, Cooler Condenser, Prefilter Inlet, and Holdup Line drain valves all isolate

B – incorrect – OG will not auto shift to Bypass Mode

C – incorrect – OG should be shifted from Auto Mode to Treat Mode at 5% power

Technical Reference(s): ARI-H13-P604-001-A5 rev 5	Reference Attached: ARI-H13-P604-001-A5 p 13

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-N64-M.6

Question Source:	Bank # Modified Bank # New	River Bend 2008
Question History:	Previous NRC Exam:	River Bend 2008
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 8

The following conditions exist:

- The plant scrammed on high reactor pressure
- The reactor is shutdown
- All feedwater pumps tripped
- Level control is on Reactor Core Isolation Cooling
- Pressure control is on the Bypass Valves

The assigned level band should be __(1)__, but can be expanded if other __(2)__ actions have a higher priority.

	__(1)__	__(2)__
A.	150” to 219”	EOP
B.	150” to 219”	ONI
C.	178” to 219”	EOP
D.	178” to 219”	ONI

QUESTION RO 8

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#		2.4.14
	Importance Rating	3.8	

K&A: Knowledge of general guidelines for EOP usage.

Generic

Explanation: **Answer C** – both EOP-1 and ONI-C71-1 direct a level band of 178-219”. Only the EOP allows expanding the level band (General Guidelines). Scram on high reactor pressure is an EOP entry.

A – incorrect – this is the expected level band when pressure control is on SRV’s.

B – incorrect – ONI’s do not allow expansion of the level band, they direct entry into the EOP’s.

D – incorrect – correct level band, but the ONI’s do not allow expansion of the level band.

Technical Reference(s): EOP-1 Bases rev 1 & EOP-1 Chart rev B.	Reference Attached: EOP-1 Bases p 27 & EOP-1 Chart (partial)

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): x

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 9

The reactor is at 96% rated thermal power with a plant down power in progress for monthly surveillance testing.

The following occurs:

- ANN PWR SUPPLY FAIL is illuminated
- Alarms that were locked in on H13-P601 have deactivated

The Control Room actions would be to dispatch an operator to __(1)__ to investigate and __(2)__.

	1	2
A.	D-1-A	suspended power maneuvering
B.	D-1-A	continue lowering reactor power
C.	ED-1-A	suspended power maneuvering
D.	ED-1-A	continue lowering reactor power

QUESTION RO 9

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	2.4.32	
	Importance Rating	3.6	

K&A: Knowledge of operator response to loss of all annunciators.

Generic

Explanation: **Answer A** – a loss of D-1-A-06 will cause ANN PWR SUPPLY FAIL window to illuminate and other annunciator windows to deactivate. Operator actions are to maintain the plant stable – suspend power maneuvering.

B & D – incorrect –the action to continuing to lower power is wrong

C & D – incorrect – ED-1-A does supply optical isolators which could cause the annunciator windows on H13-P601 to deactivate, but it will not cause ANN PWR SUPPLY FAIL window to illuminate

Technical Reference(s): ARI-H13-P680-007-E5 rev 17 & ONI-R61 rev 3	Reference Attached: ARI-H13-P680-007-E5 p 137 & ONI-R61 p 4

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3035-05(LP)-A.5

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 10

Which one of the following identifies a portion of a Plant Communication system that is reserved for emergency and Control Room communications, per PAP-0202, Communications?

A. Extension 5634 of the Private Branch Exchange (PBX) Phone System.

B. Channel 5-G of the RMT (CEI 800-MHx Trunked) System.

C. Line 5 of the Public Address (PA) System.

D. Channel F2 of the Plant Radio System.

QUESTION RO 10

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	2.4.43	
	Importance Rating	3.2	

K&A: Knowledge of emergency communications systems and techniques.

Generic

Explanation: **Answer C** – per PAP-0202, Section 3.11, the PA system line 5 is reserved.

A – incorrect – This is the PBX extension for the RO-ATC desk, which is used for all types of communications (not just emergency communications)

B – incorrect – This communication system is used by Off-Site Radiation Monitoring Teams for communications with the TSC and EOF

D – incorrect – This is the Radio frequency reserved for use by Maintenance personnel

Technical Reference(s): PAP-0202 rev 4	Reference Attached: PAP-0202 p 5

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3039-01.B

Question Source:	Bank #	Perry 2003
	Modified Bank #	
	New	
Question History:	Previous NRC Exam:	Perry 2003
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 11

The plant was operating at 100% rated thermal power when both Reactor Recirculation Pumps down shifted.

The plant is now operating in the Immediate Exit Required region of the OPRM Operable - Two Loop Power - Flow Map.

Which of the following will <u>raise</u> the likelihood of power oscillations?

A. Raise core flow.

B. Control rod insertion.

C. Feedwater temperature reduction.

D. Lower pressure regulator setpoint.

QUESTION RO 11

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295001	AK1.04
	Importance Rating	2.5	

K&A: Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Limiting cycle oscillation: Plant-Specific

Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4

Explanation: **Answer C** – lowering feedwater temperature can cause increase in reactor power and increase the likelihood of power oscillations. For this reason, RPV level is lowered during an ATWS.

A – incorrect – raising core flow lowers the possibility of power oscillations

B – incorrect – inserting control rods to lower power lowers the possibility of power oscillations

D – incorrect – reducing the pressure regulator setpoint will not have any effect on power oscillations

Technical Reference(s): EOP Bases rev 0	Reference Attached: EOP Bases p 6

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): x

Question Source:	Bank #	INL-1312
	Modified Bank #	
	New	
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 12

A Total Loss of AC power (TLAC) is in progress.

ONI-SPI D1, Maintaining System Availability, directs that the Division 3 Battery Room door be opened within two (2) hours.

Which of the following describes the location <u>and</u> the specific reason given for performing this action?

A. Control Complex 620': Dissipate Heat

B. Control Complex 620': Prevent Hydrogen build up

C. Control Complex 638': Dissipate Heat

D. Control Complex 638': Prevent Hydrogen build up

QUESTION RO 12

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295003	AK2.01
	Importance Rating	3.2	

K&A: Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF A.C. POWER and the following: Station batteries

Partial or Complete Loss of AC / 6

Explanation: **Answer A** – per ONI-SPI- D-1, battery room doors are opened to dissipate heat.

B & D – incorrect – hydrogen buildup is a concern when designing battery room ventilation, but it is not the reason given in ONI-SPI D1 for opening the door within two hours

C & D – incorrect – Control Complex 638' has Div 1 & 2 battery rooms, but not the Div 3 battery room

Technical Reference(s): ONI-SPI-D-1 rev 0 & ONI-R10 flowchart rev B	Reference Attached: ONI-SPI-D-1 p 3 & ONI-R10 flowchart

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3035-18(LP) A.4

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 13

The plant was operating at 80% rated thermal power.

Then, due to a system fault, power is lost to 125 VDC Bus ED-1-A.

Which of the following loses power <u>and</u> what is the immediate operator action?

A. High Pressure Core Spray (HPCS) Logic – Verify RCIC Operable

B. Low Pressure Core Spray (LPCS) Logic – Rack-out breaker EH1111 for LPCS pump

C. Residual Heat Removal (RHR) C Logic – Place RHR C loop on Alternate Keep-fill

D. Reactor Recirculation Breaker 3 Control Logic – Insert a manual reactor scram

QUESTION RO 13

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295004	AK3.03
	Importance Rating	3.1	

K&A: Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: Reactor SCRAM:

Partial or Total Loss of DC Pwr / 6

Explanation: **Answer D** – ED-1-A supplies control logic for Recirc Breakers 3A & 3B. When power is lost, Recirc breakers 5A & 5B trip open. Per ONI-C51, with no Recirc pumps running, place Mode Switch in Shutdown.

A – incorrect - HPCS logic is powered from ED1C

B – incorrect - LPCS logic is powered from ED1A, however, there are no required operator actions to rack-out the LPCS pump breaker

C – incorrect - RHR C logic is powered from ED1B

Technical Reference(s): ONI-R42-1 rev 7 & ONI-C51 rev 24	Reference Attached: ONI-R42-1 p 4 & ONI-C51 p 6

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-R42-F & OT-3035-05(LP)-A.1

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 14

With the reactor at 100% power, which of the following conditions will result in a reactor scram and a direct automatic transfer of the Recirculation Pumps from fast speed to slow speed?

A. Main turbine trip

B. Reactor feedwater pump trip

C. Drywell pressure high - 1.68 psig

D. Reactor water level high - Level 8

QUESTION RO 14

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295005	AA1.01
	Importance Rating	3.1	

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

Main Turbine Generator Trip / 3

Explanation: **Answer A** – A main turbine trip from >38% power will initiate a reactor scram and EOC-RPT logic will initiate a downshift of Recirc pumps. This is based on the MT Stop valve position (direct)

B – incorrect – RFPT trip will cause a FCV runback when RPV hits L4

C – incorrect – DW pressure high will cause a Rx scram, but not a direct down shift of Recirc pumps – the subsequent lowering of feedwater flow will cause a RR Pump downshift after a time delay

D – incorrect – RPV water level high will cause a Rx scram, but not a direct down shift of Recirc pumps – the subsequent lowering of feedwater flow will cause a RR Pump downshift after a time delay

Technical Reference(s): ONI-N32 rev 9 & ARI-H13-P680-004-A3 rev 14	Reference Attached: ONI-N32 p 4 & ARI-H13-P680-004-A3 p 9

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-B33-E.3

Question Source:	Bank # Modified Bank # New	INL-1294
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 15

The following plant conditions exist:

- RCIC was running for testing
- RHR loop A was running in Suppression Pool Cooling mode to support RCIC testing

The following then occurred:

- The Control Room was evacuated due to toxic gas
- All Immediate Actions of ONI-C61, Evacuation of the Control Room, have been completed
- No other operator actions have been performed.

Which of the following describes the Suppression Pool Temperature instrumentation available on the Remote Shutdown Panel(s) that is(are) indicating accurate Suppression Pool Temperature, if any, at this time?

A. No Remote Shutdown Panel Suppression Pool temperature instrumentation.

B. Only the Division 1 Remote Shutdown Panel Suppression Pool temperature instrumentation.

C. Only the Division 2 Remote Shutdown Panel Suppression Pool temperature instrumentation.

D. Both the Division 1 and Division 2 Remote Shutdown Panel Suppression Pool temperature instrumentation.

NRC Exam - 2010

QUESTION RO 15

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295016	AA2.04
	Importance Rating		

K&A: Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT: Suppression pool temperature

Control Room Abandonment / 7

Explanation: **Answer C** – Division 2 Remote Shutdown Panel instruments do not have transfer switches and are normally energized.

A – incorrect – Div 2 panel instrumentation is providing accurate information

B – incorrect – Div 1 panel instruments must be transferred to energize the instruments, this action has not been done yet

D – incorrect – same as B

Technical Reference(s): SDM-C61 rev 6	Reference Attached: SDM-C61 p 10

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3035-13(LP)-A.5

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 16

The following plant conditions existed at 08:00:

- The plant was operating at 100% rated thermal power
- NCC pumps A & B were running
- NCC pump C was tagged out for maintenance
- NCC C heat exchanger was in dry layup
- NCC B heat exchanger was removed from service due to indications of a tube leak

At 09:30, NCC temperature control valve (NCC HX A TCV) P43-F006A failed.

The Unit Supervisor entered ONI-P43, Loss Of Nuclear Closed Cooling.

The following conditions currently exist:

- Validated SPDS Drywell temperature is rising at 0.5^{o}F/minute.
- Average Drywell temperature is 138^{o}F
- Highest Drywell temperature is 148^{o}F
- Suppression Pool Average temperature is 96^{o}F rising slowly
- Suppression Pool Level is 18.1' rising slowly
- Containment pressure is 0.3 psig stable
- Drywell pressure is 0.6 psig rising slowly
- Containment Average temperature is 87^{o}F rising slowly

Based on these conditions, the following action are required immediately.

A. Insert a manual reactor scram.

B. Continue with ONI-P43 actions only.

C. Suspend ONI-P43 actions and enter EOP-2 Primary Containment Control.

D. Continue with ONI-P43 actions and enter EOP-2 Primary Containment Control.

QUESTION RO 16

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295018	2.4.8
	Importance Rating	3.8	

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

Partial or Total Loss of CCW / 8

Explanation: **Answer D** – Entry into EOP-2 is required based on Suppression Pool temperature. EOPs are the higher tier documents, but ONI actions should be taken if they don not interfere with EOP's.

A – incorrect – this action is only appropriate if all NCC flow was lost after lowering flow to 58 mlbm/hr

B – incorrect –entry condition for EOP-2 is met

C – incorrect – it is inappropriate to suspend ONI-P43 actions since entry conditions for ONI-P43 still exist

Technical Reference(s): EOP Bases rev 0 & EOP-2 Bases rev 0	Reference Attached: EOP Bases p 27 & EOP-2 Bases pp. 6 & 15

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3035-16(LP)-A.3

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 17

The following plant conditions exist:

- The reactor is operating at 50% power
- The Service Air and Instrument Air Systems are in their normal lineup

An air leak occurs resulting in the following:

- Instrument Air receiver pressure is 85 psig and lowering
- Service Air receiver pressure is 95 psig and lowering

Which of the following describes how the Service Air/Instrument Air Cross-Connect Valves, 1P52-F050 & 2P52-F050, respond to these conditions, including the bases for this response?

The Service Air/Instrument Air Cross-Connect Valves ____.

A. close to completely isolate the Service Air and Instrument Air headers

B. close to prevent a leak in the Service Air header from impacting the Instrument Air header

C. remain open; however they will close if Service Air receiver pressure lowers to 90 psig in order to completely isolate the Service Air and Instrument Air headers

D. remain open; however they will close if Instrument Air receiver pressure lowers to 80 psig in order to prevent a leak in the Service Air header from impacting the Instrument Air header

QUESTION RO 17

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295019	AA1.04
	Importance Rating	3.3	

K&A: Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Service air isolations valves:

Partial or Total Loss of Inst. Air / 8

Explanation: **Answer B** – the cross-connect valves will close when Instrument Air receiver pressure is <90 psig to protect the Instrument Air system from a leak in the Service Air system.

A – incorrect – check valves around the P52-F050 valves allow Service Air to continue to supply Instrument Air when the F050 valves are closed

C – incorrect – P52-F050 valves are closed - Service air can still supply instrument air header. Therefore, they are not completely isolated from each other

D – P52-F050 valves are closed. There are no automatic actions at 80 psig in the IA receiver.

Technical Reference(s): SOI-P51/52 rev 25 & Lesson Plan OT-COMBINED-P51_P52 rev 1	Reference Attached: SOI-P51/52 p 4 & Lesson Plan OT-COMBINED-P51_P52 p 14

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-P51_P52-E.7

Question Source:	Bank #	Perry 2002
	Modified Bank #	
	New	
Question History:	Previous NRC Exam:	Perry 2002
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 18

The following conditions exist:

- It is day 20 of a refuel outage
- Fuel Pool Cooling Cleanup (FPCC) is providing decay heat removal
- Fuel shuffle is complete
- Core verification is in progress
- Div 2 electrical outage in progress
- Reactor coolant temperature is 85^{o}F

FPCC A pump then trips on a ground fault.

Based on the above conditions, determine time to boil in the core?

Reference Provided: PBD-A016 & PDB-A017

A. 5 hours

B. 8.5 hrs

C. 20 hrs

D. 35 hrs

QUESTION RO 18

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295021	AK2.05
	Importance Rating	2.7	

K&A: Knowledge of the interrelations between LOSS OF SHUTDOWN COOLING and the following: Fuel pool cooling and cleanup system

Loss of Shutdown Cooling / 4

Explanation: **Answer D** – FPCC is the alternate decay heat removal (SDC) system. The decay heat load is 20 MBTU/hr based on curves for decay after 248 bundles discharged and water level at 23' above the RPV flange, the time to boil is approximately 35 hours. With Div 2 outage in progress, B FPCC pump is not available.

A – incorrect – plausible if use time to boil curve at vessel flange and decay heat curve prior to bundle discharge

B – incorrect – plausible if use time to boil curve at vessel flange and decay heat curve after bundle discharge

C – incorrect – plausible if use time to boil curve at 23' above flange and decay heat curve prior to bundle discharge

Technical Reference(s): PBD-A016 rev 9 & PDB-A017 rev 10	Reference Attached: PBD-A016 pp. 3 & 5 & PDB-A017 pp 9 & 11

Proposed references to be provided to applicants during examination: PBD-A016 & PDB-A017

Learning Objective (As available): OT-3035-11(LP)-A.1

Question Source:	Bank # Modified Bank # New	 x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 19

The following plant conditions exist:

- A full core offload is in progress

The following then occurred:

- An irradiated fuel bundle is being unloaded from the Fuel Handling Building IFTS Upender
- Bubbles are observed coming from the irradiated fuel bundle
- The Fuel Handling Building Evacuation Alarm Sounded

Based on these conditions, in order to minimize unnecessary exposure, immediate evacuation of ____ from the Fuel Handling Building is required.

1. Platform Operator
2. FME Coordinator
3. Fuel Handling Building Crane Operator
4. Fuel Handling Supervisor
5. Radiation protection Technician
6. Site Protection Officer
7. Spotter

A. 1, 3, & 7

B. 1, 4, & 6

C. 2, 5, & 6

D. 3, 5, & 7

QUESTION RO 19

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295023	AK3.01
	Importance Rating	3.6	

K&A: Knowledge of the reasons for the following responses as they apply to REFUELING ACCIDENTS: Refueling floor evacuation

Refueling Acc / 8

Explanation: **Answer C** – Per ONI-J11-2 Necessary Personnel are defined as those personnel necessary to place the equipment or fuel in a 'safe condition'. SOI-F11, fuel handling Platform, identifies personnel required for fuel handling in the FHB. At Perry, 'necessary personnel' are the FH Supervisor, the Platform Operator, and the Spotter – all other personnel are to be evacuated.

A – incorrect – bridge driver and spotter are necessary personnel

B – incorrect – bridge driver and FH Supervisor are necessary personnel

D – incorrect – the spotter is considered necessary personnel

Technical Reference(s): ONI-J11-2 rev 13 and SOI-F11 rev 10	Reference Attached: ONI-J11-2 pp. 5-7 and SOI-F11 p 5

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3035-14(LP)-A.5

Question Source:	Bank # Modified Bank # New	 x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 20

The following plant conditions exist:

- The plant is operating at 100% power
- No equipment is out of service

A large-break LOCA occurs in the Drywell and the following conditions exist:

- Drywell pressure is at 12 psig and lowering
- Reactor pressure is at 140 psig and lowering
- Reactor water level is at 5 inches and rising

Which of the following describes the loading sequence to limit starting transients on the 4160 V Emergency Buses?

A. RHR C starts 5 seconds from initiation signal
HPCS starts 10 seconds from initiation signal
LPCS starts 15 seconds from initiation signal

B. RHR C starts 5 seconds from initiation signal
HPCS starts 14 seconds from initiation signal
LPCS starts 19 seconds from initiation signal

C. RHR C starts immediately after initiation signal
HPCS starts 10 seconds from initiation signal
LPCS starts 15 seconds from initiation signal

D. RHR C starts immediately after initiation signal
HPCS starts 14 seconds from initiation signal
LPCS starts 19 seconds from initiation signal

QUESTION RO 20

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295024	EA1.10
	Importance Rating	3.4	

K&A: Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE: A.C. distribution

High Drywell Pressure / 5

Explanation: **Answer C** – a high DW pressure signal will initiate HPCS, LPCS, & RHR. Per Tech Spec Bases, pump starting is sequenced due to the high starting currents to control emergency bus and transformer loading.

A & B – incorrect – RHR C starts immediately.

B & D – incorrect – HPCS starts 10 seconds from initiation. Nineteen seconds for LPCS and 14 seconds for HPCS is plausible because on a loss of bus power the respective pump, start relays are energized for 19 & 14 seconds upon restoration of power.

Technical Reference(s): TS 3.3.5.1 Bases rev 7, SDM-E12 rev 9, SDM-E21 rev 10, & SDM-E22 rev 7	Reference Attached: TS 3.3.5.1 Bases p B3.3-97, SDM-E12 p 37, SDM-E21 p 21, & SDM-E22 p 22

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): x

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 21

The plant was operating at rated power when a transient occurred that resulted in the following plant parameters:

• Reactor pressure:	1040 psig
• RPV level:	50”
• Drywell temperature:	185^{o}F
• Drywell pressure:	4 psig
• Suppression Pool level:	18.5 feet
• Suppression Pool temperature:	114^{o}F
• Containment pressure:	2.5 psig
• Containment temperature:	125^{o}F

Which one of the following conditions will cause the margin to the Heat Capacity Limit (HCL) to <u>improve</u>?

A. RPV level rises

B. RPV pressure lowers

C. Suppression Pool level lowers

D. Suppression Pool temperature rises

QUESTION RO 21

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295025	EA2.03
	Importance Rating	3.9	

K&A: Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Suppression pool temperature

High Reactor Pressure / 3

Explanation: **Answer B** – lowering reactor pressure will move farther away from HCL.

A – incorrect – plausible due to misconception of effect of RPV level on HCL. Raising RPV level will have no effect on HCL margin

C – incorrect – lowering suppression pool level will reduce margin to HCL

D – incorrect – raising suppression pool temperature lowers margin to HCL

Technical Reference(s): EOP Bases rev 0	Reference Attached: EOP bases p 63

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3402-06-C.1

Question Source:	Bank #	
	Modified Bank #	Peach Bottom 2008
	New	
Question History:	Previous NRC Exam:	Peach Bottom 2008
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 22

The following plant conditions exist:

- Reactor power: 85%
- Suppression Pool Average temperature: 76^{o}F

The following then occurs:

- Safety Relief valve 1B21-F047G inadvertently opens
- ONI-B21, SRV Inadvertent Opening/Stuck Open, actions were not successful in closing the SRV
- Suppression Pool Average Temperature is rising 3^{o}F/minute
- No other operator actions are being performed

A Technical Specification requirement to immediately scram the reactor will first be met in ____.

A. 7 minutes

B. 10 minutes

C. 12 minutes

D. 15 minutes

QUESTION RO 22

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295026	2.2.22
	Importance Rating	4	

K&A: Knowledge of limiting conditions for operations and safety limits.

Suppression Pool High Water Temp. / 5

Explanation: **Answer C** – with a heat-up rate of 3 deg/min, suppression pool temperature will exceed 110^{o}F in just over 11 minutes. Per tech specs, a scram is required if suppression pool temp exceeds 110^{o}F.

A – incorrect – this is the time to exceed 95^{o}F, which is the requirement to enter TS 3.6.2.1 with no testing in progress adding heat to suppression pool

B – incorrect – this is the time to exceed 105^{o}F, which is the requirement to enter TS 3.6.2.1 with testing in progress adding heat to suppression pool

D – incorrect – this is the time to exceed 120^{o}F, which is requirement to depressurize the reactor

Technical Reference(s): Tech Spec 3.6.2.1	Reference Attached: Tech Spec 3.6.2.1 pp. 3.6-36 - 38

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3037-10.B

Question Source:	Bank #	
	Modified Bank #	Oyster Creek 2008
	New	
Question History:	Previous NRC Exam:	Oyster Creek 2008
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 23

EOP-2, Containment Control, was entered due to a scram discharge volume rupture.

Containment Spray was initiated per EOP-SPI 3.1, Containment Spray Operation.

In accordance with EOP-2, Containment Spray operation is required to be terminated ____.

A. before Containment Pressure is reduced below 0.0 psig

B. before Containment Pressure is reduced below 0.5 psig

C. after Containment Temperature is reduced below 95°F

D. after Containment Temperature is reduced below 185°F

QUESTION RO 23

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295027	EK1.03
	Importance Rating	3.8	

K&A: Knowledge of the operational implications of the following concepts as they apply to HIGH CONTAINMENT TEMPERATURE (MARK III CONTAINMENT ONLY): Containment integrity

High Containment Temperature / 5

Explanation: **Answer A** – Containment Spray must be terminated prior to reaching 0 psig in containment in order to maintain a margin to the negative design pressure of containment.

B – incorrect – 0.5 psig is a 'rule of thumb' when securing from containment spray operation

C – incorrect – 95^{o}F is the entry condition for containment temperature leg, where containment sprays can be started

D – incorrect – 185^{o}F is the containment design limit for containment temperature for ED

Technical Reference(s): EOP-2 Bases rev 0	Reference Attached: EOP-2 Bases p 10

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3402-07-C.2

Question Source:	Bank #	Perry 2007-1
	Modified Bank #	
	New	
Question History:	Previous NRC Exam:	Perry 2007-1
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 24

The following conditions exist:

- EOP-2, Containment Control, has been entered
- Suppression Pool level is 17.7 feet and lowering slowly
- Suppression Pool Average temperature is 83^{o}F
- Containment Average temperature is 96^{o}F and stable
- Drywell Average temperature is 146^{o}F and rising
- Drywell pressure is 0.6 psig and stable

Based on these conditions, the required EOP-2 action(s) is(are) to ____.

A. Maximize Suppression Pool Cooling only

B. Operate all available Drywell Cooling only

C. Operate all available Drywell Cooling and
Operate all available Containment Cooling

D. Maximize Suppression Pool Cooling and
Operate all available Drywell Cooling and
Operate all available Containment Cooling

QUESTION RO 24

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295028	EK2.04
	Importance Rating	3.6	

K&A: Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: Drywell ventilation

High Drywell Temperature / 5

Explanation: **Answer C** – Containment temperature and DW temperature exceeds the value for operating all containment and DW cooling. Per the EOP bases, the determination that DW temp can not be maintained can be made prior to exceeding the value.

A – incorrect – SP temperature is 13 deg F from the limit

B – incorrect – this is only partially correct, containment temperature exceeds the value to operate cooling

D – incorrect – SP temperature is 13 deg F from the limit and is rising slowly

Technical Reference(s): EOP Bases rev 0 & EOP-2 Bases rev 0	Reference Attached: EOP Bases p 36 & EOP-2 Bases pp 17 and 50

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3402-8.C

Question Source:	Bank #	
	Modified Bank #	
	New	x

Question History: Previous NRC Exam:

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

10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 25

A CAUTION in the EOP's warns the operator about operating RCIC with low Suppression Pool water level (<7.25 feet).

Select the statement below that describes the application of this limit.

Immediate and catastrophic equipment failure ____.

A. is expected. Operation of RCIC at a Suppression Pool level less than 7.25 feet is not permitted.

B. is not expected. Operation of RCIC at a Suppression Pool level less than 7.25 feet is not permitted.

C. is expected. Operation of RCIC at a Suppression Pool level less than 7.25 feet is permitted under certain circumstances.

D. is not expected. Operation of RCIC at a Suppression Pool level less than 7.25 feet is permitted under certain circumstances.

QUESTION RO 25

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295030	EK3.03
	Importance Rating	3.6	

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

Low Suppression Pool Wtr Lvl / 5

Explanation: **Answer D** – Equipment failure is not expected to be immediate and this limit may be disregarded if core cooling is threatened.

A & B – incorrect – This limit may be disregarded if core cooling is threatened.

A & C – incorrect – Equipment failure is not expected to be immediate.

Technical Reference(s): EOP Bases rev 0	Reference Attached: EOP Bases p 48

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3402-1-b.2

Question Source:	Bank # Modified Bank # New	Perry 2001
Question History:	Previous NRC Exam:	Perry 2001
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 26

The following conditions exist:

- A LOCA in the drywell occurred
- Bus EH11 experienced a lockout
- High Pressure Core Spray failed to initiate
- Validated RPV water level is 25 inches and lowering
- Validated RPV pressure is 900 psig and lowering
- Drywell pressure is 1.8 psig and rising
- 1B21-N695B, RPV Level – Lvl 3, failed high
- All other systems are functioning as designed.

Which of the following describes the operation of the Automatic Depressurization System (ADS) under the current conditions?

A. ADS SRVs can be opened by their individual key-lock switches <u>only</u>.

B. ADS logic automatically initiates to open ADS SRVs after 105 second time delay relay times out

C. ADS SRVs can be opened by arming and depressing the ADS LOGIC MANUAL INITIATION switches.

D. ADS logic automatically initiates to open ADS SRVs 105 seconds after the RPV level set point is reached.

QUESTION RO 26

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295031	EA1.06
	Importance Rating	4.4	

K&A: Ability to operate and/or monitor the following as they apply to REACTOR LOW WATER LEVEL: Automatic depressurization system

Reactor Low Water Level / 2

Explanation: **Answer C** – the Arm & Depress switches will work anytime the respective LPCI/LPCS pump is providing sufficient discharge pressure.

A – incorrect – the manual SRV switches are not the <u>only</u> way to open the valves

B & D – incorrect – the ADS B system will not automatically initiate with the Level 3 instrument failed high and ADS A will not automatically initiate with no LPCI/LPCS pump running

Technical Reference(s): PDB-I-005 rev 9	Reference Attached: PDB-I005 pp.13-14

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-B21C E.1

Question Source:	Bank # Modified Bank # New	 x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 27

The plant was operating at 85% rated power with the following equipment out of service:

- CRD B pump tagged out due to an oil leak

A manual reactor scram was inserted.

The following conditions now exist:

- Reactor Scram Hardcard actions were completed
- Reactor power is at 5%
- Reactor pressure is at 940 psig
- The GP1A light above the 'A' manual scram switch is illuminated
- Bus EH11 locked out
- Full Core Display indicates numerous control rods withdrawn at various positions
- SCRAM VLV AIR HEADER PRESS LO annunciator (H13-P680-5A-D6) has alarmed

Which of the following methods would successfully insert control rods for the above listed conditions?

A. EOP-SPI 1.1, PULLING SCRAM FUSES

B. EOP-SPI 1.3, MANUAL ROD INSERTION

C. EOP-SPI 1.5, VENTING CRD OVERPISTON VOLUMES

D. EOP-SPI 1.6, INCREASED COOLING WATER ΔP

QUESTION RO 27

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295037	EA2.05
	Importance Rating	4.2	

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

SCRAM Condition Present and Power Above APRM Downscale or Unknown / 1

Explanation: **Answer C** – CRD B pump is tagged and CRD A bump is not running due to EH11 bus lockout. Therefore, venting the over-piston volumes is the only method that would work.

A – incorrect – RPS did not fail to de-energize, therefore, this method would not be successful

B & D – incorrect – a CRD pump must be running to use these sections

Technical Reference(s): EOP-SPI 1.1 rev 0, EOP-SPI 1.3 rev 1, EOP-SPI 1.5 rev 0, EOP-SPI 1.6 rev 1	Reference Attached: EOP-SPI 1.1 p 2, EOP-SPI 1.3 p 2, EOP-SPI 1.5 p 2, EOP-SPI 1.6 p 2

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): 3402-03-D

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 28

The following conditions exist:

- The plant is operating at full power
- Fuel inspections are in progress in the Fuel Handling Building
- FHB HVAC Supply Fan A (M40-C001A) is in operation
- FHB HVAC Exhaust Fans A and B (M40-C002A(B)) are in operation

An event occurs that damages numerous Spent Fuel Bundles with the following results:

- All FHB Ventilation Exhaust Airborne Radiation Monitor (D17-K710) module indications are offscale high.
- Spent Fuel Pool (D21-K332) and Fuel Prep Pool (D21-K322) Area Radiation Monitor indications are reading 12,000 mR/hr.
- The radiation release rate is at the General Emergency level
- The Unit Supervisor entered EOP-3, Secondary Containment Control and EOP-5, Radioactivity Release Control

The off-site release path is through the __(1)__. The Control Room Operator would __(2)__.

	__(1)__	__(2)__
A.	Unit 1 Plant Vent	verify FHB HVAC Supply Fan A is tripped
B.	Unit 1 Plant Vent	scram the Reactor and Emergency Depressurize
C.	Unit 2 Plant Vent	verify FHB HVAC Supply Fan A is tripped
D.	Unit 2 Plant Vent	scram the Reactor and Emergency Depressurize

QUESTION RO 28

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295038	2.4.31
	Importance Rating	4.2	

K&A: Knowledge of annunciator alarms, indications, or response procedures.

High Off-site Release Rate / 9

Explanation: **Answer A** – Release path from FHB is through the UNIT 1 Plant Vent, EOP-3 requires verification of supply fans that should have tripped trip.

B & D – incorrect – this would be true if the leak was from a primary system per the EOP's

C & D – incorrect – unit 2 plant vent is not release pathway

Technical Reference(s): ARI-H13-P902-001-B1 rev 4 & EOP-3 Bases rev 0	Reference Attached: ARI-H13-P902-001-B1 p 7 & EOP-3 Bases p 12

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3402-17-D.2, OT-COMBINED-M40-B.3, & OT-3402-15-C.2

Question Source:	Bank #	Perry 2007-1
	Modified Bank #	
	New	
Question History:	Previous NRC Exam:	Perry 2007-1
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 29

The following conditions exist:

- The plant is operating at 100% power
- SAS reports that Zone 1 and Zone 2 heat detectors for Reactor Recirculation Pump A are in alarm
- The Unit Supervisor enters ONI-P54, Fire

Which one of the following actions is required for a fire in Reactor Recirculation Pump A per ONI-P54, Fire?

A. Operator confirms the CO2 System has automatically dumped CO2

B. Operator opens 1P54-F395, DW CO2 SUPPLY OTBD ISOL to commence CO2 dump

C. Operator opens 1P54-F340, CTMT CO2 SUPPLY OTBD ISOL to commence CO2 dump

D. Fire Brigade member opens 1P54-F3590, RX RCIRC PMP A SELECTOR VLV to commence CO2 dump

QUESTION RO 29

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

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

Plant Fire On Site / 8

Explanation: **Answer C** – Per ONI-P54 Immediate operator action for Recirculation pump fire is to open 1P54F340.

A – incorrect - 2 heat detectors automatically initiate CO2, but system will not dump with 1P54F340 closed

B – incorrect – wrong valve, valve is already open

D – incorrect – Fire Brigade is not required to open the selector valve for a Recirculation Pump fire

Technical Reference(s): ONI-P54 rev 15	Reference Attached: ONI-P54 p 4

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED P54-CO2-D.1.B, OT-3035-05(LP)-A.9

Question Source:	Bank #	Perry 2007-1
	Modified Bank #	
	New	

Question History:	Previous NRC Exam:	Perry 2007-1

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

10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 30

SCC informs the Control Room of a degraded grid condition and the following occurs:

1. ONI-S11, Hi/Low Voltage, is entered
2. Grid voltage and generator voltage swings are observed

What is the nominal bus voltage at which the undervoltage relays will drop out AND what is the basis for this protection?

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

B. 3800 Vac (~95% nominal), protection is provided to prevent jeopardizing the reliability of starting a diesel generator.

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

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

QUESTION RO 30

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

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

Generator Voltage and Electric Grid Disturbances / 6

Explanation: **Answer D** – 75% voltage is to protect starting safety related motors

A – incorrect – 3000 vac is protection for starting motors, not running equipment

B – incorrect – Bus voltage does not effect the reliability of the D/G starting

B & C – incorrect – 3800 volts is to protect running equipment

Technical Reference(s): OT-COMBINED-R10 (LP) rev 0	Reference Attached: OT-COMBINED-R10 (LP) pp. 49-50

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-R10-D.3

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 31

The plant is operating at 70% power when Inboard MSIV, B21-F022A, inadvertently closes.

- Reactor pressure rises 20 psig and stabilizes
- No RPS setpoints are exceeded
- No operator actions are taken

Which one of the following describes the response of the reactor to this event?

Reactor power initially ____.

A. lowers and then stabilizes at a lower value.

B. lowers and then returns to its original value.

C. rises and then stabilizes at a higher value.

D. rises and then returns to its original value.

QUESTION RO 31

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295007	AA2.02
	Importance Rating	4.1	

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

High Reactor Pressure / 3

Explanation: **Answer C** –Rx pressure rising causes voids to collapse causing a positive reactivity addition causing Rx power rises. Power will stabilize at a higher value due to the higher pressure.

A & B – incorrect – Rx power will initially rise due to void collapse, not lower

D – incorrect – Rx power will not return to its original value

Technical Reference(s): OT-3301-04 (LP) rev 4 (GFE-Rx Theory Chaper 4) and IOI-3 rev 44	Reference Attached: OT-3301-04 (LP) p49 and IOI-3 p 7

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3301-04-7 & 9

Question Source:	Bank # Modified Bank # New	Perry 2003
Question History:	Previous NRC Exam:	Perry 2003
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 32

The following conditions exist:

- The plant is operating at 50% power
- RFPT A is out of service for an oil leak

RFPT B tripped.

The following then occurred:

- The reactor operator inserted a reactor scram
- Reactor pressure peaked at 1045 psig
- The Motor Feed pump tripped on an electrical fault
- All other systems functioned as designed
- EOP-1, RPV Control, was entered

Based on the above conditions which of the following valves will be closed?

1. 1B21-F019 MSL DRN & MSIV BYP OTBD ISOL
2. 1D17-F081B CNTMT RAD MON INBD SUCT ISOL
3. 1E12-F073B RHR B HX'S SECOND VENT TO SUPR POOL
4. 1G42-F010 SPCU PUMP FIRST SUCT ISOL
5. 1P43-F055 NCC CNTMT SUPPLY OTBD ISOL
6. 1P51-F150 SA SUPPLY HDR CNTMT ISOL.

A. 3, 4, & 5

B. 2, 3, & 6

C. 1, 5, & 6

D. 1, 2, & 4

QUESTION RO 32

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295009	2.4.2
	Importance Rating	4.5	

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

Low Reactor Water Level / 2

Explanation: **Answer B** – based on the given conditions, EOP-1 should be entered when vessel hits Level 2. The Level 2 and Level 3 isolations will occur. Additionally, HPCS and RCIC will initiate on L2 to restore level and not allow level to drop to Level 1.

A & C – incorrect - 1P43-F055 isolates on L1 not L2

C & D – incorrect - 1B21-F019 isolates on L1 not L2

Technical Reference(s): EOP-1 Bases rev 1 & OAI-1703 rev 3	Reference Attached: EOP-1 Bases p22 & OAI-1703 pp. 16 & 17

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3402-02-B & OT-COMBINED-B21-NS4-D

Question Source:	Bank #	
	Modified Bank #	
	New	x

Question History: Previous NRC Exam:

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

10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 33

The plant is operating at 100% power when a trip of Containment Vessel Chilled Water Chiller A occurred.

- Containment temperature and pressure are slowly rising
- Drywell temperature and pressure are stable
- Alarm CONTAINMENT TEMP A HIGH H13-P601-020-F4 has been received
- Alarm CONTAINMENT TEMP B HIGH H13 P601-017-D2 has been received
- No EOP Entry Conditions exist

Which one of the following conditions will occur if Containment temperature and pressure continue to rise with no operator action taken?

A. Indicated Containment Upper Pool level will lower.

B. Indicated Suppression Pool level will rise.

C. Containment Vacuum Breakers will open.

D. Drywell Vacuum Breakers will open.

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QUESTION RO 33

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295011	AK1.01
	Importance Rating	4	

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

High Containment Temp / 5

Explanation: **Answer D** – with stable DW pressure, as containment pressure exceeds the DW vacuum breaker opening setpoint, the DW vacuum breakers will open

A – These conditions will have no effect on indicated Containment Upper Pool level.

B – These conditions will cause indicated Suppression Pool level to lower, not raise

C – These conditions will cause Containment Vacuum breakers to remain closed, not open.

Technical Reference(s): ARI H13-P601-20(E4) & (F4) & SDM-M16 rev 4	Reference Attached: ARI H13-P601-20(E4) (F4) pp. 71 & 83 and & SDM-M16 p 5

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-T23-K10.D & OT-COMBINED-D23-F.4

Question Source:	Bank #	Perry 2003	
	Modified Bank #		
	New		
Question History:	Previous NRC Exam:	Perry 2003	
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		x
10 CFR Part 55 Content:	55.41	x	
	55.43		

Comments: Level of Difficulty = x

QUESTION RO 34

A plant startup is in progress after completion of a refuel outage.
Plant conditions are as follows:

- Mode Switch is in STARTUP
- APRM Power is 3%
- IRMs are on Range 8

To protect the reactor from an inadvertent reactivity addition due to a control rod withdrawal accident the primary scram signal is ____.

A. SRM High-High Flux

B. IRM Neutron Flux-High

C. APRM Neutron Flux-High

D. APRM Neutron Flux-High Setdown

QUESTION RO 34

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295014	AK2.01
	Importance Rating	3.9	

K&A: Knowledge of the interrelations between INADVERTENT REACTIVITY ADDITION and the following: RPS

Inadvertent Reactivity Addition / 1

Explanation: **Answer B –** per Tech Spec Bases, IRM Neutron Flux-High trip is the primary scram signal.

A – incorrect – SRM scram is bypassed for RFO12 startup

C – incorrect – this APRM scram is active in Mode 1

D – incorrect – this is a back-up or secondary scram to the IRM scram for the given condition.

Technical Reference(s): Tech Spec Bases 3.3.1.1 rev 0	Reference Attached: Tech Spec Bases 3.3.1.1 p B 3.3-4

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-C71-F.2

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

Question History: Previous NRC Exam: Perry 2007-2

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

10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 35

The following conditions exist:

- The plant is operating at rated power
- Surveillance SVI-B21-T0061-A, RPV LOW LEVEL 1 AND 2 CHANNEL A FUNCTIONAL FOR 1B21-N681A, is in progress

The I&C tech just dialed down 1B21-N681A, RPV LEVEL 1, until it tripped.

If 1B21-N681B, RPV LEVEL 1, now fails downscale, the effect would be a ____.

A. direct reactor scram to preserve the integrity of the fuel cladding

B. Main Steam Line drain valve isolation <u>only</u> to prevent off-site dose limits from being exceeded

C. RWCU valve isolation <u>only</u> to ensure peak fuel cladding temperatures remain below the limits of 10 CFR 50.46

D. MSIV closure and subsequent reactor scram in anticipation of the complete loss of the normal heat sink and subsequent over-pressurization transient

QUESTION RO 35

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295020	AK3.01
	Importance Rating	3.8	

K&A: Knowledge of the reasons for the following responses as they apply to INADVERTENT CONTAINMENT ISOLATION: Reactor SCRAM

Inadvertent Cont. Isolation / 5 & 7

Explanation: **Answer D** – the reactor scrams on a MSIV isolation in anticipation of the complete loss of the normal heat sink and subsequent over-pressurization transient. Channels A & B give full MSIV isolation.

A – incorrect – simultaneous trips on B21-N0681 A & B channels cause all MSIVs to isolate which causes a reactor scram on MSIV position

B & C – incorrect – more than the stated actions occur

Technical Reference(s): PDB-I005 rev 9, SDM-B21(NS4) rev 6, & Tech Spec Bases 3.3.1.1 rev 0	Reference Attached: PDB-I005 pp. 1, 17-20, & 32,SDM-B21(NS4) 46, & TS Bases p B 3.3-14

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-B21(NS4)-E, I & J

Question Source:	Bank #	
	Modified Bank #	
	New	x

Question History: Previous NRC Exam:

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

10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 36

The High Pressure Core Spray (HPCS) Pump, 1E22-C001, is being operated in the CST to CST Mode for testing.

During the test, actual Suppression Pool level rose to 18.5'.

Which one of the following statements describes the effect on HPCS operation?

A. HPCS operation will be unaffected

B. HPCS pumps the Suppression Pool to the CST

C. HPCS Pump operates on minimum flow CST to Suppression Pool

D. HPCS Pump operates on minimum flow with suction on the Suppression Pool

QUESTION RO 36

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295029	EA1.02
	Importance Rating	3.1	

K&A: Ability to operate and/or monitor the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL: HPCS

High Suppression Pool Wtr Lvl / 5

Explanation: **Answer D** – Suppression Pool high level causes suction shift to the Suppression Pool. Both Test Valves Close. This loss of flow path causes the Minimum Flow Valve to open.

A – incorrect - misconception when operating in TEST, if in suppression pool test mode (vice CST to CST test mode) no affect

B – incorrect – misconception of CST Test Return Valve closure logic

C – incorrect – misconception of Suction Valve transfer logic in TEST

Technical Reference(s): ARI-H13-P601-016-G5 Rev 12 and SDM-E22A rev 7	Reference Attached: ARI-H13-P601-016-G5 p 83 and SDM-E22A pp.67 & 69

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-E22A-E.3

Question Source:	Bank #	
	Modified Bank #	Perry 2007-1
	New	
Question History:	Previous NRC Exam:	Perry 2007-1
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 37

The following conditions exist:

- The plant is shutdown for a refuel outage
- Fuel shuffle is in progress
- Containment Vessel and Drywell Purge is operating in Refuel Mode

A fuel bundle was dropped and the following annunciators alarmed on 1H13-P680:

- CNTMT VENT EXH RAD HI
- CNTMT VENT EXH RAD A/D HI HI/INOP

Observation of radiation monitors in the Control Room reveal various monitors with elevated readings but only the Containment Ventilation Exhaust Radiation Monitors D17-K609A and D17-K609D have reached their alarm setpoints.

Based on these conditions, __(1)__ Containment Vessel and Drywell Purge fans will be running and __(2)__ of the Containment Vessel and Drywell Purge containment isolation valves will be closed.

	__(1)__	__(2)__
A.	no	all
B.	no	half
C.	all	none
D.	some	half

QUESTION RO 37

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295033	EA2.01
	Importance Rating	3.8	

K&A: Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS: Area radiation levels

High Secondary Containment Area Radiation Levels / 9

Explanation: **Answer B** – the CVDWP radiation monitors are located in the Intermediate Building (secondary containment area) and will initiate an isolation signal to prevent a spread of radiation/contamination into secondary containment. A HI/HI alarm on D17-K609A and D radiation monitor causes an M14 Outboard Isolation while a HI/HI on the B and C radiation monitors causes an iM14 inboard isolation. The isolation causes of loss of suction path trip for the fans. All fans trip but only 1/2 (the outboard) valves go closed.

A – incorrect – the inboard isolation valves remain open

C – incorrect – all of the fans trip and half of the isolation valves close

D – incorrect – all of the fans trip

Technical Reference(s): ARI-H13-P680-007 rev 18 and EOP-3 Bases rev 0	Reference Attached: ARI-H13-P680-007 pp 27, 29-30 and EOP-3 Bases p 7

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3407-17-C

Question Source:	Bank # Modified Bank # New	
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 38

The following conditions exist:

- The plant is in MODE 4.
- RHR Loop A is running in the Shutdown Cooling Mode.
- RHR Loop B is running in the Suppression Pool Cooling Mode for surveillance testing

Which one of the following describes the automatic response of the RHR system if a valid RPV Level 1 Reactor water level condition occurs?

A. RHR Pump A trips.
RHR Loop B realigns to the LPCI mode.

B. RHR Pump A continues to operate in the Shutdown Cooling Mode.
RHR Loop B realigns to the LPCI mode.

C. RHR Pumps A and B trip.
RHR Loop B then shifts to the LPCI mode and RHR Pump B restarts.

D. RHR Pump A trips.
RHR Loop B continues to operate in the Suppression Pool Cooling mode.

QUESTION RO 38

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	203000	K4.09
	Importance Rating	3.1	

K&A: Knowledge of RHR/LPCI: INJECTION MODE (PLANT SPECIFIC)design feature(s) and/or interlocks which provide for the following: Surveillance for all operable components

RHR/LPCI: Injection Mode

Explanation: **Answer A** – RHR pump A trips due to RHR-SDC valves F008 and F009 going close on the L3 signal. RHR B loop will realign for LPCI injection at L1. There is no automatic realignment of RHR from SDC to LPCI.

B – incorrect – RHR pump A trips due to RHR-SDC valves F008 and F009 going close on the L3 signal

C – incorrect – RHR Pump B does not trip and then restart

D – incorrect – RHR B loop will realign for LPCI injection at L1

Technical Reference(s): ARI-H13-P601-017-C4 rev 12, ARI-H13-P601-020-F6 rev 15, PDB-I-005 rev 9, & SOI-E12 rev. 46	Reference Attached: ARI-H13-P601-017-C4 p 37, ARI-H13-P601-020-F6 p 89, PDB-I005 p 35, & SOI-E12 pp.9-10

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-E12-H.1.A

Question Source:	Bank #	INL-36851
	Modified Bank #	
	New	
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 39

RHR Loop A has just been placed into the Shutdown Cooling Mode of operation using the Normal Return Path.

The cooldown rate is excessive.

In accordance with SOI-E12, Residual Heat Removal System, which of the following actions will lower the cooldown rate?

A. Throttle close the RHR A HX'S BYPASS VALVE, E12-F048A, and throttle open the RHR A HX'S OUTLET VALVE, E12-F003A, while maintaining a system flowrate of 7000-7100 gpm.

B. Throttle open the RHR A HX'S BYPASS VALVE, E12-F048A, and throttle close the RHR A HX'S OUTLET VALVE, E12-F003A, while maintaining a system flowrate of 7000-7100 gpm.

C. Throttle open the RHR A HX'S BYPASS VALVE, E12-F048A, and throttle close the RHR A HX'S OUTLET VALVE, E12-F003A, while maintaining a system flowrate of 2575-7100 gpm.

D. Throttle ESW flow through the RHR Heat Exchanger using RHR A HX'S ESW OUTLET VALVE, P45-F068A.

QUESTION RO 39

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	205000	K5.03
	Importance Rating	2.8	

K&A: Knowledge of the operational implications of the following concepts as they apply to SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE): Heat removal mechanisms

Shutdown Cooling

Explanation: **Answer B** – this is the approved method for adjusting the cooldown rate while in Normal Mode SDC.

A – incorrect – this action will raise the cooldown rate

C – incorrect – this is the required flow rate band when using the alternate return path via E12-F042A

D – incorrect –not an approved method to control the cooldown rate per SOI-E12

Technical Reference(s): SOI-E12 rev 26	Reference Attached: SOI-E12 pp 26, 31-32

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-E12-B.4.C

Question Source:	Bank #	Perry 2001
	Modified Bank #	
	New	
Question History:	Previous NRC Exam:	Perry 2001
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 40

The following conditions exist:

- A plant startup is in progress
- Reactor power 25%
- Reactor Recirculation Flow Control Valves are full open
- APRM A is in Bypass

OPRM A receives a Core Flow signal from APRM __(1)__ and if power oscillations occur, OPRM A __(2)__ trip RPS.

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

QUESTION RO 40

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	215005	K6.04
	Importance Rating	3.2	

K&A: Knowledge of the effect that a loss or malfunction of the following will have on the OSCILATING POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM: Trip units

OPRM

Explanation: **Answer D** – when the APRM is bypassed, the associated OPRM automatically receives a core flow signal from the other APRM in the same panel. The OPRM trip is enabled at > 23.3 % power and drive flow < 63.5%.

A & C – incorrect – OPRM A receives core flow signal from APRM E

A & B – incorrect – the OPRM trip is enabled and will input a trip signal into RPS

Technical Reference(s): SOI-C51(APRM) rev 9 and ARI-H13-P601-006-A2 rev 7	Reference Attached: SOI-C51(APRM) p 14 and ARI-H13-P601-006-A2 p 7

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-C51-AP_OPRM-E.2 & 3

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 41

The Plant is operating at 75% rated power.

One hour ago the blue pressure permissive light for the LPCS Injection Valve, 1E21-F005, extinguished.

Control Room Operators confirmed the blue light bulb was good.

Which one of the following describes the operation of the LPCS Injection Valve control logic if a Loss-of Coolant Accident now occurs?

The LPCS Injection Valve ____.

A. remains closed and cannot be opened with its control switch until RPV pressure lowers to 600 psig

B. automatically opens, irrespective of RPV pressure, due to the LPCS LOCA initiation signal.

C. remains closed and cannot automatically open until RPV pressure lowers to 530 psig

D. remains closed and cannot automatically open until RPV pressure lowers to 600 psig

QUESTION RO 41

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	209001	A1.03
	Importance Rating	3.8	

K&A: Ability to predict and/or monitor changes in parameters associated with operating the LOW PRESSURECORE SPRAY SYSTEM controls including: Reactor water level

LPCS

Explanation: **Answer D** – the Blue permissive light indicates that pressure downstream of the injection valve is <600 psig and will allow the injection valve to automatically open.

A – incorrect – the injection valve control switch bypasses the pressure permissive

B – incorrect – the blue light off indicates pressure permissive is not met → no automatic opening

C – incorrect – 600 psig is LPCS injection valve open permissive, 530 psig for RHR

Technical Reference(s): ARI-H13-P601-021-A6 rev 13 and SDM-E21 rev 10	Reference Attached: ARI-H13-P601-021-A6 p 17 and SDM-E21 pp.23-25

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-E21-E.1 & 3

Question Source:	Bank #	Perry 2001-2	
	Modified Bank #		
	New		
Question History:	Previous NRC Exam:	Perry 2001-2	
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		x
10 CFR Part 55 Content:	55.41	x	
	55.43		

Comments: Level of Difficulty = x

QUESTION RO 42

HPCS is operating in CST to CST Mode for PMT

- The operator has adjusted HPCS flow to 600 gpm to shutdown HPCS to Standby Readiness per SOI-E22A Section 7.10.18, HPCS Full flow Test to CST
- The HPCS Minimum flow valve 1E22-F012 remains closed

As the Reactor Operator continues to shutdown the pump, in order to prevent HPCS pump damage caused by __(1)__, the required operator action is to __(2)__.

	__(1)__	__(2)__
A.	runout	open the HPCS PUMP MIN FLOW VALVE, 1E22-F012
B.	runout	lower flow until 1E22-F0012 automatically opens
C.	operating at shutoff head	open the HPCS PUMP MIN FLOW VALVE, 1E22-F012
D.	operating at shutoff head	lower flow until 1E22-F0012 automatically opens

QUESTION RO 42

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	209002	A2.10
	Importance Rating	2.7	

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

HPCS

Explanation: **Answer C** – HPCS min flow valve should auto open as flow is lowered to 725 gpm to prevent the pump from running at shutoff head. SOI-E22A P&L says to verify min flow valve open prior to stopping flow in CST to CST mode. NOP-OP-1002, it is the responsibility of the operator to take manual actions if automatic actions fail to occur.

A & B – incorrect – runout is a condition of excess pump flow – in this case the pump does not have the minimum required flow

B & D – incorrect – all sections of SOI-E22A that contain min flow valve operations say to verify min flow valve open – none discuss lowering pump flow below the minimum flow requirements

Technical Reference(s): SOI-E22A rev 25 & NOP-OP-1002 rev 5	Reference Attached: SOI-E22A pp.5 and 45 & NOP-OP-1002 p 43

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-E22A-E.3 & OT-3039-01-B

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 43

The following conditions exist:

- An ATWS occurred
- Standby Liquid Control (SLC) was initiated
- The SLC A Squib valve, 1C41-F004A, failed to fire
- The SLC B Pump, 1C41-C001B, failed to start
- Reactor Pressure is 1000 psig

Considering these failures, which one of the following describes the indications for SLC Pump A and B discharge pressure that would be observed on 1H13-P601.

SLC Pump A discharge pressure will be approximately __(1)__ and SLC Pump B discharge pressure will be approximately __(2)__.

	__(1)__	__(2)__
A.	1100 psig	1100 psig
B.	1100 psig	0 psig
C.	1300 psig	1300 psig
D.	1300 psig	0 psig

QUESTION RO 43

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	211000	A3.01
	Importance Rating	3.5	

K&A: Ability to monitor automatic operations of the STANDBY LIQUID CONTROL SYSTEM including: Pump discharge pressure

SLC

Explanation: **Answer A** – both instruments show discharge pressure even though only one pump starts. Since one squib valve is open, a flow path exists, thus pressure will be slightly above reactor pressure.

B – incorrect - due to the cross-connect line between the pumps, both meters will show discharge pressure

C & D – incorrect – since the pumps are rated at 1250 psig, a 1300 psig is an indication that the relief valve (1C41-F029A/B) is lifting, which is not the case since one squib valve fired

Technical Reference(s): SOI-C41 rev 18, Dwg. 302-691 rev V (Partial), & SDM-C41 rev 8	Reference Attached: SOI-C41 pp 6 & 9, Dwg. 302-691 rev V (Partial), & SDM-C41 p 8

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-C41-C.3 & OT-3402-D.2

Question Source:	Bank # Modified Bank # New	RQL-0047
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 44

At 10:00 the following conditions exist:

- The plant is operating at 100% rated power

At 10:01 the following conditions exist:

- A load rejection with a loss of turbine EHC occurred
- Safety/Relief valves responded as designed

At 10:02 the following conditions exist:

- Motor Feedwater pump is running on minimum flow
- Reactor power is 5%
- Scram Hardcard actions have been completed

At 10:10 the following conditions exist:

- All rods are in
- Reactor pressure is being controlled on Bypass Valves
- Setpoint Setdown has been reset
- Motor Feedwater pump is running in manual, feeding the RPV
- The Unit Supervisor is operating in ONI-C71-1, Reactor Scram

At this time, the Redundant Reactivity Control System logic __(1)__ be reset and RPS logic __(2)__ be reset.

	__(1)__	__(2)__
A.	can	can
B.	can not	can
C.	can	can not
D.	can not	can not

QUESTION RO 44

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	212000	2.4.11
	Importance Rating	4	

K&A: Knowledge of abnormal condition procedures.

RPS

Explanation: **Answer B** – at time 10:10 there are no RPS scram signals nor RRCS initiation signals present. Therefore, RPS can be reset. However, ONI-C71-1, Reactor Scram contains a NOTE that informs the operator that RRCS logic contains a 12 minute timer, which does not allow reset of the RRCS logic until the timer times out.

A & C – incorrect – RRCS logic can not be reset due to the 12 minute timer

C & D – incorrect – RPS logic can be reset as there are no scram signals present

Technical Reference(s): ONI-C71-2 rev 8	Reference Attached: ONI-C71-2 p 7

Proposed references to be provided to applicants during examination: None

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

Question Source:	Bank # Modified Bank # New	 x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 45

The plant is operating at 100% power with the Reactor Protection System MG SET TRANSFER switch in ALT A.

The following occurs:

- Numerous Control Room Alarms are received
- Half scram is indicated
- Outboard BOP isolation has occurred

This is an indication of a loss of power from Bus ____.

A. F1B08

B. F1C08

C. F1C12

D. F1D12

QUESTION RO 45

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	212000	K1.01
	Importance Rating	3.4	

K&A: Knowledge of the physical connections and/or cause/effect relationships between REACTOR PROTECTION SYSTEM and the following: A.C. electrical distribution

RPS

Explanation: **Answer B** – with RPS MG Set Transfer switch in ALT A position, RPS bus A is powered from F1C08. Loosing RPS bus A causes an outboard isolation and ½ scram.

A – incorrect – this is the normal power supply for RPS A

C – incorrect – this is the normal power supply for RPS B

D – incorrect – this is the alternate power supply for RPS B

Technical Reference(s): PDB-H014 rev 1, PDB-H15 rev 2, PDB-H16 rev 1, & SDM-C71 rev 10	Reference Attached: PDB-H15 p 5 & SDM-C71 p 78

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-C71 D.1

Question Source:	Bank #	
	Modified Bank #	Perry 2007-2
	New	
Question History:	Previous NRC Exam:	Perry 2007-2
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 46

What is the power source for the high voltage power supply for IRM H?

A. Division 1 ATWS Dist. Panel 1R14-S014

B. Division 2 ATWS Dist. Panel 1R14-S015

C. RPS A Bus, C71-P001

D. RPS B Bus, C71-P002

QUESTION RO 46

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	215003	K2.01
	Importance Rating	2.5	

K&A: Knowledge of electrical power supplies to the following: IRM channels/detectors

IRM

Explanation: **Answer D** – RPS B supplies IRM H.

A & B – incorrect – the ATWS panels supply the APRM's not the IRM's

C – incorrect – RPS A supplies IRM's A, C, E, & G

Technical Reference(s): SOI-C71 rev 18 & SDM-C51(IRM) rev 7	Reference Attached: SOI-C71 p 111 & SDM-C51(IRM) p 53

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-C51(IRM)-B.3.E

Question Source:	Bank #	
	Modified Bank #	
	New	x

Question History: Previous NRC Exam:

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

10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 47

The following plant conditions exist:

- A reactor startup is in progress following replacement of all fuel bundles
- Reactor Protection System shorting links are removed
- Reactor power is rising with a stable positive period of 150 seconds
- SRM Channel ‘A’ detector is stuck and will not withdraw
- SRM Channel ‘A’ indication rises to $2.5x10^5$ cps
- No operator actions are performed

Which one of the following subsequently describes SRM Channel ‘A’ indicated reactor power and reactor period?

Indicated reactor power will __(1)__. Reactor period will __(2)__

	__(1)__	__(2)__
A.	lower	become negative
B.	lower	remain stable and positive
C.	continue to rise	become shorter
D.	continue to rise	remain stable and positive

QUESTION RO 47

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	215004	K3.04
	Importance Rating	3.7	

K&A: Knowledge of the effect that a loss or malfunction of the SOURCE RANGE MONITOR (SRM) SYSTEM will have on following: Reactor power and indication

Source Range Monitor

Explanation: **Answer A** – with shorting links removed, SRM trips are active and non-coincident. As power rises above 2E+5, the reactor will scram on high flux and reactor period will be negative.

B & D – incorrect – Rx period will not remain stable and positive

C & D – incorrect – Rx power will lower due to the scram

Technical Reference(s): SOI-C51(SRM) rev 6	Reference Attached: SOI-C51(SRM) p 3

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-C51(SRM)-D.4 & OT-COMBINED –C71-O.5

Question Source:	Bank # Modified Bank # New	Perry 2002
Question History:	Previous NRC Exam:	Perry 2002
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 x 55.43	

Comments: Level of Difficulty = x

QUESTION RO 48

The following conditions exist:

- The plant is operating at 48% rated power
- Total Core Flow is 50 MLbs/Hr
- Both Recirculation are operating in fast speed

What is the current Upscale Thermal Power Trip setpoint rounded to the nearest %?

Reference Provided: PDB-A0012, Recirc Drive Flow vs. Total Core Flow

A. 107%

B. 108%

C. 111%

D. 113%

QUESTION RO 48

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	215005	K4.07
	Importance Rating	3.7	

K&A: Knowledge of AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM design feature(s) and/or interlocks which provide for the following: Flow biased trip setpoints

APRM / LPRM

Explanation: **Answer C** – the APRM Upscale Thermal Power Trip setpoint is calculated using the formula Setpoint=0.628W_r + 61% (where W_r is % drive flow) and is clamped at 111%.

A – incorrect – this is the current APRM Upscale Thermal Power Alarm setpoint

B – incorrect – this is the clamped value for the APRM Upscale Thermal Power Alarm setpoint

D – incorrect – this is the calculated setpoint, however, it is clamped at 111%

Technical Reference(s): SDM-C51 (PRM & OPRM) rev 10 and PDB A12 rev 15	Reference Attached: SDM-C51 (PRM & OPRM) pp 23-25 and PDB-A12 p 6

Proposed references to be provided to applicants during examination: PDB-A0012, Recirc Drive Flow vs. Total Core Flow

Learning Objective (As available): OT-COMBINED-C51(AP_OPRM)-D.8

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 49

The following conditions exist:

- The reactor scrammed due to a loss of the feedwater system
- High Pressure Core Spray pump tripped due to a seized shaft
- Reactor Core Isolation Cooling is maintaining RPV level at 30 inches
- Reactor pressure is 870 psig and lowering slowly

The receipt of which annunciator would indicate an <u>immediate</u> threat to RCIC's ability to maintain RPV level?

A. RCIC ISOL DIAPHRAGM RUPTURED, (H13-P601-0021-B1)

B. STEAM TUNNEL LD AMB TEMP P632, (H13-P601-0019-G4)

C. RCIC TURBINE OIL COOLER OUT TEMP HIGH, (H13-P601-0021-C4)

D. RCIC SUPR POOL SUCT VLV OPEN SUPR PL LVL HI, (H13-P601-0021-G5)

QUESTION RO 49

<table>
<tr><td rowspan="5">Examination Outline Cross-Reference</td><td>Level:</td><td>RO</td><td>SRO</td></tr>
<tr><td>Tier #</td><td>2</td><td></td></tr>
<tr><td>Group #</td><td>1</td><td></td></tr>
<tr><td>K/A#</td><td>217000</td><td>K5.07</td></tr>
<tr><td>Importance Rating</td><td>3.1</td><td></td></tr>
</table>

K&A: Knowledge of the operational implications of the following concepts as they apply to REACTOR CORE ISOLATION COOLING SYSTEM (RCIC): Assist core cooling

RCIC

Explanation: **Answer A** – an exhaust diaphragm rupture will cause an immediate RCIC turbine trip.

B – incorrect – the steam tunnel high temperature has a 29 minute time delay to isolate

C – incorrect – high RCIC lube oil temp does not cause an immediate loss of the turbine

D – incorrect – a high suppression level will cause a suction shift, but the suppression pool valve opens fully before the CST valve closes – in this case it doesn't matter which source is lined up.

Technical Reference(s): ARI-H13-P601-021 rev 13, ARI-H13-P601 rev 11, & SOI-E31 rev 8	Reference Attached: ARI-H13-P601-021 pp. 19, 39, & 89, ARI-H13-P601 p111, & SOI-E31 p 19

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-E51-D.3

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 50

The following plant conditions exist:

- The reactor is operating at 100% power
- A loss of ED-1-A occurs

Which of the following describes the ability of the SRVs to function in the Pressure-Relief Mode and in the ADS Mode?

In the Pressure-Relief Mode, __(1)__ SRVs will function and in the ADS Mode, actuation of the Div. II ADS logic __(2)__ open the ADS valves.

	__(1)__	__(2)__
A.	no	will
B.	no	will not
C.	all	will
D.	all	will not

QUESTION RO 50

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	218000	K6.06
	Importance Rating	3.4	

K&A: Knowledge of the effect that a loss or malfunction of the following will have on the AUTOMATIC DEPRESSURIZATION SYSTEM: D.C. power:

ADS

Explanation: **Answer C** – the Pressure Relief mode will function using the B solenoids and the ADS function will function using the Div 2 logic

A & B – incorrect – the pressure relief mode will still function utilizing the B solenoids.

D – incorrect – Div 2 ADS logic is independent of Div 1 power supply and can still open the ADS valves

Technical Reference(s): PDB-I005 rev 9 and SDM-B21C rev 6	Reference Attached: PDB-I005 p 14 and SDM-B21 p 22

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-B21C-E.1 & OT-COMBINED-R42-H

Question Source:	Bank #	Perry 2003
	Modified Bank #	
	New	
Question History:	Previous NRC Exam:	Perry 2003
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 51

An isolation signal was generated that resulted in the following light indications on panel H13-P622, Div 2 Aux Relay Panel Inboard Valves:

- NS4 MSL DRN ISOL INBD LOGIC TEST B21H-DS1B light on
- NS4 BOP ISOL INBD LOGIC TEST B21H-DS3B light on
- RX WTR SMPL VLV B33-F019 LOGIC TEST B21H-DS7B light off
- RHR ISOLATION INBD LOGIC TEST B21H-DS5B light off

Based on these indications, select the isolation that occurred.

Reference Provided: Picture of panel H13-P622

A. BOP isolation

B. RWCU isolation

C. RHR LOCA isolation

D. RHR Radwaste Valve isolation

QUESTION RO 51

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	223002	A1.01
	Importance Rating	3.5	

K&A: Ability to predict and/or monitor changes in parameters associated with operating the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF controls including: System indicating lights and alarms

PCIS/Nuclear Steam Supply Shutoff

Explanation: **Answer D** – the RHR Radwaste Valve isolation is Level 3 and 1.68 psig. The light goes out if isolation signal present. The candidate needs to know if the lights are on or off for the isolation.

A – incorrect – the BOP isolation is actuated on Level 2 and 1.68 psig in DW – the light would be off

B – incorrect – the RWCU isolation is actuated on Level 2 and Leak Detection signals – the BOP light would be off if hit L2

C – incorrect – the RHR LOCA isolation is actuated on Level 1 and 1.68 psig in DW – the light would be off

Technical Reference(s): IOI-18 rev 6 and DWGS 208-013-008 rev P & 208-013-012 rev FF	Reference Attached: IOI-18 p 82 and DWGS 208-013-008 & 208-013-012

Proposed references to be provided to applicants during examination: Picture of panel H13-P622

Learning Objective (As available): OT-COMBINED-B21(NS4)-M

Question Source:	Bank # Modified Bank # New	 x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 52

The plant is operating at full power.

An indication that an SRV is open is __(1)__.
In accordance with ONI-B21-1, SRV Inadvertent Opening/Stuck Open, do not attempt to close the SRV until reactor power is lowered to __(2)__ percent.

	__(1)__	__(2)__
A.	Main generator electrical output lowers	90
B.	Main generator electrical output lowers	95
C.	Indicated steam flow on the affected steam line rises	90
D.	Indicated steam flow on the affected steam line rises	95

QUESTION RO 52

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	239002	A2.03
	Importance Rating	4.1	

K&A: Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Stuck open SRV

SRVs

Explanation: **Answer A** – an indication that an SRV is open is lowering of main generator MWe – per ONI-B21-1, Rx power is lowered to 90%

B & D – incorrect – reactor power is to be lowered to less than 90% prior to attempting to close an SRV

C & D – incorrect – the flow orifices are down stream of the SRV's and indicated steam flow will lower not rise

Technical Reference(s): ONI-B21-1 rev 9	Reference Attached: ONI-B21-1 pp.3-4

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): x

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 53

The following conditions exist:

- The plant is at 100% power
- Digital Feedwater Operator Rx Level Setpoint was set at 200 inches

A manual scram is now inserted.

Which of the following describes the response of the DFWLCS following the scram?

A. Upon receipt of the scram signal, the level demand signal will be 196 inches for 10 seconds and then lower to 178 inches.

B. Upon receipt of the scram signal, the level demand signal will be 200 inches for 10 seconds and then lower to 178 inches.

C. When level reaches 178 inches, the level demand signal will be 196 inches for 10 seconds and then lower to 178 inches.

D. When level reaches 178 inches, the level demand signal will be 200 inches for 10 seconds and then lower to 178 inches.

QUESTION RO 53

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	259002	A3.06
	Importance Rating	3.0	

K&A: Ability to monitor automatic operations of the REACTOR WATER LEVEL CONTROL SYSTEM including: Reactor water level setpoint setdown following a reactor scram: Plant-Specific

Reactor Water Level Control

Explanation: **Answer D** – with the Operator Rx Level Setpoint set at 200”, when RPV level drops below L3, Setpoint Setdown logic demands the Operator Rx Level Setpoint setting for 10 seconds then lowers to 178”

A & B – incorrect – the scram signal does not initiate Setpoint Setdown logic

A & C – incorrect – the Operator Rx Level Setpoint was set at 200”, therefore the Setpoint Setdown logic demands 200” not 196”

Technical Reference(s): OT-COMBINED-C32 LP rev 4	Reference Attached: OT-COMBINED-C32 LP p 32

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-C34-F.5.c

Question Source:	Bank #	INL-3034
	Modified Bank #	
	New	
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 54

The following plant conditions exist:

- The reactor is operating at 90% power
- Annulus Exhaust Gas Treatment System (AEGTS) Train ‘B’ is in service
- An unplanned gaseous radioactive release occurs in the Annulus

Which one of the following Airborne Radiation Monitors (ABRM) would detect this radioactive release in the Annulus?

A. Unit 1 (U1) Plant Vent, 1D17-K0780

B. Unit 2 (U2) Plant Vent, 2D17-K0780

C. Intermediate Bldg Vent, D17-K730

D. Aux Bldg Vent, 1D17-K700

QUESTION RO 54

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	261000	A4.01
	Importance Rating	3.2	

K&A: Ability to manually operate and/or monitor in the control room: Off-site release levels

SGTS

Explanation: **Answer B** – The discharge of each AEGTS fan branches into two headers. One header recirculates air flow back into the annulus. The other header exhausts air to the unit's vent, AEGTS subsystem A to the Unit 1 Vent, AEGTS subsystem B to the Unit 2 Vent.

A – incorrect – AEGTS A exhausts to U1 plant vent

C & D – plausible due to misconception that AEGTS exhausts to these ventilation systems

Technical Reference(s): Dwg 912-605 rev W	Reference Attached: Dwg 912-605

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-M15-B.1

Question Source:	Bank #	
	Modified Bank #	Perry 2002
	New	
Question History:	Previous NRC Exam:	Perry 2002
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 55

The plant is operating at full power with Division 1 Diesel Generator operating in parallel with the grid when the following valid alarms were received:

- BUS EH11 VOLTAGE DEGRADATION (H13-P877-0001-B1)
- BUS EH12 VOLTAGE DEGRADATION (H13-P877-0002-B1)
- BUS EH12 BUS STRIPPED UNDERVOLTAGE (H13-P877-0002-C1)

What additional condition is required to initiate the LOOP logic?

A. 3 seconds elapses.

B. 12 seconds elapses.

C. Bus EH11 frequency at 58 hertz.

D. Bus EH12 frequency at 57 hertz.

QUESTION RO 55

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	262001	2.4.45
	Importance Rating	4.1	

K&A: Ability to prioritize and interpret the significance of each annunciator or alarm.

AC Electrical Distribution

Explanation: **Answer C** – the LOOP logic requires both bus EH11 & bus EH12 to have low voltage (3KV) or low frequency (59 Hz). The low voltage is made up for the EH12 bus, so when frequency on bus EH11 lowers below 59 Hz, the logic will be made up.

A – incorrect – this conditions is already fulfilled when the Bus Stripped alarm is received

B – incorrect – this conditions is true if a LOCA signal were present

D – incorrect – the LOOP logic is already made up for bus EH12, therefore this signal will have no effect

Technical Reference(s): SOI-S11 rev 5	Reference Attached: SOI-S11 pp. 3-4

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-R10-D.3

Question Source:	Bank # Modified Bank # New	 x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 56

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

A reactor scram could <u>not</u> be initiated prior to evacuating the Control Room.

Which one of the following describes the <u>preferred</u> method for initiating a reactor scram, including the bases for this method?

Cycle the specified ____.

A. ATWS UPS distribution panel breakers since this will not cause a MSIV closure

B. ATWS UPS distribution panel breakers since this will not cause a loss of LPRMs/APRMs

C. RPS power distribution panel breakers since this will not cause a MSIV closure

D. RPS power distribution panel breakers since this will not cause a loss of LPRMs/APRMs

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QUESTION RO 56

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	262002	K1.19
	Importance Rating	2.9	

K&A: Knowledge of the physical connections and/or cause/effect relationships between UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) and the following: Power range neutron monitoring system: Plant-Specific

UPS (AC/DC)

Explanation: **Answer A** – cycling the ATWS breakers is preferred because this method deenergizes the APRM to cause a scram and allows the MSIV's to remain open.

B – incorrect – this will cause the APRM's to lose power

C – incorrect this will cause the MSIV's to close

D – incorrect – this is not the preferred method

Technical Reference(s): ONI-C61 rev 6	Reference Attached: ONI-C61 p 5

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): x

Question Source:	Bank #	Perry 2002
	Modified Bank #	
	New	
Question History:	Previous NRC Exam:	Perry 2002
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 57

Which of the following is the power supply for the listed load?

	1N43-C007, Turb Emg Bearing Oil Pump	RHR B/C Logic	Main Generator Trip Logic
A.	D1A	ED1A	D1B
B.	D1B	ED1B	D1B
C.	D1A	ED1A	D1A
D.	D1B	ED1B	D1A

QUESTION RO 57

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	263000	K2.01
	Importance Rating	3.1	

K&A: Knowledge of electrical power supplies to the following: Major D.C. loads

DC Electrical Distribution

Explanation: **Answer D** – DC bus D1B supplies Emergency Bearing oil pump, D1A supplies the main generator trip logic, and ED1B supplies RHR B/C logic.

A – incorrect – all three selections are incorrect

B – incorrect – D1B does not supply main generator trip logic

C – incorrect – wrong selections for emergency bearing oil pump and RHR logic

Technical Reference(s): PDB-H002 rev 2, PDB-H004 rev 3, & PDB-H005 rev 2	Reference Attached: PDB-H002 p 12, PDB-H004 pp. 13-14, & PDB-H005 p 2

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-R42-B.2.C

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 58

The following conditions exist:

- A Loss of Off-Site Power (LOOP) occurred
- Divisional diesel generators are supplying their respective buses
- DIESEL GENERATOR CONTROL TRANSFER switch is in CONT RM on 1H51-P055A
- DIESEL GENERATOR CONTROL TRANSFER switch is in LOCAL on 1H51-P055B

The following alarms are then received on H13-P877:

Division 1
- DIV 1 DIESEL GENERATOR TROUBLE, (H13-P877-0001-D5)
- DG TRIP* LUBE OIL TEMP HIGH, (H13-P877-0001-C3)

Division 2
- DIV 2 DIESEL GENERATOR TROUBLE, (H13-P877-0002-D5)
- DG TRIP DIFF RELAY LOCKOUT, (H13-P877-0002-E4)

What will be the status of Buses EH11 and EH12?

	EH11	EH12
A.	Energized	Energized
B.	Energized	De-energized
C.	De-energized	Energized
D.	De-energized	De-energized

QUESTION RO 58

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	264000	K3.02
	Importance Rating	3.9	

K&A: Knowledge of the effect that a loss or malfunction of the EMERGENCY GENERATORS (DIESEL/JET) will have on following: A.C. electrical distribution

EDGs

Explanation: **Answer B** – the Lube Oil high temperature trip is bypassed with a LOOP signal present when the DG CONTROL TRANSFER switch is in CR position and the DIFF RELAY LOCKOUT trip (Div 2) is always active

A – incorrect – EH12 will be deenergized

C & D – incorrect – EH11 will be energized

Technical Reference(s): SOI-R43 rev 36, ARI-H13-P877-001 rev 10, ARI-H13-P877-002 rev 11, & ARI-H51-P054B rev 7	Reference Attached: SOI-R43 p 6, ARI-H13-P877-001 pp 31& 44, ARI-H13-P877-002 pp. 44 & 53, & ARI-H51-P054B p 97

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-R43_48-D.9

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 59

Which of the following will automatically trip on high temperature on a complete loss of Nuclear Closed Cooling?

A. Control Complex Chiller B, P47-B001B

B. Control Rod Drive Pump A, 1C11-C001A

C. Reactor Recirculation Pump A, 1B33-C001A

D. Unit 1 Instrument Air Compressor, 1P52-C001

QUESTION RO 59

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	300000	K4.03
	Importance Rating	2.8	

K&A: Knowledge of (INSTRUMENT AIR SYSTEM) design feature(s) and or interlocks which provide for the following: Securing of IAS upon loss of cooling water

Instrument Air

Explanation: **Answer D** – NCC cools the IAC lube oil. On a loss of NCC, lube oil will heat up to the trip setpoint.

A – incorrect –CC chiller B is supplied from ECC, CC chiller C is supplied from NCC

B & C – incorrect – CRD and Recirc pumps do not have automatic trips on high temp

Technical Reference(s): ARI-H13-P904 rev 9, ARI-H13-P680-004 rev 14, & ONI-P43 rev 11	Reference Attached: ARI-H13-P904 p 53, ARI-H13-P680-004 pp. 107 & 129 & ONI-P43 p 17

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-P43-H & OT-3035-16(LP)-A.3

Question Source:	Bank #	
	Modified Bank #	
	New	x

Question History: Previous NRC Exam:

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

10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 60

The following conditions exist:

- The plant is operating at 75% reactor power.
- SVI-C71-T0039, MSIV Closure Channel Functional, is in progress
- MSL B INBD MSIV B21-F022B control switch is in the TEST position.

The Control Room Operator depresses the MSL B INBD MSIV TEST pushbutton 1B21H-S3B.

Which one of the following describes the response of MSL B INBD MSIV B21-F022B?

A. Safety-Related Instrument Air bleeds off the bottom portion of the MSIV air cylinder and the top portion of the MSIV air cylinder is pressurized to stroke the MSIV closed in 3-5 seconds.

B. Instrument Air bleeds off the bottom portion of the MSIV air cylinder and the top portion of the MSIV air cylinder is pressurized to stroke the MSIV closed in 3-5 seconds.

C. Safety-Related Instrument Air bleeds off the bottom portion of the MSIV air cylinder causing the MSIV to slowly close.

D. Instrument Air bleeds off the bottom portion of the MSIV air cylinder causing the MSIV to slowly close.

QUESTION RO 60

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	300000 K1.05	
	Importance Rating	3.1	

K&A: Knowledge of the connections and / or cause effect relationships between INSTRUMENT AIR SYSTEM and the following: Main Steam Isolation Valve air

Instrument Air

Explanation: **Answer D** – air is bled off the MSIV air cylinder allowing it to slow close.

A & B – incorrect – bleeding air off the MSIV allows it to close slowly – not 3-5 seconds which is the normal stroke time

A & C – incorrect – Safety Related Instrument Air (P57) is not the source of operating air to the MSIV's

Technical Reference(s): SDM-B21/N11 rev 10	Reference Attached: SDM-B21/N11 pp.25 & 58

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-B21_N11-E.2

Question Source:	Bank #	Perry 2001
	Modified Bank #	
	New	

Question History: Previous NRC Exam: Perry 2001

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

10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 61

What is the power supply to Nuclear Closed Cooling Pump C?

A. H11

B. H21

C. XH11

D. XH21

QUESTION RO 61

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	400000	K2.01
	Importance Rating	2.9	

K&A: Knowledge of electrical power supplies to the following: CCW pumps

Component Cooling Water

Explanation: **Answer D** – XH2102 from bus XH21 is power supply for NCC pump C

A & D – incorrect – not the correct power supply

C – incorrect – this is the supply to NCC pump A

Technical Reference(s): ELI-R22 rev 8	Reference Attached: ELI-R22 p 25

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-P43-C.2

Question Source:	Bank # Modified Bank # New	 x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 62

Breaker EH1104, 4.16KV TO 480V XFMR EHF-1-A TO BUS EF-1-A, tripped.

Which one of the following has lost power?

A. Suppression Pool Cleanup Pump, 1G42-C001

B. Standby Liquid Control Pump A, 1C41-C001A

C. Reactor Water Cleanup Pump A, 1G33 C001A

D. Control Complex Chill Water Pump C, P47-C001C

QUESTION RO 62

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	211000	K2.01
	Importance Rating	2.9	

K&A: Knowledge of electrical power supplies to the following: SBLC pumps

SLC

Explanation: **Answer B** – Standby Liquid Control Pump A, 1C41-C001A is powered from Bus EF-1-A, Breaker EH1104 on Bus EH11 supplies power to Bus EF-1-A

A – incorrect – is plausible; powered from F-1-E bus

C – incorrect – is plausible; Powered from F-1-C bus

D – incorrect – is plausible; powered from EF-2-A Bus

Technical Reference(s): Dwg 206-021 rev KKKK	Reference Attached: Dwg 206-021

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-C41-E.1, OT-COMBINED-R10-C.17

Question Source:	Bank # Modified Bank # New	Perry 2007-1
Question History:	Previous NRC Exam:	Perry 2007-1
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 63

The following conditions exist:

- The plant is operating at 100% power
- Lake water temperature is 55^{o}F
- TBCC Temperature Control Valve, 1P44-F300 is at midposition

The lake rolls over causing lake water temperature to rise to 63^{o}F.

The rise in lake water temperature causes the temperature control valve to ____.

A. raise the TBCC flow through the heat exchanger

B. lower the TBCC flow through the heat exchanger

C. raise the Service Water flow through the heat exchanger

D. lower the Service Water flow through the heat exchanger

QUESTION RO 63

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	400000	A1.02
	Importance Rating	2.8	

K&A: Ability to predict and / or monitor changes in parameters associated with operating the CCWS controls including: CCW temperature

Component Cooling Water

Explanation: **Answer A** – the TBCC temperature control valve splits the TBCC flow through the HX and bypassing the HX. As the lake temperature rises, SW temperature rises. This causes the TCV to bypass less TBCC flow around the HX to maintain TBCC outlet temperature constant. This is a recent modification to the plant to throttle TBCC vs. SW flows.

B – incorrect – this is true if lake water temperature lowers

C & D – incorrect – SW is full flow through the TBCC HX's – plausible is candidate does not understand which medium is throttled

Technical Reference(s): SDM-P44 rev 6	Reference Attached: SDM-P44 pp 4 & 23

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-P44-B.2

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 64

The plant is operating at full power with the following conditions:

- A scram signal is present
- The Scram Inlet Valve, EP-126, for control rod 30-31 fails to open

Which of the following describes how control rod 30-31 is inserted?

The Control Rod Drive ball check valve repositions directing ____ the drive piston to insert the control rod.

A. reactor water above

B. reactor water below

C. accumulator water above

D. accumulator water below

QUESTION RO 64

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	201003	K1.02
	Importance Rating	2.9	

K&A: Knowledge of the physical connections and/or cause/effect relationships between CONTROL ROD AND DRIVE MECHANISM and the following: Reactor water

Control Rod and Drive Mechanism

Explanation: **Answer B** – when the scram outlet valve opens, the ball check valve will reposition since it sences accumulator pressure < Rx pressure and directs Rx water under the drive piston to insert the control rod.

A – incorrect – Rx water is directed below the piston not above

C & D – incorrect – the ball check valve does not reposition to allow accumulator water to drive the rod – the accumulator is normally lined up to drive the rod

Technical Reference(s): SDM-C22(CRDH) rev 5	Reference Attached: SDM-C22(CRDH) pp. 6 & 37

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-C11(CRDM)-C.1

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 65

Which of the following describes the operational implications of maintaining control rods within designated withdrawal distances during control rod movements above the high power setpoint?

A. Establish a 2 notch limit to mitigate the consequences of a control rod drop accident by limiting the amount and rate of reactivity increase.

B. Establish a 4 notch limit to mitigate the consequences of a control rod drop accident by limiting the amount and rate of reactivity increase.

C. Establish a 2 notch limit to provide protection for a control rod withdrawal error event to preclude a MCPR safety limit violation.

D. Establish a 4 notch limit to provide protection for a control rod withdrawal error event to preclude a MCPR safety limit violation.

QUESTION RO 65

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	201005	K5.09
	Importance Rating	3.5	

K&A: Knowledge of the operational implications of the following concepts as they apply to ROD CONTROL AND INFORMATION SYSTEM (RCIS): High power setpoints

RCIS

Explanation: **Answer C** – RC&IS enforces the limits of the rod withdrawal limiter above the HPSP and limits rod withdrawal to 2 notches to mitigate the consequences of a rod withdrawal error.

A & B – incorrect – the rod pattern controller of RC&IS is designed to mitigate the consequences of a control rod drop accident

B & D – incorrect – the 4-notch limit is enforced between the LPSP and the HPSP

Technical Reference(s): Tech Spec Bases 3.3.2.1 rev 0 & SOI-C11(RCIS) rev 26	Reference Attached: Tech Spec Bases 3.3.2.1 pp. B 3.3-42 & 43 & SOI-C11(RCIS) p 6

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-C11(RCIS)-G & J.5

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 66

The following conditions exist:

- Reactor is operating at rated power
- Reactor Recirculation FCV's are at 70% open
- APRM A is in Bypass
- APRM E ramped to 125%
- No operator actions were taken

The FCV's are now __(1)__ open .

If APRM E reading returned to normal, the FCV's will __(2)__ position.

	__(1)__	__(2)__
A.	30%	remain at the runback
B.	30%	return to the pre-runback
C.	60%	remain at the runback
D.	60%	return to the pre-runback

QUESTION RO 66

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	202002 A3.01	
	Importance Rating	3.6	

K&A: Ability to monitor automatic operations of the RECIRCULATION FLOW CONTROL SYSTEM including: Flow control valve operation: BWR-5,6

Recirculation Flow Control

Explanation: **Answer B** – the FCV's will runback 40% from the original starting point. The FCV's will return to the original position if no operator action is taken.

A & C – incorrect – if the runback was not caused by AFDL, then the FCV's would remain in the current position

C & D – incorrect – this is correct if the FCV's would have started at 100% open – plausible if the candidate thinks FCV's will close 40% from 100%

Technical Reference(s): SDM-B33 rev 9

Reference Attached: SDM-b33 p 43

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-B33-E.16

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 67

The following conditions exist:

- The control room was evacuated per ONI-C61, control Room Evacuation
- The Plant Computer lost power
- Preparation in progress to place Shutdown Cooling in service in accordance with IOI-11, Shutdown from Outside Control Room

While lowering reactor pressure to place Shutdown Cooling in service the following reactor pressure readings were obtained from 1C61-R011, Reactor Pressure, on the Remote Shutdown Panel, C61-P001, at the indicated times:

Time	Rx pressure
1200	1000 psig
1300	425 psig
1400	100 psig
1500	25 psig

Which one of the following choices completes the following statement?

The reactor cooldown rate specified in IOI-11 ____.

Reference provided – Steam Tables

A. has not been exceeded

B. was exceeded between 1200 and 1300

C. was exceeded between 1300 and 1400

D. was exceeded between 1400 and 1500

QUESTION RO 67

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	216000	K4.01
	Importance Rating	3.6	

K&A: Knowledge of NUCLEAR BOILER INSTRUMENTATION design feature(s) and/or interlocks which provide for the following: Reading of nuclear boiler parameters outside the control room

Nuclear Boiler Inst.

Explanation: **Answer C** – With the Plant computer out of service, pressure readings must be obtained from installed instrumentation on either Div 1 or Div 2 Remote Shutdown panels. Tech Spec and IOI-11 requires cooldown at <100^{o}F, cooldown from 1000-425 psig is approximately 92^{o}F, from 425-100 psig is approximately 116^{o}F, and 100-25 psig is approximately 71^{o}F per ABB steam tables. Similar readings were obtained using RPV saturation curve of IOI-11.

A – incorrect – cooldown exceeded 100^{o}F/hr between 1300 and 1400

B & D – incorrect – cooldown rate not exceeded

Technical Reference(s): IOI-11 rev 19	Reference Attached: IOI-11 p 4

Proposed references to be provided to applicants during examination: Steam Tables

Learning Objective (As available): OT-3035-05(LP)-A.10 & OT-3037-08-B

Question Source:	Bank #	Brunswick 2008	
	Modified Bank #		
	New		
Question History:	Previous NRC Exam:	Brunswick 2008	
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		x
10 CFR Part 55 Content:	55.41	x	
	55.43		

Comments: Level of Difficulty = x

QUESTION RO 68

A high Drywell pressure scram occurred.

Eleven (11) minutes later, the following plant conditions exist:

- Reactor pressure is 400 psig and lowering
- Reactor water level is +12 inches and stable
- Drywell pressure is 4 psig and slowly rising
- Containment pressure is 6 psig and slowly rising
- No operator actions have been performed

Which one of the following describes the operating condition of RHR Loop ‘A’?

RHR Loop ‘A’ is ____.

A. spraying Containment

B. operating on minimum flow; Containment Spray mode <u>can</u> be manually initiated

C. injecting into the rector vessel; Containment Spray mode <u>can</u> be manually initiated

D. operating on minimum flow; Containment Sprays mode <u>cannot</u> be manually initiated

QUESTION RO 68

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	226001 A1.07	
	Importance Rating	3.1	

K&A: Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE controls including: System pressure

RHR/LPCI: CTMT Spray Mode

Explanation: **Answer B** – the high DW pressure signal will initiate RHR A. However, Rx pressure is too high for LPCI injection and Containment Spray mode will not auto initiate until containment pressure exceeds 8 psig.

A – incorrect – Containment Spray auto initiate logic is not complete

C – incorrect – LPCI injection valve opens at 530 psig but system injection doesn't start until ~280 psig due to the discharge pressure of RHR pumps

D – incorrect – RHR Containment Spray mode can be manually initiated when drywell pressure is above 1.68 psig.

Technical Reference(s): ARI-H13-P601-020 rev 15, SOI-E12 rev 46, & SDM-E12 rev 9	Reference Attached: ARI-H13-P601-020 p 11, SOI-E12 p 9, & SDM-E12 p 97

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-E12-G.1 & G.2

Question Source:	Bank #	Perry 2002
	Modified Bank #	
	New	
Question History:	Previous NRC Exam:	Perry 2002
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 69

Refueling operations are in progress and the Inclined Fuel Transfer System (IFTS) is in operation.

The IFTS Carriage Assembly has just been raised to the RAISE FILL/DRAIN STOP position in the Fuel Handling Building.

The Bottom Valve and Drain Valve have closed.

Which one of the following describes what the Control Room operator will observe on H13-P601 meter, G43-R022A P601, POOL LEVEL A-SEPARATION?

The Upper Containment Pool water level will initially ____.

A. lower when the IFTS Transfer Tube is filled with water; water level must be manually restored with makeup water from the Condensate Transfer and Storage System

B. lower when the IFTS Transfer Tube is filled with water; water level is restored when water from the FPCC Surge Tanks are subsequently pumped back to the Upper Containment Pool

C. rise due to the displacement of water by the IFTS Carriage Assembly; water level is restored when the IFTS Carriage Assembly is subsequently lowered to the Fuel Handling Building

D. rise due to the displacement of water by the IFTS Carriage Assembly; water level is automatically restored via an automatic drain valve to the Fuel Storage Pool in the Fuel Handling Building

QUESTION RO 69

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	233000	A4.04
	Importance Rating	2.9	

K&A: Ability to manually operate and/or monitor in the control room: Pool level

Fuel Pool Cooling/Cleanup

Explanation: **Answer B** – First, upper pool level lowers due to IFTS tube filling, then level is restored when the FTT drain pumps pump water into the FPCC surge tanks and water is returned to the upper pools.

A – incorrect – The Upper Containment Pool level is restored via the Fuel Transfer Tube Drain Tank Pump to the FPCC Surge Tanks and then pumped back to the Upper Containment Pool

C & D – incorrect - Upper containment pool level will initially lower as the transfer tube is filled (until the FPCC Upper Pool return can restore pool level)

Technical Reference(s): SDM-G41 rev 5

Reference Attached: SDM-G41 pp.5 & 50

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-G41-B.1, B.2.c, & B.2.d

Question Source:	Bank #	Perry 2002
	Modified Bank #	
	New	
Question History:	Previous NRC Exam:	Perry 2002
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 70

Select the statement that describes how the Low-Low Set (LLS) function of the Safety Relief Valves (SRVs) automatically operates following a main steam isolation from high power, to maintain RPV Pressure?

A. 1B21-F051C opens at 1103 psig arming LLS which opens five LLS SRVs at 1113 psig.
1B21F051C cycles between 1073 psig and 936 psig.

B. 1B21-F051C and 1B21-F051D open at 1103 psig arming LLS.
Four LLS SRVs cycle between 1113 psig and 936 psig.
1B21F051C and 1B21-F051D cycle between 1073 psig and 926 psig.

C. 1B21-F051D opens at 1103 psig arming LLS which opens 1B21-F051C.
Four LLS SRVs cycle between 1113 psig and 946 psig.
1B21-F051C and 1B21-F051D cycle between 1033 psig and 976 psig.

D. 1B21-F051D opens at 1103 psig arming LLS which opens 1B21-F051C.
Four LLS SRVs cycle between 1113 psig and 946 psig.
1B21-F051C closes at 936 psig.
1B21-F051D cycles between 1033 psig and 926 psig.

QUESTION RO 70

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	239001	K5.09
	Importance Rating	3.4	

K&A: Knowledge of the operational implications of the following concepts as they apply to MAIN AND REHEAT STEAM SYSTEM: Decay heat removal

Main and Reheat Steam

Explanation: **Answer D** – 1B21F051D opens at 1103 psig arming LLS, which opens 1B21F051C and F047F, F051A, F051B, F051G. SRVs 47F, 51A, 51B, and 51G cycle between 946 and 1113 psig. F051C cycles between 936 and 1073 psig and F051D cycles between 926 and 1033 psig.

A – incorrect – F051C does not arm LLS logic

B – incorrect – cycle pressures for all LLS SRVs is not correct

C – incorrect – cycle pressure for F051C and D is not correct

Technical Reference(s): ONI-B21-1 rev 9, SOI-B21 rev 16, SDM-B21 rev 10	Reference Attached: ONI-B21-1 p 11, SOI-B21 p 4, SDM-B21 pp. 21 & 29

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-B21/N11-E.1 and E.2

Question Source:	Bank #	Perry 2007-1
	Modified Bank #	
	New	

Question History: Previous NRC Exam: Perry 2007-1

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

10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 71

The following conditions exist:

- A plant startup is in progress
- The main turbine is reset

If Main Turbine speed rises from 100% to 107% of rated, which of the following describes the response of the Main Turbine Control Valves (CVs) <u>and</u> Intercept Valves (IVs)?

A. The CVs <u>and</u> IVs start to throttle closed together

B. The IVs throttle closed <u>and</u> the CVs remain open

C. The IVs throttle closed first followed by the CVs

D. The CVs throttle closed first followed by the IVs

QUESTION RO 71

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	241000	A1.13
	Importance Rating	2.7	

K&A: Ability to predict and/or monitor changes in parameters associated with operating the REACTOR/TURBINE PRESSURE REGULATING SYSTEM controls including: Main turbine speed

Reactor/Turbine Pressure Regulator

Explanation: **Answer D** – The CVs will throttle closed first and will be fully closed at 105.5% of rated speed. Then the master IVs will throttle closed and all CVs will be fully closed at 107.5% of rated speed.

A – incorrect – the CVs AND IVs will not throttle together

B – incorrect – the IVs will NOT throttle closed first - the CVs will throttle closed first and will be fully closed at 105.5% of rated speed, then the master IVs will throttle closed AND all IVs will be fully closed at 107.5% of rated speed

C – incorrect – the CVs will NOT remain open - the CVs will throttle closed first and will be fully closed at 105.5% of rated speed, then the master IVs will throttle closed and all IVs will be fully closed at 107.5% of rated speed

Technical Reference(s): SDM-N32/C85 rev 6	Reference Attached: SDM-N32/C85 pp. 34-35 & 115

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-N32/C85-G

Question Source:	Bank # Modified Bank # New	Nine Mile Point 2002
Question History:	Previous NRC Exam:	Nine Mile Point 2002
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41 55.43	x

Comments: Level of Difficulty = x

QUESTION RO 72

The following conditions exist:

- The plant is operating at approximately 63% power
- RFPT B on 3 element (3E) control in AUTO
- RFPT B is operating at the suction flow limit of 23.1 Kgpm per SOI-C34, Feedwater Control System
- The Motor Feed pump is shutdown to Casing Warmup
- RFPT A is shutdown on the turning gear

The plant then entered ONI-P52, Loss Of Service And/Or Instrument Air, due to an air leak which resulted in the following:

- The source of the air leak was isolated from the Instrument Air header by closing 1P52F591, Instrument Air To Heater Bay Bldg Isol
- Instrument Air pressure in the Heater Bay is 50 psig and dropping steadily

Which one of the following describes the required operator response to the above conditions?

A. Take RECIRC FLOW CONTROLLER MINIMUM OUTPUT, 1N27-K162B_OP1 controller to 0% to close the RFP B RECIRC CONTROL VALVE and maintain vessel level 192 to 200 inches.

B. Start the Motor Feed pump per SOI-C34, Feedwater Control System, and maintain vessel level 192 to 200 inches.

C. Scram the Reactor due to lowering vessel level and enter ONI-C71-1, Reactor Scram.

D. No action required RFPT B will recover vessel level in automatic.

QUESTION RO 72

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	259001	A2.05
	Importance Rating	3	

K&A: Ability to (a) predict the impacts of the following on the REACTOR FEEDWATER SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of applicable plant air systems

Reactor Feedwater

Explanation: **Answer C** – the RFPT recirc valve fails open on a loss of air. Since the RFPT is operating at the suction limit to maintain RPV level, the > 7000 gpm going to recirc will cause RPV level to lower. Vessel level will continue to lower until power is reduced to the lower feed capacity. Scram prior to auto scram at level 3. Lowering power would not be an option since it would require a R/R pump downshift.

A – incorrect – this will have no effect as the RFPT recirc valve fails open on a loss of air

B – incorrect – motor feed pump will start but control valves will not open to provide flow to vessel due to the loss of air

D – incorrect – RFPT B will respond in auto but it is already operating at 95%, will add recirc flow of 7000+ gpm, vessel level will lower

Technical Reference(s): ONI-P52 rev 14, ARI-H13-P680-003 rev 11, & SOI-C34 rev 26	Reference Attached: ONI-P52 p 30, ARI-H13-P680-003 p 103, & SOI-C34 p 4

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-N27-K.4

Question Source:	Bank #	RQL-16886
	Modified Bank #	
	New	

Question History: Previous NRC Exam:

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

10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 73

Which of the following will cause Offgas After Filter flow to rise?

A. Moisture in adsorber trains

B. Excessive steam flow rate to SJAEs

C. Low level in condenser seal troughs

D. Prefilter Line Drain Loop Seal Level Low

QUESTION RO 73

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	271000	A4.02
	Importance Rating	2.9	

K&A: Ability to manually operate and/or monitor in the control room: System flows

Offgas

Explanation: **Answer C** – a low level in the condenser seal troughs can cause increased air in-leakage into the condensers. Increased air leakage will result in higher Offgas system flow.

A – incorrect – moisture in the adsorber trains can actually lower after filter flow by blocking the flowpath

B – incorrect – excessive steam flow to the SJAE can cause high cooler condenser temperatures, but not a change in Offgas flow

D – incorrect – a prefilter line drain loop seal low level can cause a spread of contamination, but not an increase in after filter flow

Technical Reference(s): ARI-H13-P845-001 rev 9 & ARI-H13-P680-002 rev 8	Reference Attached: ARI-H13-P845-001 p 107 & ARI-H13-P680-002 p 3

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-N62-K.2

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 74

The following conditions exist:

- The plant is shutdown for a refueling outage
- M25/26 Control Room Ventilation train A is running in NORMAL mode
- M25/26 Control Room Ventilation train B is in STANDBY

A loss of input from Control Room Atmosphere Gas Module D17-K776 to the M25/26 logic has been experienced due to an electrical switching error on buses K1A and K1N.

What is the current status of the Control Room HVAC system?

	Train A	Train B
A.	running in NORMAL mode	remains in STANDBY
B.	running in NORMAL mode	running in EMERGENCY RECIRC mode
C.	running in EMERGENCY RECIRC mode	remains in STANDBY
D.	running in EMERGENCY RECIRC mode	running in EMERGENCY RECIRC mode

QUESTION RO 74

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	290003	K6.01
	Importance Rating	2.7	

K&A: Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROOM HVAC: Electrical power

Control Room HVAC

Explanation: **Answer D** – Both trains initiate in ER on loss of radiation monitor signal.

A – incorrect – plausible – loss of power will not have any affect

B&C – incorrect – both trains initiate in ER

Technical Reference(s): SOI-M25/26 Rev 18 & dwg 208-117 sh 11 rev S	Reference Attached: SOI-M25/26 p 5 & dwg 208-117 sh 11

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-M25/26 F & G

Question Source:	Bank #	Perry 2009
	Modified Bank #	
	New	
Question History:	Previous NRC Exam:	Perry 2009
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION RO 75

The following conditions exist:

- The plant is in Mode 4
- Reactor cooldown is in progress
- SVI-B21-T1176, RCS Heatup And Cooldown Surveillance is in progress
- Current RCS temperature is 69°F

What is the consequence of cooling down the Reactor Coolant System below 70°F?

The probability of ____ rises.

Reference Provided: SVI-B21-T1176 partial – modified

A. brittle fracture

B. ductile failure

C. erosion corrosion

D. stress corrosion cracking

QUESTION RO 75

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	290002	2.2.12
	Importance Rating	3.7	

K&A: Knowledge of surveillance procedures.

Reactor Vessel Internals

Explanation: **Answer A** – operating to the left of the heatup/cooldown curve increases the probability of brittle fracture.

B & D – incorrect – A high RCS temperature increases the probability of these occurring, not a low RCS temperature.

C – incorrect – The probability of this occurring is maximum at 270°F, not below 70°F

Technical Reference(s): SVI-B21-T1176 rev 12, Tech Spec 3.4.11 Bases rev 8, & GFE(LP) Chapt. 10 rev 4	Reference Attached: SVI-B21-T1176 pp. 9 & 18, Tech Spec 3.4.11 Bases p B 3.4-56, & GFE(LP) Chapt. 10 p 23

Proposed references to be provided to applicants during examination: SVI-B21-T1176 partial – modified

Learning Objective (As available): x

Question Source:	Bank #	Perry 2003
	Modified Bank #	
	New	
Question History:	Previous NRC Exam:	Perry 2003
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION SRO 1

A staff Senior Reactor Operator had an Active and Valid license on December 31, 2009.

He then stood the following 12-hour shifts:

- Jan 12, 2010 Shift Engineer
- Jan 31, 2010 ATC
- Feb 21, 2010 ATC
- Feb 22, 2010 BOP
- Feb 23, 2010 ATC
- Mar 23, 2010 Field Supervisor
- Apr 3, 2010 Unit Supervisor

No additional shifts were stood as of April 30, 2010.

As of May 1, 2010 the staff SRO ____.

A. maintained proficiency as SRO

B. maintained proficiency as RO only

C. maintained proficiency as Shift Engineer

D. did not maintain proficiency as a licensed operator

QUESTION SRO 1

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	2.1.1	
	Importance Rating		4.2

K&A: Knowledge of conduct of operations requirements.

Generic

Explanation: **Answer D** – The SRO is required to stand 5 12-hour watches per calendar quarter in a SRO position to maintain proficiency. Alternatively, he may stand 1 SRO watch and the remaining 4 may be in a Tech Spec required RO or SRO position.

A – incorrect – plausible if candidate uses quarter rather than calendar quarter

B – incorrect – the Field Supervisor is not a required TS position – only 4 RO watches can be credited

C – incorrect – the Shift Engineer must also maintain an Active RO or SRO license per PYBP-POS-1-5

Technical Reference(s): TMA-4206 rev 12 & PYBP-POS-1-5 rev 3	Reference Attached: TMA-4206 p 17 & PYBP-POS-1-5 p 5

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-2600-01-F

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	
	55.43b.1	x

Comments: Level of Difficulty = x

QUESTION SRO 2

Refuel outage is in progress with the following conditions:

- Core alterations are in progress
- Div 2 electrical outage in progress
- RHR A loop is operating in Shutdown Cooling Mode
- Reactor cavity water level is above the top of the weir wall

What condition below will require the Unit Supervisor to suspend movement of irradiated fuel assemblies within the RPV immediately?

A. Cavity water level is at 22’10” above the fuel

B. Refuel position one-rod-out interlock is inoperable

C. RHR A is shutdown to replace breaker control power fuses

D. Control Rod 30-31 position indication channel 2 is inoperable

QUESTION SRO 2

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	2.1.36	
	Importance Rating		4.1

K&A: Knowledge of procedures and limitations involved in core alterations.

Generic

Explanation: **Answer A** – per TS 3.9.6 or 3.9.7, RPV water level needs to be ≥ 22'9" above the RPV flange. This is much lower.

B – incorrect – this is TS 3.9.2 and action is to suspend control rod withdrawal

C – incorrect – TS 3.9.8 allows RHR SDC to be removed from service for up to 2 hours per 8 hour period

D – incorrect – this is TS 3.9.4 – but per TS bases, channel 1 is the required channel

Technical Reference(s): TS 3.9.6, 3.9.7, 3.9.2, 3.9.8, 3.9.4, & TS Bases 3.9.4 rev 0	Reference Attached: TS 3.9.6 p 3.9-8, 3.9.7 p 3.9-9, 3.9.2 p 3.9-2, 3.9.8 p 3.9-10, 3.9.4 p 3.9-5, & TS Bases 3.9.4 p B 3.9-13

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3037-13-A, B, & C

Question Source:	Bank # Modified Bank # New	 x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43b.2 x	

Comments: Level of Difficulty = x

QUESTION SRO 3

The plant is operating at full power.

- A problem develops with the Residual Heat Removal ‘A’ pump and the Unit Supervisor declares RHR A INOPERABLE at 0500 on June 1
- On June 3, a fire occurs in breaker EH1208 for RHR ‘B’ pump and RHR ‘B’ is declared INOPERABLE at 1700
- RHR ‘A’ is returned to OPERABLE status three hours later at 2000

If RHR ‘B’ cannot be returned to OPERABLE status, what is the latest time that the plant must be in MODE 3?

Reference provided: Tech Spec 3.5.1

A. June 8, 0500

B. June 9, 0500

C. June 9, 1700

D. June 10, 1700

QUESTION SRO 3

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	2.2.23	
	Importance Rating		4.6

K&A: Ability to track Technical Specifications limiting conditions for operations

Generic

Explanation: **Answer C** – per TS 1.3, this qualifies for a 24 hour completion time extension – the plant is required to be in Mode 3 by June 9, 1700

A – incorrect – this is the original completion time for RHR A inop

B – incorrect – this is the time to enter TS 3.5.1 condition D – not Mode 3

D – incorrect – this is the time to enter Mode 4

Technical Reference(s): TS 3.5.1 and TS 1.3	Reference Attached: TS 3.5.1 pp 3.5-1—3.5-3 and TS 1.3 pp. 1.0-11&12

Proposed references to be provided to applicants during examination: Tech Spec 3.5.1

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

Question Source:	Bank #	
	Modified Bank #	
	New	x

Question History: Previous NRC Exam:

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

10 CFR Part 55 Content:	55.41	
	55.43b.2	x

Comments: Level of Difficulty = x

QUESTION SRO 4

The following plant conditions exist:

- Reactor Mode Switch in SHUTDOWN
- Reactor Coolant Temperature is 250°F
- Unit 2 Div. 1 battery (2R42-S002) is INOPERABLE
- Unit 1 Div.1 battery (1R42-S002) is on float
- RCIC System is in Secured Status

While performing weekly battery voltage surveillance, Electrical Maintenance discovered that the Unit 1 Division 1 battery (1R42-S002) terminal voltage to be 105 VDC.

As the Unit Supervisor, select the statement that describes the actions required.

Reference provided: Technical Specifications 3.8.4 & 3.8.5

A. Restore the Unit 1, Div. 1 battery to OPERABLE within 2 hours or be in COLD SHUTDOWN within the following 24 hours

B. Restore either Div. 1 battery to OPERABLE within 2 hours or be in COLD SHUTDOWN within the following 36 hours

C. Restore both Div.1 batteries to OPERABLE within 2 hours or be in Cold Shutdown within 36 hours

D. Declare affected required feature(s) INOPERABLE, immediately

QUESTION SRO 4

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	2.2.36	
	Importance Rating		4.2

K&A: Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.

Generic

Explanation: **Answer B** – with RCS temperature > 250 deg. F, TS 3.8.4 is applicable. LCO Bases states either the Unit 1 or Unit 2 battery.

A – incorrect – while this cycles the pump breaker, this is not the minimum requirement

B – incorrect – this also cycles the pump breaker, but is not the minimum requirement

C – incorrect – this is performed if keep-fill is lost or voiding is suspected

Technical Reference(s): Tech Spec 3.8.4 & 3.8.5 and TS Bases 3.8.4 rev 1	Reference Attached: Tech Spec 3.8.4 & 3.8.5 and TS Bases 3.8.4 p B 3.8-53

Proposed references to be provided to applicants during examination: Technical Specifications 3.8.4 & 3.8.5

Learning Objective (As available): OT-3037-12-C

Question Source:	Bank #	Perry 2009 Audit
	Modified Bank #	
	New	

Question History: Previous NRC Exam:

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

10 CFR Part 55 Content:	55.41
	55.43b.2 x

Comments: Level of Difficulty = x

QUESTION SRO 5

The following plant conditions exist:

- A plant transient results in fuel damage
- The Shift Manager declared a Site Area Emergency 18 minutes ago
- It is necessary for an operator to enter the Containment to terminate the offsite radioactive release
- Radiation Protection estimates the expected exposure for this task is 3 Rem
- The operator volunteering for the task has 500 mrem TEDE for the year

As the Shift Manager acting as the Emergency Coordinator, describe the occupational dose limit(s) that will be exceeded, including the required approval(s) for this task?

A. Only the Site Administrative Dose Limit will be exceeded;
only your approval is required.

B. Only the Site Administrative Dose Limit will be exceeded;
your approval and the Radiation Protection Manager's approval are required.

C. The Site Administrative Dose Limit and the Federal Dose Limit will be exceeded;
only your approval is required.

D. The Site Administrative Dose Limit and the Federal Dose Limit will be exceeded;
your approval and the Radiation Protection Manager's approval are required.

QUESTION SRO 5

Examination Outline Cross-Reference			
	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	2.3.4	
	Importance Rating		3.7

K&A: Knowledge of radiation exposure limits under normal or emergency conditions.

Generic

Explanation: **Answer A** – based on the 3 rem exposure estimate and the 500 mr dose received to date, it is not expected to exceed federal Dose limits, but it does exceed Perry admin dose limits of 1000 mr. since the TSC has not been activated, only the Shift Managers approval is necessary for the increased exposure.

B & D – incorrect – RP managers approval is not required

C & D – incorrect – the federal dose limits will not be exceeded

Technical Reference(s): HPI-B-003 rev 26 & NOP-OP-4201 rev 0	Reference Attached: HPI-B-003 pp. 4-5 & NOP-OP-4201 p 18

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3037-03-I

Question Source:	Bank # Modified Bank # New	Perry Audit 2003
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43b.4 x	

Comments: Level of Difficulty = x

QUESTION SRO 6

The plant is operating at rated power.

At 0700 on 3/1/2010, while reviewing the Tech Spec Rounds, the Unit Supervisor discovers the following:

- The last time that a Channel Check was performed on Control Room Airborne radiation monitor D17-K776 was 24 hours ago

Per Technical Specifications, D17-K776 is __(1)__. The surveillance is required to be performed no later than __(2)__.

Reference provided: Tech Spec 3.3.7.1

	__(1)__	__(2__
A.	INOPERABLE	1300 on 3/2/2010
B.	OPERABLE	1300 on 3/2/2010
C.	INOPERABLE	0700 on 3/2/2010
D.	OPERABLE	0700 on 3/2/2010

QUESTION SRO 6

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#		2.3.15
	Importance Rating		3.1

K&A: Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Generic

Explanation: **Answer D** – TS SR 3.0.3 allows up to 24 hours to complete a missed surveillance. TS SR 3.0.2 - 25% extension is not allowed for a missed surveillance.

A & B – incorrect –TS SR 3.0.3 allows up to 24 hours to complete a missed surveillance – 3.0.2 - 25% extension is not allowed for a missed surveillance

A & C – incorrect –TS SR 3.0.3 states if a surveillance is missed then immediate entry into the LCO is not required

Technical Reference(s): Tech Spec 3.0.3, Tech Spec Bases 3.0.3 rev 4, and Tech Spec 3.3.7.1	Reference Attached: Tech Spec 3.0.3 p 3.0-4, Tech Spec Bases 3.0.3 pp. B 3.0-12 to 14, and Tech Spec 3.3.7.1 p 3.3-70 to 73

Proposed references to be provided to applicants during examination: Tech Spec 3.3.7.1

Learning Objective (As available): OT-3037-04-J & K

Question Source:	Bank #	Grand Gulf 2009	
	Modified Bank #		
	New		
Question History:	Previous NRC Exam:	Grand Gulf 2009	
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		x
10 CFR Part 55 Content:	55.41		
	55.43b.4 x		

Comments: Level of Difficulty = x

QUESTION SRO 7

The following conditions exist:

- An Alert has been declared
- The TSC is not yet activated

Which of the below listed duties are allowed to be delegated by the Shift Manager?

1. Provide information and assistance to the Public Information Organization.
2. Determine the emergency classification including reclassification or termination.
3. Recommend protective actions for the general public to State and local County Officials.
4. Coordinate and direct the actions necessary to terminate or mitigate the effects of the emergency.
5. Request mobilization of the Corporate Planning Assistance Center (CPAC) in accordance with NOBP-LP-5001
6. Provide an interface with FirstEnergy Corporation organizational management and senior levels of outside organizations.

A. 1,2,3,

B. 2,3,4

C. 3,4,5

D. 4,5,6

QUESTION SRO 7

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	2.4.40	
	Importance Rating		4.5

K&A: Knowledge of SRO responsibilities in emergency plan implementation.

Generic

Explanation: **Answer D** – these duties are allowed to be delegated by the Emergency Coordinator who, in this case, is the Shift Manager

A & B – incorrect – Classification of the event can not be delegated

B & C – incorrect – Protective Action Recommendations can not be delegated

Technical Reference(s): EPI-A2 rev 14	Reference Attached: EPI-A2 pp 3-5

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): EPL-0801-01-3

Question Source:	Bank # Modified Bank # New	 x
Question History:	Previous NRC Exam: x	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41 55.43b.5 x	

Comments: Level of Difficulty = x

QUESTION SRO 8

The following conditions exist:

- The plant is shutdown for a refuel outage
- Core Alterations are in progress
- High Pressure Core Spray Pump is out of service for a motor change out
- Division 1 ECCS pumps are out of service for LLRT testing
- Buses EH11, EH12, & EH13 are aligned to LH-1-A
- The Perry Transmission Yard East Bus (Bus # 1) is de-energized for maintenance

A lockout occurs on Unit 1 Startup transformer

Ten (10) minutes after the Unit 1 transformer lockout the following conditions exist:

- Division 1 Diesel Generator is operating supplying Bus EH11
- Division 2 Diesel Generator failed to start
- Division 3 Diesel Generator is operating supplying Bus EH13

As the Unit Supervisor, which abnormal procedure will you enter to mitigate the consequences of the conditions above?

A. ONI-R10, LOSS OF AC POWER - SBO

B. ONI-R10, LOSS OF AC POWER - TLAC

C. ONI-R10, LOSS OF AC POWER - LOOP

D. ONI-R22, Loss OF AN ESSENTIAL AND/OR A STUB 4.16KV BUS

QUESTION SRO 8

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	295003	AA2.05
	Importance Rating		4.2

K&A: Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: Whether a partial or complete loss of A.C. power has occurred

Partial or Complete Loss of AC / 6

Explanation: **Answer B** – With HPCS pump and Div 1 ECCS pumps out of service, having the Div 1 and Div 3 DGs start is irrelevant. The plant is still in a TLAC not an SBO.

A – incorrect – SBO would be entered if HPCS pump was available

C – incorrect – LOOP would be entered if either Bus EH11 or EH12 with associated ECCS pumps available

D – incorrect – ONI-R22-1 directs entry into ONI-R10 (and exit of ONI-R22-1)

Technical Reference(s): ONI-R10 rev 9 & ONI-R22-1 rev 8	Reference Attached: ONI-R10 p 3 & ONI-R22-1 p 5

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3035-18 (LP) A-4

Question Source:	Bank # Modified Bank # New	 x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43b.5 x	

Comments: Level of Difficulty = x

QUESTION SRO 9

Given the following conditions:

• D-1-A Voltage	125 VDC
• D-1-B Voltage	100 VDC
• ED-1-A Voltage	100 VDC
• ED-1-B Voltage	0 VDC
• ED-1-C Voltage	90 VDC
• Reactor Coolant Temperature	180F
• All control rods are inserted	

Entry into Off-normal instruction(s) __(1)__ is(are) required and, the Emergency Plan classification will be __(2)__.

Reference Provided: EPI-A1 Attachments 1 and 2

	(1)	(2)
A.	ONI-R42-2, LOSS OF DC BUS ED-1-B and ONI-R42-3, LOSS OF DC BUS ED-1-C	ES-1
B.	ONI-R42-2, LOSS OF DC BUS ED-1-B and ONI-R42-3, LOSS OF DC BUS ED-1-C	EU-1
C.	ONI-R42-2, LOSS OF DC BUS ED-1-B	ES-1
D.	ONI-R42-2, LOSS OF DC BUS ED-1-B	EU-1

QUESTION SRO 9

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	295004	2.4.11
	Importance Rating		4.2

K&A: Knowledge of abnormal condition procedures.

Partial or Total Loss of DC Pwr / 6

Explanation: **Answer D** – the Entry Condition for all R42 ONI's is Bus Voltage zero. The other 'entry conditions' in the ONI describe the effects of the loss of the DC bus. Therefore, only ONI-R42-2 should be entered. Additionally, Since the Plant is in Mode 4, EU-1 is the correct E-plan classification.

A & B – incorrect – voltage on ED-1-C is greater than the ONI-R42-3 Entry Condition voltage

A & C – incorrect – ES-1 would be correct if plant was in Mode 1, 2, or 3.

Technical Reference(s): ONI-R42-2 rev 6 & EPI-A001 rev 21	Reference Attached: ONI-R42-2 p 3 & EPI-A001 p 14

Proposed references to be provided to applicants during examination: EPI-A1 Attachments 1 and 2

Learning Objective (As available): OT-3035-05(LP)-A.2

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	
	55.43b.5	x

Comments: Level of Difficulty = x

QUESTION SRO 10

Surveillance SVI-C71-T0046, TURBINE STOP VALVE CLOSURE AND TURBINE CONTROL VALVE FAST CLOSURE CHANNEL FUNCTIONAL FOR 1C71-N006A, B, C, D, E, F, G, H AND 1C71-N005A, B, C, D commenced at 12:00 on June 2.

The Unit Supervisor signed the SVI for channel INOPERABILITY at 14:00 on June 2

At 16:00 on June 2, Turbine Stop Valve #3 failed to initiate a half scram signal to RPS when tested.

The Technical Specification Required Actions are to place channel in trip by __(1)__ on June 3. The bases of the Turbine Stop Valve closure scram is to ensure the __(2)__.

Reference Provided: Technical Specification 3.3.1.1 – partial – modified

	__(1)__	__(2)__
A.	04:00	fuel peak cladding temperature remains below the limits of 10CRF50.46
B.	08:00	fuel peak cladding temperature remains below the limits of 10CRF50.46
C.	04:00	Minimum Critical Power Ratio (MCPR) Safety Limit is not exceeded
D.	08:00	Minimum Critical Power Ratio (MCPR) Safety Limit is not exceeded

QUESTION SRO 10

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	295005	AA2.03
	Importance Rating		3.1

K&A: Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP: Turbine valve position

Main Turbine Generator Trip / 3

Explanation: **Answer C** – On a turbine trip with Rx power > 38%, the Turbine Stop valves generate a Rx scram signal when they go closed. Testing of the Stop Valves includes cycling the TSV to ensure a ½ scram signal is generated when each valve is < 93% open. Time of Discovery of an INOP condition starts the clock for Required Actions. In this case, Time of Discovery was 16:00. RA is to place channel in trip within 12 hours. The bases of the TSV closure scram is to prevent exceeding the MCPR safety limit.

A & B – incorrect – the bases of the TSV scram is to not exceed MCPR

B & D – incorrect – the Required Actions must be completed within 12 hours of time of discovery – this answer is plausible if candidate believes that the 6-hour note for surveillance testing applies to completing the Required Actions

Technical Reference(s): Tech Spec 3.3-1, Tech Spec 1.3, & Tech Spec Bases B 3.3.1.1 rev 0	Reference Attached: Tech Spec 3.3-1 pp. 3.3-1&3, Tech Spec 1.3 p 1.0-11, & Tech Spec Bases B 3.3.1.1 pp. B 3.3-14&16

Proposed references to be provided to applicants during examination: Technical Specification 3.3.1.1 – partial – modified

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

Question Source:	Bank # Modified Bank # New	 x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43b.2 x	

Comments: Level of Difficulty = x

QUESTION SRO 11

Which one of the following identifies a condition that would require declaration of an ALERT only?

Reference Provided: EPI-A1 Attachments 1 and 2

A.
- RPV level is lowering slowly
- Fifty Control Rods are withdrawn at various positions
- Reactor Power is 10%
- Mode Switch is in SHUTDOWN

B.
- RPV level at 175 inches and lowering
- Control Rods 30-31, 18-55, 14-15, 46-15, 46-31, & 46-47 are at position 48
- Nuclear Instruments are fully inserted
- IRM's are indicating on range 4

C.
- Mode Switch is placed in SHUTDOWN
- Control Rod 46-15 is at position 48
- Nuclear Instruments are fully inserted
- Power is indicating middle of the source range

D.
- RPV level at 179 inches and lowering
- Mode Switch is placed in SHUTDOWN
- All Control Rods indicate 00
- Nuclear Instruments are fully inserted
- Power is indicating middle of the source range

QUESTION SRO 11

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	295006	2.4.41
	Importance Rating		4.6

K&A: Knowledge of the emergency action level thresholds and classifications.

SCRAM / 1

Explanation: **Answer B** – this is a CA1 – failure to auto scram and power below 4%

A – incorrect – this should be a CS1

C – incorrect – no auto scram setpoint exceeded – Rx is S/D under all conditions without boron

D – incorrect – no auto scram setpoint exceeded – no EAL entry

Technical Reference(s): EPI-A1 rev 21	Reference Attached: EPI-A1 pp. 14 & 23

Proposed references to be provided to applicants during examination: EPI-A1 Attachments 1 and 2

Learning Objective (As available): EPL-0804-01-4

Question Source:	Bank # Modified Bank # New	Grand Gulf 2004
Question History:	Previous NRC Exam:	Grand Gulf 2004
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43b.5 x	

Comments: Level of Difficulty = x

QUESTION SRO 12

Refueling activities are in progress.

The following conditions exist on the Refuel Floor:

- A fuel bundle just arrived from the Fuel Handling Building with the IFTS Upender vertical
- A fuel bundle is in transit from the RPV to the fuel storage racks via F15 Bridge
- An unexplained drop in upper pool level occurs

Which of the following actions is required concerning the status of the two bundles?

A. Incline the IFTS Upender and return the fuel bundle on the F15 Bridge back to any open vessel location.

B. Incline the IFTS Upender and continue the fuel movement with the F15 Bridge to the fuel storage racks.

C. Continue fuel movement with the F15 Bridge to the fuel storage racks, then, transfer the fuel bundle in IFTS to the fuel storage racks.

D. Transfer the fuel bundle in IFTS down to the Fuel Handling Building and return the fuel bundle on the F15 Bridge back to the vessel location from which it was removed.

QUESTION SRO 12

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	295023	AA2.02
	Importance Rating		3.7

K&A: Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS: Fuel pool level

Refueling Acc / 8

Explanation: **Answer B** – > The stem contains entry conditions described for ONI-E12-2 entry. The ONI also describes Safe Conditions. FTI-D009 further restricts placing a bundle back into the Rx after it has been off-loaded.

A – incorrect – Moving the bundle back to the core is an unsafe act/condition

C – incorrect – An IFTS transfer to the FHB is an unsafe act/condition

D – incorrect – An IFTS transfer to the FHB is an unsafe act/condition - moving the bundle back to the core is an unsafe act/condition

Technical Reference(s): ONI-E12-2 rev 25 & FTI-D009 rev 13	Reference Attached: ONI-E12-2 pp. 3-7 & FTI-D009 p 9

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3035-11(LP)-A.2 & OT-3602-01-D.4 & E.2

Question Source:	Bank #	RQL-0748
	Modified Bank #	
	New	
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	
	55.43b.7 x	

Comments: Level of Difficulty = x

QUESTION SRO 13

The plant was operating at 100% power when a LOCA occurred.

The following conditions now exist:

- LPCI A and LPCS pumps are injecting
- Adequate core cooling is being maintained
- NO other ECCS pumps are available
- Suppression Pool temperature is rising

Based on the information from the attached SPDS screen print, select the statement below that describes the use of RHR 'A' Loop for Suppression Pool cooling.

Reference Provided: SPDS screen print

A. LPCI 'A' must be diverted to Suppression Pool Cooling to ensure that Suppression Pool temperature is maintained below the Heat Capacity Limit, since LPCS can maintain adequate core cooling through spray cooling alone.

B. LPCI 'A' may be diverted to Suppression Pool Cooling as long as LPCS is able to maintain RPV water level above -25 inches.

C. LPCI 'A' must be diverted to Suppression Pool Cooling, irrespective of adequate core cooling, when neither Suppression Pool temperature nor Reactor pressure can not be maintained below the Heat Capacity Limit (HCL)

D. LPCI 'A' may be diverted to Suppression Pool Cooling as long as LPCS is able to maintain RPV water level above -42.5 inches

QUESTION SRO 13

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	295026	2.1.19
	Importance Rating		3.8

K&A: Ability to use plant computers to evaluate system or component status.

Suppression Pool High Water Temp. / 5

Explanation: **Answer B** – The Print shows the scram was successful and ED was performed. Injection sources needed for adequate core cooling are not allowed to be diverted to suppression pool cooling. However, adequate core cooling is defined as maintaining RPV level > -25" with injection. So, if LPCS can maintain RPV level > -25" adequate core cooling is assured.

A & C – incorrect – no directions are given tin the EOP's to divert injection sources required for adequate core cooling to suppression pool cooling to avoid exceeding HCL.

D – incorrect – the – 42.5" is allowable if no injection to the RPV is available

Technical Reference(s): EOP-Bases rev 0 & EOP-2 Bases rev 0	Reference Attached: EOP-Bases pp 33-35 & EOP-2 Bases pp. 22-23

Proposed references to be provided to applicants during examination: SPDS screen print

Learning Objective (As available): 3402-01-C.1, C.30, C.1.c

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	
	55.43b.5	x

Comments: Level of Difficulty = x

QUESTION SRO 14

While the Unit Supervisor was reviewing Technical Specification Rounds he discovered the following:

- Containment Average Air Temp 78^{o}F
- Suppression Pool validated level 17.2 feet
- Half of the suppression pool temperature detectors on H13P868 and H13-P869 read 79^{o}F
- Half of the suppression pool temperature detectors on H13P868 and H13-P869 read 85^{o}F

Based on the readings from Tech Spec Rounds, select the correct statement below.

A. All Suppression Pool temperature instruments are OPERABLE

B. The Suppression Pool temperature instruments reading 79^{o}F are OPERABLE

C. The Suppression Pool temperature instruments reading 85^{o}F are OPERABLE

D. None of the Suppression Pool temperature instruments are OPERABLE

QUESTION SRO 14

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	295030	EA2.02
	Importance Rating		3.9

K&A: Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: Suppression pool temperature.

Low Suppression Pool Wtr Lvl

Explanation: **Answer C** – Eight of sixteen SP level instruments are uncovered and exposed to the containment atmosphere when SP level drops below 17.33'. In order for the SP level instruments to perform their intended function (Operable) they must be submerged.

A – incorrect – the instruments not submerged are not operable

B – incorrect – these instruments are reading containment air temperature and are not operable

D – incorrect – the instruments still submerged are operable – plausible is candidate is confused about which sets of instruments are reading accurately

Technical Reference(s): Tech Spec 1.0, Tech Spec Bases 3.3 rev 1, & Dwg 240-082 rev K	Reference Attached: Tech Spec 1.0 p 1.0-5, Tech Spec Bases 3.3 p B 3.3-54 & Dwg 240-082

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-D23.C & D

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	
	55.43b.2	x

Comments: Level of Difficulty = x

QUESTION SRO 15

The following conditions exist:

- Drywell pressure is 2.0 psig
- Reactor pressure is 870 psig
- Reactor is shutdown

Given these conditions, EOP-2 Primary Containment Control contains steps to have the Unit Supervisor direct prevention of LPCS and LPCI injection if not needed for adequate core cooling.

What is the basis for this action?

A. Facilitate RPV level control.

B. Facilitate RPV pressure control.

C. Reduce thermal stress on the Reactor vessel caused by cold water injection.

D. Prevent the possibility of a reactor power excursion large enough to severely damage the core.

QUESTION SRO 15

<table>
<tr><td rowspan="5">Examination Outline Cross-Reference</td><td>Level:</td><td>RO</td><td>SRO</td></tr>
<tr><td>Tier #</td><td></td><td>1</td></tr>
<tr><td>Group #</td><td></td><td>2</td></tr>
<tr><td>K/A#</td><td>295010</td><td>2.4.18</td></tr>
<tr><td>Importance Rating</td><td></td><td>4.0</td></tr>
<tr><td colspan="4">K&A: Knowledge of the specific bases for EOPs.</td></tr>
<tr><td colspan="4">High Drywell Pressure / 5</td></tr>
<tr><td colspan="4">Explanation: Answer A – > Per the bases, low pressure systems are prevented so as not to complicate RPV water level control.

B – incorrect – plausible since injection of cold water can also affect RPV pressure, however, not per the Bases

C – incorrect – controlling cooldown rates controls thermal stress – not per the Bases

D – incorrect – this is true if the reactor is not shutdown – Rx s/d given in stem</td></tr>
<tr><td colspan="2">Technical Reference(s): EOP-2 Bases rev 0</td><td colspan="2">Reference Attached: EOP-2 Bases pp. 38-39</td></tr>
<tr><td colspan="4">Proposed references to be provided to applicants during examination: None</td></tr>
<tr><td colspan="4">Learning Objective (As available): OT-3402-01-C.44 & -08-B.1</td></tr>
<tr><td>Question Source:</td><td colspan="3">Bank #
Modified Bank #
New Monticello 2002</td></tr>
<tr><td>Question History:</td><td colspan="3">Previous NRC Exam: Monticello 2002</td></tr>
<tr><td>Question Cognitive Level:</td><td colspan="3">Memory or Fundamental Knowledge x
Comprehension or Analysis</td></tr>
<tr><td>10 CFR Part 55 Content:</td><td colspan="3">55.41
55.43b.5 x</td></tr>
<tr><td colspan="4">Comments: Level of Difficulty = x</td></tr>
</table>

QUESTION SRO 16

The following conditions exist:

- The plant is operating at rated power
- A loss of drywell cooling occurs
- Drywell average air temperature has reached 150°F and continues to slowly rise

Per the Bases for Technical Specification 3.6.5.5, Drywell Air Temperature, if this condition is not corrected, then exceeding the ____ during a design basis LOCA can not be ensured.

A. limit for hydrogen concentration in the drywell

B. drywell pressure suppression capability

C. heat capacity of the suppression pool

D. drywell design temperature limit

QUESTION SRO 16

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		2
	K/A#	295012	AA2.01
	Importance Rating		3.9

K&A: Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: Drywell temperature.

High Drywell Temperature / 5

Explanation: **Answer D** – > The Tech Spec Bases for high DW temperature states the design drywell temperature of 330 degrees cannot be ensured following a design basis LOCA if drywell temperature is not maintained below 145 degrees.

A - incorrect – this is the containment HDOL limit

B - incorrect – the DW pressure suppression capability is a function of DW bypass leakage

C - incorrect - the Heat capacity limit is a function of RPV pressure and suppression pool temperature

Technical Reference(s): Tech Spec Bases 3.6.5.5 rev 7	Reference Attached: Tech Spec Bases 3.6.5.5 p B 3.6-148

Proposed references to be provided to applicants during examination: None

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

Question Source:	Bank #	Grand Gulf 2009
	Modified Bank #	
	New	
Question History:	Previous NRC Exam:	Grand Gulf 2009
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	
	55.43b.2 x	

Comments: Level of Difficulty = x

QUESTION SRO 17

The following plant conditions exist:

- Reactor water level is -10 inches
- Reactor pressure is 900 psig
- Reactor power is 5%
- Drywell pressure is + 1.1 psig
- Drywell temperature is 130^{o}F
- Containment pressure is +1.5 psig.
- Containment temperature is 100^{o}F
- Suppression Pool temperature is 116^{o}F
- Suppression Pool level is 24.8 feet
- All appropriate EOP's have been entered

As the Unit Supervisor, which one of the following actions should you direct, including the bases for this action?

Reference Provided: Technical Specification 3.3.1.1

A. Open the Main Turbine Bypass Valves and depressurize the reactor; the RPV cannot be permitted to remain at pressure if operation of SRVs may cause the SRV discharge line or associated components to fail.

B. Open the Main Turbine Bypass Valves and depressurize the reactor; if primary containment water level rises above the elevation of the SRV solenoids, the SRVs may no longer be operable.

C. Open the ADS valves and depressurize the reactor; the RPV cannot be permitted to remain at pressure if operation of SRVs may cause the SRV discharge line or associated components to fail.

D. Open the ADS valves and depressurize the reactor; if primary containment water level rises above the elevation of the SRV solenoids, the SRVs may no longer be operable.

QUESTION SRO 17

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		2
	K/A#	295029	2.1.32
	Importance Rating		4

K&A: Ability to explain and apply system limits and precautions.

High Suppression Pool Wtr Lvl / 5

Explanation: **Answer C** – > The SRV Tail Pipe Level Limit is 24.5'. Operation of SRV's with SP level above this could result in damage to the SRV discharge lines and direct pressurization of containment. Since the reactor is not shutdown, Anticipate ED is not allowed. ED by SRV is required.

A & B – incorrect – using the turbine bypass valves to anticipate ED is not allowed when Rx not shutdown

D – incorrect – the high SP level can affect the MSIV drain valves not the SRV solenoids

Technical Reference(s): EOP-2 Bases rev 0	Reference Attached: EOP-2 Bases p 35

Proposed references to be provided to applicants during examination: None

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

Question Source:	Bank #	Perry 2003 Audit
	Modified Bank #	
	New	

Question History: Previous NRC Exam:

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

10 CFR Part 55 Content:	55.41
	55.43b.5 x

Comments: Level of Difficulty = x

QUESTION SRO 18

The following conditions exist:

- First startup after refueling outage in progress
- The IRM Linearity Check was performed

The table below shows the IRM linearity data after final adjustments by I&C.

IRM LINEARITY TABLE				
IRM	Range 6 (40 to 75)	Range 6 Divided By 3.1623	Range 7	Sat / Unsat ± 3/125
A	65	20.5547	21	S / U
E	58	18.3411	19	S / U
C	49	15.4951	19	S / U
G	55	17.3924	14	S / U
B	45	14.2301	13	S / U
F	65	20.5547	22	S / U
D	50	15.8113	16	S / U
H	51	16.1275	20	S / U

Based on the above data, which of the following actions is required?

Reference provided: Technical Specification 3.3.1.1 – Partial (Modified)

A. Within 1 hour, insert a half scram.

B. Within 6 hours, either place one IRM channel in trip or insert a half scram.

C. Within 12 hours, place the plant in MODE 3.

D. Within 12 hours, either place one IRM channel in trip or insert a half scram.

QUESTION SRO 18

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A#	215003	A2.05
	Importance Rating		3.5

K&A: Ability to (a) predict the impacts of the following on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Faulty or erratic operation of detectors/system

IRM

Explanation: **Answer D** – IRM channels C, G, & H are out of spec. Channels G & H are on trip system B. TS require 3 of 4 channels per trip system. TS 3.3.1.1 Condition A is entered since only Trip System B is below minimum number of channels. The channel is placed in trip IAW the Trip Instructions. Or the trip system is placed in trip (half scram inserted)

A – incorrect – This the action from Condition H which is not required at this time - incorrect use of the table

B – incorrect – This is the action from Condition B - it is a misconception that both trip systems are affected; Trip System B still has three channels

D – incorrect – This is the action from Condition C, which is not entered since no loss of function has occurred

Technical Reference(s): IOI-1 rev 30 & Tech Spec 3.3.1.1	Reference Attached: IOI-1 28-29 & Tech Spec 3.3.1.1p 3.3-1

Proposed references to be provided to applicants during examination: Technical Specification 3.3.1.1 – Partial (Modified)

Learning Objective (As available): OT-COMBINED-C51_IRM-J.1

Question Source:	Bank #	RQL-0934
	Modified Bank #	
	New	
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	
	55.43b.2 x	

Comments: Level of Difficulty = x

QUESTION SRO 19

The following conditions exist:

- The plant is operating at full power
- Work was performed on the 1E51-F031, RCIC PUMP SUPR PL SUCT ISOL
- 1E51-F031 is now down powered
- RCIC is in standby with suction aligned to the Condensate Storage Tank

A Technician Specification function of RCIC is to __(1)__.
RCIC is __(2)__.

Reference Provided: PMT data

	__(1)__	__(2)__
A.	maintain coolant inventory, as well as vessel level, if a small break occurs in the RPV while the RCS is still pressurized	OPERABLE
B.	maintain coolant inventory, as well as vessel level, if a small break occurs in the RPV while the RCS is still pressurized	INOPERABLE
C.	operate either automatically or manually following a loss of coolant flow from the feedwater system to provide adequate core cooling and control of RPV water level	OPERABLE
D.	operate either automatically or manually following a loss of coolant flow from the feedwater system to provide adequate core cooling and control of RPV water level	INOPERABLE

QUESTION SRO 19

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A#	217000	2.1.27
	Importance Rating		4

K&A: Knowledge of system purpose and/or function.

RCIC

Explanation: **Answer D** – The function of RCIC is to provide water for adequate core cooling if feedwater is isolated. The Suppression Pool suction is INOP because it is down powered. The Tech Spec required suction source is the Suppression Pool. With RCIC aligned to the CST and the SP suction INOP, the TS required source is unable to support the TS required function and RCIC is INOP.

A & C – incorrect – RCIC is INOP due to not being on the SP

A & B – incorrect – this is the Tech Spec function of HPCS

Technical Reference(s): Tech Spec Bases 3.5.3 rev 5 and Tech Spec Bases 3.5.1 rev 0	Reference Attached: Tech Spec Bases p B 3.5-21 & B 3.5-1

Proposed references to be provided to applicants during examination: PMT data

Learning Objective (As available): OT-3037-09-B

Question Source:	Bank #	
	Modified Bank #	
	New	x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	
	55.43b.2	x

Comments: Level of Difficulty = x

QUESTION SRO 20

The following conditions exist:

- The plant is at 100% power
- I&C is performing surveillance SVI-B21-T0072-C, MAIN STEAM LINE LOW PRESSURE CHANNEL C CALIBRATION FOR 1B21-N076C
- All other MSL Low Pressure instruments are OPERABLE
- The technician reports to you that he is unable to adjust the 1B21-N076C instrument within the allowable value.

Based on this information, ____.

Reference Provided: Technical Specification 3.3.6.1 – partial – modified

A. enter TS 3.3.6.1 Condition A and place channel in trip within 12 hours

B. enter TS 3.3.6.1 Condition A and place channel in trip within 24 hours

C. no action is required at this time; the allowable value is <u>not</u> the required condition

D. no action is required at this time; only 2 of the 4 channels are required to be operable

QUESTION SRO 20

<table>
<tr><td rowspan="5">Examination Outline Cross-Reference</td><td>Level:</td><td>RO</td><td>SRO</td></tr>
<tr><td>Tier #</td><td></td><td>2</td></tr>
<tr><td>Group #</td><td></td><td>1</td></tr>
<tr><td>K/A#</td><td>223002</td><td>A2.08</td></tr>
<tr><td>Importance Rating</td><td></td><td>3.1</td></tr>
</table>

K&A: 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: Surveillance testing

PCIS/Nuclear Steam Supply Shutoff

Explanation: **Answer B** – per Tech Spec Bases, the appropriate condition and required action must be entered if any one of the 4 MSL low presure transmitters is INOP. Twenty-four hours is correct because this is for Function 1b

A – incorrect – 12 hours is for Functions 2b, 5b, & 5d

C – incorrect –- plausible if misinterprets Allowable Value for Leave As Is Zone

D – incorrect – required channels are 2 per trip system, with 4 channels total in 2 trip systems

Technical Reference(s): Tech Spec 3.3.6.1 & Tech Spec Bases B 3.3.6.1 rev 4, SVI-B21-T0072-C rev 5	Reference Attached: Tech Spec 3.3.6.1 pp. 48 & 54 and Tech Spec Bases p B 3.3-142

Proposed references to be provided to applicants during examination: Technical Specification 3.3.6.1 – partial – modified

Learning Objective (As available): OT-COMBINED-B21(NS4)-I

Question Source:	Bank # Modified Bank # New	 x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43b.2 x	

Comments: Level of Difficulty = x

QUESTION SRO 21

The following conditions exist:

- The reactor scrammed due to a small-break LOCA
- The only available injection source is from the Condensate Transfer system
- To maximize injection, Emergency Depressurization was initiated approximately 20 minutes ago and all ADS SRVs were verified open
- The SRV OPEN annunciator just reset

You have directed the Reactor Operator to verify the status of the ADS SRVs.

The ADS SRVs are reported to be ____.

A. closed based on stable SRV tailpipe temperatures. You would direct the panel operators to open SRVs using their control switches.

B. closed based on SRV tailpipe temperatures slowly lowering. You would direct the panel operators to use alternate methods of depressurizing the reactor vessel.

C. open based on SRV tailpipe temperatures of approximately 250°F and stable. Injection is occurring. You would direct operators to monitor reactor vessel level.

D. open based on SRV tailpipe temperatures of approximately 330°F and slowly rising due to the lack of injection. You would direct operators to open additional SRVs and continue to monitor for injection.

QUESTION SRO 21

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A#	239002	2.4.46
	Importance Rating		4.2

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

SRVs

Explanation: **Answer C** – > With ED occurring 20 minutes ago and the only injection source being CTS, RPV pressure will lower to < 30 psig. This will cause the SRV OPEN annunciator to reset and the SRV Open/Close lights to change state. The SRVs are verified open by observing SRV tailpipe temperature of 250°F which corresponds to reactor pressure of ~25 psig.

A & B incorrect – SRVs are still open

D – incorrect – tailpipe temperature of 330°F corresponds to normal reactor pressure

Technical Reference(s): ARI-H13-P601-019 rev 11 & ABB Steam Tables	Reference Attached: ARI-H13-P601-019 p 17

Proposed references to be provided to applicants during examination: None

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

Question Source:	Bank #	Perry 2005
	Modified Bank #	
	New	

Question History: Previous NRC Exam: Perry 2005

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

10 CFR Part 55 Content:	55.41	
	55.43b.5	x

Comments: Level of Difficulty = x

QUESTION SRO 22

The following conditions exist:

- The Mode Switch is in SHUTDOWN
- All control rods are inserted
- Reactor coolant temperature is 250°F
- Buses EH11 and EH13 are aligned to the Preferred Source
- Bus EH12 is aligned to the Alternate Preferred Source
- Division 2 Diesel generator is out of service and tagged out
- AEGT System A is in operation

A grid disturbance then occurs resulting in the following:

- Loss of power to the Unit 2 Startup Transformer
- A non-valid Division 2 RHR-LOCA signal is generated

No operator action have been performed.

The status of the A and B AEGT systems is __(1)__, requiring you to enter __(2)__ 3.6.4.3 AEGTS LCO in accordance with OAI-1701, Tracking Of LCOs.

	(1)	(2)
A.	AEGTS A is running AEGTS B is running	a Potential or NO
B.	AEGTS A is running AEGTS B is not running	an Active
C.	AEGTS A is not running AEGTS B is running	a Potential or NO
D.	AEGTS A is not running AEGTS B is not running	an Active

QUESTION SRO 22

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A#	261000	A2.07
	Importance Rating		2.8

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

SGTS

Explanation: **Answer B** – A loss of the Unit 2 S/U transformer results in the loss of Bus EH12 since Div. 2 DG is tagged out. Therefore, AEGT train (B) will not start on a LOCA signal. Since the LOCA signal is a Div 2 signal, it will not affect the Div 1 AEGT fan. Although the plant is shutdown, RCS temperature is >200°F which is Mode 3. Therefore, TS 3.6.4.3 is applicable.

A – incorrect – AEGTS B train is Not running due to the loss of 480VAC power – this is an active LCO

C – incorrect – AEGTS A train continues to run and AEGT B can not run – this is an active LCO

D – incorrect – AEGTS A train continues to run

Technical Reference(s): oai-1701 rev 11, Tech Spec 3.6.4.3 and Tech Spec Bases B 3.6.4.3 rev 7	Reference Attached: Tech Spec 3.6.4.3 p 3.6.56 and Tech Spec Bases p B 3.6-119

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): x

Question Source:	Bank # Modified Bank # New	 x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43b.2 x	

Comments: Level of Difficulty = x

QUESTION SRO 23

The following conditions exist:

- The plant is operating at 100% power with a high load line
- FMEOD (Fraction of Maximum Extended Operating Domain) at 0.985.

During a planned power reduction to 80% using the Recirc Flow Control Valves, the SCC Dispatcher requests Perry to hold power at 90%.

A short time later, the Reactor Operator reports that FMEOD is 0.992.

The Unit Supervisor is required to restore FMEOD to ≤ 0.990 by ____.

A. <u>only</u> inserting Cram Rods

B. <u>only</u> inserting control rods per Reactor Engineering direction

C. entering ONI-C51, Unplanned Change in Reactor Power or Reactivity <u>and</u> inserting Cram Rods

D. entering ONI-C51, Unplanned Change in Reactor Power or Reactivity <u>and</u> inserting control rods per ONI direction

QUESTION SRO 23

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		2
	K/A#	201005	2.1.32
	Importance Rating		4.0

K&A: 2.1.32 Ability to explain and apply system limits and precautions.

RCIS

Explanation: **Answer B** – Per IOI-3, P&L 2.2, if FMEOD is > 0.990, then insert control rods to restore FMEOD to <0.990 per Reactor Engineering direction.

A – incorrect – IOI-3, P&L 2.2 does not direct use of Cram Rods

C & D – incorrect – per ONI-C51Entry Conditions and IOI-3, P&L 2.2, ONI-C51 is only entered when FMEOD is > 1.000

Technical Reference(s): IOI-3 rev 24 & ONI-C51 Chart rev I	Reference Attached: ONI-C51 p 5 & ONI-C51 flowchart

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3035-08(LP)-B.4.B

Question Source:	Bank # Modified Bank # New	Perry 2007-1
Question History:	Previous NRC Exam:	Perry 2007-1
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 x
10 CFR Part 55 Content:	55.41 55.43b.1 x	

Comments: Level of Difficulty = x

QUESTION SRO 24

The plant was operating at rated power, when a transient occurred, resulting in the following conditions:

- All control rods are inserted
- Drywell pressure is 3.0 psig and stable
- Reactor pressure is 950 psig and stable
- RPV water level is (zero) 0 inches and lowering slowly
- No injection sources are currently available
- RWCU Isolations have been bypassed per EOP-SPI 2.6, Bypass Of RWCU Isolations
- RWCU blowdown rate is 60 gpm and stable
- RWCU Recirculation and Blowdown Modes are being used to control RPV pressure
- The Unit Supervisor entered EOP-4-3, Steam Cooling

Based on these conditions, Emergency Depressurization should __(1)__ delayed. And, with regard to the RWCU system, the Unit Supervisor will direct the Reactor Operator to __(2)__.

	__(1)__	__(2)__
A.	be	Secure Blowdown Mode and maintain Recirculation Mode
B.	be	Maintain Blowdown Mode and secure Recirculation Mode
C.	not be	Secure Blowdown Mode and maintain Recirculation Mode
D.	not be	Maintain Blowdown Mode and secure Recirculation Mode

QUESTION SRO 24

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		2
	K/A#	204000	A2.12
	Importance Rating		2.8

K&A: Ability to (a) predict the impacts of the following on the REACTOR WATER CLEANUP SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Excessive drain flow rates

RWCU

Explanation: **Answer A** – Based on RPV level trend, Emergency Depressurization will be required prior to reaching -25 inches. RWCU is draining 60 gpm from vessel inventory. This is resulting in lowering RPV level. When RPV level drops below 0" with no injection sources operating, Steam Cooling is entered. When in Steam Cooling, it is appropriate to secure RWCU from blowdown mode to preserve RPV inventory. With no injections sources available, ED should be delayed as long as possible.

B & D – incorrect – blowdown mode should be secured

C & D – incorrect – with no injection sources available, ED should be delayed as long as possible

Technical Reference(s): EOP-1 Bases rev 1	Reference Attached: EOP-1 Bases pp. 47, 55, & 76

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3402-01-C-29 & 02-F,

Question Source:	Bank #	
	Modified Bank #	Dresden 2009
	New	
Question History:	Previous NRC Exam:	Dresden 2009
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	
	55.43b.5	x

Comments: Level of Difficulty = x

QUESTION SRO 25

The following conditions exist:

- The reactor is shutdown
- Control Rod Drive Hydraulic system is shutdown

Which of the following conditions require system manipulations to be directly supervised by the Shift Manager or Unit Supervisor per OAI-0201, Operations General Instructions and Operating Practices?

A. Adjusting FPCC Upper Pool flow while in MODE 3

B. Placing a RWCU Filter/Demin in service while in MODE 4

C. Shifting NCC heat exchangers from ‘A’ to ‘C’ while in MODE 4

D. Placing RHR B loop in Suppression Pool Cooling with RHR A loop is in Shutdown Cooling while in MODE 3

QUESTION SRO 25

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		2
	K/A#	216000	2.1.9
	Importance Rating		4.5

K&A: 2.1.9 Ability to direct personnel activities inside the control room.

Nuclear Boiler Inst.

Explanation: **Answer D** – With CRDH shutdown, reference leg purge is out of service. With reference leg purge out of service, accuracy of the RPV water level instruments is questioned. OAI-0201 requires compensatory measures when performing activities with a potential to drain the vessel with the reference leg purge is out of service while in Modes 1, 2, or 3. With RHR in SDC, valve manipulations on any part of the RHR system must be supervised by the SM or US.

A – incorrect – with plant not in Mode 5, adjusting upper pool flow does not affect RPV level

B – incorrect – OAI-0201 specifies RWCU system manipulations in Mode 3 must be supervised – not Mode 4

C – incorrect – per Risk Management, if the plant was in Mode 5 and NCC was affecting Decay Heat removal, this would require supervisors oversight

Technical Reference(s): OAI-0201 rev 20	Reference Attached: OAI-0201 pp. 13-14

Proposed references to be provided to applicants during examination: None

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

Question Source:	Bank # Modified Bank # New	 x
Question History:	Previous NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41 55.43b.5 x	

Comments: Level of Difficulty = x

RO Question 18

PERRY NUCLEAR POWER PLANT	Procedure Number: PDB-A0016	
Title: Decay Heat Curve	Use Category: In-Field Reference	
	Revision: 9	Page: 1 of 7

DECAY HEAT CURVE

Functional Location (1J11)

Plant Data Book

Effective Date: 1-29-09

Preparer: Patrick Curran / 11-24-08
Date

RO Question 18

1.0 REFERENCES

1.1 Discretionary

Calculation FM-060, Decay Heat Analysis for Perry RFO12

1.2 Obligations

None

Commitments addressed in this document:

None

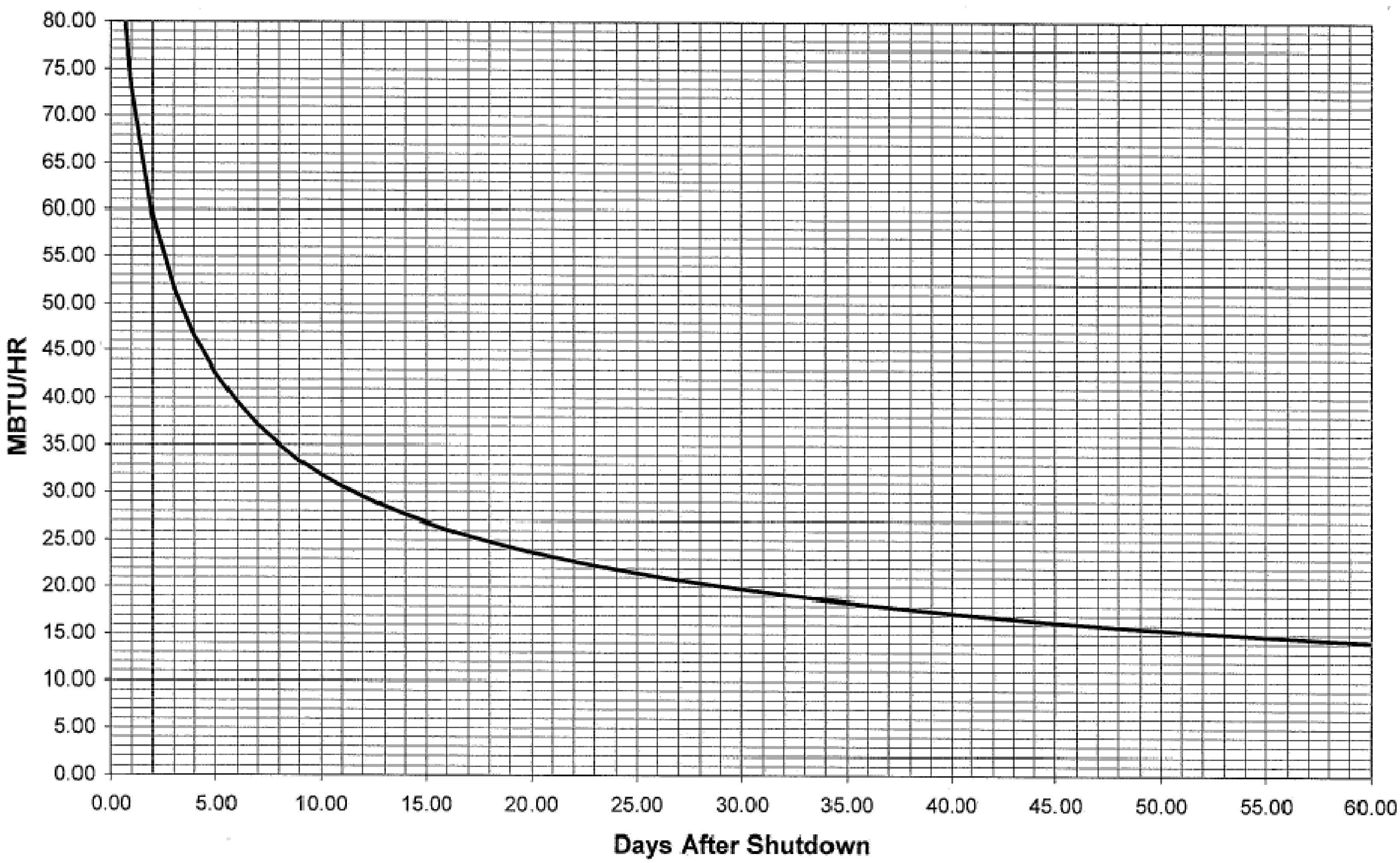
2.0 SCOPE OF REVISION

Rev. 9 1. Update Decay Heat Curves for RFO12.

RO Question 18

PERRY NUCLEAR POWER PLANT	Number: PDB-A0016	
Title: Decay Heat Curve	Use Category: In-Field Reference	
	Revision: 9	Page: 3 of 7

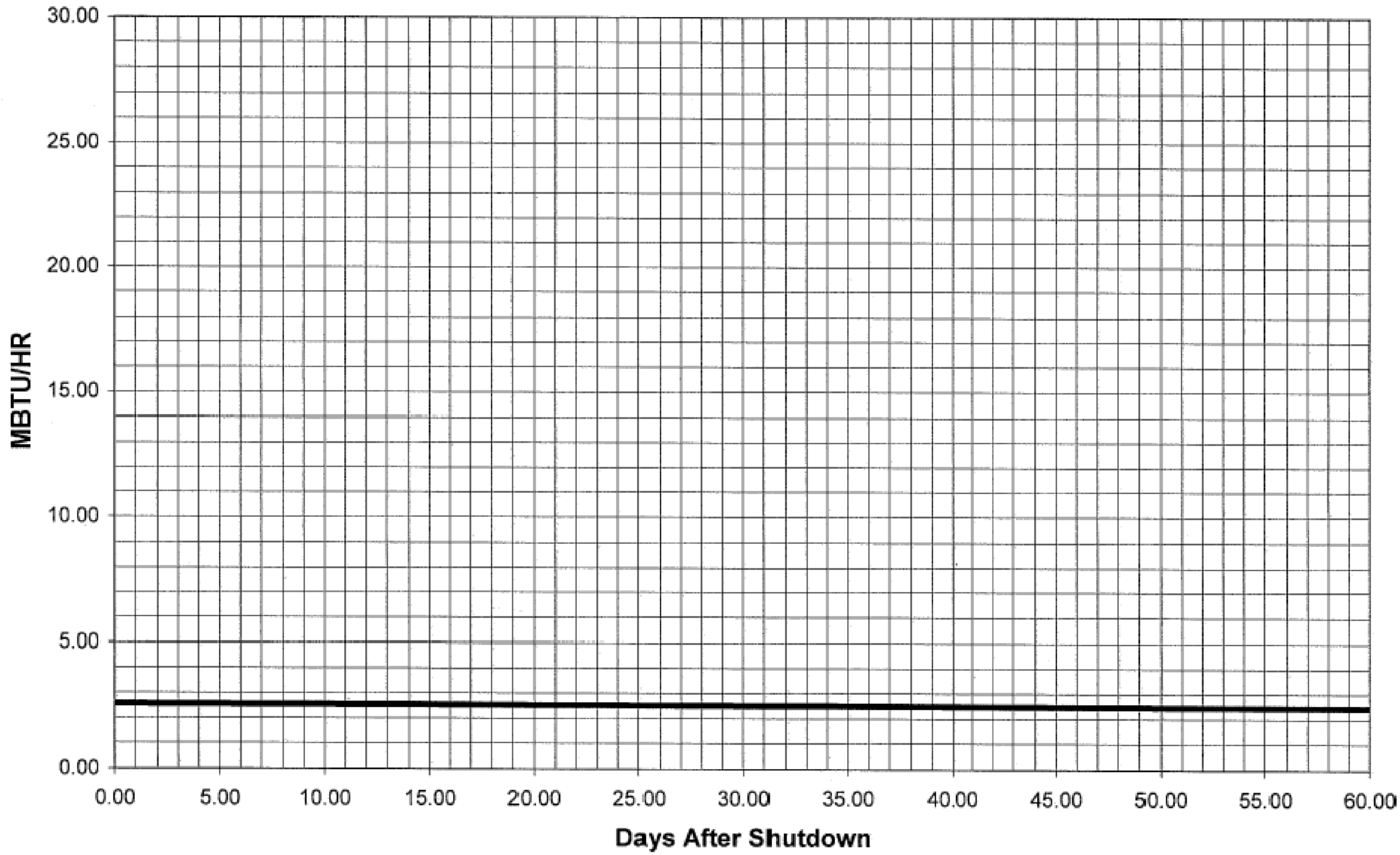
RFO12 In-Vessel Decay Heat Before Fuel Discharge



RO Question 18

PERRY NUCLEAR POWER PLANT		Number: PDB-A0016	
Title: Decay Heat Curve		Use Category: In-Field Reference	
		Revision: 9	Page: 4 of 7

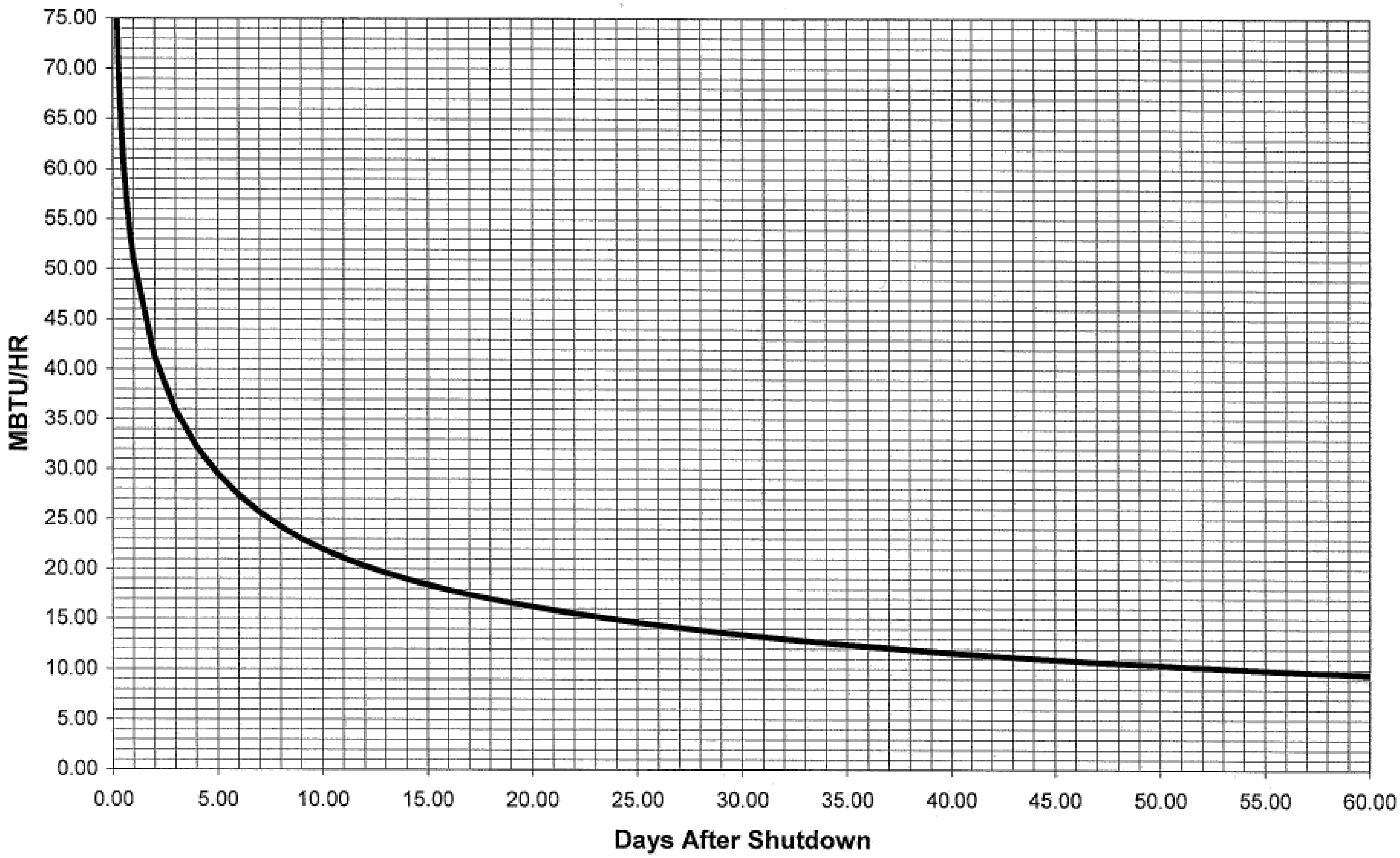
RFO12 Fuel Pool Decay Heat Before Fuel Discharge



RO Question 18

PERRY NUCLEAR POWER PLANT	Number: PDB-A0016	
Title: Decay Heat Curve	Use Category: In-Field Reference	
	Revision: 9	Page: 5 of 7

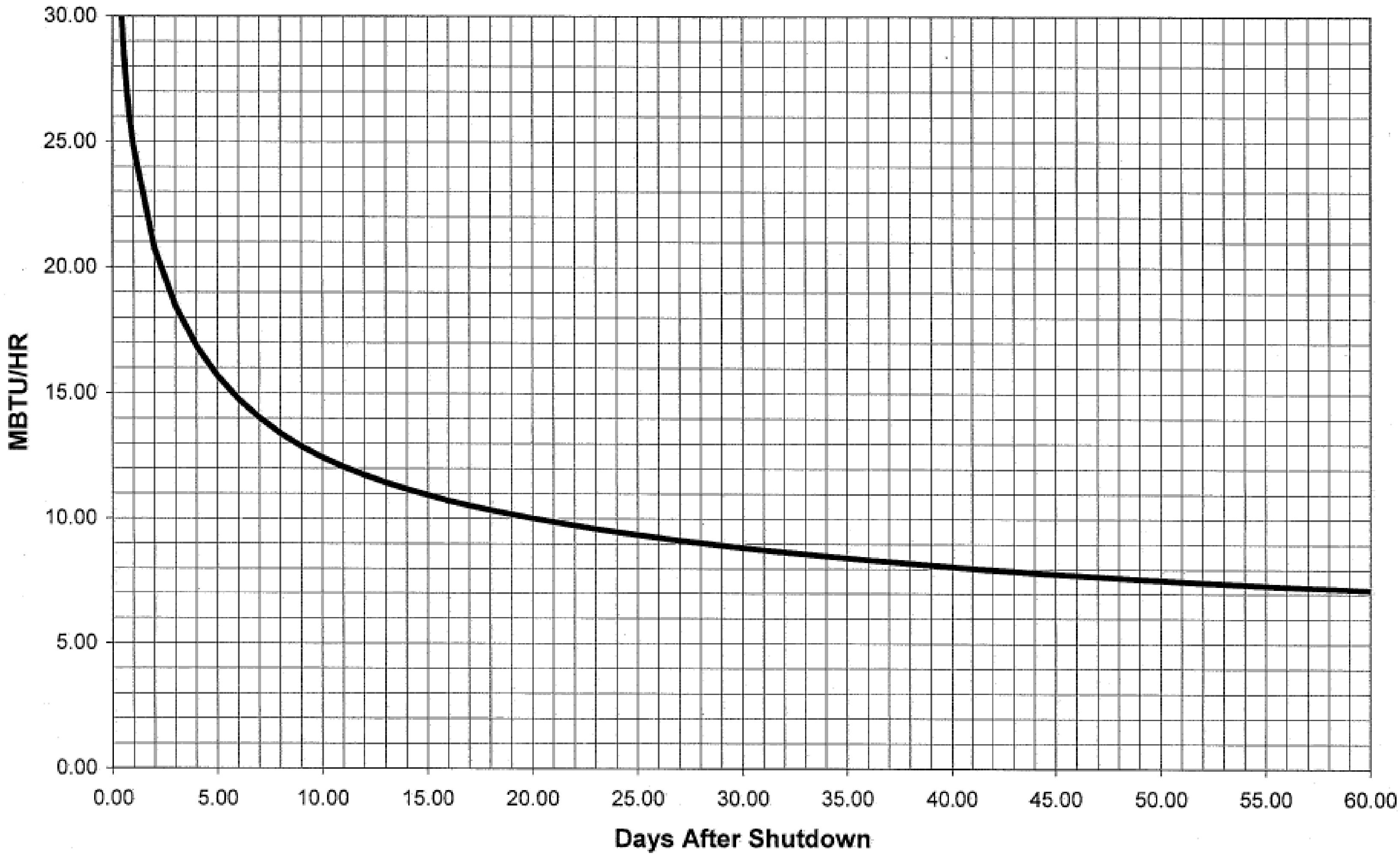
RFO12 In-Vessel Decay Heat After 280 Fuel Bundle Discharge



RO Question 18

PERRY NUCLEAR POWER PLANT	Number: PDB-A0016	
Title: Decay Heat Curve	Use Category: In-Field Reference	
	Revision: 9	Page: 6 of 7

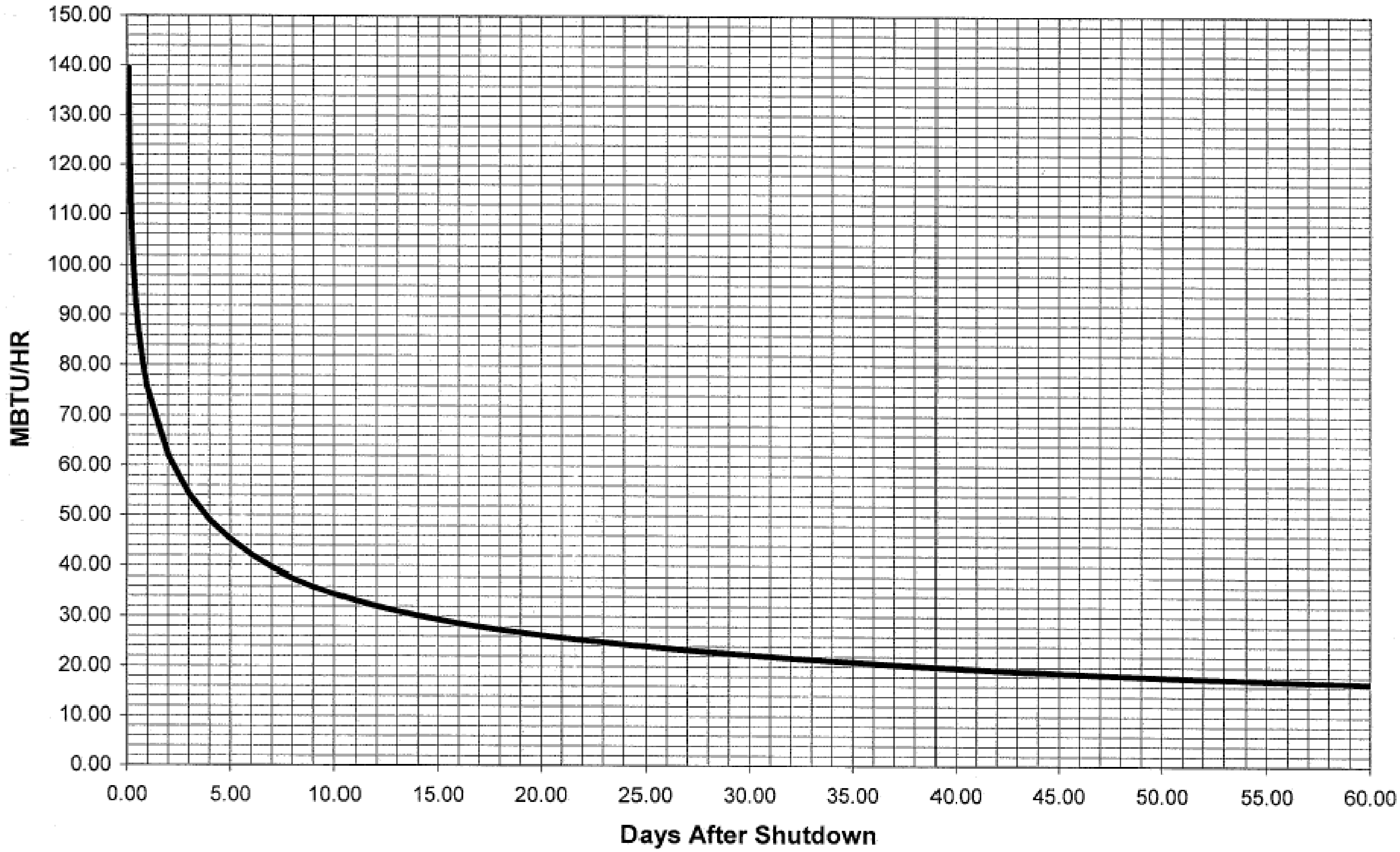
RFO12 Fuel Pool Decay Heat After Adding 280 Fuel Discharged Bundles



RO Question 18

PERRY NUCLEAR POWER PLANT		Number: PDB-A0016
Title: Decay Heat Curve		Use Category: In-Field Reference
	Revision: 9	Page: 7 of 7

RFO12 Fuel Pool Decay Heat After Adding 748 Discharged Fuel Bundles



RO Question 18

PERRY NUCLEAR POWER PLANT	Procedure Number: PDB-A0017	
Title: Pool Heatup Curves	Use Category: In-Field Reference	
	Revision: 10	Page: 1 of 19

POOL HEATUP CURVES

Functional Location (J11)

Plant Data Book

Effective Date: 2-18-09

Preparer: A. Widmer / 1-18-09
Date

1.0 PURPOSE

Provide plant personnel the information with the time for either the reactor or fuel handling building pools to reach a specified temperature. Given the plant configuration, decay heat load, and location of the fuel from the previous operating cycle, the time it would take to reach bulk coolant saturation temperature with **no** Decay Heat Removal (or Spent Fuel Pool Cooling) systems in operation. In the case of the Fuel Handling Building pools, information is provided regarding time to reach 150^{o}F as well as 212^{o}F.

2.0 REFERENCES

2.1 Discretionary

Calculation G41-038 Revision 10

Calculation G41-042 Revision 10

Plant Data Book PDB-A0016

PAP-1925 Shutdown Defense In Depth Assessment and Management

NSAC/175L - EPRI Outage Risk Assessment and Management (ORAM) Program “Safety Assessment of BWR Risk During Shutdown Operations”

2.2 Obligations

None

Commitments addressed in this document:

None

3.0 DETAILS

In the event that Decay Heat Removal system is lost, the following graphs can be utilized to determine the time it takes to get to a specified temperature. The graphs are decay heat vs. time. The decay heat value may be determined from Plant Data Book PDB-A0016 for a given plant configuration and day after shutdown. The decay heat value can be used herein to determine the applicable heat-up time given the initial water temperature is known.

4.0 SCOPE OF REVISION

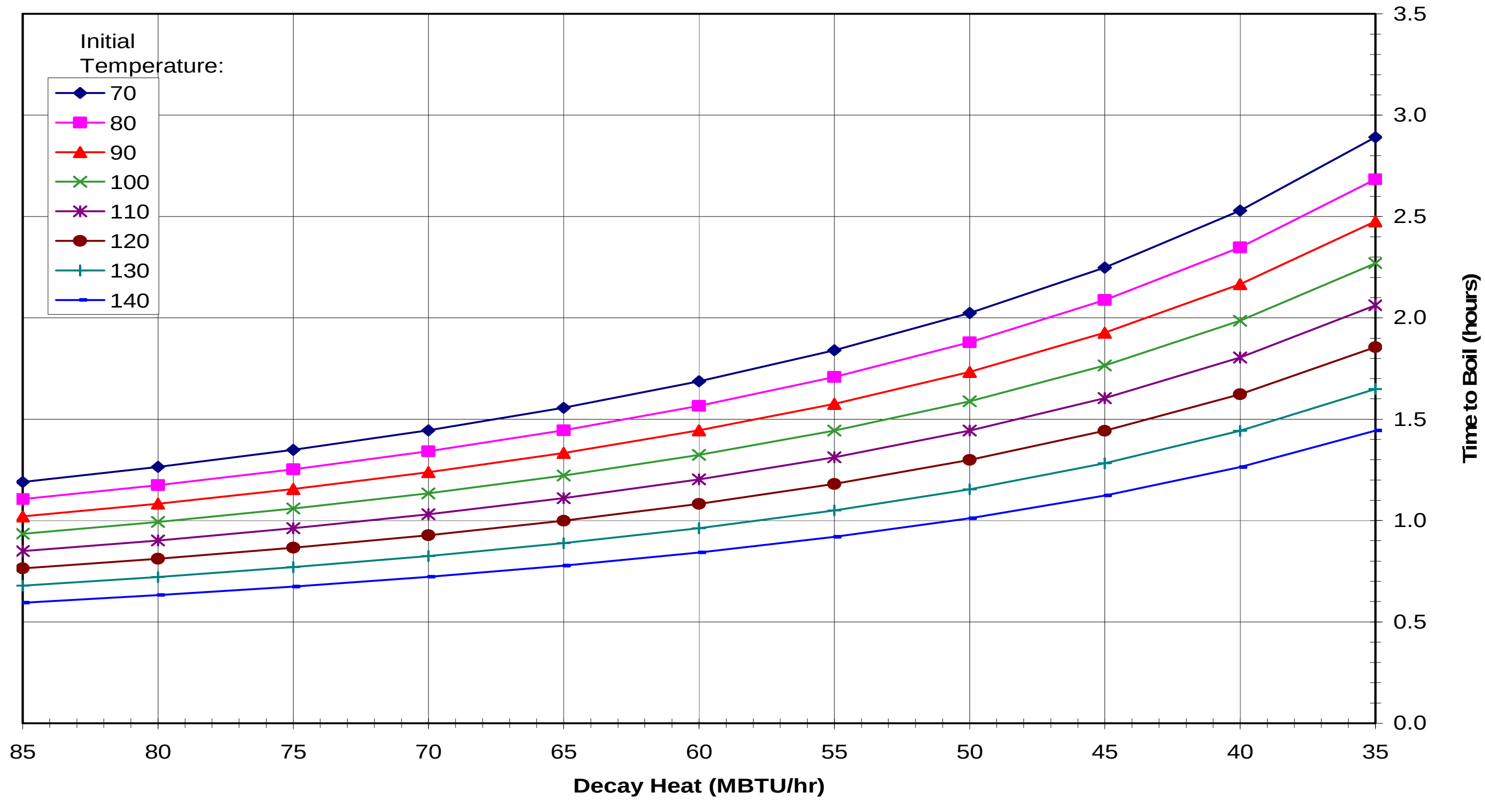
Rev. 10
1. Updated curves to reflect decay heat vs. time.
2. Revised format in accordance with NOP-SS-3007.

RO Question 18

PERRY NUCLEAR POWER PLANT	Number: PDB-A0017	
Title: Pool Heatup Curves	Use Category: In-Field Reference	
	Revision: 10	Page: 4 of 19

TIME-TO-BOIL CURVES FOR REACTOR VESSEL

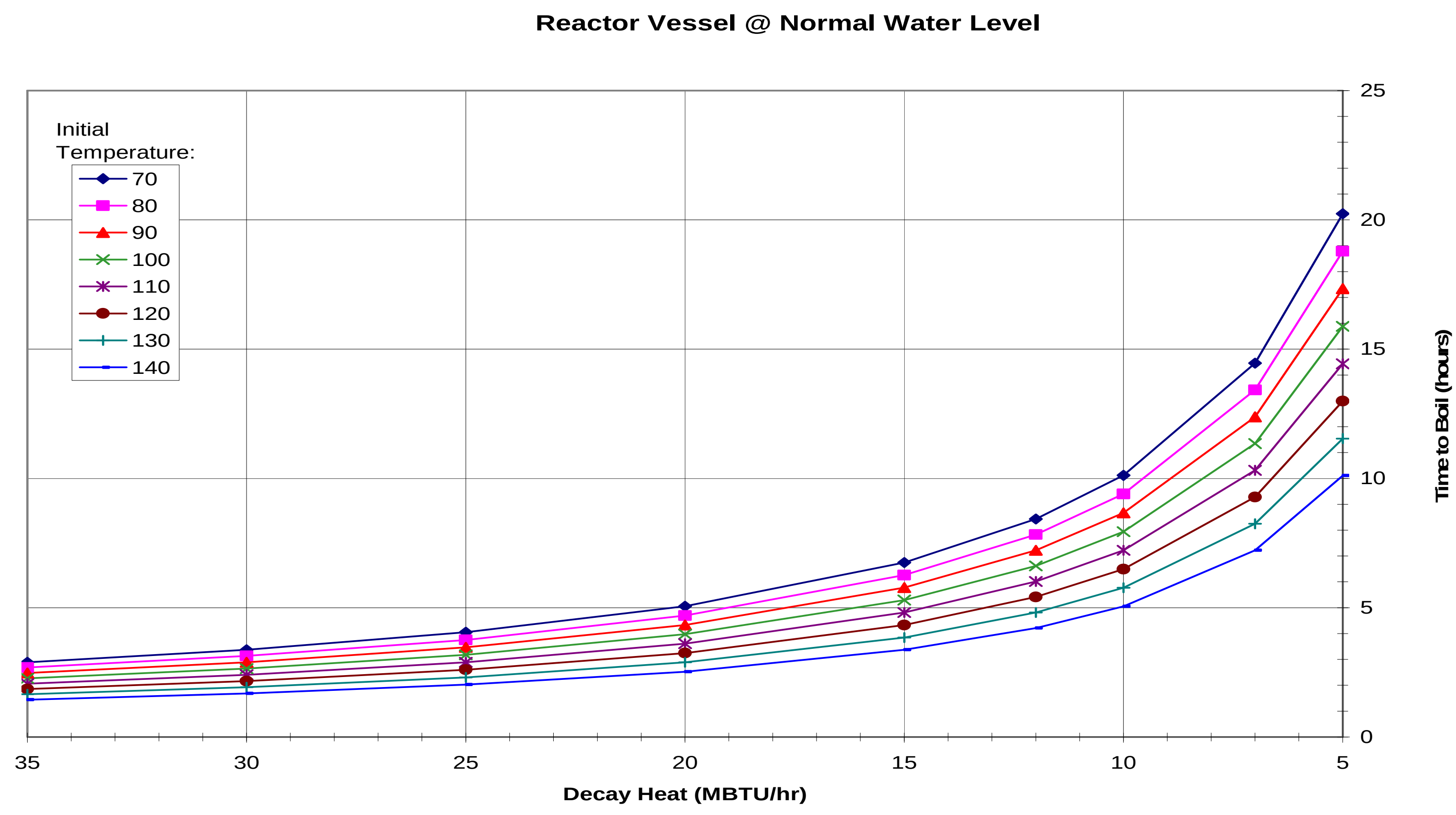
Reactor Vessel @ Normal Water Level



RO Question 18

PERRY NUCLEAR POWER PLANT	Number: PDB-A0017	
Title: Pool Heatup Curves	Use Category: In-Field Reference	
	Revision: 10	Page: 5 of 19

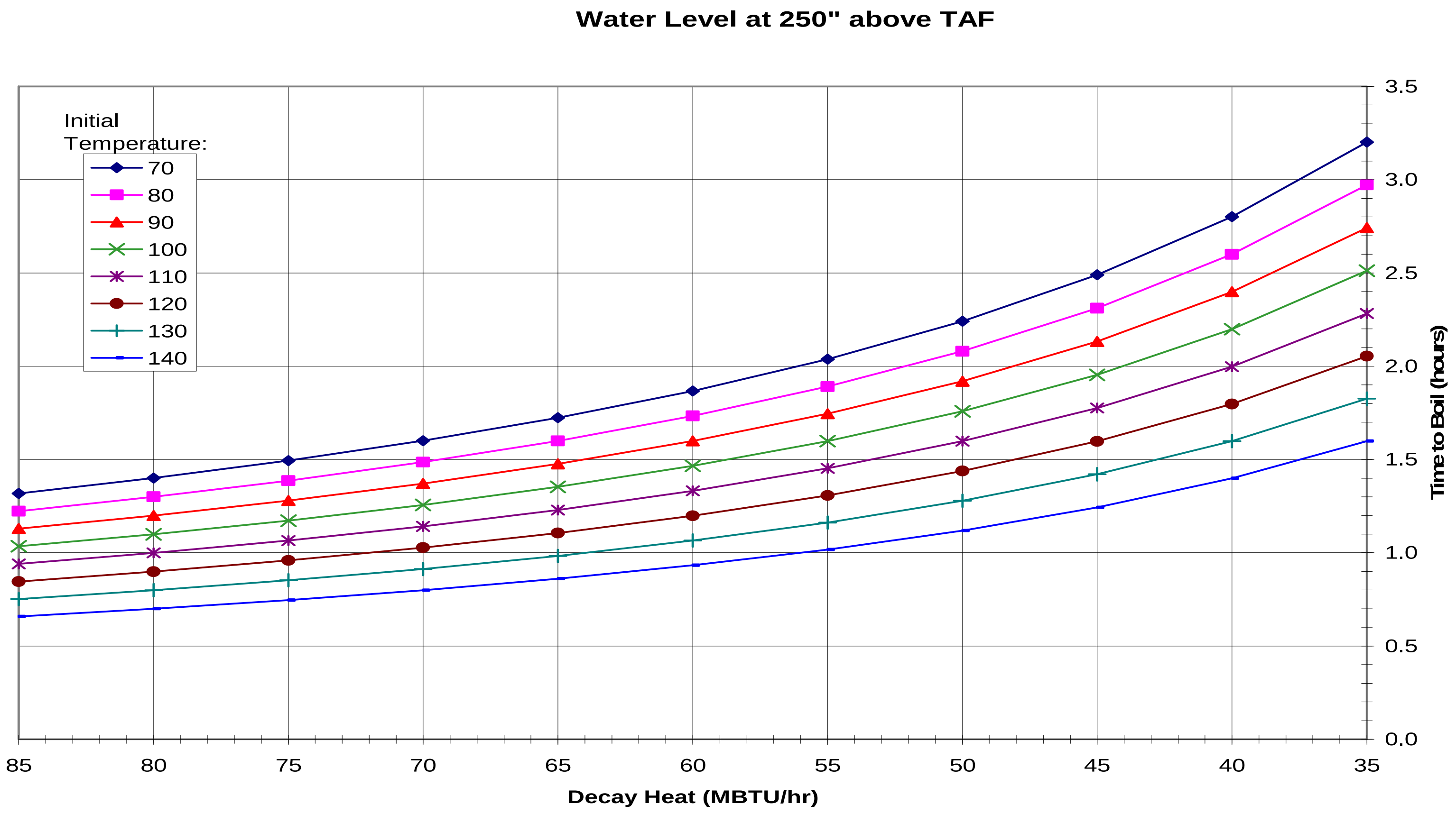
TIME-TO-BOIL CURVES FOR REACTOR VESSEL



RO Question 18

PERRY NUCLEAR POWER PLANT	Number: PDB-A0017	
Title: Pool Heatup Curves	Use Category: In-Field Reference	
	Revision: 10	Page: 6 of 19

TIME-TO-BOIL CURVES FOR REACTOR VESSEL

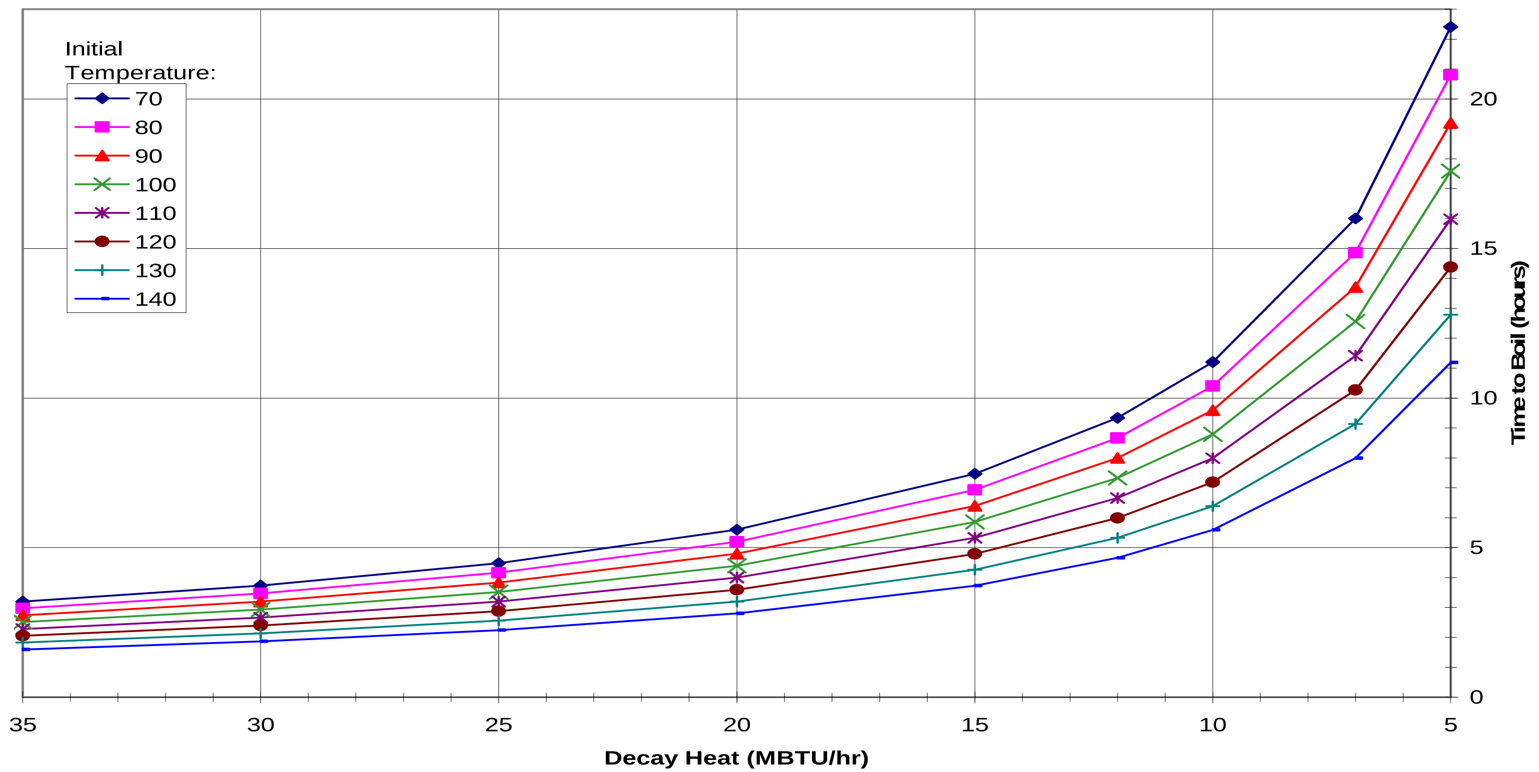


RO Question 18

PERRY NUCLEAR POWER PLANT		Number: PDB-A0017
Title: Pool Heatup Curves	Use Category: In-Field Reference	
	Revision: 10	Page: 7 of 19

TIME-TO-BOIL CURVES FOR REACTOR VESSEL

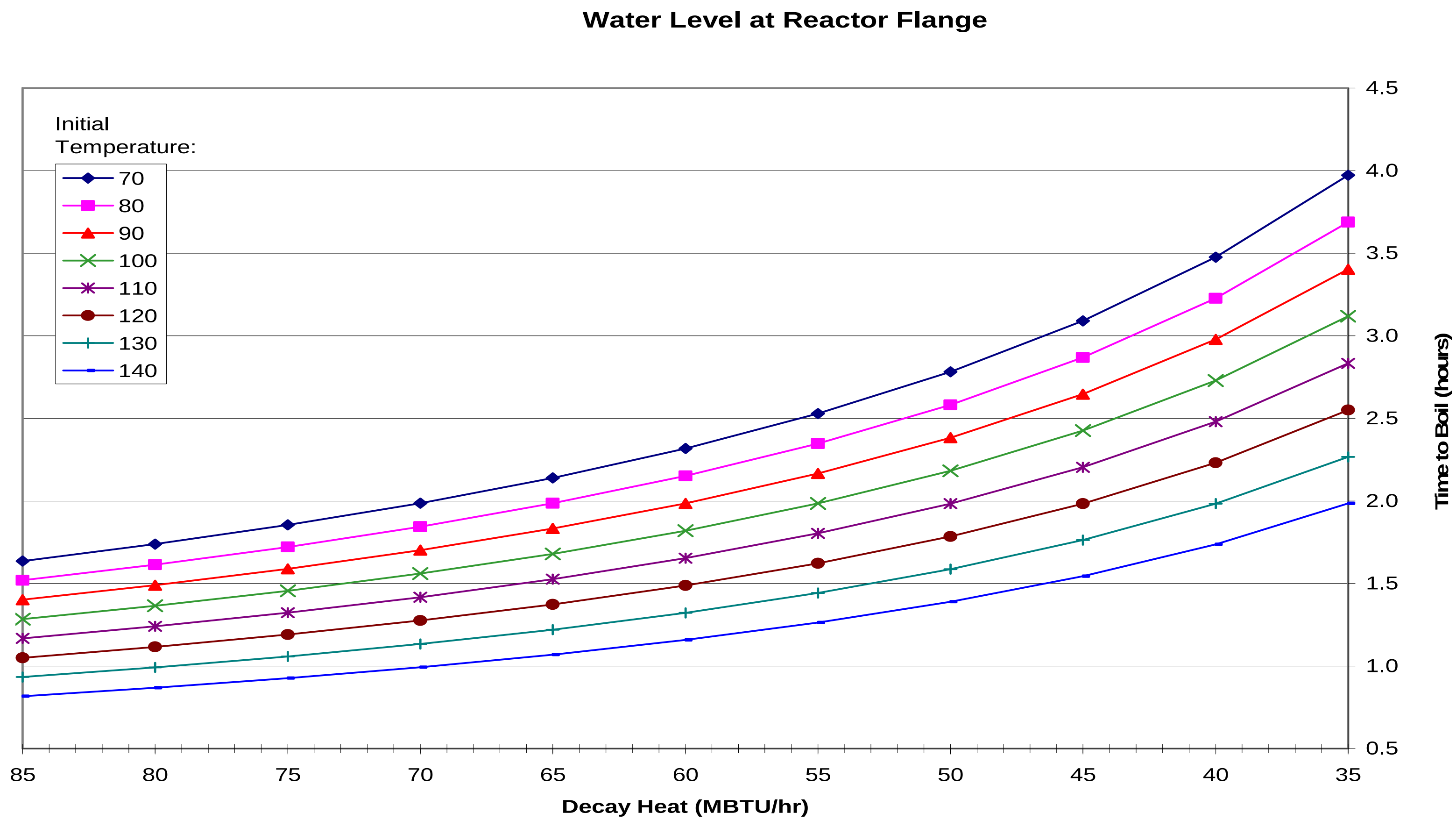
Water Level at 250" above TAF



RO Question 18

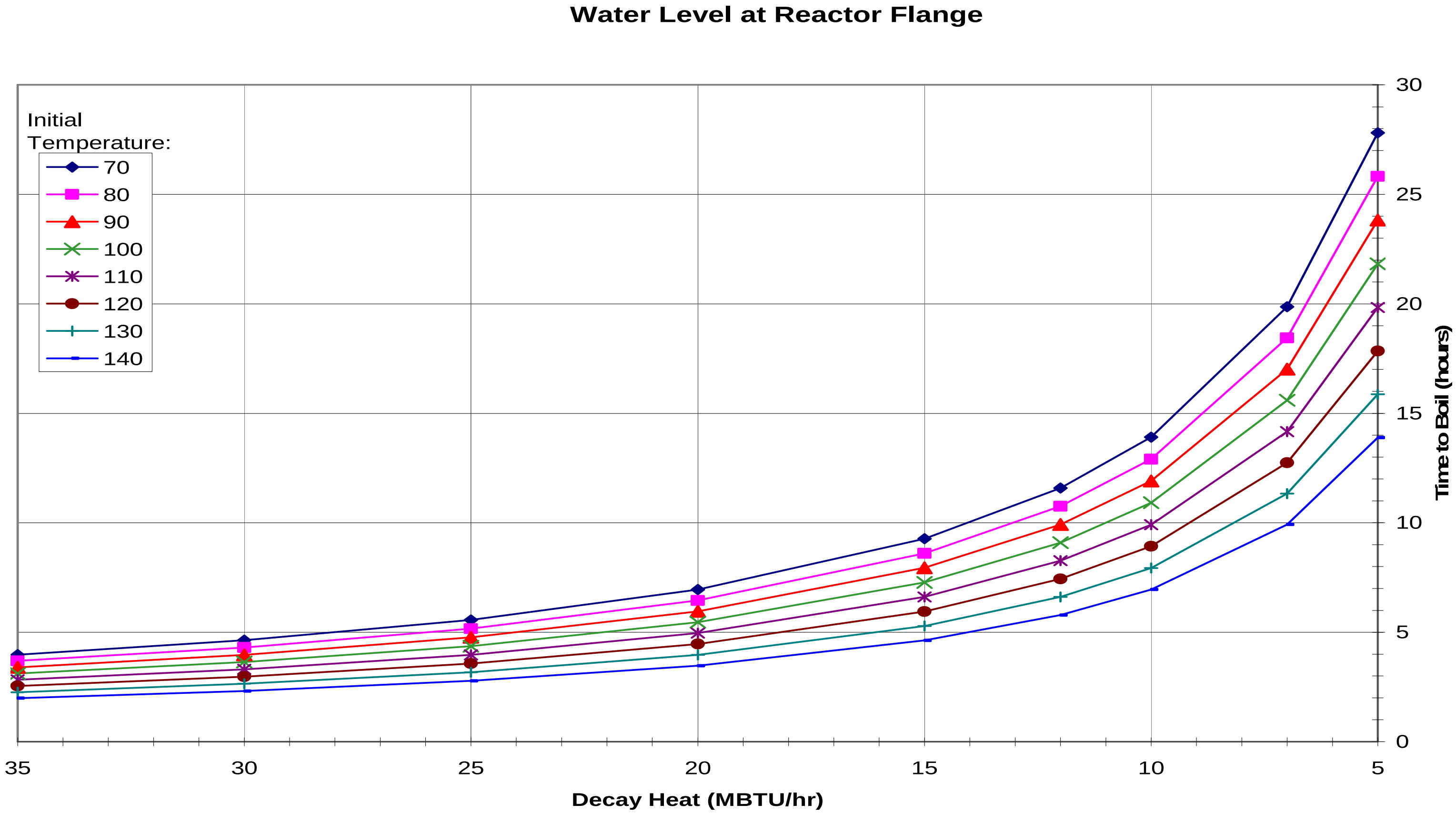
PERRY NUCLEAR POWER PLANT	Number: PDB-A0017	
Title: Pool Heatup Curves	Use Category: In-Field Reference	
	Revision: 10	Page: 8 of 19

TIME-TO-BOIL CURVES FOR REACTOR VESSEL



RO Question 18

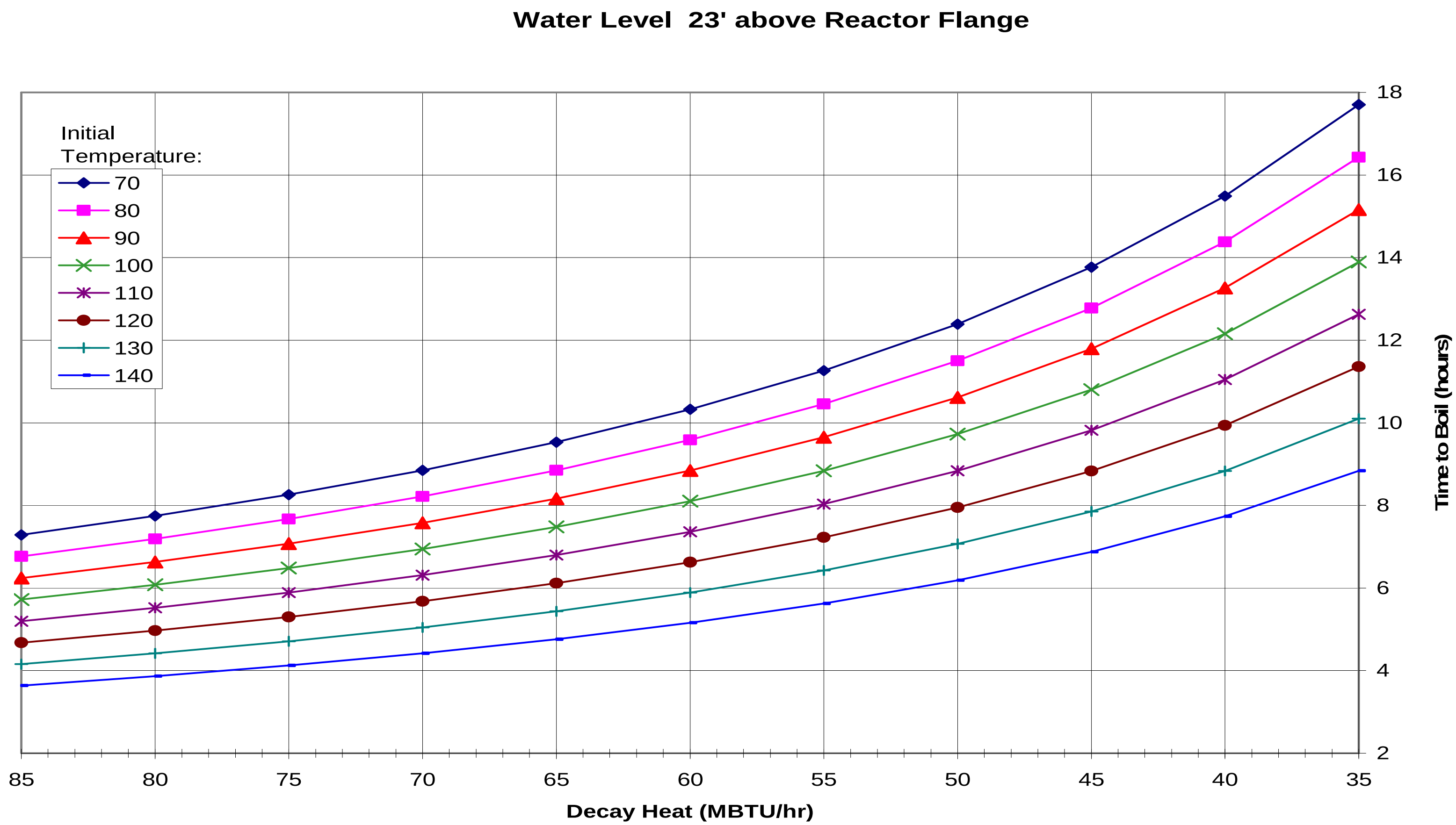
TIME-TO-BOIL CURVES FOR REACTOR VESSEL



RO Question 18

PERRY NUCLEAR POWER PLANT		Number: PDB-A0017
Title: Pool Heatup Curves	Use Category: In-Field Reference	
	Revision: 10	Page: 10 of 19

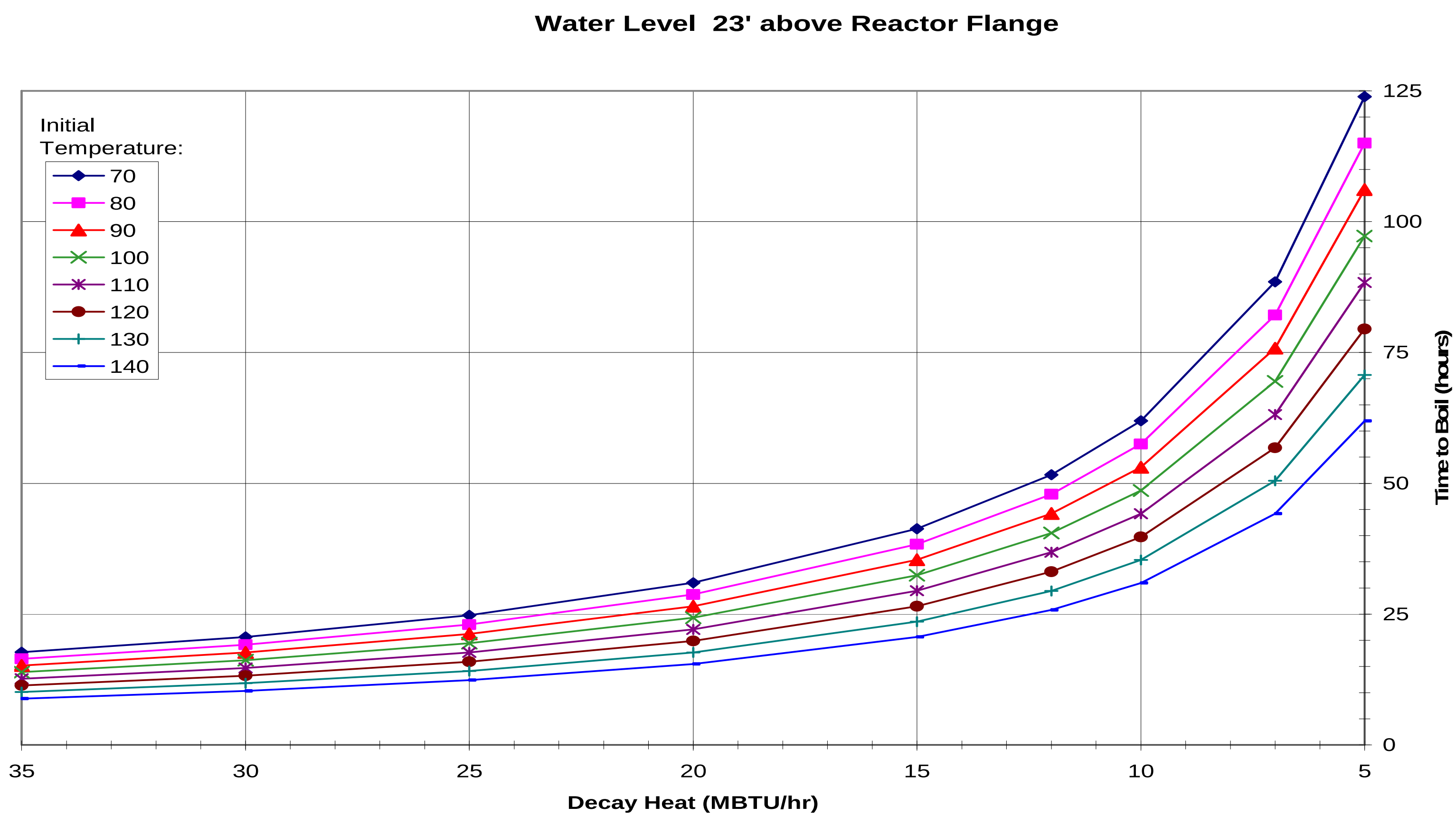
TIME-TO-BOIL CURVES FOR REACTOR VESSEL



RO Question 18

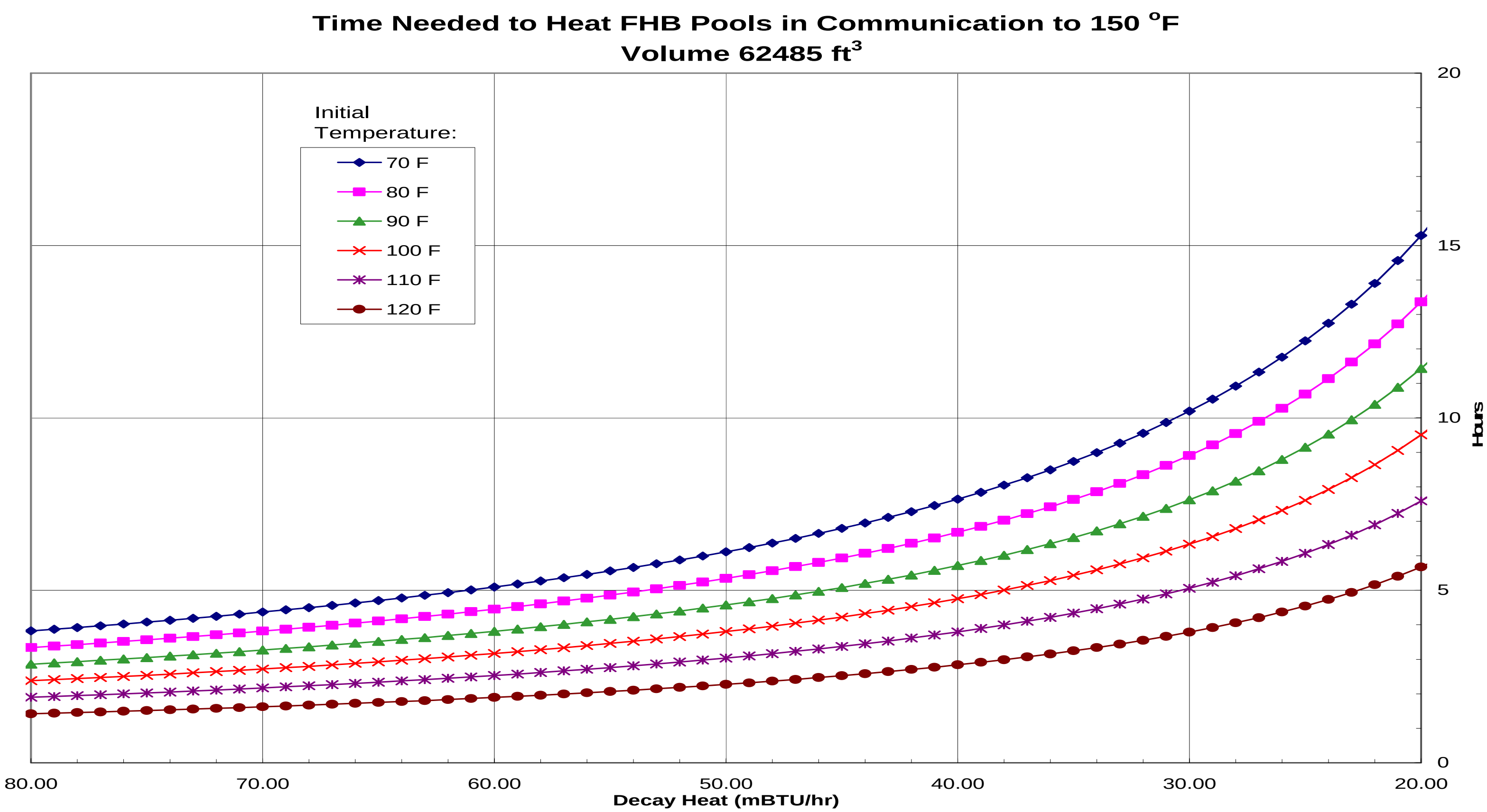
PERRY NUCLEAR POWER PLANT	Number: PDB-A0017	
Title: Pool Heatup Curves	Use Category: In-Field Reference	
	Revision: 10	Page: 11 of 19

TIME-TO-BOIL CURVES FOR REACTOR VESSEL



RO Question 18

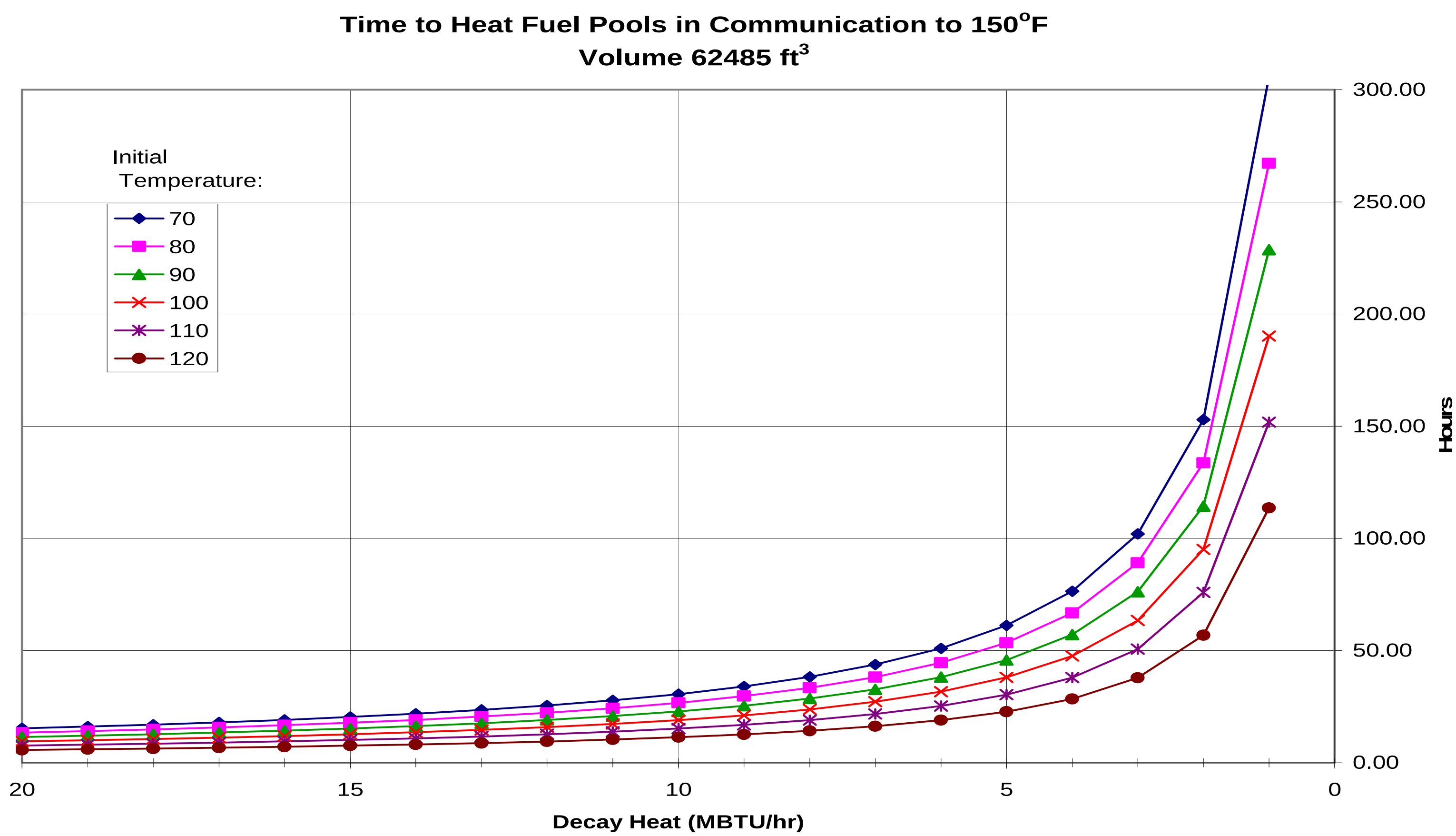
TIME TO HEAT FUEL HANDLING BUILDING POOLS TO 150°F



RO Question 18

PERRY NUCLEAR POWER PLANT	Number: PDB-A0017	
Title: Pool Heatup Curves	Use Category: In-Field Reference	
	Revision: 10	Page: 13 of 19

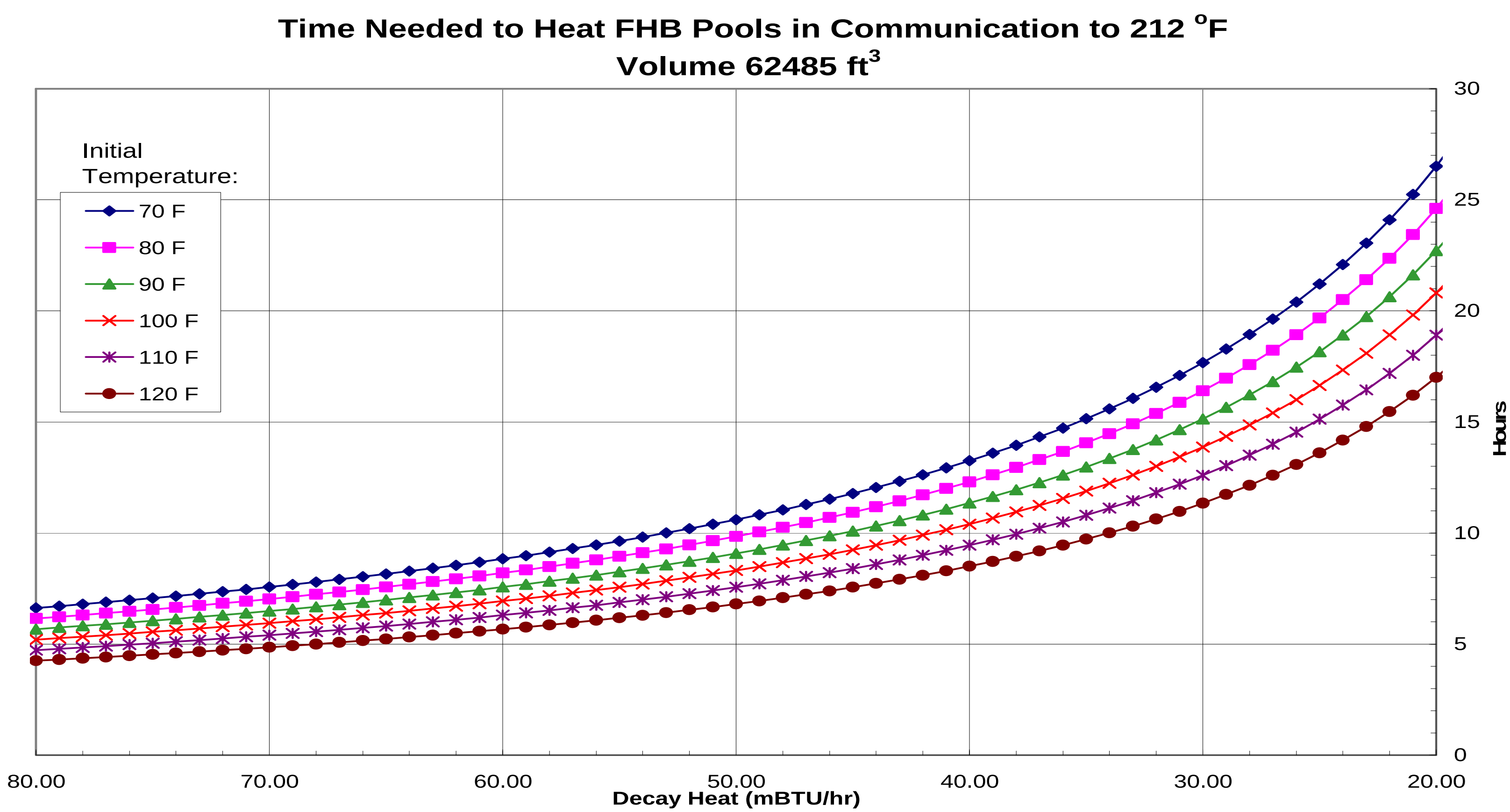
TIME TO HEAT FUEL HANDLING BUILDING POOLS TO 150^{O}F



RO Question 18

PERRY NUCLEAR POWER PLANT	Number: PDB-A0017	
Title: Pool Heatup Curves	Use Category: In-Field Reference	
	Revision: 10	Page: 14 of 19

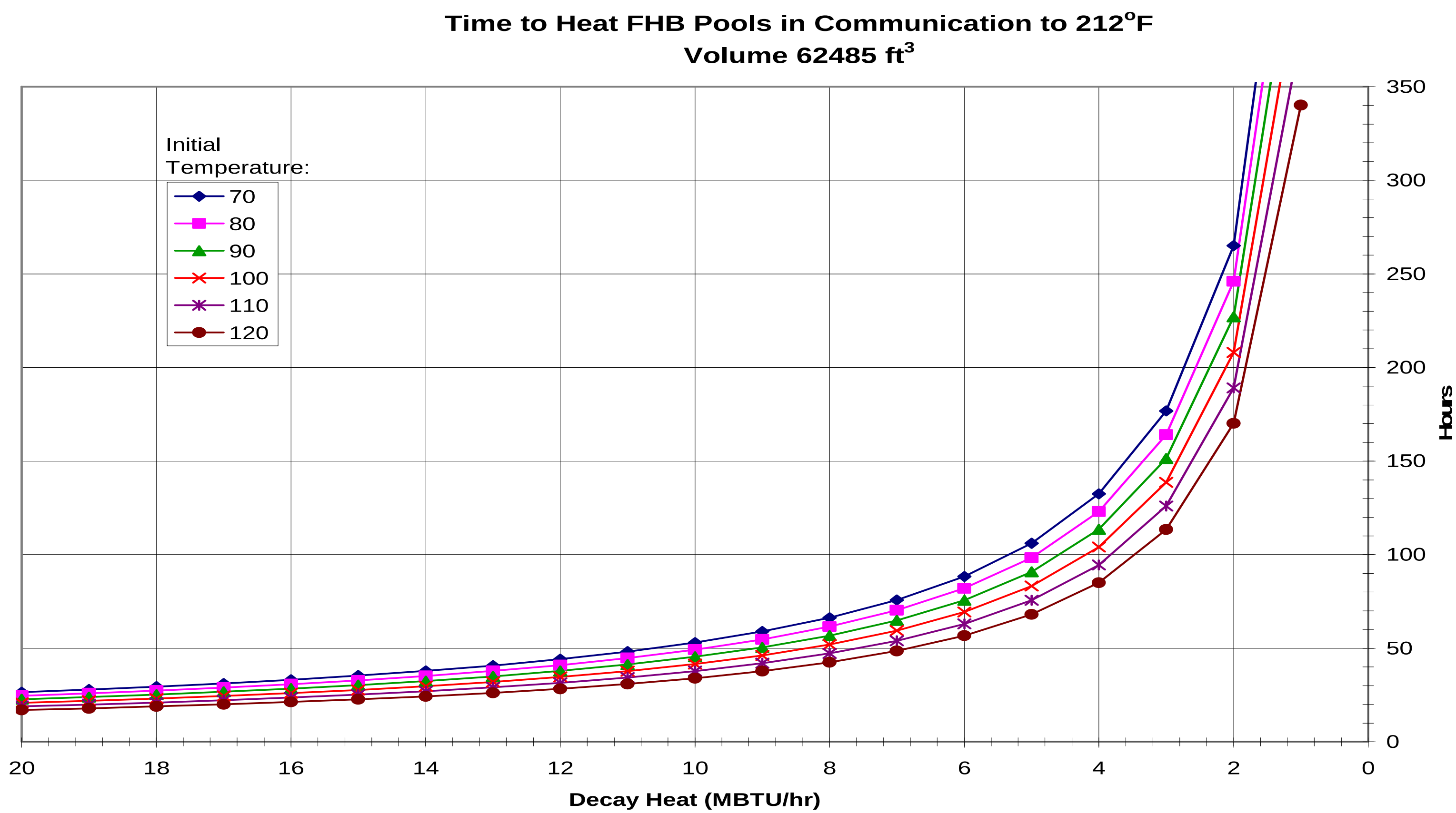
TIME TO HEAT FUEL HANDLING BUILDING POOLS TO 212^{O}F



RO Question 18

PERRY NUCLEAR POWER PLANT	Number: PDB-A0017	
Title: Pool Heatup Curves	Use Category: In-Field Reference	
	Revision: 10	Page: 15 of 19

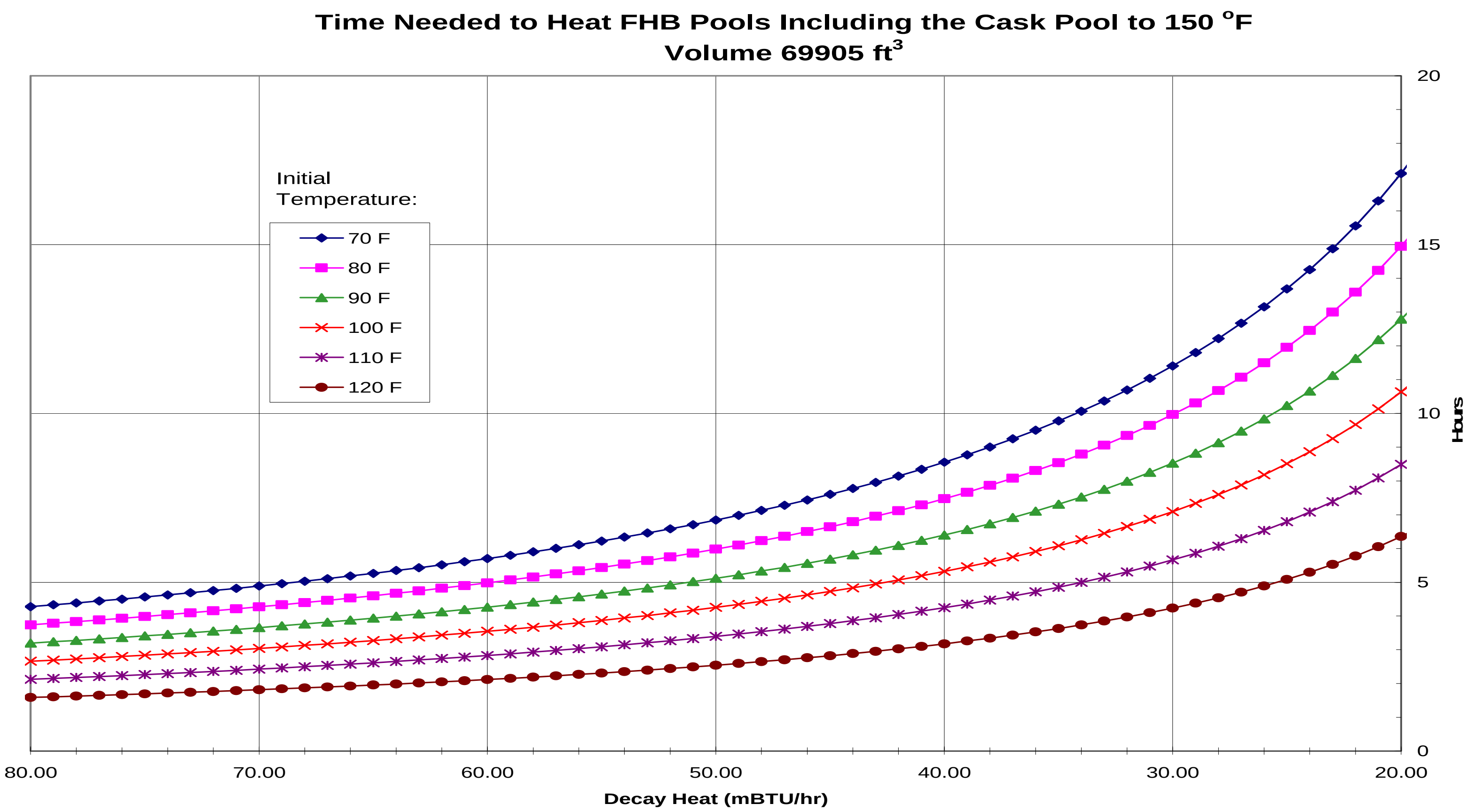
TIME TO HEAT FUEL HANDLING BUILDING POOLS TO 212^{O}F



RO Question 18

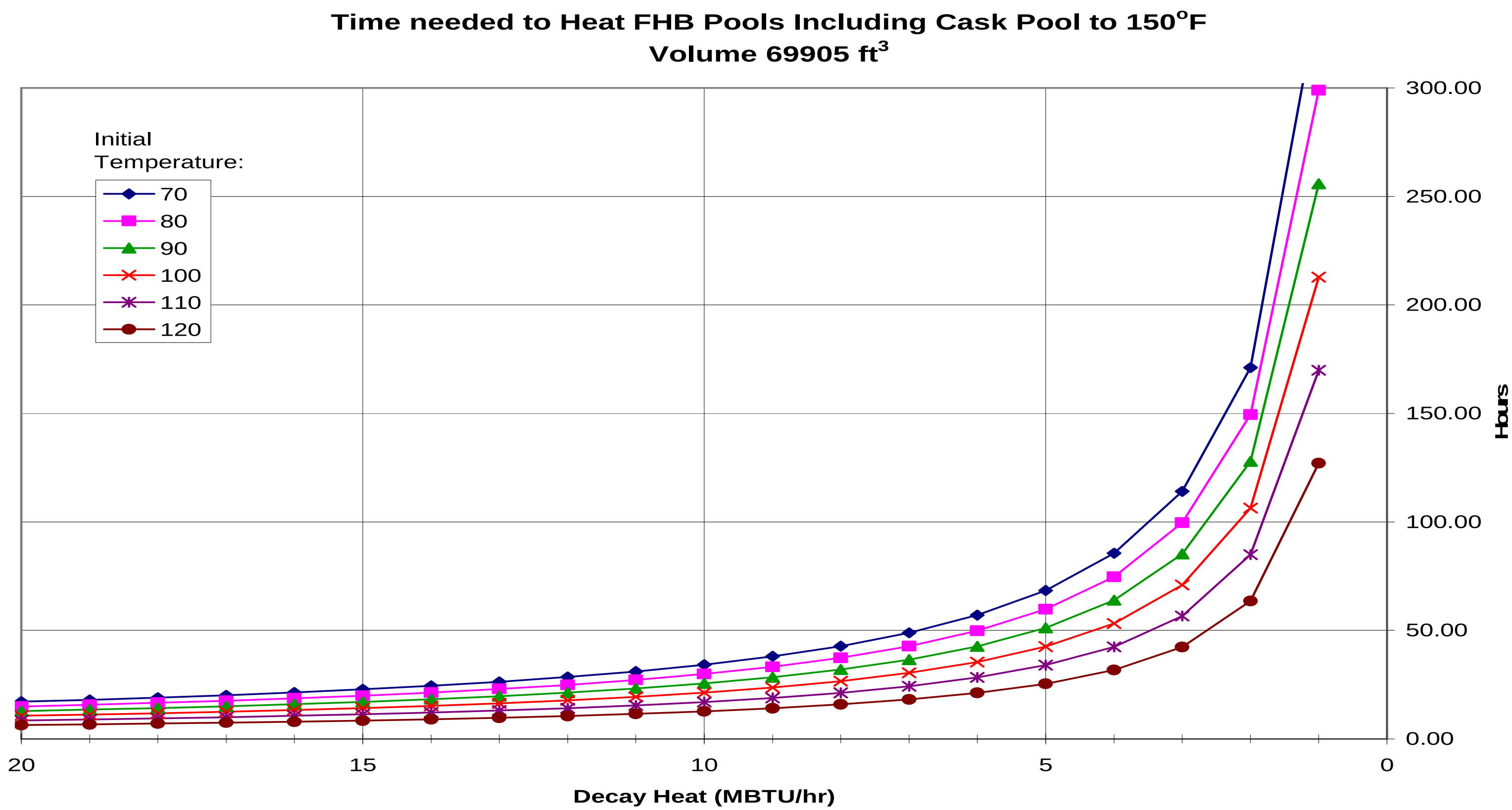
PERRY NUCLEAR POWER PLANT	Number: PDB-A0017	
Title: Pool Heatup Curves	Use Category: In-Field Reference	
	Revision: 10	Page: 16 of 19

TIME TO HEAT FUEL HANDLING BUILDING POOLS (INCLUDING CASK POOL) TO 150^{O}F



RO Question 18

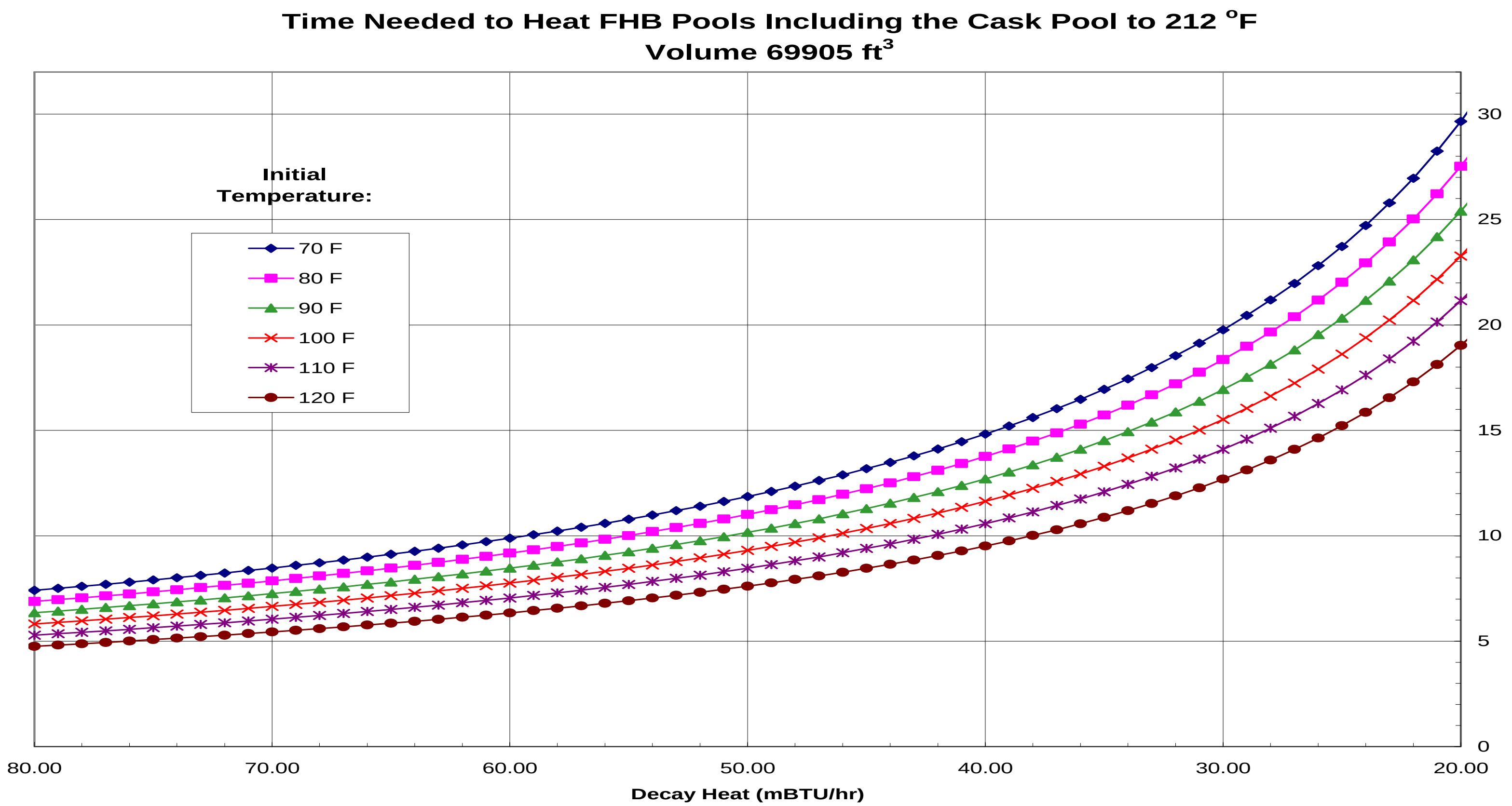
TIME TO HEAT FUEL HANDLING BUILDING POOLS (INCLUDING CASK POOL) TO 150^{O}F



RO Question 18

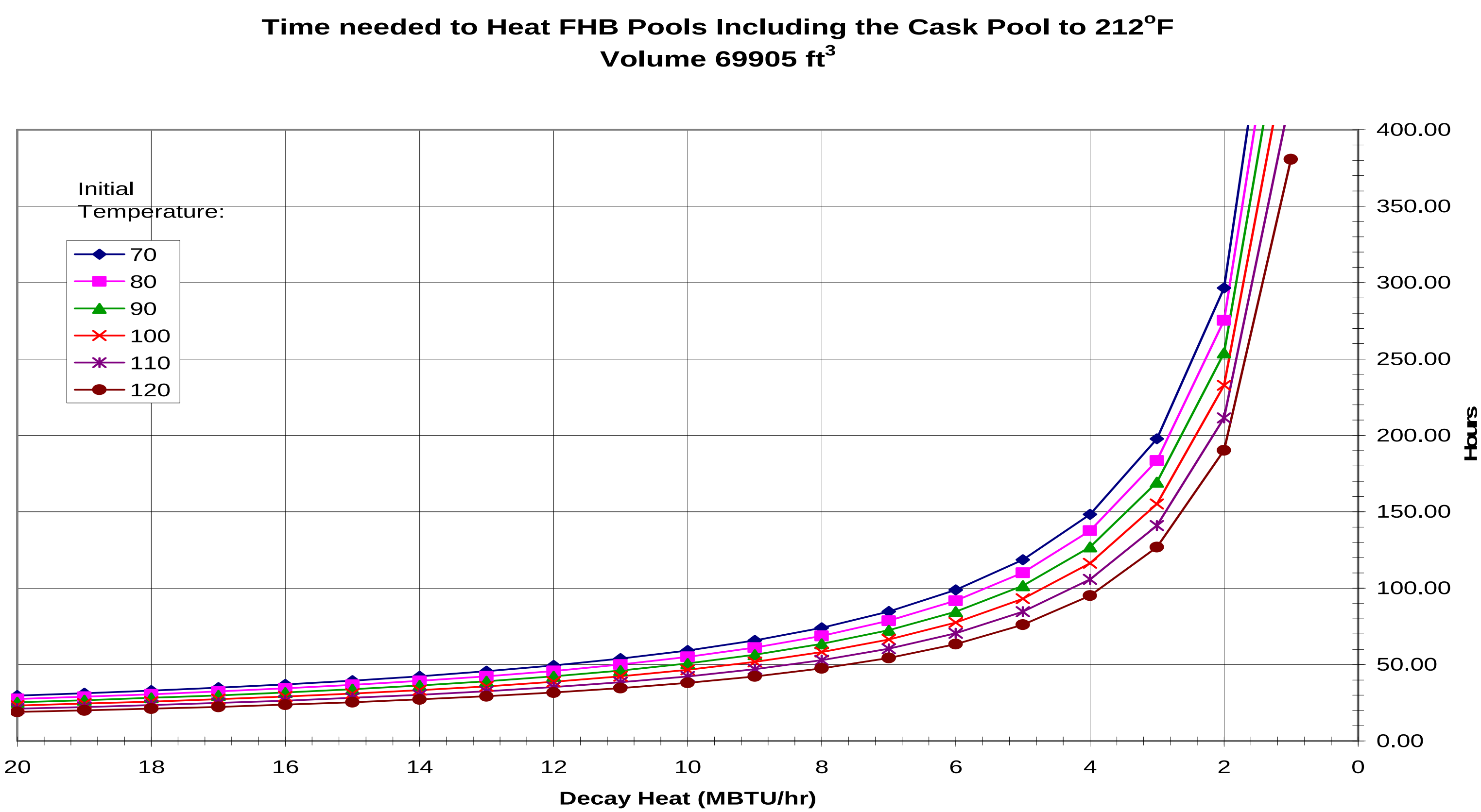
PERRY NUCLEAR POWER PLANT	Number: PDB-A0017	
Title: Pool Heatup Curves	Use Category: In-Field Reference	
	Revision: 10	Page: 18 of 19

TIME TO HEAT FUEL HANDLING BUILDING POOLS (INCLUDING CASK POOL) TO 212°F



RO Question 18

TIME TO HEAT FUEL HANDLING BUILDING POOLS (INCLUDING CASK POOL) TO 212^{O}F



RO Question 48

PERRY NUCLEAR POWER PLANT		Procedure Number: PDB-A0012
Title: Recirc Drive Flow vs. Total Core Flow		Use Category: General Skill Reference
	Revision: 15	Page: 1 of 6

RECIRC DRIVE FLOW VS. TOTAL CORE FLOW

Functional Location (B33)

Plant Data Book

Effective Date: 9-20-07

Preparer: P. Curran / 8-24-07
Date

RO Question 48

1.0 REFERENCES

1.1 Discretionary

None

1.2 Obligations

SVI-F41-T3008, Reactor Recirculation System Flow Data Verification performed May 2005 and June/July 2007.

SVI-B33-T1160, Jet Pump Operability

Technical Specification SR 3.4.3.1.b

ICS/SDS Displays RFPERF and RSPERF.

Commitments addressed in this document:

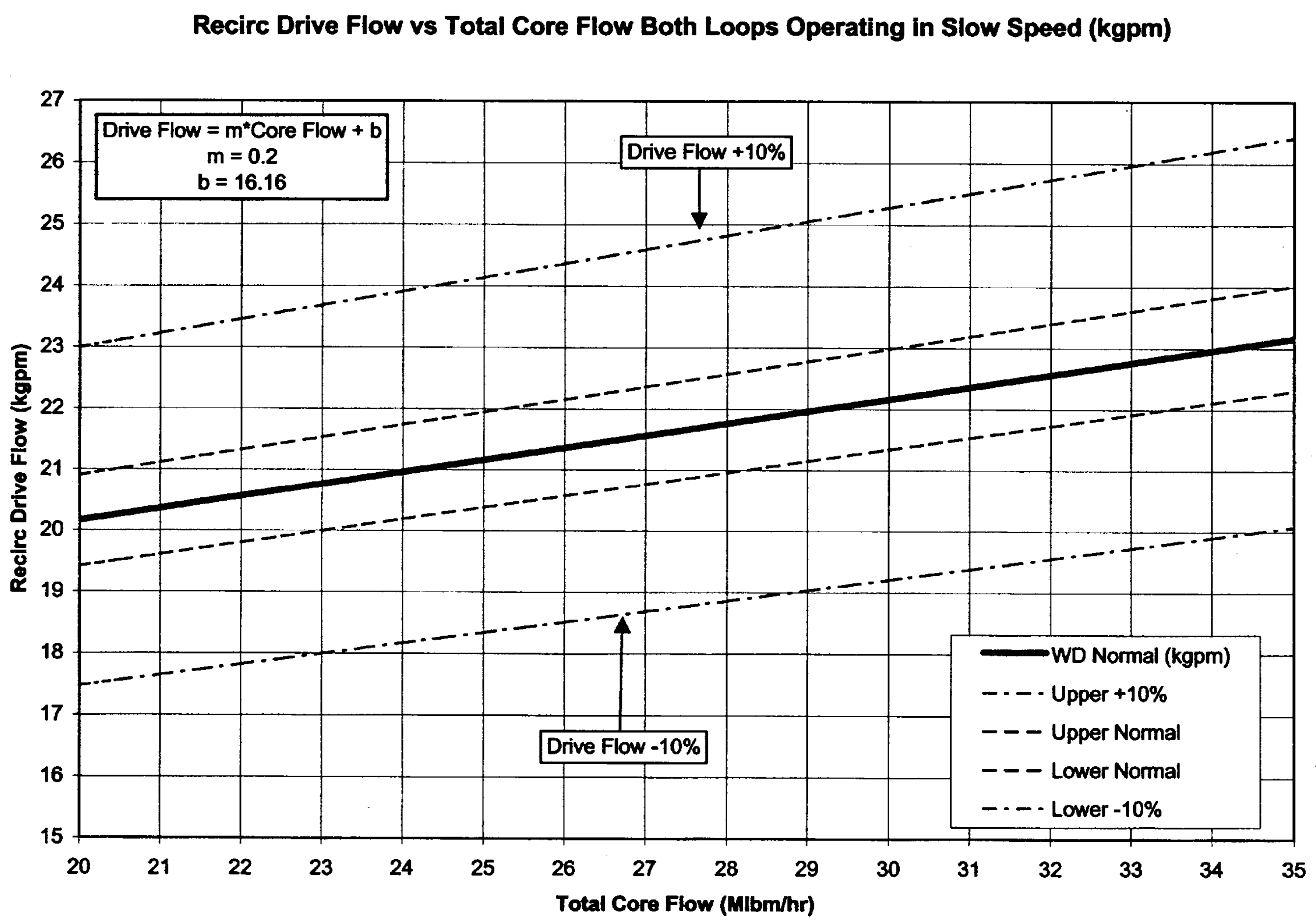
None

2.0 SCOPE OF REVISION

Rev. 15
1. Revised format in accordance with NOP-SS-3007.
2. Updated references to include the June/July 2007 performance of SVI-F41-T3008. There were no changes to the graphs.

RO Question 48

A0012 Slow Speed (Kgpm)

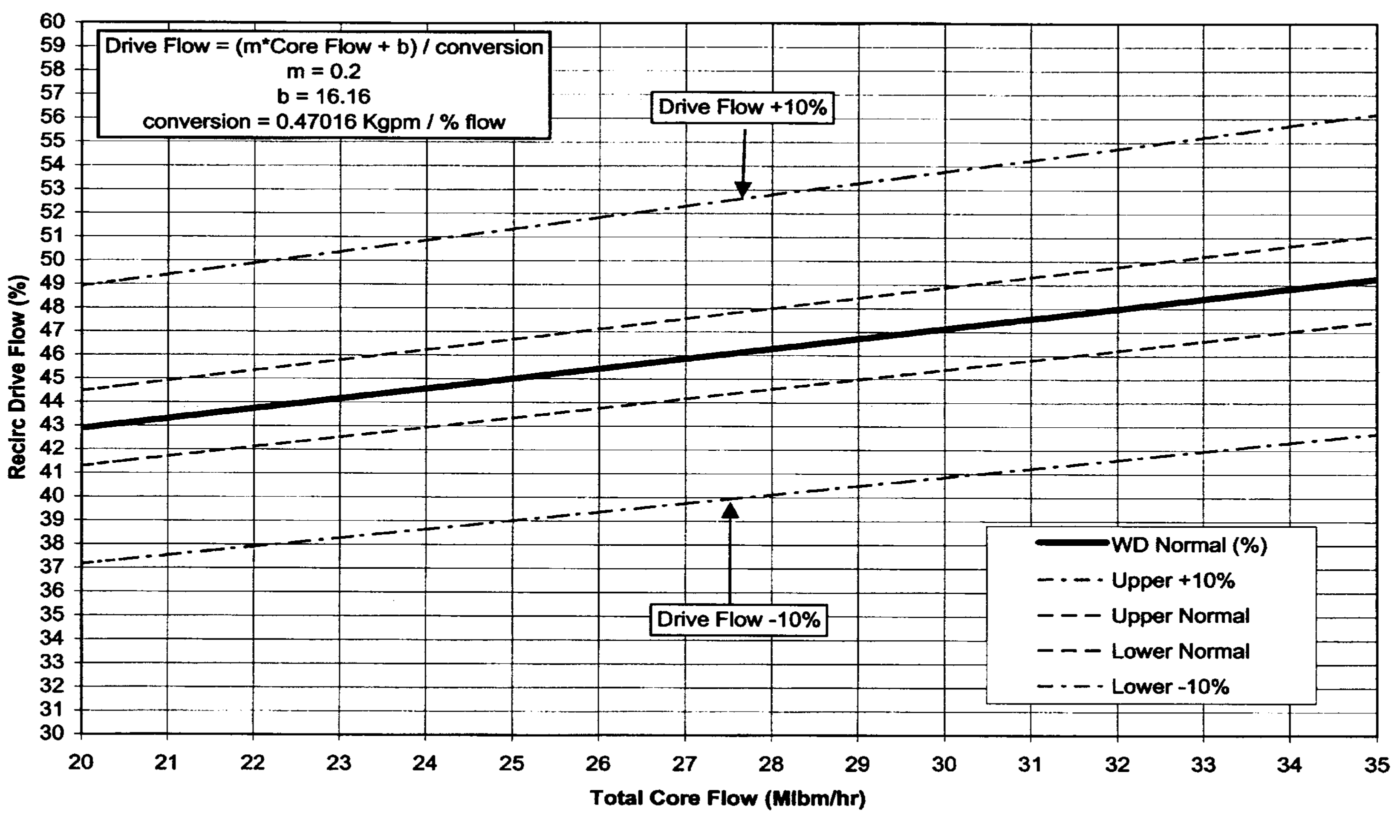


RO Question 48

PERRY NUCLEAR POWER PLANT	Procedure Number: PDB-A0012	
Title: Recirc Drive Flow vs. Total Core Flow	Use Category: General Skill Reference	
	Revision: 15	Page: 4 of 6

A0012 Slow Speed (%)

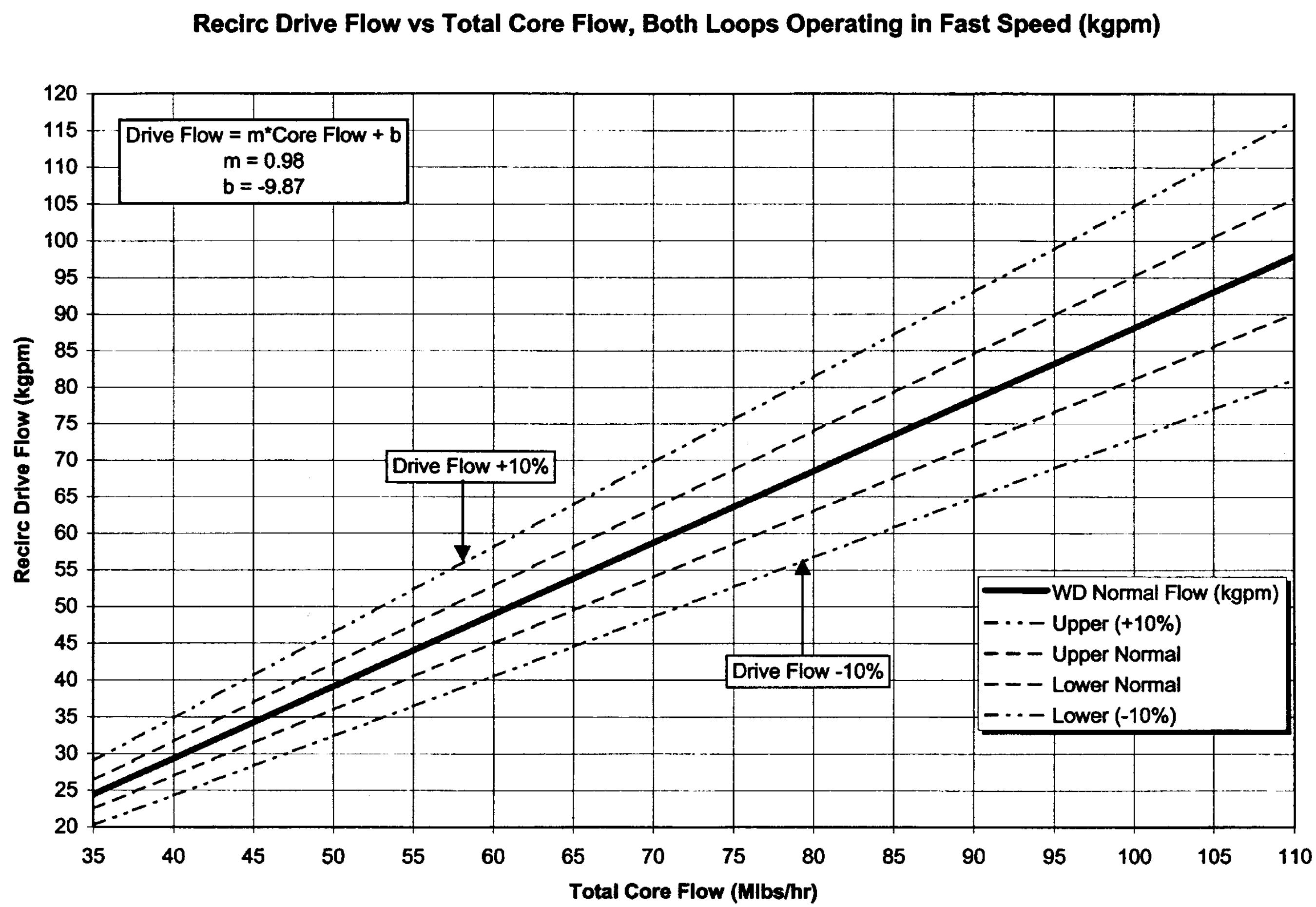
Recirc Drive Flow vs Total Core Flow Both Loops Operating in Slow Speed (%)



RO Question 48

PERRY NUCLEAR POWER PLANT	Procedure Number: PDB-A0012	
Title: Recirc Drive Flow vs. Total Core Flow	Use Category: General Skill Reference	
	Revision: 15	Page: 5 of 6

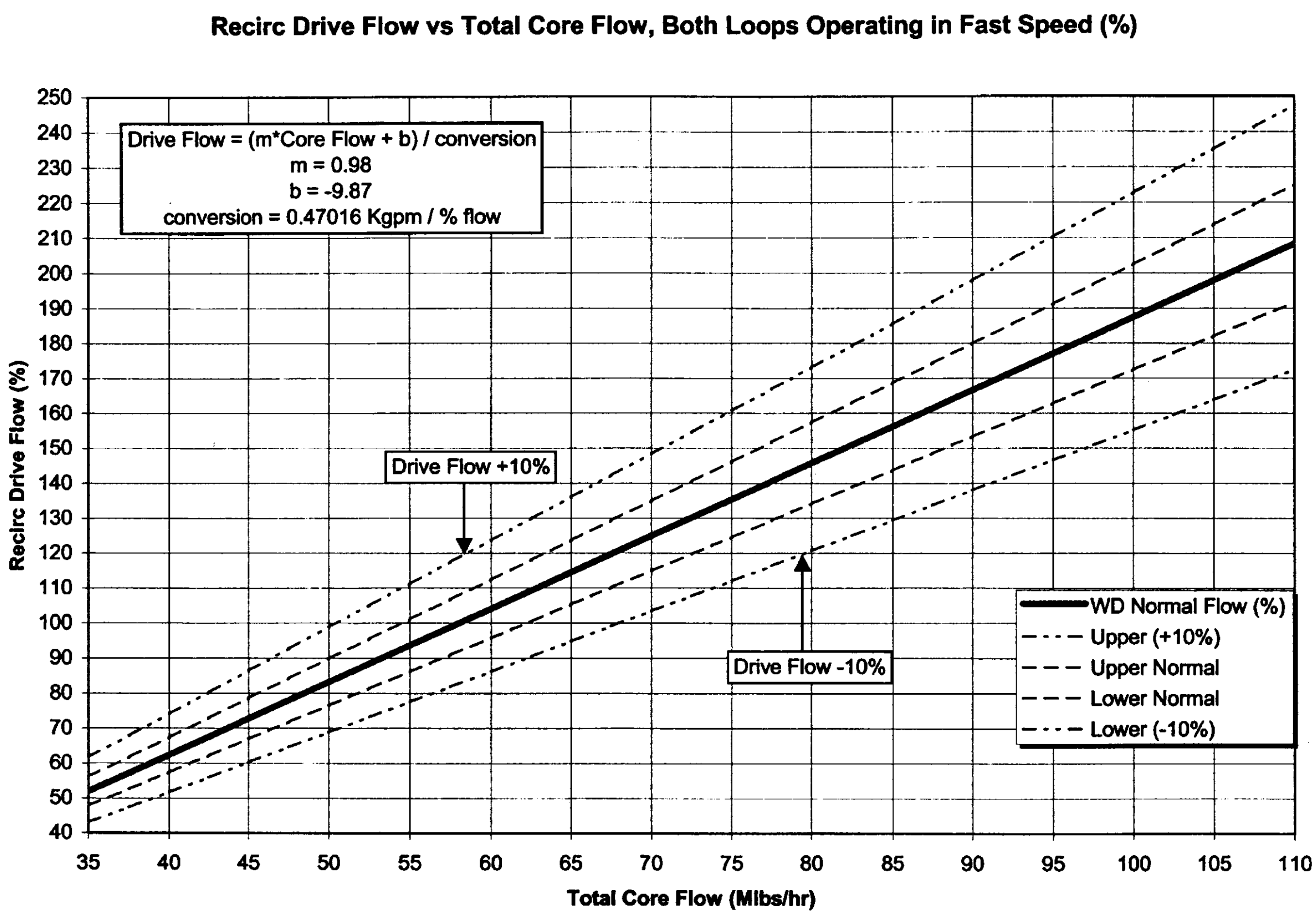
A0012 Fast Speed (Kgpm)



RO Question 48

PERRY NUCLEAR POWER PLANT	Procedure Number: PDB-A0012	
Title: Recirc Drive Flow vs. Total Core Flow	Use Category: General Skill Reference	
	Revision: 15	Page: 6 of 6

A0012 Fast Speed (%)



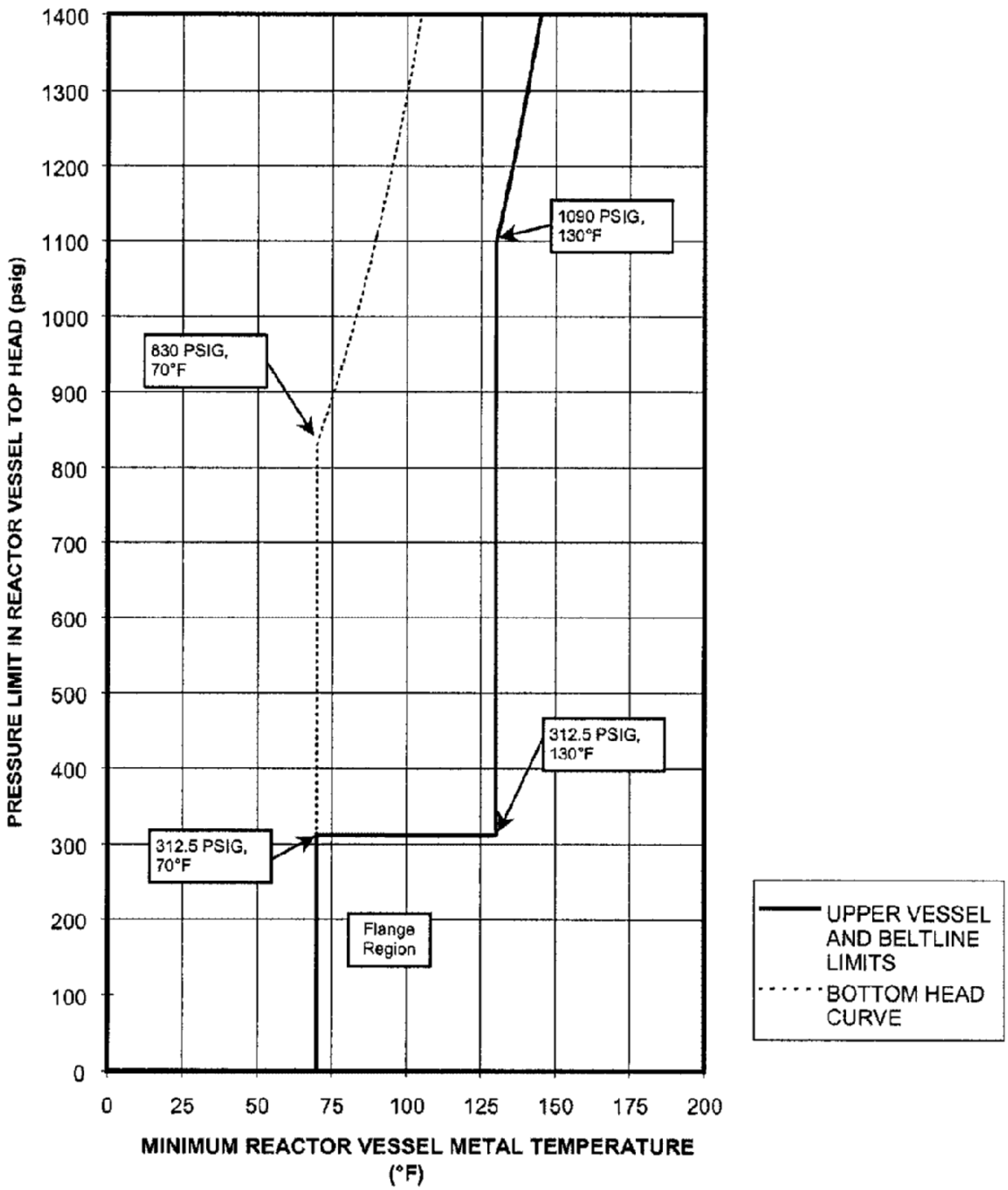
RO Question 51

H13-P622

NS4 MSL DRN ISOL INBD LOGIC TEST B21H-DS1B	NS4 BOP ISOL INBD LOGIC TEST B21H-DS3B	RX WTR SMPL VLV B33-F019 LOGIC TEST B21H-DS7B	RHR ISOLATION INBD LOGIC TEST B21H-DS5B
Light on	**Light on**	**Light off**	**Light off**

RO Question 75

P/T Limits - Non-Nuclear Heatup/Cooldown (Curve B) Valid up to 22 EFPY
Technical Specification Figure 3.4.11-1(b)



SRO Question 3

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.1 ECCS–Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of eight safety/relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ 150 psig.

ACTIONS

-----NOTE-----
LCO 3.0.4.b is not applicable to HPCS.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low pressure ECCS injection/spray subsystem inoperable.	A.1 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	7 days
B. High Pressure Core Spray (HPCS) System inoperable.	B.1 Verify by administrative means RCIC System is OPERABLE when RCIC is required to be OPERABLE.	1 hour
	<u>AND</u>	
	B.2 Restore HPCS System to OPERABLE status.	14 days

(continued)

SRO Question 3

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two ECCS injection subsystems inoperable. OR One ECCS injection and one ECCS spray subsystem inoperable.	C.1 Restore one ECCS injection/spray subsystem to OPERABLE status.	72 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	12 hours 36 hours
E. One ADS valve inoperable.	E.1 Restore ADS valve to OPERABLE status.	14 days
F. One ADS valve inoperable. AND One low pressure ECCS injection/spray subsystem inoperable.	F.1 Restore ADS valve to OPERABLE status. OR F.2 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	72 hours 72 hours
G. Two or more ADS valves inoperable. OR	G.1 Be in MODE 3. AND	12 hours (continued)

SRO Question 3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. (continued) Required Action and associated Completion Time of Condition E or F not met.	G.2 Reduce reactor steam dome pressure to $\leq$ 150 psig.	36 hours
H. HPCS and Low Pressure Core Spray (LPCS) Systems inoperable. OR Three or more ECCS injection/spray subsystems inoperable. OR HPCS System and one or more ADS valves inoperable. OR Two or more ECCS injection/spray subsystems and one or more ADS valves inoperable.	H.1 Enter LCO 3.0.3.	Immediately

SRO Question 4

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources—Operating

LCO 3.8.4 The Division 1, Division 2, and Division 3 DC electrical power subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Division 1 or 2 DC electrical power subsystem inoperable.	A.1 Restore Division 1 and 2 DC electrical power subsystems to OPERABLE status.	2 hours
B. Division 3 DC electrical power subsystem inoperable.	B.1 Declare High Pressure Core Spray System inoperable.	Immediately
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	12 hours
	AND	
	C.2 Be in MODE 4.	36 hours

SRO Question 4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage is ≥ 129 V on float charge.	7 days
SR 3.8.4.2	Verify no visible corrosion at battery terminals and connectors. OR Verify battery connection resistance is ≤ 5.0 E-5 ohm for inter-cell connections, ≤ 5.0 E-5 ohm for inter-rack connections, ≤ 5.0 E-5 ohm for inter-tier connections, ≤ 5.0 E-5 ohm for terminal connections; for Div 1 and Div 2 and ≤ 1.0 E-4 ohm for inter-cell connections, ≤ 1.0 E-4 ohm for inter-rack connections, ≤ 1.0 E-4 ohm for inter-tier connections, ≤ 1.0 E-4 ohm for terminal connections, for Div 3.	92 days
SR 3.8.4.3	Verify battery cells, cell plates, and racks show no visual indication of physical damage or abnormal deterioration that could degrade battery performance.	24 months
SR 3.8.4.4	Remove visible corrosion, and verify battery cell to cell and terminal connections are coated with anti-corrosion material.	24 months

(continued)

SRO Question 4

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.5	Verify battery connection resistance is ≤ 5.0 E-5 ohm for inter-cell connections, ≤ 5.0 E-5 ohm for inter-rack connections, ≤ 5.0 E-5 ohm for inter-tier connections, ≤ 5.0 E-5 ohm for terminal connections; for Div 1 and Div 2 and ≤ 1.0 E-4 ohm for inter-cell connections, ≤ 1.0 E-4 ohm for inter-rack connections, ≤ 1.0 E-4 ohm for inter-tier connections, ≤ 1.0 E-4 ohm for terminal connections for Div 3.	24 months
SR 3.8.4.6	Verify each required Division 1 and 2 battery charger supplies ≥ 400 amps at ≥ 125 V for ≥ 8 hours; and each required Division 3 battery charger supplies ≥ 50 amps at ≥ 125 V for ≥ 8 hours.	24 months
SR 3.8.4.7	------------------NOTE----------------- SR 3.8.4.8 may be performed in lieu of SR 3.8.4.7 once per 60 months. --------------------------------------- Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.	24 months

(continued)

SRO Question 4

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.4.8	Verify battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test.	60 months AND ------NOTE------ Only applicable when battery shows degradation or has reached 85% of expected life. --------------- 18 months

SRO Question 4

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources—Shutdown

LCO 3.8.5 The following DC electrical power subsystems shall be OPERABLE:

a. One Class 1E DC electrical power subsystem capable of supplying one division of the Division 1 or 2 onsite Class 1E electrical power distribution subsystem(s) required by LCO 3.8.8, "Distribution Systems - Shutdown";

b. One Class 1E battery or battery charger, other than the DC electrical power subsystem in LCO 3.8.5.a, capable of supplying the remaining Division 1 or Division 2 onsite Class 1E DC electrical power distribution subsystem when required by LCO 3.8.8; and

c. The Division 3 DC electrical power subsystem capable of supplying the Division 3 onsite Class 1E DC electrical power distribution subsystem, when the Division 3 onsite Class 1E DC electrical power distribution subsystem is required by LCO 3.8.8.

APPLICABILITY: MODES 4 and 5,
During movement of recently irradiated fuel assemblies in the primary containment or fuel handling building.

SRO Question 4

ACTIONS

-----NOTE-----
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required DC electrical power subsystems inoperable.	A.1 Declare affected required feature(s) inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2.2 Suspend movement of recently irradiated fuel assemblies in the primary containment and fuel handling building.	Immediately
	AND	
	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately
	AND	
	A.2.4 Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately

SRO Question 4

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.5.1	------------------NOTE------------------ The following SRs are not required to be performed: SR 3.8.4.4, SR 3.8.4.6, SR 3.8.4.7, and SR 3.8.4.8. -- For DC sources required to be OPERABLE, the following SRs are applicable: SR 3.8.4.1 SR 3.8.4.4 SR 3.8.4.7 SR 3.8.4.2 SR 3.8.4.5 SR 3.8.4.8. SR 3.8.4.3 SR 3.8.4.6	In accordance with applicable SRs

SRO Question 6

3.3 INSTRUMENTATION

3.3.7.1 Control Room Emergency Recirculation (CRER) System Instrumentation

LCO 3.3.7.1 The CRER System instrumentation for each Function in Table 3.3.7.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.7.1-1.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.7.1-1 for the channel.	Immediately
B. As required by Required Action A.1 and referenced in Table 3.3.7.1-1.	B.1 Declare associated CRER subsystem inoperable.	1 hour from discovery of loss of CRER initiation capability in both trip systems
	<u>AND</u>	
	B.2 Place channel in trip.	24 hours

(continued)

SRO Question 6

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. As required by Required Action A.1 and referenced in Table 3.3.7.1-1.	C.1 Provide alternate method of control room radiation monitoring.	24 hours
	AND	
	C.2 Restore the inoperable monitor to an OPERABLE status.	7 days
D. Required Action and associated Completion Time of Condition B or C not met.	D.1 Place the associated CRER subsystem in the emergency recirculation mode of operation.	1 hour
	OR	
	D.2 Declare associated CRER subsystem inoperable.	1 hour

SRO Question 6

SURVEILLANCE REQUIREMENTS

-----NOTES-----

1. Refer to Table 3.3.7.1-1 to determine which SRs apply for each Function.
2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains CRER initiation capability.

SURVEILLANCE		FREQUENCY
SR 3.3.7.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.7.1.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.7.1.3	Calibrate the trip unit.	92 days
SR 3.3.7.1.4	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.7.1.5	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

SRO Question 6

Table 3.3.7.1-1 (page 1 of 1)
Control Room Emergency Recirculation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level — Low Low Low, Level 1	1,2,3, (a)	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4 SR 3.3.7.1.5	≥ 14.3 inches
2. Drywell Pressure — High	1,2,3	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4 SR 3.3.7.1.5	≤ 1.88 psig
3. Control Room Ventilation Radiation Monitor	1,2,3, (b)	1	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.4 SR 3.3.7.1.5	≤ 800 cpm

(a) During operations with a potential for draining the reactor vessel.

(b) During operations with a potential for draining the reactor vessel, and movement of recently irradiated fuel assemblies in the primary containment or fuel handling building.

SRO QUESTIONS 9 & 11

INITIATING CONDITION INDEX

EVENT CATEGORY	MODE	UNUSUAL EVENT	MODE	ALERT	MODE	SITE AREA EMERGENCY	MODE	GENERAL EMERGENCY
A: FISSION PRODUCT BARRIER DEGRADATION	1 2 3 4 5	Fuel clad degradation **Page 16 - AU1**	1 2 3	Any loss or challenge to the Fuel Clad barrier. **Page 19 (FPB Matrix) - AA1**	1 2 3 4 5	Loss of RPV water level that has or will uncover fuel. **Page 18 - AS1**	1 2 3	Loss of two barriers, AND a loss or challenge to the third barrier. **Page 19 (FPB Matrix) - AG1**
	1 2 3	Reactor Coolant System leakage. **Page 17 - AU2**	1 2 3	Any loss or challenge to the Reactor Coolant System barrier. **Page 19 (FPB Matrix) - AA2**	1 2 3	<u>Either</u> a challenge or loss of <u>both</u> the Fuel Clad barrier AND Reactor Coolant System barrier. **Page 20 (FPB Matrix) - AS2**		
	1 2 3	Any loss or challenge to the Containment barrier. **Page 19 (FPB Matrix) - AU3**			1 2 3	Challenge to <u>either</u> the Fuel Clad barrier OR Reactor Coolant System barrier, AND the loss of any additional barrier. **Page 20 (FPB Matrix) - AS3**		
B: LOSS OF DECAY HEAT REMOVAL FUNCTIONS		**NOT APPLICABLE**	4 5	Inability to maintain plant in COLD SHUTDOWN **Page 20 - BA1**	1 2 3	Complete loss of functions needed to achieve COLD SHUTDOWN **Page 21 - BS1**		**NOT APPLICABLE**
C: LOSS OF SHUTDOWN FUNCTIONS OR FAILURE TO SHUTDOWN	1 2 3	Inability to reach required shutdown within Technical Specification limits **Page 22 - CU1**	1 2	Failure to initiate or complete an automatic Reactor Scram once an RPS function is required. **Page 23 - CA1**	1	Failure to initiate or complete an automatic Reactor Scram once an RPS function is required, AND a manual Scram was <u>NOT</u> successful. **Page 24 - CS1**	1	Failure to initiate or complete a successful shutdown, AND indication of an extreme challenge to the ability to cool the core. **Page 26 - CG1**
D: A.C. POWER LOSS	1 2 3 4 5 D	Loss of all offsite power to Division 1 and 2 EH Essential Busses for greater than 15 minutes. **Page 28- DU1**	1 2 3	Power capability to Division 1 and 2 EH Essential Busses reduced to a single power source for greater than 15 minutes, such that any additional single failure would result in a Station Blackout. **Page 29 - DA1**	1 2 3	Loss of all offsite power AND onsite power to Division 1 and 2 EH Essential Busses for greater than 15 minutes. **Page 31 - DS1**	1 2 3	Prolonged loss of all offsite power AND onsite power to Division 1 and 2 EH Essential Busses, AND continuing degradation of core cooling capability. **Page 32 - DG1**
			4 5 D	Loss of all offsite power AND onsite power to Division 1 and 2 EH Essential Busses for greater than 15 minutes. **Page 30 - DA2**				
E: D.C. POWER DEGRADATION	4 5	Degradation of Division 1 and 2 essential DC power for greater than 15 minutes. **Page 34 - EU1**		**NOT APPLICABLE**	1 2 3	Degradation of Division 1 and 2 essential DC power for greater than 15 minutes. **Page 35 - ES1**		**NOT APPLICABLE**
F: FIRE OR EXPLOSION	1 2 3 4 5 D	Fire within a Safe Shutdown Building <u>NOT</u> extinguished within 15 minutes. **Page 36 - FU1**	1 2 3 4 5 D	Fire OR explosion affecting the operability of plant safety systems required to establish or maintain safe shutdown. **Page 38 - FA1**		**NOT APPLICABLE**		**NOT APPLICABLE**
	1 2 3 4 5 D	Explosion affecting a Safe Shutdown Building. **Page 37 - FU2**						
G: INCREASED PLANT RADIATION LEVELS	1 2 3 4 5 D	Unexpected increase in plant radiation levels. **Page 40 - GU1**	1 2 3 4 5 D	Increases in radiation levels within Safe Shutdown Buildings that impede operation of systems required to maintain safe operations OR to establish or maintain COLD SHUTDOWN. **Page 42 - GA1**		**NOT APPLICABLE**		**NOT APPLICABLE**
	1 2 3 4 5 D	Uncontrolled fuel pool or Reactor Cavity water level decrease with irradiated fuel remaining covered. **Page 41 - GU2**	1 2 3 4 5 D	Damage to irradiated fuel. **Page 43 - GA2**				

INITIATING CONDITION INDEX

EVENT CATEGORY	MODE	UNUSUAL EVENT	MODE	ALERT	MODE	SITE AREA EMERGENCY	MODE	GENERAL EMERGENCY
H: INCREASED RADIATION RELEASE TO THE ENVIRONMENT	1 2 3 4 5 D 1 2 3 4 5 D	Any unplanned release of gaseous radioactivity to the environment that exceeds two times the ODCM Control limit for 60 minutes or greater. **Page 46 - HU1** Any unplannned release of liquid radioactivity to the environment that exceeds two times the ODCM Control limit for 60 minutes or greater. **Page 47 - HU2**	1 2 3 4 5 D 1 2 3 4 5 D	Any unplanned release of gaseous radioactivity to the environment that exceeds 200 times the ODCM Control limit for 15 minutes or greater. **Page 48 - HA1** Any unplanned release of liquid radioactivity to the environment that exceeds 200 times the ODCM Control limit for 15 minutes or greater. **Page 49 HA2**	1 2 3 4 5 D	Site Boundary dose resulting from an actual or imminent release of gaseous radioactivity that exceeds 100 mRem TEDE dose OR 500 mRem CDE Child Thyroid dose for the actual OR projected duration of the release. **Page 50 - HS1**	1 2 3 4 5 D	Site Boundary dose resulting from an actual or imminent release of gaseous radioactivity that exceeds 1000 mRem TEDE OR 5000 mRem CDE Child Thyroid dose for the actual or projected duration of the release. **Page 51 - HG1**
I: CONTROL ROOM EVACUATION		**NOT APPLICABLE**	1 2 3 4 5 D	Control Room Evacuation has been initiated. **Page 52 - IA1**	1 2 3 4 5 D	Control Room evacuation has been initiated, AND plant control <u>CANNOT</u> be established within 15 minutes. **Page 53 - IS1**		**NOT APPLICABLE**
J: LOSS OF ANNUNCIATORS OR INDICATIONS	1 2 3	Loss of most annunciators or indication in the Control Room for greater than 15 minutes. **Page 54 - JU1**	1 2 3	Loss of most annunciators or indication in the Control Room with <u>either</u>: (1) a significant transient in progress; OR (2) compensatory indications are <u>NOT</u> available. **Page 55 - JA1**	1 2 3	Inability to monitor a significant transient in progress. **Page 56 - JS1**		**NOT APPLICABLE**
K: LOSS OF COMMUNICATIONS	1 2 3 4 5 D 1 2 3 4 5 D	Loss of onsite OR in-plant communications capabilities. **Page 58 - KU1** Significant degradation of offsite communications capabilities. **Page 59 - KU2**		**NOT APPLICABLE**		**NOT APPLICABLE**		**NOT APPLICABLE**
L: NATURAL OR DESTRUCTIVE PHENOMENA	1 2 3 4 5 D	Natural OR destructive phenomena affecting the Protected Area boundary. **Page 60 - LU1**	1 2 3 4 5 D	Natural OR destructive phenomena affecting Safe Shutdown Buildings. **Page 61 - LA1**		**NOT APPLICABLE**		**NOT APPLICABLE**
M: RELEASE OF TOXIC OR FLAMMABLE GAS	1 2 3 4 5 D	Release of toxic OR flammable gasses affecting the Protected Area boundary deemed detrimental to the safe operation of the plant. **Page 62- MU1**	1 2 3 4 5 D	Release of toxic OR flammable gases within a Safe Shutdown Building which jeopardizes operation of systems required to maintain safe operations OR to establish or maintain COLD SHUTDOWN. **Page 63- MA1**		**NOT APPLICABLE**		**NOT APPLICABLE**
N: SECURITY EVENTS	1 2 3 4 5 D	Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant. **Page 64 - NU1**	 1 2 3 4 5 D 1 2 3 4 5 D	**NOT APPLICABLE** Notification of an airborne attack threat. **Page 66 - NA2** HOSTILE ACTION within the OWNER CONTROLLED AREA **Page 67 - NA3**	 1 2 3 4 5 D	**NOT APPLICABLE** HOSTILE ACTION within the PROTECTED AREA **Page 69 - NS2**	1 2 3 4 5 D	HOSTILE ACTION resulting in loss of physical control of the facility. **Page 70 - NG1**
O: EMERGENCY COORDINATOR'S JUDGEMENT	1 2 3 4 5 D	Other conditions existing, which in the judgement of the Emergency Coordinator, warrant declaration of an Unusual Event. **Page 72- OU1**	1 2 3 4 5 D	Other conditions existing, which in the judgement of the Emergency Coordinator, warrant declaration of an Alert. **Page 73 - OA1**	1 2 3 4 5 D	Other conditions existing, which in the judgement of the Emergency Coordinator, warrant declaration of a Site Area Emergency. **Page 74- OS1**	1 2 3 4 5 D	Other conditions existing, which in the judgement of the Emergency Coordinator, warrant declaration of a General Emergency. **Page 75 - OG1**

Category A: Fission Product Barrier Degradation

Initiating Conditions	Entry Criteria	
AU1 **Fuel clad degradation** **Applicable Modes**: **1 2 3 4 5**	High Offgas pretreatment air activity greater than the Technical Specification 3.7.5.	Reactor Coolant System sample indicates activity greater than Technical Specification 3.4.8 limits.

AU1

UNUSUAL EVENT

NOTE

Fuel clad degradation is NOT an issue when the Reactor is defueled. Damage to fuel in spent fuel pools is addressed in GU1.

<table>
<tr><th>Initiating Conditions</th><th colspan="3">Entry Criteria</th></tr>
<tr><td rowspan="2">AU2

Reactor Coolant System leakage.

Applicable Modes:
1 | 2 | 3 | | |</td><td rowspan="2">Greater than 10 gpm unidentified leakage in Drywell.</td><td rowspan="2">Greater than 30 gpm total leakage in Drywell averaged over the previous 24 hour period.</td><td>Greater than 30 gpm total leakage in Drywell.</td></tr>
<tr><td>Greater than 2 gpm increase in unidentified leakage within the previous 24 hour period.</td></tr>
</table>

AU2

UNUSUAL EVENT

<table>
<tr><th colspan="6">Initiating Conditions</th><th>Entry Criteria</th></tr>
<tr><td colspan="6" rowspan="2">AS1

Loss of RPV water level that has or will uncover fuel

Applicable Modes:</td><td>RPV water level <u>CANNOT</u> be maintained greater than 0 inches.</td></tr>
<tr><td rowspan="2">Reactor is “shutdown under all conditions without boron”.</td></tr>
<tr><td>1</td><td>2</td><td>3</td><td>4</td><td>5</td><td></td></tr>
</table>

<u>AS1</u>

SITE AREA EMERGENCY

<u>NOTE</u>

AS1 is applicable only to non-ATWS situations in which RPV level was <u>NOT</u> intentionally lowered per <EOP-01A> as a means of power control. Refer to Event Category ”C” for classification under ATWS conditions in which RPV water level is intentionally lowered.

PNPP No. 9855 Rev. 8/26/08

FISSION PRODUCT BARRIER MATRIX

EPLAN/EPI-A1/PSI-0019

FUEL CLAD BARRIER		REACTOR PRESSURE VESSEL LEVEL	DRYWELL RADIATION	REACTOR COOLANT SYSTEM ACTIVITY	EMERGENCY COORDINATOR JUDGEMENT
	LOSS CRITERIA	Entry into SAG-1, Primary Containment Flooding.[3]	Drywell radiation monitor reading greater than 4,000 Rem/hr.	Sample activity is equal to or greater than 300 uCi/gm dose equivalent iodine-131.[5]	Any condition that, in the judgment of the Emergency Coordinator indicates loss of the Fuel Clad barrier.[1]
	CHALLENGE CRITERIA	RPV level is less than 0 inches. [2] RPV level CANNOT be determined.	NOT APPLICABLE	NOT APPLICABLE	Any condition that, in the judgment of the Emergency Coordinator indicates a challenge to the Fuel Clad barrier.[1]

INSTRUCTIONS

1. For each of the three barriers, determine if any **LOSS** or **CHALLENGE** criteria have been met.
2. Compare the barrier **LOSS(es)** and **CHALLENGE(s)** to the initiating conditions listed, and make the appropriate event declaration.

REACTOR COOLANT SYSTEM BARRIER		REACTOR PRESSURE VESSEL LEVEL	DRYWELL RADIATION	REACTOR PRESSURE CONTROL			DRYWELL PRESSURE	REACTOR COOLANT SYSTEM BYPASS	EMERGENCY COORDINATOR JUDGEMENT
	LOSS CRITERIA	RPV level less than 0 inches. [2]	Drywell radiation monitor reading greater than 135 Rem/hr.	SRV stuck open. Sample activity is equal to or greater than 300 uCi/gm dose equivalent Iodine-131.[5]	An SRV is being cycled to control RPV pressure.	Emergency Depressurization is required.	Drywell pressure greater than 1.68 psig. Indication of RCS leakage inside the Drywell.	MSL break outside containment exceeding one or more MSIV Tech. Spec. allowable values. Containment penetration does NOT isolate on a valid closure signal. Immediate Operator actions in the Control Room are NOT successful in isolating affected penetration.	Any condition that, in the judgment of the Emergency Coordinator, indicates loss of the RCS barrier.[1]
	CHALLENGE CRITERIA	NOT APPLICABLE	NOT APPLICABLE	NOT APPLICABLE			NOT APPLICABLE	Unisolable primary system discharging outside Containment per either EOP-03 or EOP-05. [4]	Any condition that, in the judgment of the Emergency Coordinator, indicates a challenge to the RCS barrier.[1]

CONTAINMENT BARRIER		REACTOR PRESSURE VESSEL LEVEL	CONTAINMENT RADIATION	CONTAINMENT HYDROGEN	CONTAINMENT PRESSURE		CONTAINMENT ISOLATION		EMERGENCY COORDINATOR JUDGEMENT
	LOSS CRITERIA	Entry into SAG-1, Primary Containment Flooding [3]	NOT APPLICABLE	Intentional venting of Containment required per SAG-2	Intentional venting of Containment required per EOP-02		Containment penetration does NOT isolate on a valid closure signal. Immediate Operator actions in the Control Room are NOT successful in isolating affected penetration. Pathway to the environment exists via penetration.	Unisolable primary system discharging outside Containment per either EOP-03 or EOP-05. [4]	Any condition that, in the judgment of the Emergency Coordinator, indicates loss of the Containment barrier.[1] (Loss of the Containment barrier may include a rapid unexplained decrease in Containment pressure following an initial increase.)
	CHALLENGE CRITERIA	NOT APPLICABLE	Containment radiation monitor reading greater than 20,000 Rem/hr.	NOT APPLICABLE	In the UNSAFE region on the HCL figure.	Containment pressure is greater than 15 psig and increasing.	NOT APPLICABLE		Any condition that, in the judgment of the Emergency Coordinator, indicates a challenge to the Containment barrier.[1]

INITIATING CONDITIONS

UNUSUAL EVENT

AU3 Any loss or challenge to the Containment barrier.
Modes: 1 2 3

ALERT

AA1 Any loss or challenge to Fuel Clad barrier.
Modes: 1 2 3

AA2 Any loss or challenge to the Reactor Coolant System barrier.
Modes: 1 2 3

SITE AREA EMERGENCY

AS2 Either a challenge or loss of both the Fuel Clad barrier AND Reactor Coolant System barrier.
Modes: 1 2 3

AS3 Challenge to either the Fuel Clad barrier OR Reactor Coolant System barrier, AND the loss of any additional barrier.
Modes: 1 2 3

GENERAL EMERGENCY

AG1 Loss of two barriers, AND a loss or challenge to the third barrier.
Modes: 1 2 3

NOTE: If the criteria for loss has been met, then by definition, the challenge criteria has been met for that same fission product barrier.

FOOTNOTES:
1. Those thresholds for which a LOSS or CHALLENGE is determined to be IMMINENT (i.e., within the next 3 hours), classify as though the threshold(s) has been exceeded.
2. RPV level is less than 0 inches is both a FUEL CLAD BARRIER CHALLENGE CRITERIA and a REACTOR COOLANT SYSTEM BARRIER LOSS CRITERIA.
3. Entry into SAG-1, Primary Containment Flooding is both a FUEL CLAD BARRIER LOSS CRITERIA and a CONTAINMENT BARRIER LOSS CRITERIA.
4. Unisolable primary system discharging outside containment per EOP-03 and/or EOP-05 is both a REACTOR COOLANT SYSTEM BARRIER CHALLENGE CRITERIA and a CONTAINMENT BARRIER LOSS CRITERIA.
5. Sample activity is equal to or greater than 300 uCi/gm dose equivalent Iodine-131 is both a FUEL CLAD BARRIER LOSS CRITERIA and a contributor to a REACTOR COOLANT SYSTEM BARRIER LOSS CRITERIA.

Category B: Loss of Decay Heat Removal Functions

<table>
<tr><th colspan="6">Initiating Conditions</th><th colspan="2">Entry Criteria</th><th></th></tr>
<tr><td colspan="6" rowspan="3">BA1

Inability to maintain plant in COLD SHUTDOWN

Applicable Modes:</td><td colspan="2">Loss of Shutdown Cooling Mode function for RHR loop A.</td><td rowspan="4"><u>BA1</u>

A
L
E
R
T</td></tr>
<tr><td colspan="2">Loss of Shutdown Cooling Mode function for RHR loop B.</td></tr>
<tr><td rowspan="2">RCS temperature exceeds COLD SHUTDOWN limit of 200°F per Technical Specification Table 1.1-1.</td><td rowspan="2">Uncontrolled temperature rise approaching 200°F RCS temperature.</td></tr>
<tr><td></td><td></td><td></td><td>4</td><td>5</td><td></td></tr>
</table>

NOTES

- The IC remains applicable for situations in which an increase in RCS temperature greater than 200°F results in a change to Mode 3.
- The above criteria is met as soon as it becomes known that sufficient cooling CANNOT be restored to maintain temperature below 200°F regardless of the current temperature. The intent of BA1 is NOT to classify based on an unplanned excursion above 200°F when heat removal capability is available.
- “Uncontrolled” means that RCS temperature increase is NOT the result of planned actions by plant staff.

<table>
<tr><th>Initiating Conditions</th><th>Entry Criteria</th></tr>
<tr><td rowspan="2">BS1

Complete loss of functions needed to achieve COLD SHUTDOWN

Applicable Modes:
1 | 2 | 3</td><td>RHR Loops A and B are <u>NOT</u> capable of lowering RPV temperature.</td></tr>
<tr><td>The plant is operating in the UNSAFE Region on the HCL figure.</td></tr>
</table>

BS1

SITE AREA EMERGENCY

Category C: Loss of Shutdown Functions or Failure to Shutdown

Initiating Conditions	Entry Criteria	
CU1 **Inability to reach required shutdown within Technical Specification limits** **Applicable Modes**: **1** | **2** | **3**	Plant is NOT brought to the required operating mode within the Technical Specification Required Action Completion Time following entry into an LCO.	**CU1** **UNUSUAL EVENT**

NOTE

Declaration should be made because of equipment failures that prevent the performance of an orderly shutdown or failure to meet the shutdown completion time from the time discovered and a required action being entered. Declaration of an Unusual Event is based on the time at which the specified completion time period elapses and is NOT related to how long a condition may have existed before it was discovered.

Initiating Conditions	Entry Criteria	
CA1 **Failure to initiate or complete an automatic Reactor Scram once an RPS function is required** **Applicable Modes**: **1** **2**	Actuation of RPS has occurred or should have occurred.	Actuation of RRCS has occurred or should have occurred.
	The reactor is <u>NOT</u> "shutdown under all conditions without boron."	

CA1

A L E R T

<u>NOTES</u>

- CA1 is applicable if <u>either</u> Mode 1 or 2 existed when the transient started and <u>NOT</u> the mode which exists at the time of classification.
- Entry criteria is applicable for actions taken by an Operator to manually initiate <u>either</u> RPS or RRCS prior to or after exceeding an automatic actuation setpoint.

<table>
<tr><th>Initiating Conditions</th><th colspan="6">Entry Criteria</th></tr>
<tr><td rowspan="3">CS1

Failure to initiate or complete an automatic Reactor Scram once an RPS function is required, AND a manual Scram was <u>NOT</u> successful

Applicable Modes:
1</td><td colspan="3">Actuation of RPS has occurred or should have occurred.</td><td colspan="3">Actuation of RRCS has occurred or should have occurred.</td></tr>
<tr><td colspan="6">The reactor is <u>NOT</u> “shutdown under all conditions without boron.”</td></tr>
<tr><td colspan="2">Manual operator actions taken at 1H13-P680 to insert control rods were <u>NOT</u> successful in lowering Reactor power to less than 4%.</td><td colspan="2">Reactor power <u>CANNOT</u> be determined.</td><td colspan="2">Suppression Pool temperature is greater than 110°F.</td></tr>
</table>

CS1

SITE AREA EMERGENCY

NOTE

Refer to next page.

NOTES

- CS1 is applicable if Mode 1 existed when the transient started and NOT the mode which exists at the time of classification. Refer to CA1 for Mode 2 applicability.

- "Manual Operator actions" are defined as any set of actions by the Reactor Operator at 1H13-P680 which results in a scram signal. These actions include placing the Reactor Mode Switch in the SHUTDOWN position, arming and depressing the RPS Manual Scram push buttons, and arming and depressing the RRCS Manual ARI push buttons. Injection of boron is NOT considered in reducing reactor power below 4%.

- A concurrent challenge to the ability to cool the core would escalate this event to General Emergency per CG1.

<table>
<tr><th colspan="6">Initiating Conditions</th><th colspan="6">Entry Criteria</th></tr>
<tr><td colspan="6" rowspan="4">CG1

Failure to initiate or complete a successful shutdown, AND indication of an extreme challenge to the ability to cool the core

Applicable Modes:</td><td colspan="3">Actuation of RPS has occurred or should have occurred.</td><td colspan="3">Actuation of RRCS has occurred or should have occurred.</td></tr>
<tr><td colspan="6">The reactor is <u>NOT</u> “shutdown under all conditions without boron.”</td></tr>
<tr><td colspan="2">Manual operator actions taken at 1H13-P680 to insert control rods were <u>NOT</u> successful in lowering Reactor power to less than 4%.</td><td colspan="2">Reactor power <u>CANNOT</u> be determined.</td><td colspan="2">Suppression Pool temperature is greater than 110°F.</td></tr>
<tr><td colspan="6" rowspan="2">Any of the following conditions exist:

• Entry into <SAG-1>, Primary Containment Flooding.
• In the UNSAFE region on the HCL figure.</td></tr>
<tr><td>1</td><td></td><td></td><td></td><td></td><td></td></tr>
</table>

CG1

G E N E R A L E M E R G E N C Y

NOTES

- CG1 is applicable if Mode 1 existed when the transient started and <u>NOT</u> the mode which exists at the time of classification. Refer to CA1 for Mode 2 applicability.
- Entry criteria is applicable for actions taken by an Operator to manually initiate <u>either</u> RPS or RRCS prior to or after exceeding an automatic actuation setpoint.

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Category D: A. C. Power Loss

<table>
<tr><th>Initiating Conditions</th><th>Entry Criteria</th><th></th></tr>
<tr><td rowspan="3">DU1

Loss of all offsite power to Division 1 and 2 EH Essential Busses for greater than 15 minutes

Applicable Modes:
1 | 2 | 3 | 4 | 5 | D</td><td>ONI-R10 entered for a Loss of Off-site Power (LOOP).</td><td rowspan="3"><u>DU1</u>

UNUSUAL EVENT</td></tr>
<tr><td><u>Either</u> of the following power sources <u>CANNOT</u> be made available within 15 minutes for energizing bus EH11:

• Normal Preferred
• Alternate Preferred</td></tr>
<tr><td><u>Either</u> of the following power sources <u>CANNOT</u> be made available within 15 minutes for energizing bus EH12:

• Normal Preferred
• Alternate Preferred</td></tr>
</table>

<u>NOTE</u>

Failure of <u>either</u> bus EH11 or EH12 to be supplied from its respective diesel generator is evaluated for escalation to an Alert under DA1 for Modes 1, 2 and 3. Failure of <u>both</u> busses EH11 and EH12 to be supplied from their respective diesel generators (Station Black Out) is evaluated for escalation to an Alert under DA2 for Modes 4 and 5 and to a Site Area Emergency under DS1 for Modes 1, 2 and 3.

Initiating Conditions	Entry Criteria	
DA1 **Power capability to Division 1 and 2 EH Essential Busses reduced to a single power source for greater than 15 minutes, such that any additional single failure would result in Station Blackout** **Applicable Modes:** **1 2 3**	Essential AC power reduced to only <u>one</u> of the following power sources for greater than 15 minutes: • Normal Preferred • Alternate Preferred • Division 1 Diesel Generator • Division 2 Diesel Generator Loss of the single remaining power source will result in a loss of AC power to <u>both</u> busses EH11 and EH12.	**<u>DA1</u>** **A L E R T**

NOTES

- Escalation to a Site Area Emergency is evaluated under DS1, for Operating Modes 1, 2 and 3, based on a total loss of AC power to <u>both</u> busses EH11 and EH12.
- A total loss of AC power to busses EH11 and EH12 while in Operating Modes 4 and 5 is classified as an Alert under DA2. No escalation path exists to a Site Area Emergency for Operating Modes 4 and 5.

<table>
<tr><th colspan="6">Initiating Conditions</th><th>Entry Criteria</th></tr>
<tr><td colspan="6" rowspan="3">DA2

Loss of all offsite power AND onsite power to Division 1 and 2 EH Essential Busses for greater than 15 minutes.

Applicable Modes:</td><td><u>Both</u> busses EH11 and EH12 <u>CANNOT</u> be energized from the Normal Preferred source within 15 minutes.</td></tr>
<tr><td><u>Both</u> busses EH11 and EH12 <u>CANNOT</u> be energized from the Alternate Preferred source within 15 minutes.</td></tr>
<tr><td rowspan="2"><u>Both</u> busses EH11 and EH12 <u>CANNOT</u> be energized from the Associated Diesel Generator source within 15 minutes.</td></tr>
<tr><td></td><td></td><td></td><td>4</td><td>5</td><td>D</td></tr>
</table>

<u>**DA2**</u>

A L E R T

Initiating Conditions	Entry Criteria
DS1 **Loss of all offsite power AND onsite power to Division 1 and 2 EH Essential Busses for greater than 15 minutes**	Both busses EH11 and EH12 CANNOT be energized from the Normal Preferred source within 15 minutes.
	Both busses EH11 and EH12 CANNOT be energized from the Alternate Preferred source within 15 minutes.
Applicable Modes: **1 2 3**	Both busses EH11 and EH12 CANNOT be energized from the Associated Diesel Generator source within 15 minutes.

DS1

SITE AREA EMERGENCY

NOTE

Escalation to a General Emergency is evaluated under DG1 for Modes 1, 2 and 3, based on a continuing degradation of core cooling capability.

<table>
<tr><th>Initiating Conditions</th><th colspan="3">Entry Criteria</th><th></th></tr>
<tr><td rowspan="4">DG1

Prolonged loss of all offsite power AND onsite power to Division 1 and 2 EH Busses, AND continuing degradation of core cooling capability

Applicable Modes:
1 | 2 | 3</td><td colspan="3"><u>Both</u> busses EH11 and EH12 <u>CANNOT</u> be energized from the Normal Preferred source.</td><td rowspan="4"><u>DG1</u>

GENERAL EMERGENCY</td></tr>
<tr><td colspan="3"><u>Both</u> busses EH11 and EH12 <u>CANNOT</u> be energized from the Alternate Preferred source.</td></tr>
<tr><td colspan="3"><u>Both</u> busses EH11 and EH12 <u>CANNOT</u> be energized from the Associated Diesel Generator source.</td></tr>
<tr><td>Restoration of power to either of the following busses is <u>NOT</u> likely in less than four hours:

• EH11
• EH12</td><td>RPV water level less than 0 inches.</td><td>RPV water level <u>CANNOT</u> be determined.</td></tr>
</table>

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Category E: D. C. Power Degradation

<table>
<tr><th>Initiating Conditions</th><th>Entry Criteria</th></tr>
<tr><td rowspan="2">EU1

Degradation of Division 1 and 2 essential DC power for greater than 15 minutes

Applicable Modes:
| | | | 4 | 5 | |</td><td>Voltage on ED-1-A buss is less than 105 VDC for greater than 15 minutes.</td></tr>
<tr><td>Voltage on ED-1-B buss is less than 105 VDC for greater than 15 minutes.</td></tr>
</table>

EU1

UNUSUAL EVENT

NOTE

The same set of conditions as described in this EAL would be classified as Site Area Emergency under ES1 if they occurred during Modes 1, 2, or 3.

<table>
<tr><th colspan="6">Initiating Conditions</th><th>Entry Criteria</th></tr>
<tr><td colspan="6" rowspan="2">ES1

Degradation of Division 1 and 2 essential DC power for greater than 15 minutes

Applicable Modes:</td><td>Voltage on ED-1-A buss is less than 105 VDC for greater than 15 minutes.</td></tr>
<tr><td rowspan="2">Voltage on ED-1-B buss is less than 105 VDC for greater than 15 minutes.</td></tr>
<tr><td>1</td><td>2</td><td>3</td><td></td><td></td><td></td></tr>
</table>

ES1

S I T E

A R E A

E M E R E N G C Y

Category F: Fire or Explosion

<table>
<tr><th colspan="6">Initiating Conditions</th><th colspan="2">Entry Criteria</th><th rowspan="4"><u>FU1</u>

UNUSUAL EVENT</th></tr>
<tr><td colspan="6" rowspan="2">FU1

Fire within a Safe Shutdown Building <u>NOT</u> extinguished within 15 minutes

Applicable Modes:</td><td colspan="2">Fire within any Safe Shutdown Building.</td></tr>
<tr><td rowspan="2">Fire <u>CANNOT</u> be extinguished within 15 minutes of the verification of alarm.</td><td rowspan="2">Fire <u>CANNOT</u> be extinguished within 15 minutes of the notification received in the Control Room from plant personnel that a fire exists.</td></tr>
<tr><td>1</td><td>2</td><td>3</td><td>4</td><td>5</td><td>D</td></tr>
</table>

<u>NOTE</u>

Verification in this context means those actions taken in the Secondary Alarm Station (SAS) to determine that the alarm is <u>NOT</u> spurious. Verification includes the receipt of multiple or independent alarms or confirmation of a single detector by visual inspection of the affected area by a first responder. List of Safe Shutdown Buildings is found in Section 3 “Definitions”, sub-step 3.10.

Initiating Conditions						Entry Criteria
FU2 **Explosion affecting a Safe Shutdown Building** **Applicable Modes**:						Report by plant personnel confirming the occurrence of an explosion within the Protected Area resulting in visible damage to a Safe Shutdown Building.
1	**2**	**3**	**4**	**5**	**D**	

FU2

U N U S U A L E V E N T

NOTE

No attempt is made to assess the magnitude of the damage. The occurrence of the explosion with reports of damage (deformation/scorching) is sufficient for declaration. Actual damage to safe shutdown equipment is covered under Alert FA1. List of Safe Shutdown Buildings is found in Section 3 "Definitions", sub-step 3.10.

<table>
<tr><th colspan="6">Initiating Conditions</th><th colspan="2">Entry Criteria</th><th></th></tr>
<tr><td colspan="6" rowspan="2">FA1

Fire OR explosion affecting the operability of plant safety systems required to establish or maintain safe shutdown

Applicable Modes:</td><td colspan="2"><u>Either</u> of the following has been confirmed:

• Fire in a Safe Shutdown Building.
• Explosion in a Safe Shutdown Building.</td><td rowspan="3"><u>FA1</u>

A
L
E
R
T</td></tr>
<tr><td rowspan="2">Plant personnel at the scene report visible damage to safe shutdown equipment or components.</td><td rowspan="2">Affected safe shutdown system indicates degraded performance.</td></tr>
<tr><td>1</td><td>2</td><td>3</td><td>4</td><td>5</td><td>D</td></tr>
</table>

NOTE

The inclusion of a "report of visible damage" should NOT be interpreted as mandating a lengthy damage assessment prior to classification. NO attempt is made in this EAL to assess the actual magnitude of damage beyond the immediate area. The occurrence of the explosion or fire with reports of evidence of damage (e.g., deformation, scorching) is sufficient for declaration. List of Safe Shutdown Buildings is found in Section 3 "Definitions", sub-step 3.10.

<u>NOTE</u>

Safe Shutdown System/Equipment refers to equipment identified in the Safe Shutdown Capability Report. This is the minimum list of equipment required to achieve and maintain COLD SHUTDOWN (including all auxiliary equipment such as AC/DC power, cooling water and instrumentation). A detailed list is provided in the <Appendix R Evaluation - Safe Shutdown Capability Report>.

Safe Shutdown System/Equipment list: (Division 1 and 2 only)

- Reactor Protection System
- Control Rod Drive Hydraulics
- Automatic Depressurization System/SRV
- Reactor Core Isolation Cooling
- Low Pressure Core Spray
- Low Pressure Coolant Injection - A/B/C
- Suppression Pool Cooling
- Shutdown Cooling
- Safety-Related Instrument Air
- Emergency Service Water
- Emergency Service Water Screen Wash
- Emergency Service Water Pump House Ventilation
- ECCS Pump Room Cooling Systems
- Diesel Generator Building Ventilation
- Stand-by Diesel Generator (DG)
- DG Fuel Oil Storage/Transfer
- Electrical Power Distribution
- Emergency Closed Cooling Pump Area Cooling
- Emergency Closed Cooling
- Control Complex Chilled Water
- MCC, Switchgear and Miscellaneous Electrical Equipment Areas HVAC System
- Battery Room Exhaust
- Control Room HVAC and Emergency Recirculation System

(Reference: <NUMARC/NESP-007> (Rev. 2), Unusual Event HA2)

Category G: Increased Plant Radiation Levels

<table>
<tr><th>Initiating Conditions</th><th colspan="2">Entry Criteria</th><th></th></tr>
<tr><td rowspan="2">GU1

Unexpected increase in plant radiation levels

Applicable Modes:
1 | 2 | 3 | 4 | 5 | D</td><td>A valid area radiation monitor (D21) reading increases by a factor of 1000 over normal levels.</td><td>Health Physics surveys indicate an increase by a factor of 1000 times over normally expected area radiation levels.</td><td rowspan="2"><u>GU1</u>

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T</td></tr>
<tr><td colspan="2">In-plant radiation level increase <u>CANNOT</u> be attributed to <u>any</u> of the following:

• the start-up and operation of plant equipment or systems within design parameters.
• the planned movement of radioactive materials.
• the planned movement of shielding (i.e., plugs, lead shot, etc.)</td></tr>
</table>

NOTE

“Normal” area radiation levels can be considered as the highest reading in the past 24-hour period, excluding the current peak value.

Initiating Conditions	Entry Criteria					
GU2 **Uncontrolled fuel pool or Reactor Cavity water level decrease with irradiated fuel remaining covered.** **Applicable Modes:**	Uncontrolled decrease in <u>one or more</u> of the following fuel pools containing irradiated fuel: • Reactor Cavity • FHB Fuel Storage and Preparation Pool • FHB Fuel Transfer Pool • FHB Spent Fuel Storage Pool • FHB Cask Pit • CNTMT Fuel Storage Pool • CNTMT Fuel Transfer Pool					
1	**2**	**3**	**4**	**5**	**D**	

<u>GU2</u>

U N U S U A L E V E N T

<table>
<tr><th colspan="6">Initiating Conditions</th><th colspan="2">Entry Criteria</th><th></th></tr>
<tr><td colspan="6" rowspan="2">GA1
Increases in radiation levels within Safe Shutdown Buildings that impede operation of systems required to maintain safe operations OR to establish or maintain COLD SHUTDOWN

Applicable Modes:</td><td rowspan="3">Area radiation levels of greater than 15 mRem/hr in any of the following areas:

• Control Room
• Central Alarm Station</td><td>Area radiation levels of greater than 6000 mRem/hr in a Safe Shutdown Building, as determined by <u>either</u>:

• area radiation surveys
• installed or portable radiation monitors</td><td rowspan="3"><u>GA1</u>

A
L
E
R
T</td></tr>
<tr><td rowspan="2">Access is required to maintain safe operation or perform a safe shutdown, as determined by the Shift Manager.</td></tr>
<tr><td>1</td><td>2</td><td>3</td><td>4</td><td>5</td><td>D</td></tr>
</table>

<u>NOTE</u>

This IC addresses increased radiation levels that impede necessary access to operating stations or other areas containing equipment that <u>must</u> be operated manually in order to maintain safe operation or perform a safe shutdown. It is this impaired ability to operate the plant that results in the actual or potential substantial degradation of the level of safety of the plant.

<table>
<tr><th colspan="6">Initiating Conditions</th><th colspan="2">Entry Criteria</th></tr>
<tr><td colspan="6" rowspan="2">GA2

Damage to irradiated fuel

Applicable Modes:</td><td>Damage to irradiated fuel</td><td rowspan="3">Water level observed to be below top of the gate sill separating <u>any</u> of the following containing irradiated fuel:

• Reactor Cavity
• FHB Fuel Storage and Preparation Pool
• FHB Fuel Transfer Pool
• FHB Spent Fuel Storage Pool
• FHB Cask Pit
• CNTMT Fuel Storage Pool
• CNTMT Fuel Transfer Pool</td></tr>
<tr><td rowspan="2">A valid HIGH alarm on <u>one or more</u> of the following radiation monitors:

• SPENT FUEL POOL D21-K332
• UPPER POOL AREA 1D21-K083
• FUEL PREP POOL D21-K322
• FHB VENT EXH GAS D17-K716
• CNTMT ATMOS GAS 1D17-K686</td></tr>
<tr><td>1</td><td>2</td><td>3</td><td>4</td><td>5</td><td>D</td></tr>
</table>

<u>GA2</u>

A L E R T

<u>NOTE</u>

The intent of this EAL is to allow observations from plant personnel to be factored into the declaration decision and is <u>not</u> intended to direct an entry into an area solely to observe pool level. The gate sill is the lip between the pools where the bottom of the gate would sit if installed.

NOTE

This EAL is only applicable to emergency conditions and is NOT applicable to pre-planned evolutions such as a reactor drain down to remove or re-install the reactor head during refueling operations.

NOTE

Damage to irradiated fuel is defined as; a degraded fuel bundle that results in the release of fission product gasses normally present in the fuel rod gap to the surrounding environment.

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Category H: Increased Radiation Release to the Environment

<table>
<tr><th colspan="6">Initiating Conditions</th><th colspan="2">Entry Criteria</th><th></th></tr>
<tr><td colspan="6" rowspan="3">HU1

Any unplanned release of gaseous radioactivity to the environment that exceeds two times the ODCM Control limit for 60 minutes or greater.

Applicable Modes:</td><td>A valid reading greater than TWO times the HIGH alarm setpoint on <u>one or more</u> of the following plant gaseous effluent monitors lasting greater than or equal to 60 minutes:

• PLANT VENT GAS 1D17-K786
• OG VENT PIPE GAS 1D17-K836
• TB/HB VENT GAS 1D17-K856
• PLANT VENT GAS 2D17-K786</td><td>Routine or as required sample analysis indicates a release rate greater than two times ODCM 3.11.2.1 limits.

The release lasts for greater than or equal to 60 minutes.</td><td rowspan="4"><u>HU1</u>

UNUSUAL EVENT</td></tr>
<tr><td colspan="2"></td></tr>
<tr><td>Chemistry sample analysis methods <u>CANNOT</u> confirm within 60 minutes of receipt of the HIGH alarm, on <u>one or more</u> of the plant gaseous effluent monitors, that effluent levels are less than two times ODCM 3.11.2.1 limits.</td><td></td></tr>
<tr><td>1</td><td>2</td><td>3</td><td>4</td><td>5</td><td>D</td><td></td><td></td></tr>
</table>

<u>NOTE</u>

It is <u>NOT</u> intended that the release be averaged over 60 minutes. Further, the Emergency Coordinator should <u>NOT</u> wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release will exceed TWO times the ODCM Control 3.11.2.1 limit for greater than 60 minutes.

<table>
<tr><th>Initiating Conditions</th><th colspan="3">Entry Criteria</th><th></th></tr>
<tr><td rowspan="2">HU2

Any unplanned release of liquid radioactivity to the environment that exceeds two times the ODCM Control limit for 60 minutes or greater.

Applicable Modes:
1 | 2 | 3 | 4 | 5 | D</td><td>A valid reading greater than 1.2E3 cpm above background for <u>one or more</u> of the following liquid process monitors lasting greater than or equal to 60 minutes:

• EMERGENCY SERVICE WATER LOOP A D17-K604

• EMERGENCY SERVICE WATER LOOP B D17-K605</td><td>A valid reading greater than 20 times the HIGH-HIGH alarm setpoint on NCC HX and RW EFF Common Process Radiation Recorder 0D17-R0170.</td><td>Routine or as required sample analysis indicates a release rate greater than two times ODCM Control 3.11.1.1 limits.</td><td rowspan="2"><u>HU2</u>

UNUSUAL EVENT</td></tr>
<tr><td>Chemistry sample analysis methods <u>CANNOT</u> confirm within 60 minutes of receipt of the HIGH-HIGH alarm, on <u>either</u> ESW Loop A or B radiation monitors, that liquid release levels are less than two times the ODCM Control 3.11.1.1 limits.</td><td>Release CANNOT be terminated within 60 minutes of exceeding NCC HX and RW EFF Common Process Radiation Recorder HIGH-HIGH alarm setpoints.</td><td>The release lasts for greater than or equal to 60 minutes.</td></tr>
</table>

<u>NOTE</u>

It is <u>NOT</u> intended that the release be averaged over 60 minutes. Further, the Emergency Coordinator should <u>NOT</u> wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release will exceed TWO times the ODCM Control 3.11.1.1 limit for greater than 60 minutes.

<table>
<tr><th>Initiating Conditions</th><th colspan="4">Entry Criteria</th></tr>
<tr>
<td rowspan="2">HA1

Any unplanned release of gaseous radioactivity to the environment that exceeds 200 times the ODCM Control limit for 15 minutes or greater

Applicable Modes:
1 | 2 | 3 | 4 | 5 | D</td>
<td>A valid reading greater than 200 times the HIGH alarm setpoint on <u>one or more</u> of the following plant gaseous effluent monitors:*

• PLANT VENT GAS 1D17-K786
• OG VENT PIPE GAS 1D17-K836
• TB/HB VENT GAS 1D17-K856
• PLANT VENT GAS 2D17-K786</td>
<td rowspan="2">A valid offscale high reading on <u>one or more</u> of the following plant gaseous effluent monitors:*

• PLANT VENT GAS 1D17-K786
• OG VENT PIPE GAS 1D17-K836
• TB/HB VENT GAS 1D17-K856
• PLANT VENT GAS 2D17-K786</td>
<td>Routine or as required sample analysis indicates a release rate greater than 200 times ODCM Control 3.11.2.1 limits.</td>
<td rowspan="2">Portable survey instruments indicate radiation levels of greater than 10 mRem/hr at the Site Boundary for greater than or equal to 15 minutes.</td>
</tr>
<tr>
<td>The valid reading lasts for greater than or equal to 15 minutes.</td>
<td>The release lasts for greater than or equal to 15 minutes.</td>
</tr>
</table>

<u>HA1</u>

ALERT

* Perform an emergency dose calculation (i.e., CADAP run) within 15 minutes to determine if the Site Area Emergency entry criteria is met.

<u>NOTE 1</u>

ODCM [coolant activity] methodology is used to determine the threshold for this EAL. The Site Area Emergency threshold for EAL, HS1, is established using a clad damage source term. As a result, if the EAL threshold for the Alert is met AND there is clad damage, then the Site Area Emergency thresholds may be exceeded.

<u>NOTE 2</u>

It is <u>NOT</u> intended that the release be averaged over 15 minutes. Rather, the Emergency Coordinator should declare the event as soon as it is determined that the release will exceed 200 times the ODCM Control 3.11.2.1 limit for greater than 15 minutes.

<table>
<tr><th>Initiating Conditions</th><th colspan="3">Entry Criteria</th><th></th></tr>
<tr><td rowspan="2">HA2

Any unplanned release of liquid radioactivity to the environment that exceeds 200 times the ODCM Control limit for 15 minutes or greater

Applicable Modes:
1 | 2 | 3 | 4 | 5 | D</td><td>A valid reading greater than 1.2E5 cpm above background for <u>one or more</u> of the following liquid process monitors:

• EMERGENCY SERVICE WATER LOOP A D17-K604

• EMERGENCY SERVICE WATER LOOP B D17-K605</td><td>A valid Reading greater than 2000 times the HIGH-HIGH alarm setpoint on NCC HX and RW EFF Common Process Radiation Recorder 0D17R0170.</td><td>Routine or as required sample analysis indicates a release rate greater than 200 times ODCM Control 3.11.1.1 limits.</td><td rowspan="2"><u>HA2</u>

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<tr><td>The reading lasts greater than or equal to 15 minutes.</td><td>Release <u>CANNOT</u> be terminated within 15 minutes of exceeding NCC HX and RW EFF Common Process Radiation Recorder HIGH-HIGH alarm setpoints.</td><td>The release lasts for greater than or equal to 15 minutes.</td></tr>
</table>

<u>NOTE</u>

It is <u>NOT</u> intended that the release be averaged over 15 minutes. Rather, the Emergency Coordinator should declare the event as soon as it is determined that the release will exceed 200 times the ODCM Control 3.11.1.1 limit for greater than 15 minutes.

<table>
<tr><th>Initiating Conditions</th><th colspan="3">Entry Criteria</th></tr>
<tr>
<td rowspan="2">HS1

Site Boundary dose resulting from an actual or imminent release of gaseous radioactivity that exceeds 100 mRem TEDE dose OR 500 mRem CDE Child Thyroid dose for the actual or projected duration of the release

Applicable Modes:
1 | 2 | 3 | 4 | 5 | D</td>
<td>A valid indication greater than the listed reading for one or more of the following plant gaseous effluent monitors:

• PLANT VENT GAS 1D19-K300 3.8E-1 µCi/cc
• OG VENT GAS 1D19-K400 2.2E0 µCi/cc
• TB/HB VENT GAS 1D17-K856 1.6E4 cpm
• PLANT VENT GAS 2D19-K300 6.0E-1 µCi/cc</td>
<td rowspan="2">Emergency dose calculations, using actual meteorology indicate that one or more of the following are met at the Site Boundary:

• Greater than 100 mRem TEDE
• Greater than 500 mRem CDE Child Thyroid</td>
<td>Field survey results indicate that one or more of the following have been met at the Site Boundary:

• Greater than 100 mRem/hr Whole Body
• Greater than 500 mRem CDE Child Thyroid</td>
</tr>
<tr>
<td>Emergency dose calculations CANNOT confirm, within 15 minutes of exceeding limit, that levels at the Site Boundary are less than 100 mRem TEDE and 500 mRem CDE Child Thyroid dose using actual meteorology.</td>
<td>Dose rates are expected to continue for greater than or equal to 1 hour.</td>
</tr>
</table>

HS1

SITE AREA EMERGENCY

<table>
<tr><th>Initiating Conditions</th><th colspan="3">Entry Criteria</th><th></th></tr>
<tr><td rowspan="2">HG1

Site Boundary dose resulting from an actual or imminent release of gaseous radioactivity that exceeds 1000 mRem TEDE dose OR 5000 mRem CDE Child Thyroid dose for the actual or projected duration of the release

Applicable Modes:
1 | 2 | 3 | 4 | 5 | D</td><td>A valid indication greater than the reading listed for <u>one or more</u> of the following plant gaseous effluent monitors:

• PLANT VENT GAS 1D19-K300 3.8E0 µCi/cc
• OG VENT GAS 1D19-K400 2.2E1 µCi/cc
• TB/HB VENT GAS 1D17-K856 1.6E5 cpm
• PLANT VENT GAS 2D19-K300 6.0E0 µCi/cc</td><td rowspan="2">Emergency dose calculations, using actual meteorology indicate that <u>one or more</u> of the following are met at the Site Boundary:

• Greater than 1000 mRem TEDE
• Greater than 5000 mRem CDE Child Thyroid</td><td>Field survey results indicate that <u>one or more</u> of the following have been met at the Site Boundary:

• Greater than 1000 mRem/hr Whole Body
• Greater than 5000 mRem CDE Child Thyroid</td><td rowspan="2"><u>HG1</u>

GENERAL EMERGENCY</td></tr>
<tr><td>Emergency dose calculations <u>CANNOT</u> confirm, within 15 minutes of exceeding above limit, that levels at the Site Boundary are less than 1000 mRem TEDE and 5000 mRem CDE Child Thyroid dose using actual meteorology.</td><td>Dose rates are expected to continue for greater than or equal to 1 hour.</td></tr>
</table>

<u>NOTE</u>

Exceeding the entry criteria for HG1 may require the initiation of an RPV emergency depressurization per <EOP-05>. Ensure Shift Manager is notified immediately whenever the above entry criteria for a General Emergency is met.

Category I: Control Room Evacuation

<table>
<tr><th colspan="6">Initiating Conditions</th><th>Entry Criteria</th><th></th></tr>
<tr><td colspan="6">IA1

Control Room evacuation has been initiated

Applicable Modes:</td><td rowspan="2">Entry into <ONI-C61>.</td><td rowspan="2"><u>IA1</u>

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T</td></tr>
<tr><td>1</td><td>2</td><td>3</td><td>4</td><td>5</td><td>D</td></tr>
</table>

NOTE

An inability to establish plant control from outside the Control Room will escalate this event to a Site Area Emergency per IS1.

<table>
<tr><th colspan="6">Initiating Conditions</th><th>Entry Criteria</th></tr>
<tr><td colspan="6" rowspan="2">IS1

Control Room evacuation has been initiated, AND plant control <u>CANNOT</u> be established within 15 minutes.

Applicable Modes:</td><td>Entry into <ONI-C61>.</td></tr>
<tr><td>Within 15 minutes of entry into <ONI-C61>, Operator(s) located at the remote shutdown controls <u>CANNOT</u> establish control of <u>one or more</u> of the following parameters per <IOI-11>:

• RPV level
• RPV pressure
• Suppression Pool temperature
• Reactor power
• Decay heat removal, if required</td></tr>
<tr><td>1</td><td>2</td><td>3</td><td>4</td><td>5</td><td>D</td><td></td></tr>
</table>

<u>**IS1**</u>

SITE AREA EMERGENCY

<u>NOTE</u>

A maximum 15 minute time frame for the physical transfer of control of "required" systems was established by <NUMARC/NESP-007>. Control at the Remote Shutdown Areas is accomplished by the repositioning of control transfer switches per <IOI-11>. Control is assumed unless indication of the absence of control is present.

Category J: Loss of Annunciators or Indication

<table>
<tr><th colspan="6">Initiating Conditions</th><th colspan="2">Entry Criteria</th></tr>
<tr><td colspan="6" rowspan="2">JU1

Loss of most annunciators or indication in the Control Room for greater than 15 minutes

Applicable Modes:</td><td>Unplanned loss of most Control Room annunciators for greater than 15 minutes.</td><td>Unplanned loss of most Control Room indication for greater than 15 minutes.</td></tr>
<tr><td colspan="2" rowspan="2">In the Shift Manager’s opinion, increased surveillance is warranted to safely operate the plant.</td></tr>
<tr><td>1</td><td>2</td><td>3</td><td></td><td></td><td></td></tr>
</table>

JU1

UNUSUAL EVENT

NOTE

Quantification of “most” is left to the Shift Manager. It is NOT intended that plant personnel perform a detailed count of the instrumentation lost, but rather make a judgment call with approximately 75% being the threshold.

<table>
<tr><th>Initiating Conditions</th><th colspan="2">Entry Criteria</th><th></th></tr>
<tr><td rowspan="3">JA1

Loss of most annunciators or indication in the Control Room with <u>either</u>: (1) a significant transient in progress, OR (2) compensatory indicators are <u>NOT</u> available.

Applicable Modes:
1 | 2 | 3</td><td>Unplanned loss of most Control Room annunciators for greater than 15 minutes.</td><td>Unplanned loss of most Control Room indication for greater than 15 minutes.</td><td rowspan="3"><u>JA1</u>

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T</td></tr>
<tr><td colspan="2">In the Shift Manager’s opinion, increased surveillance is warranted to safely operate the plant.</td></tr>
<tr><td>A significant plant transient is in progress.</td><td>Compensatory indications, i.e., Integrated Computer System (ICS), are <u>NOT</u> available.</td></tr>
</table>

<u>NOTES</u>

- Quantification of “most” is left to the Shift Manager. It is <u>NOT</u> intended that plant personnel perform a detailed count of the instrumentation lost, but rather make a judgment call with approximately 75% being the threshold.
- A “significant transient” includes response to automatic OR manually initiated functions such as scrams, runbacks involving greater than 25% thermal power change, ECCS injection, or thermal power oscillations of 10% or greater.

<table>
<tr><th>Initiating Conditions</th><th colspan="2">Entry Criteria</th></tr>
<tr><td rowspan="4">JS1

Inability to monitor a significant transient in progress

Applicable Modes:
1 | 2 | 3</td><td>Loss of most Control Room annunciators.</td><td>Loss of most Control Room indication.</td></tr>
<tr><td colspan="2">Compensatory indicators, i.e., Integrated Computer System (ICS), are <u>NOT</u> available.</td></tr>
<tr><td colspan="2">A significant transient is in progress.</td></tr>
<tr><td colspan="2">Sufficient indication is <u>NOT</u> available to directly monitor plant critical safety parameters for EOPs entered due to the transient.</td></tr>
</table>

<u>**JS1**</u>

SITE AREA EMERGENCY

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Category K: Loss of Communications

<table>
<tr><th colspan="6">Initiating Conditions</th><th>Entry Criteria</th><th></th></tr>
<tr><td colspan="6" rowspan="2">KU1

Loss of onsite OR in-plant communications capabilities

Applicable Modes:</td><td>Loss of <u>all</u> five Plant Public Address System channels.</td><td rowspan="3"><u>KU1</u>

UNUSUAL EVENT</td></tr>
<tr><td rowspan="2">Loss of <u>all</u> of the following Plant Radio System channels:

• Channel 1
• Channel 2
• Channel 3</td></tr>
<tr><td>1</td><td>2</td><td>3</td><td>4</td><td>5</td><td>D</td></tr>
</table>

<table>
<tr><th colspan="6">Initiating Conditions</th><th>Entry Criteria</th><th></th></tr>
<tr><td colspan="6" rowspan="2">KU2

Significant degradation of offsite communications capabilities

Applicable Modes:</td><td>Loss of the State and County Notification Circuit (5-way) reported to the Control Room.</td><td rowspan="3"><u>KU2</u>

U N U S U A L

E V E N T</td></tr>
<tr><td rowspan="2">Loss of offsite long distance calling capability on <u>three or more</u> of the following systems circuits for greater than 15 minutes:

• Control Room private (259-) lines
• Private Branch Exchange, Service Building (“5000”) Switch
• Private Branch Exchange, Warehouse Building (“6000”) Switch
• Company Off-Premise Exchange</td></tr>
<tr><td>1</td><td>2</td><td>3</td><td>4</td><td>5</td><td>D</td></tr>
</table>

<u>NOTES</u>

- A loss of the “5 Way” Circuit refers to the inability to contact <u>one or more</u> of the four offsite contacts: the State of Ohio, and the counties of Ashtabula, Geauga, and Lake. Testing to determine “5-Way” operability or to initiate circuit restoration actions are governed under <PSI-0007>.

- Direct (259-) off-site calling capability from the Control Room via private lines refer to: autodialer at the US console, and private (259-) lines on the superphones and line at the SAS console.

Category L: Natural or Destructive Phenomena
<P00067>

<table>
<tr><th>Initiating Conditions</th><th colspan="6">Entry Criteria</th></tr>
<tr><td rowspan="2">LU1

Natural OR destructive phenomena affecting the Protected Area boundary

Applicable Modes:
1 | 2 | 3 | 4 | 5 | D</td><td colspan="2">Control Room receives report from plant personnel who felt an earthquake.</td><td rowspan="2">Report by plant personnel confirming <u>either</u> of the following within the Protected Area boundary:
• tornado strike
• plane or train crash</td><td rowspan="2">High sustained winds greater than 70 mph for equal to or greater than
15 minutes.</td><td colspan="2">Indications in the Control Room of a Main Turbine Trip or failure</td></tr>
<tr><td>WHITE event indicator light on local Seismic Monitoring Panel 0H51-P021.</td><td>AMBER light(s) on Seismic Monitoring Panel 0H13-P969.</td><td>Turbine casing penetration.</td><td>Catastrophic damage to generator seals.</td></tr>
</table>

<u>LU1</u>

UNUSUAL EVENT

<table>
<tr><th>Initiating Conditions</th><th colspan="5">Entry Criteria</th></tr>
<tr>
<td rowspan="2">LA1

Natural OR destructive phenomena affecting Safe Shutdown Buildings

Applicable Modes:
1 | 2 | 3 | 4 | 5 | D</td>
<td>Control Room receives report from plant personnel who felt an earthquake.</td>
<td rowspan="2">Report of visible damage to any Safe Shutdown Building caused by <u>any</u> of the following:
• tornado
• aircraft, barge or train crash
• other natural or destructive phenomena</td>
<td rowspan="2">Sustained high winds with a velocity greater than 90 mph for 15 minutes or longer.</td>
<td rowspan="2">Greater than <EOP-03> Maximum Safe Operating Value for Area Water Level (internal flooding)</td>
<td>Report by plant personnel confirming a turbine failure which results in penetration of the turbine casing.</td>
</tr>
<tr>
<td><u>Either</u> of the following indications present:
• YELLOW seismic switch indicator light on local Seismic Monitoring Panel OH51-P021.
• RED light on Seismic Monitoring Panel OH13-P969.</td>
<td>Missile generated from the turbine failure resulting in <u>either</u>:
• damage to Safe Shutdown equipment
• penetration of a Safe Shutdown Bldg.</td>
</tr>
</table>

LA1

ALERT

Category M: Release of Toxic or Flammable Gases

<table>
<tr><th colspan="6">Initiating Conditions</th><th colspan="2">Entry Criteria</th></tr>
<tr><td colspan="6" rowspan="2">MU1

Release of toxic OR flammable gases affecting the Protected Area boundary deemed detrimental to the safe operation of the plant

Applicable Modes:</td><td>Toxic or flammable gas concentrations detected within the Protected Area.</td><td rowspan="3">Report by local, county, or State officials for a potential evacuation of site personnel based on an offsite event.</td></tr>
<tr><td rowspan="2">Normal operation of the plant is impeded due to access restrictions.</td></tr>
<tr><td>1</td><td>2</td><td>3</td><td>4</td><td>5</td><td>D</td></tr>
</table>

MU1

U N U S U A L   E V E N T

NOTE

A toxic or flammable gas release is considered to be impeding normal operations due to access restrictions if it is of sufficient magnitude that access to areas normally accessed to plant operator rounds is restricted. It also includes releases where access to these areas is possible only through the use of protective equipment, such as respirators since this limits the operators visibility and mobility thereby affecting “normal” plant operations.

<table>
<tr><th colspan="6">Initiating Conditions</th><th colspan="3">Entry Criteria</th><th></th></tr>
<tr><td colspan="6" rowspan="2">MA1

Release of toxic OR flammable gases within a Safe Shutdown Building which jeopardizes operation of systems required to maintain safe operations OR to establish or maintain COLD SHUTDOWN

Applicable Modes:</td><td colspan="3">Entry of toxic or flammable gases into Safe Shutdown Buildings or Areas.</td><td rowspan="3"><u>MA1</u>

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T</td></tr>
<tr><td rowspan="2">Toxic gas in concentrations considered life-threatening</td><td rowspan="2">Flammable gas estimated or determined to be in explosive concentrations</td><td rowspan="2">Plant personnel <u>NOT</u> able to perform actions necessary to establish and maintain Mode 4 while utilizing appropriate protective equipment.</td></tr>
<tr><td>1</td><td>2</td><td>3</td><td>4</td><td>5</td><td>D</td></tr>
</table>

<u>NOTE</u>

This IC addresses increased toxic or flammable gas levels that impede necessary access to operating stations or other areas containing equipment that must be operated manually in order to maintain safe operation or perform a safe shutdown. It is this impaired ability to operate the plant that results in the actual or potential substantial degradation of the level of safety of the plant.

Category N: Security Events

<table>
<tr><th>Initiating Conditions</th><th colspan="3">Entry Criteria</th><th></th></tr>
<tr><td>NU1

Confirmed SECURITY CONDITION OR threat which indicates a potential degradation in the level of safety of the plant

Applicable Modes:
1 | 2 | 3 | 4 | 5 | D</td><td>SECURITY CONDITION that does NOT involve a HOSTILE ACTION as reported by the Security Shift Supervisor.</td><td>A credible site specific security threat notification.</td><td>A validated notification from NRC providing information of an aircraft threat.</td><td><u>NU1</u>

UNUSUAL EVENT</td></tr>
</table>

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<table>
<tr><th colspan="6">Initiating Conditions</th><th>Entry Criteria</th></tr>
<tr><td colspan="6" align="center">NA2

Notification of an Airborne Attack Threat

Applicable Modes:</td><td rowspan="2">A validated notification from the NRC of an airliner attack threat less than 30 minutes away.</td></tr>
<tr><td>1</td><td>2</td><td>3</td><td>4</td><td>5</td><td>D</td></tr>
</table>

<u>**NA2**</u>

A
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<u>NOTE</u>

Airliner is meant to be a large aircraft with the potential for causing significant damage to the plant. The status and size of the plane may be provided by NORAD through the NRC.

Initiating Conditions						Entry Criteria	
NA3 **HOSTILE ACTION within the Owner Controlled Area** **Applicable Modes**:						A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervisor.	**NA3** **A L E R T**
1	**2**	**3**	**4**	**5**	**D**		

NOTES

- HOSTILE ACTION is defined as an act toward a nuclear power plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included.
- HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the owner controlled area).

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Initiating Conditions	Entry Criteria
NS2 **HOSTILE ACTION within the PROTECTED AREA** **Applicable Modes**:	A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor.

1	2	3	4	5	D

NS2

SITE AREA EMERGENCY

NOTES

- HOSTILE ACTION is defined as an act toward a nuclear power plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included.

- HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the owner controlled area).

Initiating Conditions	Entry Criteria	
NG1 **HOSTILE ACTION resulting in loss of physical control of the facility** **Applicable Modes**: **1 2 3 4 5 D**	A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions.	A HOSTILE ACTION has caused failure of spent fuel cooling systems and imminent fuel damage is likely for freshly off-loaded reactor core in the pool.

NG1

GENERAL EMERGENCY

NOTES

- HOSTILE ACTION is defined as an act toward a nuclear power plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included.
- HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the owner controlled area).

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Category O: Emergency Coordinator's Judgment

<table>
<tr><th colspan="6">Initiating Conditions</th><th>Entry Criteria</th><th></th></tr>
<tr><td colspan="6">OU1

Other conditions existing, which in the judgment of the Emergency Coordinator, warrant declaration of an Unusual Event

Applicable Modes:</td><td rowspan="2">Events are in process or have occurred which indicate a potential degradation of the level of safety of the plant.</td><td rowspan="2">OU1

UNUSUAL EVENT</td></tr>
<tr><td>1</td><td>2</td><td>3</td><td>4</td><td>5</td><td>D</td></tr>
</table>

NOTE

For those cases where the degradation in the level of safety of the plant is tied to equipment or system malfunctions, the decision that the component is degraded should be based upon its functionality and NOT its operability.

<table>
<tr><th colspan="6">Initiating Conditions</th><th>Entry Criteria</th></tr>
<tr><td colspan="6">OA1

Other conditions existing, which in the judgment of the Emergency Coordinator, warrant declaration of an Alert

Applicable Modes:</td><td rowspan="2">Events are in progress or have occurred which indicate an actual or potential degradation of systems needed for the protection of the public and which warrant increased monitoring of plant functions.</td></tr>
<tr><td>1</td><td>2</td><td>3</td><td>4</td><td>5</td><td>D</td></tr>
</table>

<u>OA1</u>

A L E R T

<u>NOTE</u>

This IC is intended to address unanticipated conditions <u>NOT</u> addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Coordinator to fall under the Alert emergency class. This includes a determination by the Emergency Coordinator that additional assistance similar to that provided by the TSC and OSC staffs, <u>including a transfer of the Emergency Coordinator responsibilities to the TSC</u>, is necessary for the event to be effectively mitigated. Transfer of Emergency Coordinator duties for classification, offsite notifications and PAR decisions, is used as an initiator since an event significant enough to warrant transfer of command and control is a substantial reduction in the level of safety of the plant. It is not intended to declare an Alert if Emergency Coordinator responsibilities are transferred out of the Control Room for convenience.

Initiating Conditions						Entry Criteria
OS1 **Other conditions existing, which in the judgment of the Emergency Coordinator, warrant declaration of a Site Area Emergency** **Applicable Modes**:						Other conditions exist which indicate an actual or likely major failure of plant functions needed for protection of the public.
1	**2**	**3**	**4**	**5**	**D**	

OS1

SITE AREA EMERGENCY

<table>
<tr><th colspan="6">Initiating Conditions</th><th colspan="3">Entry Criteria</th></tr>
<tr><td colspan="6">OG1

Other conditions existing, which in the judgment of the Emergency Coordinator, warrant declaration of a General Emergency

Applicable Modes:</td><td rowspan="2">Other conditions exist which indicate an actual or imminent substantial core degradation with the potential loss of Containment integrity.</td><td rowspan="2">Potential for an uncontrolled release which can reasonably be expected to be greater than 1 Rem TEDE at the Site Boundary.</td><td rowspan="2">Potential for an uncontrolled release which can reasonably be expected to be greater than 5 Rem CDE Child Thyroid at the Site Boundary.</td></tr>
<tr><td>1</td><td>2</td><td>3</td><td>4</td><td>5</td><td>D</td></tr>
</table>

<u>OG1</u>

G E N E R A L

E M E R G E N C Y

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

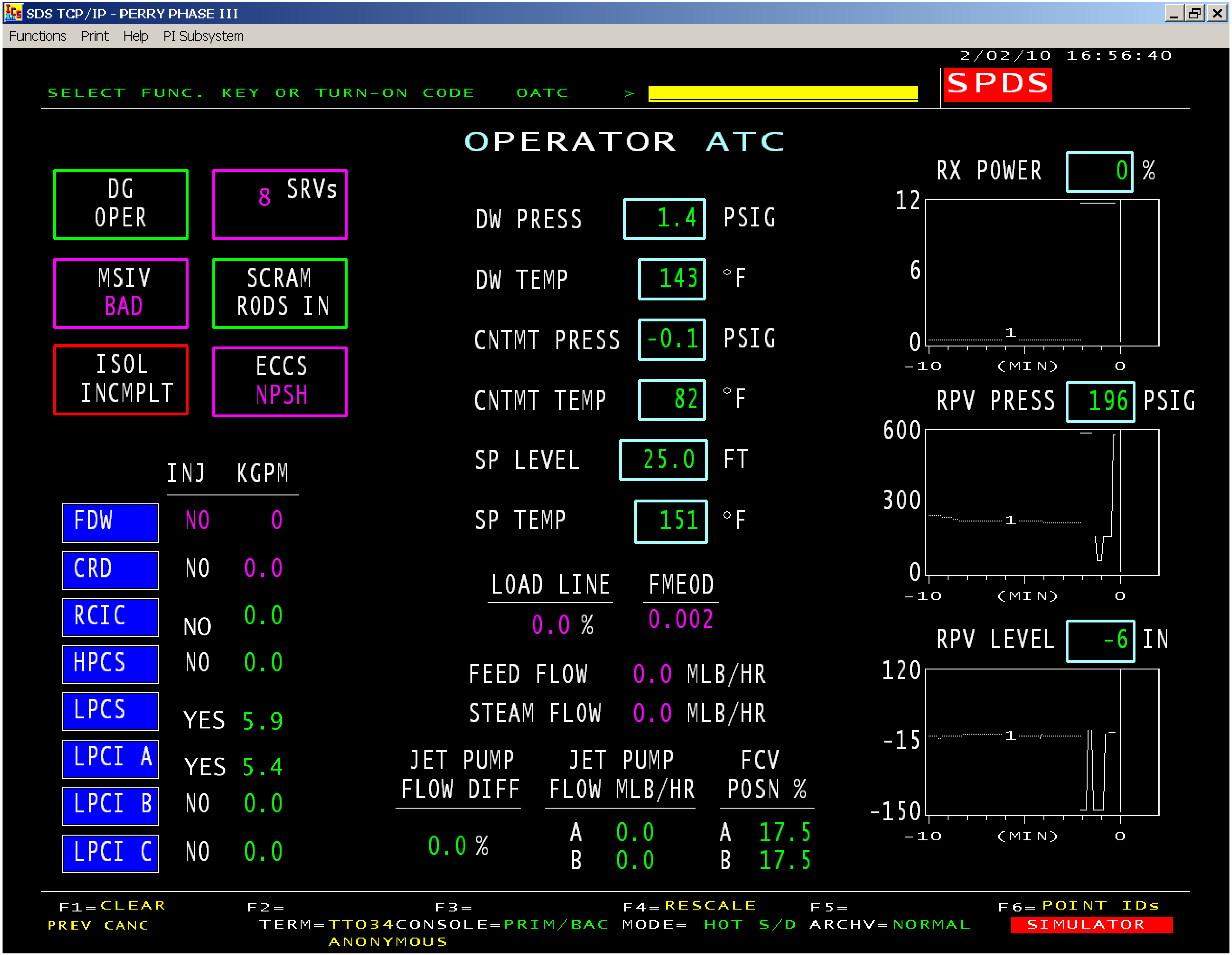
ACTIONS

-------------------------------------NOTE-------------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip. OR A.2 Place associated trip system in trip.	12 hours 12 hours

SURVEILLANCE REQUIREMENTS

-------------------------------------NOTES-------------------------------------
1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.
2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.



SRO Question 13

SRO Question 17

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip.	12 hours
	<u>OR</u>	
	A.2 Place associated trip system in trip.	12 hours
B. One or more Functions with one or more required channels inoperable in both trip systems.	B.1 Place channel in one trip system in trip.	6 hours
	<u>OR</u>	
	B.2 Place one trip system in trip.	6 hours
C. One or more Functions with RPS trip capability not maintained.	C.1 Restore RPS trip capability.	1 hour

(continued)

SRO Question 17

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1	Be in MODE 3.	12 hours
I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	I.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SRO Question 17

SURVEILLANCE REQUIREMENTS

-----------------------------------NOTES-----------------------------------

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.
2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.1.4	------------------NOTE------------------ Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -- Perform CHANNEL FUNCTIONAL TEST.	7 days
SR 3.3.1.1.6	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to withdrawing SRMs from the fully inserted position
SR 3.3.1.1.7	------------------NOTE------------------ Only required to be met during entry into MODE 2 from MODE 1. -- Verify the IRM and APRM channels overlap.	7 days

(continued)

SRO Question 17

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.13	-----NOTES----- 1. Neutron detectors are excluded. 2. For IRMs, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. ----- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.1.15	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.1.19	Perform CHANNEL FUNCTIONAL TEST	31 days

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux - High	2	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 122/125 divisions of full scale
	5(a)	3	I	SR 3.3.1.1.1 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.19	≤ 122/125 divisions of full scale
b. Inop	2	3	H	SR 3.3.1.1.4 SR 3.3.1.1.15	NA
	5(a)	3	I	SR 3.3.1.1.15 SR 3.3.1.1.19	NA

SRO Question 18

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip. OR A.2 Place associated trip system in trip.	12 hours 12 hours
B. One or more Functions with one or more required channels inoperable in both trip systems.	B.1 Place channel in one trip system in trip. OR B.2 Place one trip system in trip.	6 hours 6 hours
C. One or more Functions with RPS trip capability not maintained.	C.1 Restore RPS trip capability.	1 hour

(continued)

SRO Question 18

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1	Be in MODE 3.	12 hours

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux - High	2	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.15	
b. Inop	2	3	H	SR 3.3.1.1.4 SR 3.3.1.1.15	

PMT Data

1. Suction Valve Post Maintenance Tests: 1E51-F010 (EO, EC, ST), 1E51-F031 (EO, EC, ST).

   a. Close 1E51-F010, RCIC PUMP CST SUCTION VALVE, measure stroke time and record the following:

      1) Exercise Closed Results:
         ☒ SAT ☐ UNSAT

      2) Stroke Time - Closed direction.

         __24.8___ seconds
         Reference Value = 25.0 seconds
         Acceptable Range = 21.3 to 28.7 seconds
         ☒ SAT ☐ UNSAT

   b. Open 1E51-F031, RCIC PUMP SUPR PL SUCT ISOL, measure stroke time and record the following:

      1) Exercise Open Results:
         ☒ SAT ☐ UNSAT

      2) Stroke Time - Open direction.

         __19.5___ seconds
         Reference Value = 16.9 seconds
         Acceptable Range = 14.4 to 19.4 seconds
         ☐ SAT ☒ UNSAT

   c. Close 1E51-F031, RCIC PUMP SUPR PL SUCT ISOL, measure stroke time, and record the following:

      1) Exercise Closed Results:
         ☒ SAT ☐ UNSAT

      2) Stroke Time - Closed direction.

         __18.9__ seconds
         Reference Value = 17.0 seconds
         Acceptable Range = 14.5 to 19.5 seconds
         ☒ SAT ☐ UNSAT

PMT Data

d. Open 1E51-F010, RCIC PUMP CST SUCTION VALVE, measure stroke time, and record the following:

1) Exercise Open Results:
☒ SAT ☐ UNSAT

2) Stroke Time - Open direction.

__23.8__ seconds
Reference Value = 24.2 seconds
Acceptable Range = 20.6 to 27.8 seconds
☒ SAT ☐ UNSAT

SRO Question 20

3.3 INSTRUMENTATION

3.3.6.1 Primary Containment and Drywell Isolation Instrumentation

LCO 3.3.6.1 The primary containment and drywell isolation instrumentation for each Function in Table 3.3.6.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6.1-1.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip.	12 hours for Functions 2.b, 5.b, and 5.d AND 24 hours for Functions other than Functions 2.b, 5.b, and 5.d
B. One or more automatic Functions with isolation capability not maintained.	B.1 Restore isolation capability.	1 hour

(continued)

SRO Question 20

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Enter the Condition referenced in Table 3.3.6.1-1 for the channel.	Immediately
D. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	D.1 Isolate associated main steam line (MSL).	12 hours
	<u>OR</u>	
	D.2.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	D.2.2 Be in MODE 4.	36 hours

SRO Question 20

Table 3.3.6.1-1 (page 1 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS
1. **Main Steam Line Isolation**				
b. **Main Steam Line Pressure - Low**	1	2	E	SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6

(continued)

SRO Question 20

Table 3.3.6.1-1 (page 2 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS
2. Primary Containment and Drywell Isolation				
b. Drywell Pressure — High	1,2,3	$2^{(b)}$	H	

(continued)

(b) Required to initiate the drywell isolation function.

SRO Question 20

Table 3.3.6.1-1 (page 6 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS
5. RHR System Isolation				
b. Reactor Vessel Water Level	1,2(g),3(g)	2	F	
	2(e),3(e),4,5	2(f)	J	
d. Drywell Pressure — High	1,2,3	2	F	

(e) With reactor vessel steam dome pressure less than the RHR cut in permissive pressure.

(f) Only one trip system required in MODES 4 and 5 with RHR Shutdown Cooling System integrity maintained.

(g) With reactor vessel steam dome pressure greater than or equal to the RHR cut in permissive pressure.