Answer Sheet

Exam Title:ILT SRO NRC Written Exam answer sheet

Name ____________________ Date __________

1	a b ● d	26	a b ● d	51	a ● c d	76	a b ● d
2	● b c d	27	● b c d	52	● b c d	77	● b c d
3	a b c ●	28	a b c ●	53	a b c ●	78	● b c d
4	● b c d	29	a b ● d	54	a ● c d	79	a b c ●
5	● b c d	30	a b ● d	55	a ● c d	80	a b c ●
6	a b ● d	31	● b c d	56	● b c d	81	a ● c d
7	a b c ●	32	● b c d	57	a b c ●	82	a b ● d
8	a ● c d	33	a ● c d	58	a b c ●	83	● b c d
9	a b ● d	34	a ● c d	59	a b c ●	84	a ● c d
10	a b c ●	35	a b ● d	60	a ● c d	85	a b c ●
11	a b c ●	36	a ● c d	61	a b c ●	86	a ● c d
12	● b c d	37	● b c d	62	a b c ●	87	a b ● d
13	a b ● d	38	a b ● d	63	a b ● d	88	● b c d
14	a b ● d	39	● b c d	64	a b c ●	89	● b c d
15	a ● c d	40	a ● c d	65	a ● c d	90	a ● c d
16	a b ● d	41	a b c ●	66	a b ● d	91	a ● c d
17	a b ● d	42	a b c ●	67	● b c d	92	a b c ●
18	● b c d	43	a b ● d	68	a b ● d	93	a ● c d
19	a b ● d	44	a b ● d	69	a b c ●	94	a b c ●
20	a b c ●	45	● b c d	70	● b c d	95	a b ● d
21	a b ● d	46	a b c ●	71	a b ● d	96	a b c ●
22	a ● c d	47	a ● c d	72	a b c ●	97	● b c d
23	a ● c d	48	a b ● d	73	a b c ●	98	a b ● d
24	a b c ●	49	● b c d	74	a b c ●	99	a b ● d
25	● b c d	50	a b c ●	75	a ● c d	100	a ● c d

Examination Outline Cross-Reference	Level	RO
263000 (SF6 DC) DC Electrical Distribution	Tier #	2
	Group #	1
Knowledge of the effect that a loss or malfunction of the D.C. ELECTRICAL DISTRIBUTION will have on following:	K/A #	K3.03
	Rating	3.4

K3.03 Systems with D.C. components (i.e. valves, motors, solenoids, etc.)

Question 1

The plant is at 100% power.

The following indications are present in the control room.

- 125 VDC Bus1A is reading 0 VDC
- Annunciator C-1/B-2, 125V DC BUS 1A Ground is in alarm

If the plant were to scram in this condition, 4160 V Bus 1F would be ___(1)___.

The ___(2)___ MSIV DC PCIS Group 1 isolation solenoids would be deenergized.

A. (1) energized
(2) inboard

B. (1) energized
(2) outboard

C. (1) deenergized
(2) inboard

D. (1) deenergized
(2) outboard

Answer: C

Explanation:
125 VDC 1A supplies the control power for 4160 V Bus 1F breakers. Losing the control power would prevent the automatic swap of power sources following a SCRAM from the NSST to the SSST, ESST, or the associated DG. The SCRAM would cause a loss of power on the normal power supply (Normal Transformer) through 4160 V Bus 1A. The loss of control power would prevent the 4160 V 1F bus from swapping to a different power supply. DG-1 would also be unable to start with a loss of DC 1A.

The Inboard MSIV DC PCIS Gr. 1 isolation solenoids are powered by 125 VDC 1A.

A is wrong. Part 1 is plausible if the applicant forgets that DG-1 is unable to start or that 4160 V Bus 1F will not transfer to the Startup or Emergency transformer. Part 2 is correct for the reasons given in explanation.

B is wrong. Part 1 is correct for the reasons given in explanation. Part 2 is plausible because the outboard MSIV DC PCIS Gr. 1 solenoids are energized

C is correct .

D is wrong. Part 1 for the reasons stated in the A distractor and Part 2 for the reasons stated in distractor B.

Technical References:
Procedure 2.2A_125DC.DIV1, revision 7, page 14

References to be provided to applicants during exam:
None.

Learning Objective: COR002-07-02, DC Electrical Distribution, Revision 35, Enabling objective 8.c

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference		Level	RO
205000 (SF4 SCS) Shutdown Cooling		Tier #	2
		Group #	1
Ability to (a) predict the impacts of the following on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:		K/A #	A2.08
		Rating	3.3
A2.08 Loss of heat exchanger cooling			

Question 2

Given the following:

- The plant is in Mode 4
- RHR loop A is operating in shutdown cooling mode
- RHR Division 1 Service Water is in service
- RHR Service Water is then lost to the A RHR heat exchanger
- The Control Room Supervisor directs Reactor Recirculation pump A started

Which of the below are correct in accordance with Abnormal Procedure 2.4SDC, Shutdown Cooling Abnormal?

A. (1) When reactor coolant temperature approaches 212ºF, close the reactor head vent valves.
   (2) Bypass RHR flow around RHR heat exchanger A until SW flow is restored.

B. (1) When reactor coolant temperature approaches 212ºF, close the reactor head vent valves.
   (2) Immediately place alternative decay heat removal systems in service until SW flow is restored.

C. (1) There would be a loss of circulation throughout the reactor core and thermal stratification would take place.
   (2) Bypass RHR flow around RHR heat exchanger A until SW flow is restored.

D. (1) There would be a loss of circulation throughout the reactor core and thermal stratification would take place.
   (2) Immediately place alternative decay heat removal systems in service until SW flow is restored.

Answer: A

Explanation:
A is correct because 1. At 212ºF the reactor would be in Mode 4 and the head vent valves would need to be closed; without the reactor open to atmospheric pressure, as the reactor coolant temperature increases, reactor pressure will increase and 2. Precaution and Limitation 2.14 and Procedure 2.4SDC, Attachment 1 direct operators - If RHR SW lost to in

service RHR HX, bypass RHR flow around HX until SW flow restored per Procedure 2.4SDC. This will prevent boiling water in tube side of HX which will cause a water hammer when SW flow is restored.℗[8]

B is wrong because 1. The RHR pump is still operating. Per lesson plan COR002-23-02-S-OPS, the SDC mode could fail causing a loss of cooling to the reactor during refueling operations. Due to decay heat production, reactor water and metal temperatures would rise. There would be a loss of circulation throughout the reactor core and thermal stratification would take place (if the RR pump was not operating). The upper portion of water in the reactor could heat up to the boiling point without the Control Room operator being aware of the situation. D is also wrong because part 2 is incorrect as Abnormal Procedure 2.4SDC, Shutdown Cooling Abnormal, Attachment 7 directs operators for contingencies to "Consider placing alternate decay heat removal systems in service per Procedure 2.1.20.2." This is plausible because it is guidance contained in the Shutdown Cooling Abnormal procedure.

C is wrong because 1. The CRS had RR pump A started. Per lesson plan COR002-23-02-S-OPS The SDC mode could fail causing a loss of cooling to the reactor during refueling operations. Due to decay heat production, reactor water and metal temperatures would rise. There would be a loss of circulation throughout the reactor core and thermal stratification would take place (if the RR pump was not operating). The upper portion of water in the reactor could heat up to the boiling point without the Control Room operator being aware of the situation. And is plausible because 2. Is correct.

D is wrong because part 2 is incorrect as Abnormal Procedure 2.4SDC, Shutdown Cooling Abnormal, Attachment 7 directs operators for contingencies to "Consider placing alternate decay heat removal systems in service per Procedure 2.1.20.2." This is plausible because it is guidance contained in the Shutdown Cooling Abnormal procedure. C is also plausible because part 1 is correct.

Technical References:
Document where the correct answer is found (Reference, Revision, Page number)

- System Operating Procedure 2.2.69.2, RHR System Shutdown Operations, Rev 106, Precaution and Limitation 2.14, page 3
- Abnormal Procedure 2.4SDC, Shutdown Cooling Abnormal, Rev 17, Attachment 1 and Attachment 7, pages 8 and 25
- Technical Specifications Bases 3.4.8, page 3.4-39

References to be provided to applicants during exam:
None.

Learning Objective:
COR002-23-02, Residual Heat Removal System, Revision 36, Enabling Objective 8.r

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

10CFR Part 55 Content: 55.41.5

Examination Outline Cross-Reference	Level	RO
215003 (SF7 IRM) Intermediate-Range Monitor	Tier #	2
	Group #	1
	K/A #	K6.04
Knowledge of the effect that a loss or malfunction of the following will have on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM:	Rating	3.0
K6.04 Detectors		

Question 3

Plant is starting up from an outage.

Mode switch is in START & HOT STBY

IRM F loses power.

What alarms do you expect to come in on Panel 9-5, Annunciator 9-5-1?

A. IRM DOWNSCALE and IRM RPS CH A UPSCALE TRIP OR INOP **ONLY**

B. IRM UPSCALE and IRM RPS CH B UPSCALE TRIP OR INOP **ONLY**

C. IRM UPSCALE, IRM DOWNSCALE, and IRM RPS CH A UPSCALE TRIP OR INOP **ONLY**

D. IRM UPSCALE, IRM DOWNSCALE, and IRM RPS CH B UPSCALE TRIP OR INOP **ONLY**

Answer: D

Explanation:
A is wrong because the upscale alarm comes in also and this is the wrong RPS channel.
B is wrong because the downscale alarm comes in also due to the detector reading 0, but this is the correct RPS channel.
C is wrong because this is the wrong RPS channel, but other alarms are correct.
D is correct because all 3 alarms will come in.

Meets the K/A because there are only a few malfunctions for an IRM detector. They can lose power, burn up, read high or read low. Since losing power is losing the detector, it meets the K/A.

Rev 1 changed the detector from E to F so D could remain the correct answer and have logical sequence of choices. Added complexity with determining the correct RPS channel, so changed to higher cognitive.

Technical References:
COR002-12-02, Intermediate Range Monitor, Revision 16, p. 15 and 22

References to be provided to applicants during exam:
None.

Learning Objective:
COR002-12-02, Intermediate Range Monitor, Revision 16, Enabling Objectives 7.b and 7.d

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
264000 (SF6 EGE) Emergency Generators (Diesel/Jet) EDG	Tier #	2
	Group #	1
	K/A #	K4.02
Knowledge of EMERGENCY GENERATORS (DIESEL/JET) design feature(s) and/or interlocks which provide for the following:	Rating	4.0

K4.02 Emergency generator trips (Emergency/LOCA)

Question 4

After receiving a start signal, an emergency diesel generator will trip if the diesel generator fails to reach ___(1)___ rpm within ___(2)___ seconds.

A. (1) 280
(2) 20

B. (1) 280
(2) 30

C. (1) 665
(2) 20

D. (1) 665
(2) 30

Answer: A

Explanation:
A is correct.
B is wrong. Part 1 is correct, Part 2 is plausible because the examinee can confuse override of the high vibrations trip bypass of 30 seconds with the incomplete sequence trip.
C is wrong. Part 1 is Plausible because 665 RPM is the overspeed trip for the diesel generator. Part 2 is correct. The diesel overspeed trip is 665 rpm and will also trip the diesel.
D is wrong for the reasons stated in distractors B and C.

Technical References:
COR002-08-02, Diesel Generators, Revision 37, page 30 to 34

References to be provided to applicants during exam:
None

Learning Objective: COR002-08-02, Diesel Generators, Revision 37, Enabling objective 9.b.

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

Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
259002 (SF2 RWLCS) Reactor Water Level Control	Tier #	2
	Group #	1
	K/A #	K1.15
Knowledge of the physical connections and/or cause-effect relationships between REACTOR WATER LEVEL CONTROL SYSTEM and the following:	Rating	3.2

K1.15 Recirculation flow control system

Question 5

Given the following initial conditions:

- Reactor power is 100 %
- Reactor Recirculation pump motor generator A scoop tube is locked out for corrective maintenance

One minute later A RFP trips which results in the following conditions:

- Reactor water level is 27 inches Narrow Range
- Total steam flow is 9.25 Mlbm/hr

With no operator action, both recirculation pumps will _____.

A. remain at current speed

B. run back to 22%

C. run back to 45%

D. run back until total steam flow is less than 9 Mlbm/hr

Answer: A

Explanation:
A is correct because with the A feed pump scoop tube locked out, the RVLCS logic will block the runback.

B is wrong because 22% is the speed controller's automatic setpoint for the speed limiter if the recirculation pump main discharge valve is not fully open or the feedwater flow is less than 20% of rated flow; and is plausible because it is an automatic setpoint of the reactor recirculation motor generator speed during abnormal conditions.

C is wrong because the reactor vessel level control system (RVLCS) logic will block the runback when an RRMG scoop tube is locked out or the plant is operating in single loop mode; and is plausible because it would be the correct answer with both scoop tubes in their normal position - the limiter logic would operate to runback reactor recirculation pump speed until within the capacity of the operating pumps or the limit of 45% speed demand is reached.

D is wrong because the RVLCS logic will block the runback when an RRMG scoop tube is locked out or the plant is operating in single loop mode; and is plausible because if both scoop tubes were in their normal position then this would be a partially correct answer as the limiter logic would operate to runback reactor recirculation pump speed until within the capacity of the operating pumps or the limit of 45% speed demand is reached.

Technical References:
UFSAR section VII, part 9.5.3, page VII-9-3 – VII-9-4

References to be provided to applicants during exam:
None.

Learning Objective:
COR002-32-02, Reactor Vessel Level Control, Revision 24, Enabling Objective 9.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41.5	

Examination Outline Cross-Reference		Level	RO
215003 (SF7 IRM) Intermediate-Range Monitor		Tier #	2
		Group #	1
		K/A #	K5.03
Knowledge of the operational implications of the following concepts as they apply to INTERMEDIATE RANGE MONITOR (IRM) SYSTEM:		Rating	3.0

K5.03 Changing detector position

Question 6

Given the following:

- A reactor startup is in progress

___(1)___ is used to withdraw each IRM, and the detector should be withdrawn ___(2)___.

A. (1) Startup Procedure, 2.1.1
(2) promptly to reduce detector coating burnup

B. (1) Startup Procedure, 2.1.1
(2) before placing the Mode switch in Run to ensure proper overlap with APRMs

C. (1) Intermediate Range Monitoring System, 4.1.2
(2) promptly to reduce detector coating burnup

D. (1) Intermediate Range Monitoring System, 4.1.2
(2) before placing the Mode switch in Run to ensure proper overlap with APRMs

Answer: C

Explanation:
C. is correct. Per references 2.1.1, 4.1.2, and training materials, AFTER mode switch is placed in run the IRMs are directed to be withdrawn IAW procedure 4.1.2 and the promptly aspect is discussed in the training material because it reduces the active coating burnup in the detector.

A is wrong because procedure 2.1.1 directs you to procedure 4.1.2 to withdraw the detectors (the steps are not in the startup procedure). Plausible if you don’t remember where the steps are located. The part 2 information is correct as stated in the training materials.

B is wrong because both parts are incorrect. Plausibility for part 1 is that you are in the startup procedure. Part 2 information is incorrect but plausible because you just went to RUN in the step above the withdraw IRMs step.

D is wrong because the part 2 information is incorrect (see above discussion).

Technical References:
Startup Procedure, 2.1.1, Revision 200, page 32.

Intermediate Range Monitoring System, 4.1.2, revision 24, page 5.
COR002-12-02, Intermediate Range Monitor, revision 16, page 28.

References to be provided to applicants during exam:
None.

Learning Objective:
COR002-12-02, Intermediate Range Monitor, Revision 16, Enabling objective 3.a

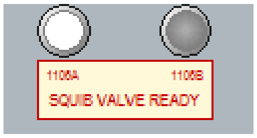
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41.2	

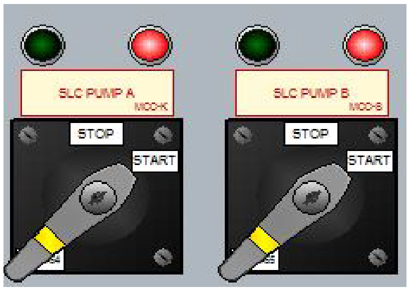
Examination Outline Cross-Reference	Level	RO
211000 (SF1 SLCS) Standby Liquid Control	Tier #	2
	Group #	1
Ability to manually operate and/or monitor in the control room:	K/A #	A4.08
	Rating	4.2

A4.08 System initiation

Question 7

After placing the Panel 9-5 control switches for SLC Pump A and SLC Pump B to START, you observe that the Squib Valve Ready DS-3A (1106A) light is **ON** and Squib Valve Ready DS-3B (1106B) is **OFF**.





Which of the following correctly describes the SLC system response to this condition?

Squib valve ___(1)___ is open and injection into the core is being supplied by ___(2)___ SLC pump(s).

A. (1) SLC-14A (LOOP A SQUIB VALVE)
   (2) One

B. (1) SLC-14A (LOOP A SQUIB VALVE)
   (2) Two

C. (1) SLC-14B (LOOP B SQUIB VALVE)
   (2) One

D. (1) SLC-14B (LOOP B SQUIB VALVE)
   (2) Two

Answer: D

Explanation:
The DS-3A and DS-3B white lights indicate continuity and that the valve is closed. The DS-3A or DS-3B light being out indicate that its associated squib valve has fired and opened.

The SLC pumps can pump through either squib valve after it has fired. The SLC trains are cross connected and this is not how other safety systems are designed. Normally a closed valve would prevent a pump on the same train from being able to inject into the core.

A is wrong because Squib Valve A is closed and two pumps are injecting into the core. Plausible because of the unique design feature described above.
B is wrong because Squib Valve A is closed. Plausible because two pumps are injecting into the core
C is wrong because two pumps are injecting into the core. Plausible because 14B is the open squib valve.
D is correct.

Technical References:
Procedure 2.2.74 Standby Liquid Control System, revision 56, page 11

References to be provided to applicants during exam:
None.

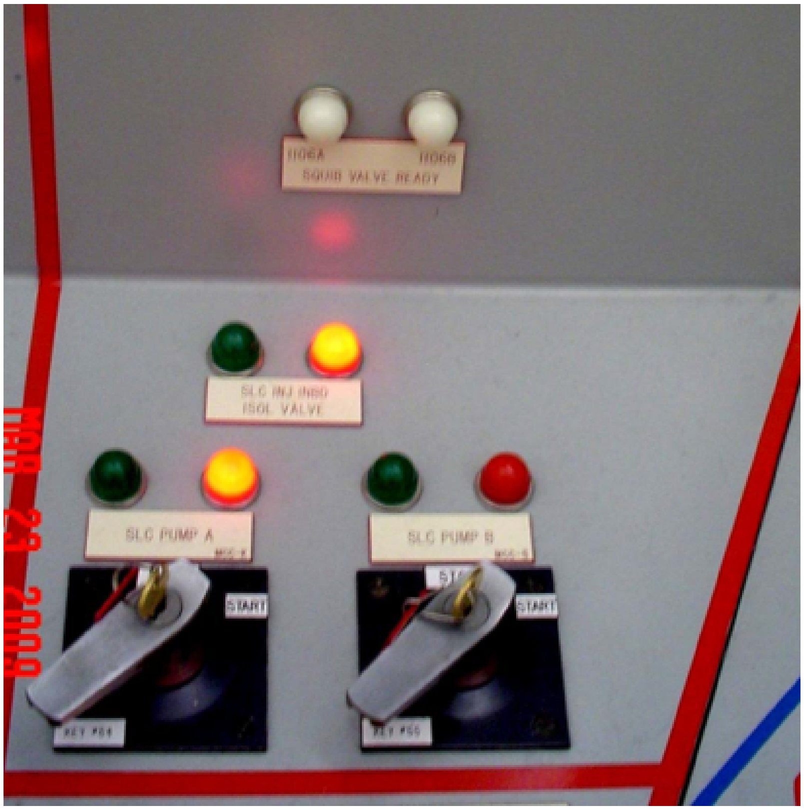
Learning Objective: COR002-29-02, Standby Liquid Control, Revision 28, Enabling objective 8.8

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 2009 NRC Exam #19
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41.7	

PARENT QUESTION

QUESTION: RO 19

During ATWS conditions the following indications are noted on 9-5.



What is the status of SLC injection?

a. SLC pumps are not injecting
b. SLC Pump 1A is injecting through Squib Valve 14A only
c. SLC Pump 1A is injecting through Squib Valve 14B only
d. SLC Pump 1A is injecting through Squib Valve 14A and Squib Valve 14B

ANSWER: RO 19

d. SLC Pump 1A is injecting through Squib Valve 14A and Squib Valve 14B

Both Squib lights are out this indicates both squib valves fired. With the red light on this indicates SLC pump A is running. With both lights out for SLC pump B this indicates the breaker is tripped for SLC pump B.

Examination Outline Cross-Reference		Level	RO
215004 (SF7 SRMS) Source-Range Monitor		Tier #	2
		Group #	1
Ability to manually operate and/or monitor in the control room:		K/A #	A4.01
		Rating	3.9
A4.01 SRM count rate and period			

Question 8

Given the following:

- The Reactor is shutdown
- The Reactor mode switch is in Refuel
- Core offload has begun – moving the second fuel bundle is in progress
- Welders are working on CRDMs under vessel in the drywell
- Source range monitor counts on **ALL** channels are quickly increasing and decreasing

Source range counts are expected to be not less than __(1)__ but are being influenced by __(2)__.

A. (1) 3 cps
(2) Fuel shuffles

B. (1) 3 cps
(2) Electro Magnetic Interference (EMI)

C. (1) 100 cps
(2) Fuel shuffles

D. (1) 100 cps
(2) Electro Magnetic Interference (EMI)

Answer: B

Explanation:
A is wrong because fuel shuffles may cause SRM counts as high as 100 cps but not on all channels simultaneously, and actual neutron counts (from fuel movement) will not decrease as quickly as they rise; and is plausible because: during refueling ≥ 3 cps is about the expected source range counts (correct), and because the first fuel bundle has been moved.

B is correct because ≥3 cps is the expected source range counts and because if all channels are quickly increasing and decreasing the cause is EMI as actual neutron counts will not decrease as quickly as they rise; when SRM channel(s) responds at exactly the same time as energy is dissipated by a known EMI source (welding machines), then EMI source is most likely the cause.

C is wrong because fuel shuffles may cause SRM counts as high as 100 but not on all channels simultaneously, and because actual neutron counts (from fuel movement) will not decrease as quickly as they rise.

D is wrong because fuel shuffles may cause SRM counts as high as 100 cps but not on all channels simultaneously; and is plausible because EMI is correct.

Technical References:
Document where the correct answer is found (Reference, Revision, Page number)

- procedure 4.4.1, Instrumentation Operations, Source Range Monitoring System, Attachment A
- procedure 2.2.31, Fuel Handling – Refueling Platform, page 15
- Technical Specification 3.3.1.2

References to be provided to applicants during exam:
None.

Learning Objective:
COR002-30-02, Source Range Monitor, Revision 17, Enabling Objective 7.d

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	3
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
218000 (SF3 ADS) Automatic	Tier #	2
Depressurization	Group #	1
	K/A #	2.2.12
2.2.12 Knowledge of surveillance procedures	Rating	3.7

Question 9

The plant is in Mode 1 performing testing using procedure 6.ADS.201, “ADS Manual Valve Actuation (IST).”

In accordance with 6.ADS.201, which of the following is the FIRST average suppression pool temperature which would require termination of the testing?

A. 92°F

B. 99°F

C. 106°F

D. 113°F

Answer: C

Explanation:
A is plausible because greater than or equal to 90°F **service water** wouldn’t allow testing either. This could be confused for suppression pool temperature if an applicant misremembers the prerequisite.
B is plausible because greater than 95°F would require suppression pool cooling placed in service. An applicant may believe that testing would need to be terminated until suppression pool cooling is in service.
C is correct because according to the continuous actions attachment of 6.ADS.201 any temp above 105°F requires termination of the testing and this is the immediate required action for LCO 3.6.2.1.
D is plausible because it is above 105°F. 110°F is a trigger point for placing the mode switch in shutdown. This would also stop the testing. Since the question asks for the first temp a subset or accepting 2 correct answers should have been avoided.

Also, suppression pool temperatures are between 70 and 85°F during normal operation depending on the season. Temperatures in the above range could also be used to support balance of answers.

Technical References:
6.ADS.201, “ADS Manual Valve Actuation (IST),” Attachment 4
LCO 3.6.2.1 Suppression Pool Average Temperature – 1 hour or less Tech spec

References to be provided to applicants during exam:
None.

Learning Objective:
OPS Nuclear Pressure Relief / COR002-16-02, Revision 21, Enabling objective 3.k

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
209001 (SF2, SF4 LPCS) Low-Pressure Core Spray Knowledge of the effect that a loss or malfunction of the following will have on the LOW PRESSURE CORE SPRAY SYSTEM: K6.04 D.C. power	Tier#	2
	Group#	1
	K/A #	K6.04
	Rating	2.8
	Revision	0
Revision Statement:		

Question 10

Plant is at 100% power.

A Non-Licensed Operator has reported finding the supply breaker for 125 VDC Panel BB3 open on 125 VDC Distribution Panel B

While investigating the open DC feed to panel BB3, a steam leak develops in the drywell resulting in an automatic reactor scram.

Annunciator 9-3-1/A-7, CORE SPRAY A LOGIC ACTUATED and Annunciator 9-3-3/A-4, CORE SPRAY B LOGIC ACTUATED are alarming.

'A' Core Spray pump starts after a ___(1)___ second time delay and 'B' Core Spray pump ___(2)___ be manually started from panel 9-3.

A. (1) 5
(2) CAN

B. (1) 5
(2) CANNOT

C. (1) 10
(2) CAN

D. (1) 10
(2) CANNOT

Answer: D
Explanation: Core Spray Pump 'A' starts after a 10 second delay; Core Spray Pump 'B' CANNOT be manually started from panel 9-3. De-energizing 125 VDC Panel BB3 results in a loss of 125V DC control power to Core Spray Pump 'B', which prevents the 4160V supply breakers from closing automatically in the event of a LPCI initiation signal or manually from the control room. Core spray Pump 'A' is unaffected because DC control power for the 'A' pump breakers is supplied by 125 VDC Panel AA3
Distractors: Answer A Part 1 is plausible because 5 seconds start time in the time for Low pressure RHR pumps to start. An examinee can confuse the start times of the 2 different ECCS pumps. It is wrong because the sequential loading for core spray is 10 seconds. Part 2 is plausible if the examinee confuses the power loss and believes this is a power loss to the automatic initiation

only thinking that control power is available. It is wrong because BB3 does not take away the initiation logic, but it removes the control power to the ECCS breakers. Answer B Part 1 is plausible for the reasons stated in Distractor A. Part 2 is correct. Answer C Part 1 is correct. Part 2 is plausible for the reasons stated in Distractor A.		
Technical References: 2.2.9 CORE SPRAY SYSTEM (REV 86) 2.2.69 RESIDUAL HEAT REMOVAL SYSTEM (REV 103) 5.3DC125 LOSS OF 125 VDC (REV 45) COR0020602 OPS CORE SPRAY SYSTEM, COR0020702 OPS DC ELECTRICAL DISTRIBUTION; 3019 4160V Switchgear Elementary Diagram 3025 4160V Switchgear Elementary Diagram		
References to be provided to applicants during exam: none		
Learning Objective: COR0020602001050H Describe the Core Spray system design features and/or interlocks that provide for the following: Automatic system initiation COR0020702001080P Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following: AC Electrical Distribution		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	14486
	New	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) (7)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	
PSA Applicability:		
Top 10 Risk Significant System – Emergency DC Power		

PARENT QUESTION

QUESTION: 22 14486 (1 point(s))

Given the following:

Plant is in mode 1 at 100% power with no equipment inoperable or unavailable.

An auxiliary operator has reported finding the supply breaker for 125 VDC Panel AA3 open on 125 VDC Distribution Panel A.

While investigating the open DC feed to panel AA3, a steam leak develops in the drywell resulting in an automatic reactor scram.

Annunciator 9-3-1/A-7, CORE SPRAY A LOGIC ACTUATED and Annunciator 9-3-3/A-4, CORE SPRAY B LOGIC ACTUATED are alarming.

Which one of the following describes the response of the Core Spray Pumps?

Core spray Pump…

A. A starts after a 10 second delay; Core spray Pump B MUST be manually started from panel 9-3.

B. B starts after a 10 second delay; Core spray Pump A MUST be manually started from panel 9-3.

C. A starts after a 10 second delay; Core spray Pump B CANNOT be manually started from panel 9-3.

D. B starts after a 10 second delay; Core spray Pump A CANNOT be manually started from panel 9-3.

ANSWER: 22 14486

D. is correct. Core spray Pump B starts after a 10 second delay; Core spray Pump A CANNOT be manually started from panel 9-3. De-energizing 125 VDC Panel AA3 results in a loss of 125V DC control power to Core Spray Pump A, which prevents the 4160V supply breakers from closing automatically in the event of a LPCI initiation signal or manually from the control room. Core spray Pump B is unaffected because DC control power for the B pump breakers is supplied by 125 VDC Panel BB3.

Examination Outline Cross-Reference 217000 (SF2, SF4 RCIC) Reactor Core Isolation Cooling	Level RO Tier # 2 Group # 1 K/A # K3.03 Rating 3.5

Knowledge of the effect that a loss or malfunction of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) will have on following:

K3.03 Decay heat removal

Question 11

RCIC has been placed in service for RPV pressure control. RCIC-FIC-91, RCIC Flow Controller, is in Auto with a setpoint of 200 GPM.

RCIC-FIC-91 is then deenergized due to an I&C technician error. Power is restored by the technician 30 seconds later.

What is the flow rate of RCIC one minute after power is restored?

A. 0 GPM

B. 80 GPM

C. 200 GPM

D. 400 GPM

Answer: D

Explanation:
RCIC-FIC-91 has a controller reset feature if the controller loses power for more than 15 seconds. The controller will reset it self with a demand of 400 GPM if the controller loses power for more than 15 seconds. The loss of power for 30 seconds will cause RPV pressure to lower when power is restored to the controller because RCIC flow will be higher than before the failure.

A loss of flow transmitter input into RCIC-FIC-91 will cause a maximum speed demand on the RCIC turbine. This will cause RPV pressure to lower.

A is wrong. Plausible if the applicant thinks that a loss of power will cause the controller to have a demand setting of 0 GPM
B is wrong. Plausible because this is the flow rate that the minimum flow valve opens
C is wrong. Plausible because this would be the setting if power to the controller were lost for less than 15 seconds.
D is correct.

Technical References:
COR002-18-02, Reactor Core Isolation Cooling, Revision 32, page 41 and 49

References to be provided to applicants during exam:
None.

Learning Objective: COR002-18-02, Reactor Core Isolation Cooling, Revision 32, Enabling Objective 10.o

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	4
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
263000 (SF6 DC) DC Electrical Distribution	Tier #	2
	Group #	1
Knowledge of D.C. ELECTRICAL DISTRIBUTION design feature(s) and/or interlocks which provide for the following:	K/A #	K4.02
K4.02 Breaker interlocks, permissives, bypasses and cross ties	Rating	3.1

Question 12

Given the following:

- Reactor power is 100%
- No maintenance is in progress
- An undervoltage condition occurs on 4160v bus 1F
- The Plant Management System's normally aligned power supply is also lost due to a blown fuse
- Power is then returned to 4160v bus 1F

Plant Management Information System (PMIS) power is transferred automatically from _____.

A. MDP2 to the PMIS battery. 15 minutes later, is automatically transferred to MCC-L.

B. MDP2 to the PMIS battery. 15 minutes later, may be manually transferred to MCC-L.

C. MCC-L to the PMIS battery. 15 minutes later, is automatically transferred to MDP2.

D. MCC-L to the PMIS battery. 15 minutes later, may be manually transferred to MDP2.

Answer: A

Explanation:
A is correct because MDP2 is the normally aligned power supply and because bus 1F was lost there is a 15 minute lockout on MCC-L, therefore power will be provided by the battery and then automatically transferred to MCC-L once the 15 minutes is over.

B is wrong because at the end of the 15 minute period power will automatically transfer to MCC-L; and is plausible because MDP2 is the normally aligned power source.

C is wrong because MDP2 is the normal power supply not MCC-L and is plausible because after the 15 minute period power is automatically transferred.

D is wrong because MDP2 is the normal power source not MCC-L and power is automatically transferred not manually transferred; and is plausible because MDP2 and MCC-L are the two power sources that are normally powering PMIS.

Technical References:
USAR section VIII part 5
Ops DC Electrical Distribution, Lesson COR002-07-02 rev 35

References to be provided to applicants during exam:
None.

Learning Objective:
COR002-07-02, DC Electrical Distribution, Revision 35, Enabling Objective 6.t

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
261000 (SF9 SGTS) Standby Gas Treatment	Tier #	2
	Group #	1
Ability to monitor automatic operations of the STANDBY GAS TREATMENT SYSTEM including:	K/A #	A3.02
	Rating	3.2

A3.02 Fan start

Question 13

Given the following:

- SGT fan 1E is in AUTO and not running.
- SGT fan 1F is in STANDBY and not running.
- SGT subsequently receives an automatic initiation signal.

Which of the following describes the response of the SGT fans to this condition?

A. Both fans start near simultaneously and will remain running until manual operator action is taken to secure one or both fans.

B. Both fans start near simultaneously, but fan 1F will cycle off after a 15 second delay if total system flow is 1958 CFM and rising.

C. Fan 1E starts immediately. Fan 1F will subsequently start after a delay if train 1E flow drops below 800 CFM.

D. Fan 1E starts immediately. Fan 1F will subsequently start after a 15 second delay if train 1E flow is 1200 CFM and lowering.

Answer: C

Explanation:

A SGT fan which is in AUTO will start immediately upon receipt of an automatic initiation signal [high drywell pressure ($\leq$ 1.84 psig); low-low reactor water level ($\geq$ -42 inches); high radiation in the exhaust plenum initiation ($\leq$ 49 mR/hr)]. A fan which is in STANDBY will start if flow in the opposite train drops below 800 cfm and an initiation signal is present, <u>OR</u> the opposite fans control switch is in RUN and its flow is <800 cfm. Facility rep confirmed that there is a 25-50 second time delay before 800 cfm lo flow relay picks up to start a STANDBY fan. See drawing 3038 SH 7.

A is wrong because Fan 1F would not immediately start if in STANDBY. Plausible because this is how the system would behave if both fans were in AUTO mode. Applicant may believe that as an ESF component, SGT starts on receipt of an automatic initiation in both AUTO and STBY modes.

B is wrong because Fan 1F would not immediately start if in STANDBY. Plausible because 1958 CFM is the setpoint for SGT UNIT HIGH FLOW K2/A3 alarm, and some systems cycle off and on when in a standby mode (such as instrument air). (Could also use a setpoint of 1780 CFM, which is the rated flow for 1 fan.)

C is correct.

D is wrong because Fan 1F will not start until system flow is 800 CFM and dropping. Plausible because 1200 CFM lowering is the setpoint for SGT UNIT LOW FLOW K2/B3 alarm, as measured on FI-545 total system flow.

Technical References:
OPS Standby Gas Treatment/COR002-28-02 Rev 26 Student Guide

References to be provided to applicants during exam:
None.

Learning Objective:
COR002-28-02, Standby Gas Treatment, Revision 26, Enabling objective 8.a

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
300000 (SF8 IA) Instrument Air	Tier #	2
	Group #	1
Knowledge of (INSTRUMENT AIR SYSTEM) design feature(s) and or interlocks which provide for the following:	K/A #	K4.03
	Rating	2.8
K4.03 Securing of IAS upon loss of cooling water		

Question 14

The plant is in shutdown with Instrument Air isolation valve IA-1936 (air supply to the Plant Air Compressor Cooling Water Valves) closed to support a clearance order. Subsequently, a loss of Reactor Equipment Cooling (REC) occurs.

What air compressor(s) no longer have cooling water supplied to the air compressor(s) ?

A. Air Compressor A only

B. Air Compressor C only

C. Air Compressors A and B

D. Air Compressors B and C

Answer: C

Explanation:
IA-1936, SAC REC/TEC AOV HEADER ISOLATION is closed in the stem. This will align cooling of Air Compressors A and B to REC and Air Compressor C to TEC

A is wrong. Plausible because REC is the normal cooling water source for Air Compressor A.
B is wrong. Plausible because REC is the alternate cooling water source for Air Compressor C.
C is correct
D is wrong. Plausible because the applicant could incorrectly think that B and C Air Compressors are cooled by REC with air isolated. Normally Air Compressors B & C are cooled by TEC

Technical References:
COR001-17-01, Plant Air, Revision 34, page 16

References to be provided to applicants during exam:
None.

Learning Objective: COR001-17-01, Plant Air, Revision 34, Enabling objectives 9.a and 9.b

Question Source: Bank #

(note changes; attach parent)	Modified Bank # New	3982
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 3
10CFR Part 55 Content:	55.41.7	

PARENT QUESTION

QUESTION: 39 3982 (1 point(s))

The plant is in MODE 5 when Instrument Air isolation valve IA-1936 (air supply to the Plant Air Compressor Cooling Water Valves) is inadvertently closed.

What is the status of Air Compressor cooling?

a. Compressor "A" is cooled by TEC.
Compressors "B" **AND** "C" are cooled by REC.

b. Compressors "A" **AND** "B" are cooled by REC.
Compressor "C" is cooled by TEC.

c. Compressor "A" is cooled by REC.
Compressors "B" **AND** "C" are cooled by TEC.

d. **ALL** Air Compressors are without cooling.

ANSWER: 39 3982

b. Compressors "A" **AND** "B" are cooled by REC.
Compressor "C" is cooled by TEC.

Examination Outline Cross-Reference	Level	RO
262001 (SF6 AC) AC Electrical Distribution	Tier #	2
	Group #	1
Ability to monitor automatic operations of the A.C. ELECTRICAL DISTRIBUTION including:	K/A #	A3.01
	Rating	3.1

A3.01 Breaker tripping

Question 15

Given the following:

- The plant is at 100% power
- The main generator trips
- Breaker 1AS fails to close
- 4160v bus 1A loses power

With no operator action, 4160v bus 1F power supply will automatically shift initially to the ___(1)___. If after 14 seconds bus 1F is without power, bus 1F power supply will automatically shift to the ___(2)___.

A. (1) emergency transformer
(2) startup transformer

B. (1) emergency transformer
(2) emergency diesel generator 1

C. (1) startup transformer
(2) emergency transformer

D. (1) startup transformer
(2) emergency diesel generator 1

Answer: B

Explanation:
A is wrong because after shifting to the emergency transformer if 10 seconds with no power elapses then supply power shifts to EDG 1 not the startup transformer; is plausible because the startup transformer would normally supply bus 1A which supplies bus 1F following a main generator trip and other loads such as instrument air shift to the startup transformer.

B is correct because with the 1AS breaker failure (startup transformer supply breaker), bus 1A loses power (which is the normal power supply to bus 1F) so 1F supply power will automatically shift to the emergency transformer and after 10 additional seconds of no power will automatically shift to emergency diesel generator 1.

C is wrong because power initially shifts to the emergency transformer not the startup transformer and because the backup alternate power source is EDG 1 not the emergency transformer; is plausible because the emergency transformer is one of the correct alternate power sources and because normally the startup transformer could power bus 1A which powers bus 1F.

D is wrong because power initially shifts to the emergency transformer not the startup transformer; is plausible because EDG1 is the correct second alternate power source.

Technical References:
USAR section VIII, page VIII-4-2

References to be provided to applicants during exam:
None.

Learning Objective:
COR001-01-01, AC Electrical Distribution , Revision 50, Enabling objective 9.g

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/**Fundamental**	
	Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
218000 (SF3 ADS) Automatic Depressurization	Tier #	2
	Group #	1
	K/A #	K5.01
	Rating	3.8

Knowledge of the operational implications of the following concepts as they apply to AUTOMATIC DEPRESSURIZATION SYSTEM:

K5.01 ADS logic operation

Question 16

Plant conditions are such that the ADS initiation logic is satisfied, and ADS blowdown is in progress. One RHR pump and one Core Spray pump are operating. All 6 ADS valves are open.

Which of the following will cause all 6 ADS valves to close **AND** stay closed?

A. Reactor Water Level rising to 0 inches

B. RHR pump discharge pressure lowering to 100 psig

C. Placing both ADS INHIBIT switches in the INHIBIT position

D. Pressing and releasing both Timer Reset pushbuttons simultaneously

Answer: C

Explanation:
D is incorrect but plausible if one does not recall that pressing both timer reset buttons will close the valves, but they will reopen 109 seconds later.
A is incorrect but plausible if one does not recall that the reactor water level switches are sealed in.
B is incorrect but plausible if one does not recall that all low pressure pumps must be secured to stop a blowdown.
C is correct.

Technical References:
COR002-16-02, Nuclear Pressure Relief, Revision 21, Pages 19 - 21

References to be provided to applicants during exam:
None.

Learning Objective:
COR002-16-02, Nuclear Pressure Relief, Revision 21, Enabling objective 6.a

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

Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 3
10CFR Part 55 Content:	55.41.5	

Examination Outline Cross-Reference	Level	RO
203000 (SF2, SF4 RHR/LPCI) RHR/LPCI: Injection Mode	Tier #	2
	Group #	1
	K/A #	2.4.18
2.4.18 Knowledge of the specific bases for EOPs	Rating	3.3

Question 17

The Crew is implementing EOP-1A, RPV Control. Drywell pressure is above 1.84 psig.

In accordance with EOP-1A, operators will prevent injection from CS and LPCI pumps not required to assure adequate core cooling when the RPV pressure is above a minimum pressure of ___(1)___ to prevent ___(2)___.

A. (1) 295 psig
(2) unnecessary injection that would complicate efforts to control RPV level

B. (1) 295 psig
(2) inducing a large power excursion large enough to severely damage the core

C. (1) 350 psig
(2) unnecessary injection that would complicate efforts to control RPV level

D. (1) 350 psig
(2) inducing a large power excursion large enough to severely damage the core

Answer: C

Explanation:
A is wrong. Part 1 is plausible because 295 psig is the discharge pressure of the LPCI pumps. Part 2 is correct.
B is wrong. Part 1 is plausible for the reason stated above. Part 2 is plausible because preventing a large power excursion is an additional basis if the reactor is not shutdown
C is correct
D is wrong. Both parts are wrong but plausible for the reasons stated above.

Technical References:
AMP-TBD00, CNS PSTG/SATG Technical Basis, revision 10, page B-6-34

References to be provided to applicants during exam:
None.

Learning Objective: INT008-06-05, OPS EOP Flowchart 1A – RPV Control, RPV Pressure, Revision 30, Enabling objective 11.

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

Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
400000 (SF8 CCS) Component Cooling Water	Tier #	2
	Group #	1
Knowledge of the physical connections and/or cause-effect relationships between CCWS and the following:	K/A #	K1.01
	Rating	3.2

K1.01 Service water system

Question 18

Given the following:

- Reactor has been scrammed due to a loss of all REC pumps
- The control room supervisor has directed you to initiate service water back up to REC per Emergency Procedure 5.2REC, Loss of REC

After placing the SW TO REC DIV 1 CROSSTIE switch to OPEN, you would ensure the _____ are OPEN.

A. North Critical Loop Service Water supply and return valves

B. South Critical Loop Service Water supply and return valves

C. North Critical Loop REC supply and return valves

D. South Critical Loop REC supply and return valves

Answer: A

Explanation:
A is correct because Attachment 6, step 1.9 has operators ensure that SW-MO-886, SW SUPPLY TO NORTH CRITICAL LOOP and SW-MO-888, SW RETURN FROM REC NORTH CRITICAL LOOP, are open (the Division 1 SW supply and return valves).

B is wrong because the south loop valves are for Division 2; is plausible because if the question asked for Division 2 then it would be correct.

C is wrong because the Reactor Equipment Cooling Valves must be closed to accomplish the task, not opened; is plausible because these are Division 1 valves that must be manipulated.

D is wrong because these are Division 2 valves and because the Reactor Equipment Cooling valves must be closed not opened; is plausible because these valves must be manipulated to initiate Service Water backup on Division 2.

Technical References:
Emergency Procedure 5.2REC, Loss of REC, Attachment 6 page 13, steps 1.9 and 1.8

References to be provided to applicants during exam:
None.

Learning Objective:
COR002-19-02, Reactor Equipment Cooling Revision 31, Enabling objectives 4.a and 4.b

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	**Memory**/Fundamental	2
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
212000 (SF7 RPS) Reactor Protection	Tier #	2
	Group #	1
Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	K/A #	A2.11
	Rating	4.0
A2.11 Main steamline isolation valve closure		

Question 19

The plant is operating at 50% power due to performing 6.MS.201, Main Steam Isolation Valve Operability Test (IST).

While closing MS-AOV-80B using Section 4, MSIV Spring Only Closure Tests, the following occurs:

- MS-AOV-86D fails CLOSED.

(1) Which RPS trip system will be affected by this configuration?

(2) Using 6.MS.201 what method will you use to reopen MS-AOV-80B?

A. (1) RPS A
(2) Release MS-AOV-80B MSIV TEST pushbutton

B. (1) RPS A
(2) Place MS-AOV-80B control switch to the OPEN position

C. (1) RPS B
(2) Release MS-AOV-80B MSIV TEST pushbutton

D. (1) RPS B
(2) Place MS-AOV-80B control switch to the OPEN position

Answer: C

Explanation:

The chart below shows the RPS configuration of closing 2 steam lines. Closing 1 steam line has no affect while closing any 3 and obviously 4 results in an RPS actuation.

Steam Lines Closed	Channel Affected	RPS result
A and B	A1	½ scram channel A
A and C	B1	½ scram channel B
A and D	None	No change
B and C	None	No change

B and D	B2	½ scram channel B
C and D	A2	½ scram channel A

A is wrong because wrong channel but correct action.
B is wrong because wrong channel and action. If the MSIV timing test were being performed the action would be correct. This section of the procedure has you close the valve by taking the handswitch to close.
C is correct because correct channel and action. During the MSIV spring closure section of the procedure the MSIV test pushbutton is used. Since the handswitch for the MSIV is in the open position, when you release the test pushbutton the MSIV will open.
D is wrong because correct channel but wrong action.

Technical References:
6.MS.201, Main Steam Isolation Valve Operability Test (IST), Revision 24, p. 11

References to be provided to applicants during exam:
None.

Learning Objective:
COR002-14-02, Main Steam, Revision 30, Enabling objectives 3.p, 7.f, 11.a
COR002-21-02, Reactor Protection System, Revision 25, Enabling objectives 4.d, 5.n, 8.f, 10.c, 12.b

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	4
10CFR Part 55 Content:	55.41.5	

Examination Outline Cross-Reference	Level	RO
223002 (SF5 PCIS) Primary Containment Isolation/Nuclear Steam Supply Shutoff Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF will have on following: (CFR: 41.7 / 45.4) K3.07 Reactor pressure	**Tier#**	2
	Group#	1
	K/A #	223002 K3.07
	Rating	3.7
	Revision	0
Revision Statement:		

Question 20

The plant is at 72% power.

Contacts fail closed inside the TEST button on Panel 9-3 for MSIV 80A, as though the TEST button was depressed.

Which one of the following describes the effect of this failure on MSIV 80A and how reactor pressure is affected?

A. MS-AOV-80A FAST closes, reactor pressure initially rises, then returns to the original value.

B. MS-AOV-80A FAST closes, reactor pressure rises until the reactor scram setpoint is reached, then pressure drops to a lower value.

C. MS-AOV-80A closes slowly, reactor pressure does not change as Turbine Control Valves throttle open.

D. MS-AOV-80A closes slowly, reactor pressure rises and stabilizes at a higher value, Bypass valves remain closed.

Answer: D
Explanation: There are four Main Steam Lines, with a rated total steam flow of 9.56 Mlbm/hr. Each steam line has two series MSIVs. MSIV 80 A is the inboard MSIV on MSL A. Each MSIV has a 2-position (CLOSE/AUTO OPEN) control switch and an adjacent TEST button. MSIVs close in 3-5 seconds when their control switch is placed to CLOSE; however, it takes >30 seconds for an MSIV to fully close when its TEST button is depressed. The test solenoid is used to "slow close" the MSIV for testing. With the test button on Panel 9-3 depressed, the solenoid is positioned to bleed air from the bottom of the air cylinder through an orifice to cause the valve to close slowly. When the button is released, the solenoid is positioned to allow the valve to re-open. If the switch fails such that contacts are closed, the associated MSIV will fully close and remain closed. As the MSIV closes, reactor pressure rises due to the reduction in steam flow in MSL A. As reactor pressure rises, steam flow in the other three steam lines rises until MSIV 80A is fully closed, ending the rise in

reactor pressure. With one steam line isolated, each unisolated line will have 133% of the original flow. At 72% power, this is less than the 142.7% Hi Steam Flow Trip; therefore, no Group 1 isolation signal is generated for this event. Below 75% power, the RPS trip setpoints for high reactor pressure or APRM high flux will not be reached.		
Distracters: Answer A is plausible with respect to MSIV 80A closing fast, because failure of its control switch in the CLOSE position will result in fast closure. This part is wrong because the TEST button energizes a solenoid valve that includes an orifice that limits air bleed-off from the MSIV actuator, resulting in slow closure. This answer is plausible with respect to the effect on reactor pressure because MSIV closure results in a rise in reactor pressure and because steam flow in the other three steam lines rises. An examinee may believe reactor pressure returns to the original value due to steam flow diverting to the other three steam lines and choose this answer. This part is wrong because the elevated reactor pressure is the cause of the rise in steam flow in the other three MSLs, and reactor pressure stabilizes at a higher value due to the reduction in effective MSL cross section and increased friction losses. Answer B is plausible and wrong with respect to MSIV 86A fast closure for the reason stated for distractor A. It is plausible with respect to the effect on reactor pressure because a Caution before step 5.4 of procedure 6.MS.201 states Reactor scram and/or Group 1 Isolation may occur if MSIV TEST button depressed greater than 20 seconds while at or near rated power. It is wrong because the plant is designed to stay on line for a single MSIV closure, since the unaffected MSLs are capable of accommodating the increased steam flow without reaching the MSL High Flow Group 1 isolation setpoint. Procedure 2.4MSIV states a scram on high reactor pressure may occur is power is >75%, but power is below 75% as an initial condition for this question. Answer C is plausible with respect to the effect on reactor pressure because reactor pressure rises, and if that was the initiator, Turbine Control Valves would open to control equalizing header pressure, and, hence, reactor pressure essentially steady. The examinee who does not understand the integrated effect of MSIV closure on upstream and downstream steam flow and pressure may choose this answer. It is wrong because reactor pressure rises due to the reduction in steam flow through MSL A. Also, equalizing header pressure initially lowers due to lowering flow in MSL A, so Turbine Control Valves initially throttle closed.		
Technical References: Lesson plan COR002-14-02 [Ops Main Steam](Rev 30), GE drawing 791E266 sh 10, Procedure 6.MS.201 [Main Steam Isolation Valve Operability Test(IST)](Rev 24), procedure 2.4MSIV [Inadvertent MSIV Closure](Rev 10)		
References to be provided to applicants during exam: None		
Learning Objective: COR002-14-02 Obj LO-6b, Given a specific Main Steam system malfunction, determine the effect on any of the following: Reactor pressure		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X

10CFR Part 55 Content:	55.41(b)(3)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	
PSA Applicability		
Top 10 Risk Significant System – Primary Containment - Isolation		

Examination Outline Cross-Reference	Level RO
262002 (SF6 UPS) Uninterruptable Power Supply (AC/DC)	Tier # 2
	Group # 1
	K/A # K6.02
Knowledge of the effect that a loss or malfunction of the following will have on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.):	Rating 2.8
K6.02 D.C. electrical power	

Question 21

The normal power supply has been lost to the 120 VAC Vital AC No-Break Power Panel (NBPP).

NBPP will automatically transfer from _____ to its alternate power supply.

A. MCC-R

B. MCC-L

C. 250 VDC Bus A

D. 250 VDC Bus B

Answer: C

Explanation:
A is wrong. Plausible because the applicant could think MCC-R is the normal power supply and not the alternate.
B is wrong. Plausible because MCC-L is the alternate power supply for PMIS another inverter in the plant.
C is correct.
D is wrong. Plausible because can mix up 250 A and B as the normal power supply to the inverter.

Technical References:
COR001-01-01, AC Electrical Distribution, Revision 50, page 84

References to be provided to applicants during exam:
None.

Learning Objective: COR001-01-01, AC Electrical Distribution, Revision 50, Enabling objective 4.f

Question Source:	Bank #	1635
(note changes; attach parent)	Modified Bank #	
	New	

Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
209001 (SF2, SF4 LPCS) Low-Pressure Core Spray	Tier #	2
	Group #	1
	K/A #	A1.01
Ability to predict and/or monitor changes in parameters associated with operating the LOW PRESSURE CORE SPRAY SYSTEM controls including:	Rating	3.4
A1.01 Core spray flow		

Question 22

Given the following:

- The plant was scrammed due to a LOCA
- Reactor water level is -103 inches and slowly lowering
- Reactor pressure is 400 psig
- Drywell pressure is 1.86 psig

With no operator action, the core spray pump flow is __________.

A. 0 gpm

B. Minimum flow

C. 2120 gpm

D. 4750 gpm

Answer: B

Explanation:
A is wrong because both pumps would be running since only one valid auto start signal is required to get an autostart and because at 400 psig the pumps would not be injecting into the core spray flow would be at minimum flow; is plausible because reactor water level is not low enough to auto start core spray and drywell containment is close to the setpoint for an autostart.

B is correct because both pumps would be running at minimum flow rate. Reactor pressure is higher than the discharge pressure of the core spray pumps, and with flow into the reactor vessel less than 2120 gpm the minimum flow control valves will be open for each pump.

C is wrong but plausible because the examinee could confuse the min flow operating flowrate with injection flow rate

D is wrong but plausible because both pumps auto start on receipt of either high drywell pressure (1.84 psig) or reactor water level (-113 inches or lower), and at reactor pressures less than 350 psig, core spray will commence injection and at the operability limit for flow rate the examinee could chose this answer.

Technical References:
USAR section VII part 4.5.4.2, page VII-4-13

References to be provided to applicants during exam:
None.

Learning Objective:
COR002-06-02, Core Spray System, Revision 27, Enabling objective 5.e

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/**Fundamental**	
	Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41.5	

Examination Outline Cross-Reference		Level	RO
239002 (SF3 SRV) Safety Relief Valves		Tier #	2
		Group #	1
Knowledge of electrical power supplies to the following:		K/A #	K2.01
		Rating	2.8

K2.01 SRV solenoids

Question 23

What are the normal and alternate power supplies to the SRV solenoids?

A. Normal - AA1
Alternate - BB1

B. Normal - AA2
Alternate - BB2

C. Normal - AA2
Alternate - None

D. Normal - BB2
Alternate - None

Answer: B

Explanation:
B is correct because all normal solenoid power comes from division 1 power, AA2. The ADS logic power is split between AA2 and BB2, but not solenoid power.

A is wrong because of above. AA1 and BB1 are 125vdc buses but neither bus powers any SRV solenoids.
B is correct because all SRV solenoids are normally powered from AA2, and their alternate power is BB2.
C is wrong because they do have different power supplies for the solenoids, but if they confuse the solenoids with logic power then they might think they are the same. A train logic does not have an alternate power supply, but B train logic does.
D is wrong because BB2 powers the solenoids in the event of a loss of AA2, but credible if you remember that only one division powers both but get them backwards. A train logic does not have an alternate power supply, but B train logic does.

Technical References:
COR002-16-02, Nuclear Pressure Relief, revision 21, page 27.

References to be provided to applicants during exam:
None.

Learning Objective:
COR002-16-02, Nuclear Pressure Relief, revision 21, Enabling objective 2.b

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	2
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.3	

Examination Outline Cross-Reference	Level	RO
206000 (SF2, SF4 HPCIS) High Pressure Coolant Injection	Tier #	2
	Group #	1
	K/A #	K5.02
Knowledge of the operational implications of the following concepts as they apply to HIGH PRESSURE COOLANT INJECTION SYSTEM:	Rating	2.8
K5.02 Turbine shaft sealing		

Question 24

When the gland seal exhauster for the HPCI system is out of service, HPCI may be operated in an emergency situation for a maximum of ___(1)___, and while operating in this condition, HPCI ___(2)___ considered inoperable.

A. (1) 8 hours
(2) is

B. (1) 8 hours
(2) is not

C. (1) 12 hours
(2) is

D. (1) 12 hours
(2) is not

Answer: D

Explanation:
A is incorrect. Part 1 is plausible because the licensee is required to make an 8 hour 10CFR50.72 notification if HPCI becomes Inoperable. Part 2 is plausible because one could believe HPCI inoperable with an inoperable component.
B is incorrect. Part 1 is plausible because of the reasons stated above. Part 2 is correct.
C is incorrect. Part 1 is correct. Part 2 is plausible for the reasons stated above.
D is correct.

This is not an SRO level question since one is not determining operability. This is a statement not only in the System Operating Procedure, but in the System Lesson associated with the specific learning objective.

Technical References:
COR002-11-02, High Pressure Coolant Injection, Revision 38, page 18
2.2.33, High Pressure Coolant Injection System, Revision 84, page 44

References to be provided to applicants during exam:
None.

Learning Objective:

COR002-11-02, High Pressure Coolant Injection, Revision 38, Enabling objective 10.c

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 3
10CFR Part 55 Content:	55.41.5	

Examination Outline Cross-Reference	Level	RO
217000 (SF2, SF4 RCIC) Reactor Core Isolation Cooling	Tier #	2
	Group #	1
	K/A #	K2.03
Knowledge of electrical power supplies to the following:	Rating	2.7

K2.03 RCIC flow controller

Question 25

What is the power supply to the RCIC flow controller, RCIC-FIC-91?

A. 125 Vdc Panel AA2

B. 125 Vdc Panel AA3

C. 125 Vdc Panel BB2

D. 125 Vdc RCIC starter rack

Answer: A

Explanation:
A is correct because 125 Vdc Panel AA2 is the power supply to the RCIC flow controller.

B is wrong because the correct power source is 125 Vdc Panel AA2; is plausible because it is a 125 Vdc RCIC system power supply

C is wrong because the correct power source is 125 Vdc Panel AA2; is plausible because it is a 125 Vdc RCIC system power supply.

D is wrong because the correct power source is 125 Vdc Panel AA2; is plausible because it is a 125 Vdc RCIC system power supply.

Technical References:
Lesson COR002-18-02 rev 32, Ops Reactor Core Isolation Cooling, page 51

References to be provided to applicants during exam:
None.

Learning Objective:
COR002-18-02, Reactor Core Isolation Cooling, Revision 32, Enabling Objective 6.c

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

	Comprehensive/Analysis
10CFR Part 55 Content:	55.41.7

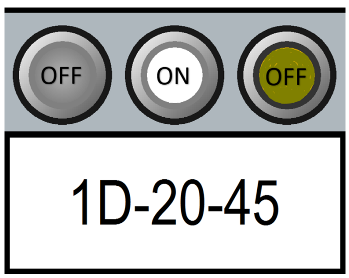
Examination Outline Cross-Reference	Level	RO
215005 (SF7 PRMS) Average Power Range Monitor/Local Power Range Monitor	Tier #	2
	Group #	1
	K/A #	A1.05
	Rating	3.3

Ability to predict and/or monitor changes in parameters associated with operating the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM controls including:

A1.05 Lights and alarms

Question 26

You observe the following indications for LPRM 1D-20-45 on Panel 9-14:



(1) Which annunciator do you expect will be illuminated on Panel 9-5?
(2) Which of the panels will auto reset after the condition clears?

A. (1) LPRM UPSCALE
   (2) Panel 9-5 Annunciator

B. (1) LPRM UPSCALE
   (2) Panel 9-14 Lights

C. (1) LPRM DOWNSCALE
   (2) Panel 9-5 Annunciator

D. (1) LPRM DOWNSCALE
   (2) Panel 9-14 Lights

Answer: C

Explanation:
A is wrong because see C.
B is wrong because see C.
C is correct because the lights above LPRM are unmarked, but their order is bypass, downscale, upscale. If the middle light is lit, then a downscale condition has occurred. This would drive the LPRM downscale annunciator to illuminate. The lights above the LPRM are seal-in lights which would have to be reset in order to clear. The annunciator would auto reset after the condition clears.
D is wrong because see C

Technical References:
Local Power Range Monitor / COR002-13-02, Revision 19, Page 13-14

References to be provided to applicants during exam:
Picture of the lights for an LPRM

Learning Objective:
COR002-13-02, Local Power Range Monitor, Revision 19, Enabling objectives 6a and 3c

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41.5	

Examination Outline Cross-Reference	Level	RO
216000 (SF7 NBI) Nuclear Boiler Instrumentation	Tier #	2
	Group #	2
	K/A #	K5.12
Knowledge of the operational implications of the following concepts as they apply to NUCLEAR BOILER INSTRUMENTATION:	Rating	3.2
K5.12 Effects on level indication due to rapid changes in void fraction		

Question 27

The plant is at 10% power. The Main Turbine bypass valves are at 25% open and controlling RPV pressure stable at 926 psig. A malfunction in the DEH has caused one Main Turbine Bypass valve to close rapidly.

Indicated wide range reactor water level will ___(1)___ due to a rapid change in ___(2)___.

A. (1) lower
(2) void fraction

B. (1) lower
(2) temperature coefficient

C. (1) rise
(2) void fraction

D. (1) rise
(2) temperature coefficient

Answer: A

Explanation:
A is correct
B is wrong. Plausible because RPV water level will initially lower if the TBV close
C is wrong. Plausible because RPV level will rise if the TBVs rapidly open and void coefficient is the reason why RPV water level is changing
D is wrong. Plausible because RPV water level will increase if the TBVs rapidly open

Technical References:
COR002-15-02, Nuclear Boiler Instrumentation, Revision 28, page 42

References to be provided to applicants during exam:
None.

Learning Objective:
COR002-15-02, Nuclear Boiler Instrumentation, Revision 28, Learning Objective 4.k

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	23293

	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41.5	

PARENT QUESTION

QUESTION: 45 23293 (1 point(s))

The plant is operating at 50% power when a DEH malfunction causes the Main Turbine Bypass valves to rapidly open.

Initially how is flow resistance in the core region affected?

Initially how is indicated wide range reactor water level affected?

a. increase, increase

b. increase, decrease

c. decrease, increase

d. decrease, decrease

ANSWER: 45 23293

a. increase, increase

Examination Outline Cross-Reference		Level	RO
288000 (SF9 PVS) Plant Ventilation		Tier #	2
		Group #	2
Knowledge of the effect that a loss or malfunction of the PLANT VENTILATION SYSTEMS will have on following:		K/A #	K3.02
		Rating	2.9
K3.02 Reactor building temperature			

Question 28

Given the following:

- A reactor scram occurs due to a small break LOCA.
- Annunciator 9-3-2, HPCI Logic Actuated, is in alarm
- HV-FAN-(FC-R-1G), FCU FC-R-1G, is in Auto
- HPCI Steam Supply Block valve, MO-14, fails to fully open

With no operator action, temperature in the __(1)__ will reach maximum **safe** operating temperature at __(2)__ºF in accordance with EOP 5A.

A. (1) SE Quad
(2) 175

B. (1) SE Quad
(2) 195

C. (1) SW Quad
(2) 175

D. (1) SW Quad
(2) 195

Answer: D

Explanation:
A is wrong because the HPCI pump is in the SW Quad and because the maximum safe operating temperature is 195ºF; is plausible because the SE Quad is close to the SW Quad and because 175ºF is close to 195ºF.

B is wrong because the HPCI pump is in the SW Quad; and is plausible because 195ºF is correct.

C is wrong because IAW EOP5A the maximum safe operating room in all CSCS pump rooms is 195ºF not 175ºF; is plausible because SW Quad is correct.

D is correct because HPCI Room FCU-G (SW Quad) auto starts when HPCI-MO-14 is full open and the FCU-G control switch is in AUTO

Technical References:
COR002-11-02 rev 38, page 21-22

EOP 5A

References to be provided to applicants during exam:
None.

Learning Objective:
COR002-11-02, High Pressure Coolant Injection, Revision 38, Enabling objective 10.j

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/**Fundamental** **Comprehensive**/Analysis	3
10CFR Part 55 Content:	55.41.5	

Examination Outline Cross-Reference	Level	RO
201001 (SF1 CRDH) CRD Hydraulic	Tier #	2
	Group #	2
Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROD DRIVE HYDRAULIC System:	K/A #	K6.06
	Rating	2.8
K6.06 Component cooling water systems		

Question 29

The plant is in Mode 4.
REC pressure instrument REC-PS-452A, REC SYSTEM LOW PRESSURE, spuriously fails low.

(1) What impact will this have on CRD pumps?

(2) What CRD pump component will fail first if cooling from REC is NOT restored??

A. (1) CRD pumps will lose cooling immediately.
(2) thrust bearing

B. (1) CRD pumps will lose cooling immediately.
(2) pump seal

C. (1) CRD pumps will lose cooling following a 40 second delay.
(2) thrust bearing

D. (1) CRD pumps will lose cooling following a 40 second delay.
(2) pump seal

Answer: C

Explanation:
REC-MO-700, NON-CRITICAL HEADER SUPPLY ISO VLV, shuts on low REC supply header pressure < 61 psig, with a 40 second delay. CRD pumps are supplied by the non-critical supply header. The REC system cools the CRDH pump lube oil cooler and thrust bearing. The lube oil then cools and lubricates other pump bearings and components. If REC cooling was lost for a long enough period of time and the pump continued to run, bearing clearances would be reduced to the point of pump seizure.

A is wrong because CRD pumps lose cooling after 40 seconds. Second part is correct.
B is wrong because CRD pumps lose cooling after 40 seconds, and REC does not cool CRD pump seals. Part 2 is plausible because REC supplies seal cooling for RWCU pumps, RHR pumps, and Reactor Recirc pumps.
C is correct.
D is wrong because REC does not cool CRD pump seals. First part is correct.

Technical References:
COR002-19-02 , Reactor Equipment Cooling, Revision 31
COR002-04-02, Control Rod Drive Hydraulics, Revision 30

References to be provided to applicants during exam:
None.

Learning Objective:
COR002-04-02, Control Rod Drive Hydraulics, Revision 30, Enabling objective 11.d

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
256000 (SF 2 CDS) Condensate 2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation	**Tier#**	2
	Group#	2
	K/A #	256000 2.1.23
	Rating	4.3
	Revision	0
Revision Statement:		

Question 30

Plant startup is in progress.

- 'C' Condensate Pump is running

Procedure 2.2.6 CONDENSATE SYSTEMS cautions that ___(1)___ Condensate Pump should be the second pump placed in service because ___(2)___.

A. (1) 'A'
(2) this will prevent loss of both condensate pumps due to loss of 4160V Bus A

B. (1) 'A'
(2) time delays for Condensate Pumps B and C suction pressure trips are longer than time delay for Condensate Pump A

C. (1) 'B'
(2) this will prevent loss of both condensate pumps due to loss of 4160V Bus A

D. 1) 'B'
(2) time delays for Condensate Pumps B and C suction pressure trips are longer than time delay for Condensate Pump A

Answer: C
Explanation: Both 'A' and 'C' Condensate pumps are powered from 4160 VAC A Bus so the caution states the preferred lineup is 'B' Condensate pump with either 'A' or 'C' Condensate pump
Distracters: Answer A Part 1is plausible because if the examinee confuses the combination with the TEC pumps, he could choose A Condensate pump as the correct answer. It is wrong because B condensate pump needs to be in combination with either A or C Condensate pumps and C Condensate pump is already running. Part 2 is correct. Answer B Part 1 is plausible for the reasons given for distractor A. Part 2 is plausible if the examinee confuses the trip setpoints of the Condensate Booster pumps. It is wrong because the caution is due to a loss of power to 'A' 4160 bus.

Answer D Part 1is plausible for the reasons stated for distractor A. Part 2 is plausible for the reasons stated in Distractor B.		
Technical References: Lesson plan OPS Condensate and Feedwater / COR002-02-02 (Rev 38) 2.2.6 CONDENSATE SYSTEMS (Rev 98)		
References to be provided to applicants during exam: none		
Learning Objective: COR002-09-02 Obj LO-4d, Describe how the DEH Control system operates to control the following: Pressure setpoint/pressure demand		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	
PSA Applicability:		
N/A		

Examination Outline Cross-Reference	Level	RO
290002 (SF4 RVI) Reactor Vessel Internals	Tier #	2
	Group #	2
Ability to (a) predict the impacts of the following on the REACTOR VESSEL INTERNALS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	K/A #	A2.04
	Rating	3.7
A2.04 Excessive heatup/cooldown rate		

Question 31

A maximum heat up and cool down rate of 100°F/hr is established to minimize cyclic stresses on pressure vessel components.

The most limiting components in the reactor vessel are the ___(1)___.

IAW Technical Specifications 3.4.9, if a cool down rate of 100°F/hr is exceeded in mode 3, restore the cool down rate to within limits in a maximum of ___(2)___.

A. (1) Feedwater nozzles
(2) 30 minutes

B. (1) Feedwater nozzles
(2) 1 hour

C. (1) CRD Housing stub tube welds
(2) 30 minutes

D. (1) CRD Housing stub tube welds
(2) 1 hour

Answer: A

Explanation:

There is a specific objective for operators to know the basis for the heatup and cooldown rates. Tech spec 3.4.9 requires restoring parameter(s) to within limits in 30 minutes.

A is correct.
B is incorrect but plausible since part 1 is correct. Part 2 is plausible as 1 hour is a required action time for RO applicants to know.
C is incorrect but plausible because part 2 is correct. Part 1 is plausible because CRD housing stub tube welds are one of the areas of concern for the ΔT between the vessel steam dome and bottom head drain.
C is incorrect but plausible Part 1 is plausible because CRD housing stub tube welds are one of the areas of concern for the ΔT between the vessel steam dome and bottom head drain. Part 2 is plausible as 1 hour is a required action time for RO applicants to know.

Technical References:
Technical Requirements Manual, Appendix A, page 10
Technical Specification 3.4.9

References to be provided to applicants during exam:
None.

Learning Objective: COR001-15-01, Nuclear Boiler, Revision 31, Enabling objective 10

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	3
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.5	

Examination Outline Cross-Reference	Level	RO
215002 (SF7 RBMS) Rod Block Monitor	Tier #	2
	Group #	2
Knowledge of electrical power supplies to the following:	K/A #	K2.03
	Rating	2.8

K2.03 APRM channels

Question 32

Identify the electrical power supply to the APRM channels.

A. RPS

B. 24/48 VDC

C. 120V Vital Bus CPP

D. 120V Vital Bus NBPP

Answer: A

Explanation:
A is correct.
B is incorrect but plausible because 24VDC supplies power to LPRM "UPSC” and "DNSC" lights on the full core display .
C is incorrect but plausible because CPP supplies power to the ”UPSCL TRIP/INOP”, “UPSCL ALM”, “DNSCL”, “BYPASS” lights on the apron section of Panel 9-5, and RR-FR-163.
D is incorrect but plausible because NBPP supplies recorder power.

Technical References:
COR002-01-02, Average Power Range Monitor, Revision 27, Page 19

References to be provided to applicants during exam:
None.

Learning Objective:
COR002-01-02, Average Power Range Monitor, Revision 27, Enabling Objective 6.b

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	2040
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	2
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7	

PARENT QUESTION

QUESTION: 30 2040 (1 point(s))

Identify the electrical power supply to the LPRM channels.

a. NBPP

b. 24/48 VDC

c. 120V Vital Bus CPP

d. RPS

ANSWER: 30 2040

d. RPS

Examination Outline Cross-Reference	Level	RO
201003 (SF1 CRDM) Control Rod and Drive Mechanism	Tier #	2
	Group #	2
	K/A #	A3.01
Ability to monitor automatic operations of the CONTROL ROD AND DRIVE MECHANISM including:	Rating	3.7
A3.01 Control rod position		

Question 33

The plant is operating at full power.

You have been tasked with moving control rod 18-27 one notch from position 12 to position 14.

Using the rod movement control switch, the control rod position indication will change from position 12 to positions _____.

A. 13, 14, 15, 14

B. 11, 12, 13, 14

C. 11, 12, 13, 14, 15, 14

D. 11, 10, 11, 12, 13, 14

Answer: B

Explanation:
A is wrong because the order is incorrect. Plausible if someone forgets an insert signal is generated before the withdraw signal to force the CRDM off its collet fingers. Also, going past 14 is plausible due to the settle function of CRDM. Testing if someone thinks the control rod goes past the intended position and settles back to the intended position.
B is correct because the sequence is correct. The insert signal forces the control to position 11 and the display will show all the individual reed switched the control rod moves past. The control rod won't go past the intended position if the switch is operated correctly.
C is wrong because the order is incorrect. Plausible if someone gets the insert correct but is confused on the settle function of a control rod.
D is wrong because the order is incorrect. Plausible if someone thinks the rod has to go in a full step instead of a half step.

Technical References:
COR002-05-02, Control Rod Drive Mechanism, Revision 14
Simulator. The text isn't explicit on RPIS during rod movement.

References to be provided to applicants during exam:
None.

Learning Objective: COR002-05-02, Control Rod Drive Mechanism, Revision 14, Enabling objective 12.a.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	2
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
259001 (SF2 FWS) Feedwater	Tier #	2
	Group #	2
Ability to predict and/or monitor changes in parameters associated with operating the REACTOR FEEDWATER SYSTEM controls including:	K/A #	A1.06
	Rating	2.7

A1.06 Feedwater heater level

Question 34

The plant is at 100% power. The Feedwater Heater A-3 level controller has malfunctioned and caused a loss of air to the actuators for LCV-62A, Heater A-3 to Heater A-2 level control valve and LCV 62B, Heater A-3 to Condenser 1A heater dump to the condenser valve.

Level in Feedwater Heater A-2 will initially ___(1)___and the level in Feedwater Heater A-3 will_____(2)_____.

A. (1) Rise
(2) Rise

B. (1) Rise
(2) Lower

C. (1) Lower
(2) Rise

D. (1) Lower
(2) Lower

Answer: B

Explanation:
LCV-62A and LCV-62B (level control valve and heater dump to condenser) will both fail open on a loss of air. Initially level in Feedwater Heater 1-A-2 will rise until its level controller responds to the increased drain flow from 1-A-3. Feed Water Heater 1-A-3 dump to the condenser also fails open and the level in Feedwater Heater 1-A-3 will lower.

A is wrong. Plausible because the applicant could incorrectly think that the loss of air position for LCV-62B is closed
B is correct.
C is wrong. Plausible because the applicant could incorrectly think that the loss of air position for LCV-62B is closed.
D is wrong. Plausible because the applicant could incorrectly think that the loss of air position for LCV-62A is closed.

Technical References:
COR0010401R28, Extraction Steam Heater Drains, revision 29, page 22 and 23.

References to be provided to applicants during exam:
None.

Learning Objective: COR0010401R28, Extraction Steam Heater Drains, revision 28, LO4c

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41.5	

Examination Outline Cross-Reference	Level	RO
219000 (SF5 RHR SPC) RHR/LPCI:	Tier #	2
Torus/Suppression Pool Cooling Mode	Group #	2
	K/A #	A4.01
Ability to manually operate and/or monitor in the control room:	Rating	3.8

A4.01 Pumps

Question 35

Following a large break LOCA, RHR pump A is the only operating RHR pump, operating in Suppression Pool Cooling Mode. To prevent pump run out, the maximum flow rate allowed from RHR pump A is _____ gpm.

A. 7700

B. 8000

C. 8400

D. 11500

Answer: C

Explanation:
A is wrong but plausible because 7700 gpm is the minimum expected pump flow.
B is wrong but plausible because 8000 gpm is what operators would throttle flow down to when switching from two pump operation to one pump operation.
C is correct because flows above 8400 gpm may cause pump run out.
D is wrong but plausible because the system operating procedure notes that when placing a second RHR subsystem during accident conditions that suppression pool cooling is not limited to 11500 gpm.

Technical References:
System Operating Procedure 2.2.69.3, RHR Suppression Pool Cooling and Containment Spray, rev 51, sections 2.6-2.7, page 3

References to be provided to applicants during exam:
None.

Learning Objective:
COR002-23-02, Residual Heat Removal System, Revision 32, Enabling objective 3.i

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

Comprehensive/Analysis

10CFR Part 55 Content: 55.41.7

Examination Outline Cross-Reference		Level	RO
286000 (SF8 FPS) Fire Protection		Tier #	2
		Group #	2
Knowledge of FIRE PROTECTION SYSTEM design feature(s) and/or interlocks which provide for the following:		K/A #	K4.04
		Rating	3.6

K4.04 Personnel safety during halon and/or carbon dioxide system actuation

Question 36

A fire was detected in the service water pump room.

What is the **MINIMUM** time, in seconds, before the service water pump room fire suppression system will release its contents?

A. 10

B. 20

C. 30

D. 50

Answer: B

Explanation:
A is wrong because this is the time delay for the supplemental diesel generator engine room.
B is correct because this is the time delay for this room. As stated, in the learning objectives the time delays are for personnel evacuation from the room.
C is wrong because this is not the time delay for the room. This time doesn't correlate to any room but was used because it is a common time.
D is wrong because this is the time delay for the diesel generator and day tank rooms.

The table below outlines all the rooms, with time delay, which have an oxygen removal suppression system.

Room/System	Time Delay (sec)	Suppression System
Main Turbine	30	Low Pressure CO_2
Diesel Generator Room	50	High Pressure CO_2
Service Water Pump Room	20	Halon
Computer Room	20	Halon
Simulator Complex	20	Halon
Supplemental Diesel Generator Engine Room	10	INERGEN

Technical References:
COR001-05-01, Fire Protection System, Revision 36, p. 53

References to be provided to applicants during exam:

None.

Learning Objective: COR001-05-01, Fire Protection System, Revision 36, Enabling objective 8.m

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference 241000 (SF3 RTPRS) Reactor/Turbine Pressure Regulating		Level	RO
		Tier #	2
		Group #	2
		K/A #	K3.01
Knowledge of the effect that a loss or malfunction of the REACTOR/TURBINE PRESSURE REGULATING SYSTEM will have on following: K3.01 Reactor power		Rating	4.1

Question 37

The plant is operating at 80% with the following equipment tagged out for maintenance:

- DEH pump A
- TEC pump C

The following annunciator is received:

- 4160 V Bus 1B BRK SS1B Trip, C-3/F-7

Which one of the following identifies the impact of this annunciator under the current plant conditions?

A. The reactor will automatically Scram due to a Main Turbine trip

B. A manual reactor Scram is required due to the loss of Turbine Equipment Cooling

C. Reactor power is required to be reduced due to degraded Main Condenser vacuum

D. The reactor automatically stabilizes at a lower power level due to the reduced Feedwater flow

Answer: A

Explanation:
The loss of 480V B results in the loss of MCC-F and the loss of the only remaining DEH pump. A turbine trip at this power level results in an automatic reactor Scram.

A is correct.
B is wrong. Plausible due to TEC pump 1C losing power but already tagged out for maintenance. The candidate that confuses TEC pump power supplies would select this answer.
C is wrong. Plausible due to 2 Circulating pumps being powered from Bus 1B. The candidate that confuses which power supply is lost would select this answer.
D is wrong. Plausible due to a Condensate & Condensate Booster pump being powered from Bus 1B. The candidate that confuses which power supply is lost would select this answer.

Technical References:

Procedure 2.3_C-3 (Panel C - Annunciator C-3), Rev. 57
Procedure 5.3AC480 (480 VAC BUS FAILURE), Rev.43
Procedure 2.4TEC (TEC Abnormal), Rev.30
Procedure 2.4VAC (Loss Of Condenser Vacuum), Rev.27

References to be provided to applicants during exam:
None.

Learning Objective: COR0011402R25, revision 25, enabling objective 5.b.

Question Source:	Bank #	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	CNS 2015-12 Q61
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
230000 (SF5 RHR SPS) RHR/LPCI:	Tier #	2
Torus/Suppression Pool Spray Mode	Group #	2
	K/A #	K1.01
Knowledge of the physical connections and/or cause-effect relationships between RHR/LPCI: TORUS/SUPPRESSION POOL SPRAY MODE and the following:	Rating	3.6

K1.01 Suppression pool

Question 38

Given the following:

- A large break LOCA has occurred
- Drywell Pressure is 1.8 psig
- You have been directed to place RHR train A in Containment Spray Mode
- Reactor Vessel Water Level is -190.5 inches CFZ
- LPCI signal is sealed in

In order to spray containment using water from the suppression pool, you will need _________ to take spray valve control to place the RHR system in containment spray mode.

1. the LPCI initiation signal cleared
2. to place the CONTMT COOLING 2/3 CORE VALVE CONTROL PERMISSIVE in MANUAL OVERRIDE
3. switch the CONTMT COOLING VLV CONTROL PERMISSIVE switch to MANUAL and release
4. drywell pressure to increase

A. 1 and 2 only

B. 1 and 3 only

C. 2, 3, and 4 only

D. 1, 3, and 4 only

Answer: C

Explanation:
A is wrong because containment pressure needs to be 2 psig (#4 is needed), the LPCI initiation signal needs to be sealed in – making #1 incorrect, the contmt vlv control permissive switch needs to be placed in Manual and then released (#3 is correct); is plausible because the core is less than 2/3 covered necessitating the containment cooling 2/3 core valve control permissive to be in manual override (#2 is correct).
B is wrong because the LPCI signal needs to be sealed in not cleared and because drywell pressure needs to increase to 2 psig or more; and is plausible because choice #3 is correct.

C is correct because to take spray valve control requires: LPCI sealed in, 2/3 core covered or the containment cooling valve control permissive in manual override, drywell pressure 2 psig or higher, and the containment cooling valve control permissive to manual.
D is wrong because the LPCI signal needs to be sealed in not cleared; is plausible because choices 3, and 4 are correct as discussed above.

Technical References:
COR002-23-02, Residual Heat Removal System, Revision 36, page 23
SOP 2.2.69.3, RHR Suppression Pool Cooling and Containment Spray, Revision 51, section 11, page 34

References to be provided to applicants during exam:
None.

Learning Objective: Document learning objective if possible.
COR002-23-02, Residual Heat Removal System, Revision 36, Enabling objectives 3.p and 3.q

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	3
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
295001 (APE 1) Partial or Complete Loss of Forced Core Flow Circulation	Tier #	1
	Group #	1
	K/A #	AA2.03
Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION:	Rating	3.3

AA2.03 Actual core flow

Question 39

Given the following conditions:

- The "A" Recirculation Pump has tripped.
- MO-53A, "A" Recirculation Pump discharge valve was closed and is now open
- LOOP A JET PUMP FLOW (FI-92A) indicates 2 Mlbm/hr
- LOOP B JET PUMP FLOW (FI-92B) indicates 35 Mlbm/hr

Annunciator E-3 on Panel 9-4-3, RECIRC LOOP A OUT OF SERVICE, is alarming

Which choice below describes the expected values for Total Core Flow as indicated on Panel 9-5 Recorder NBI-FRDPR-95 **AND** Actual Core Flow?

A. Both indicated and actual core flows are 33 Mlbm/hr

B. Both indicated and actual core flows are 37 Mlbm/hr

C. Indicated core flow is 33 Mlbm/hr and actual core flow is 37 Mlbm/hr

D. Indicated core flow is 37 Mlbm/hr and actual core flow is 33 Mlbm/hr

Answer: A

Explanation:
The alarm procedure directs entry into 2.2.68.1, which discusses the method of determining core flow depending on the difference between FI-92A and FI-92B. The quick and simple check to determine JP Flow Summing network accuracy is to subtract the flow in the idle loop (as read on NBI-FI-92A or B) from the flow in the active loop. The difference between the two indications is total core flow, which can be used to determine if indicated Total Core Flow (NBI-FRDPR-95) is accurate or faulty and if indicated core flow on the Power to Flow Map (PMIS) is accurate of faulty.
A is correct.
B is incorrect but plausible if one believes you sum the two flows to arrive at both indicated and actual core flows.
C is incorrect the first part is correct. The second part is plausible if one believes you sum the two flows to arrive at actual core flow.
D is incorrect the second part is correct. The first part is plausible if one believes you sum the two flows to arrive at indicated core flow.

Technical References:
COR002-22-02, Reactor Recirculation, Revision 35, page 50
System Operating Procedure 2.2.68.1, Reactor Recirculation System Operations, Revision 89, page 44
Alarm Procedure 2.3-9-4-3, Revision 33, page 63

References to be provided to applicants during exam:
None.
COR002-22-02, Reactor Recirculation, Revision 35, Enabling objective 6.b
Learning Objective: Document learning objective if possible.

Question Source:	Bank #	5339
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
295019 (APE 19) Partial or Complete Loss of Instrument Air / 8 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: (CFR: 41.10 / 43.5 / 45.13) AA2.02 Status of safety-related instrument air system loads	**Tier#**	1
	Group#	1
	K/A #	295019 AA2.02
	Rating	3.6
	Revision	0
Revision Statement:		

Question 40

The plant is at 100% power.

SGT A is operating to support a quarterly HPCI surveillance with the following conditions:

- Torus venting is aligned through SGT A
- SGT A discharge header flow is 1000 scfm

How are SGT A discharge flow and Torus flow path ultimately affected by the loss of instrument air?

A. SGT A flow rises, Torus venting continues

B. SGT A flow rises, Torus vent path isolates

C. SGT A flow lowers, Torus venting continues

D. SGT A flow lowers, Torus vent path isolates

Answer: B
Explanation: With SGT A flow initially at 1000 scfm, air operated dP control valve SGT-DPCV-546A, SGT A FLOW/RX BLDG DP CONTROL is in an intermediate position. SGT-DPCV-546A fails fully open upon loss of instrument air, and discharge flow would rise to ~2000 scfm. Torus exhaust outboard isolation valve PC-AO-245 is aligned open to establish the vent path from the Torus, but it fails closed on a loss of instrument air. Therefore, the Torus vent path isolates.
Distracters: Answer A is correct with respect to SGT A flow rising. It is plausible with respect to the Torus vent path because many valves fail open upon loss of instrument air, including SGT-AO-249, SGT-AO-251 and SGT-DPCV-546A in the flow path for the stated alignment. It is wrong because PC-AO-245 fails closed on a loss of instrument air, isolating the Torus vent path. Answer C is plausible with respect to SGT A flow because many valves fail closed upon loss of instrument air, including PC-AO-245 in the flow path for the stated alignment. The examinee who believes SGT A inlet and outlet valves SGT-AO-249 and SGT-AO-251 or flow

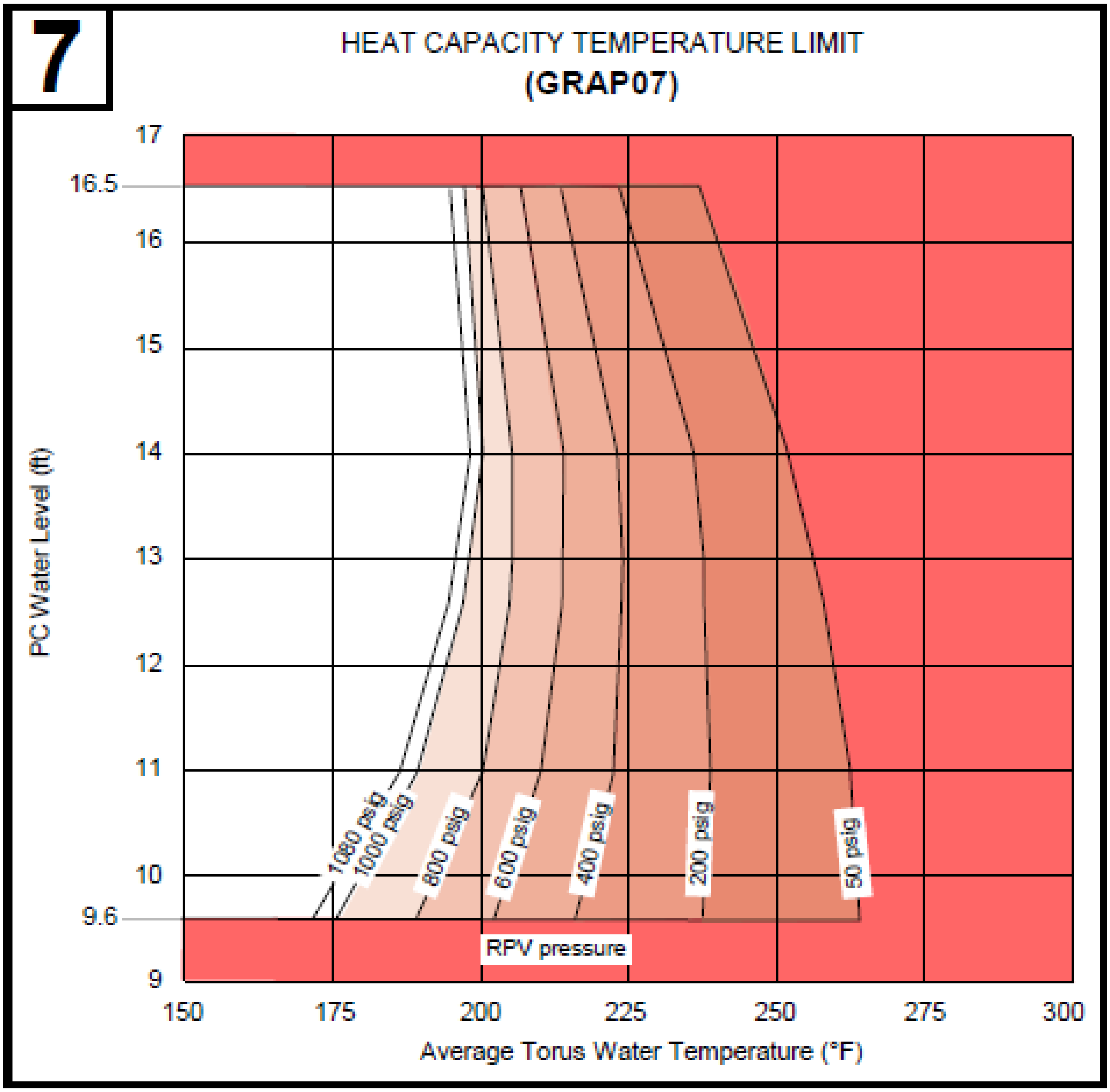
control valve SGT-DPCV-546A fail closed upon loss of instrument air will choose this answer. It is plausible and wrong with respect to the Torus vent path for the same reason stated for distractor A. Answer D is plausible and wrong with respect to SGT A flow for the reasons given for distractor C.		
Technical References: Procedure 2.2.73 [Standby Gas Treatment System](Rev 60), procedure 5.2AIR [Loss of Instrument Air](Rev 23), B&R Drawing (P&ID) 2022 sh 1 [Primary Containment Cooling & Nitrogen Inerting System]		
References to be provided to applicants during exam: none		
Learning Objective: COR002-28-02 Obj. LO-10d, Predict the consequences of the following on the Standby Gas Treatment System: Plant Air System failures		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)((7)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	
PSA Applicability		
Top 10 Risk Significant System – Primary Containment Isolation (PC-AO-245)		

Examination Outline Cross-Reference	Level	RO
295026 (EPE 3) Suppression Pool High Water Temperature	Tier #	1
	Group #	1
	K/A #	EK2.06
Knowledge of the interrelations between SUPPRESSION POOL HIGH WATER TEMPERATURE and the following:	Rating	3.5

EK2.06 Suppression pool level

Question 41

The reactor has failed to scram.
Reactor pressure is 400 psig, stable.
Suppression Pool temperature is 220°F, stable.
Which Suppression Pool water level exceeds the Heat Capacity Temperature Limit (HCTL) under these conditions?



A. 11 feet

B. 12.5 feet

C. 14 feet

D. 15.5 feet

Answer: D

Explanation:

High Torus water temperature is addressed by EOP-3A. Emergency Depressurization is required if the HCTL (Graph 7) of the suppression pool is exceeded. The unsafe zone of the HCTL is to the right of the curve corresponding to actual RPV pressure. At a RPV pressure of 400 psig and Suppression Pool Temperature of 220°F, of the levels listed in the answers, the only SP level value that is in the unsafe zone of the HCTL is 15.5 feet.

All distractors have plausibility because the shape of the HCTL curve for 400 psig. RPV pressure becomes more limiting at lower than normal SP levels, just as it becomes more limiting at higher than normal SP levels.

Technical References: EOP-6A, Rev 19, EOPSAG Graph 7 (HCTL)(Rev 19)

References to be provided to applicants during exam:
EOPSAG Graph 7

Learning Objective:
INT008-06-18, EOP and SAG Graphs and Cautions, Revision 28, Enabling bjective 3

Question Source:	Bank #	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	CNS 2017-03 Q13
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
295018 (APE 18) Partial or Complete Loss of CCW	Tier #	1
	Group #	1
	K/A #	AK1.01
Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER:	Rating	3.5
AK1.01 Effects on component/system operations		

Question 42

Given the following:

- Reactor power is 100%
- REC pumps 1A, 1B, and 1C are running
- REC Pump 1A Low Disch Alarm is in alarm
- REC System Low Pressure Alarm is in alarm
- REC Header pressure is 61 psig and slowly lowering
- The REC system responds as designed

In accordance with Emergency Procedure 5.2REC, Loss of REC, ___(1)___ will automatically close after a ___(2)___ second time delay?

A. (1) REC-AO-710, RWCU Non-Regen HX Inlet
(2) 20

B. (1) REC-AO-710, RWCU Non-Regen HX Inlet
(2) 40

C. (1) REC-MO-1329, Augmented Rad Waste Supply
(2) 20

D. (1) REC-MO-1329, Augmented Rad Waste Supply
(2) 40

Answer: D

Explanation:
A is wrong. Part 1 is wrong but plausible because REC-AO-710 is a REC system valve that needs to be closed during a Loss of REC but doesn't close automatically. Part 2 is wrong but plausible because 20 seconds is the time delay for a REC pump in standby to start.

B is wrong. Part 1 is wrong but plausible for the reason stated above. Part 2 is correct.

C is wrong. Part 1 is correct. Part 2 is wrong but plausible for the reason stated above.

D is correct because it will automatically close after 40 seconds less than 61 psig.

Technical References:
Emergency Procedure 5.2REC, Loss of REC, Revision 19, sections 1-3, page 1

References to be provided to applicants during exam:
None.

Learning Objective: Document learning objective if possible.
COR002-19-02, Reactor Equipment Cooling, Revision 31, Enabling objective 11.c

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41.8	

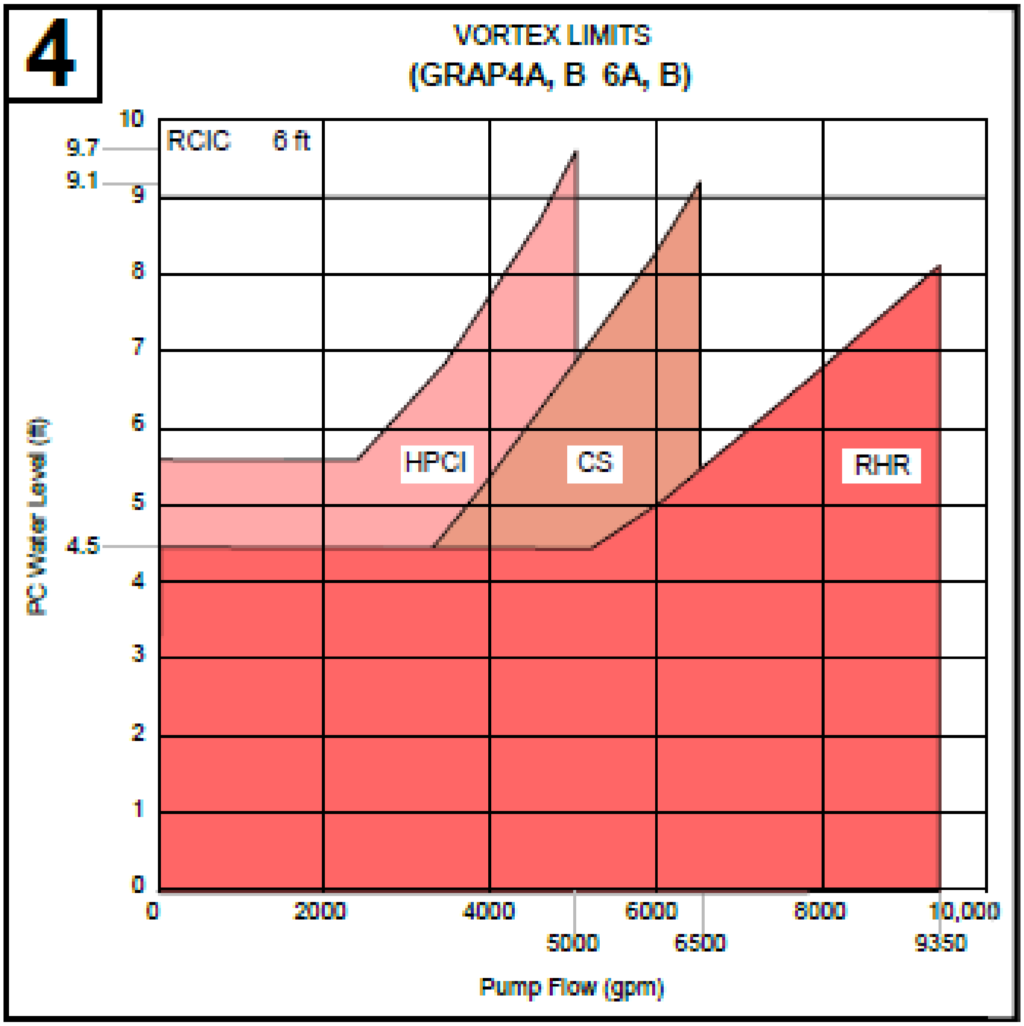
Examination Outline Cross-Reference	Level	RO
295030 (EPE 7) Low Suppression Pool Water Level	Tier #	1
	Group #	1
	K/A #	2.1.25
2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc.	Rating	3.9

Question 43

The plant has experienced an earthquake resulting in the following conditions:

- RPV level is +10 inches and stable.
- RPV Pressure is 200 psig and stable (Controlled by HPCI).
- Torus level is 11.5 feet and lowering fast.
- RHR Pumps A & D flows are 6500 gpm each.
- CS Pump A flow is 6000 gpm.
- HPCI flow is 4000 gpm in pressure control.
- RCIC flow is 400 gpm.

IAW Caution 3, which system will reach its vortex limit FIRST as Torus level continues to lower?



A. RCIC

B. HPCI

C. Core Spray

D. RHR

Answer: C

Explanation:
Caution 3 reminds the operator of potential equipment damage when operating above NPSH & Vortex limits. The vortex limits are defined to be the lowest suppression pool water level above which air entrainment is not expected to occur in pumps taking suction on the pool. These levels are functions of pump flow. Exceeding the limits can lead to air entrainment at the pump suction strainers. Since Core Spray is operating at 6000 gpm, its vortex limit would be reached first at ~ 8.5 feet. RCIC is reached at 6 feet. RHR would be reached at 5.5 feet. HPCI is not allowed to be operated below 11 feet due to direct steam exhaust to the torus air space.

B is incorrect due to CS reaching the vortex limit first. This choice is plausible due to not recognizing HPCI being required to be secured below 11’ torus level due to exhaust steam discharging directly to the torus air space. The candidate that does not recognize the requirement to secure HPCI or considers 9.1’ (specifically identified next to HPCI) would select this answer.
A is incorrect due to CS reaching the vortex limit first. This choice is plausible due to not recognizing RCIC vortex limit of 6’. The candidate that does not recognize RCICs vortex limit of 6’ would select this answer.
D is incorrect due to CS reaching the vortex limit first. This choice is plausible due to CS & RHR vortex limits being easily confused. The candidate that confuses CS & RHR vortex limits would select this answer.

Technical References:
AMP-TBD00 Tech. Basis – App. B (CNS PSTG/SATG Appendix B Technical Bases), Rev. 10EOP 3A (Primary Containment Control), Rev. 18
Emergency Operating Procedure 5.8 Attachment 2 (EOP and SAG Graphs), Rev. 17

References to be provided to applicants during exam: EOP Vortex Limits (Graphs 4A,B 6A,B)

Technical References:
AMP-TBD00 Tech. Basis – App. B (CNS PSTG/SATG Appendix B Technical Bases), Rev. 10EOP 3A (Primary Containment Control), Rev. 18
Emergency Operating Procedure 5.8 Attachment 2 (EOP and SAG Graphs), Rev. 17

Learning Objective: OPS EOP and SAG Graphs and Cautions / INT008-06-18, Revision 28, Enabling Objective 3

Question Source:	Bank #	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	CNS 2015-12 Q15
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
295016 (APE 16) Control Room Abandonment	Tier #	1
	Group #	1
Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following:	K/A #	AK2.01
	Rating	4.4

AK2.01 Remote shutdown panel:

Question 44

Which one of the following actions can be performed ENTIRELY by the ASD Operator from the Alternate Shutdown panel in the event the Control Room becomes uninhabitable due to toxic fumes during Mode 1?

A. Prevent RCIC injection

B. Operate all Low-Low Set valves

C. Place HPCI in pressure control mode

D. Place RHR Suppression Pool Cooling in service

Answer: C

Explanation:
Of the actions listed, only HPCI has all controls necessary for pressure control mode located on ASD panels. Other listed actions are either fully or in part only performed from locations other than the ASD panel room.
A is wrong but plausible because this action is performed from outside of the control room for control room abandonment. Like ASD panel actions, this action is also performed from a location in the control building. It is wrong because it is accomplished by the control building operator placing the RCIC ISOLATION switch to ISOLATE in the Auxiliary Relay Room.
B is wrong but plausible because there are two LLS SRVs and there are controls for three SRVs on the ASD panel. It is wrong because only one LLS SRV (71F) can be controlled from the ASD panel.
C is correct. Of the actions listed, only HPCI has all controls necessary for pressure control mode located on ASD panels. Other listed actions are either fully or in part only performed from locations other than the ASD panel room.
D is wrong plausible because controls for all RHR loop B valves necessary to establish the SPC lineup are located on the ASD panel. It is wrong because RHR Pump D used for SPC is not controlled from the ASD panel but must be started locally at the pump breaker in the critical switchgear room, and RHRSW to RHR B heat exchanger is aligned from locations other than the ASD room.

Technical References:
5.1ASD, Alternate Shutdown, Revision 19, COR002-16-02, Ops Nuclear Pressure Relief, Revision 21

References to be provided to applicants during exam:
None.

Learning Objective:
COR002-34-02, Alternate Shutdown, Revision 24, Enabling objective 9

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	CNS 2020-08 Q45
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
700000 (APE 25) Generator Voltage and Electric Grid Disturbances	Tier #	1
	Group #	1
	K/A #	AA1.01
Ability to operate and/or monitor the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES:	Rating	3.6
AA1.01 Grid frequency and voltage		

Question 45

REFERENCE PROVIDED

Due to grid instabilities, the following condition exists:

- The voltage regulator is in OFF
- Real power is 750 MW
- Reactive power is -350 MVAR
- Hydrogen pressure is 50 psig

In order to return the generator to within the limits of the main generator capability curve, operators will place the __(1)___ switch in the ___(2)___ direction on Panel C.

A. (1) GEN BASE ADJUST
(2) RAISE

B. (1) GEN BASE ADJUST
(2) LOWER

C. (1) GEN VOLTAGE ADJUST
(2) RAISE

D. (1) GEN VOLTAGE ADJUST
(2) LOWER

Answer: A

Explanation:
A is correct because the power factor is outside of the normal .85 lagging to .95 leading band – to correct it procedure 2.2.14, 22 kV Electrical System, states that in manual mode the voltage regulator may be controlled per step 11.2 To pick up positive (OUT) MAIN GENERATOR MVAR (clockwise), place GEN BASE ADJUST switch to RAISE.

B is wrong because the direction the GEN BASE ADJUST switch should be taken is RAISE not LOWER; is plausible because this is the correct switch.

C is wrong because the switch to be operated is the GEN BASE ADJUST switch not the GEN VOLTAGE ADJUST switch due to the manual mode of operation; is plausible because RAISE is the correct direction and because this would be the correct switch if the voltage regulator were in Automatic.

D is wrong because this is the incorrect switch and the incorrect direction; is plausible because the GEN VOLTAGE ADJUST switch adjusts MVAR in the AUTOMATIC mode of operation; is plausible because this switch usually adjusts MVAR at power and if the operating parameters are graphed incorrect then lower might be the desired correction.

Technical References:
Operations Procedure 2.2.14, 22 kV Electrical System, section 11 and Attachment 1, pages 10 and 45

References to be provided to applicants during exam:
Main Generator Capability Curve (PMIS01)

Learning Objective:
COR001-13-01, Main Generator and Auxiliaries, Revision 35, Enabling objective 11.b

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 2017 NRC Exam #63
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/**Analysis**	 4
10CFR Part 55 Content:	55.41.5	

Examination Outline Cross-Reference	Level	RO
295031 (EPE 8) Reactor Low Water Level	Tier #	1
	Group #	1
Knowledge of the interrelations between REACTOR LOW WATER LEVEL and the following:	K/A #	EK2.03
	Rating	4.2

EK2.03 Low pressure core spray

Question 46

The plant was operating at 100% power when a LOCA has occurred.

(1) The Core Spray Pumps will automatically start when RPV level drops to what minimum level?
(2) Following Core Spray auto actuation, what is the maximum Reactor pressure at which the Core Spray injection valves open?

A. (1) -42 inches
(2) 291 psig

B. (1) -42 inches
(2) 436 psig

C. (1) -113 inches
(2) 291 psig

D. (1) -113 inches
(2) 436 psig

Answer: D

Explanation:

The Core Spray Pumps will automatically start when RPV level is -113 inches or Drywell pressure = 1.84 psig.

The inboard and outboard Core Spray injection valves (MO-11/12A/B) are automatically opened and interlocked open when Core Spray is initiated and reactor pressure is $\geq$ 291 psig and $\leq$ 436 psig. (Core Spray injection piping upstream of MO-11A(B) is designed for 500 psig pressure.)

A is wrong. -42 inches is the Level 2 low-low setpoint at which RCIC and HPCI initiate, but CS doesn't initiate until Level 1 low-low-low = -113 inches. Also the maximum allowed value below which CS injection valves are allowed to open is 436 psig. 291 psig is the minimum allowed value.

B is wrong. Part (2) is correct, but for part (1) -42 inches is the Level 2 low-low setpoint at which RCIC and HPCI initiate, but CS doesn't initiate until Level 1 low-low-low = -113 inches.

C is wrong. Part (1) is correct, but for part (2) the maximum allowed value below which CS injection valves are allowed to open is 436 psig. 291 psig is the minimum allowed value.

D is correct.

Technical References:
COR002-06-02, Core Spray System, Revision 27

References to be provided to applicants during exam:
None.

Learning Objective:
COR002-06-02, Core Spray System, Revision 27, Enabling objectives 5.a and 5.h

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	2
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
295025 (EPE 2) High Reactor Pressure	Tier #	1
	Group #	1
Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following:	K/A #	EK2.09
	Rating	3.9
EK2.09 Reactor power		

Question 47

The plant is operating at 100% power, when an outboard MSIV disc-stem separation occurs.

Which one of the following is the FIRST to automatically Scram the reactor due to this failure?

A. MSIV closure

B. APRM High Flux

C. Low RPV water level

D. High Reactor Pressure

Answer: B

Explanation:
MSIV stem disc separation results in the MSIV disc rapidly closing causing a high pressure transient collapsing core steam bubbles and a spike in reactor power. This power spike is seen by all the APRMS and a reactor Scram on high flux results. The outboard MSIV's are located in the steam tunnel further down the steam lines than the inboard MSIVs so the pressure perturbation is less than an inboard MSIV stem disc separation.

A is incorrect because APRM high flux automatically Scrams the reactor. Plausible due to closing one MSIV at rated power causing steam flow in the remaining steam lines to rise. The candidate that believes the rise in flow through the other Main Steam Lines will cause a Group 1 isolation would select this answer.
B is correct.
C is incorrect because APRM high flux automatically Scrams the reactor. Plausible because RPV low level is normally received following a reactor Scram. The candidate confuses the sequence of events would select this answer.
D is incorrect because APRM high flux automatically Scrams the reactor. Plausible because rising pressure results from an MSIV stem disc separation. The candidate that confuses the sequence of events would select this answer.

Technical References:
LER 89-001, Unplanned automatic Scram due to APRM high flux resulting from separation of an MSIV disc from its stem

References to be provided to applicants during exam:
None.

Learning Objective:

COR002-14-02, Main Steam, Revision 30, Enabling Objective 6.d

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	CNS 2015-12 Q12
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 3
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
295021 (APE 21) Loss of Shutdown Cooling	Tier #	1
	Group #	1
Knowledge of the operational implications of the following concepts as they apply to LOSS OF SHUTDOWN COOLING:	K/A #	AK1.04
	Rating	3.6
AK1.04 Natural circulation		

Question 48

Given the following:

- The plant is in Mode 5
- The Reactor Pressure Vessel head is removed
- A loss of shutdown cooling has occurred
- RHR train A was in service in SDC mode
- Refueling in progress
- Spent fuel pool gates are removed and all reactor well weirs are overflowing
- Fuel pool cooling is in operation
- RHR-TR-131, RHR Heat Exchanger Temperature Recorder, is in service

In accordance with procedure OP 2.2.69.2, RHR System Shutdown Operations, operators will establish natural circulation by throttling open _________.

A. RHR-MO-66A, HX BYPASS VLV

B. FPC-30, FUEL STORAGE POOL RECIRC

C. FPC-33, REACTOR WELL RECIRCULATION

D. RHR-MO-39A, SUPPR POOL COOLING/TORUS SPRAY OUTBD VLV

Answer: C

Explanation:
A is wrong because valve FPC-33 initiates natural circulation; is plausible because this valve is opened when placing RHR train A in SDC mode per the procedure in effect.
B is wrong because valve FPC-33 initiates natural circulation; is plausible because it is the correct system and is a valve operated by the procedure in effect to intertie the fuel pool cooling system with the RHR system.
C is correct because step 17.8 for Establishing Natural Circulation directs operators to throttle open FPC-33.
D is wrong valve FPC-33 initiates natural circulation; is plausible because it is a valve in the system that was providing SDC and is aligned in accordance with the procedure in effect when placing RHR train A in service (closed).

Technical References:
Procedure 2.2.69.2, RHR System Shutdown Operations, rev 106, sections 4, 15, and 17

References to be provided to applicants during exam:

None.

Learning Objective: Document learning objective if possible.
COR002-23-02, Residual Heat Removal System, Revision 36, Enabling objective 9.d

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	3
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.8	

Examination Outline Cross-Reference	Level	RO
295003 (APE 3) Partial or Complete Loss of AC Power	Tier #	1
	Group #	1
	K/A #	AK1.02
Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER:	Rating	3.1
AK1.02 Load shedding		

Question 49

The plant is at 100% power.

Diesel generator 2 has been removed from service for a 12-year maintenance overhaul.

A loss of all offsite power occurs.

The supplemental diesel generator cannot be started.

Which of the following motor control centers will need to be manually loaded on the available bus?

A. MCC-N

B. MCC-S

C. MCC-CA

D. MCC-MR

Answer: A

Explanation:
A is correct because this is one of a series of MCC that have an undervoltage coil that will trip the supply breaker if undervoltage is detected for greater than 5.5 seconds. The diesel should take ~10 seconds to start and load onto the bus which will cause the MCC to trip. This MCC is on the 1F bus.
B is wrong because this MCC doesn't have an undervoltage coil. This MCC is also on the 1G bus which can't be powered due to DG 2 being out of service.
C is wrong because this MCC is on the 1F but does not have an undervoltage coil. This MCC will not trip on a LOOP.
D is wrong because this MCC is on the 1G bus but does have an undervoltage coil.

Technical References:
COR001-01-01, AC Electrical Distribution, Revision 50, p. 109

References to be provided to applicants during exam:
None.

Learning Objective:
COR001-01-01, AC Electrical Distribution, Revision 50, Enabling Objective 15.a

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	3
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.8	

Examination Outline Cross-Reference	Level	RO
295028 (EPE 5) High Drywell Temperature Ability to operate and/or monitor the following as they apply to HIGH DRYWELL TEMPERATURE: EA1.03 Drywell cooling system	Tier#	1
	Group#	1
	K/A #	295028 EA1.03
	Rating	3.9
	Revision	1
Revision Statement:		

Question 50

The Plant has scrammed due to high drywell pressure.

- Drywell Pressure is 13 psig
- CRS has entered EOP 3A

Drywell Fan coil units will have their control switches taken to **OVERRIDE** when drywell temperature cannot be maintained below __(1)__ºF.

IAW EOP 3A, secure the drywell FCUs when __(2)__ .

A. (1) 135
(2) PC containment H2 concentration cannot be maintained below 1% concentration

B. (1) 135
(2) Drywell temperature cannot be maintained below 280°F and Drywell Sprays are required

C. (1) 150
(2) PC containment H2 concentration cannot be maintained below 1% concentration

D. (1) 150
(2) Drywell temperature cannot be maintained below 280°F and Drywell Sprays are required

Answer: D		
Explanation: D is correct because EOP 3A states at 150 ºF to OPERATE ALL AVAILABLE DRYWELL COOLING (defeat isolation interlocks if necessary) so this is where you take the control switch to override and defeat the interlocks. At 280 ºF drywell sprays are required and this requires the drywell FCUs to be secured. Above 12.1 psig DWZL I always met so it makes the graph for meeting DWZL not required.		
Distracters: Answer A Part 1 is plausible because the drywell cooling system is designed to maintain temperature at 135 ºF but is wrong because EOP 3A doesn't require ED until temperature cannot be restored and maintained below 340°F, Part 2 is plausible examinees can confuse the requirement to secure PC ventilation with securing drywell FCUs. Answer B Part 1 is plausible for the reasons stated in distractor A, Part 2 is correct. Answer C Part 1 is correct. Part 2 is plausible for the reasons stated in distractor A		
Technical References: EOPSAG [EOP/SAG Graphs] (REV 17) EOP 3A [Primary Containment Control] (REV 18)		
References to be provided to applicants during exam: None		
Learning Objective: INT008-06-13Obj 4. State the basis for primary containment control actions as they apply to the following. a. Specific setpoints		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	
PSA Applicability: N/A		

Examination Outline Cross-Reference	Level	RO
295005 (APE 5) Main Turbine Generator Trip	Tier #	1
	Group #	1
Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP:	K/A #	AA2.04
	Rating	3.7

AA2.04 Reactor pressure

Question 51

The plant is operating at near rated power when the following occurred:

- Main Generator MVARs begin to steadily increase
- Doniphan contacts you to inform you the grid is becoming unstable
- PCB 3310 trips open followed by PCB 3312
- RV-71F is inoperable and cannot be opened

After the scram is complete reactor pressure will be automatically controlled between _____.

A. 876 and 1040 psig

B. 835 and 1010 psig

C. 926 and 1025 psig

D. 1004 and 1034 psig

Answer: B

Explanation:
This is a load reject situation. RPV pressure would rapidly rise before it tripped on RPV high pressure. On a trip like this low-low set (LLS) should activate since an SRV should open and RPS high pressure signal is received. Didn't move "psig and" and "psig." to the stem because I think the question is easier to read this way. Tried to encapsulate the chart below for describing the range of LLS.

A is wrong because of Rev 3 statement.
B is correct because of Rev 3 statement.
C is wrong because of Rev 3 statement.
D is wrong because of Rev 3 statement.

Technical References:
Procedure 2.2.77.1, Digital Electro-Hydraulic (DEH) Control System, Revision 42, p. 49

References to be provided to applicants during exam:
None.

Learning Objective:
COR002-16-02, Nuclear Pressure Relief, Revision 21, Enabling objectives 1.c and 3.j.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	4
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
295037 (EPE 14) Scram Condition Present and Reactor Power Above APRM Downscale or Unknown	Tier #	1
	Group #	1
	K/A #	EA1.06
	Rating	4.1

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

EA1.06 Neutron monitoring system

Question 52

Given the following:

- The plant is starting up
- The Mode Switch is in STARTUP
- Reactor power is 6%
- Reactor pressure is 830 psig
- A Reactor Power spike occurs and power peaks at 15.5%
- Annunciator APRM UPSCALE is in alarm
- Annunciator APRM RPS CH A UPSCALE TRIP OR INOP is in alarm
- Annunciator APRM RPS CH B UPSCALE TRIP OR INOP is in alarm

With no operator action, what is the final plant condition?

A. Scram due to reactor power

B. Scram due to MSIV closure

C. Rod block due to reactor power

D. No automatic action occurs with the mode switch in STARTUP

Answer: A

Explanation:
A is correct because power exceeded the fixed trip reference value of 14.5% causing a reactor scram.
B is wrong because the reactor will trip on Power in this condition; is plausible because at 830 psig if the mode switch were taken to RUN, then the reactor would trip on a MSIV closure.
C is wrong because power exceeded the fixed trip reference value of 14.5% which will scram the reactor (final condition); is plausible because a rod block signal came in at 11.5% power.
D is wrong because automatic action will occur for the given plant conditions; is plausible because the fixed trip reference is higher in RUN (125%) than in STARTUP/HOT STANDBY (14.5%).

Technical References:

Instrumentation Operations Procedure 4.1.3, Average Power Range Monitoring System, rev 26, pages 7-8

References to be provided to applicants during exam:
None.

Learning Objective:
COR002-01-02, Average Power Range Monitor , rev 27, Enabling objective 13.c

Question Source: (note changes; attach parent)	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
295023 (APE 23) Refueling Accidents	Tier #	1
	Group #	1
Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS:	K/A #	AA1.01
	Rating	3.3

AA1.01 Secondary containment ventilation

Question 53

Day 10 of a refueling outage and a valid 9-4-1/E-4 RX BLDG VENT HI-HI RAD signal is received from an accident on the refueling floor.

EOP 5A was entered. The refueling floor was secured and now the highest reading Reactor Building vent exhaust rad monitor is 7 mR/hr and decreasing.

Both trains of standby gas treatment failed to start in automatic and manually. You've been directed to restart reactor building ventilation.

Per 5.8.20, EOP Plant Temporary Modifications, what are the required actions to restart reactor building ventilation?

A. Install EOP jumpers
Reset radiation monitors

B. Reset radiation monitors
Depress PCIS GROUP 6 DIV 1 and DIV 2 ISOLATION reset pushbuttons

C. Install EOP jumpers
Reset radiation monitors
Turn ISOL RESET CHANNEL A and CHANNEL B handswitches to reset

D. Reset radiation monitors
Turn ISOL RESET CHANNEL A and CHANNEL B handswitches to reset
Depress PCIS GROUP 6 DIV 1 and DIV 2 ISOLATION reset pushbuttons

Answer: D

Explanation:
A is wrong because due to the current readings of less than 10 mR/hr jumper installation isn't required. In addition to resetting the rad monitors you would also have to reset the group 6 isolation. Because there is a note in procedure 5.8.20 that talks about the interlocks are not defeated when the rad monitors are reset this is plausible if someone misremembers the note.
B is wrong but plausible because in accordance with the training material these are the only **required** steps to reset the group six. However, the procedure requires turning the handswitches reset is required by the procedure.
C is wrong because jumper installation is not required. The 2 other actions are from procedure 5.8.20. According to the training material the isolation handswitch reset is not necessary since an F or an A signal was not received at the same time. Plausible if someone only remembers the procedure steps and doesn't recognize an F or A signal was

not generated. If the question was IAW 5.8.20 and radiation levels higher than 10 mR/hr then C would be correct.
D is correct.

Technical References:
COR002-03-02, Containment, Revision 35, p. 48
5.8.20, EOP Plant Temporary Modifications, Revision 21, Section 6

References to be provided to applicants during exam:
None.

Learning Objective: COR002-03-02, Containment, Revision 35, Enabling objective 13.e

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	2
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference		Level	RO
295006 (APE 6) Scram		Tier #	1
		Group #	1
Knowledge of the reasons for the following responses as they apply to SCRAM:		K/A #	AK3.01
		Rating	3.8

AK3.01 Reactor water level response

Question 54

The plant is running at 100% power. Reactor water level is being controlled in 3 element control. All RPV water level instruments are indicating correctly.

Then, a reactor scram occurs.

RPV pressure is stable and being automatically controlled by the turbine bypass valves.

The reactor vessel level control system (RVLCS) level setpoint is set to ___(1)___ inches and then commences ramping to +25" at ___(2)___ inches per minute.

A. (1) -15
(2) 2.5

B. (1) -15
(2) 5

C. (1) +5
(2) 2.5

D. (1) +5
(2) 5

Answer: B

Explanation:
A is incorrect. It is plausible because part 1 is correct since the stem states all RPV level instruments are reading correctly. Part 2 is plausible because 2.5" per minute is the ramp rate if NBI-LT-LT59D were invalid
B is correct. According to the training material, if NBI-LT-LT59D is valid, level setpoint is set to -15" and then commences ramping to 25" at 5" per minute.
C is incorrect. It is plausible because both parts would be correct if NBI-LT-LT59D were invalid.
D is incorrect. Part 1 is plausible because this is correct initial setpoint if NBI-LT-LT59D were invalid. Part 2 is the correct ramp rate.

Technical References:
COR002-32-02, Reactor Vessel Level Control, Revision 24, Page 22

References to be provided to applicants during exam:
None.

Learning Objective:
COR002-32-02, Reactor Vessel Level Control, Revision 24, Enabling objective 6.c

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 3
10CFR Part 55 Content:	55.41.5	

Examination Outline Cross-Reference	Level	RO
295038 (EPE 15) High Offsite Radioactivity Release Rate	Tier #	1
	Group #	1
	K/A #	EK3.03
Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE: (CFR: 41.5 / 45.6)	Rating	3.7

EK3.03 Control room ventilation isolation

Question 55

Given the following:

- A LOCA has occurred
- Alarm RX BLDG VENT HI-HI RAD, 9-4-1/E-4 is in due to a valid high radiation signal

The Control Room Emergency Filter system is actuated due to a ___(1)___ setpoint of ___(2)___ mr/hr. **(Use actual plant values)**

A. (1) Reactor Building Exhaust Plenum High-High Radiation
(2) 5

B. (1) Reactor Building Exhaust Plenum High-High Radiation
(2) 10

C. (1) Control Room High Radiation
(2) 5

D. (1) Control Room High Radiation
(2) 10

Answer: B

Explanation:
A is wrong because Part 2 is wrong for the reason stated in distractor A. Part 1 is correct.
B is correct.
C is wrong due Control Room ventilation isolates due to RB Exh Hi-Hi Rad setpoint of 10 mr/hr. This choice is plausible due to Control Room Hi Rad requiring entry into procedure 5.1RAD which provides guidance to manually align Control Room ventilation and CR Hi Rad providing isolation signals at other BWRs. Part 2 is plausible because this is the alarm setpoint for RB Exh Hi.
D is wrong due Control Room ventilation isolates due to RB Exh H-Hi Rad setpoint of 10 mr/hr. Part 1 is plausible because of the reason stated in distractor A. Part 2 is correct.

Technical References:
COR001-08-01, Heating, Ventilation, and Air Conditioning, Revision 29, pages 87, 103
COR002-03-02, Containment, Revision 35, page 41, 46-48
COR001-18-01, Radiation Monitoring, Revision 28, page 70

References to be provided to applicants during exam:

None.

Learning Objective:
COR002-03-02, Containment, Revision 35, Enabling objective 6.s

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 2015-11 NRC exam Q 18
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 3
10CFR Part 55 Content:	55.41.5	

Examination Outline Cross-Reference	Level	RO
295004 (APE 4) Partial or Total Loss of DC Power	Tier #	1
	Group #	1
	K/A #	AK3.03
Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER:	Rating	3.1

AK3.03 Reactor SCRAM

Question 56

With the reactor in mode 2, a loss of both Divisions of 24 VDC occurs.

What is ONE reason for an AUTOMATIC reactor SCRAM for this event?

A. All IRM units deenergize

B. Group 6 Isolation occurs

C. Loss of both reactor recirc pumps

D. Both ARI Scram solenoid valves energize

Answer: A

Explanation:
A is correct because the actual reason for the reactor scram is IRM inop trips on all IRMs. Since this is an RPS scram and there is power to the 125VDC bus and they will energize causing the scram.
B is incorrect but plausible because the reactor is in mode 2, and so the mode switch would be in START & Hot Stby. With this condition and a loss of both 24 VDC buses, a Group 6 isolation occurs but does not result in a reactor SCRAM.
C is wrong because the 125 VDC system provides power to operate the Recirc Pump drive motor and field breaker for the MG set. Therefore, they are unaffected by a loss of 24 vdc power.
D is wrong because the ARI solenoids are powered from 125vdc so are not affected by loss of 24 vdc power.

Technical References:
COR002-07-02, DC Electrical Distribution Student Text, Revision 35
COR002-21-02, OPS Reactor Protection System Student Text, Revision 25

References to be provided to applicants during exam:
None.

Learning Objective:
COR002-07-02, DC Electrical Distribution Student Text, Revision 35, Enabling objective 8.j

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

	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
600000 (APE 24) Plant Fire On Site	Tier #	1
	Group #	1
Knowledge of the reasons for the following responses as they apply to PLANT FIRE ON SITE:	K/A #	AK3.04
	Rating	2.8
AK3.04 Actions contained in the abnormal procedure for plant fire on site		

Question 57

There is a fire in the control room and 5.4FIRE S/D has been entered. Before leaving Control Room, the Control Room Operator must place RFLO pumps in PULL-TO-LOCK in a maximum of _____ seconds following the SCRAM.

A. 10

B. 20

C. 47

D. 57

Answer: D

Explanation:
A is wrong. Plausible because 5.4FIRE S/D directs that RHR-MO-34B must be opened within 10 seconds after pump start, if RHR MO 16B is closed.
B is wrong. Plausible because 5.4FIRE S/D directs placing the two Diesel Generator LOCAL GOVERNOR MOTOR CONTROL SWITCHES, to RAISE for 20 seconds.
C is wrong. Though the time is not based in 5.4FIRE S/D, it is plausible because it there is a 10 second difference between C and D, maintaining a distractor balance similar to A and B.
D is correct.

Technical References:
Procedure 5.4 Fire S-D, Fire induced shutdown from outside the control room, revision 76, pg. 2.

References to be provided to applicants during exam:
None.

Learning Objective:
INT032-01-34 OPS CNS Abnormal Procedures (RO) Fire, revision 10, enabling objective G

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No

Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference		Level	RO
295024 High Drywell Pressure		Tier #	1
		Group #	1
2.4.31 Knowledge of annunciator alarms, indications, or response procedures		K/A #	2.4.31
		Rating	4.2

Question 58

Given the following:

- Reactor power is being raised from 50% to 75%
- Drywell Temperature is 100ºF and slowly rising
- Drywell pressure is 0.65 psig and slowly rising
- 9-5-2/F-3, HIGH DRYWELL PRESSURE is in alarm

What is the correct operator action?

A. Trip and isolate both reactor feedwater pumps.

B. Scram and enter Procedure 2.1.5, REACTOR SCRAM.

C. Perform rapid power reduction and reduce core flow to $40x10^6$ lbs/hr.

D. Check drywell fan coil unit operation and enter procedure 2.4PC, Primary Containment Control.

Answer: D

Explanation:
A is wrong because Abnormal Procedure 2.4MC-RF has an entry condition if a leak in the feedwater or condensate system is known - at the rate drywell temperature and pressure are rising, a feedwater leak would be small and the control room operator could not positively determine it to be a feedwater leak; is plausible because the symptoms indicate a small leak could be occurring.
B is wrong because Alarm 9-5-2/F-3, HIGH DRYWELL PRESSURE has a note that states, "IF drywell pressure cannot be maintained below 1.5 psig, THEN SCRAM and enter Procedure 2.1.5," and no information in the stem would lead the operator to believe that passing drywell pressure of 1.5 psig is imminent or cannot be corrected; is plausible because it is direction from the applicable alarm response procedure just not for the given conditions.
C is wrong because Procedure 2.4PC does direct this action but at a drywell pressure of 0.75 psig not 0.65 psig; is plausible because lowering RR pump speed to lower power will lower the heat input into the drywell as the RR pumps are a large heat load on containment so the operator that is aware of this characteristic would select this answer.
D is correct because alarm 9-5-2/F-3 provide guidance for checking FCU operation and because procedure 2.4PC gives direction to ensure all FCU control switches are in "run".

Technical References:
Alarm 9-5-2/F-3, HIGH DRYWELL PRESSURE

References to be provided to applicants during exam:
None.

Learning Objective:
COR002-03-02, Containment System , Revision 35, Enabling objectives 16.a and 16.b

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	CNS 2014-07 Q11
Question Cognitive Level:	Memory/Fundamental **Comprehensive**/Analysis	 3
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
295034 (EPE 11) Secondary Containment Ventilation High Radiation	Tier #	1
	Group #	2
	K/A #	EK1.02
Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION:	Rating	4.1

EK1.02 Radiation releases

Question 59

An offsite radioactivity release is in progress through the reactor building ventilation. The high radiation secondary containment isolation failed to occur.

Entry into EOP 5A, RADIOACTIVITY RELEASE CONTROL, is required when an EAL classification of ___(1)___ is reached due to the offsite radioactivity release.

With the offsite radioactive release in progress from secondary containment to the public, Emergency Depressurization is REQUIRED prior to reaching this EAL Classification ___(2)___.

A. (1) Notification of Unusual Event
   (2) Site Area Emergency

B. (1) Notification of Unusual Event
   (2) General Emergency

C. (1) Alert
   (2) Site Area Emergency

D. (1) Alert
   (2) General Emergency

Answer: D

Explanation:
This is RO knowledge because the question is asking entry conditions to an EOP and a condition requiring an ED. (Memory/Fundamental)

EOP 5A states the entry conditions for the Radioactivity Release Control is Offsite radioactivity release rate above offsite gaseous release rate which requires Alert and ED is required before offsite gaseous radioactivity release rate reaches that which requires a General Emergency BUT ONLY IF primary system is discharging into area outside primary and secondary containments.

A is incorrect. Part one is plausible because NOUE is an Entry level EAL. Part 2 is plausible because a CRS could ED at SAE EAL classification but is not required to be done prior to SAE.
B is incorrect. Part 1 is incorrect but plausible as explained in distractor A. Part 2 is correct.

C is incorrect. Part 1 is correct. Part 2 is plausible for the reasons stated in Distractor A

Technical References:
EOP-5A, Secondary Containment Control, Revision 19

References to be provided to applicants during exam:
None

Learning Objective:
INT008-06-17, EOP Flowchart 5A – Secondary Containment Control / Radioactive Release Control, Revision 25, Enabling objective 6

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

Examination Outline Cross-Reference	**Level**	RO
295017 (APE 17) Abnormal Offsite Release Rate / 9 2.4.50 Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. (CFR: 41.10 / 43.5 / 45.3)	**Tier#**	1
	Group#	2
	K/A #	295017 G2.4.50
	Rating	4.2
	Revision	
Revision Statement: Rev 1 – swapped parts 1 and 2 per CE comments.		

Question 60

The plant is at 100% power when the following annunciator is received:

OFFGAS DILUTION FAN A LOW FLOW	PANEL/WINDOW: K-1/C-3

The standby dilution fan FAILS to start.

The operator presses and holds OG LOW DILUTION FLOW ISOL OVERRIDE button IAW Alarm Card K-1/C-3.

One minute later, the following annunciator is received due to rising release rate:

OFFGAS TIMER INITIATED	PANEL/WINDOW: 9-4-1/C-4

(1) What is the setpoint for Offgas Radiation Monitors RMP-RM-150A(B) that triggered annunciator 9-4-1/C-4, OFFGAS TIMER INITIATED?

AND

(2) If conditions do not change, how long will it be until AOG isolates?

A. (1) 1.58E3 mR/hr
(2) 5 minutes

B. (1) 1.58E3 mR/hr
(2) 15 minutes

C. (1) 6.7E1 mR/hr
(2) 5 minutes

D. (1) 6.7E1 mR/hr
(2) 15 minutes

Answer: B

Explanation:
This question requires knowledge of Offgas Rad Monitor RMP-RM-150A(B) setpoints and understanding of the function of the OG LOW DILUTION FLOW ISOL OVERRIDE button. Offgas dilution fans supply dilution air to reduce the hydrogen concentration in the ERP. When the AOG system is not in service, they also maintain a suitable exit velocity at the top of the ERP. Normally, one fan is in operation, with the other fan in standby. AOG isolates on a low dilution flow condition after a 5 minute time delay. If both OG Dilution fans are lost and AOG is in service, the OG LOW DILUTION FLOW ISOL OVERRIDE button may be pressed and held to override the low flow isolation. Alarm Card K-1/C-3 directs pressing and holding the OG LOW DILUTION FLOW ISOL OVERRIDE button if both dilution fans are lost and AOG has not yet isolated. Loss of dilution flow causes higher concentration of noble gases in the offgas steam, resulting in elevated radiation levels, Alarm 9-4-1/C-4, OFFGAS TIMER INITIATED is activated when Offgas Rad Monitors RMP-RM-150A and B reached their hi-hi trip setpoint, 1.58E3 mR/hr. This causes AOG to isolate after a 15 minute time delay.

Distracters:
Answer A part 1 is correct. Part 2 is plausible and wrong for the same reason given for distractor A.

Answer C part 1 is plausible because it represents the Offgas Radiation high setpoint for RMP-RM-150A, which causes annunciator 9-4-1/C-5, OFFGAS HIGH RAD. It is wrong because Offgas Radiation hi-hi, with a setpoint of 1.58E3 mR/hr, causes annunciator 9-4-1/C-4. Part 2 is plausible because the time delay associated with low dilution flow is 5 minutes. It is wrong because the time delay associated with hi-hi offgas radiation is 15 minutes. With the OG LOW DILUTION FLOW ISOL OVERRIDE button held depressed, as given in the stem, AOG will not isolate after 5 minutes on low dilution flow, but only on hi-hi radiation, after 15 minutes.

Answer D part 1 is plausible and wrong for the same reason given for distractor A. Part 2 is correct.

Technical References: Lesson plan COR001-16-01 [Ops Off Gas](Rev 35), Alarm Card K-1/C-3 [Offgas Dilution Fan A Low Flow](Rev 15), Alarm Card 9-4-1/C-4 [Offgas Timer Initiated](Rev 59), Alarm Card 9-4-1/C-5 [Offgas High Rad]((Rev 59)

References to be provided to applicants during exam: none

Learning Objective: COR001-16-01 Obj LO-13b, Given plant conditions, determine if the following should occur: AOG Auto Isolation,

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

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X

10CFR Part 55 Content:	55.41(b)(11)
Level of Difficulty:	3

SRO Only Justification:	N/A
PSA Applicability	
N/A	

Examination Outline Cross-Reference		Level	RO
295008 (APE 8) High Reactor Water Level		Tier #	1
		Group #	2
Knowledge of the reasons for the following responses as they apply to HIGH REACTOR WATER LEVEL:		K/A #	AK3.02
		Rating	3.6

AK3.02 Reactor SCRAM

Question 61

The plant is at 100% power.

The following indications are present;
- All narrow range RPV level instruments are oscillating between 47" and 51"
- Annunciator 9-5-2/F-1, Reactor Water Level High is in alarm

What actions will the control room perform?

A. Enter Procedure 2.4RXLVL, place Master level controller in "MAN"

B. Enter Procedure 2.4RXLVL, place level control switch to "1 ELEMENT CONT"

C. Enter Procedure 2.1.5, SCRAM the reactor, and maintain RPV level with RFPs

D. Enter Procedure 2.1.5, SCRAM the reactor, and ensure main turbine, RFPTs, HPCI, and RCIC are tripped

Answer: D

Explanation:
Procedure 2.3_9-5, Panel 9-5 Annunciator response directs the operators to SCRAM the reactor if RPV level is above 50".

Procedure 2.4RXLVL, attachment 4 states, 1.3.2 "The language used in the associated procedure steps (i.e., "cannot be maintained") means that if RPV level ever goes below 12" during the event, the reactor shall be manually scrammed or if RPV level ever goes above 50" during the event, the reactor shall be manually scrammed and any operating turbines stopped. The language does not provide any tolerance or allowance for exceeding these values"

A is incorrect. Plausible because procedure 2.4RXLVL directs the operators to place the Master level controller in "MAN" if level control is still erratic after placing control switch to 1 element control.
B is incorrect. Plausible because this would be the correct action for oscillating RPV levels as long as the level does not exceed 50"
C is incorrect. Plausible because normally once would control RPV level with the RFPs.
D is correct.

Technical References:

Procedure 2.4RXLVL, RPV Water Level Control Trouble, Revision 28, page 1 and Attachment 4
Procedure 2.3_9-5-2, Panel 9-5 - Annunciator 9-5-2, Revision 49, page 63
Procedure 2.1.5 Reactor Scram, Revision 77, Attachment 3

References to be provided to applicants during exam:
None.

Learning Objective:
INTO032-01-04, Administrative Procedures General Operating Procedures, Revision 10, Enabling objective D.9

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	3
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.5	

Examination Outline Cross-Reference	Level	RO
295022 (APE 22) Loss of Control Rod Drive Pumps Ability to operate and/or monitor the following as they apply to LOSS OF CRD PUMPS: AA1.01 CRD hydraulic system	**Tier#**	1
	Group#	2
	K/A #	295022 AA1.01
	Rating	3.1
	Revision	1
Revision Statement: Replaced original question due to error identified in its supporting procedure		

Question 62

The plant is at 100% power.

CRD Pump B is in service.

The following annunciator is received:

CRD PUMP B BREAKER TRIP	PANEL/WINDOW: 9-5-2/C-6

(1) Which action at Panel 9-5 is required first IAW Alarm Card 9-5-2/C-6?

AND

(2) While restoring CRD flow, which one of the following conditions must be avoided to prevent control rods from inadvertently drifting in?

A. (1) Place CRD Pump A control switch to START
(2) High CRD Drive Water dP

B. (1) Place CRD Pump A control switch to START
(2) High CRD Cooling Water dP

C. (1) Place CRD Flow Controller [CRD-FC-301] in MAN
(2) High CRD Drive Water dP

D. (1) Place CRD Flow Controller [CRD-FC-301] in MAN
(2) High CRD Cooling Water dP

Answer: D

Explanation:
The primary mitigative strategy of Alarm Card 9-5-2/C-6 is to place the standby CRD pump in operation. The first step in this process is to place the CRD Flow Controller in MANUAL in order to close the in-service Flow Control Valve. Upon loss of the running CRD pump and CRD flow, the controller would have sensed flow below the setpoint, and in the normal operating mode, BALANCE, it would cause the FCV to fully open. The FCV must be fully closed before starting the standby CRD pump to prevent a pressure surge in the CRD Cooling Water header that could result in producing high dP across CRDM drive seals, resulting in control rods drifting in. CRD Cooling Water enters the CRD beneath the CRDM piston, as would Drive Water flow during an insert command. Procedure 2.4CRD states "Cooling Water dP > 25 psid may cause inadvertent control rod movement", which would manifest as control rods drifting in.

Distracters:
Answer A Part 1 is plausible for the examinee who does not remember the CRD FCV must first be closed before starting the standby CRD pump. It is wrong because the first step in closing the CRD FCV is to place the controller in MANUAL to enable closing the FCV, in order to prevent causing control rod drifts when the CRD pump is started. Part 2 is plausible to the examinee who does not understand the CRD piping configuration and associates high Drive Water dP with control rod movement or who confuses the result of high Drive Water dp (i.e. excessive rod speeds) with inadvertent rod drifts. It is wrong because high Drive Water dP is inconsequential without a drive command present, since CRD directional control valves are closed and no motive force is applied to the CRDM drive piston.

Answer B part 1 is plausible and wrong for the same reason given for distractor A. Part 2 is correct.

Answer C part 1 is correct. Part 2 is plausible and wrong for the same reason given for distractor A.

Technical References: Alarm Card 9-5-2/C-6 [CRD Pump B Breaker Trip](Rev 51), B&R dwg 2039 [CRD P&ID], Procedure 2.4CRD [CRD Trouble](Rev 20)

References to be provided to applicants during exam: none

Learning Objective: COR002-04-02 Obj LO-5c, Briefly describe the following concepts as they apply to the CRDH system: Pressure indication; LO-11i, Predict the consequences a malfunction of the following would have on the CRDH system: CRDH pump trip

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

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X

10CFR Part 55 Content:	55.41(b)(6)

Level of Difficulty:	3
SRO Only Justification:	N/A
PSA Applicability	
N/A	

Examination Outline Cross-Reference	Level	RO
295033 (EPE 10) High Secondary Containment Area Radiation Levels	Tier #	1
	Group #	2
	K/A #	EK2.03
Knowledge of the interrelations between HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS and the following:	Rating	3.7
EK2.03 Secondary containment ventilation		

Question 63

What is the Technical Specification 3.3.6.2-required setpoint for Secondary Containment isolation on Reactor Building Ventilation Exhaust Plenum radiation?

A. ≤ 5 mrem/hr

B. ≤ 23 mrem/hr

C. ≤ 49 mrem/hr

D. ≤ 60 mrem/hr

Answer: C

Explanation:

[This should be RO level because knowledge of the TS setpoints for trip functions is an extension of above the line information, since the above the line LCO/applicability directs the operator to the table]

Secondary containment isolation occurs on any of the following three conditions:

- Reactor Vessel Water Level - Low Low Level 2
- Drywell Pressure – High
- Reactor Building Ventilation Exhaust Plenum Radiation – HI ≤ 49 mrem/hr

RX BLDG Vent exhaust plenum radiation HI HI also initiates SGT.

A is wrong because this is the alarm setpoint for RX BLDG VENT HI RAD, but not RX BLDG VENT HI HI RAD. It is also the setpoint for Admin Bldg HI RAD alarm, and various other area rad alarms in the Reactor Building.
B is wrong because this is the alarm setpoint for RX BLDG RHR PUMP ROOM (SW) AREA RAD HIGH. I like this distractor because it's not a clean factor of 5 or 10, just like the correct answer.
C is correct. TS setpoint is <= 49 mrem/hr, although the ACTUAL setpoint in the plant is 10 mrem/hr.
D is wrong because this is the setpoint for RX BLDG CRD NORTH HCU AREA RAD HIGH setpoint.]

Technical References:
Technical Specification 3.3.6.2, Amendment 260

Student Lesson Plan OPS Radiation Monitoring/COR001-18-01 Rev 28
Student Lesson Plan OPS Standby Gas Treatment/COR002-28-02

References to be provided to applicants during exam:
None.

Learning Objective:
COR001-18-01, Radiation Monitoring, Revision 28, Enabling objectives 3.d and 5.r

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	3
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.9	

Examination Outline Cross-Reference		Level	RO
295015 (APE 15) Incomplete Scram		Tier #	1
		Group #	2
Ability to determine and/or interpret the following as they apply to INCOMPLETE SCRAM:		K/A #	AA2.02
		Rating	4.1

AA2.02 Control rod position

Question 64

- The crew is responding to an ATWS event.
- You have been directed to insert all control rods using Procedure 5.8.3, Alternate Rod Insertion Methods.
- The RO has verified that the scram valves are open.

Which method of control rod insertion is attempted first?

A. Vent the scram air header

B. Individually scram control rods

C. Vent the individual CRD over piston areas

D. Drain the SDV and scram the Reactor manually

Answer: D

Explanation:
1.3 This procedure will attempt alternate control rod insertion (in this order) using the following methods concurrent with manual rod insertion:
- 1.3.1 Drain the SDV and scram the Reactor manually if the scram valves are open (hydraulic lock on the SDV).
- 1.3.2 Vent the scram air header (failure of RPS).
- 1.3.3 Individually scram control rods.
- 1.3.4 Vent the individual CRD over piston areas.

A is wrong.
B is wrong.
C is wrong.
D is correct.

Technical References:
Procedure 5.8.3, Alternate Rod Insertion Methods, Revision17, page 16

References to be provided to applicants during exam:
None.

Learning Objective:
INT008-06-06, EOP Flowchart 6A – RPV Pressure & Power (Failure-to-Scram), Revision 27, Enabling objective 9

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	2
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
295009 (APE 9) Low Reactor Water Level	Tier #	1
	Group #	2
Knowledge of the reasons for the following responses as they apply to LOW REACTOR WATER LEVEL:	K/A #	AK3.01
	Rating	3.2

AK3.01 Recirculation pump run back

Question 65

Given the following:

- The plant is at 100% reactor power
- Reactor Recirculation Pumps are not locked out
- RFP A trips on low lube oil pressure

A runback towards ___(1)___ will be initiated but will terminate if total steam flow falls below a maximum of ___(2)___ Mlbm/hr.

A. (1) 45%
(2) 8.25

B. (1) 45%
(2) 9

C. (1) 40%
(2) 8.25

D. (1) 40%
(2) 9

Answer: B

Explanation:
A is incorrect. Part 1 is correct. Part 2 is plausible because 8.25 would be correct if condensate or condensate booster pump discharge were low or both RFP suction pressures were low. Since there are no indications in the stem with condensate, booster, of feed pump pressures, it would require assumptions not in the stem to believe they are low.
B is correct.
C is incorrect. Part 1 is plausible because 40% is an additional runback. Part 2 is plausible for the reason stated in distractor A.
D is incorrect. Part 1 is plausible for the reason stated in distractor C. Part 2 is correct.

Technical References: Operations Procedure 2.2.68, Reactor Recirculation, Revision 89, page 57, Attachment 1

References to be provided to applicants during exam:
None.

Learning Objective:

COR002-22-02, Reactor Recirculation, Revision 35, Enabling objective 10.I

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 1252
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental **Comprehensive**/Analysis	 3
10CFR Part 55 Content:	55.41.5	

PARENT QUESTION

QUESTION: 10 1252 (1 point(s))
The plant was operating normally at 90% power when a single reactor feed pump trips. Reactor water level drops to +25" (NR).

Based upon these conditions, the Reactor Recirculation pumps will be . . .

a. tripped.

b. operating at □ 45% speed.

c. operating at □ 22% of rated speed.

d. operating with the scoop tubes locked out.

ANSWER: 10 1252
b. operating at □ 45% speed.

REFERENCE: STCOR002-22-02, page 29, section II.H.8, rev. 15.

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

Question 66

Given the following conditions:

- A standard pancake Geiger-Muller detector is being used to perform a whole body frisk
- Background radiation is at 105 counts per minute (cpm)

Which of the following is the MINIMUM reading on the detector at which an individual is considered to be contaminated in accordance with radiation protection procedure 9.EN-RP-104, Personnel Contamination?

A. 125 cpm

B. 175 cpm

C. 225 cpm

D. 275 cpm

Answer: C
:
Explanation:
A is wrong but plausible if background radiation level is not counted when answering the question.
B is wrong but plausible if thought contamination was evident at 50 cpm above background.
C is correct; minimum value that is at least 100 cpm above background.
D is wrong but plausible if thought contamination was evident at 150 cpm above background.

Technical References:
Radiation Protection Procedure 9.EN-RP-104, “Personnel Contamination,” Page 37

References to be provided to applicants during exam:
None.

Learning Objective: LP 032-01-100, “OPS CNS Administrative Procedures Radiation Protection,” Enabling Objective F.1.

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

	Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41.12	

Examination Outline Cross-Reference	Level	RO
2.1.26 Knowledge of industrial safety procedures (such as rotating equipment, electrical, high temperature, high pressure, caustic, chlorine, oxygen and hydrogen)	Tier #	3
	Group #	
	K/A #	2.1.26
	Rating	3.4

Question 67

You have been tasked with racking out a 125 VDC breaker.

What personnel protective equipment (PPE) is required by Procedure 0.36.8, Electrical Safety Rule Book to perform this task?

A. flash suit and hood

B. 100% cotton clothing and arc face shield

C. voltage rated gloves and fire resistant clothing

D. voltage rated gloves and 100% Cotton clothing

Answer: A

Explanation:
A is correct.
B is wrong. Plausible because this is required for opening a panel and installing temp grounds in higher voltage systems.
C is wrong. Plausible because this would be required for work near energized parts in higher voltage systems.
D is wrong. Plausible because this would be required for Removing/installing a 120 VAC breaker.

Technical References:
Procedure 0.36.8, revision 22, pages 4 and 5

References to be provided to applicants during exam:
None.

Learning Objective:

Question Source: (note changes; attach parent)	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	2
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
2.2.38 Knowledge of conditions and limitations in the facility license	Tier #	3
	Group #	
	K/A #	2.2.38
	Rating	3.6

Question 68

What is the minimum power level that LCO 3.2.1, Average Planar Linear Heat Generation (APLHGR), is applicable?

A. 9.85%

B. 15%

C. 25%

D. 29.5%

Answer: C

Explanation:
A is wrong because below this number is when TS 3.1.6, Rod Pattern Control, requires the operable control rods to comply with the requirements of the banked position withdrawal sequence; is plausible because it is a TS applicability limit.
B is wrong because 15% RTP is when LCO 3.6.3.1, Primary Containment Oxygen Concentration, becomes applicable; is plausible because it is a TS applicability limit.
C is correct because this is the TS 3.2.1 requirement.
D is wrong because this is the RPS setpoint to initiate a reactor scram on turbine stop valve closure; is plausible because it is an important setpoint and is in TS 3.3.1.1.

Technical References:
Technical Specifications 3.2.1, 3.6.3.1, 3.1.6, and 3.3.1.1

References to be provided to applicants during exam:
None.

Learning Objective:
INT007-05-03, Technical Specifications 3.2, Power Distribution Limits, Revision 12, Enabling objective 1

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

10CFR Part 55 Content: 55.41.10

Examination Outline Cross-Reference		Level	RO
2.1.1 Knowledge of conduct of operations requirements		Tier #	3
		Group #	
		K/A #	2.1.1
		Rating	3.8

Question 69

Which of the following positions is required to report to the control room within 10 minutes of being notified to report?

A. Fire Marshall

B. Fire Brigade Leader

C. Temporary Operator

D. Work Control Operator

Answer: D

Explanation:
A is wrong because this is an engineering position and does not have any time requirements to report when called. Used for balance 2 fire positions, 2 operator positions.
B is wrong because this position is not required to report. This position is usually assigned to the WCO but isn't required. Didn't put in information about not assuming this is their only position because I didn't want to lead anyone to the answer.
C is wrong because there are no time requirements for the person to report. Plausible because this person is called by the shift manager.
D is correct because as described in procedure 2.0.3 this position along with shift manager and STE must report in 10 minutes. Decided this was RO level because this position requires an active RO license.

Technical References:
Procedure 2.0.3, "Conduct of Operations," Rev. 103, p. 12

References to be provided to applicants during exam:
None.

Learning Objective:
INT032-01-03, Administrative Procedure Conduct of Operations and General Alarm Procedures (Formal Classroom/Pre-OJT Training), Revision 11, Enabling objective C.1.a.8

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

	Comprehensive/Analysis
10CFR Part 55 Content:	55.41.10

Examination Outline Cross-Reference	Level	RO
2.1.4 Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, “no-solo” operation, maintenance of active license status, 10CFR55, etc.	**Tier#**	3
	Group#	
	K/A #	2.1.4
	Rating	3.3
	Revision	2
Revision Statement:		

Question 70

IAW procedure 2.0.7, LICENSED OPERATOR ACTIVE/REACTIVATION/MEDICAL STATUS MAINTENANCE PROGRAM, an inactive Reactor Operator is required to stand a minimum of ___(1)___ under instruction watches with a qualified RO to **REACTIVATE** his license.

Standing under instruction as either ATCO/RO (At the Controls Operator/ Reactor Operator) or ___(2)____ can be counted as under instruction watches to **REACTIVATE** as Reactor Operator.

A. (1) 4
(2) BOP (Balance of Plant) ONLY

B. (1) 4
(2) BOP (Balance of Plant) OR WCO (Work Control Operator)

C. (1) 5
(2) BOP (Balance of Plant) ONLY

D. (1) 5
(2) BOP (Balance of Plant) OR WCO (Work Control Operator)

Answer: A
Explanation: Per 2.0.7 LICENSED OPERATOR ACTIVE/REACTIVATION/MEDICAL STATUS MAINTENANCE PROGRAM, 4 watches under instruction as either a ATC or BOP to get credit to reactivate as a Licensed Reactor Operator and four watches are required to be stood.
Distracters: Answer B Part 1 is correct Part 2 is plausible because , Qualified ROs can stand ATCO, BOP, or WCO once qualified but only ATC Operator and BOP get credit for reactivation. Answer C Part 1 is plausible because 5 watches is the number of watches needed to maintain the RO license but only 4 watches under instruction are needed to reactivate the RO license to stand the watch. Part 2 is correct. Answer D Part 1 is plausible for the reasons stated in Distractor C. Part 2 is plausible for the reasons stated in Distractor B
Technical References: 2.07 LICENSED OPERATOR ACTIVE/REACTIVATION/MEDICAL STATUS MAINTENANCE PROGRAM
References to be provided to applicants during exam: none
Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.(10)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	
PSA Applicability:		
N/A		

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

Question 71

The Shift Manager has declared a General Emergency. You have been tasked with restoring a piece of critical equipment to service. The work area that you will be working in has a radiation dose rate of 10 rem/hr.

What is your maximum allowed stay time per Procedure 5.7.12, Emergency Exposure Control?

A. 12 minutes

B. 30 minutes

C. 60 minutes

D. 150 minutes

Answer: C

Explanation:
A is wrong. Plausible because the administrative annual guideline is 2R and you would receive this after 12 minutes (Procedure 9.ALARA.1)
B is wrong. Plausible because the emergency limit of 5R applies to people collecting samples or performing surveys and you would receive this after 30 minutes.
C is correct.
D is wrong. Plausible because the dose allowance during a declared emergency to prevent imminent core damage is 25R.

Technical References:
Procedure 5.7.12, Emergency Radiation Exposure Control, Revision17, page 8
Procedure 9.ALARA.1, Dosimetry Administration, Revision 48, page 17

References to be provided to applicants during exam:
None.

Learning Objective: INT032-01-15, Radiation Protection and Chemistry Procedures, Enabling objective L.1.a.1

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41.12	

Examination Outline Cross-Reference	Level	RO
2.2.39 Knowledge of less than or equal to one hour Technical Specification action statements for systems.	Tier #	3
	Group #	
	K/A #	2.2.39
	Rating	3.9

Question 72

Given the following:

- The plant has just entered Mode 4 for a planned maintenance outage
- Reactor Engineering informs the Shift Manager that Shutdown Margin is not within Technical Specification limits

Which action(s) below are operators required to take to be in full compliance with TS 3.1.1?

(1) Immediately initiate action to restore SDM to within limits
(2) Immediately initiate action to insert all insertable control rods
(3) Within 1 hour initiate action to restore secondary containment to operable
(4) Within 1 hour initiate action to restore one train of SGT to operable
(5) Within 1 hour initiate action to restore isolation capability in each required secondary containment penetration flow path not isolated

A. (1), (3) and (4) only

B. (2), (4), and (5) only

C. (1), (3), (4) and (5)

D. (2), (3), (4), and (5)

Answer: D

Explanation:
A is wrong because TS 3.1.1.D requires actions 2-5 be completed; is plausible because action 1 would be correct if the plant was in mode 1 or 2, and because actions 3 and 4 are correct actions.
B is wrong because TS 3.1.1.D requires actions 2-5 be completed; is plausible because actions 2, 4, and 5 are correct answers.
C is wrong because TS 3.1.1.D requires actions 2-5 be completed; is plausible because action 1 would be correct in modes 1 or 2.
D is correct because TS 3.1.1.D require all of these actions via “AND” logic connectors.

Technical References:
TS 3.1.1.D

References to be provided to applicants during exam:
None.

Learning Objective:

INT007-05-02, Technical Specification 3.1 Reactivity Control Systems, Revision 14, Enabling objective 4.b

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 2359
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41.10	

PARENT QUESTION

QUESTION: 318 2359 (2 point(s))

Given the following plant conditions:

- A Reactor scram occurred during a plant heatup
- The Reactor mode switch in SHUTDOWN
- Reactor coolant temperature is 221 °F
- Both Reactor Building exhaust dampers failed to close following a valid isolation signal during the scram
- Three (3) withdrawn control rods failed to insert
- Shutdown margin is 0.32% k/k with the highest worth control rod determined analytically

What actions are required **IMMEDIATELY** per Technical Specification LCO 3.1.1?

a. Initiate action to insert all insertable control rods.

b. Initiate action to restore secondary containment to operable status.

c. Initiate action to restore one (1) standby gas treatment system to operable status.

d. Initiate action to restore isolation capability in each required secondary containment penetration flow path not isolated.

ANSWER: 318 2359

a. Initiate action to insert all insertable control rods.

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

Question 73

During the performance of pre-startup step requirements (Precautions and Limitations) in startup procedure 2.1.1, the requirements for additional operators during reactivity manipulations are one Reactivity Management SRO and __(1)__ AND __(2)__ ONLY.

A. (1) One extra SRO on-shift (not including the SM)
   (2) One extra RO on-shift for additional board manipulations

B. (1) One extra SRO on-shift (not including the SM)
   (2) One Licensed Operator for control manipulation verification

C. (1) One Licensed operator for control manipulation verification
   (2) One extra RO to assist during heavy workloads

D. (1) One Licensed Operator for control manipulation verification
   (2) One Station Operator to assist during heavy workloads

Answer: D

Explanation:
A is wrong because the additional requirement beyond the Reactivity SRO (in the stem of the question and given) is One Licensed Operator for control manipulation verification AND One Station Operator to assist during heavy workloads.
B is wrong because (see A above)
C is wrong because (see A above)
D is correct because per P and L step 2.7, “From time MODE 2 is entered until first RFP discharge valve is open, following additional Operator coverage is required:

> 2.7.1 One SRO to act as Reactivity Manager during reactivity manipulations. During periods when reactivity is not being manipulated, Operator may provide oversight of other activities or perform other duties as directed by Control Room Supervisor.
> 2.7.2 One Licensed Operator dedicated to verifying control rod movements or verification of control manipulation during reactivity manipulations. During periods when reactivity is not being manipulated, Operator may perform other duties as directed by Control Room Supervisor.
> 2.7.3 One Station Operator to assist duty Crew during times when work load prevents duty Crew from performing manipulations in a timely manner. When not needed to assist duty Crew, Operator is to tour plant being observant to potential plant problems.

This is RO knowledge because the RO has to know who else is required to assist during a reactor startup other than the supervisory position (which was placed in the stem for that reason).
Technical References:
Plant Startup Procedure 2.1.1, P and L step 2.7, revision 197, page 4
INT032-01-04, General Operating Procedures Training, Revision 10

References to be provided to applicants during exam:
None.

Learning Objective: INT032-01-04, General Operating Procedures Training, Revision 10, Enabling objectives A1 and A2

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	2
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
2.4.27 Knowledge of “fire in the plant” procedures	Tier #	3
	Group #	
	K/A #	2.4.27
	Rating	3.4

Question 74

A fire has been confirmed in the Turbine Building. The control room supervisor has entered Procedure 5.1INCIDENT, Site Emergency Incident. As the control room operator, you are directed to announce the fire to station personnel.

What direction(s) would you provide over the Gaitronics?

Direct the _____.

A. Fire Brigade and Utility Fire Brigade to an exterior entrance closest to the fire and the Incident Commander to the scene of the fire

B. Fire Brigade and Utility Fire Brigade to a specific fire locker designated by the control room operator and the Turbine Building NLO to the scene of the fire

C. Fire Brigade to a specific fire locker designated by the control room operator and the Utility Fire Brigade and Fire Brigade Leader to the location of the fire

D. Fire Brigade to a specific fire locker designated by the control room operator; the Utility Fire Brigade to an exterior entrance closest to the fire and all other personnel to remain clear of the area

Answer: D

Explanation:
Per Emergency Procedure 5.1INCIDENT (Rev 40) Attachment 1 (Control Room Operator), Step 1.2.1

ATTENTION, A FIRE HAS BEEN DETECTED IN THE (location of fire). FIRE BRIGADE RESPOND TO THE (location of fire locker) FIRE EQUIPMENT LOCKER. UTILITY FIRE BRIGADE RESPOND TO (location of exterior entrance closest to fire). ALL OTHER PERSONNEL REMAIN CLEAR OF THE FIRE RESPONSE AREA. MINIMIZE NON-EMERGENCY COMMUNICATIONS WITH THE CONTROL ROOM.

A is wrong. Plausible because the utility fire brigade is dispatched to the closest outside entrance.
B is wrong. Plausible because the fire brigade is sent to a fire locker in the plant.
C is wrong. Plausible because the fire brigade is sent to a fire locker in the plant.
D is correct.

Technical References:
Procedure 5.1 Incident, revision 40, Attachment 1

References to be provided to applicants during exam:
None.

Learning Objective:
INT0320134, Abnormal Procedures (RO) Fire, Revision 10, Enabling objective G

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	2012 Exam Q 75
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	2
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.10	

PARENT QUESTION

QUESTION: 75

A fire has been confirmed in the Turbine Building. The CRS has entered Procedure 5.1INCIDENT, SITE EMERGENCY INCIDENT. As the control room operator you are directed to announce the fire to station personnel. What are your actions for these conditions?

Sound the Fire Alarm Pulse Tone for ten (10) seconds and direct the …

a. Fire Brigade and Utility Fire Brigade to a specific fire locker designated by the control room operator and the Turbine Building NLO to the scene of the fire.

b. Fire Brigade to a specific fire locker designated by the control room operator and the Utility Fire Brigade and Fire Brigade Leader to the location of the fire.

c. Fire Brigade and Utility Fire Brigade to an exterior entrance closest to the fire and the Incident Commander to the scene of the fire.

d. Fire Brigade to a specific fire locker designated by the control room operator; the Utility Fire Brigade to an exterior entrance closest to the fire and the Turbine Building NLO to the scene of the fire.

ANSWER: 75

d. Fire Brigade to a specific fire locker designated by the control room operator; the Utility Fire Brigade to an exterior entrance closest to the fire and the Turbine Building NLO to the scene of the fire.

Examination Outline Cross-Reference	Level	RO
2.4.5 Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions	Tier #	3
	Group #	
	K/A #	2.4.5
	Rating	3.7

Question 75

If an EOP directs an explicit system operation per a 5.8 EOP Support Procedure while an abnormal operating procedure is already in use for that system, then _____.

A. that operation may be executed solely within the abnormal operating procedure

B. transition shall be made from the abnormal operating procedure to the 5.8 EOP Support Procedure

C. the abnormal operating procedure and the 5.8 EOP Support Procedure may be executed concurrently

D. operators may use skill of the craft to operate the system

Answer: B

Explanation:
A is wrong because EOPs and SAGs are the highest tier of procedures and the transition out of the abnormal procedure into the 5.8 procedure is required per procedures 2.0.1.2, Operations Procedure Policy, and 5.8, Emergency Operating Procedures; is plausible because abnormal procedures may direct some system operations.
B is correct because procedures 2.0.1.2, Operations Procedure Policy, and 5.8, Emergency Operating Procedures, require that a transition to the 5.8 procedure be made.
C is wrong because this specific situation is an exception to the rule that "EOPs and abnormal procedures (and others) may be performed concurrently"; is plausible because per procedure 2.0.1.2 in most situations "Alarm/Abnormal/Emergency/System Operating/Instrument Operating Procedures may be carried out concurrently with an EOP."
D is wrong because skill of the craft is allowed to be used for non-EOP procedures and/or when no specific reference is given; is plausible because skill of the craft is allowed in some circumstances.

Technical References:
Procedure 2.0.1.2, rev 47, Operations Procedure Policy, section 2.4-2.5, page 2
From Emergency Operating Procedure 5.8, rev 45, Emergency Operating Procedures, section 3.4, page 4

References to be provided to applicants during exam:
None.

Learning Objective:
INT032-01-01, CNS Procedures Volume 0, Administrative Procedures, Revision 17, Enabling objective R.2

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	SRO
295028 (EPE 5) High Drywell Temperature	Tier #	1
	Group #	1
2.2.12 Knowledge of surveillance procedures	K/A #	2.2.12
	Rating	4.1

Question 76

Plant is in Mode 1

Your crew has assumed the watch and the crew has started taking the daily shift logs.

The turnover sheet indicates three drywell fan cooling units are running with the fourth set to be returned in 4 hours. Drywell temperatures are elevated and are expected to return to normal when the fourth drywell cooling fan is placed in service.

The Reactor Operator reports the following to you on the status of the average drywell temperature indicator points:

Indicator	Quality Code	Color	Health Status
SPDS0202	ALM	Red	Healthy
SPDS0104	NCAL	Magenta	Unhealthy
SPDS0110	DALM	Green	Healthy
SPDS0051	SUB	Blue	Healthy

You direct the Reactor Operator to complete the daily shift logs using point ___(1)___ in accordance with procedure ___(2)___.

A. (1) SPDS0202
(2) 2.1.12, Control Room Data

B. (1) SPDS0104
(2) 6.PC.604, Average Drywell Temperature Manual Determination

C. (1) SPDS0110
(2) 6.LOG.601, Daily Surveillance Log - Modes 1, 2, and 3

D. (1) SPDS0051
(2) NEDC 89-142, Method for Determination of Average Bulk Drywell Temperature

Answer: C

Explanation:
A is wrong since there are only 3 fan cooling units in service point 0202 cannot be used even though it has a healthy status according to Attachment 5 of 6.LOG.601. Also, the procedure isn't credited for surveillances making it incorrect also.
B is wrong since 0104 has an unhealthy quality code and because the procedure wouldn't be used because 6.LOG.601 can still be met with a different point.
C is correct since 0104 has an unhealthy code then point 0110 is required since it has a healthy quality code

D is wrong because 0051 isn't used in any of the credited procedures and because the procedure referenced is a manual calculation performed by engineering it shouldn't be used either since a simple procedure is available.

According to 2.6.3PMIS, Attachment 2, the preferred order of parameters is 0202, 0104, 0110, or 0051 with a healthy quality.

I wanted to use a 2X2 to highlight the depth of knowledge needed to answer the question. I highlighted the operator because I couldn't remember the Cooper naming convention.

SRO-only due to testing the depth of procedure knowledge and not just the overall goal even though the answer is the regular procedure.

K/A match because these are the inputs used to monitor for high drywell temperature.

Technical References:
6.LOG.601, Daily Surveillance Log - Modes 1, 2, and 3, Attachment 5, Primary Containment Instruments Notes
2.6.3, PMIS, Attachment 2

References to be provided to applicants during exam:
None

Learning Objective:
INT032-01-03, Administrative Procedure Conduct of Operations and General Alarm Procedures (Formal Classroom/Pre-OJT Training), Revision 11, Enabling objective C.1.a.6

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 3
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference	Level	SRO
295004 (APE 4) Partial or Total Loss of DC Power	Tier #	1
	Group #	1
	K/A #	AA2.01
Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER:	Rating	3.6
AA2.01 Cause of partial or complete loss of D.C. power		

Question 77

The plant is at 100% power.

Annunciator C-1/A-2, 125V DC SWGR BUS 1A BLOWN FUSE alarms due to a blown fuse supplying Panel AA3.

Assuming TS are entered as specified by procedure 5.3DC125 [Loss of 125 VDC],

Mode 4 is REQUIRED to be entered within no later than ________.

A. 37 hours

B. 41 hours

C. 44 hours

D. 8 days, 12 hours

Answer: A

Explanation:

This question requires knowledge of specific requirements in procedure attachments required to be implemented by procedure 5.3DC125, knowledge of TS bases, and application of TS actions including LCO 3.0.3 requirements.
TS 3.8.7 governs DC distribution systems as well as AC distribution systems during Modes 1, 2, and 3. 125 VDC panel BB3 is fed from 125 VDC Bus 1A. 125 VDC Bus 1A is specifically listed in TS Table 3.8.7-1, but distribution Panel AA3 is not. TS 3.8.7 bases states The loss of electrical loads associated with buses NOT specifically listed in TS Table 3.8.7-1 may not result in a complete loss of redundant safety function necessary to shut down the reactor and maintain it in a safe condition. Therefore, should one or more of these buses become inoperable due to a failure not affecting the Operability of a bus listed in Table 3.8.7-1 (e.g., a breaker supplying a single MCC fails open), the individual loads on the bus would be considered inoperable, and the appropriate Conditions and Required Actions of the LCOs governing the individual loads would be entered.

125 VDC panel AA3 supplies breaker control power for Div 1 4160 VAC powered systems supplied from 4160V Bus 1F. These systems include Core Spray Pump A, DG1 (output breaker), RHR subsystem A (including RHR Pump A), RHR subsystem B (including RHR

Pump B), RHR SWBPs A and C, Service Water Pumps A and C. Procedure 5.3DC125 Att. 5 is entered for loss of 125 VDC Panel AA3. Step 1.7 of Att. 5 states to enter TS 3.0.3, consistent with TS 3.5.1 Action H.1 for Core Spray A and RHR Pumps A and B inoperable, LCO 3.0.3 requires initiating action within 1 hour to place the unit in Mode 2 within 7 hours, Mode 3 within 13 hours, and Mode 4 within 37 hours.

Answers that involve TS other than LCO 3.0.3 are plausible since only one Div 1 125 VDC panel lost power. It is reasonable that an unprepared examinee may believe the loss of one power supply within one electrical division would not require entry into TS 3.0.3.

Answer B is plausible because DG1 is inoperable due to loss of output breaker control power and RHR Pumps in both RHR loops are inoperable. With one DG inoperable, TS 3.8.1 Action B.2 requires declaring supported features inoperable within 4 hours when redundant required features are inoperable. The unprepared examinee may conclude redundant required features are inoperable, since both RHR loops are impacted, and believe TS 3.0.3 entry is required, but only after 4 hours. (4 hours + 37 hours = 41 hours) It is wrong because LCO 3.0.3 is required to be entered and allows only 37 hours to be in Mode 4, and the stem asks for the earliest time required for Mode 3 entry.

Answer C is plausible because RHR Pumps A and B are affected, each Suppression Pool Cooling loop is affected. For two SPC loops inoperable, TS 3.6.2.3 Action B.1 allows 8 hours to restore one SPC loop operable before Condition C must be entered. TS 3.6.2.3 Action C.1 requires entry into Mode 3 within 12 hours and Mode 4 within 36 hours. (8 hrs + 36 hrs = 44 hours) It is wrong because LCO 3.0.3 is required to be entered and allows only 37 hours to be in Mode 4, and the stem asks for the earliest time required for Mode 3 entry.

Answer D is plausible because it reflects the time that would be attained by an examinee who confuses the effects of 125 VDC power loss with that of 250 VDC Bus 1A and would believe only LPCI Loop A is affected. This would be also consistent with TS 3.8.7 Action D, which requires declaring supported features inoperable. The TS 3.5.1 Action A for one LPCI Loop inoperable would require restoring LPCI operable within 7 days or being in Mode 3 within 12 hours and Mode 4 within 36 hours per Action B. (7 days + 36 hours = 8 days, 12 hours) It is wrong because 125 VDC affects operability of more required features than 250 VDC, and TS 3.0.3 is ultimately required.

Technical References:
Procedure 5.3DC125, Loss of 125 VDC, Revision 40
TS 3.0.3, Limiting Condition for Operation (LCO) Applicability
TS 3.8.1, AC Sources – Operating
TS 3.8.7, Distribution Systems – Operating, and bases, TS 3.6.2.3, RHR Suppression Pool Cooling

References to be provided to applicants during exam:
None.

Learning Objective: COR0020602R27, DC Electrical Distribution, Enabling Objective 2

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

Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 3
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference	Level	SRO
295018 (APE 18) Partial or Complete Loss of CCW	Tier #	1
	Group #	1
	K/A #	AA2.01
	Rating	3.4

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER:

AA2.01 Component temperatures

Question 78

Given the following:

- The plant is at 100% power
- Main generator stator temperature is 210ºF and slowly rising
- The highest bearing metal temperature is 210ºF and steady
- Annunciator 9-5-2/C-4 TSV & TCV CLOSURE TRIP BYP CHAN A/B is clear

The plant must enter procedure ___(1)___ and ___(2)___.

A. (1) 2.4GENH2
(2) scram

B. (1) 2.4GENH2
(2) lower power

C. (1) 2.4TURB
(2) scram

D. (1) 2.4TURB
(2) lower power

Answer: A

Explanation:
A is correct because attachment 6 of 2.4GENH2 requires a reactor scram for stator temperature ≥210ºF with alarm 9-5-2/C-4 clear.
B is wrong because attachment 6 of 2.4GENH2 requires the reactor to be scrammed for the given conditions; is plausible because for the given stator temperature 2.4GENH2 is the correct procedure.
C is wrong because entry conditions for procedure 2.4TURB require rising bearing metal temperatures not steady temperature and because a manual turbine trip is not procedurally required until bearing metal temperature is 225ºF not 210ºF; is plausible because a reactor scram is required to address the stator temperature (by a different procedure, 2.4GENH2).
D is wrong because entry conditions for procedure 2.4TURB require rising bearing metal temperatures not steady temperature; is plausible because bearing metal temperature is slightly elevated and lowering power would seem reasonable if an operator is unfamiliar with the procedure.

Technical References:
2.4GENH2, Generator or Hydrogen Abnormal, Revision 35, Attachment 6
2.4TURB, Main Turbine Abnormal, Revision 34, Attachment 11

References to be provided to applicants during exam:
None.

Learning Objective:
COR001-13-01, Main Generator and Auxiliaries, Revision 35, Enabling Objective 8.a

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	23490
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	3
10CFR Part 55 Content:	55.43.5	

PARENT QUESTION

QUESTION: 180 23490 (1 point(s))

The plant is operating at near rated power when the RO reports rising generator hydrogen temperatures. The RO also reports that bearing and oil temperatures are normal and TEC temperature and pressure are normal.

What is required?

a. Enter 2.4TURB only.

b. Enter 2.4GENH2 only.

c. Enter 2.4TEC and 2.4GENH2.

d. Enter 2.4TEC and 2.4TURB.

ANSWER: 180 23490

b. Enter 2.4GENH2 only.

Examination Outline Cross-Reference	Level	SRO
295006 (APE 6) Scram	Tier #	1
	Group #	1
2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation	K/A #	2.1.7
	Rating	4.7

Question 79

REFERENCE PROVIDED

The Main Turbine trips while operating at 31% Power.

Two minutes later, ALL Mitigating Task Scram Actions per 2.1.5 (Reactor Scram) are completed with the following indications reported:

- All Main Turbine Bypass Valve positions are at 95%.
- Reactor Pressure is steady at 985 psig.

(1) What is the current reactor power based upon Main Turbine Bypass valve status?

(2) What is the HIGHEST Emergency Action Level (EAL) required to be declared IAW Procedure 5.7.1 (Emergency Classification)?

A. (1) Between 10% and 15%.
(2) Alert

B. (1) Between 20% and 25%.
(2) Alert

C. (1) Between 10% and 15%.
(2) Site Area Emergency

D. (1) Between 20% and 25%.
(2) Site Area Emergency

Answer: D

Explanation:
Requires evaluation of Main Turbine Bypass valve position following a valid reactor scram signal to determine reactor power. The SRO must then make an operational judgment based upon these conditions to apply them to the Emergency Plan. Three Main Turbine Bypass valves have the capacity to reject ~25% rated steam flow (~ 8% per BPV). With all BPVs at 95% open for the given reactor pressure, a correlation is made to a reactor power between 20% and 25%. With a valid automatic scram signal (MT Trip >30%) and all Mitigating Task Scram Actions complete (all manual actions taken at the reactor control console do not shut down the reactor as indicated by reactor power > 3%) the EAL for a Site Area Emergency is exceeded (SS2.1).

A is incorrect due to reactor power being between 20% and 25% and the SAE EAL threshold being exceeded. This answer is plausible if BPV capacity is unknown or miscalculated and the Mitigating Task Scram Actions were successful or final reactor power were changed to be less than 3% (BPVs at 15% open - making choice correct). The candidate who incorrectly identifies reactor power following a scram due to BPV position and does not recognize all manual actions to shut down the reactor from the control room were not successful would select this answer.
B is incorrect due to the SAE EAL threshold being exceeded. This answer is plausible if the Mitigating Task Scram Actions were successful or final reactor power were changed to be less than 3% (BPVs at 15% open - making choice correct). The candidate who correctly identifies reactor power following a scram due to BPV position and does not recognize all manual actions to shut down the reactor from the control room were not successful would select this answer.
C is incorrect due to reactor power being between 20% and 25%. This answer is plausible if BPV capacity is unknown or miscalculated. The candidate who incorrectly identifies reactor power following a scram due to BPV position and correctly identifies the highest EAL threshold exceeded would select this answer.

Technical References:

Procedure 2.1.5, Reactor Scram, Rev. 77
Procedure 5.7.1, Emergency Classification, Rev. 64
EPIP 5.7.1 Attachment 4, Rev. 18

References to be provided to applicants during exam:
EPIP 5.7.1 Attachment 4, Rev. 18

Learning Objective:
INT032-01-04, General Operating Procedures, Revision , Enabling objective D.4

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference	Level	SRO
295026 (EPE 3) Suppression Pool High Water Temperature / 5 2.4.41 Knowledge of the emergency action level thresholds and classifications. (CFR: 41.10 / 43.5 / 45.11)	**Tier#**	1
	Group#	1
	K/A #	295026 G2.4.41
	Rating	4.6
	Revision	1
Revision Statement: Rev 1 – Added EPIPEALCOLD as provided reference per CE comment.		

Question 80

REFERENCE PROVIDED

An event is in progress following a manual scram with the following conditions:

- Reactor power is 25%
- Reactor water level is -100 inches, stable
- Reactor pressure is 1000 psig, slowly rising
- Suppression Pool water level is 12 feet, slowly lowering
- Suppression Pool temperature is 200°F, slowly rising
- Drywell pressure is 1.9 psig, slowly rising

What is the HIGHEST emergency classification required for these conditions?

A. Notification of Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Answer: D
Explanation: The conditions given represent a failure to scram with power > 3% and operation in the unsafe zone of the Heat Capacity Temperature Limit (HCTL) curve. EAL SG2.1 (General Emergency) is met for these conditions.
Distracters: Answer A is plausible because operation in the unsafe region of HCTL, by itself, would constitute a Potential Loss of primary containment, which requires an Alert per EAL FU1.1. It is wrong because ATWS conditions also exist, which combined with operation in the unsafe region of HCTL, requires a General Emergency per EAL SG2.1. Answer B is plausible because drywell pressure is above and reactor level is -100 inches. Drywell pressure > 1.84 psig due to RCS leakage requires an Alert per EAL FA1.1. The

examinee who believes these conditions represent a loss of RCS integrity and does not recognize operation in the unsafe zone of HCTL or that ATWS conditions >3% power exist may choose this answer. It is wrong for the same reason stated for distractor A. Answer C is plausible because ATWS conditions >3% power require a Site Area Emergency per EAL SS2.1. It is wrong for the same reason given for distractor A.		
Technical References: EOP/SAG Graph 7, Heat Capacity Temperature Limit (Rev 17), Emergency Action Level Matrix, EPIPEALHOT (Rev 18)		
References to be provided to applicants during exam: EOP/SAG Graph 7, Heat Capacity Temperature Limit (Rev 17), Emergency Action Level Matrix, EPIPEALHOT (Rev 18) and EPIPEALCOLD (Rev 18)		
Learning Objective: ERO001-01-14 EO-4e, State the method and requirements for determining an appropriate classification: Given a copy of EPIP 5.7.1 and hypothetical abnormal plant symptoms, indications, or events, identify any and all EALs which have been exceeded and specify the appropriate emergency classification.		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3	
SRO Only Justification:		
This question requires knowledge of administrative procedures that require implementation and coordination of emergency implementing procedures.		
PSA Applicability		
N/A		

Examination Outline Cross-Reference		Level	SRO
295031 (EPE 8) Reactor Low Water Level		Tier #	1
		Group #	1
Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL:		K/A #	EA2.01
		Rating	4.6

EA2.01 Reactor water level

Question 81

The plant is operating when a station blackout occurred. All control rods fully inserted on the scram. The following conditions are noted by the crew:

- Reactor pressure is 1000 psig and being controlled by manual SRV actuation.
- Drywell pressure is 4 psig and very slowly rising.
- Drywell temperature is 250°F and slowly rising.
- Suppression Pool temperature is 175°F and rising slowly.
- Primary Containment level is 12.5' and stable.
- Reactor water level is -165" CFZ and slowly lowering.
- HPCI and RCIC have failed and cannot be made to inject.

What action is required?

A. Exit EOP-1A and perform RPV Flooding.

B. Exit EOP-1A and perform Steam Cooling.

C. Exit EOP-1A and Emergency Depressurize.

D. Continue in EOP-1A and Anticipate Emergency RPV Depressurization.

Answer: B

Explanation:
No injection sources are available and reactor level is -165" Corrected FZ. EOP-1A directs that that EOP-1A RPV Pressure and RPV level control actions be terminated (Override in RC/P-1 and RC/P-2) and enter EOP-1B, 12 where conditions of steam cooling exist.

A is wrong but plausible because this is the override action if RPV level cannot be determined. RPV level can be determined.
B is correct as stated above.
C is wrong but plausible because entering EOP-2A is the override action. However, the override in EOP-2A directs entry into EOP-1B before Emergency Depressurizing is commenced.
D is wrong but plausible because this is the override action if RPV depressurization is required.

Technical References:
EOP-1A, RPV Control, Revision 22
EOP-1B, Alternate Level / Pressure Control, Revision 2

EOP-2A, Emergency RPV Depressurization, Revision 20

References to be provided to applicants during exam:
None.

Learning Objective:
INT008-06-05, EOP Flowchart 1A – RPV Control, RPV Pressure, Revision 30, Enabling objective 13

Question Source:	Bank #	23182
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	4
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference		Level	SRO
295005 (APE 5) Main Turbine Generator Trip		Tier #	1
		Group #	1
2.1.20 Ability to interpret and execute procedure steps		K/A #	2.1.20
		Rating	4.6

Question 82

The plant is operating at rated power with the Startup Transformer out of service for maintenance.

The Main Turbine trips due to loss of vacuum.

(1) What power source automatically reenergizes the Critical Busses?

(2) What action is required IAW 5.3EMPWR (Emergency Power During MODES 1, 2, or 3)?

A. (1) Diesel Generators
(2) Direct DCC to perform the CNS-Black Plant Procedure.

B. (1) Diesel Generators
(2) Coordinate with DCC to backfeed through the Normal Transformer.

C. (1) Emergency Transformer
(2) Direct DCC to perform the CNS-Black Plant Procedure.

D. (1) Emergency Transformer
(2) Coordinate with DCC to backfeed through the Normal Transformer.

Answer: C

Explanation:
With the Startup Station Service Transformer out of service, the Normal Station Transformer is providing power to Critical Buses 1F and 1G through 4160V Buses 1A and 1B. When the main generator trips, the Normal Station Service Transformer becomes de-energized. Buses 4160 1A and 1B become de-energized which for one second de-energizes 1F and 1G. The Emergency Station Service Transformer repowers 1F and 1G directly. The Diesel Generators receive a start signal because of the short-lived (1 second) de-energization of 4160V buses1F and 1G. With all 4160V buses are de-energized for a short period of time a Station Blackout condition exists. However, due to the short lived duration, the proper procedure to enter is 5.3EMPWR. A common misconception is that only procedure 5.3SBO has guidance for directing DCC to enter the CNS Black Plant procedure. Procedure 5.3EMPWR Attachment 3 directs DCC to enter the CNS Black Plant procedure.

A is incorrect because the Critical Buses are energized from the Emergency Service Station Transformer. This answer is plausible if the order in which emergency power supplies energize the Critical Buses is confused or if the stem were changed to reflect a Loss of Offsite Power (LOOP - the emergency transformer is unavailable). The candidate who confuses the order in which emergency power supplies energize the Critical Buses and correctly identifies 5.3EMPWR directs the DCC to enter the Black Plant procedure would select this option.

B is incorrect because the Critical Buses are energized from the Emergency Service Station Transformer and backfeed is not directed in 5.3 EMPWR. This answer is plausible if the order in which emergency power supplies energize the Critical Buses is confused or if the stem were changed to reflect a Loss of Offsite Power (LOOP – the emergency transformer is unavailable) AND due to the Normal transformer being available for backfeed (Off Site power remains available). The candidate who confuses the order in which emergency power supplies energize the Critical Buses and does not know backfeed through the Normal transformer is only directed in 5.3SBO would select this answer.

D is incorrect because backfeed is not directed in 5.3 EMPWR. This answer is plausible to the Normal transformer being available for backfeed (Off Site power remains available). The candidate who correctly identifies the order in which emergency power supplies energize the Critical Buses and does not know backfeed through the Normal transformer is only directed in 5.3SBO would select this answer.

Technical References:
Procedure 5.3EMPWR (Emergency Power During Modes 1, 2, or 3), Rev. 68.
Procedure 5.3SBO (Station Blackout) Rev. 46

References to be provided to applicants during exam:
None.

Learning Objective:
COR001-01-01, AC Electrical Distribution, Revision 50, Terminal objective

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	CNS 2015-04 Q81
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 3
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference	Level	SRO
295029 (EPE 6) High Suppression Pool Water Level	Tier #	1
	Group #	2
	K/A #	2.4.6
2.4.6 Knowledge of EOP mitigation strategies	Rating	4.7

Question 83

The following conditions exist during a LOCA:

- Reactor pressure is 150 psig, stable.
- Reactor water level is -160" CFZ, lowering 1 inch/min.
- Torus water level is 16.2 ft., rising 0.2 inch/min.
- Condensate Booster Pump A and Condensate Pump A are the only available injection source to the RPV.

Which of the following strategies is the CRS required to direct for the current conditions?

A. Continue injection with the Condensate system IAW EOP-1A.
Emergency depressurize IAW EOP-2A.

B. Continue injection with the Condensate system IAW EOP-1A.
Reduce Torus water level using RHR IAW 2.2.69.3.

C. Secure Condensate system injection IAW EOP-3A.
Emergency depressurize IAW EOP-2A.

D. Secure Condensate system injection IAW EOP-3A.
Reduce Torus water level using RHR IAW 2.2.69.3.

Answer: A

Explanation:
This question requires prioritization of potentially contradictory EOP actions. When SP level cannot be maintained below 16.0 ft, EOP-3A step SP/L-5 directs securing injection systems that take suction from outside primary containment, if adequate core cooling can be assured. In this case, RPV level is below -160 inches and lowering; therefore, Condensate pump A injection is necessary for adequate core cooling. Adequate core cooling is maintained, for the present, by level above -183 inches with Condensate A injection. Since level is lowering and no other injection systems are available, EOP-1A step RC/L-15 should be answered NO, that level cannot be maintained above -183 inches, the point at which adequate core cooling will be lost, resulting in Emergency Depressurization is required. Thus, answer A is correct.

Distracters that include securing condensate pump A, which takes suction from the hotwell, are plausible, since EOP-3A step SP/L-5 directs securing injection systems that take suction from outside primary containment. This includes answers C and D. However, these answers are wrong because that injection should only be secured if other systems are available to assure adequate core cooling are available, but the stem states only Condensate pump A is available.

Answers that include reducing Torus water level using RHR are plausible since EOP-3A steps SP/L-1 and SP/L-3 direct using RHR to control SP level, in this case by rejecting water to radwaste or to the condenser. This includes answers B and D. These are wrong because for the given RPV level, a Group 2 isolation would be present, preventing opening of the RHR reject valves, and there is no provision to defeat that interlock.

Technical References:
EOP-1A, RPV Control; EOP-3A, Primary Containment Control, EPGs rev 3

References to be provided to applicants during exam:
None.

Learning Objective:
INT0320126Q0Q0100, Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	CNS 2015-11 Q85
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 3
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference	Level	SRO
295035 (EPE 12) Secondary Containment	Tier #	1
High Differential Pressure	Group #	2
	K/A #	EA2.01
Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE:	Rating	3.9
EA2.01 Secondary containment pressure		

Question 84

The plant is operating at 100% power when the following conditions occur:

- A Group 6 isolation has occurred on low RPV level
- Reactor Building differential pressure is +0.10 inches of water
- The maximum area temperatures in any quadrant are approximately 180ºF and steady
- Reactor Building Exhaust Rad Monitors are reading between 4 to 7 mR/hr and steady

What strategy is appropriate for the CRS?

A. Enter Procedure 2.1.5, REACTOR SCRAM, to lower radiation being released into the environment

B. Enter EOP 5A and defeat isolation interlocks as necessary in order to restart Rx Bldg HVAC and restore Secondary Containment differential pressure

C. Enter EOP-1A to ensure the plant is scrammed and EOP-2A to Emergency depressurize the vessel in order to restore Secondary Containment temperatures.

D. Enter Procedure 5.8.2, RPV DEPRESSURIZATION SYSTEMS, to anticipate Emergency depressurization with the main steam lines in order to lower energy being released into Secondary Containment

Answer: B

Explanation:
Since Secondary Containment differential pressure is less than -0.25 inches of water and the Reactor building ventilation isolated due to a Group 6 isolation caused by RPV level <-42 inches EOP-5A has an override to check the Rx Bldg Exhaust ventilation rad monitors and ensure they are less than 10 mR/hr and restart normal Rx Bldg ventilation to aid in temperature and pressure control. These instructions are located in an override to the concurrent steps for controlling Secondary Containment Temperature, Radiation and Water Level. Restoring Reactor Building HVAC aids in cooling the building but will restore secondary containment pressure to a negative value.
A is wrong. Plausible an applicant might choose this answer if they misinterpret the radiation levels as being too high. This answer is plausible because the actions stated are appropriate if conditions were of a higher order.
B is correct.
C is wrong. Plausible because an applicant might choose this answer if they misinterpret the

temperatures in the area of the leak and thought they were above the Max Safe levels of 195ºF which require an emergency depressurization.
D is wrong. Plausible because an applicant might choose this answer if they misinterpret the temperatures in the area of the leak and thought they were approaching the Max Safe levels of 195ºF which would allow them to anticipate emergency depressurization.

Technical References:
EOP-5A, Secondary Containment Control, Revision 19

References to be provided to applicants during exam:
None.

Learning Objective: INT008-06-17, EOP Flowchart 5A Secondary Containment Control and Radioactivity Release Control, Revision 25, Enabling objective 6

Question Source:	Bank #	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	CNS 2014-07 Q85
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	4
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference	Level	SRO
295013 (APE 13) High Suppression Pool Temperature	Tier #	1
	Group #	2
	K/A #	AA2.02
	Rating	3.5

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

AA2.02 Localized heating/stratification

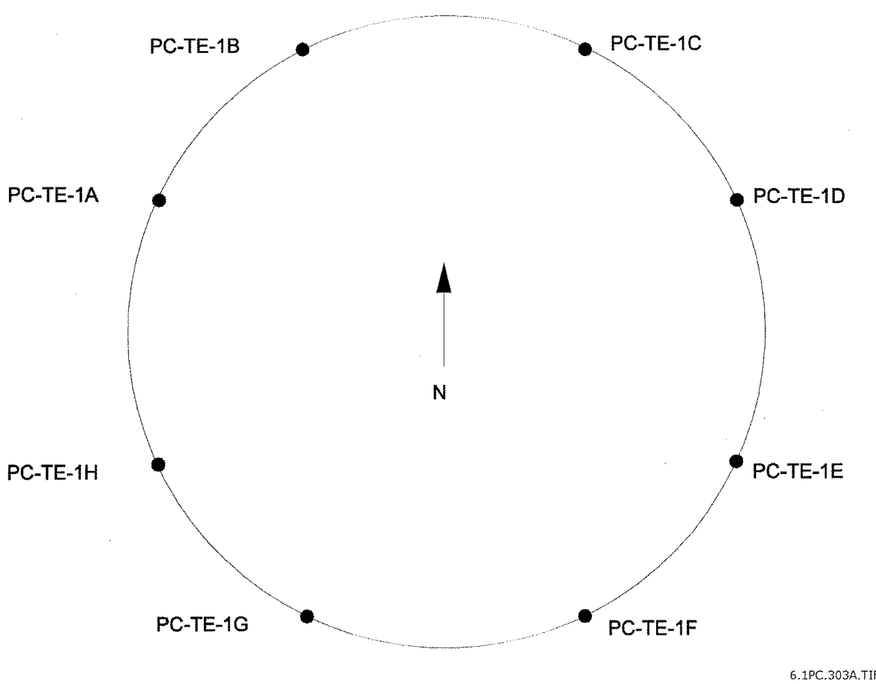
Question 85

REFERENCE PROVIDED

Given the following:

- Alarm J-1/A-1, SUPPR POOL DIV I WATER HIGH TEMP is in alarm
- PC-TE-1A, PC-TE-1B, PC-TE-1D, and PC-TE-1F have failed off-scale high
- I&C is troubleshooting
- All other RTDs are operating as designed

Post Accident Monitoring Instrumentation is ___(1)___ because ___(2)___.



A. (1) operable
(2) division II is operable

B. (1) operable
   (2) suppression pool temperature is adequately monitored with four RTDs operable

C. (1) inoperable
   (2) four RTDs are inoperable

D. (1) inoperable
   (2) PC-TE-1A and PC-TE-1B are inoperable

Answer: D

Explanation:
A is plausible for the reason given, because there is no impact to Div 2 RTD's. The reference does not reveal how many RTDs there are per division. The examinee must understand the two required channels are comprised of 8 RTDs for Div 1 and another 8 RTDs for Div 2 and that both Div 1 and Div 2 are required to be OPERABLE (all TS Bases knowledge). An examinee who does not understand the definition of "channel" for this instrumentation and who overlooks the reference to note (c) may believe a single OPERABLE RTD comprises a required "channel" and choose this answer.
B is plausible for the reason given, four RTDs are OPERABLE, as stated in note (C). An examinee who overlooks note (c) regarding adjacent RTDs, or who fails to recognize PC-TE-1A and PC-TE-1B are adjacent to one another, or who does not understand what is meant by "with no two adjacent RTDs inoperable" may choose this answer. (This may seem to be too simple, but it is no different than other Tech Spec questions where TS Condition statements are provided as a reference, and all the examinee must do is interpret the failures given and select the appropriate TS Condition statement. Here, the examinee is given a list of inoperable RTDs, and they must interpret that data with respect to a note, similar to a TS Condition statement.)
C is plausible because four of the eight RTDs shown in the stem diagram are inoperable. An examinee who does not know the definition of "channel" from TS Bases and overlooks the reference to note (c) may believe, with four RTDs inoperable, there are insufficient OPERABLE RTDs to meet the LCO. You may want to reword the answer "a total of four RTDs are inoperable".
D is correct because two adjacent RTDs are inoperable making Division I inoperable because this would impact the ability to detect localized heating (for example in one quadrant).

Technical References:
Technical Specification 3.3.3.1 table 3.3.3.1-1, Revision 4/3/19, page 3.3-25
Technical Specification Bases, Revision 10/15/19, page B3.3-66

References to be provided to applicants during exam:
Technical Specification 3.3.3.1 table 3.3.3.1-1, Revision 4/3/19, page 3.3-25 (table only)

Learning Objective:
INT-007-05-04, CNS Tech Spec 3.3, Instrumentation, Revision 24, Enabling objective 1

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

Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	3
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference	Level	SRO
262001 (SF6 AC) AC Electrical Distribution	Tier #	2
	Group #	1
Ability to (a) predict the impacts of the following on the A.C. ELECTRICAL DISTRIBUTION; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	K/A #	A2.04
	Rating	4.2
A2.04 Types of loads that, if deenergized, would degrade or hinder plant operation		

Question 86

The plant is in Mode 3 due to exceeding technical specification allowed outage time for damage to the ESST and DG-2 emergent maintenance.

The grid operator reports the 161kV line voltage is degrading, but stable voltage on the 69kV line.

161kV line voltage suddenly drops to zero.

SW-P-1A trips and SW-P-1C fails to start from the control room manually.

Five minutes have elapsed.

As the CRS you would enter _____.

A. 5.2SW and direct closing SW-P-1C breaker locally

B. 5.3EMPWR and direct stopping DG-1, THEN transition to 5.3SBO

C. 5.3AC-OUTAGE and direct placing SDG in service to power one safety bus

D. 5.3GRID and direct swapping the SSST to the 69kV line, THEN transition to 5.3EMPWR and direct stopping DG-1

Answer: B

Explanation:
A is wrong because 5.2SW does not have direction on starting service water locally. Plausible because 5.2SW entry is required for loss of SW pumps and if someone thinks they get the breaker closed in 5 minutes.
B is correct because 5.3EMPWR would be entered for loss of 4160 v buses 1A, 1B, and 1E due to loss of 161kV line, and DG1 would auto start and energize 4160V bus 1F. Per 5.3EMPWR, DG1 is required to be emergency stopped with no SW for 5 minutes. Stopping DG1 would result in loss of the only 4160V bus, 1F, requiring entry into 5.3SBO, which directs closing a SW pump breaker locally and restarting DG1.
C is wrong because the procedure is only entered if in modes 4 or 5. If the procedure was entered then placing the supplemental diesel in service would be correct if ESST and DG1

were unavailable. Since DG1 isn't unavailable yet, then SDG should not be placed in service. Plausible if someone thinks DG1 is unavailable.
D is wrong because only the ESST can be powered from the 69kV line. If 5.3GRID was entered, then an attempt to change voltage using load tap changer would be the correct answer. Plausible if someone forgets only the ESST has 2 power sources

Technical References:
5.3SBO, Station Blackout, Revision 46, p. 1, step 4.1.2

References to be provided to applicants during exam:
None.

Learning Objective: Document learning objective if possible.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	3
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference		Level	SRO
205000 (SF4 SCS) Shutdown Cooling		Tier #	2
		Group #	1
2.4.4 Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures		K/A #	2.4.4
		Rating	4.7

Question 87

The plant is shutdown in Mode 5. Fuel shuffles are in progress.

- RHR pump A is running in the SDC mode of operation
- RHR Pump C is in standby
- Reactor coolant temperature is 100 °F

RHR subsystem A was temporarily removed from service at 1200 at the request of the fuel handlers. The reactor coolant temperature is rising at a rate of 20 °F an hour.

(1) According to tech specs, what time must a RHR pump be placed back in service?

The control room restarted RHR Pump A with RHR flow established at 8,000 gallons per minute through the RHR heat exchanger. The crew notices that reactor coolant temperature is continuing to rise. One of the reactor operators receives a report that the service water inlet to the in-service RHR heat exchanger is closed..

(2) What procedure will the control room use to respond to malfunction?

A. (1) 1330
(2) 2.4SDC

B. (1) 1330
(2) 2.2.69.2 RHR System Shutdown Operations

C. (1) 1400
(2) 2.4SDC

D. (1) 1400
(2) 2.2.69.2 RHR System Shutdown Operations

Answer: C

Explanation:

Procedure 2.2.69.2, step 13.2.1 allows for SDC to be shutdown for less than 1.5 hours, time to boil divided by two, or RCS exceeding 190 °F. T.S. 3.9.7 allows for 2 hours. Precaution and limitation 2.14 states "If RHR SW lost to in service RHR HX, bypass RHR flow around HX until SW flow restored per Procedure 2.4SDC. This will prevent boiling water in tube side of HX which will cause a water hammer when SW flow is restored."

A is incorrect. Plausible because the procedure allows no more than 1.5 hours, and Part 2 is correct.
B is incorrect. Plausible because the procedure allows no more than 1.5 hours, and Part 2 is correct, and 2.2.69.2 is the controlling document initially and it directs the operators to restore SW using 2.4SDC
C is correct
D is wrong. Plausible Part 1 is correct, and because 2.2.69.2 is the controlling document and it directs the operators to restore SW using 2.4SDC.

Technical References:
2.2.69.2, RHR system Shutdown Operations, revision 106
2.4SDC, Shutdown Cooling Abnormal, revision 17

References to be provided to applicants during exam:
None.

Learning Objective:
COR002-23-02, Residual Heat Removal System, Revision 36, Enabling objective 8.r

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 3
10CFR Part 55 Content:	55.43.2	

Examination Outline Cross-Reference		Level	SRO
400000 (SF8 CCS) Component Cooling Water		Tier #	2
		Group #	1
2.4.2 Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions		K/A #	2.4.2
		Rating	4.6

Question 88

Given the following:

- The plant is at 100% reactor power
- REC SYSTEM LOW PRESSURE is in alarm - for PS-452A, DIV I
- REC SURGE TANK LOW LEVEL is in alarm
- A field operator reports a 1 gpm leak from the REC surge tank
- All other systems are operating as designed

__(1)__ is one of three valves automatically closed after a time delay; and the Technical Specification limit on REC leakage is based on __(2)__.

A. (1) REC-MO-702, DRYWELL SUPPLY ISOLATION
(2) maintaining 30 days of inventory in the REC surge tank

B. (1) REC-MO-702, DRYWELL SUPPLY ISOLATION
(2) a 14 day completion time providing lower plant risk than shutting down with an operable but degraded REC system

C. (1) REC-MO-713, HX B OUTLET
(2) maintaining 30 days of inventory in the REC surge tank

D. (1) REC-MO-713, HX B OUTLET
(2) a 14 day completion time providing lower plant risk than shutting down with an operable but degraded REC system

Answer: A

Explanation:
A is correct because REC System Low Pressure alarm from PS-452A automatically closes valves REC-MO-700, REC-MO-702, and REC-MO-1329 and because the TS bases state that the REC leakage specification is to maintain a 30 day inventory in the surge tank without refilling it.
B is wrong because the 14 day allowed completion time for risk is the action if the REC was inoperable in conjunction with one train of SW also being inoperable; is plausible because it is part of the REC Technical Specification allowed completion times and because part 1 is correct.
C is wrong because REC-MO-713 is automatically closed by PS-452B2 not PS452A; is plausible because part 2 is correct.
D is wrong because REC-MO-713 is automatically closed by PS-452B2 not PS452A and because the 14 day allowed completion time for risk is the action if the REC was inoperable in conjunction with one train of SW also being inoperable; is plausible because PS-452B2

does automatically shut REC isolation valves and because the 14 day completion time is part of the REC Technical Specification allowed completion time.

Technical References:
TS Bases section 3.7.3, page B 3.7.13

References to be provided to applicants during exam:
None.

Learning Objective:
COR002-19-02, Reactor Equipment Cooling, Revision 31, Enabling objective 11.c

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	3
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference	Level	SRO
215004 (SF7 SRMS) Source-Range Monitor	Tier #	2
	Group #	1
2.1.32 Ability to explain and apply system limits and precautions	K/A #	2.1.32
	Rating	4.0

Question 89

What is the basis for Technical Specification 3.3.1.2, Source Range Monitoring Instrumentation, for the minimum required number of operable Source Range Monitor Instruments while in MODE 2 and IRMs on Range 2 or below?

A. To provide monitoring and indication before and during approach to criticality ONLY.

B. To provide indication, alarm, control rod block and scram functionality during approach to criticality.

C. To ensure that Safety Function 1 (reactivity control) is maintained during the applicable safety analysis events IAW the USAR.

D. To provide adequate representation in all four quadrants of the core during those periods when reactivity changes are occurring throughout the core.

Answer: A

Explanation:
A is correct. Per the bases, information on page B 3.3-32, it states that “However, this LCO specifies OPERABILITY requirements only for monitoring and indication functions of the SRMs.”
B is wrong because per the bases other TS areas contain the aspects of alarm and rod block information (such as 3.9.1, 3.3.1.1 for RPS, 3.3.2.1 for control rod blocks, etc).
C is wrong because SRMs have no safety function and are not assumed to function during any USAR design basis accident or transient analysis.
D is wrong because only three are required per the LCOs, therefore all four quadrants cannot be monitored with only three SRMs.

Notes for the KA match-minimum number of SRMs is a system limit/precaution so it meets the KA. The explain the limit aspect is contained in the bases for the limit.

Technical References:
Technical Specification Bases, page B 3.3-32, Amendment 11/25/12
INT007-05-04, CNS Tech Specs 3.3, Instrumentation, Revision 24

References to be provided to applicants during exam:
None.

Learning Objective: INT007-05-04, CNS Tech Specs 3.3, Instrumentation, Revision 24, Enabling objective 2

Question Source: Bank #

(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	3
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.2	

Examination Outline Cross-Reference	Level	SRO
300000 (SF8 IA) Instrument Air Ability to (a) predict the impacts of the following on the INSTRUMENT AIR SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: A2.01 Air Dryer and filter malfunctions	**Tier#**	2
	Group#	1
	K/A #	300000 A2.01
	Rating	2.8
	Revision	0
Revision Statement:		

Question 90

The plant is at 100% power with RCIC tagged out of service.

Instrument air header pressure lowers to 75 psig due to failure of the in-service air dryer/filter.

IAW with Procedure 5.2AIR, Loss of Instrument Air,

The CRS is required to execute Procedure 5.2AIR, IA Pressure Loss, Attachment 2 ___(1)___ (concurrently with / after) procedure body instructions.

AND

MSIVs are required to be closed ___(2)___ (before/after) HPCI is placed in service IAW Procedure 2.2.33.1, High Pressure Coolant Injection System, for level control.

A. (1) concurrently with
(2) before

B. (1) concurrently with
(2) after

C. (1) after
(2) before

D. (1) after
(2) after

Answer: B
Explanation: B is correct as stated in the information section of 5.2AIR that subsequent actions are to be performed with the attachments. At 77 psig you are required to transfer level control before you take away the high pressure source by closing the MSIVs and losing the RFPs.
Distracters: Answer A part 1 is correct. Part 2 distractor is plausible because RFPTs are not removed from service until after MSIVs are closed. The reactor is scrammed before MSIVs are closed, so the inventory makeup requirement is low when MSIVs are directed to be closed. An examinee may believe there is a sense of urgency to close MSIVs before they drift close on loss of air and that the sequence is to close MSIVs, place HPCI in service, then trip RFPTs. It is wrong because 5.2AIR step 4.9.2 states place HPCI/RCIC in service, then step 4.9.3 states close MSIVs. Answer C Part 1 distractor plausible because Att. 2 is only performed when IA pressure is too low to support continued operation. An examinee may believe it contains the priority actions. This answer is wrong because 5.2AIR Att 3 states Attachment 2 is to be performed in conjunction with the procedure body instructions. Part 2 is plausible for the reasons listed in distractor A. Answer D part 1 is plausible for the reasons listed in distractor A. Part 2 is plausible for the reasons listed in distractor A.
Technical References: 5.2AIR Loss of Instrument Air 2.2.59 PLANT AIR SYSTEMS 2.3_AIR DRYER
References to be provided to applicants during exam: none
Learning Objective:
COR0011702001070A Given a specific Plant Air system malfunction, determine the effect on any of the following: a. Plant operation

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	

	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3	
SRO Only Justification:		
Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps.		
PSA Applicability:		
N/A		

Examination Outline Cross-Reference	Level	SRO
204000 (SF2 RWCU) Reactor Water Cleanup	Tier #	2
	Group #	2
Ability to (a) predict the impacts of the following on the REACTOR WATER CLEANUP SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	K/A #	A2.10
	Rating	2.8

A2.10 Valve closures

Question 91

The current time is 0830 and the plant is at 100% power. RWCU A is in service.

The following indications are observed 5 minutes later:

- Annunciator alarm 9-4-2/B-4, RWCU PUMP A LOW FLOW
- RWCU-MO-15 is stuck in mid-position

RWCU-MO-18 must be closed with power removed no later than _____.

A. 0935

B. 1235

C. 1635

D. 2035

Answer: B

Explanation:

The RWCU flow path has two PCIVs. RWCU-MO-15 and RWCU-MO-18 are the primary containment isolation valves (PCIVs) for RWCU. They are required to be Operable in Mode 1. Only one is Inoperable and thus only Condition A is required to be entered. The other PCIV is required to be closed within 4 hours

RWCU-MO-15 is off its open seat and stuck mid position

A is wrong but plausible because one hour is the time requirement if both RWCU-MO-15 and RWCU-MO-18 were Inoperable.
B is correct.
C is wrong but plausible because eight hours is the time requirement for main steam lines.
D is wrong but plausible because twelve hours is the time requirement for EFCVs.

Technical References:
Technical Specification Bases, Revision 10/15/19, page 3.6-19

References to be provided to applicants during exam:
None.

Learning Objective:
COR001-20-01, Reactor Water Cleanup, Revision 25, Enabling objective 2

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 3
10CFR Part 55 Content:	55.43.1	

Examination Outline Cross-Reference	Level	RO
226001 (SF5 RHR CSS) RHR/LPCI: Containment Spray Mode	Tier #	2
	Group #	2
	K/A #	A2.11
	Rating	3.0

Ability to (a) predict the impacts of the following on the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

A2.11 Motor operated valve failures

Question 92

REFERENCE PROVIDED

Given the following:

- The plant was scrammed from 100% power due to a LOCA
- RHR-MO-65B, HX-B INLET VLV is closed and will not open
- RHR pump C tripped on overcurrent
- Drywell pressure is 7 psig and slowly rising
- Torus pressure is 10 psig and slowly rising
- PC water level is 8 feet and stable
- EOP 3A has been entered

While lining up for Torus Spray per EOP 3A using RHR pump A, the RO reports that RHR-MO-39A, SUPPR POOL COOLING/TORUS SPRAY VLV, is closed and will not open.

Operators should declare ___(1)___ and ___(2)___.

A. (1) the A train of Containment Spray inoperable
(2) emergency vent the PC per EOP 5.8.18, Primary Containment Venting for PCPL, PSP

B. (1) the A train of Containment Spray inoperable
(2) emergency depressurize per EOP 5.8.2, RPV Depressurization Systems

C. (1) both trains of Containment Spray inoperable
(2) emergency vent the PC per EOP 5.8.18, Primary Containment Venting for PCPL, PSP

D. (1) both trains of Containment Spray inoperable
(2) emergency depressurize per EOP 5.8.2, RPV Depressurization Systems

Answer: D

Explanation:
A is wrong because EOP-3A directs operators to emergency depressurize under these

conditions not vent the primary containment, and because both trains of Containment Spray are inoperable per the TS bases; and is plausible because train A of Containment Spray is inoperable and because venting the primary containment is the next step in the flowchart (once conditions are met on graph 11 – which they are not at this time).
B is wrong because Both trains of Containment spray are inoperable. Containment Spray B is also inoperable due to the fact that the HX outlet valve will not open and the HX is required to consider the subsystem operable.
C is wrong because EOP-3A directs operators to Emergency Depressurize under these conditions not vent the primary containment; is plausible because both trains of Containment Spray are inoperable.
D is correct because both trains of Containment Spray are inoperable by the definition of operable in TS bases 3.6.1.9 and because torus pressure cannot be maintained below PSP during a postulated accident since level has lowered below 9.5 feet in the Torus, therefore emergency depressurization is necessary.

Technical References:
EOP-3A
TS Bases 3.6.1.9

References to be provided to applicants during exam:
EOP Graph 10

Learning Objective:
INT008-06-13, EOP Flow Chart 3A, Revision 23, Enabling objective 11

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	3
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference	Level	SRO
202001 (SF1, SF4 RS) Recirculation	Tier #	2
	Group #	2
2.4.6 Knowledge of EOP mitigation strategies	K/A #	2.4.6
	Rating	4.7

Question 93

Scram signal present due to high drywell pressure. The following conditions are present:

Drywell pressure is 12 psig and stable
Torus pressure is 8 psig and stable
Drywell temperature is 270°F and rising at 5°F per minute
APRM data is not available
IRM are reading 100/125 of full scale on range 6 on all IRM
RPV water level is at 0 inches after being below -42 inches WR for 7 seconds
Recirc pumps are running at minimum speed

You order the ATC operator to trip the recirc pumps due to _____.

A. torus spray is required

B. drywell spray is required

C. reactor power being above 3%

D. RPV water level dropped below -42 inches WR for 7 seconds

Answer: B

Explanation:
A is wrong because in EOP 3A torus spray is required before you reach 10 psig in the torus. Since torus pressure is stable spray is not required.
B is correct because in EOP 3A drywell spray is required before you reach 280°F. Since temperatures are rising this block is met. Whenever you spray the drywell you are required to make sure all drywell electrical components are turned off.
C is wrong because in EOP 6A the block asks if power is above 3% or cannot be determined then trip the recirc pumps. That IRM level corresponds to ~1% power so tripping of recirc pumps is not required.
D is wrong because recirc pumps should trip if water level is below level 2 after a 9 second time delay. This isn't in an EOP and a setpoint so could be considered RO knowledge distractor. I think it is useful here for verifying an automatic action has occurred in reference to conduct of ops. The only other mention of recirc pumps in EOP is in 6A to runback the pumps in an ATWS if the turbine is online. Didn't want to have 3 trips and 1 runback as answers.

SRO-only due to knowledge of diagnostic steps and decision points.

Technical References:
EOP 3A, "Primary Containment Control"

References to be provided to applicants during exam:
None.

Learning Objective: INT008-06-13,, OPS EOP Flowchart 3A - Primary Containment Control, Revision 23, Enabling objectives 11 and 12

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	3
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference	Level	SRO
2.2.21 Knowledge of pre- and post-maintenance operability requirements	Tier #	3
	Group #	
	K/A #	2.2.21
	Rating	4.1

Question 94

Instrument and Controls (I&C) technicians are performing a calibration of an instrument covered by technical specifications.

While performing a calibration, the I&C technicians report to you that the AS FOUND condition is outside the administrative limits.

What actions are required by Procedure 0.26, Surveillance Program?

A. report to SM or initiate a Condition Report for the existing condition only

B. it shall be identified as a discrepancy, immediately reported to SM, and a Condition Report initiated only

C. it shall be identified as a discrepancy, immediately reported to SM, a Condition Report initiated, and system/component declared inoperable by SM only

D. it shall be identified as a discrepancy, immediately reported to the SM, a Condition Report initiated, and action taken as specified in controlling document only

Answer: D

Explanation:
A is wrong. Plausible because this would be the correct action if the instrument were new and being calibrated prior to being placed in service
B is wrong. Plausible because this would be the correct action if a calibration were being performed on an instrument not covered by TS, TRM, or ODAM
C is wrong. Plausible because this would be the correct action if the AS FOUND or AS LEFT data was outside the limits of TS, TRM, or ODAM
D is correct.

Technical References:
Procedure 0.26, Surveillance Program, revision 71

References to be provided to applicants during exam:
None.

Learning Objective:
INT0320101R17, revision 17, enabling objective G.1.j.

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

	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
10CFR Part 55 Content:	55.43.2	

Examination Outline Cross-Reference	Level	SRO
2.4.47 Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material	Tier #	3
	Group #	
	K/A #	2.4.47
	Rating	4.2

Question 95

REFERENCE PROVIDED

At 1000, reactor startup is in progress with the following:

- The Mode Switch is in Startup
- All SRMs are indicating 3 x 10^2 CPS
- All IRMs on Range 2
- SRM ‘A’ is declared **INOPERABLE**.

At 1010, the following conditions exist:

- All SRMs are declared **INOPERABLE** based on a common mode failure
- All IRMs on Range 2

Identify which of the following complete the requirement(s) of Technical Specification 3.3.1.2, Source Range Monitor Instrumentation?

A. Restore three SRMs to operable status by 1410 ONLY

B. Suspend rod withdrawal at 1010 and restore two SRMs to operable status by 1410

C. Suspend rod withdrawal at 1010 and restore three SRMs to operable status by 1410

D. Fully insert all insertable control rods and place the Mode Switch in SHUTDOWN no later than 1110

Answer: C

Explanation:
A is wrong because action statements A.1 must be entered for one or more required SRMs inoperable AND B.1 for three required SRMs inoperable; is plausible because action A.1 is one of the two required TS actions and the time limit is correct.
D is wrong because action statements A.1 must be entered for one or more required SRMs inoperable AND B.1 for three required SRMs inoperable is plausible because it would be partially correct in Mode 3 or Mode 4.
C is correct because action A.1 to restore all SRMs to operable and action B.1 to suspend rod withdrawal are the correct actions. By restoring all SRM’s to operable status this achieves the requirement of A.1 to restore all required SRM to operable status which is 3.
B is wrong because all SRMs must be returned to operable not 2; is plausible because in Modes 3 or 4 only two SRMs would be required to be returned to operable.

Technical References:
Technical Specification 3.3.1.2, Source Range Monitor (SRM) Instrumentation, Amendment No. 178

References to be provided to applicants during exam:
Technical Specification 3.3.1.2 without table 3.3.1.2-1

Learning Objective:
INT007-05-04, Technical Specification 3.3, Instrumentation, Revision 24, Enabling objective 3

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 Grand Gulf 2009 NRC Exam Q87
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 2
10CFR Part 55 Content:	55.43.5	

PARENT QUESTION

Question 87

Use your provided references to answer this question.

At 0800, a reactor startup is in progress with the following:

- All SRMs are indicating 2 x 10^2 CPS
- All IRMs are on Range 2

At 0800, SRM 'A' is declared inoperable.

At 0810, the following conditions exist:

- All SRMs are declared inoperable based on a common mode failure
- All IRMs are on Range 2

Which of the following describes the **EARLIEST** action required by Technical Specification 3.3.1.2, Source Range Monitor Instrumentation?

A. Restore all SRMs to an OPERABLE status no later than 1200.

B. Suspend rod withdrawal at 0810.

C. Begin implementing the actions as required by Tech Spec 3.0.3 at 0810.

D. Fully insert all insertable control rods and place the Mode Switch in SHUTDOWN no later than 0910.

Answer: B

Examination Outline Cross-Reference	Level	SRO
2.2.37 Ability to determine operability and/or availability of safety related equipment	Tier #	3
	Group #	
	K/A #	2.2.37
	Rating	4.6

Question 96

It has just been discovered that a Surveillance with an 18 Month Frequency was missed last refueling outage six months ago. There are no indications of any problems with the related equipment.

Is the equipment OPERABLE? Why or why not?

A. NO. The affected equipment was inoperable four months ago when the Surveillance Requirement was missed.

B. NO. The affected equipment is immediately inoperable at the time it is known that the Surveillance Requirement was not performed within the required frequency.

C. YES. A Surveillance Requirement may be missed one time within a cycle, as long as it is done at the next scheduled time plus 25% at the latest.

D. YES. Up to 24 hours or up to the limit of the specified Frequency, whichever is greater, may be taken to perform the missed Surveillance Requirement.

Answer: D

Explanation:
SR 3.0.3 - If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

A is incorrect but plausible. Inoperabilities are not determined at the last scheduled date although it would be reasonable to believe it should.

B is incorrect but plausible. It is reasonable to believe the equipment is inoperable at the time of discovery.

C is incorrect but plausible. While the equipment is operable, the 1.25 times Frequency for performance (SR 3.0.2) does not apply to missed surveillances.

D is correct.

Technical References:
Technical Specification SR 3.0.3

References to be provided to applicants during exam:
None.

Learning Objective:
INT007-05-01, Introduction to Technical Specifications, Revision 26, Enabling objective 3.f

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	CNS 2012-10 Q95
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 2
10CFR Part 55 Content:	55.43.2	

Examination Outline Cross-Reference	Level	SRO
2.3.6 Ability to approve release permits	Tier #	3
	Group #	
	K/A #	2.3.6
	Rating	3.8

Question 97

IAW Procedure 8.8.1, Liquid Radioactive Waste Discharge Authorization,

the Shift Manager is completing Section 4, Shift Manager Approval to Release, of the Liquid Radioactive Waste Discharge form for release of the Floor Drain Sample Tank (FDST).

(1) What is the minimum dilution flow rate required in order to approve this discharge?

AND

(2) What is the minimum action required if liquid discharge flow rate rises above the limit specified on the Liquid Radioactive Waste Discharge Form?

A. (1) 159,000 gpm
(2) Immediately terminate the discharge

B. (1) 159,000 gpm
(2) Immediately reduce discharge flow rate to within the specified limit

C. (1) 198,000 gpm
(2) Immediately terminate the discharge

D. (1) 198,000 gpm
(2) Immediately reduce discharge flow rate to within the specified limit

Answer: A

Explanation:
This question tests SRO knowledge required for authorizing initiation of and for controlling liquid radioactive effluent discharges. Procedure 8.8.11 states the minimum dilution flow rate for liquid radioactive discharges is 159,000 gpm. If either dilution flow lowers below the minimum or liquid effluent flow rate rises above the limit listed on the discharge permit, the release must be immediately terminated.

Answer B part 1 is correct. Part 2 is plausible because there are other actions that allow restoring a parameter to within limits before more drastic actions are required, such as for many TS actions. It is wrong because Procedure 8.8.11 requires immediate termination of the release.

Answer C part 1 is plausible because it represents the flow rate of one CW pump listed on procedure 8.8.11 Attachment 1. It is wrong because it is not the minimum dilution rate, as required by the stem. Part 2 is correct.

Answer D part 1 is plausible and wrong for the reasons stated for distractor C. Part 2 is plausible and wrong for the reasons stated for distractor B.

Technical References: procedure 8.8.11, Liquid Radioactive Waste Discharge Authorization

References to be provided to applicants during exam: None

Learning Objective: INT0320115 EO-B3, State the number of Circulating Water Pumps required to be in service during liquid radioactive discharges.
Technical References: procedure 8.8.11, Liquid Radioactive Waste Discharge Authorization

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
10CFR Part 55 Content:	55.43.4	

C. Facility Licensee Procedures Required To Obtain Authority for Design and Operating Changes in the Facility [10 CFR 55.43(b)(3)]

Some examples of SRO exam items for this topic include the following:

- screening and evaluation processes under 10 CFR 50.59, "Changes, Tests and Experiments"
- administrative processes for temporary modifications
- administrative processes for disabling annunciators
- administrative processes for the installation of temporary instrumentation
- processes for changing the plant or plant procedures

Section IV provides an example of a satisfactory SRO-only question related to this topic.

D. Radiation Hazards That May Arise during Normal and Abnormal Situations, including Maintenance Activities and Various Contamination Conditions [10 CFR 55.43(b)(4)]

Some examples of SRO exam items for this topic include the following:

- process for gaseous/liquid release approvals (i.e., release permits)
- analysis and interpretation of radiation and activity readings as they pertain to the selection of administrative, normal, abnormal, and emergency procedures
- analysis and interpretation of coolant activity, including comparison to emergency plan criteria and/or regulatory limits

SRO-only knowledge should not be claimed for questions that can be answered *solely* based on RO knowledge of radiological safety principles (e.g., radiation work permit requirements, stay time, and DAC hours).

E. Assessment of Facility Conditions and Selection of Appropriate Procedures during Normal, Abnormal, and Emergency Situations [10 CFR 55.43(b)(5)]

This 10 CFR 55.43 topic involves both (1) assessing plant conditions (normal, abnormal, or emergency) and then (2) selecting a procedure or section of a procedure to mitigate or recover, or with which to proceed. One area of SRO-level knowledge (with respect to selecting a procedure) is knowledge of the content of the procedure versus knowledge of the procedure's overall mitigative strategy or purpose.

4.19.2.8 Calculate '2 x Alarm Limit'.

a. Record value in Attachment 1, Section 4.

4.19.2.9 Calculate '200 x Alarm Limit'.

a. Record value in Attachment 1, Section 4.

4.19.2.10 Record Maximum liquid waste discharge rate on Attachment 1, Section 3.

4.19.2.11 Date and sign hand calculations, and file per Step 4.22.

<u>NOTE</u> – When completed, Attachment 1, Section 3, provides formal authorization to Shift Manager to perform release of radioactive liquid effluent per dilution requirements specified.

4.20 Record time and date, and sign Attachment 1, Section 3.

4.21 (Checked By) Validate Chemistry portion of the Liquid Waste Discharge has been correctly completed.

Performed By: ________

Checked By: ________

4.22 Place following items, if available, in Liquid Waste Discharge Log:

4.22.1 Attachment 1 - copy.

4.22.2 Isotopic analysis results - gamma spectrometer printout.

4.22.3 Accumulated Personal Dose Report - computer printout.

4.22.4 Alarm setpoint - computer printout or hand calculation.

4.23 Return Attachment 1 to Shift Manager to effect release.

<u>NOTE</u> – Exact flows cannot be pre-determined for this procedure. They have to be determined dependent on plant operating condition at time release is affected.

4.24 Shift Manager determines circulating water pump requirement for proper flow to discharge canal to comply with required dilution flow.

4.25 Shift Manager shall tag running circulating water pumps with DISCHARGE IN PROGRESS tags to ensure pumps remain running until completion of discharge.

4.26 IF dilution flow falls below value specified in Attachment 1, Section 3, THEN immediately secure discharge.

ATTACHMENT 1 LIQUID RADIOACTIVE WASTE DISCHARGE FORM

Section 1. REQUEST FOR ANALYSIS OF RADIOACTIVE LIQUID WASTE PRIOR TO DISCHARGE

To: Chemistry From: Shift Manager Tank To Be Discharged: ______

Started Recirculation For Sample: Time: ______ Date: ______

Recirculation Of Tank Complete: Time: ______ Date: ______

Estimated Volume To Be Discharged: ______

Shift Manager: ______ Time: ______ Date: ______

Section 2. THIS SECTION TO BE COMPLETED BY CHEMISTRY

Monitor Source Check

Informed Control Room And Performed Source Check Initials: ______

Monitor Background: ______ Monitor Source Check Value: ______

Sample Point: ______ Sample Time: ______ Date: ______

Signature: ______

Section 3. AUTHORIZATION TO RELEASE RADIOACTIVE LIQUID WASTE

To: Shift Manager From: Chemistry Release Authorization Number: ______

Total μCi/ml: ______

Total Concentration is < 1.0E-02 μCi/ml YES/NO

Signature: ______

31 Day Dose, Percent Of Annual Limit For Each Value Is ≤ 2.0E+00 YES/NO

Signature: ______

You Are Authorized To Release Subject Tank With Following Restrictions:

Maximum Liquid Waste Discharge Rate (gpm): ______

Minimum Dilution Flow To Canal (gpm): 159,000 ______

Discharge Monitor Alarm Setpoint (μCi/ml): ______

NOTE – Terminate Discharge If Above Specifications Cannot Be Maintained. Contents Of This Tank Are Within Chemical Parameters For Discharge.

Chemistry: ______ Time: ______ Date: ______

Section 4. SHIFT MANAGER APPROVAL TO RELEASE

4.1 Circle Appropriate Discharge Canal Flow Rate:

NUMBER OF OPERATING CW PUMPS	AVERAGE CW DISCHARGE FLOWRATE (gpm)	
	DE-ICING	NO DE-ICING
4	378,600	631,000
3	308,400	514,000
2	193,200	322,000
1	118,800	198,000

4.2 To: Operations Personnel From: Shift Manager

The Subject Tank Contents Are Approved For Release Subject To The Following Restrictions:

1) Maximum Liquid Disch Rate: __________ gpm (Section 3)

2) Minimum Dilution Flow To Canal Of: __________ gpm (Section 3)

3) Alarm Limits Specified (Section 3)

2 x Alarm Limit: ____________________

200 x Alarm Limit: __________________

4) Tank Volume Verified: ____________ (Compare To Section 1)

5) DISCHARGE IN PROGRESS Tags Installed On Running Circ Water Pumps.

Approval To Release:

Shift Manager: ______________________________ Time: __________ Date: __________

Examination Outline Cross-Reference	Level	SRO
2.1.36 Knowledge of procedures and limitations involved in core alterations.	Tier #	3
	Group #	
	K/A #	2.1.36
	Rating	4.1

Question 98

Which activity REQUIRES Refuel Floor Supervisor permission during Core Alterations?

A. Shifting shutdown cooling trains

B. Suspending fuel handling operations

C. Allowing access to the fuel handling area on the refuel floor

D. Using greater than 10 gallons of demineralized water on the refuel floor

Answer: C

Explanation:

Answer A is incorrect because Refuel Floor Supervisor permission is not required to allow shutdown cooling operations. This answer is plausible because shifting shutdown cooling trains can affect water clarity, which could cause the Refuel Floor Supervisor to delay fuel handling.
Answer C is plausible because Refuel Floor Supervisor permission is required to recommence fuel handling operations IAW Attachment 4 (Reset Checklist) which shall be used each time the normal fuel handling process is stopped/interrupted. It is wrong because fuel handling may be immediately suspended due to a variety of reasons, such as equipment failure, without Refuel Floor Supv permission.
Answer B is correct because movement of fuel within the RPV is a Core Alteration. Refuel SRO responsibilities are, by nature, generic. Procedure 2.2.31 [Fuel Handling – Refueling Platform] step 2.10 states access to fuel handling area on refueling floor and to overhead bridge crane when fuel handling is in-progress shall be limited to authorized personnel, and that individual authorization is determined by the Refuel Floor Supervisor. Step 3.3.1 states the Refuel Floor Supervisor must be a SRO when Core Alterations are in progress.
Answer D is incorrect because Refuel Floor Supervisor permission is not required to use greater than 10 gallons of demineralized water on the refuel floor. This choice is plausible due to the Refuel floor SRO is required to brief available refueling floor personnel on limiting demineralized water usage and requirement to notify Control Room if using > 50 gallons demineralized water each shift. The applicant who confuses briefing vs. giving permission would choose this answer.

Technical References:
Procedure 2.2.31, Fuel Handling – Refueling Platform, Revision 56

References to be provided to applicants during exam:
None.

Learning Objective:

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	CNS 2017-03 Q95
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
10CFR Part 55 Content:	55.43.6	

Examination Outline Cross-Reference	Level	SRO
2.4.37 Knowledge of the lines of authority during implementation of the emergency plan	Tier #	3
	Group #	
	K/A #	2.4.37
	Rating	4.1

Question 99

The following conditions exist:

- CNS has experienced a major accident
- Nebraska Public Power District has implemented its Emergency Plan for Cooper Nuclear Station
- General Emergency has been declared
- Severe Accident Guidelines (SAGs) have been entered and are being implemented
- All required emergency plan positions have been staffed

Which individual is provided with the decision-making authority for providing direction to the control room related to accident mitigation actions?

A. Emergency Director

B. Technical Support Center Director

C. Operations Coordinator

D. Shift Manager

Answer: C

Explanation:
A is wrong, but plausible if thought the ED has ultimate authority of EP activities.
B is wrong, but plausible because the Operations Director is part of the TSC team, and it may be believed that the TSC Director has authority over the Operations Coordinator.
C is correct.
D is wrong, but plausible because this would be correct prior to implementation of SAMGs.

Technical References:
NPPD Emergency Plan for CNS, Page 36

References to be provided to applicants during exam:
None.

Learning Objective: LP ERO001-01-13, EP Fundamental – Emergency Response, Revision 1, Enabling objective 1C

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

	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference	Level	SRO
2.1.35 Knowledge of the fuel-handling responsibilities of SROs. (CFR: 41.10 / 43.7)	**Tier#**	3
	Group#	
	K/A #	G2.1.35
	Rating	3.9
	Revision	0
Revision Statement:		

Question 100

The plant is in Mode 5.

IAW Procedure 2.0.3 [Conduct of Operations], the Refueling SRO is required to be directly in charge of which one of the following refueling activities?

A. Inserting a new LPRM string into the core

B. Removing a depleted fuel bundle from the core

C. Removing a control rod blade from a defueled cell

D. Using more than 50 gallons of demineralized water

Answer: B

Explanation:
This is a modified version of 2020-4 ILT NRC Q#100. It was modified by changing the stem from which activity requires Refuel SRO permission to which activity is the Refuel SRO in direct charge of and by changing the correct answer and two distractors.

Procedure 2.0.3 step 12.3.1 states an active Licensed SRO (Refuel Floor SRO) with no other concurrent duties shall be directly in charge of core alterations. TS defines a core alteration as:

the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS:

a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and

b. Control rod movement, provided there are no fuel assemblies in the associated core cell.

Removing a spent fuel bundle from the core is a core alteration.

Distracters:
Answer A is plausible because LPRMs contain U-235 and an examinee could consider a new LPRM detector string to be fuel. It is wrong because movement of LPRMs is specifically exempted from the definition of CORE ALTERATION, so Refueling SRO supervision is not required.

Answer C is plausible because removal of a control rod from a cell containing fuel is a CORE ALTERATION. It is wrong because removal of a control rod is not considered to be a CORE ALTERATION if the cell contains no fuel assemblies.

Answer D is plausible because the Refueling SRO is required to brief refueling floor personnel to notify the control room if >50 gpm of demin water is required to be used. It is wrong because the Refueling SRO is not required to be directly in charge of using demin water on the refuel floor.

Technical References: TS 1.1 [Definitions], Procedure 2.0.3 [Conduct of Operations](Rev 104), procedure 2.1.20.1 [Restoration from Refueling](Rev 43)

References to be provided to applicants during exam: none

Learning Objective: INT0231002001160A Identify the administrative duties and responsibilities of the each of the following: Refueling Floor Supervisor

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	2020-4 NRC Exam #100
	New	

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X

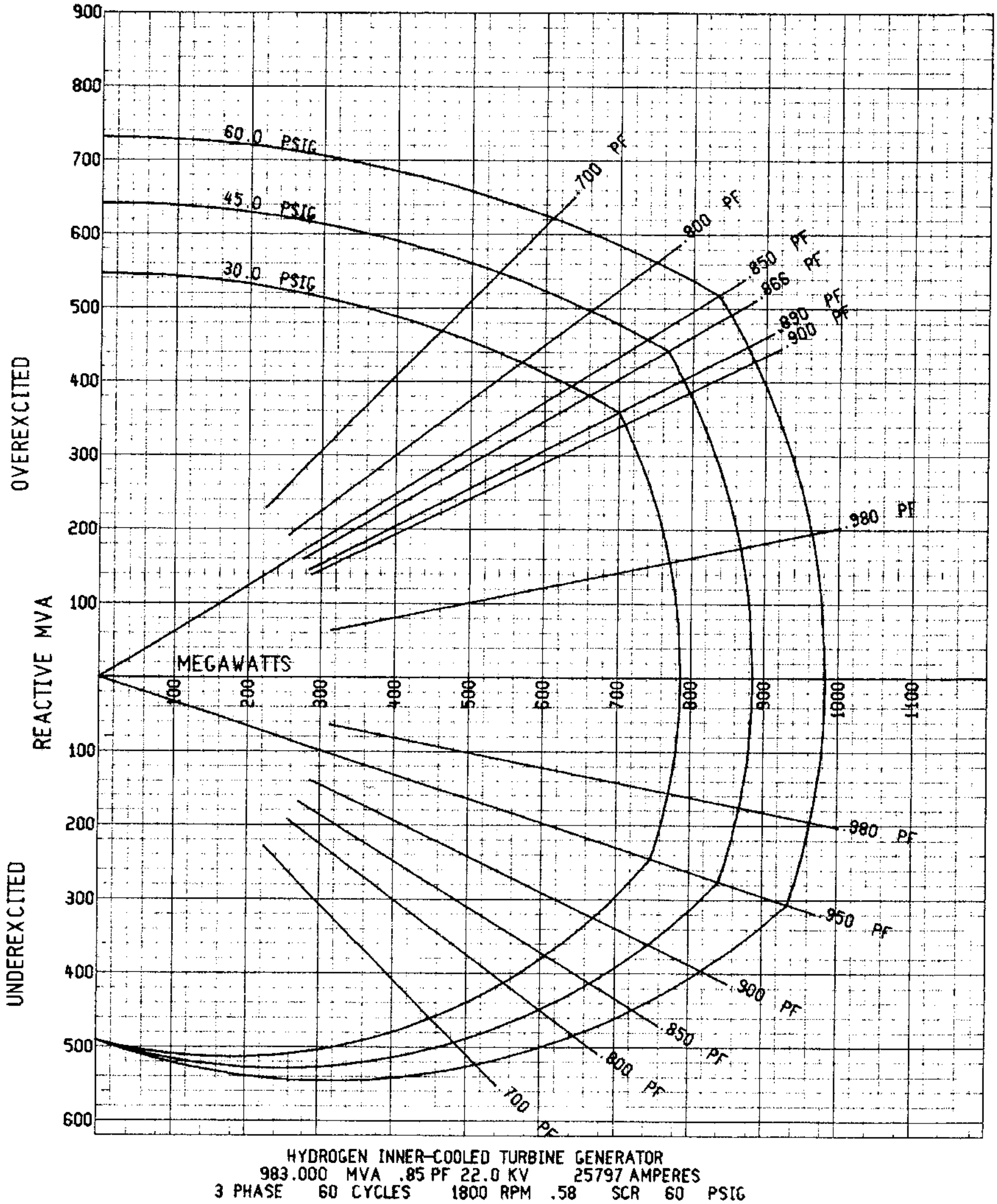
10CFR Part 55 Content:	55.43(b)(7)
Level of Difficulty:	2
SRO Only Justification:	

This question requires knowledge of Refuel floor SRO responsibilities.

PSA Applicability

N/A

REFERENCES



OVEREXCITED
REACTIVE MVA
UNDEREXCITED
MEGAWATTS
60.0 PSIG
45.0 PSIG
30.0 PSIG
.700 PF
.800 PF
.850 PF
.866 PF
.890 PF
.900 PF
.980 PF
.980 PF
.950 PF
.900 PF
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1100
HYDROGEN INNER-COOLED TURBINE GENERATOR
983.000 MVA .85 PF 22.0 KV 25797 AMPERES
3 PHASE 60 CYCLES 1800 RPM .58 SCR 60 PSIG

Figure 1

GENERAL EMERGENCY | SITE AREA EMERGENCY | ALERT | UNUSUAL EVENT

A – Abnorm. Rad Release / Rad Effluent

1 – Offsite Rad Conditions

AG1.1 (1 2 3 4 5 DEF)
Any valid gaseous monitor reading > Table A-1 column "GE" for ≥ 15 min. (Note 1)

AS1.1 (1 2 3 4 5 DEF)
Any valid gaseous monitor reading > Table A-1 column "SAE" for ≥ 15 min. (Note 1)

AA1.1 (1 2 3 4 5 DEF)
Any valid gaseous monitor reading > Table A-1 column "Alert" for ≥ 15 min. (Note 2)

AU1.1 (1 2 3 4 5 DEF)
Any valid gaseous monitor reading > Table A-1 column "UE" for ≥ 60 min. (Note 2)

AG1.2 (1 2 3 4 5 DEF)
Dose assessment using actual meteorology indicates doses > 1 Rem TEDE or > 5 Rem thyroid CDE at or beyond the site boundary

AS1.2 (1 2 3 4 5 DEF)
Dose assessment using actual meteorology indicates doses > 0.1 Rem TEDE or > 0.5 Rem thyroid CDE at or beyond the site boundary

AA1.2 (1 2 3 4 5 DEF)
Any valid liquid effluent monitor reading > Table A-1 column "Alert" for ≥ 15 min. (Note 2)

AU1.2 (1 2 3 4 5 DEF)
Any valid liquid effluent monitor reading > Table A-1 column "UE" for ≥ 60 min. (Note 2)

AG1.3 (1 2 3 4 5 DEF)
Field survey results indicate closed window dose rates > 1 Rem/hr expected to continue for ≥ 60 min. at or beyond the site boundary (Note 1)
OR
Analyses of field survey samples indicate thyroid CDE > 5 Rem for 1 hr of inhalation at or beyond the site boundary

AS1.3 (1 2 3 4 5 DEF)
Field survey indicates closed window dose rate > 0.1 Rem/hr that is expected to continue for ≥ 60 min. at or beyond the site boundary (Note 1)
OR
Field survey sample analysis indicates thyroid CDE > 0.5 Rem for 1 hr of inhalation at or beyond the site boundary

AA1.3 (1 2 3 4 5 DEF)
Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 200 x ODAM limits for ≥ 15 min. (Note 2)

AU1.3 (1 2 3 4 5 DEF)
Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 2 x ODAM limits for ≥ 60 min. (Note 2)

2 – Onsite Rad Conditions & Spent Fuel Pool Events

Table A-1 Effluent Monitor Classification Thresholds

	Monitor	GE for ≥ 15 min.	SAE for ≥ 15 min.	ALERT for ≥ 15 min.	UE for ≥ 60 min.
GASEOUS	ERP	3.50E+08 μCi/sec	3.50E+07 μCi/sec	2.80E+06 μCi/sec	2.24E+05 μCi/sec
GASEOUS	Rx Bldg Vent	3.50E+07 μCi/sec	3.50E+06 μCi/sec	5.45E+05 μCi/sec	8.48E+04 μCi/sec
GASEOUS	Turb Bldg Vent	3.50E+07 μCi/sec	3.50E+06 μCi/sec	5.62E+05 μCi/sec	9.02E+04 μCi/sec
GASEOUS	RW / ARW Bldg Vent	3.50E+07 μCi/sec	3.50E+06 μCi/sec	5.64E+05 μCi/sec	9.08E+04 μCi/sec
LIQUID	Rad Waste Effluent	-----	-----	200 x calculated alarm values*	2 x calculated alarm values*
LIQUID	Service Water Effluent	-----	-----	4.80E-04 μCi/cc	4.80E-06 μCi/cc

* with effluent discharge **not** isolated

AA2.1 (1 2 3 4 5 DEF)
Damage to irradiated fuel **OR** loss of water level (uncovering irradiated fuel outside the RPV) that causes **EITHER** of the following:
Valid RMA-RA-1 Fuel Pool Area Rad reading > 5.0E+04 mR/hr
OR
Valid RMP-RM-452 A-D Rx Bldg Vent Exhaust Plenum Hi-Hi alarm

AU2.1 (1 2 3 4 5 DEF)
Unplanned water level drop in the reactor cavity or spent fuel pool as indicated by **any** of the following:
- LI-86 (calibrated to 1001' elev.)
- Spent fuel pool low level alarm
- Visual observation

AND
Valid area radiation monitor reading rise on RMA-RA-1 or RMA-RA-2

AA2.2 (1 2 3 4 5 DEF)
A water level drop in the reactor refueling cavity or spent fuel pool that will result in irradiated fuel becoming uncovered

AU2.2 (1 2 3 4 5 DEF)
Unplanned valid area radiation monitor reading or survey results rise by a factor of 1,000 over normal levels*

* Normal levels can be considered as the highest reading in the past 24 hours excluding the current peak value

3 – MCR/CAS Rad

AA3.1 (1 2 3 4 5 DEF)
Dose rates > 15 mRem/hr in **EITHER** of the following areas requiring continuous occupancy to maintain plant safety functions:
Main Control Room (RM-RA-20)
OR
CAS

H – Hazards & Other Conditions Affecting Plant Safety

Table H-1 Safe Shutdown Areas
- Reactor Building
- Control Building
- Service Water Pump Room
- Diesel Generator Building
- Cable Expansion Room

1 – Natural or Destructive Phenomena

HA1.1 (1 2 3 4 5 DEF)
Seismic event > 0.1g as indicated by the Seismic Monitor System free field sensor(s) or Alarm B-3/A-1 EMERGENCY SEISMIC HIGH LEVEL
AND
Earthquake confirmed by **any** of the following:
- Earthquake felt in plant
- National Earthquake Information Center
- Control Room indication of degraded performance of systems required for the safe shutdown of the plant

HU1.1 (1 2 3 4 5 DEF)
Seismic event identified by any **two** of the following:
- The Seismic Monitor System free field sensor(s) actuated or Alarm B-3/B-1 SEISMIC EVENT
- Earthquake felt in plant
- National Earthquake Information Center

HA1.2 (1 2 3 4 5 DEF)
Tornado striking or high winds ≥ 100 mph resulting in **EITHER**:
Visible damage to **any** Table H-1 area structure containing safety systems or components
OR
Control Room indication of degraded performance of safety systems

HU1.2 (1 2 3 4 5 DEF)
Tornado striking within Protected Area boundary
OR
Sustained high winds ≥ 100 mph

HA1.3 (1 2 3 4 5 DEF)
Main turbine failure-generated projectiles result in **EITHER**:
Visible damage to or penetration of **any** Table H-1 area structure containing safety systems or components
OR
Control Room indication of degraded performance of safety systems

HU1.3 (1 2 3 4 5 DEF)
Main turbine failure resulting in casing penetration or damage to turbine or generator seals

HA1.4 (1 2 3 4 5 DEF)
Flooding in **any** Table H-1 area resulting in **EITHER**:
An electrical shock hazard that precludes access to operate or monitor safety equipment
OR
Control Room indication of degraded performance of safety systems

HU1.4 (1 2 3 4 5 DEF)
Flooding in **any** Table H-1 area that has the potential to affect safety-related equipment required by Technical Specifications for the current operating mode

HA1.5 (1 2 3 4 5 DEF)
High river/forebay water level > 902' MSL
OR
Low river/forebay level < 865' MSL

HU1.5 (1 2 3 4 5 DEF)
High river/forebay water level > 899' MSL
OR
Low river level/forebay < 870' MSL

HA1.6 (1 2 3 4 5 DEF)
Vehicle crash resulting in **EITHER**:
Visible damage to **any** Table H-1 area structure containing safety systems or components
OR
Control Room indication of degraded performance of safety systems

Notes
1. The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.
   If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values.
   (See EAL AS1.2/AG1.2.) Do **not** delay declaration awaiting dose assessment results.
2. The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.
3. The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.
4. Containment Closure is the action taken to secure primary or secondary containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment Closure requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety.
5. Manual scram methods for EAL SA2.1 and EAL SS2.1 are the following:
   - Reactor Scram push buttons
   - Reactor Mode switch in SHUTDOWN
   - Manual or auto actuation of ARI
6. See Table F-1, Fission Product Barrier Matrix, for possible escalation above the Unusual Event due to RCS Leakage.
7. If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should **not** be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.
8. The time required to establish the backfeed is likely longer than the specified time interval. If off-normal plant conditions have already established the backfeed, its power to the safety-related buses may be considered an offsite power source.

2 – Fire or Explosion

HA2.1 (1 2 3 4 5 DEF)
Fire or explosion resulting in **EITHER**:
Visible damage to **any** Table H-1 area containing safety systems or components
OR
Control Room indication of degraded performance of safety systems

HU2.1 (1 2 3 4 5 DEF)
Fire in **any** Table H-1 area **not** extinguished within 15 min. of Control Room notification or receipt of a valid Control Room alarm due to fire (Note 3)

HU2.2 (1 2 3 4 5 DEF)
Explosion within the Protected Area

3 – Hazardous Gas

HA3.1 (1 2 3 4 5 DEF)
Access to **any** Table H-1 area is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shut down the reactor (Note 7)

HU3.1 (1 2 3 4 5 DEF)
Toxic, corrosive, asphyxiant or flammable gases in amounts that have or could affect normal plant operations

HU3.2 (1 2 3 4 5 DEF)
Recommendation by local, county or state officials to evacuate or shelter site personnel based on an offsite event

4 – Security

HG4.1 (1 2 3 4 5 DEF)
A hostile action has occurred such that plant personnel are unable to operate equipment required to maintain safety functions
OR
A hostile action has caused failure of Spent Fuel Cooling Systems and imminent fuel damage is likely for a freshly off-loaded reactor core in pool

HS4.1 (1 2 3 4 5 DEF)
A hostile action is occurring or has occurred within the **Protected Area** as reported by the Security Shift Supervisor

HA4.1 (1 2 3 4 5 DEF)
A hostile action is occurring or has occurred within the **Owner Controlled Area** as reported by the Security Shift Supervisor
OR
A validated notification from NRC of an airliner attack threat within 30 min. of the site

HU4.1 (1 2 3 4 5 DEF)
A security condition that does **not** involve a hostile action as reported by the Security Shift Supervisor
OR
A credible site-specific security threat notification
OR
A validated notification from NRC providing information of an aircraft threat

5 – Control Room Evacuation

General Emergency: None

HS5.1 (1 2 3 4 5 DEF)
Control Room evacuation has been initiated
AND
Control of the plant **cannot** be established within 15 min.

HA5.1 (1 2 3 4 5 DEF)
Procedure 5.1ASD, Alternate Shutdown, or Procedure 5.4FIRE-S/D, Fire Induced Shutdown From Outside the Control Room, requires Control Room evacuation

Unusual Event: None

6 – Judgment

HG6.1 (1 2 3 4 5 DEF)
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve **EITHER**:
Actual or imminent substantial core degradation or melting with potential for loss of containment integrity
OR
Hostile action that results in an actual loss of physical control of the facility

Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels (1 Rem TEDE and 5 Rem thyroid CDE) beyond the site boundary

HS6.1 (1 2 3 4 5 DEF)
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve **EITHER**:
An actual or likely major failures of plant functions needed for protection of the public
OR
Hostile action that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public

Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels (1 Rem TEDE and 5 Rem thyroid CDE) beyond the site boundary

HA6.1 (1 2 3 4 5 DEF)
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve **EITHER**:
An actual or potential substantial degradation of the level of safety of the plant
OR
A security event that involves probable life threatening risk to site personnel or damage to site equipment because of hostile action

Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels beyond the site boundary

HU6.1 (1 2 3 4 5 DEF)
Other conditions exist which in the judgment of the Emergency Director indicate that **EITHER**:
Events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant
OR
A security threat to facility protection has been initiated

No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs

E – ISFSI

General Emergency: None
Site Area Emergency: None
Alert: None

EU1.1 (N/A)
Damage to a loaded cask confinement boundary

Modes: 1 Power Operation | 2 Startup | 3 Hot Shutdown | 4 Cold Shutdown | 5 Refueling | DEF Defueled

Cooper Nuclear Station
Emergency Action Level Matrix
EPIP 5.7.1 Attachment 4, Rev. 18

GENERAL EMERGENCY | SITE AREA EMERGENCY | ALERT | UNUSUAL EVENT

C – Cold SD/ Refuel System Malfunct.

1 – Loss of AC Power

General Emergency: None
Site Area Emergency: None

CA1.1 (4 5 DEF)
Loss of **all** offsite and **all** onsite AC power (Table C-4) to critical 4160V buses 1F and 1G for ≥ 15 min. (Note 3)

CU1.1 (4 5)
AC power capability to critical 4160V buses 1F and 1G reduced to a single power source (Table C-4) for ≥ 15 min. such that **any** additional single failure would result in loss of **all** AC power to critical buses (Note 3)

2 – RPV Level

CG2.1 (4 5)
RPV level < -158 in. for ≥ 30 min. (Note 3)
AND
Any Containment Challenge indication, Table C-5

CS2.1 (4 5)
With Containment Closure **not** established, RPV level < -48 in. (Note 4)

CA2.1 (4 5)
RPV level < -42 in.
OR
RPV level **cannot** be monitored for ≥ 15 min. (Note 3) with **any** unexplained RPV leakage indication, Table C-1

CU2.1 (4)
RPV level **cannot** be restored and maintained > +3 in. for ≥ 15 min. (Note 3) due to RCS leakage

CG2.2 (4 5)
RPV level **cannot** be monitored for ≥ 30 min. (Note 3) with core uncovery indicated by
EITHER:
Unexplained RPV leakage indication, Table C-1
OR
Erratic Source Range Monitor indication
AND
Any Containment Challenge indication, Table C-5

CS2.2 (4 5)
With Containment Closure established, RPV level < -158 in. (Note 4)

CS2.3 (4 5)
RPV level **cannot** be monitored for ≥ 30 min. (Note 3) with a loss of inventory as indicated by **EITHER**:
Unexplained RPV leakage indication, Table C-1
OR
Erratic Source Range Monitor indication

CU2.2 (5)
Unplanned RPV level drop for ≥ 15 min (Note 3) below **EITHER**:
RPV flange (LI-86: 206 in. normal calibration, 113.75 in. elevated calibration)
OR
RPV level band when the RPV level band is established below the RPV flange

CU2.3 (5)
RPV level **cannot** be monitored with **any** unexplained RPV leakage indication, Table C-1

3 – RCS Temp.

General Emergency: None
Site Area Emergency: None

CA3.1 (4 5)
Any unplanned event results in **EITHER**:
RCS temperature > 212°F for > Table C-3 duration (Note 4)
OR
RPV pressure increase > 10 psig due to loss of RCS cooling

CU3.1 (4 5)
Any unplanned event results in RCS temperature > 212°F due to loss of decay heat removal capability

CU3.2 (4 5)
Loss of **all** RCS temperature and RPV level indication for ≥ 15 min. (Note 3)

4 – Comm.

General Emergency: None
Site Area Emergency: None
Alert: None

CU4.1 (4 5 DEF)
Loss of **all** Table C-2 onsite (internal) communication methods affecting the ability to perform routine operations
OR
Loss of **all** Table C-2 offsite (external) communication methods affecting the ability to perform offsite notifications

5 – Inadvertent Criticality

General Emergency: None
Site Area Emergency: None
Alert: None

CU5.1 (4 5)
An unplanned sustained positive period observed on nuclear instrumentation

6 – Loss of DC Power

General Emergency: None
Site Area Emergency: None
Alert: None

CU6.1 (4 5)
< 105 VDC bus voltage indications on **all** Technical Specification required 125 VDC buses for ≥ 15 min. (Note 3)

Table C-1 RPV Leakage Indications
- Drywell equipment drain sump level rise
- Drywell floor drain sump level rise
- Reactor Building equipment drain sump level rise
- Reactor Building floor drain sump level rise
- Suppression Pool water level rise
- RPV make-up rate rise
- Observation of unisolable RCS leakage

Table C-2 Communications Systems

System	Onsite (internal)	Offsite (external)
Station Intercom System "Gaitronics"	X	
Site UHF Radio Consoles	X	
Radio Paging System	X	
Alternate Intercom	X	
CNS On-Site Cell Phone System	X	X
Telephone system (PBX)	X	X
Federal Telecommunications System (FTS 2001)		X
Local Telephones (C.O. Lines)		X
CNS State Notification Telephones		X
Satellite Telephones		X

Table C-3 RCS Reheat Duration Thresholds

* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is **not** applicable

1. RCS intact (Containment Closure N/A)	60 min.*
2. Containment Closure established **AND** RCS **not** intact	20 min.*
3. Containment Closure **not** established **AND** RCS **not** intact	0 min.

Table C-4 AC Power Sources

Offsite
- Startup Station Service Transformer
- Emergency Station Service Transformer
- Backfeed 345 kv line through Main Power Transformer to the Normal Station Service Transformer (Note 8)

Onsite
- DG-1
- DG-2

Table C-5 Containment Challenge Indications
- Containment Closure **not** established (Note 4)
- Deflagration concentrations exist inside PC
  - ≥ 6% H_2 in drywell or torus
    AND
  - ≥ 5% O_2 in drywell or torus
- Unplanned rise in PC pressure
- Secondary Containment area radiation > 1000 mR/hr (EOP-5A Table 10)

EAL Identifier
XXX.X
Category (A, H, S, F, C, E)
Emergency classification (G, S, A, U)
Subcategory number (1 if no subcategory)
Sequential number within subcategory/classification

MODE 4, 5 or DEF

Cooper Nuclear Station Emergency Action Level Matrix

Columns: GENERAL EMERGENCY | SITE AREA EMERGENCY | ALERT | UNUSUAL EVENT

A – Abnorm. Rad Release / Rad Effluent

1 – Offsite Rad Conditions

AG1.1 (Modes 1 2 3 4 5 DEF)
Any valid gaseous monitor reading > Table A-1 column "GE" for ≥ 15 min. (Note 1)

AG1.2 (Modes 1 2 3 4 5 DEF)
Dose assessment using actual meteorology indicates doses > 1 Rem TEDE or > 5 Rem thyroid CDE at or beyond the site boundary

AG1.3 (Modes 1 2 3 4 5 DEF)
Field survey results indicate closed window dose rates > 1 Rem/hr expected to continue for ≥ 60 min. at or beyond the site boundary (Note 1)
OR
Analyses of field survey samples indicate thyroid CDE > 5 Rem for 1 hr of inhalation at or beyond the site boundary

AS1.1 (Modes 1 2 3 4 5 DEF)
Any valid gaseous monitor reading > Table A-1 column "SAE" for ≥ 15 min. (Note 1)

AS1.2 (Modes 1 2 3 4 5 DEF)
Dose assessment using actual meteorology indicates doses > 0.1 Rem TEDE or > 0.5 Rem thyroid CDE at or beyond the site boundary

AS1.3 (Modes 1 2 3 4 5 DEF)
Field survey results indicate closed window dose rate > 0.1 Rem/hr that is expected to continue for ≥ 60 min. at or beyond the site boundary (Note 1)
OR
Field survey sample analysis indicates thyroid CDE > 0.5 Rem for 1 hr of inhalation at or beyond the site boundary

AA1.1 (Modes 1 2 3 4 5 DEF)
Any valid gaseous monitor reading > Table A-1 column "Alert" for ≥ 15 min. (Note 2)

AA1.2 (Modes 1 2 3 4 5 DEF)
Any valid liquid effluent monitor reading > Table A-1 column "Alert" for ≥ 15 min. (Note 2)

AA1.3 (Modes 1 2 3 4 5 DEF)
Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 200 x ODAM limits for ≥ 15 min. (Note 2)

AU1.1 (Modes 1 2 3 4 5 DEF)
Any valid gaseous monitor reading > Table A-1 column "UE" for ≥ 60 min. (Note 2)

AU1.2 (Modes 1 2 3 4 5 DEF)
Any valid liquid effluent monitor reading > Table A-1 column "UE" for ≥ 60 min. (Note 2)

AU1.3 (Modes 1 2 3 4 5 DEF)
Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 2 x ODAM limits for ≥ 60 min. (Note 2)

2 – Onsite Rad Conditions & Spent Fuel Pool Events / 3 – MCR/CAS Rad

Table A-1 Effluent Monitor Classification Thresholds

	Monitor	GE for ≥ 15 min.	SAE for ≥ 15 min.	ALERT for ≥ 15 min.	UE for ≥ 60 min.
GASEOUS	ERP	3.50E+08 µCi/sec	3.50E+07 µCi/sec	2.80E+06 µCi/sec	2.24E+05 µCi/sec
GASEOUS	Rx Bldg Vent	3.50E+07 µCi/sec	3.50E+06 µCi/sec	5.45E+05 µCi/sec	8.48E+04 µCi/sec
GASEOUS	Turb Bldg Vent	3.50E+07 µCi/sec	3.50E+06 µCi/sec	5.62E+05 µCi/sec	9.02E+04 µCi/sec
GASEOUS	RW / ARW Bldg Vent	3.50E+07 µCi/sec	3.50E+06 µCi/sec	5.64E+05 µCi/sec	9.08E+04 µCi/sec
LIQUID	Rad Waste Effluent	-----	-----	200 x calculated alarm values*	2 x calculated alarm values*
LIQUID	Service Water Effluent	-----	-----	4.80E-04 µCi/cc	4.80E-06 µCi/cc

* with effluent discharge **not** isolated

AA2.1 (Modes 1 2 3 4 5 DEF)
Damage to irradiated fuel **OR** loss of water level (uncovering irradiated fuel outside the RPV) that causes **EITHER** of the following:
Valid RMA-RA-1 Fuel Pool Area Rad reading > 5.0E+04 mR/hr
OR
Valid RMP-RM-452 A-D Rx Bldg Vent Exhaust Plenum Hi-Hi alarm

AA2.2 (Modes 1 2 3 4 5 DEF)
A water level drop in the reactor refueling cavity or spent fuel pool that will result in irradiated fuel becoming uncovered

AU2.1 (Modes 1 2 3 4 5 DEF)
Unplanned water level drop in the reactor cavity or spent fuel pool as indicated by **any** of the following:
- LI-86 (calibrated to 1001' elev.)
- Spent fuel pool low level alarm
- Visual observation

AND
Valid area radiation monitor reading rise on RMA-RA-1 or RMA-RA-2

AU2.2 (Modes 1 2 3 4 5 DEF)
Unplanned valid area radiation monitor reading or survey results rise by a factor of 1,000 over normal levels*

* Normal levels can be considered as the highest reading in the past 24 hours excluding the current peak value

AA3.1 (Modes 1 2 3 4 5 DEF)
Dose rates > 15 mRem/hr in **EITHER** of the following areas requiring continuous occupancy to maintain plant safety functions:
Main Control Room (RM-RA-20)
OR
CAS

H – Hazards & Other Conditions Affecting Plant Safety

1 – Natural or Destructive Phenomena

Table H-1 Safe Shutdown Areas
- Reactor Building
- Control Building
- Service Water Pump Room
- Diesel Generator Building
- Cable Expansion Room

HA1.1 (Modes 1 2 3 4 5 DEF)
Seismic event > 0.1g as indicated by the Seismic Monitor System free field sensor(s) or Alarm B-3/A-1 EMERGENCY SEISMIC HIGH LEVEL
AND
Earthquake confirmed by **any** of the following:
- Earthquake felt in plant
- National Earthquake Information Center
- Control Room indication of degraded performance of systems required for the safe shutdown of the plant

HA1.2 (Modes 1 2 3 4 5 DEF)
Tornado striking or high winds ≥ 100 mph resulting in **EITHER**:
Visible damage to **any** Table H-1 area structure containing safety systems or components
OR
Control Room indication of degraded performance of safety systems

HA1.3 (Modes 1 2 3 4 5 DEF)
Main turbine failure-generated projectiles result in **EITHER**:
Visible damage to or penetration of **any** Table H-1 area structure containing safety systems or components
OR
Control Room indication of degraded performance of safety systems

HA1.4 (Modes 1 2 3 4 5 DEF)
Flooding in **any** Table H-1 area resulting in **EITHER**:
An electrical shock hazard that precludes access to operate or monitor safety equipment
OR
Control Room indication of degraded performance of safety systems

HA1.5 (Modes 1 2 3 4 5 DEF)
High river/forebay water level > 902' MSL
OR
Low river/forebay level < 865' MSL

HA1.6 (Modes 1 2 3 4 5 DEF)
Vehicle crash resulting in **EITHER**:
Visible damage to **any** Table H-1 area structure containing safety systems or components
OR
Control Room indication of degraded performance of safety systems

HU1.1 (Modes 1 2 3 4 5 DEF)
Seismic event identified by any **two** of the following:
- Seismic Monitor System free field sensor(s) actuated or Alarm B-3/B-1 SEISMIC EVENT
- Earthquake felt in plant
- National Earthquake Information Center

HU1.2 (Modes 1 2 3 4 5 DEF)
Tornado striking within Protected Area boundary
OR
Sustained high winds ≥ 100 mph

HU1.3 (Modes 1 2 3 4 5 DEF)
Main turbine failure resulting in casing penetration or damage to turbine or generator seals

HU1.4 (Modes 1 2 3 4 5 DEF)
Flooding in **any** Table H-1 area that has the potential to affect safety-related equipment required by Technical Specifications for the current operating mode

HU1.5 (Modes 1 2 3 4 5 DEF)
High river/forebay water level > 899' MSL
OR
Low river level/forebay < 870' MSL

Notes
1. The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values. (See EAL AS1.2/AG1.2.) Do **not** delay declaration awaiting dose assessment results.
2. The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.
3. The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.
4. Containment Closure is the action taken to secure primary or secondary containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment Closure requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety.
5. Manual scram methods for EAL SA2.1 and EAL SS2.1 are the following:
   - Reactor Scram push buttons
   - Reactor Mode switch in SHUTDOWN
   - Manual or auto actuation of ARI
6. See Table F-1, Fission Product Barrier Matrix, for possible escalation above the Unusual Event due to RCS Leakage.
7. If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should **not** be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.
8. The time required to establish the backfeed is likely longer than the specified time interval. If off-normal plant conditions have already established the backfeed, its power to the safety-related buses may be considered an offsite power source.

2 – Fire or Explosion

HA2.1 (Modes 1 2 3 4 5 DEF)
Fire or explosion resulting in **EITHER**:
Visible damage to **any** Table H-1 area containing safety systems or components
OR
Control Room indication of degraded performance of safety systems

HU2.1 (Modes 1 2 3 4 5 DEF)
Fire in **any** Table H-1 area **not** extinguished within 15 min. of Control Room notification or receipt of a valid Control Room alarm due to fire (Note 3)

HU2.2 (Modes 1 2 3 4 5 DEF)
Explosion within the Protected Area

3 – Hazardous Gas

HA3.1 (Modes 1 2 3 4 5 DEF)
Access to **any** Table H-1 area is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shut down the reactor (Note 7)

HU3.1 (Modes 1 2 3 4 5 DEF)
Toxic, corrosive, asphyxiant or flammable gases in amounts that have or could affect normal plant operations

HU3.2 (Modes 1 2 3 4 5 DEF)
Recommendation by local, county or state officials to evacuate or shelter site personnel based on an offsite event

4 – Security

HG4.1 (Modes 1 2 3 4 5 DEF)
A hostile action has occurred such that plant personnel are unable to operate equipment required to maintain safety functions
OR
A hostile action has caused failure of Spent Fuel Cooling Systems and imminent fuel damage is likely for a freshly off-loaded reactor core in pool

HS4.1 (Modes 1 2 3 4 5 DEF)
A hostile action is occurring or has occurred within the **Protected Area** as reported by the Security Shift Supervisor

HA4.1 (Modes 1 2 3 4 5 DEF)
A hostile action is occurring or has occurred within the **Owner Controlled Area** as reported by the Security Shift Supervisor
OR
A validated notification from NRC of an airliner attack threat within 30 min. of the site

HU4.1 (Modes 1 2 3 4 5 DEF)
A security condition that does **not** involve a hostile action as reported by the Security Shift Supervisor
OR
A credible site-specific security threat notification
OR
A validated notification from NRC providing information of an aircraft threat

5 – Control Room Evacuation

General Emergency: None

HS5.1 (Modes 1 2 3 4 5 DEF)
Control Room evacuation has been initiated
AND
Control of the plant **cannot** be established within 15 min.

HA5.1 (Modes 1 2 3 4 5 DEF)
Procedure 5.1ASD, Alternate Shutdown, or Procedure 5.4FIRE-S/D, Fire Induced Shutdown From Outside the Control Room, requires Control Room evacuation

Unusual Event: None

6 – Judgment

HG6.1 (Modes 1 2 3 4 5 DEF)
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve **EITHER**:
Actual or imminent substantial core degradation or melting with potential for loss of containment integrity
OR
Hostile action that results in an actual loss of physical control of the facility

Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels (1 Rem TEDE and 5 Rem thyroid CDE) beyond the site boundary

HS6.1 (Modes 1 2 3 4 5 DEF)
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve **EITHER**:
An actual or likely major failures of plant functions needed for protection of the public
OR
Hostile action that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to equipment needed for the protection of the public

Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels (1 Rem TEDE and 5 Rem thyroid CDE) beyond the site boundary

HA6.1 (Modes 1 2 3 4 5 DEF)
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve **EITHER**:
An actual or potential substantial degradation of the level of safety of the plant
OR
A security event that involves probable life threatening risk to site personnel or damage to site equipment because of hostile action

Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels beyond the site boundary

HU6.1 (Modes 1 2 3 4 5 DEF)
Other conditions exist which in the judgment of the Emergency Director indicate that **EITHER**:
Events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant
OR
A security threat to facility protection has been initiated

No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs

E – ISFSI

General Emergency: None | Site Area Emergency: None | Alert: None

EU1.1 (N/A)
Damage to a loaded cask confinement boundary

S – System Malfunct.

1 – Loss of Power

SG1.1 (Modes 1 2 3)
Loss of **all** offsite and **all** onsite AC power (Table S-3) to critical 4160V buses 1F and 1G
AND EITHER:
Restoration of at least one emergency bus in < 4 hours is **not** likely
OR
RPV level **cannot** be restored and maintained > -158 in. or **cannot** be determined

SS1.1 (Modes 1 2 3)
Loss of **all** offsite and **all** onsite AC power (Table S-3) to critical 4160V buses 1F and 1G for ≥ 15 min. (Note 3)

SA1.1 (Modes 1 2 3)
AC power capability to critical 4160V buses 1F and 1G reduced to a single power source (Table S-3) for ≥ 15 min. such that **any** additional single failure would result in loss of **all** AC power to critical buses (Note 3)

SU1.1 (Modes 1 2 3)
Loss of **all** offsite AC power (Table S-3) to critical 4160V buses 1F and 1G for ≥ 15 min. (Note 3)

2 – ATWS Criticality

SG2.1 (Modes 1 2)
Automatic and **all** manual scrams were **not** successful
AND
Reactor power ≥ 3%
AND EITHER of the following exist or have occurred due to continued power generation:
RPV level **cannot** be restored and maintained > -183 in. or **cannot** be determined
OR
Average torus water temperature and RPV pressure **cannot** be maintained within the Heat Capacity Temperature Limit (EOP/SAG Graph 7)

SS2.1 (Modes 1 2)
An automatic scram failed to shut down the reactor
AND
Manual actions taken at the reactor control console (Note 5) do **not** shut down the reactor as indicated by reactor power ≥ 3%

SA2.1 (Modes 1 2)
An automatic scram failed to shut down the reactor
AND
Manual actions taken at the reactor control console (Note 5) successfully shut down the reactor as indicated by reactor power < 3%

SU2.1 (Modes 3)
An unplanned sustained positive period observed on nuclear instrumentation

3 – Inability to Reach Shutdown Conditions

General Emergency: None | Site Area Emergency: None | Alert: None

SU3.1 (Modes 1 2 3)
Plant is **not** brought to required operating mode within Technical Specifications LCO action statement time

4 – Inst.

General Emergency: None

SS4.1 (Modes 1 2 3)
Loss of > approximately 75% of annunciators or indicators associated with safety systems on Control Room Panels 9-3, 9-4, 9-5, and C for ≥ 15 min. (Note 3)
AND
Any significant transient is in progress, Table S-1
AND
Compensatory indications are unavailable

SA4.1 (Modes 1 2 3)
Unplanned loss of > approximately 75% of annunciators or indicators associated with safety systems on Control Room Panels 9-3, 9-4, 9-5, and C for ≥ 15 min. (Note 3)
AND EITHER:
Any significant transient is in progress, Table S-1
OR
Compensatory indications are unavailable

SU4.1 (Modes 1 2 3)
Unplanned loss of > approximately 75% of annunciators or indicators associated with safety systems on Control Room Panels 9-3, 9-4, 9-5, and C for ≥ 15 min. (Note 3)

5 – Fuel Clad Degradation

General Emergency: None | Site Area Emergency: None | Alert: None

SU5.1 (Modes 1 2 3)
SJAE monitor > 1.58E+3 mR/hr

SU5.2 (Modes 1 2 3)
Coolant activity ≥ 4.0 µCi/gm dose equivalent I-131

6 – RCS Leakage

General Emergency: None | Site Area Emergency: None | Alert: None

SU6.1 (Modes 1 2 3)
Unidentified or pressure boundary leakage > 10 gpm
OR
Identified leakage > 30 gpm
(Note 6)

7 – Loss of DC Power

General Emergency: None

SS7.1 (Modes 1 2 3)
< 105 VDC bus voltage indications on **all** vital 125 VDC buses (1A and 1B) for ≥ 15 min. (Note 3)

Alert: None | Unusual Event: None

8 – Comm.

General Emergency: None | Site Area Emergency: None | Alert: None

SU8.1 (Modes 1 2 3)
Loss of **all** Table S-2 onsite (internal) communication capability affecting the ability to perform routine operations
OR
Loss of **all** Table S-2 offsite (external) communication methods affecting the ability to perform offsite notifications

F – Fission Product Barriers

FG1.1 (Modes 1 2 3)
Loss of **any** two barriers
AND
Loss or potential loss of third barrier (Table F-1)

FS1.1 (Modes 1 2 3)
Loss or potential loss of **any** two barriers (Table F-1)

FA1.1 (Modes 1 2 3)
Any loss or **any** potential loss of either **Fuel Clad** or **RCS** (Table F-1)

FU1.1 (Modes 1 2 3)
Any loss or **any** potential loss of **Primary Containment** (Table F-1)

Table F-1 Fission Product Barrier Matrix©[4]

	Fuel Clad Barrier – Loss	Fuel Clad Barrier – Potential Loss	Reactor Coolant System Barrier – Loss	Reactor Coolant System Barrier – Potential Loss	Primary Containment Barrier – Loss	Primary Containment Barrier – Potential Loss
A. RPV Level	1. SAG 1 entry is required due to **any** of the following: • Non-Failure-to-Scram: o RPV water level **cannot** be restored and maintained > -183 in., or o RPV water level **cannot** be restored and maintained ≥ -209 in. and **no** core spray subsystem flow can be restored and maintained ≥ 4,750 gpm • Failure-to-Scram: o RPV water level **cannot** be restored and maintained > -183 in. and core steam flow **cannot** be restored and maintained > 800,000 lbm/hr • Core damage is occurring due to loss of core cooling	8. RPV level **cannot** be restored and maintained > -158 in. or **cannot** be determined	10. RPV level **cannot** be restored and maintained > -158 in. or **cannot** be determined	None	None	25. SAG 1 entry is required
B. PC Pressure / Temperature	None	None	11. PC pressure > 1.84 psig due to RCS leakage	None	19. PC pressure rise followed by a rapid unexplained drop in PC pressure 20. PC pressure response **not** consistent with LOCA conditions	26. PC pressure > 56 psig and rising 27. Deflagration concentrations exist inside PC ≥ 6% H_2 in drywell or torus (or **cannot** be determined) **AND** ≥ 5% O_2 in drywell or torus (or **cannot** be determined) 28. Average torus water temperature and RPV pressure **cannot** be maintained within the Heat Capacity Temperature Limit (EOP/SAG Graph 7)
C. Isolation	None	None	12. Release pathway exists outside primary containment resulting from isolation failure in **any** of the following (excluding normal process system flowpaths from an unisolable system): • Main steam line • HPCI steam line • RCIC steam line • RWCU • Feedwater	16. RCS leakage > 50 gpm inside the drywell 17. Unisolable primary system discharge outside primary containment as indicated by exceeding **any** secondary containment Maximum Normal Operating temperature or radiation value (EOP-5A Tables 9 and 10)	21. Failure of **all** valves in **any** one line to close **AND** Direct downstream pathway to the environment exists after PC isolation signal 22. Intentional PC venting per EOPs 23. Unisolable primary system discharge outside PC as indicated by exceeding **any** secondary containment Maximum Safe Operating temperature or radiation value (EOP-5A Tables 9 and 10)	None
D. ERD	None	None	13. Emergency RPV depressurization is required	None	None	None
E. Rad	2. Drywell radiation monitor (RMA-RM-40A/B) > 2.50E+03 Rem/hr 3. Primary coolant activity > 300 µCi/gm dose equivalent I-131 4. Main Steam Line Radiation Monitor Readings ≥ Hi-Hi Alarm Setpoint 5. ≥ 1.5E4 mrem/hr on SJAE monitor 6. Non-LOCA with DW Rad Monitor reading > 115 REM/hr	None	14. Drywell radiation monitor (RMA-RM-40A/B) > 2.40E+02 Rem/hr - LOCA	None	None	29. Drywell radiation monitor (RMA-RM-40A/B) > 5.00E+04 Rem/hr
F. Judgment	7. **Any** condition in the opinion of the Emergency Director that indicates loss of the Fuel Clad barrier.	9. **Any** condition in the opinion of the Emergency Director that indicates potential loss of the Fuel Clad barrier	15. **Any** condition in the opinion of the Emergency Director that indicates loss of the RCS barrier	18. **Any** condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier	24. **Any** condition in the opinion of the Emergency Director that indicates loss of the PC barrier	30. **Any** condition in the opinion of the Emergency Director that indicates potential loss of the PC barrier

Table S-1 Significant Transients
- Reactor scram
- Runback > 25% thermal power
- Electrical load rejection > 25% full electrical load
- ECCS injection
- Thermal power oscillations > 10%

Table S-2 Communications Systems

System	Onsite (internal)	Offsite (external)
Station Intercom System "Gaitronics"	X	
Site UHF Radio Consoles	X	
Radio Paging System	X	
Alternate Intercom	X	
CNS On-Site Cell Phone System	X	X
Telephone system (PBX)	X	X
Federal Telecommunications System (FTS 2001)		X
Local Telephones (C.O. Lines)		X
CNS State Notification Telephones		X
Satellite Telephones		X

Table S-3 AC Power Sources

Offsite
- Startup Station Service Transformer
- Emergency Station Service Transformer
- Backfeed 345 kv line through Main Power Transformer to the Normal Station Service Transformer (Note 8)

Onsite
- DG-1
- DG-2
- Main Generator

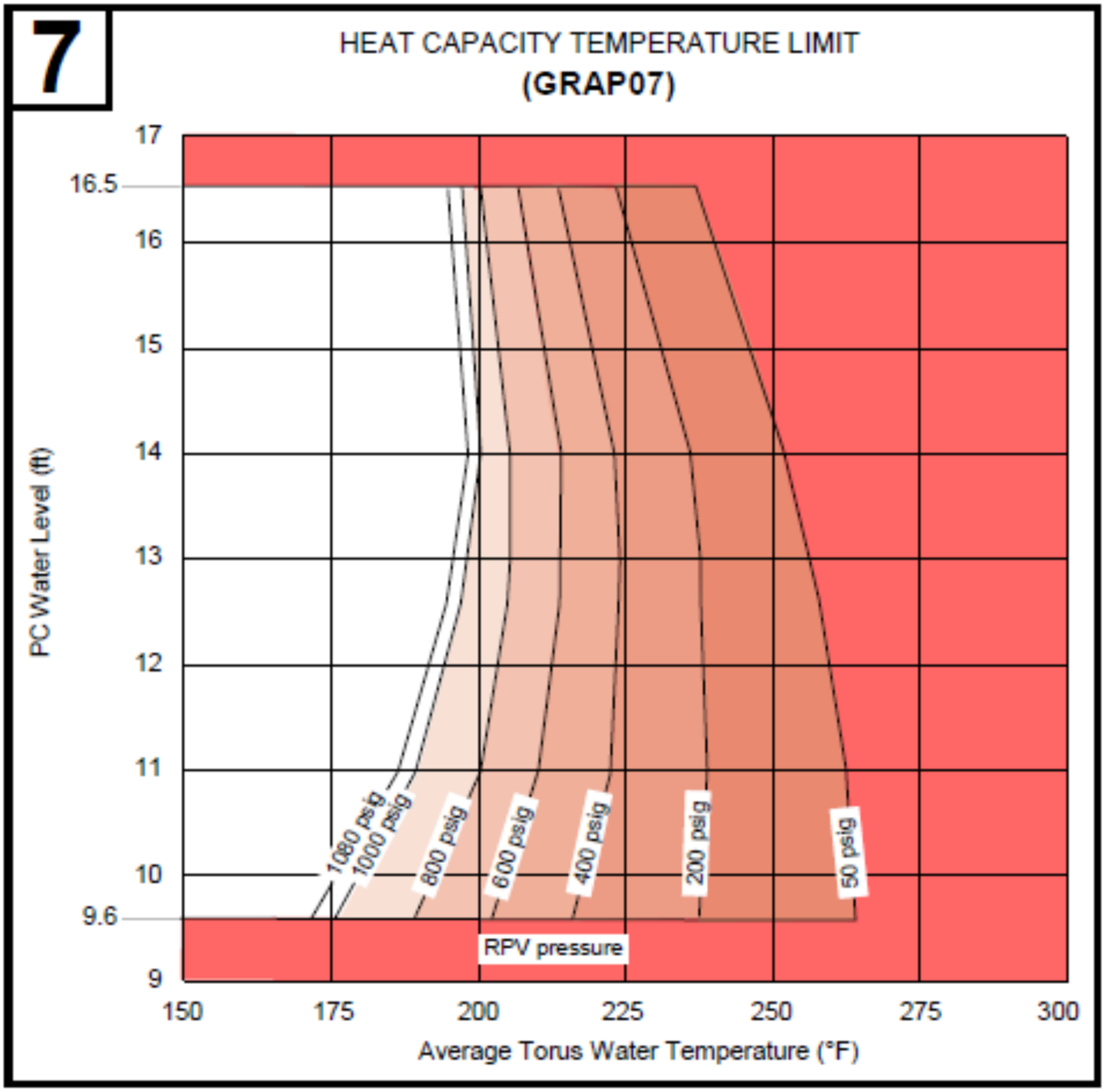
EAL Identifier

XXX.X
- Category (A, H, S, F, C, E)
- Emergency classification (G, S, A, U)
- Subcategory number (1 if no subcategory)
- Sequential number within subcategory/classification

Modes: 1 Power Operation | 2 Startup | 3 Hot Shutdown | 4 Cold Shutdown | 5 Refueling | DEF Defueled

Cooper Nuclear Station
Emergency Action Level Matrix
EPIP 5.7.1 Attachment 4, Rev. 18

MODE 1, 2 or 3



7
HEAT CAPACITY TEMPERATURE LIMIT
(GRAP07)
17
16.5
16
15
14
13
12
11
10
9.6
9
PC Water Level (ft)
1080 psig
1000 psig
800 psig
600 psig
400 psig
200 psig
50 psig
RPV pressure
150
175
200
225
250
275
300
Average Torus Water Temperature (°F)

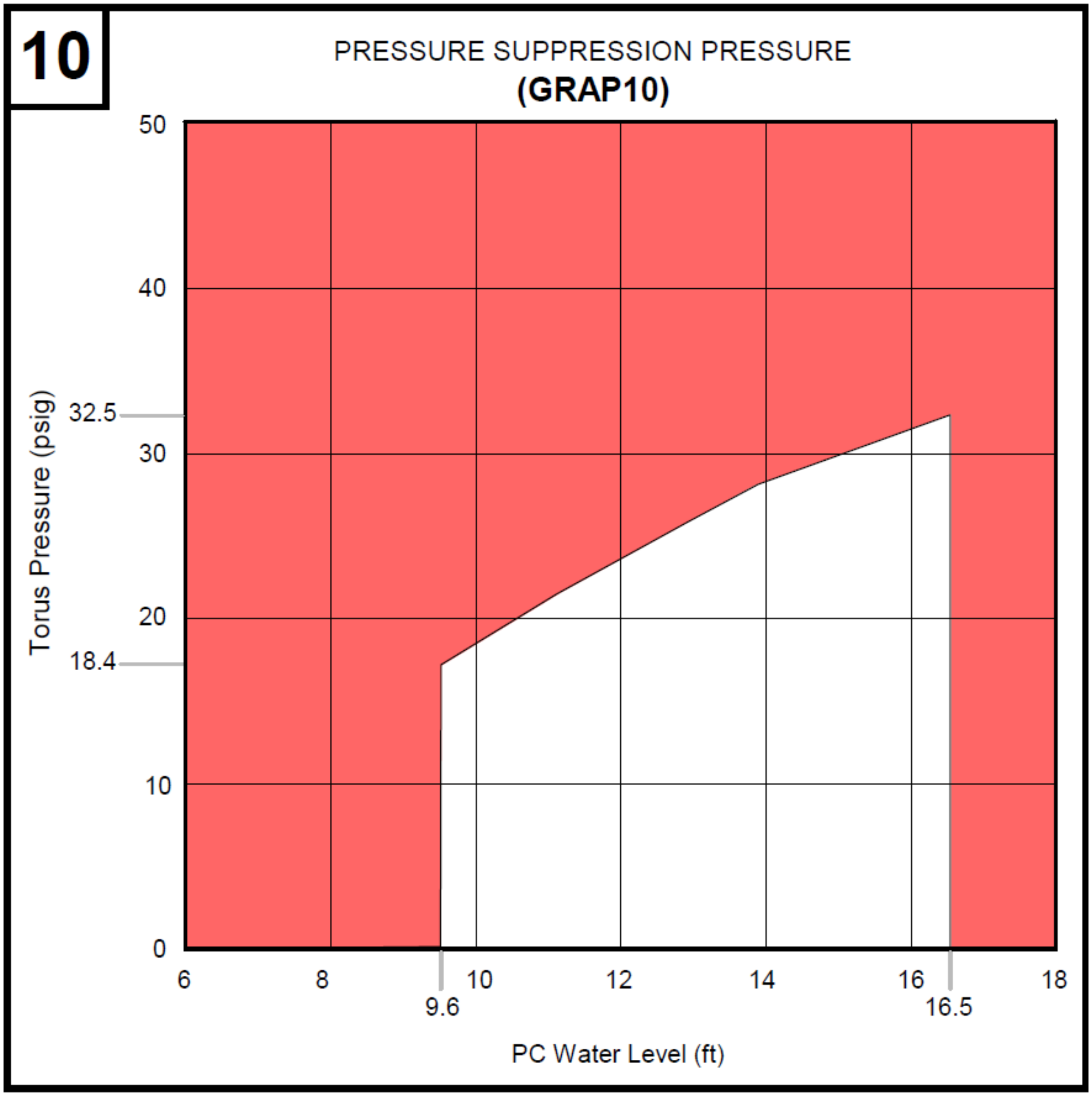

10
PRESSURE SUPPRESSION PRESSURE
(GRAP10)
Torus Pressure (psig)
50
40
32.5
30
20
18.4
10
0
6
8
9.6
10
12
14
16
16.5
18
PC Water Level (ft)

Table 3.3.3.1-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

FUNCTION	REQUIRED CHANNELS	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1
1. Reactor Pressure	2	E
2. Reactor Vessel Water Level		
a. Fuel Zone	2	E
b. Wide Range	2	E
c. Steam Nozzle	1	F
3. Suppression Pool Level (Wide Range)	2	E
4. Primary Containment Gross Radiation Monitors	2	F
5. PCIV Position	2 per penetration flow path[a][b]	E
6. Primary Containment H_2 & O_2 Analyzer	2	E
7. Primary Containment Pressure		
a. Drywell Narrow Range	2	E
b. Drywell Wide Range	2	E
c. Suppression Chamber Wide Range	2	E
8. Suppression Pool Water Temperature	2[c]	E

(a) Not required for isolation valves whose associated penetration flow path is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

(b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.

(c) A channel requires a minimum of four resistance temperature detectors (RTDs) to be OPERABLE with no two adjacent RTDs inoperable.

3.3 INSTRUMENTATION

3.3.1.2 Source Range Monitor (SRM) Instrumentation

LCO 3.3.1.2 The SRM instrumentation in Table 3.3.1.2-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.2-1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required SRMs inoperable in MODE 2 with intermediate range monitors (IRMs) on Range 2 or below.	A.1 Restore required SRMs to OPERABLE status.	4 hours
B. Three required SRMs inoperable in MODE 2 with IRMs on Range 2 or below.	B.1 Suspend control rod withdrawal.	Immediately
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One or more required SRMs inoperable in MODE 3 or 4.	D.1	Fully insert all insertable control rods.	1 hour
	AND		
	D.2	Place reactor mode switch in the shutdown position.	1 hour
E. One or more required SRMs inoperable in MODE 5.	E.1	Suspend CORE ALTERATIONS except for control rod insertion.	Immediately
	AND		
	E.2	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately