Niagara Mohawk

October 21, 1999

NMP-95715

Regional Administrator
USNRC Region I
475 Allendale Road
King of Prussia, PA 19406

Subject: Draft Examination Submittal for Nine Mile Point Unit 2 Initial Operator Examinations

Dear Mr. Miller,

In response to the NRC Corporate Notification letter dated July 30, 1999 and NUREG 1021 Revision 8, Niagara Mohawk Power Corporation is required to submit the draft examination materials by October 22, 1999.

Enclosed are the draft written examinations for RO and SRO tests, operating tests, and associated quality checklists. Also enclosed are the requested reference materials.

Please withhold these examination materials from public disclosure until after the examinations have been completed.

If you have any questions regarding the submittal, please contact Mr. Jerry Bobka at 315-349-2569.

Sincerely,

L E Pisano

Louis E. Pisano
Manager Training – Nuclear

/cld

Enc.

xc: Herb Williams

RO #	written	level 1	level 2	level 3	a	b	c	d	dir	mod	new	ro only	comments
1	1		1			1					1		
2	1		1					1			1		
3	1		1					1			1		
4	1	1				1					1		
5	1		1				1				1		
6	1		1			1					1		
7	1		1					1			1		
8	1		1				1			1			
9	1		1			1					1		
10	1	1					1				1		
11	1		1					1	1				
12	1			1	1						1		
13	1		1				1				1		
14	1	1				1				1			
15	1		1		1						1		
16	1		1		1				1				
17	1		1		1						1		
18	1			1				1			1		
19	1	1				1			1				
20	1		1		1						1		
21	1	1			1						1	1	
22	1	1						1			1	1	
23	1	1			1						1		
24	1		1					1			1		
25	1	1				1					1	1	
26	1		1				1				1	1	
27	1		1			1			1				
28	1	1			1						1		
29	1	1						1			1	1	
30	1	1				1					1		
31	1	1					1				1	1	
32	1	1					1				1	1	
33	1		1					1			1		
34	1		1				1				1		
35	1	1			1						1		
36	1		1				1				1		
37	1		1		1						1		
38	1		1				1				1	1	
39	1		1			1			1				
40	1		1					1			1		
41	1	1					1				1		
42	1		1			1					1		
43	1		1					1	1				
44	1		1		1						1		
45	1	1			1						1		
46	1		1		1						1		
47	1	1						1			1	1	
48	1		1				1		1			1	
49	1	1				1			1				
50	1	1			1						1		
51	1		1			1					1	1	
52	1			1				1			1		
53	1		1		1						1		
54	1		1				1				1	1	
55	1	1			1				1				
56	1	1					1				1	1	
57	1		1					1			1		
58	1		1		1						1	1	
59	1		1					1			1		
60	1	1				1					1		
61	1		1					1			1	1	
62	1		1		1						1		
63	1		1					1			1		
64	1		1					1			1		
65	1		1			1					1		
66	1		1			1			1			1	
67	1	1				1			1			1	
68	1	1				1			1				

RO #	written	level 1	level 2	level 3	a	b	c	d	dir	mod	new	ro only	comments
69	1		1		1						1	1	
70	1		1					1			1	1	
71	1	1						1	1				
72	1		1				1				1	1	
73	1	1			1						1		
74	1		1		1						1	1	
75	1	1				1					1		
76	1		1		1				1				
77	1	1				1			1				
78	1	1						1			1		
79	1		1				1				1	1	
80	1	1					1				1		
81	1		1			1					1	1	
82	1		1			1					1	1	
83	1	1				1					1		
84	1	1			1				1			1	
85	1	1					1		1			1	
86	1		1			1					1	1	
87	1	1					1				1		
88	1	1					1				1		
89	1	1					1				1		
90	1	1				1					1		
91	1		1					1			1	1	
92	1	1				1					1	1	
93	1			1	1						1	1	
94	1		1					1			1		
95	1	1						1			1		
96	1	1					1				1		
97	1	1			1						1	1	
98	1	1					1				1		
99	1	1				1					1	1	
100	1	1						1			1		
	100	44	52	4	25	27	23	25	17	2	81	32	

SRO #	written	level 1	level 2	level 3	a	b	c	d	dir	mod	new	sro	sro specific	comments
1	1		1		1				1					
2	1		1				1				1	1	1	Apply T.S.
3	1		1					1			1			
4	1	1					1				1	1		
5	1		1					1			1			
6	1		1				1				1	1		
7	1	1				1					1			
8	1		1				1				1			
9	1		1		1						1			
10	1		1			1					1			
11	1		1					1			1			
12	1		1				1			1				
13	1	1					1			1		1	1	Control Room evacuation
14	1			1		1					1	1	1	E-plan / SED actions
15	1		1				1				1			
16	1		1			1					1			
17	1	1					1				1			
18	1		1		1						1	1		
19	1	1					1				1	1	1	Control of plant testing
20	1			1		1					1	1	1	ECCS vortex/NPSH
21	1	1			1						1			
22	1		1					1	1					
23	1		1				1				1	1		
24	1			1	1						1			
25	1		1			1					1	1	1	E-plan / Rad Release
26	1		1				1				1			
27	1	1				1				1				
28	1		1		1						1			
29	1		1		1						1			
30	1		1			1					1			
31	1			1				1			1			
32	1	1				1			1					
33	1	1			1						1			
34	1		1					1			1			
35	1	1					1				1	1		
36	1		1					1			1			
37	1		1			1			1					
38	1	1					1				1	1		
39	1		1		1						1	1	1	E-plan / SED responsibilities
40	1	1				1					1			
41	1	1			1						1			
42	1		1				1				1			
43	1		1		1						1	1	1	fire protection USAR reqmts
44	1		1					1			1			
45	1	1					1				1			
46	1		1			1					1			
47	1		1					1	1					
48	1		1		1						1			
49	1	1			1						1			
50	1		1		1						1			
51	1	1				1			1					
52	1	1			1						1			
53	1			1				1			1			
54	1		1		1						1			
55	1	1			1				1					
56	1		1					1			1			
57	1		1		1						1	1	1	Apply T.S.
58	1	1					1		1			1	1	license limits not exceeded
59	1		1			1					1	1	1	consequences of actions
60	1		1					1			1			
61	1			1	1						1	1	1	in-depth EOP knowledge
62	1		1		1						1			
63	1	1				1					1			
64	1		1					1			1			
65	1		1					1			1			
66	1		1			1					1	1	1	EOP use and direction
67	1		1		1						1			
68	1		1			1			1					

SRO #	written	level 1	level 2	level 3	a	b	c	d	dir	mod	new	sro	sro specific	comments
69	1		1			1					1	1	1	control of plant evolutions
70	1		1				1				1	1	1	assess plant conditions
71	1	1				1			1					
72	1	1			1						1			
73	1	1				1					1			
74	1		1		1				1					
75	1	1				1			1					
76	1	1						1			1			
77	1	1					1				1			
78	1	1						1			1	1	1	challenges to design bases
79	1	1				1					1			
80	1		1			1					1			
81	1	1			1						1	1	1	roles and responsibilities
82	1	1						1	1					
83	1	1					1				1			
84	1	1					1				1	1	1	command and control
85	1	1					1				1			
86	1	1					1				1			
87	1	1				1					1			
88	1		1					1			1	1	1	Apply T.S.
89	1		1		1						1	1	1	Apply T.S.
90	1		1				1				1	1	1	control of maintenance
91	1	1						1			1	1	1	procedure change process
92	1	1					1				1	1	1	LCO tracking / logging
93	1	1			1						1	1	1	authorizing dose extensions
94	1		1					1			1			
95	1	1						1			1			
96	1	1					1				1			
97	1		1					1			1	1	1	assess for TS compliance
98	1	1					1				1			
99	1	1			1						1	1	1	EOP use
100	1	1						1			1			
	100	43	51	6	27	24	26	23	13	3	84	32	26	

Reference Material Listing for Unit 2 Written Exam
on 12/6/99

RO Test

Question No.	Reference Material
RO3 SRO5	N2-EOP-RPV, "RPV Control" N2-EOP-PC, "Primary Containment Control
RO4 SRO7	N2-EOP-RPV, "RPV Control' (entry conditions blacked out) N2-EOP-PC, "Primary Containment Control (entry conditions blacked out" N2-EOP-SC/RR, "Secondary Containment Control Radioactivity Release Control' (entry conditions blacked out)
RO9 SRO16	All EOP Graphs (full-size) and ruler
RO12 SRO24	N2-EOP-C5, "Failure to Scram"
RO13 SRO26	N2-EOP-PCH, Rev. 0 N2-EOP-PC, "Primary containment Control"
RO14 SRO27	Power to flow map (Ref dwg EM-950 A/B)
RO18 SRO31	O2-OPS-001-259-2-02, Figure 1, Rev. 0
RO20 SRO9	EOP's without entry conditions
RO26	Attachment 1 of N2-SOP-30, "Control Rod Drive Failures, Flow Diagram"
RO31	EPIP-EPP-02, Attachment 1.0
RO36 SRO42	EOP's without entry conditions
RO51	N2-EOP-RPV, Rev. 8, Figure A, Figure C
RO58	EOP-s without entry conditions

Reference Material Listing for Written Exam

SRO Only Test

Question No.	Reference Material
SRO 2	Technical Specification 3.8.1
SRO14	EPIP-EPP-08, Attachment 1 and Table 1.1
SRO18	EOP's without entry conditions
SRO20	N2-EOP-6, Attachment 29
SRO23	EOP's without entry conditions
SRO25	EPIP-EPP-02, Attachment 1, Rev. 8
SRO39	EPIP-EPP-02, Attachment 1, Rev. 8
SRO 43	USAR, Rev. 10, Section 9A.3.6.1 USAR, Rev. 10, Table 9A.3-18
SRO57	Technical Specifications 3.4.3, 3.4.3.1/4.4.3.1
SRO61	RPV Control EOP
SRO88	Technical Specifications, Section 3.7.1.1
SRO89	N2-FHP-13.1 Technical Specifications, Section 3.9
SRO99	All EOP's with the entry conditions blacked out

Reference Material Listing for Unit 2 Written Exam
On 12/6/99

SRO Test

Question No.	Reference Material
SRO 2	Technical Specification 3.8.1
RO3 SRO5	N2-EOP-RPV, "RPV Control" N2-EOP-PC, "Primary Containment Control
RO4 SRO7	N2-EOP-RPV, "RPV Control' (entry conditions blacked out) N2-EOP-PC, "Primary Containment Control (entry conditions blacked out" N2-EOP-SC/RR, "Secondary Containment Control Radioactivity Release Control' (entry conditions blacked out)
RO20 SRO9 SRO14	EOP's without entry conditions EPIP-EPP-08, Attachment 1 and Table 1.1
RO9 SRO16	All EOP Graphs (full-size) and ruler
SRO18	EOP's without entry conditions
SRO20	N2-EOP-6, Attachment 29
SRO23	EOP's without entry conditions
RO12 SRO24	N2-EOP-C5, "Failure to Scram"
SRO25	EPIP-EPP-02, Attachment 1, Rev. 8
RO13 SRO26	N2-EOP-PCH, Rev. 0 N2-EOP-PC, "Primary containment Control"
RO14 SRO27	Power to flow map (Ref dwg EM-950 A/B)
RO18 SRO31	O2-OPS-001-259-2-02, Figure 1, Rev. 0
SRO39	EPIP-EPP-02, Attachment 1, Rev. 8
RO36 SRO42	EOP's without entry conditions
SRO 43	USAR, Rev. 10, Section 9A.3.6.1 USAR, Rev. 10, Table 9A.3-18
SRO57	Technical Specifications 3.4.3, 3.4.3.1/4.4.3.1
SRO61	RPV Control EOP
SRO88	Technical Specifications, Section 3.7.1.1

Reference Material Listing for Unit 2 Written Exam
On 12/6/99

SRO Test

Question No.	Reference Material
SRO89	N2-FHP-13.1 Technical Specifications, Section 3.9
SRO99	All EOP's with the entry conditions blacked out

U.S. Nuclear Regulatory Commission
Site-Specific
Written Examination

Applicant Information

Name:	Region: I
Date: December 6, 1999	Facility/Unit: Nine Mile Point / Unit 2
License Level: RO	Reactor Type: GE
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. The passing grade requires a final grade of at least 80.00 percent. Examination papers will be collected five hours after the examination starts.

Applicant Certification

All work done of this examination is my own. I have neither given nor received aid.

____
Applicant's Signature

Results

Examination Value	100 Points
Applicant's Score	____ Points
Applicant's Grade	____ Percent

U.S. Nuclear Regulatory Commission
Site-Specific
Written Examination

Applicant Information

Name:	Region: I
Date: December 6, 1999	Facility/Unit: Nine Mile Point / Unit 2
License Level: SRO	Reactor Type: GE
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. The passing grade requires a final grade of at least 80.00 percent. Examination papers will be collected five hours after the examination starts.

Applicant Certification

All work done of this examination is my own. I have neither given nor received aid.

____
Applicant's Signature

Results

Examination Value	100 Points
Applicant's Score	______ Points
Applicant's Grade	______ Percent

Question # RO 1 SRO 30

		RO	SRO
Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	1	1
	Group #	1	1
	K/A #	295005 2.1.33	295005 2.1.33
	Importance Rating	3.4	4.0

Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications.

Proposed Question:

While operating at 100% power an electrical transient in the 345 KV Scriba Switchyard caused a full load reject at NMP2. After the initial actions were taken the STA determined the reactor was shutdown by the Alternate Rod Insertion function of RRCS.

Which one of the following states the significance of this event?

a. A reactivity anomoly has occurred.

b. An engineered safety feature has failed to actuate.

c. The Generator output breakers R-925 and R-230 failed to trip open.

d. Control Valve fast closure failed and the turbine tripped on overspeed.

Proposed Answer: b.

Explanation (Justification of Distractors):

a. A reactivity anomoly has NOT occurred.
c. This would not cause this event to be significant.
d. This would be expected for a failure of the low ETS oil pressure trip.

Technical Reference(s): Technical Specifications 2.2.1 and 3.3.1

Proposed references to be provided to applicants during the examination:

None

Learning Objective:	02-OPS-001-245-2-01, EO-11.0
Question Source:	Bank # Modified Bank # New New
Question History:	Previous NRC Exam Previous Test / Quiz
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis 2
10CFR Part 55 Content:	43.2 / 45.3 / 45.2

Comments:

Question # RO 2 SRO 3

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	1	1
	Group #	1	1
	K/A #	295006 AA2.06	295006 AA2.06
	Importance Rating	3.7	3.8

Ability to determine and/or interpret the following as they apply to SCRAM: Cause of reactor SCRAM.

Proposed Question:

The plant is operating at 100% power when the following events occur:

- Reactor Scrams
- Turbine Stop Valves CLOSE
- Turbine Control Valves CLOSE
- Turbine Bypass Valves OPEN
- Several SRVs OPEN

Based on these actions occurring within a few seconds, without any operator actions, which one of the following conditions caused this transient?

a. Nuclear Instruments failed causing RPS to trip.

b. An instrument failure caused a Recirculation FCV runback.

c. Main Steam Line Radiation exceeded the High-High setpoint.

d. A feedwater malfunction caused RPV level to exceed Level 8.

Proposed Answer: d. Exceeding RPV Level 8 causes a Turbine Trip which closes the TCVs and TSVs causing a scram and over-pressure condition.

Explanation (Justification of Distractors):

a. This would NOT cause a turbine trip therefore the TCVs would NOT close and the SRVs would NOT open.
b. The flow control valve closure results in level rise, but not to Level 8.
c. This event also causes an MSIV isolation which would not open the bypass valves.

Technical Reference(s): USAR. Chapter 15, Section 2, Table 15.2-3 (attached)
N2-SOP-06, FEEDWATER FAILURES
N2-SOP-21, TURBINE TRIP

Proposed references to be provided to applicants during the examination:

None

Learning Objective: 02-OPS-001-212-2-00, EO-5.0

Question Source:

Bank #	New
Modified Bank #	New
New	New

Question History:

Previous NRC Exam	New
Previous Test / Quiz	New

Question Cognitive Level:

Memory of Fundamental Knowledge	
Comprehension or Analysis	2

10CFR Part 55 Content: 41.10 / 43.5 / 45.13

Comments:

Question # RO 3 SRO 5

		RO	SRO
Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	1	1
	Group #	1	1
	K/A #	295007 AK3.03	295007 AK3.03
	Importance Rating	3.4	3.5

Knowledge of the reasons for the following responses as they apply to High Reactor Pressure: RCIC operation: Plant-Specific.

Proposed Question:

The EOPs have been entered following a plant trip due to a loss of all operating Circulating Water Pumps. The following conditions exist:

- A Group 1 isolation signal has occurred.
- RPV pressure is 1050 psig and rising.

Which one of the following systems should be used for reactor pressure control?

a. Shutdown Cooling

b. Turbine Bypass Valves

c. Main Steam Line Drains

d. Reactor Core Isolation Cooling

Proposed Answer: d. RPV CONTROL, Step P-5, RCIC is an alternate system that can be used.

Explanation (Justification of Distractors):

a. SRVs are not available because there is no power to the C solenoids
b. Bypass Valves are NOT available following MSIV closure
c. Main steam line drains also isolate with Group 1 isolation

Technical Reference(s): N2-EOP-RPV, RPV CONTROL
N2-EOP-PC, PRIMARY CONTAINMENT CONTROL

(Attach if not previously provided)

Proposed references to be provided to applicants during the examination:

N2-EOP-RPV, RPV CONTROL
N2-EOP-PC, PRIMARY CONTAINMENT CONTROL

Learning Objective:	02-OPS-006-344-2-01, TO-13.0, TO-15	
Question Source:	Bank #	
	Modified Bank #	
	New	New
Question History:	Previous NRC Exam	
	Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge	
	Comprehension or Analysis	2
10CFR Part 55 Content:	41.5 / 45.6	

Comments:

Question # RO 4 SRO 7

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	1	1
	Group #	1	1
	K/A #	Generic 295009 2.4.4	Generic 295009 2.4.4
	Importance Rating	4.0	4.3
Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.			

Proposed Question:

Which one of the following responses contains the EOP(s) required to be entered if a loss of Drywell Cooling causes Drywell pressure to rise to 1.7 psig"?

a. N2-EOP-RPV, RPV CONTROL only.

b. N2-EOP-RPV, RPV CONTROL and N2-EOP-PC, PRIMARY CONTAINMENT CONTROL only.

c. N2-EOP-PC, PRIMARY CONTAINMENT CONTROL and N2-EOP-SC, SECONDARY CONTAINMENT CONTROL only.

d. N2-EOP-PC, PRIMARY CONTAINMENT CONTROL, N2-EOP-RPV, RPV CONTROL and N2-EOP-SC, SECONDARY CONTAINMENT CONTROL only.

Proposed Answer: b. Entry into N2-EOP-RPV and N2-EOP-PC are required.

Explanation (Justification of Distractors):

a. Entry into N2-EOP-RPV only is NOT complete entry into N2-EOP-PC is also required.
c. Entry into N2-EOP-PC only is NOT complete entry into N2-EOP-RPV is also required. Entry into N2-EOP-SC is NOT required.
d. Entry into N2-EOP-SC is NOT required.

NOTE: Any EOPs supplied for the exam must have the entry conditions blanked out.

Technical Reference(s):	N2-EOP-RPV, "RPV CONTROL" and N2-EOP-PC, "PRIMARY CONTAINMENT CONTROL".

Proposed references to be provided to applicants during the examination:

N2-EOP-RPV, "RPV CONTROL" (Entry conditions blacked out)
N2-EOP-PC, "PRIMARY CONTAINMENT CONTROL" (Entry conditions blacked out)
N2-EOP-SC/RR, "SECONDARY CONTAINMENT CONTROL RADIOACTIVITY RELEASE CONTROL" (Entry conditions blacked out)

Learning Objective:	02-OPS-006-344-2-01, EO-1.0 02-OPS-006-344-2-04, EO-1.0	
Question Source:	Bank #	New
	Modified Bank #	New
	New	New
Question History:	Previous NRC Exam	New
	Previous Test / Quiz	New
Question Cognitive Level:	Memory of Fundamental Knowledge	1
	Comprehension or Analysis	
10CFR Part 55 Content:	4.10 / 43.2 / 45.6	

Comments:

Question # RO 5 SRO 8

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	1	1
	Group #	1	1
	K/A #	295010 AA1.02	295010 AA1.02
	Importance Rating	3.6	3.6
Ability to operate and/or monitor the following as they apply to high drywell pressure: Drywell floor and equipment drain sumps.			

Proposed Question:

The unit operating at 80% power. The last drywell floor drain pump to operate was pump 2DFR-P1A. Both the 2DFR-P1A and 2DFR-P1B pump control switches are in the NORMAL-AFTER-STOP position. The following annunciators are received:

- 873111, DRWL FLR DRN TANK 1 LEVEL HI-HI
- 603140, DRYWELL PRESSURE HIGH/LOW

Which one of the following describes the status of the Drywell Floor Drain System as drywell pressure rises from 0.75 psig to 1.8 psig?

a. **Both** the P1A and P1B pumps will be off and remain off.

b. **Only** the P1B pump will pump the sump until the system isolates.

c. **Both** the P1A and P1B pumps will pump the sump until the system isolates.

d. **Only** the P1B pump will pump the sump and it will continue to pump until manually stopped.

Proposed Answer: c.

Explanation (Justification of Distractors):
When the hi-hi level alarm is received, the P1B pump is already running and the P1A pump starts. They will pump the sump until the LOCA signal (1.68 psig drywell pressure) automatically isolates the system.

Technical Reference(s): N2-OP-67, Rev 02, Section B.2, Section H.1.0
N2-ARP-01, Rev 00, 873111

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-291-2-01, EO-4b, EO-4c	
Question Source:	Bank # Modified Bank # New	New
Question History:	Previous NRC Exam Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	2
10CFR Part 55 Content:	55.41.7 55.45.6	

Comments:

Question # RO 6 SRO 10

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	1	1
	Group #	1	1
	K/A #	295014 AA1.02	295014 AA1.03
	Importance Rating	3.6	3.8

Ability to operate and/or monitor the following as they apply to INADVERTENT REACTIVITY ADDITION: Recirculation flow control system

Proposed Question:

During a power ascension the reactor is at 82% power while raising power with Recirculation Flow. The operator is attempting to manually open the Recirculation Loop "A" FCV with the 602 Panel Flow Controller but is unable because the servo valve at the HPU is stuck. After numerous attempts to open the FCV the servo valve becomes free and rapidly moves to the open position and sticks there. The FCV begins to fully open.

Which one of the following actions will remedy this situation?

a. Startup the standby HPU.

b. Secure and isolate the HPU.

c. Place the Loop Controller in AUTO and close the FCV.

d. Lower the Loop Controller in MANUAL until the FCV is closed.

Proposed Answer: b.

Explanation (Justification of Distractors):

a. This will have no effect on a stuck servo valve
c. This will have no effect on a stuck servo valve
d. With the stuck servo valve this will have no effect

Technical Reference(s): N2-SOP-08, UNPLANNED POWER CHANGES

Proposed references to be provided to applicants during the examination:

None

Learning Objective:	02-OPS-001-202-2-02, EO-2.0, 3.0, 8.0	
Question Source:	Bank # Modified Bank # New	New
Question History:	Previous NRC Exam Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	2
10CFR Part 55 Content:	41.7 / 45.6	

Comments:

Question # RO 7 SRO 11

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	1	1
	Group #	1	1
	K/A #	295015 AK2.11	295015 AK2.11
	Importance Rating	3.5	3.7
Knowledge of the interrelations between INCOMPLETE SCRAM and the following: Instrument Air			

Proposed Question:

During a reactor startup a Scram Discharge Volume (SDV) High Level Scram occurred. The following conditions exist:

- 53% of the control rods remain in the core at various positions
- Some movement was observed on all control rods
- Scram solenoid power lights are OFF
- Scram Valves have been verified Open at the HCUs

Based on these conditions EOP-6, Attachment 14, directs manually initiating additional scrams. Which one of the following is the basis for this action?

a. Allows additional scrams at lower reactor pressures.

b. Provides another scram to totally vent the scram air header.

c. Closes the scram valves to allow recovery of the CRD pumps.

d. Resets the scram to establish air to the SDV vent and drain valves.

Proposed Answer: d. Eliminates the hydraulic lock by opening the SDV vent and drain valves to drain the SDV.

Explanation (Justification of Distractors):

a. There is no basis for scraming at lower pressure.
b. The Scram air header is vented now.
c. The CRD pumps are operating.

Technical Reference(s):

N2-EOP-6, Attachment 14, Sect 3.3
EOP Basis

Proposed references to be provided to applicants during the examination:

None

Learning Objective:	02-OPS-006-344-2-01, EO-3.0
Question Source:	Bank # Modified Bank # New New
Question History:	Previous NRC Exam Previous Test / Quiz
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis 2
10CFR Part 55 Content:	41.7 / 45.8

Comments:

Question # RO 8 SRO 12

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	1	1
	Group #	1	1
	K/A #	295015 AA1.02	295015 AA1.02
	Importance Rating	4.0	4.2
Ability to operate and/or monitor the following as they apply to Incomplete Scram: RPS			

Proposed Question:

The plant is operating at 100% power when a failure of an RPS relay occurs causing the following conditions to exist on the 2CEC*PNL603:

- Only 3 out of 4 solenoid lights are **ON** for the "A" RPS Trip System
- All 4 solenoid lights for the "B" RPS Trip System are **ON**

Which one of the following describes the immediate plant impact of an RPS "B" System trip?

a. All control rods will insert.

b. No control rods will insert.

c. One quarter of the control rods will insert.

d. Three quarters of the control rods will insert.

Proposed Answer: c.

Explanation (Justification of Distractors):

a. Three quarters of the A scram solenoids are energized and these rods will NOT scram.
b. One quarter of the rods will have both scram solenoids de-energized and will scram.
d. Three quarters of the A scram solenoids are energized and these rods will NOT scram.

Technical Reference(s): 02-OPS-001-212-2-00, Figures 5, 6A, 6B
N2-OP-97, REACTOR PROTECTION SYSTEM

(Attach if not previously provided)

Proposed references to be provided to applicants during the examination:

N/A

Learning Objective: 02-OPS-001-212-2-00, EO-2.0, EO-8.0

Question Source:

Bank #
Modified Bank # Q8154
New

Question History:

Previous NRC Exam No
Previous Test / Quiz No

Question Cognitive Level:

Memory of Fundamental Knowledge
Comprehension or Analysis 2

10CFR Part 55 Content: 41.7 / 45.6

Comments:

QUESTION 259. (Point value: 1.00 , T.R.A.I.N. Q8154)

The plant is operating at 100% Reactor Power, the following conditions exist.

- A failure of RPS channel sensor relay K14M has just occurred.
- for the 'A' RPS trip system (P603)
  - 3 out of 4 solenoid lights are on
  - The 'A' white solenoid light is out
- All 4 solenoid lights are on for the 'B' RPS trip system (P603)

Which one of the following describes the <u>immediate</u> plant impact a scram signal in the 'B' RPS trip system would have?

a. A full Scram occurs and all control rods insert.

b. A half Scram occurs with no control rod motion.

c. A quarter of the control rods insert.

d. There will be no effect on the plant

Answer:

C

KA/Setting:
NMPC KA #: 0.00
Setting : C1

Related Training:
O2 -OPS -001-212-2-00 Rev 0

Related Items:
GE, DRW, N2, 807E166TY, , Sh 10
NMPC, LP, N2, O2-OPS-001-212-2-00, 0, EO-8.0
NRC, NUREG, NA, 1123, 4.1/4.1, 212000 K3.06

Question # RO 9 SRO 16

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	1	1
	Group #	1	1
	K/A #	295024 EK3.04	295024 EK3.04
	Importance Rating	3.7	4.1

Knowledge of the reasons for the following responses as they apply to high drywell pressure: emergency depressurization.

Proposed Question:

A reactor scram due to a LOCA has occurred. The following conditions exist:

• Reactor pressure	400 psig
• Reactor water level (actual) 0 inches	stable
• Drywell pressure	16 psig
• Drywell temperature	250°F
• Suppression chamber pressure	17 psig
• Suppression pool temperature	135°F
• Suppression pool water level	201 feet

Which one of the following is assured by performing an RPV Blowdown under the current plant conditions?

a. Ensure the suppression chamber design temperature is not exceeded.

b. Ensure that steam does not accumulate in the suppression chamber air space.

c. Ensure that containment vent valves can be opened and closed to reject heat from and to vent the containment.

d. Ensure opening an SRV will not result in exceeding the capability of the SRV tail pipe, quencher, or associated supports.

Proposed Answer: b.

Explanation (Justification of Distractors):
The Pressure Suppression Pressure is being challenged.
a. Reason for depressurizing prior to exceeding the HCTL.
c. Reason for depressurizing prior to exceeding the PCPL.
d. Reason for depressurizing prior to exceeding the SRVTPL.

Technical Reference(s):	N2-EOP-PC, Rev 8 NMP2-EOP Bases Document, Section C	

Proposed references to be provided to applicants during the examination:

All the EOP Graphs (full size) and ruler.

Learning Objective:	O2-OPS-006-344-2-04, # 3	
Question Source:	Bank # Modified Bank # New	 New
Question History:	Previous NRC Exam Previous Test / Quiz	New New
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 2
10CFR Part 55 Content:	55.41.5 55.45.6	

Comments:

Question # RO 10 SRO 17

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	1	1
	Group #	1	2
	K/A #	295025 EK1.05	295025 EK1.05
	Importance Rating	4.4	4.7
Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE: Exceeding Safety Limits.			

Proposed Question:

During the conduct of N2-OSP-RPV-@002, REACTOR PRESSURE VESSEL AND ALL CLASS 1 SYSTEMS LEAKAGE TEST, reactor pressure is raised to 1375 psig.

Which one of the following describes the significance of this event?

a. The reactor pressure vessel warranty has been voided.

b. A principle safety barrier has been significantly degraded.

c. A Technical Specification safety limit has been exceeded.

d. Conditions existed that are outside of station procedures.

Proposed Answer: c. 1325 psig in the steam dome safety limit has been violated

Explanation (Justification of Distractors):

a. Not a valid concern
b. Not a valid concern
d. Not a valid concern

Technical Reference(s): T.S. 2.1.3 and 6.7

Proposed references to be provided to applicants during the examination:

None

Learning Objective:	02-OPS-008-362-2-01
Question Source:	Bank # New Modified Bank # New
Question History:	Previous NRC Exam Previous Test / Quiz
Question Cognitive Level:	Memory of Fundamental Knowledge 1 Comprehension or Analysis
10CFR Part 55 Content:	55.41 8 55.41.10

Comments:

Question # RO 11 SRO 22

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	1	1
	Group #	1	1
	K/A #	295031 EK2.08	295031 EK2.08
	Importance Rating	4.2	4.3

Knowledge of the interrelations between reactor low water level and the following: Automatic depressurization system.

Proposed Question:

A LOCA has occurred and NO operator action has been taken. The following conditions have been present for 2 minutes:

- RPV level indicates -100 inches on the Fuel Zone range
- Reactor pressure is 300 psig
- Drywell pressure is 22 psig

Assume ALL equipment operates as designed. Which one of the following describes the current status of the ADS valves, and the actions necessary to close or maintain them closed?

The ADS valves are ...

a. **open.**
Div. I and Div. II DISABLE key lock switches placed in ON.

b. **closed.**
Div. I and Div. II DISABLE key lock switches placed in ON.

c. **closed.**
Div. I and Div. II SEAL-IN RESET pushbuttons depressed every 90 seconds.

d. **open.**
Div. I and Div. II DISABLE key lock switches placed in ON <u>and then</u> Div. I and Div. II SEAL-IN RESET pushbuttons are depressed.

Proposed Answer: d.

Explanation (Justification of Distractors):

a. The automatic initiation circuit bypasses the key lock switches.

b. The valves are open. IF the valves were closed, the action indicated would maintain them closed.

c. The valves are open. IF the valves were closed, the action indicated would maintain them closed.

Technical Reference(s): N2-OP-34, Rev 07, Section B.2.3

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-218-2-01, EO-4b,4d	
Question Source:	Bank #	Q15787
	Modified Bank #	
	New	
Question History:	Previous NRC Exam	
	Previous Test / Quiz	(wk 15 exam)
Question Cognitive Level:	Memory of Fundamental Knowledge	
	Comprehension or Analysis	2
10CFR Part 55 Content:	55.41.7	
	55.45.8	

Comments:

Question # RO 12 SRO 24

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	1	1
	Group #	1	1
	K/A #	295037 EK1.02	295037 EK1.02
	Importance Rating	4.1	4.3

Knowledge of the operational implications of the following concepts as they apply to scram condition present and reactor power above APRM downscale or unknown: Reactor water level effects on reactor power.

Proposed Question:

Note: Reactor water levels are <u>indicated</u>.
An ATWS is in progress. Following the actions to terminate and prevent all RPV injection, injection was reestablished when reactor water level reached –10 inches because APRM downscales were received. Conditions at the time were:

- Reactor water level –10 inches
- Reactor power 3%
- Reactor pressure 1000 psig
- Suppression pool temperature 120°F and rising slowly
- 2 SRVs are open
- Control rod insertion has NOT been established
- SLS failed to inject and CANNOT be started
- No alternate boron system is injecting

One (1) minute later, reactor water level has risen to +30 inches. Which one of the following describes the required operator actions including why?

a. Terminate and prevent injection except for boron, CRD, and RCIC because reactor power is above 4%.

b. Perform an RPV Blowdown because reactor water level CANNOT be maintained in the required band.

c. Lower level using the preferred ATWS systems to the assigned reactor water level band because adequate core cooling is NOT assured.

d. Assign a new level band and maintain reactor water level between +30 inches and –45 inches using alternate ATWS systems for improved control.

Proposed Answer: a.

The rise in reactor water level will cause reactor power to rise above 4%. The conditions of the override step for terminate and prevent injection are met and must be performed.

Explanation (Justification of Distractors):

d. The rise in reactor water level will cause reactor power to rise above 4%. Action must be taken to lower reactor power.

b. Reactor water level is above TAF and adequate core cooling is assured. Emergency depressurization is NOT required.

c. Level is above TAF, adequate core cooling is assured.

Technical Reference(s): N2-EOP-C5, Rev 8

Proposed references to be provided to applicants during the examination:

N2-EOP-C5, Failure to Scram.

Learning Objective: O2-OPS-006-344-2-17, EO-2

Question Source:	Bank # Modified Bank # New	 New
Question History:	Previous NRC Exam Previous Test / Quiz	New New
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 3
10CFR Part 55 Content:	55.41.6 55.41.7 55.45.8	

Comments:

12-2

Question # RO 13 SRO 26

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	1	1
	Group #	1	1
	K/A #	500000 EK1.01	500000 EK1.01
	Importance Rating	3.3	3.9

Knowledge of the operational implications of the following concepts as they apply to high containment hydrogen concentrations: Containment integrity.

Proposed Question:

A LOCA has occurred and the following conditions exist:

- Drywell H2 concentration is 7%
- Suppression Chamber H2 concentration is 4%

- Drywell O2 concentration is 4%
- Suppression chamber O2 concentration is 6%

In accordance with the EOPs, which one of the following describes the Primary Containment H2/O2 deflagration limit status and required actions?

The Primary Containment H2/O2 concentration is ...

a. **below** the deflagration limit. A Reactor scram and emergency depressurization is required.

b. **below** the deflagration limit. A Reactor scram and emergency depressurization is **NOT** required.

c. **above** the deflagration limit. A Reactor scram and emergency depressurization is required.

d. **above** the deflagration limit. A Reactor scram and emergency depressurization is **NOT** required.

Proposed Answer: c.

Explanation (Justification of Distractors):
The limits, 6%, H2 and 5%, O2 in either the suppression chamber or drywell are the limits for the primary containment. Combustible limit exceeded requires a reactor scram and emergency depressurization.

Technical Reference(s): N2-EOP-PCH, Rev 0
NMP2 EOP Bases, EOP-PCH

Proposed references to be provided to applicants during the examination:

N2-EOP-PCH, Rev 0
N2-EOP-PC, Rev 8

Learning Objective: O2-OPS-006-344-2-23, EO-2, EO-3,

Question Source:	Bank #	
	Modified Bank #	
	New	New
Question History:	Previous NRC Exam	New
	Previous Test / Quiz	New
Question Cognitive Level:	Memory of Fundamental Knowledge	2
	Comprehension or Analysis	
10CFR Part 55 Content:	55.41.8, 41.9, 41.10	
	55.45.5	

Comments:

13-2

Question # RO 14 SRO 27

		RO	SRO
Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	1	1
	Group #	2	2
	K/A #	295001 AA2.01	295001 AA2.01
	Importance Rating	3.5	3.8

Ability to determine and/or interpret the following as they apply to partial or complete loss of forced core flow circulation: Power/flow map.

Proposed Question:

The plant is operating at 85% power above the 100% rod line. A fault occurs causing the "A" RCS pump to trip to off.

- The resulting core flow is 44 x 10^6 lbm/hr.

Which one of the following describes the required immediate actions?

a. Raise recirc flow to at least 50 x 10^6 lbm/hr in accordance with N2-SOP-29.

b. Place the Reactor Mode Switch in Shutdown and follow the actions of N2-SOP-101C.

c. Monitor APRMs and LPRMs for indication of thermal hydraulic oscillations and scram the reactor if oscillations exist.

d. Reduce power to less than 70%, Notify I&C Dept. & Reactor Engineering, verify RCS pump A speed is zero and shut FCV 17A.

Proposed Answer: b.

If above the 100% rod line and total core flow is ≤ 45% (49 mlb/hr), then scram the reactor per N2-SOP-101C.

Explanation (Justification of Distractors):

a. A reactor scram is required per N2-SOP-29. This is a subsequent action.
c. A reactor scram is required per N2-SOP-29. This is a subsequent action.
d. A reactor scram is required per N2-SOP-29. This is a subsequent action.

Technical Reference(s): N2-SOP-29, Rev 00, Section 3.0

Proposed references to be provided to applicants during the examination:

Power to flow map (Ref dwg EM-950A/B)

Learning Objective: O2-OPS-006-SOP-2-01-29, TO-12, EO-2

Question Source:
Bank #
Modified Bank # Q8211
New

Question History:
Previous NRC Exam
Previous Test / Quiz

Question Cognitive Level:
Memory of Fundamental Knowledge 1
Comprehension or Analysis

10CFR Part 55 Content:
55.41.10
55.43.5
55.45.13

Comments:

QUESTION 275. (Point value: 1.00 , T.R.A.I.N. Q8211)

The plant is operating at 85% power. The Red Rod line sign is posted. A fault occurs causing the "A" RCS pump to trip to off. The resulting core flow is 45 x 10^6 lbm/hr. Which one of the following best describes the immediate actions to be taken?

a. No Action, allow parameters to stabilize for further assessment.

b. Immediately raise recirc flow to at least 50 x 10^6 lbm/hr with the B Recirc flow control valve.

c. Immediately place the Reactor Mode Switch in Shutdown, follow SOP101C actions.

d. Reduce power to less than 70%, Notify I&C Dept. & Reactor Engineering, verify RCS pump A speed is zero and shut FCV 17A

Answer:

C

KA/Setting:
NMPC KA #: 0.00
Setting : C1

Related Training:
O2 -OPS -001-202-2-01 Rev 0

Related Items:
NMPC, LP, N2, O2-OPS-001-202-2-01, 0, EO-7d
NMPC, PROC, N2, N2-OP-29, 06, SECT H.2.0
NRC, NUREG, NA, 1123, 0, 295001AA2.03
NRC, NUREG, NA, 1123, 0, 295001G K/A7
NRC, NUREG, NA, 1123, 0, 295001GKA/10

Question # RO 15 SRO 28

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	1	1
	Group #	1	1
	K/A #	295002 AA1.07	295002 AA1.07
	Importance Rating	3.1	2.9
Ability to operate and/or monitor the following as they apply to a LOSS OF MAIN CONDENSER VACUUM: condenser circulating water system			

Proposed Question:

With the plant operation at 100% power the following conditions exist:

- Circulating Water Pumps "A", "B", "C", "E", "F" are in operation
- Circulating Water Pump "C" TRIPs

Which one of the following actions is required?

a. Scram and trip the Main Turbine.

b. Verify Circulating Water system in Mode 1.

c. Determine the cause of the trip and re-start the pump.

d. Confirm delta vacuum between any two condensers is <4" Hg.

Proposed Answer: a.
Main turbine must be tripped if a condenser is NOT receiving CW, when the C pump trips the center (B) condenser has no CW (C & D are secured).

Explanation (Justification of Distractors):
See above

Technical Reference(s):

N2-ARP-01, 851301

Proposed references to be provided to applicants during the examination:

None

Learning Objective:	02-OPS-001-275-2-00, EO-3.0	
Question Source:	Bank # Modified Bank # New	 New
Question History:	Previous NRC Exam Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 2
10CFR Part 55 Content:	41.7 / 45.6	

Comments:

Question # RO 16 SRO 1

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	1	1
	Group #	2	1
	K/A #	295003 AK2.03	295003 AK2.03
	Importance Rating	3.7	3.9

Knowledge of the interrelations between Partial or Complete Loss of A.C. power and the following: A.C. electrical distribution system.

Proposed Question:

The plant is operating at 100% power with the normal AC distribution lineup. An overcurrent condition occurs on the **Reserve Transformer B** resulting in actuation of its protective relaying.

Which one of the following states the plant AC busses immediately de-energized as a result of the automatic fault isolation?

a. 2ENS*SWG103 (Div. II)

b. 2ENS*SWG101 (Div. I)

c. 2ENS*SWG103 (Div. II) and 2NPS-SWG003

d. 2ENS*SWG101 and 2ENS*SWG102 (Div. I & III)

Proposed Answer: a.

Explanation (Justification of Distractors):
b. SWG101 is supplied by Reserve Trans. A, SWG001 is NOT affected
c. SWG003 is NOT affected
d. SWG101 is supplied by Reserve Trans. A, SWG102 is NOT affected

Technical Reference(s):	N2-OP-70, Sect. B and Attachment 2 N2-OP-71A, Sect. B
	(Attach if not previously provided)

Proposed references to be provided to applicants during the examination:

None

Learning Objective:	NMPC, LP, N2, O2-OPS-001-262-2-01, 0, EO-8	
Question Source:	Bank # Modified Bank # New	Q8094
Question History:	Previous NRC Exam Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 2
10CFR Part 55 Content:	55.41 55.43	

Comments:

Question # RO 17 SRO 29

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	1	1
	Group #	2	2
	K/A #	295004 AK1.02	295004 AK1.02
	Importance Rating	3.2	3.2

Knowledge of the operational implications of the following concepts as they apply to Partial or Complete Loss of D.C. Power: Redundant D.C. power supplies: Plant-Specific

Proposed Question:

The plant is operating at 75% power when a fault in the Division I 125 VDC Battery, 2BYS*BAT2A, causes the following:

- Battery Breaker to Division I DC Switchgear, 2BYS*SWG002A, trips **OPEN**.
- Charger 2BYS*CHGR2A1, Output Breaker to Division I DC Switchgear, 2BYS*SWG002A, trips **OPEN**.

What is the effect on plant operation **and** what must be done to restore power to the Division I 125 VDC Bus or loads?

a. Immediate scram is required, DC power can be restored using the standby charger.

b. A plant shutdown should be started, DC power can be restored using the standby charger.

c. Orderly plant shutdown is required, DC power can **NOT** be restored until the battery is available.

d. Stop all activity that could result in a plant trip, DC power can **NOT** be restored until the battery is available.

Proposed Answer: a. Plant must be scrammed because both recirculation pumps tripped (SOP-29 and SOP-4)). Power can be restored by placing the alternate charger in service (SOP-4).

Explanation (Justification of Distractors):

See above

Technical Reference(s): N2-SOP-04, and N2-SOP-29.

Proposed references to be provided to applicants during the examination:

None

Learning Objective:	02-OPS-001-263-2-01, EO-8.0	
Question Source:	Bank # Modified Bank # New	New
Question History:	Previous NRC Exam Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 2
10CFR Part 55 Content:	55.41 55.43	

Comments:

Question # RO 18 SRO 31

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	1	1
	Group #	2	2
	K/A #	295008 AK1.03	295008 AK1.03
	Importance Rating	3.2	3.2

Knowledge of the operational implications of the following concepts as they apply to high reactor water level: Feed flow / steam flow mismatch.

Proposed Question:

Given the following conditions:

- Reactor power is steady at **50%**
- Reactor water level is 182 inches
- Reactor Vessel Level Control System is in 3-element control
- Reactor level detector channel "A" is selected

The Channel "B" feedwater flow **SIGNAL** fails to ZERO.

Which one of the following describes the result?

a. Scram on Main Turbine trip.

b. Scram on reactor water level trip.

c. Reactor water level stabilizes at a lower level.

d. Reactor water level stabilizes at a higher level.

Proposed Answer: d.

Explanation (Justification of Distractors):

a. Level rises but should not reach the high level setpoint at this steam flow.
b. Level rises
c. Level rises

Technical Reference(s): N2-OP-03, Rev 13, Section B

Proposed references to be provided to applicants during the examination:

02-OPS-001-259-2-02, Figure 1, Rev 0

Learning Objective: O2-OPS-001-259-2-02, EO-8.0

Question Source:	Bank #		
	Modified Bank #	New	
	New	New	
Question History:	Previous NRC Exam	New	
	Previous Test / Quiz	New	
Question Cognitive Level:	Memory of Fundamental Knowledge		
	Comprehension or Analysis		3
10CFR Part 55 Content:	55.41.8		
	55.41.10		

Comments:

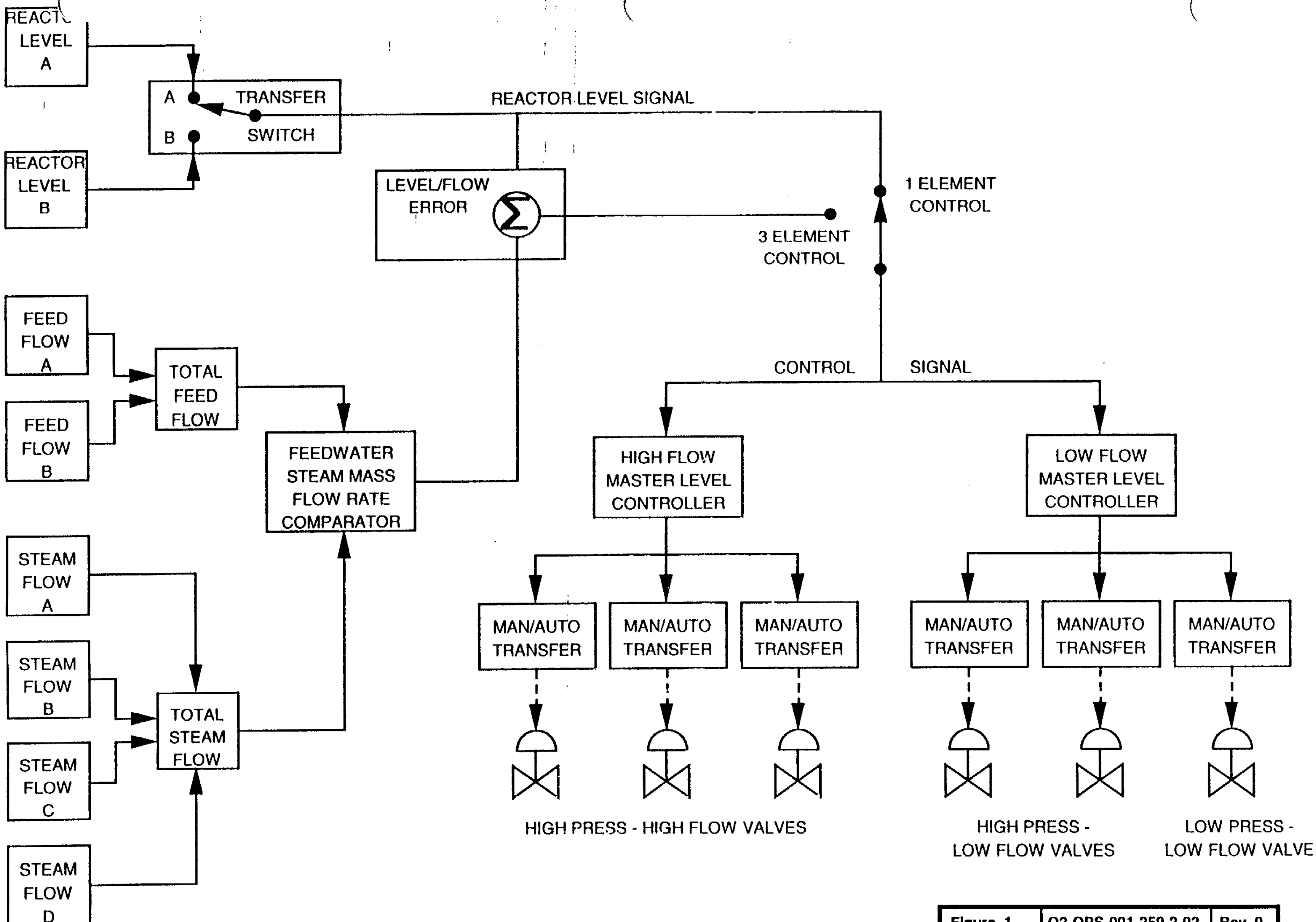


Figure 1	02-OPS-001-259-2-02	Rev 0
TITLE	**Feedwater Control**	

18-3

Question # RO 19 SRO 32

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	1	1
	Group #	2	2
	K/A #	295012 AK2.01	295012 AK2.01
	Importance Rating	3.4	3.5

Knowledge of the interrelations between high drywell temperature and the following: Drywell ventilation.

Proposed Question:

The plant is operating at rated power: Drywell temperature is 140°F and slowly rising.

Which one of the following actions is required to stabilize drywell temperature?

a. Align service water to the drywell unit coolers.

b. Start the standby RPV top head area unit cooler.

c. Align alternate drywell cooling to the drywell unit coolers.

d. Throttle open CCP outlet valves to the DRS unit coolers.

Proposed Answer: b.

Explanation (Justification of Distractors):

a. The additional cooling is to align CCP to the top head area unit cooler and start it. Cannot lineup service water to the DRS unit coolers.

c. The additional cooling is to align CCP to the top head area unit cooler and start it. ADC is only permitted in MODE 4 and MODE 5.

d. The CCP valves are full open and will trip the unit coolers if not full open.

Technical Reference(s): N2-OP-60, H.2.0

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-222-2-01, EO-4b, EO-7d, EO-8
Question Source:	Bank # Modified Bank # Q10383 New
Question History:	Previous NRC Exam Previous Test / Quiz
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis 1
10CFR Part 55 Content:	55.41.7 55.45.8

Comments:

Question # RO 20 SRO 9

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	1	1
	Group #	2	1
	K/A #	295013 AK3.01	295013 AK3.01
	Importance Rating	3.6	3.8

Knowledge of the reasons for the following responses as they apply to HIGH SUPPRESSION POOL TEMPERATURE: suppression pool cooling operation.

Proposed Question:

A steam line break has occurred in the Primary Containment. During the scram **several control rods failed to fully insert**. The following conditions exist:

- RPV Level is 167 inches
- RPV Pressure is 420 psig
- Drywell Pressure is 7.0 psig
- Drywell Temperature is 180°F
- Suppression Chamber Pressure is 2 psig
- Suppression Pool Temperature is 106°F

Which one of the following Residual Heat Removal System lineups is appropriate for these conditions?

a. System “A” and “B” in suppression pool cooling.

b. System “A” in suppression pool cooling with “B” in LPCI.

c. System “A” and “B” in drywell and suppression chamber spray.

d. System “A” in suppression pool cooling and “B” in drywell spray.

Proposed Answer: a. N2-EOP-PC directs starting all available suppression pool cooling.

Explanation (Justification of Distractors):

b. B loop should also be started in SP cooling, RPV makeup is NOT needed.
c. Both loops should be in suppression pool cooling
d. Both loops should be in suppression pool cooling

Technical Reference(s):

N2-EOP-PC
EOP Basis

Proposed references to be provided to applicants during the examination:

EOPs without entry conditions

Learning Objective:	02-OPS-006-344-2-04, EO-2.0	
Question Source:	Bank # Modified Bank # New	 New
Question History:	Previous NRC Exam Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 2
10CFR Part 55 Content:	41.5 / 45.6	

Comments:

Question # RO 21

Examination Outline Cross-Reference	Level	RO
	Tier #	1
	Group #	2
	K/A #	295016 AA1.03
	Importance Rating	3.0

Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT: RPIS

Proposed Question:

A rapidly spreading fire forced evacuation of the control room. During the evacuation it was **NOT** possible to verify **ALL RODS IN**. Which of the following methods is available to determine control rod positions?

a. Demand an OD-7 at the remote computer.

b. Verify all HCU accumulator pressures less than 860 psig.

c. Perform continuity checks at the RPIS termination cabinets.

d. Determine ALL RODS IN at the RWM Computer Display Chassis.

Proposed Answer: a.

Explanation (Justification of Distractors):

b. De-pressurized accumulators do NOT insure rods are in.
c. Termination cabinets are in the control room.
d. Does not exist

Technical Reference(s): N2-SOP-78, Section 4.1

Proposed references to be provided to applicants during the examination:

None

Learning Objective:	02-OPS-006-SOP-2-01	
Question Source:	Bank # Modified Bank # New	 New
Question History:	Previous NRC Exam Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	1
10CFR Part 55 Content:	41.7 / 45.6	

Comments:

Question # RO 22

Examination Outline Cross-Reference	Level	RO
	Tier #	1
	Group #	2
	K/A #	295017 AA2.01
	Importance Rating	2.9

Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: off-site release rate: plant specific.

Proposed Question:

The plant is in Mode 5 unloading the reactor core in preparation for refueling. A tornado strikes the site and several of the panels on the refueling floor are torn free and fall from the building.

Which one of the following describes the type of release and the release path?

a. Monitored release from the secondary containment only.

b. Unmonitored release from the secondary containment only.

c. Monitored release from the primary and secondary containment.

d. Unmonitored release from the primary and secondary containment.

Proposed Answer: d.

Explanation (Justification of Distractors):

a. The release is unmonitored
b. The release is outside the secondary containment
c. The release is unmonitored and outside the secondary containment

Technical Reference(s): N2-OP-79, Rev 07, Section B

Proposed references to be provided to applicants during the examination:

None

Learning Objective:	02-OPS-001-223-2-04, EO-4.0	
Question Source:	Bank # Modified Bank # New	 NEW
Question History:	Previous NRC Exam Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	1
10CFR Part 55 Content:	41.10 / 43.5 / 45.13	

Comments:

Question # RO 23 SRO 33

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	1	1
	Group #	2	2
	K/A #	295018 AK3.07	295018 AK3.07
	Importance Rating	3.1	3.2

Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF CCW: cross connecting with backup systems.

Proposed Question:

During a long term loss of Reactor Building Closed Loop Cooling Water (CCP) which one of the following lists the loads that can be cooled by backup systems to CCP?

a. RHR Pump Seal Coolers and Spent Fuel Pool Cooling

b. Reactor Building Drain Coolers and RHR Motor Coolers

c. Reactor Building Ventilation and CRD Pump Seal Coolers

d. CRD Pump Seal and Oil Coolers and Recirc Pump Motor Coolers

Proposed Answer: a.

Explanation (Justification of Distractors):

b. Drain Cooler cannot be supplied by service water, there are no RHR motor coolers

c. Reactor Bldg is a SW load

d. No backup on the CRD pump, there used to be a backup to the Recirc Pumps but it was abandoned.

Technical Reference(s):

N2-OP-13 Attachment 1

Proposed references to be provided to applicants during the examination:

None

Learning Objective:	02-OPS-001-208-2-0, EO-5.0	
Question Source:	Bank # Modified Bank # New	 New
Question History:	Previous NRC Exam Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	1
10CFR Part 55 Content:	41.5 / 45.6	

Comments:

Question # RO 24 SRO 34

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	1	1
	Group #	2	2
	K/A #	295019 2.4.48	295019 2.4.48
	Importance Rating	3.5	3.8

Ability to interpret control room indications to verify the status and operation of system, and understand how operator actions and directives affect plant and system conditions.

Proposed Question:

The following annunciators are in alarm:

- 851229, INSTR AIR SYSTEM TROUBLE
- 851218, INST AIR RCVR TK 3 PRESS LOW
- 851208, INST AIR RCVR TK 2 PRESS LOW
- 851239, SER AIR SYS 2IAS-AOV171 CLOSED

The Compressor Selector Switch is in position, **CAB**

Which one of the following states the Air Compressors that should be operating for these conditions?

a. C and A

b. C and B

c. A and B

d. C, A, B

Proposed Answer: d. The Lag Compressors starts at 100 psig, the Backup starts at 85 psig. The Inst Air RCVR #2 alarm comes in at 85 psig so all three compressors should be running.

Explanation (Justification of Distractors):

See above

Technical Reference(s): N2-SOP-19, 0, Sect. 2.0

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	02-OPS-001-279-2-00, EO-5.0, 8.0	
Question Source:	Bank # Modified Bank # New	 NEW
Question History:	Previous NRC Exam Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 2
10CFR Part 55 Content:	55.43.5 55.45.12	

Comments:

Question # RO 25

Examination Outline Cross-Reference	Level	RO
	Tier #	1
	Group #	2
	K/A #	295020 2.4.11
	Importance Rating	3.4

Knowledge of abnormal condition procedures.

Proposed Question:

A relay failure caused a Division I Group 8 isolation. Which one of the following Special Operating Procedures (SOP) is required to be entered?

a. N2-SOP-11, Loss of Service Water.

b. N2-SOP-60, Loss of Drywell Cooling.

c. N2-SOP-13, Total Loss of CCP System.

d. N2-SOP-30, Control Rod Drive Failures.

Proposed Answer: b.

Explanation (Justification of Distractors):

a. Service water is not affected by group 8.
c. CCP is not affected by group 8.
d. RDS not affected by group 8.

Technical Reference(s): N2-SOP-60, Rev 01, Section 4.4

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-006-SOP-2-01-29, TO-12, EO-3	
Question Source:	Bank # Modified Bank # New	 New
Question History:	Previous NRC Exam Previous Test / Quiz	New New
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	1
10CFR Part 55 Content:	55.41.10 55.43.5 55.45.13	

Comments:

Question # RO 26

Examination Outline Cross-Reference	Level	RO
	Tier #	1
	Group #	2
	K/A #	295022 AA2.02
	Importance Rating	3.3

Ability to determine and/or interpret the following as they apply to LOSS OF CRD PUMPS: CRD system status.

Proposed Question:

The plant is operating at 100% power. Several annunciators have been received in the last few minutes, including:

- 603309, CRD PUMP 1A SUCTION PRESS LOW
- 603308, CRD PUMP 1A/1B AUTO TRIP
- 603446, CRD PUMP DISCH HEADER PRESSURE LOW
- 603311, CRD CHARGING WTR PRESSURE LOW
- 603441, ROD DRIVE ACCUMULATOR TROUBLE

A check of the full core display on P603 indicates 6 (six) amber accumulator lights for fully withdrawn control rods are **ON**.

Which one of the following actions is required **FIRST**?

a. Start the standby CRD pump then restore the CRD system.

b. Reduce recirculation flow to minimum and scram the reactor.

c. Dispatch an operator to the accumulators to determine pressure.

d. Declare associated control rods inoperable and enter Technical Specifications LCO.

Proposed Answer: c. Accumulator pressure must be locally verified >940 psig.

Explanation (Justification of Distractors):

a. Can't be started until suction filter is swapped.
b. Not required until CRD accumulator status is determined.
d. Not required until CRD accumulator status is determined.

Technical Reference(s):

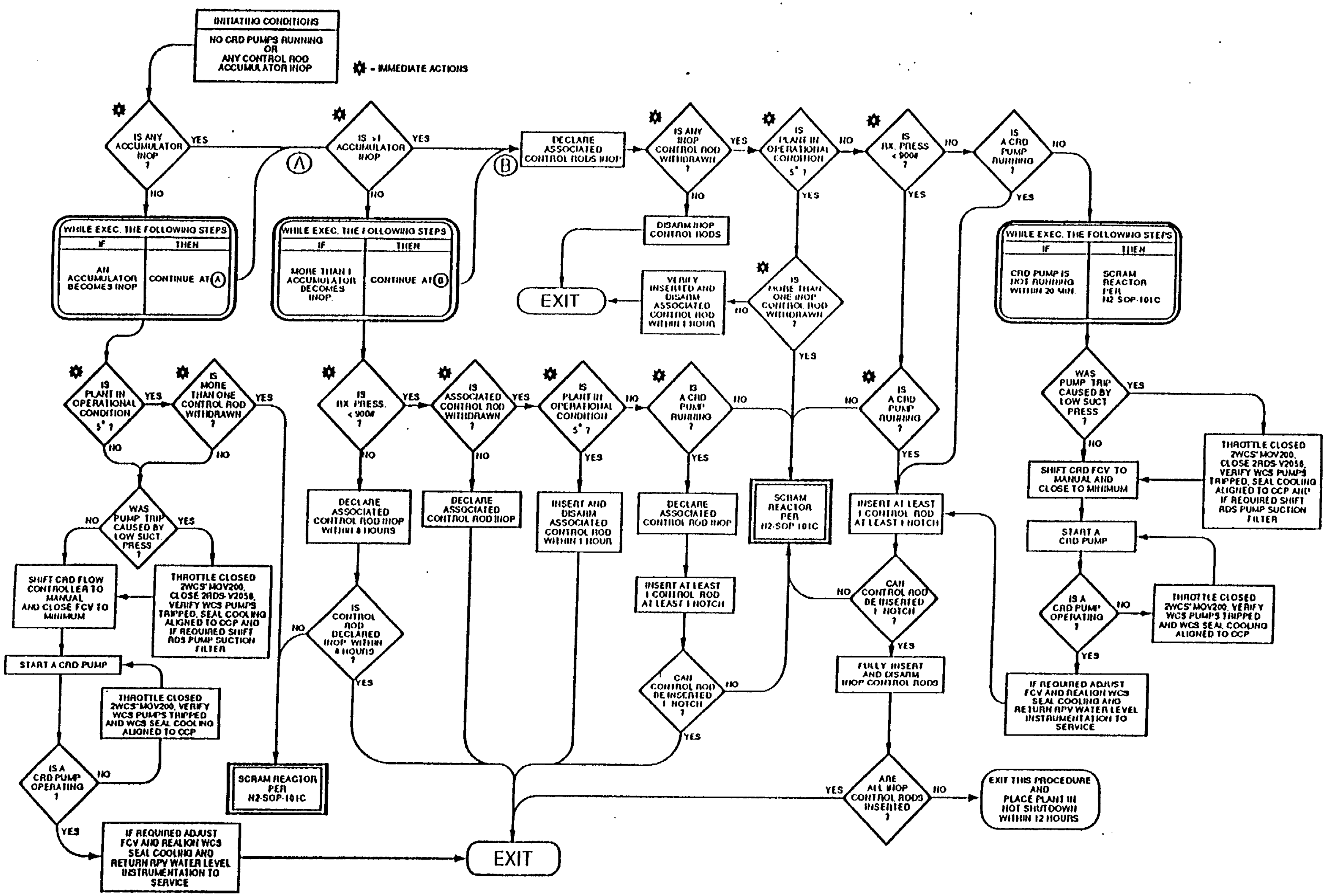
N2-SOP-30, T.S. 34.1.3.5

Proposed references to be provided to applicants during the examination:

Attachment 1 of N2-SOP-30, CONTROL ROD DRIVE FAILURES, FLOW DIAGRAM

Learning Objective:	02-OPS-001-201-2-01, EO-8.0	
Question Source:	Bank #	
	Modified Bank #	
	New	NEW
Question History:	Previous NRC Exam	
	Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge	
	Comprehension or Analysis	2
10CFR Part 55 Content:	41.10 / 43.5 / 45.13	

Comments:


INITIATING CONDITIONS
NO CRD PUMPS RUNNING OR ANY CONTROL ROD ACCUMULATOR INOP
- IMMEDIATE ACTIONS
IS ANY ACCUMULATOR INOP ?
IS >1 ACCUMULATOR INOP
DECLARE ASSOCIATED CONTROL RODS INOP
IS ANY INOP CONTROL ROD WITHDRAWN ?
IS PLANT IN OPERATIONAL CONDITION 5 ?
IS RX. PRESS < 900#
IS A CRD PUMP RUNNING ?
WHILE EXEC. THE FOLLOWING STEPS
IF
THEN
AN ACCUMULATOR BECOMES INOP
CONTINUE AT A
MORE THAN 1 ACCUMULATOR BECOMES INOP.
CONTINUE AT B
DISARM INOP CONTROL RODS
EXIT
VERIFY INSERTED AND DISARM ASSOCIATED CONTROL ROD WITHIN 1 HOUR
IS MORE THAN ONE INOP CONTROL ROD WITHDRAWN ?
CRD PUMP IS NOT RUNNING WITHIN 20 MIN.
SCRAM REACTOR PER N2-SOP-101C
IS PLANT IN OPERATIONAL CONDITION 5 ?
IS MORE THAN ONE CONTROL ROD WITHDRAWN ?
IS RX. PRESS. < 900# ?
IS ASSOCIATED CONTROL ROD WITHDRAWN ?
IS PLANT IN OPERATIONAL CONDITION 5 ?
IS A CRD PUMP RUNNING ?
IS A CRD PUMP RUNNING ?
WAS PUMP TRIP CAUSED BY LOW SUCT PRESS ?
DECLARE ASSOCIATED CONTROL ROD INOP WITHIN 8 HOURS
DECLARE ASSOCIATED CONTROL ROD INOP
INSERT AND DISARM ASSOCIATED CONTROL ROD WITHIN 1 HOUR
DECLARE ASSOCIATED CONTROL ROD INOP
SCRAM REACTOR PER N2-SOP-101C
INSERT AT LEAST 1 CONTROL ROD AT LEAST 1 NOTCH
SHIFT CRD FCV TO MANUAL AND CLOSE TO MINIMUM
THROTTLE CLOSED 2WCS*MOV200, CLOSE 2RDS-V2058, VERIFY WCS PUMPS TRIPPED, SEAL COOLING ALIGNED TO CCP AND IF REQUIRED SHIFT RDS PUMP SUCTION FILTER
START A CRD PUMP
WAS PUMP TRIP CAUSED BY LOW SUCT PRESS ?
SHIFT CRD FLOW CONTROLLER TO MANUAL AND CLOSE FCV TO MINIMUM
THROTTLE CLOSED 2WCS*MOV200, CLOSE 2RDS-V2058, VERIFY WCS PUMPS TRIPPED, SEAL COOLING ALIGNED TO CCP AND IF REQUIRED SHIFT RDS PUMP SUCTION FILTER
START A CRD PUMP
THROTTLE CLOSED 2WCS*MOV200, VERIFY WCS PUMPS TRIPPED AND WCS SEAL COOLING ALIGNED TO CCP
IS A CRD PUMP OPERATING ?
SCRAM REACTOR PER N2-SOP-101C
IS CONTROL ROD DECLARED INOP WITHIN 8 HOURS ?
INSERT AT LEAST 1 CONTROL ROD AT LEAST 1 NOTCH
CAN CONTROL ROD BE INSERTED 1 NOTCH ?
CAN CONTROL ROD BE INSERTED 1 NOTCH ?
FULLY INSERT AND DISARM INOP CONTROL RODS
IS A CRD PUMP OPERATING ?
THROTTLE CLOSED 2WCS*MOV200, VERIFY WCS PUMPS TRIPPED AND WCS SEAL COOLING ALIGNED TO CCP
IF REQUIRED ADJUST FCV AND REALIGN WCS SEAL COOLING AND RETURN RPV WATER LEVEL INSTRUMENTATION TO SERVICE
ARE ALL INOP CONTROL RODS INSERTED ?
EXIT THIS PROCEDURE AND PLACE PLANT IN HOT SHUTDOWN WITHIN 12 HOURS
IF REQUIRED ADJUST FCV AND REALIGN WCS SEAL COOLING AND RETURN RPV WATER LEVEL INSTRUMENTATION TO SERVICE
EXIT
YES
NO

26-3

Question # RO 27 SRO 37

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	1	1
	Group #	2	2
	K/A #	295028 EK1.01	295028 EK1.01
	Importance Rating	3.5	3.7

Knowledge of the operational implications of the following concepts as they apply to high drywell temperature: reactor water level measurement.

Proposed Question:

A leak into the primary containment atmosphere has developed. As drywell temperatures rise, which one of the following describes the effect on the indicated reactor water level compared to the actual reactor water level?

a. Indicated level is lower on all level instruments.

b. Indicated level is higher on all level instruments.

c. Indicated level is lower on narrow range instruments and higher on all other instruments.

d. Indicated level is lower on wide range instruments and higher on all other instruments.

Proposed Answer: b.

Explanation (Justification of Distractors):

High drywell temperature effect will be the same for all RPV level instruments. As temperature rises, the indicated level will be higher than the actual reactor water level.

Technical Reference(s): NMP2 EOP Technical Bases for RPV Control

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-216-2-01, EO-5	
Question Source:	Bank #	Q8328
	Modified Bank #	
	New	
Question History:	Previous NRC Exam	
	Previous Test / Quiz	(Week 10 exam)
Question Cognitive Level:	Memory of Fundamental Knowledge	2
	Comprehension or Analysis	
10CFR Part 55 Content:	55.41.5	

Comments:

Question # RO 28 SRO 21

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	1	1
	Group #	2	2
	K/A #	295030 EK3.06	295030 EK3.06
	Importance Rating	3.6	3.8

Knowledge of the reasons for the following responses as they apply to low suppression pool water level: Reactor scram.

Proposed Question:

Which one of the following is the reason for requiring a plant shutdown and RPV blowdown if suppression pool water level CANNOT be maintained above elevation 192 feet?

a. Protect the primary containment from over pressurization if a LOCA occurs.

b. Prevent the loss of HPCS and RCIC as injection sources due to loss of NPSH.

c. Protect the primary containment from excessive upward pressure on the drywell floor if drywell sprays are initiated.

d. Prevent exceeding suppression chamber downcomer design differential pressures if suppression chamber sprays are initiated.

Proposed Answer: a.

A loss of pressure suppression function of the suppression chamber, direct pressurization of the suppression chamber air space, and insufficient NPSH for pumps taking a suction on the suppression pool.

Explanation (Justification of Distractors):

d. This would result from a high suppression pool water level.
b. RCIC and HPCS suction will not be lost. Condensate storage tank is the normal injection source.
c. This would result from drywell spray with a high suppression pool water level.

Technical Reference(s): NMP2-EOP-Basis Document, Section E.

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-006-344-2-04, EO-3	
Question Source:	Bank # Modified Bank # New	 New
Question History:	Previous NRC Exam Previous Test / Quiz	New New
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	1
10CFR Part 55 Content:	55.41.5 55.45.6	

Comments:

28-2

Question # RO 29

Examination Outline Cross-Reference	Level	RO
	Tier #	1
	Group #	2
	K/A #	295033 EK3.04
	Importance Rating	4.0

Knowledge of the reasons for the following responses as they apply HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS: personnel evacuation.

Proposed Question:

While operating at 89% power the following events occur:

- 851244, REACTOR BLDG AREA RADN MON ACTIVATED.
- The annunciator is confirmed to be caused by a **red** high alarm on 2RMS-RE105, TIP EQUIP AREA.
- It is confirmed there are **NO** known activities being performed in the TIP area.

The CSO enters EPIP-EPP-21, RADIATION EMERGENCIES, and directs you to announce a radiation emergency area evacuation of the TIP area.

Which one of the following is the basis for this announcement?

a. Prevents the spread of contamination.

b. Directs Radiation Protection to the TIP area.

c. Initiates an accountability of personnel in the area.

d. Lowers radiation exposures to personnel in the area.

Proposed Answer: d.

Explanation (Justification of Distractors):
a. Not used for contamination control.
b. Directs RP to the Control Room
c. There is NO accountability for this event

Technical Reference(s):

N2-ARP-01
EOP BASIS
EPIP-EPP-21

Proposed references to be provided to applicants during the examination:

None

Learning Objective:

Question Source:	Bank #	
	Modified Bank #	
	New	NEW
Question History:	Previous NRC Exam	
	Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge	
	Comprehension or Analysis	1
10CFR Part 55 Content:	41.5 / 45.6	

Comments:

Question # RO 30 SRO 40

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	1	1
	Group #	2	2
	K/A #	295034 2.4.17	295034 2.4.17
	Importance Rating	3.1	3.8
Knowledge of EOP terms and definitions.			

Proposed Question:

Which one of the following defines the EOP term, MAXIMUM SAFE VALUE?

Exceeding the value...

a. requires entry into the E-plan and declaration of a Site Area Emergency.

b. will directly threaten equipment important to safety or personnel safety.

c. is symptomatic of offnormal conditions that could degrade into an emergency.

d. indicates a primary system discharging into the reactor building and requires emergency depressurization.

Proposed Answer: b.

Explanation (Justification of Distractors):

a. Exceeding the value is NOT a trigger for entry into the E-plan. E-plan entry is based on effluent monitor levels. This would be true if a primary system is discharging into the reactor building and the MAX SAFE VALUE is exceeded.

c. This is from the definition of Maximum Normal, which is no longer used.

d. Other abnormal and emergency conditions can be the source. Emergency depressurization is required only if the source cannot be isolated and two (2) or more areas are above the MAX SAFE VALUE for the same parameter.

Technical Reference(s): NMP2-EOP-Basis Document, Section E.
EPIP-EPP-02, Attachment 1, Rev 8 (Emergency Action Level Matrix / Unit 2)

Proposed references to be provided to applicants during the examination:

None.

Learning Objective: O2-OPS-006-344-2-08, EO-3

Question Source:	Bank #	
	Modified Bank #	
	New	New
Question History:	Previous NRC Exam	New
	Previous Test / Quiz	New
Question Cognitive Level:	Memory of Fundamental Knowledge	1
	Comprehension or Analysis	
10CFR Part 55 Content:	55.41.10	
	55.45.13	

Comments:

30-2

Question # RO 31

Examination Outline Cross-Reference	Level	RO
	Tier #	1
	Group #	2
	K/A #	295038 EK2.05
	Importance Rating	3.7

† Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: Site emergency plan.

Proposed Question:

Which one of the following events would require an emergency event classification?

a. A fire is reported in the site warehouse.

b. Loss of Line 5 <u>or</u> 6 with a loss of the associated EDG.

c. A radioactivity release with a site boundary TEDE of 15 mr/hr.

d. A reactor scram where RPV water level lowers to 100 inches.

Proposed Answer: c.

Explanation (Justification of Distractors):

This off-site release rate would require an entry into EPIP-EPP-02.

Technical Reference(s): EPIP-EPP-02

Proposed references to be provided to applicants during the examination:

EPIP-EPP-02, Attachment 1

Learning Objective:	02-OPS-006-344-2-12, EO-1.0	
Question Source:	Bank #	
	Modified Bank #	
	New	NEW
Question History:	Previous NRC Exam	
	Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge	1
	Comprehension or Analysis	
10CFR Part 55 Content:	41.7 / 45.8	

Comments:

Question # RO 32

Examination Outline Cross-Reference	Level	RO
	Tier #	1
	Group #	2
	K/A #	600000 AA1.08
	Importance Rating	2.7

Ability to operate and/or monitor the following as they apply to plant fire on site: Fire fighting equipment used on each class of fire.

Proposed Question:

Which one of the following describes the response of the fire protection system if the fire detection system senses a fire in zone 333XL, DIV 1 SWGR ROOM?

a. Deluge system actuated and the fixed foam system pump is operating.

b. Fire computer prints an alarm tape and the motor-driven fire pump is running.

c. Local horn and light actuate, and after 30 seconds carbon dioxide is discharged.

d. Local alarm and strobe light actuate after halon flow is detected in the zone discharge line.

Proposed Answer: c.

Explanation (Justification of Distractors):

a. 333XL indicates that this zone uses CO2 – indicated by the "L". Foam would not be used for an electrical fire.

b. 333XL indicates that this zone uses CO2 – indicated by the "L". Fire protection water would not be used for an electrical fire.

d. 333XL indicates that this zone uses CO2 – indicated by the "L". Halon is not used in this area. CO2 has a pre-discharge sequence before discharge but halon does not. The local alarm and light for halon are actuated when the zone discharge line pressure switch detects halon flow.

Technical Reference(s):	N2-OP-47, Rev 04, Section B N2-OP-45, Rev 05, Section B

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-286-2-01, EO-4b, EO-4c, EO-4d	
Question Source:	Bank # Modified Bank # New	 New
Question History:	Previous NRC Exam Previous Test / Quiz	New New
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	1
10CFR Part 55 Content:	55.41.7	

Comments:

32-2

Question # RO 33 SRO 36

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	1	1
	Group #	3	2
	K/A #	295021 AK2.04	295021 AK2.04
	Importance Rating	3.0	3.1

Knowledge of the interrelations between LOSS OF SHUTDOWN COOLING and the following: component cooling water systems: plant specific.

Proposed Question:

The plant is making preparations to startup following refueling outage. The following conditions exist:

- The Residual Heat Removal system is **NOT** available
- The main condenser is **NOT** available
- The reactor has been shutdown for 10 weeks

Because of the low core decay heat load that exists, the decision is made to use Alternate Decay Heat Removal. Which one of the following systems will be used in this lineup?

a. Safety Relief Valves

b. Condensate/Feedwater

c. Main Steam Line Drains

d. Reactor Building Closed Loop Cooling Water

Proposed Answer: d. Used to cool the Non-Regen H/X when WCS is lined up to recirculate reactor coolant.

Explanation (Justification of Distractors):

a. Cannot be used without RHR
b. Cannot be used without the condenser
c. Cannot be used without the condenser

Technical Reference(s):	N2-SOP-31, Sect. 4.2 N2-OP-37, Sect. H.5.0

Proposed references to be provided to applicants during the examination:

None

Learning Objective:	02-OPS-001-201-2-01, EO-3.00		
Question Source:	Bank # Modified Bank # New	 NEW	
Question History:	Previous NRC Exam Previous Test / Quiz		
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis		 2
10CFR Part 55 Content:	41.7./.45.8		

Comments:

Question # RO 34 SRO 15

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	1	1
	Group #	2	1
	K/A #	295023 AA1.07	295023 AA1.07
	Importance Rating	3.6	3.6

Ability to operate and monitor the following as they apply to REFUELING ACCIDENTS: Standby Gas Treatment/FRVS

Proposed Question:

The plant is in a refueling outage when a design bases DROPPED FUEL ASSEMBLY ACCIDENT occurs. Standby Gas Treatment (GTS) Train "A" is maintaining Secondary Containment integrity.

Which one of the following describes the consequence of this accident on GTS Train "A" operation?

a. GTS Fan breaker trips.

b. Clogging of the HEPA filter.

c. High charcoal adsorber temperatures.

d. Moisture builds up in the filters and adsorbers.

Proposed Answer: c.

Explanation (Justification of Distractors):

a. GTS Fan will not be effected by the iodine.
b. The HEPA filter will not be effected by the gas
d. Moisture does not reach the filters or adsorbers

Technical Reference(s): N2-OP-61B, Sect B

Proposed references to be provided to applicants during the examination:

None

Learning Objective:	02-OPS-001-261-2-01, EO-3.0	
Question Source:	Bank #	
	Modified Bank #	
	New	NEW
Question History:	Previous NRC Exam	
	Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge	
	Comprehension or Analysis	2
10CFR Part 55 Content:	55.41 7 / 45.6	

Comments:

Question # RO 35 SRO 41

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	1	1
	Group #	3	2
	K/A #	295035 EK3.02	295035 EK3.02
	Importance Rating	3.3	3.5

Knowledge of the reasons for the following responses as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: secondary containment ventilation response.

Proposed Question:

During full power operation a sudden cold spell causes Reactor Building Differential Pressure to lower from –0.47 in WG to –0.35 in WG. Which one of the following actions is required to restore Reactor Building Differential Pressure to the same value that existed before the cold spell?

a. Throttle closed Manual Supply Damper 2HVR-DMPV72.

b. Secure one of the Reactor Building Supply Fans, FN 1A(B,C).

c. Start a second Above Refueling Floor Exhaust Fan, FN 5A(B).

d. Manually close Vent Supply Air Recirc Dampers 2HVR-MOD17A/B

Proposed Answer: a. Closing DMP72 allow less air into the reactor building making it more negative.

Explanation (Justification of Distractors):

b. Interlocks prevent stopping a supply fan in this lineup.
c. Interlocks prevent starting a second fan.
d. Closing the dampers will make the Rx. Bldg more positive

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Technical Reference(s):	N2-OP-52, Sect. F.1.0	

Proposed references to be provided to applicants during the examination:

None

Learning Objective:	02-OPS-001-288-2-03, EO-3.0, 6.0	
Question Source:	Bank #	
	Modified Bank #	
	New	NEW
Question History:	Previous NRC Exam	
	Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge	
	Comprehension or Analysis	1
10CFR Part 55 Content:	41.5./.45.6	

Comments:

Question # RO 36 SRO 42

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	1	1
	Group #	3	2
	K/A #	295036 EA2.03	295036 EA2.03
	Importance Rating	3.4	3.8

Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: Cause of the high water level.

Proposed Question:

Following a LOCA, the following plant conditions exist:

- CRD is maximized for RPV injection
- RHR loops "A" and "B" are in suppression pool cooling
- SFC is maintaining fuel pool temperature
- WCS is being used for RPV pressure control

If a Reactor Building sump reaches the **High-High level** setpoint and cannot be restored and maintained below the High-High level, which one of the following systems should be isolated.

a. CRD

b. LPCI

c. SFC

d. WCS

Proposed Answer: c.

Explanation (Justification of Distractors):

a. CRD is needed for EOP actions
b. LPCI is needed for EOP actions
d. WCS is needed for EOP actions

Technical Reference(s): N2-EOP-BASES, SECT F1

Proposed references to be provided to applicants during the examination:

EOPs without entry conditions

Learning Objective:	O2-OPS-006-344-2-08, 0, EO-2.0	
Question Source:	Bank # Modified Bank # New	 NEW
Question History:	Previous NRC Exam Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 2
10CFR Part 55 Content:	41.10, 43.5, 45.13	

Comments:

Question # RO 37 SRO 67

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	2	2
	Group #	1	2
	K/A #	201001 K2.05	201001 K2.05
	Importance Rating	4.5	4.5

Knowledge of electrical power supplies to the following: Alternate rod insertion valve solenoids: Plant-Specific.

Proposed Question:

Which one of the following statements describes how a total loss of power from Div. I 125 VDC will effect the automatic initiation of Alternate Rod Insertion (ARI) during an RRCS initiation?

	Div I Actuates	Div II Actuates	Number of ARI valves that open
a.	No	Yes	4
b.	Yes	No	8
c.	Yes	Yes	4
d.	Yes	Yes	8

Proposed Answer: a. ARI valves are energized to Open and the Logic is energized to actuate. Loss of power will prevent activation of Div I and failure of it's four valves to Open.

Explanation (Justification of Distractors):

See explanation in proposed answer

Technical Reference(s):

Logic from 02-OPS-001-294-2-08, also N2-OP-36B, Sect. B. and Attachment 1

Proposed references to be provided to applicants during the examination:

None

Learning Objective: 02-OPS-001-294-2-08, EO-3.0, 5.0

Question Source:	Bank # Modified Bank # New	 NEW
Question History:	Previous NRC Exam Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 2
10CFR Part 55 Content:	41.7	

Comments:

Question # RO 38

Examination Outline Cross-Reference	Level	RO
	Tier #	2
	Group #	1
	K/A #	201001 A1.03
	Importance Rating	2.9

Ability to predict and/or monitor changes in parameters associated with operating the CONTROL ROD DRIVE HYDRAULIC SYSTEM controls including: CRD system flow.

Proposed Question:

The plant is operating at 60% power with the Control Rod Drive (CRD) Flow Controller in **AUTO** set for 63 gpm. Which one of the following describes how the CRD Flow Control Valve responds to a reactor scram?

a. Opens then partially closes to control flow as the SDV is pressurized.

b. Opens then partially closes to control flow to recharge the accumulators.

c. Closes then partially opens to control flow when the scram is reset.

d. Closes then partially opens to control flow when control rods reach position "00".

Proposed Answer: c.
The flow element for the FCV is located upstream of the charging water header and the FCV. On a scram flow through the charging header rises and the FCV closes on sensed high flow. When the scram is reset the FCV will open as the flow directly into the reactor is stopped and the accumulators recharge.

Explanation (Justification of Distractors):

a. See explanation in proposed answer
b. FCV will not throttle closed until the scram is reset, which allows the accumulators to recharge.
d. See explanation in proposed answer.

Technical Reference(s): 02-OPS-001-201-2-01, Figure 1

Proposed references to be provided to applicants during the examination:

None

Learning Objective: 02-OPS-001-201-2-01, EO-3.00

Question Source:	Bank # Modified Bank # New	 NEW
Question History:	Previous NRC Exam Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 2
10CFR Part 55 Content:	55.41.5 55.45.5	

Comments:

Question # RO 39 SRO 68

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	2	2
	Group #	1	2
	K/A #	201002 A2.04	201002 A2.04
	Importance Rating	3.2	3.1

Ability to (a) predict the impacts of the following on the reactor manual control system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Control rod block.

Proposed Question:

A reactor startup is in progress with thermal power below the LPSP. The RWM and RSCS are both OPERABLE.

The next step requires that a control rod be moved from position 12 (Bank insert limit) to position 24 (Bank withdraw limit). When the control rod is positioned, its final position is 26 because of a double-notch.

Regarding ONLY the RWM, which one of the following describes the actions necessary to return the control rod to its bank withdraw limit (position 24)?

a. Bypass the RWM, then return the control rod to position 24.

b. Using the insert pushbutton, returns the control rod to position 24.

c. Bypass the control rod in the RWM, then return it to position 24.

d. Using the RWM enter a substitute control rod position, then return it to position 24.

Proposed Answer: b.

Explanation (Justification of Distractors):
The RWM will enforce a withdraw block and indicate an insert error. No insert or select block is actuated. The control rod can be inserted.

Technical Reference(s): N2-OP-95A, Rev 04, Section B

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-201-2-02, EO-4b,4d	
Question Source:	Bank #	Q15749
	Modified Bank #	
	New	
Question History:	Previous NRC Exam	
	Previous Test / Quiz	Week 11 exam
Question Cognitive Level:	Memory of Fundamental Knowledge	
	Comprehension or Analysis	2
10CFR Part 55 Content:	55.41.5	
	55.45.6	

Comments:

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Question # RO 40 SRO 44

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	2	2
	Group #	1	1
	K/A #	202002 A3.01	202002 A3.01
	Importance Rating	3.6	3.4

Ability to monitor automatic operations of the recirculation flow control system including: Flow control valve operation.

Proposed Question:

The plant is operating at 100% power. A shutdown of the "B" Recirculation Flow Control Valve Hydraulic Power Unit occurs. The following conditions currently exist:

- Total Core Flow 107.5 mlbs/hr
- Jet Pump Loop "A" Flow 53.5 mlbs/hr
- Jet Pump Loop "B" Flow 54.0 mlbs/hr
- FCV "A" 83% Open
- FCV "B" 84% Open

Which one of the following limits will be challenged if **NO** operator action is taken?

a. Rated core flow

b. Rated reactor power

c. Recirculation pump amperes

d. Jet pump loop flow mismatch

Proposed Answer: d.
"B" FCV will drift CLOSED causing a mismatch between A and B Recirculation loop Jet Pump flows which are required to be within 5% by T.S. 3.4.1.3.

Explanation (Justification of Distractors):
a. Core Flow may lower but it is not the primary concern under these conditions.
b. Rated power may lower due to the B FCV closing, but under these conditions it is not the primary concern.
c. Not a concern under these conditions.

Technical Reference(s): N2-OP-29, REACTOR RECIRCULATION SYSTEM, SECT D.28.0 and T.S. 3.4.1.3

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-202-2-02, EO-4c	
Question Source:	Bank # Modified Bank # New	 NEW
Question History:	Previous NRC Exam Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 2
10CFR Part 55 Content:	55.41.7 55.45.7	

Comments:

Question # RO 41 SRO 45

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	2	2
	Group #	1	1
	K/A #	203000 K5.01	203000 K5.01
	Importance Rating	2.7	2.9

Knowledge of the operational implications of the following concepts as they apply to RHR/LPCI INJECTION MODE: testable check valve operation.

Proposed Question:

Which one of the following methods is used to **confirm** RPV injection flow during an automatic LPCI injection using RHR "A"?

a. Injection Valve RHS*MOV24A opens.

b. Minimum Flow Valve RHS*MOV4A opens.

c. Testable Check Valve RHS*AOV16A opens.

d. RPV pressure lowers to within 130 psid of RHR pressure.

Proposed Answer: c.

Explanation (Justification of Distractors):

a. No indication of flow
b. This indicates flow but not where it's going
d. Does not indicate flow or flowpath

Technical Reference(s): N2-OP-31, Sect F.2.0

Proposed references to be provided to applicants during the examination:

None

Learning Objective:	02-OPS-001-205-2-00, EO-3.0, 9.0
Question Source:	Bank # Modified Bank # New NEW
Question History:	Previous NRC Exam Previous Test / Quiz
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis 1
10CFR Part 55 Content:	55.41.7 55.45.7

Comments:

Question # RO 42 SRO 46

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

Knowledge of the physical connections and/or cause effect relationships between LOW PRESSURE CORE SPRAY and the following: Suppression Pool

Proposed Question:

During a refueling outage involving Core Shroud work, Low Pressure Core Spray (CSL) is lined up with its suction from Residual Heat Removal (RHS). The CSL Manual Suction Isolation Valve, 2CLS*V121, is verified closed and all other CSL valves are in their **Standby Lineup**.

While in this lineup RPV level lowers while suppression pool level rises. Which one of the following is the cause?

a. RHS pump suction valve is open.

b. CSL minimum flow valve is open.

c. CSL flow back through the RHS pump.

d. Jockey pump flow to the CSL test return valve.

Proposed Answer: b.

Explanation (Justification of Distractors):

a. This valve could be open, no flow from the vessel will occur.
c. The suctions are cross-connected, there is no CSL flow into RHS.
d. Test return valve is normally closed.

Technical Reference(s): N2-OP-32, Sect D.11.0

Proposed references to be provided to applicants during the examination:

None

Learning Objective:	02-OPS-001-209-2-00, EO-6.0	
Question Source:	Bank # Modified Bank # New	 NEW
Question History:	Previous NRC Exam Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 2
10CFR Part 55 Content:	55.41.7 / 45.7 / 45.8	

Comments:

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Question # RO 43 SRO 47

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	2	2
	Group #	1	1
	K/A #	209001 K1.09	209001 K1.09
	Importance Rating	3.2	3.4

Knowledge of the physical connections and/or cause-effect relationships between low pressure core spray and the following: Nuclear boiler instrumentation.

Proposed Question:

The plant is operating at 60% power. A high drywell pressure causes a reactor scram. Plant conditions are as follow:

- RPV level 165 inches rising slowly
- RPV pressure 1005 psig and stable
- Drywell pressure 2.3 psig
- Turb. bypass valves available

Which one of the following describes the Low Pressure Core Spray (CSL) system status?

a. Pump shutdown in the standby lineup.

b. Pump shutdown with injection valve open.

c. Pump running with the injection valve open.

d. Pump running with the injection valve closed.

Proposed Answer: d.

Explanation (Justification of Distractors):

a. LPCS will be running on minimum flow

b. LPCS auto starts on high drywell pressure at 1.68 psig.

c. Injection valve remains closed until RCS pressure lowers to within 88 psig of the LPCS discharge pressure.

Technical Reference(s): N2-OP-32, Rev 06, Section B, Section F.2.0

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-209-2-00, EO-4c	
Question Source:	Bank #	Q13325
	Modified Bank #	
	New	
Question History:	Previous NRC Exam	
	Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge	
	Comprehension or Analysis	2
10CFR Part 55 Content:	55.41.7, 55.41.8	
	55.45.7	

Comments:

Question # RO 44 SRO 48

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	2	2
	Group #	1	1
	K/A #	209002 A1.03	209002 A1.03
	Importance Rating		

Ability to predict and/or monitor changes in the parameters associated with operating the high pressure core spray system (HPCS) controls including: Reactor water level.

Proposed Question:

Following a loss of feedwater, High Pressure Core Spray (HPCS) initiated on low reactor water level. When reactor water level is at 190 inches indicated, the operator closes CSH*MOV107, PMP 1 INJECTION VLV, and an AMBER light above the control switch lights.

Subsequently, when RPV water level lowers to 140 inches CSH*MOV107, PMP 1 INJECTION VLV, control switch is placed to OPEN.

Assuming no other HPCS controls are operated, which one of the following describes the reactor water level response?

a. Rises to 202.3 inches and then lowers.

b. Lowers to 108.8 inches and then rises.

c. Rises to 202.3 inches and continues to rise.

d. Lowers to 108.8 inches and continues to lower.

Proposed Answer: a.

Explanation (Justification of Distractors):

b. If level lowered to 108.8 inches before the injection valve was manually opened, HPCS would automatically inject. However, HPCS injects when the injection valve is opened at 140 inches.

c. Although the injection valve is manually overridden, it will still automatically close on high RPV water level.

d. HPCS injects when the injection valve is opened.

Technical Reference(s): N2-OP-33,

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-202-2-02, EO-4c	
Question Source:	Bank # Modified Bank # New	New
Question History:	Previous NRC Exam Previous Test / Quiz	New New
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	2
10CFR Part 55 Content:	55.41.5 55.45.5	

Comments:

Question # RO 45 SRO 49

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	2	2
	Group #	1	1
	K/A #	211000 K4.03	211000 K4.03
	Importance Rating	3.8	3.9

Knowledge of SLC System design feature(s) and/or interlocks which provide for keeping sodium pentaborate in solution.

Proposed Question:

As temperature in the Reactor Building lowers which one of the following design features prevent sodium pentaborate from coming out of solution?

a. SLC tank heater and piping heat tracing.

b. Reactor Building heating and ventilation system.

c. SLC area heaters and SLC air supply sparger.

d. SLC air supply sparger and Reactor Building heating and ventilation system.

Proposed Answer: a. Heater and heat tracing maintain temperature between 75°F and 85°F.

Explanation (Justification of Distractors):

b. HVAC is not designed or designated to maintain SLC Temperatures
c. Area heaters are not designed to maintain SLC Temperatures
d. Sparger is for mixing only

Technical Reference(s): N2-OP-36A, STANDBY LIQUID CONTROL SYSTEM, SECT B

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-211-2-00, EO-4c	
Question Source:	Bank #	
	Modified Bank #	
	New	NEW
Question History:	Previous NRC Exam	
	Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge	1
	Comprehension or Analysis	
10CFR Part 55 Content:	55.41.7	

Comments:

Question # RO 46 SRO 50

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	2	2
	Group #	1	1
	K/A #	212000 K4.07	212000 K4.07
	Importance Rating	4.1	4.1
Knowledge of REACTOR PROTECTION SYSTEM design feature(s) and/or interlocks which provide for the following: Manual system activation trip			

Proposed Question:

During a reactor startup Intermediate Range Monitors (IRM) "C" and "G" become inoperative without causing a reactor scram. It has been decided to continue the startup while I&C makes repairs. You are directed to MANUALLY TRIP the associated RPS trip channel.

Which one of the following methods is used?

a. Arm and depress the A2 Manual Scram Pushbutton.

b. Arm and depress the B2 Manual Scram Pushbutton.

c. Place "C" or "G" IRM drawer mode switch in Standby.

d. Place both "C" and "G" IRM drawers mode switches in Standby.

Proposed Answer: a.

Explanation (Justification of Distractors):
b. This is the wrong channel
c. This is not in accordance with N2-OP-97
d. This is not in accordance with N2-OP-97

Technical Reference(s): N2-OP-97, Sect H.1.0

Proposed references to be provided to applicants during the examination:

None

Learning Objective:	02-OPS-001-212-2-00, EO-3.0, 5.0
Question Source:	Bank # Modified Bank # New NEW
Question History:	Previous NRC Exam Previous Test / Quiz
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis 2
10CFR Part 55 Content:	55 41.7

Comments:

46-2

Question # RO 47

Examination Outline Cross-Reference	Level	RO
	Tier #	2
	Group #	1
	K/A #	215003 K5.03
	Importance Rating	3.0

Knowledge of the operational implications of the following concepts as they apply to INTERMEDIATE RANGE MONITOR (IRM) SYSTEM: changing detector position

Proposed Question:

With the Mode Switch in STARTUP control rods are being withdrawn. The reactor has just been declared critical when the "A" Intermediate Range Monitor detector fully withdraws from the core.

Which one of the following describes the plant response?

a. Half Scram caused by IRM High High.

b. Half Scram caused by IRM Inoperative.

c. Control Rod Block caused by IRM High.

d. Control Rod Block caused by IRM detector position.

Proposed Answer: d. Detector NOT fully inserted causes a rod block, not a scram

Explanation (Justification of Distractors):

a. IRM will go lower, possible downscale alarm, no scram
b. IRM will be inoperative but no scram will occur
c. IRM will go lower it may cause IRM low block but NOT IRM high.

Technical Reference(s): N2-ARP-01, Ann 603442

Proposed references to be provided to applicants during the examination:

None

Learning Objective: 02-OPS-001-215-2-04, EO-8.0

Question Source:	Bank # Modified Bank # New	 NEW
Question History:	Previous NRC Exam Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 1
10CFR Part 55 Content:	41.5./.45.3	

Comments:

Question # RO 48

Examination Outline Cross-Reference	Level	RO
	Tier #	2
	Group #	1
	K/A #	215004 K2.01
	Importance Rating	2.6

Knowledge of the electrical power supplies to the following:
SRM channels/detectors.

Proposed Question:

The plant is in Cold Shutdown, following a Refueling outage. The RPS shorting links are removed.

Which one of the following describes the effect of de-energizing 24/48 VDC Panel 2BWS-PNL300B on Neutron Monitoring System (NMS) and Reactor Protection System (RPS)?

a. Only a half scram because of the power loss to some SRMs.

b. Only a half scram because of the power loss to some IRMs.

c. A full scram occurs because of the power loss to some SRMs.

d. RPS is energized because the battery charger supplies the NMS.

Proposed Answer: c.

Explanation (Justification of Distractors):

a. Because the shorting links are removed, RPS is in a non-coincident mode. The loss of any SRM or IRM causes a full scram.

b. Because the shorting links are removed, RPS is in a non-coincident mode. The loss of any SRM or IRM causes a full scram.

d. The charger will not supply power.

Technical Reference(s): N2-SOP-04, Rev 00, Attachment 7

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-215-2-03, EO-4a, EO-8 O2-OPS-001-215-2-04, EO-8	
Question Source:	Bank # Modified Bank # New	Q13265
Question History:	Previous NRC Exam Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	2
10CFR Part 55 Content:	55.41.7	

Comments:

Question # RO 49 SRO 51

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	2	2
	Group #	1	1
	K/A #	215004 K3.02	215004 K3.02
	Importance Rating	3.4	3..4

Knowledge of the effect that a loss or malfunction of the source range monitor (SRM) system will have on the following: Reactor manual control: plant-specific.

Proposed Question:

A reactor startup is in progress. All Intermediate Range Monitors (IRMs) are on range 2 <u>except</u> IRM "B" which is on range 3. The Source Range Monitor (SRM) detectors are being withdrawn.

Which one of the following describes the response if SRM "C" count rate lowers to 70 cps while it is being withdrawn?

a. A half scram on RPS "A" occurs.

b. A control rod block is generated.

c. SRM "C" downscale light turns on.

d. SRM "C" detector drive will deenergize.

Proposed Answer: b.

Explanation (Justification of Distractors):

a. Scram signal is upscale high high at $2x10^5$ cps.

c. Downscale occurs at 3 cps.

d. Detector drives normally remain energized until electrical power is removed.

Technical Reference(s): N2-OP-92, Rev 04, Section E.2.0

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-215-2-02, EO-4c	
Question Source:	Bank #	Q13173
	Modified Bank #	
	New	
Question History:	Previous NRC Exam	
	Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge	1
	Comprehension or Analysis	
10CFR Part 55 Content:	55.41.7	
	55.45.44	

Comments:

Question #

RO 50 SRO 52

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	2	2
	Group #	1	1
	K/A #	215005 K1.14	215005 K1.14
	Importance Rating	2.8	2.9

Knowledge and the physical connections and/or cause-effect relationships between Average Power Range Monitor / Local Power Range Monitor System and the following: Reactor vessel.

Proposed Question:

To be considered operable, each APRM is required to have a minimum total number of LPRM inputs as well as a minimum number of operable LPRM inputs from each detector level.

Which one of the following describes the basis for this requirement?

a. Ensure the APRM will provide a good representation of average core power.

b. Ensure the APRM averaging circuit has enough inputs to provide valid 3D Monicore calculations.

c. Ensure the combined LPRM signals will provide on scale readings, even at lower power levels.

d. Ensure the combined LPRM signals will provide automatic protection to prevent exceeding local thermal limits.

Proposed Answer: a.

Explanation (Justification of Distractors):

b. 3D Monicore uses LPRMS but is not the bases for the inoperative condition.

c. Not a consideration for the conditions provided.

d. Local or individual LPRMs provide local power indication to the RBM. The RBM enforces the protective actions.

Technical Reference(s): N2-OP-92, Section B

Proposed references to be provided to applicants during the examination:

None.

Learning Objective: O2-OPS-001-215-2-05, EO-1, EO-5

Question Source:	Bank #	
	Modified Bank #	
	New	New
Question History:	Previous NRC Exam	New
	Previous Test / Quiz	New
Question Cognitive Level:	Memory of Fundamental Knowledge	1
	Comprehension or Analysis	
10CFR Part 55 Content:	55.41.2	
	55.45.2	

Comments:

Question # RO 51

Examination Outline Cross-Reference	Level	RO
	Tier #	-
	Group #	-
	K/A #	216000 A3.01
	Importance Rating	3.4

Ability to monitor automatic operation of the nuclear boiler instrumentation including the following: Relationship between the meter/recorder readings and actual parameter values. Plant specific.

Proposed Question:

During conduct of the EOPs, the following parameters exist:

- Reactor pressure 40 psig
- Drywell pressure 8 psig
- Drywell temperature (highest) 250°F
- Suppression Pool temperature 105°F
- Rx Building temperature (highest) 150°F

If actual reactor water level is at the top of active fuel (TAF), which one of the following describes the status of RPV level instrumentation?

a. No level instruments are available.

b. Fuel Zone level instruments are available.

c. Wide Range level instruments are available.

d. Upset Range level instruments are available.

Proposed Answer: b.

Explanation (Justification of Distractors):

a. In the safe region of the RPV saturation curve (EOP Figure D) and the temperatures are below the maximum allowed. The fuel zone instruments are above the minimum indicated level.

c. below minimum indicated level

d. below minimum indicated level

Technical Reference(s):	N2-EOP-RPV, Rev 8, Figure A, Figure C N2-EOP-6, Rev 05, Attachment 28		

Proposed references to be provided to applicants during the examination:

N2-EOP-RPV, Rev 8, Figure A, Figure C

Learning Objective:	O2-OPS-001-216-2-01, EO-9 O2-OPS-006-344-2-01, EO-2		
Question Source:	Bank # Modified Bank # New	New	
Question History:	Previous NRC Exam Previous Test / Quiz	New New	
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis		2
10CFR Part 55 Content:	55.41.7 55.45.7		

Comments:

51-2

Question # RO 52 SRO 53

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	2	2
	Group #	1	1
	K/A #	216000 K3.01	216000 K3.01
	Importance Rating	4.0	4.3

Knowledge of the effect that a loss or malfunction of the NUCLEAR BOILER INSTRUMENTATION will have on the following: Reactor Protection System

Proposed Question:

The plant is at 100% power with reactor water level transmitter, **B22-N680A**, (RPS Narrow Range) failed downscale.

Prior to removing the transmitter from service the equalizing valve for reactor water level transmitter, **B22-N680D**, (RPS Narrow Range) is fully opened by I&C.

Assume **NO** operator actions are taken. Which one of the following describes the effects of these failures?

a. The RFPs and the Main Turbine will trip.

b. Only a reactor low level alarm is received.

c. Only a reactor high level alarm is received.

d. A half scram is received on RPS trip system "A".

Proposed Answer: d. Half scram from A side level

Explanation (Justification of Distractors):
a. Only one channel is high
b. This would cause a half scram
c. This would cause a half scram

Technical Reference(s): N2-OP-34, Attachment 1

Proposed references to be provided to applicants during the examination:

None

Learning Objective:	02-OPS-001-216-2-00, EO-2.0, 5.0, 8.0	
Question Source:	Bank # Modified Bank # New	 NEW
Question History:	Previous NRC Exam Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 3
10CFR Part 55 Content:	55 41.7	

Comments:

Question #		RO 53	SRO 54
Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	2	2
	Group #	1	1
	K/A #	217000 K6.03	217000 K6.03
	Importance Rating	3.5	3.5

Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR CORE ISOLATION COOLING (RCIC): Suppression Pool water supply.

Proposed Question:

The Reactor Core Isolation Cooling (RCIC) pump suction is lined up to the Suppression Pool. Following a loss of feedwater RCIC receives an initiation signal. Several seconds later a Suppression Pool low level occurs.

Which one of the following is the expected RCIC response?

a. RCIC initiates and injects. CST suction valve (MOV129) does **NOT** open.

b. RCIC initiates but pump discharge to the reactor (MOV126) does **NOT** open.

c. RCIC initiates then trips on low suction pressure when Suppression Pool suction valve (MOV136) closes.

d. RCIC initiates and injects. CST suction valve (MOV129) opens and Suppression Pool suction valve (MOV136) closes.

Proposed Answer: a. The only way to swap back to the CST suction is manually, RCIC stays lined up to the SP and injects

Explanation (Justification of Distractors):

b. Pump Discharge will open and inject SP water.
c. Suction valves do not change position.
d. Suction valves do not change position.

Technical Reference(s): N2-OP-35 Sect. F

Proposed references to be provided to applicants during the examination:

None

Learning Objective: 02-OPS-001-217-2-00, EO-3.0

Question Source:	Bank # Modified Bank # New	 NEW
Question History:	Previous NRC Exam Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 2
10CFR Part 55 Content:	41.7 / 45.7	

Comments:

Question # RO 54

Examination Outline Cross-Reference	Level	RO
	Tier #	2
	Group #	1
	K/A #	217000 A4.09
	Importance Rating	3.7

Ability to manually operate and/or monitor in the control room System Pressure.

Proposed Question:

Following a scram Reactor Core Isolation Cooling (RCIC) was manually initiated and used for RPV level control. As RPV level rose to 180 inches, 2ICS*FV108, TEST BYPASS TO CONDENSATE STOR TK was opened to control RCIC flow.

Currently the following conditions exist:

- RPV Pressure 880 psig
- RPV Level 121 inches
- RCIC Pump Discharge Pressure 720 psig
- RCIC Flow Controller is in MANUAL
- RCIC Flow 600 gpm

You are directed to raise RPV level with RCIC. Which one of the following is necessary?

a. Throttle open 2ICS*FV108, TEST BYPASS TO CONDENSATE STOR TK.

b. Place Flow Controller in AUTO, then Open 2ICS*MOV126, PMP 1 DISCH TO REACTOR.

c. Close 2ICS*FV108, TEST BYPASS TO CONDENSATE STOR TK, then adjust RCIC speed with the Flow Controller.

d. Open 2ICS*MOV126, PMP 1 DISCH TO REACTOR and close 2ICS*FV108, TEST BYPASS TO CONDENSATE STOR TK.

Proposed Answer: c.

Explanation (Justification of Distractors):

a. This will place RCIC in the correct lineup but the RCIC speed is too low
b. This will not change anything except controller position
d. Opening MOV126 is not required.

Technical Reference(s): N2-OP-35 Sect. F

Proposed references to be provided to applicants during the examination:

None

Learning Objective: 02-OPS-001-217-2-00, EO-3.0

Question Source:	Bank #	
	Modified Bank #	
	New	NEW
Question History:	Previous NRC Exam	
	Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge	
	Comprehension or Analysis	2
10CFR Part 55 Content:	41.7 / 45.5 / 45.6 / 45.7 / 45.8	

Comments:

Question # RO 55 SRO 55

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	2	2
	Group #	1	1
	K/A #	218000 K2.01	218000 K2.01
	Importance Rating	3.1	3.3
Knowledge of electrical power supplies to the following: ADS logic.			

Proposed Question:

The unit is operating at 100% power. ALL high pressure and low pressure ECCS systems are in standby.

Div. 1 DC power from 2BYS*PNL201A is lost.

Which one of the following describes the ability of the SRVs to function in the pressure-relief mode and in the ADS mode?

a. **NO** SRVs will function in the pressure-relief mode. Actuation of Div. II ADS logic or placing the Div. II ADS valves key lock switches (Panel H13-P631) to open will open the ADS valves.

b. **NO** SRVs will function in the pressure-relief mode. Actuation of Div. II ADS logic opens the ADS valves. Placing the Div. II ADS valve key lock switches (PNL H13-P631) to open will NOT open the ADS valves.

c. **ALL** SRVs will function in the pressure-relief mode. Actuation of Div. II ADS logic or placing the Div. II ADS valve key lock switches (Panel H13-P631) to open will open the ADS valves.

d. **ALL** SRVs will function in the pressure-relief mode. Actuation of Div. II ADS logic opens the ADS valves. Placing the Div. II ADS valve key lock switches (Panel H13-P631) to open will NOT open the ADS valves.

Proposed Answer: a.

Explanation (Justification of Distractors):

b. Placing the Div II keylock switches to open will open the ADS valves.
c. No SRVs function in the pressure-relief mode.
d. No SRVs function in the pressure-relief mode. Placing the Div II keylock switches to open will open the ADS valves.

Technical Reference(s):	N2-OP-34, Rev 07, Section B.2.2, B.2.3 N2-SOP-04, Rev 00, Attachment 1	

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-218-2-01, EO-4b, EO-4c	
Question Source:	Bank # Modified Bank # New	Q15788
Question History:	Previous NRC Exam Previous Test / Quiz	 Week 15 exam
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	1
10CFR Part 55 Content:	55.41.7	
Comments:		

Question #		RO 56
Examination Outline Cross-Reference	Level	RO
	Tier #	2
	Group #	1
	K/A #	223001 K6.01
	Importance Rating	3.6

Knowledge of the effect that a loss or malfunction of the following will have on the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES: Drywell Cooling.

Proposed Question:

While operating at full power a loss of circuit 2DRSA04 trips all the drywell coolers. Which one of the following is the principle effect within the primary containment?

a. Water level instruments become inaccurate causing a scram.

b. High temperatures at the drywell head require a manual scram.

c. The drywell overheats and pressure rises requiring a shutdown.

d. Recirculation pump motors overheat requiring a power reduction.

Proposed Answer: c.

Explanation (Justification of Distractors):

a. Other actions must be taken long before this becomes a consideration.
b. The principle problem is drywell pressure.
d. Motors will heat up but drywell temp and press is principle problem.

Technical Reference(s): N2-SOP-60, Sect. D.3.0

Proposed references to be provided to applicants during the examination:

None

Learning Objective: 02-OPS-001-222-2-01, EO-6.0

Question Source:	Bank #	
	Modified Bank #	
	New	NEW
Question History:	Previous NRC Exam	
	Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge	
	Comprehension or Analysis	1
10CFR Part 55 Content:	41.7 / 45.7	

Comments:

Question # RO 57 SRO 56

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	2	2
	Group #	1	1
	K/A #	223001 2.4.45	223001 2.4.45
	Importance Rating	3.3	3.6

Ability to prioritize and interpret the significance of each annunciator or alarm.

Proposed Question:

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

- 602309, RWCU PUMP ROOM A TEMPERATURE HIGH

High temperature is confirmed. Assuming that all systems function as designed, which one of the following describes the primary containment response and the required operator actions?

a. Only group 7 isolates. Verify the running WCS pump trips.

b. Only group 6 isolates. Establish a leak path for WCS pump seals.

c. Group 6 and group 7 isolate. Manually scram the reactor per N2-SOP-101C, Reactor Scram.

d. Group 6 and group 7 isolate. Enter N2-EOP-SC, Secondary Containment Control.

Proposed Answer: d.

Explanation (Justification of Distractors):

a. Both the inboard WCS isolation valve (group 7) and the outboard WCS isolation valve (group 6) close. A single trip from high area temperature will isolate both valves (groups). Verify the running pump trips is an ARP action.

b. Both the inboard WCS isolation valve (group 7) and the outboard WCS isolation valve (group 6) close. A single trip from high area temperature will isolate both valves (groups). Establishing a seal leak path is necessary in response to the annunciator and automatic response.

c. A reactor scram is only required if a the primary system discharging into the reactor building cannot be isolated. With the system responding as designed, the leak will be isolated.

Technical Reference(s):	N2-ARP-01, Rev 00, 602309 N2-OP-83, Rev 03, Attachment 2, Group 6 N2-OP-83, Rev 03, Attachment 2, Group 7	

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-223-2-02, EO-2, EO-5 O2-OPS-006-344-2-08, EO-1 O2-OPS-001-204-2-01, EO-4a, EO-4b	
Question Source:	Bank # Modified Bank # New	 New
Question History:	Previous NRC Exam Previous Test / Quiz	New New
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 2
10CFR Part 55 Content:	55.43.5 55.45.3 55.45.12	

Comments:

57-

Question # RO 58

Examination Outline Cross-Reference	Level	RO
	Tier #	2
	Group #	1
	K/A #	223002 2.1.32
	Importance Rating	3.4
Ability to explain and apply system limits and precautions.		

Proposed Question:

N2-OP-83, PRIMARY CONTAINMENT ISOLATION SYSTEM, contains the following precaution and limitation:

If a system isolation has occurred due to a valid signal, the problem must be determined and corrected prior to resetting or bypassing the isolation signal, unless directed to do otherwise by the Emergency Operating Procedures.

Which one of the following is operator actions is allowed by this precaution?

a. Defeat RPV low pressure isolations to allow injection systems to operate following a LOCA.

b. Bypass drywell pressure isolations to use reactor water cleanup for RPV level control.

c. Reset and re-open the MSIVs to relieve RPV pressure when RPV Blowdown is anticipated.

d. Override the reactor building ventilation isolations to remove smoke from a fire in drywell.

Proposed Answer: a.

Explanation (Justification of Distractors):

b. WCS isolations may be bypassed for pressure control.
c. MSIVs should not be opened with evidence of fuel failure.
d. There is no trip on smoke, and this is not included in the EOPs

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Technical Reference(s): N2-OP-83 and EOPs

Proposed references to be provided to applicants during the examination:

EOPs without entry conditions

Learning Objective: 02-OPS-001-221-2-01, EO-6.0

Question Source:	Bank # Modified Bank # New	 NEW
Question History:	Previous NRC Exam Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 2
10CFR Part 55 Content:	41.10 / 43.2 / 45.12	

Comments:

Question # RO 59 SRO 60

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	2	2
	Group #	2	1
	K/A #	241000 K6.01	241000 K6.01
	Importance Rating	2.8	2.9

Knowledge of the effect that a loss or malfunction of the following will have on the Reactor Regulating System: A.C. electrical power.

Proposed Question:

Reactor startup in progress. The Main turbine has been rolled to 1800 rpm and is at set speed:

- SET SPEED light is on
- SPEED INCREASING light is off

Before the generator can be synchronized, 2VBB-UPS1A power to the Electro-Hydraulic Control (EHC) system is lost.

Which one of the following describes the effect on the Main Turbine and bypass valves?

a. Turbine trip with bypass valves open.

b. Turbine trip and bypass valves close.

c. Turbine is at 1800 rpm with the bypass valves closed.

d. Turbine is at 1800 rpm with bypass valves controlling pressure.

Proposed Answer: d.

Explanation (Justification of Distractors):
The PMG will maintain power to the EHC electrical control circuitry. If turbine speed was lower, then the EHC system would lose power causing an all valves closed signal.

Technical Reference(s): N2-SOP-71, Section 1.0

Proposed references to be provided to applicants during the examination:

None.

Learning Objective: O2-OPS-001-248-2-00, EO-5

Question Source:	Bank #		
	Modified Bank #		
	New	New	
Question History:	Previous NRC Exam	New	
	Previous Test / Quiz	New	
Question Cognitive Level:	Memory of Fundamental Knowledge		
	Comprehension or Analysis		2
10CFR Part 55 Content:	55.41.7		
	55.45.7		

Comments:

Question # RO 60 SRO 73

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	2	2
	Group #	1	2
	K/A #	259001 2.4.49	259001 2.4.49
	Importance Rating	4.0	4.0

Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

Proposed Question:

The plant is operating at 80% power when one of the two operating Reactor Feedwater Pumps (RFP) trips. Which one of the following describes the immediate operator actions?

a. If recirculation pumps do not shift to slow speed, then scram the reactor.

b. If recirculation flow has not lowered automatically, then manually reduce Recirc flow.

c. Start the standby RFP and if level is NOT stable, then control RPV level in manual.

d. Perform the actions for a reactor scram and establish RPV level control using the running RFP.

Proposed Answer: b.

Explanation (Justification of Distractors):

a. The recirc pumps should automatically runback to lower reactor power within the capability of one RFP. The IOA is to verify the automatic action occurs and if not, then to take the action.

c. There is no requirement to start a reactor feedwater pump. The appropriate action is to establish plant conditions (power) within the capability of one RFP.

d. This is a subsequent operator action taken after it is determined that the cause is a failure of the feedwater level control system.

Technical Reference(s): N2-SOP-06, Section 3

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-006-SOP-01-29, TO-2, EO-2	
Question Source:	Bank #	
	Modified Bank #	
	New	New
Question History:	Previous NRC Exam	New
	Previous Test / Quiz	New
Question Cognitive Level:	Memory of Fundamental Knowledge	1
	Comprehension or Analysis	
10CFR Part 55 Content:	55.41.7	

Comments:

60-2

Question # RO 61

Examination Outline Cross-Reference	Level	RO
	Tier #	2
	Group #	1
	K/A #	259002 K3.05
	Importance Rating	2.8

Knowledge of the effect that a loss or malfunction of the Reactor Water Level Control System will have on the following: Recirculation flow control system.

Proposed Question:

The unit is operating at 100% power when a failure of the FWCS causes reactor water level to rise and keep rising. NO operator action is taken.

Which one of the following describes the status of the RCS system pumps and flow control valves when plant conditions are stable?

a. Pumps are tripped with their FCVs in loop manual.

b. Pumps are in fast speed with their FCVs in the motion inhibit.

c. Pumps are at minimum speed with their FCVs in loop manual.

d. Pumps are in slow speed with their FCVs at the 20% valve position.

Proposed Answer: d.

Explanation (Justification of Distractors):
When the RPV high level trip is received, the main turbine RFPs trip. The reactor scrams on turbine SV/CV fast closure. When the RPV level lowers to the RPV low level alarm, the automatic runback circuit is actuated to the FCVs. When RPV level lowers to 159.3" or the RFPs are secured for greater than 15 seconds, the RCS pumps automatically shift to slow speed.

Technical Reference(s): N2-ARP-01, Rev 00, 602226, 602222

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-202-2-02, EO-4c, EO-5	
Question Source:	Bank #	
	Modified Bank #	
	New	New
Question History:	Previous NRC Exam	New
	Previous Test / Quiz	New
Question Cognitive Level:	Memory of Fundamental Knowledge	
	Comprehension or Analysis	2
10CFR Part 55 Content:	55.41.7	
	55.45.4	

Comments:

Question # RO 62 SRO 62

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	2	2
	Group #	1	1
	K/A #	261000 A4.07	261000 A4.07
	Importance Rating	3.1	3.2

Ability to manually operate and/or monitor in the Control Room: system flow

Proposed Question:

Hi drywell pressure causes a trip of the Reactor Building Ventilation System. Both trains of Standby Gas Treatment (GTS) automatically start.

After verifying both trains of GTS are **NOT** required it is decided to shutdown Train "B". The Train "B" INITIATION control switch is placed in **AUTO AFTER STOP**. The following conditions are observed:

- GTS*MOV1B, INLET FROM RX BLDG VENTILATION goes closed
- GTS*AOV2B, TRAIN B INLET VLV goes closed
- GTS*AOV3B, FAN 1B DISCHARGE ISOL VLV remains open
- GTS*FN1B, SBGTS FAN stops

With the high drywell pressure condition still present, which one of the following will occur (assuming **NO** operator action)?

GTS train "B"...

a. restarts and restores Reactor Building differential pressure to –0.25 inches WG or more negative.

b. restarts with flow less than rated, Reactor Building differential pressure is –0.05 inches WG.

c. remains off and GTS Train "A" flow raises to 4000 scfm, Reactor Building differential pressure is –0.05 inches WG.

d. remains off and GTS Train "A" flow raises to 4000 scfm, Reactor Building differential pressure is -0.25 inches WG or more negative.

Proposed Answer: A. The B train restarts because the initiation signal is still present and the A train can not maintain Rx Bldg differential. The valve failure doesn't effect SBGTS flow because both fans are running and no short cycle flow path exists

Explanation (Justification of Distractors):

b. The B train will auto start and restore d/p to –0.25 in. WG.
c. Short cycle flow path is eliminated and d/p will be restored.
d. A short cycle flow path has been created by the B fan discharge isolation valve remaining open there will be no flow and Rx Bldg d/p will lower.

Technical Reference(s): N2-OP-61.B, Sects D.10 and H.2

Proposed references to be provided to applicants during the examination:

None

Learning Objective:	02-OPS-001-261-2-01, EO-6.0, EO-8.0	
Question Source:	Bank #	New
	Modified Bank #	
	New	
Question History:	Previous NRC Exam	
	Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge	
	Comprehension or Analysis	2
10CFR Part 55 Content:	55.41 8	

Comments:

Question # RO 63 SRO 64

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	2	2
	Group #	1	1
	K/A #	264000 A1.03	264000 A1.03
	Importance Rating	2.8	2.9

Ability to predict and/or monitor changes in parameters associated with operating the EMERGENCY GENERATORS (DIESEL/JET) controls including: Operating voltages, currents, and temperatures.

Proposed Question:

Emergency Diesel Generator 1 (EDG1) is running paralleled to the grid for the monthly load test. EDG1 parameters are:

- Voltage 4160 v
- Load 4400 kw
- Frequency 60.0 hz

A LOCA signal is received. Which one of the following describes the EDG1 voltage, load, and frequency one (1) minute later?

a. Voltage and frequency are lower, load is higher.

b. Voltage is zero, load is downscale, frequency is upscale.

c. Voltage is the same, load is zero, frequency is higher.

d. Voltage and frequency are the same, load is zero.

Proposed Answer: d.

Explanation (Justification of Distractors):
a. EDG1 is NO longer paralleled to the grid.
b. EDG1 is NOT tripped.
c. EDG1 is running unloaded but NOT in the droop mode as indicated.

Technical Reference(s): N2-OP-101A, Section B

Proposed references to be provided to applicants during the examination:

None.

Learning Objective: O2-OPS-001-264-2-01, EO-7

Question Source:	Bank #	
	Modified Bank #	
	New	New
Question History:	Previous NRC Exam	New
	Previous Test / Quiz	New
Question Cognitive Level:	Memory of Fundamental Knowledge	
	Comprehension or Analysis	2
10CFR Part 55 Content:	55.41.5	
	55.45.5	

Comments:

63-2

Question # RO 64 SRO 65

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	2	2
	Group #	1	1
	K/A #	264000 A3.06	264000 A3.06
	Importance Rating	3.1	3.2

Ability to monitor automatic operations of the EMERGENCY GENERATORS (DIESEL/JET) including: Cooling water system operations.

Proposed Question:

The Division I Emergency Diesel Generator (EDG) has been supplying its loads for ten (10) minutes following a LOCA. The following conditions exist:

- 852118, EDG 1 SERVICE WATER INLET PRESS LOW, alarms
- Pressure sensed at 2SWP*PT66A is 20 psig

Which one of the following describes an effect on EDG1?

a. EDG1 trips on high oil temperatures.

b. EDG1 trips on low service water flow.

c. EDG1 continues to operate with a higher service water flow.

d. EDG1 continues to operate with higher jacket water temperature.

Proposed Answer: d.

Explanation (Justification of Distractors):

a. Oil temperatures will rise but the trips are bypassed because of the LOCA start.

b. There is no trip on a loss of service water. The automatic action that occurs is the 2SWP*MOV66A closes to preserve service water to the other EDG.

c. The automatic action that occurs is the 2SWP*MOV66A closes to preserve service water to the other EDG.

Technical Reference(s): N2-ARP-01, Rev 00, 852118

Proposed references to be provided to applicants during the examination:

None.

Learning Objective: O2-OPS-001-264-2-01, EO-7.d.2

Question Source:	Bank #	
	Modified Bank #	
	New	New
Question History:	Previous NRC Exam	New
	Previous Test / Quiz	New
Question Cognitive Level:	Memory of Fundamental Knowledge	
	Comprehension or Analysis	2
10CFR Part 55 Content:	55.41.7	
	55.45.7	

Comments:

Question # RO 65 SRO 80

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	2	2
	Group #	2	3
	K/A #	201003 K6.01	201003 K6.01
	Importance Rating	3.3	3.3

Knowledge of the effect that a loss or malfunction of the following will have on the control rod and drive mechanism: Control rod drive hydraulic system.

Proposed Question:

The plant is operating at 50% power with the following CRD system indications:

• Drive water differential pressure	265 psid
• Drive flow	0.0 gpm
• Charging Header pressure	1450 psig
• CRD system flow	50 gpm

When attempting to insert control rod 18-19, drive water flow is observed at 0.0 gpm. When attempting to withdraw control rod 18-19, drive water flow is observed at 2.0 gpm. The control rod does **NOT** move.

Which one of the following describes the cause of the above indications?

Directional Control Valve ...

a. SOV123, Insert Supply, is stuck open.

b. SOV123, Insert Supply, is stuck closed.

c. SOV122, Withdrawal Supply, is stuck open.

d. SOV122, Withdrawal Supply, is stuck closed.

Proposed Answer: b.

Explanation (Justification of Distractors):
a. 123 stuck open would provide continuous insert flow
c. 122 stuck open would provide continuous withdrawal flow
d. 122 stuck closed would prohibit withdrawal flow but allow insert flow.

Technical Reference(s): N2-OP-30, Section H.1.0

Proposed references to be provided to applicants during the examination:

None.

Learning Objective: O2-OPS-001-201-2-01, EO-4a, EO-8

Question Source:	Bank #	
	Modified Bank #	
	New	New
Question History:	Previous NRC Exam	New
	Previous Test / Quiz	New
Question Cognitive Level:	Memory of Fundamental Knowledge	
	Comprehension or Analysis	2
10CFR Part 55 Content:	55.41.7	
	55.45.7	

Comments:

Question # RO 66

Examination Outline Cross-Reference	Level	RO
	Tier #	2
	Group #	2
	K/A #	202001 A4.11
	Importance Rating	3.2

Ability to manually operate and/or monitor in the control room: Seal pressures: plant-specific.

Proposed Question:

The plant is operating at 20% power when the following annunciators alarm:

- 602109, RECIRC PUMP 1A OUTER SL LEAK HIGH
- 602115, RECIRC PUMP 1A SEAL STAGING FLOW HIGH/LOW

The following indications are observed:

- Seal leakage 1.7 gpm
- Seal staging flow 1.9 gpm
- Upper seal staging pressure 250 psig
- Lower seal staging pressure 950 psig

Which one of the following describes the state of the 1A Recirculation Pump seals?

a. The lower seal failed.

b. The upper seal failed.

c. Both the upper and lower seal failed.

d. The seal staging flow orifice is clogged.

Proposed Answer: b.

Explanation (Justification of Distractors):

a. Upper cavity pressure will approach lower cavity pressure.
c. Pressure in both seals would lower. The upper seal pressure is too high.
d. Will not receive outer seal leak high alarm for this condition.

Technical Reference(s): N2-OP-29, Section H.5.0

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-202-2-01, EO-3, EO-4b, EO-8	
Question Source:	Bank # Modified Bank # New	Q9104
Question History:	Previous NRC Exam Previous Test / Quiz	 Week 12 exam
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 2
10CFR Part 55 Content:	55.41.7 55.45.5	

Comments:

66-2

Question # RO 67

Examination Outline Cross-Reference	Level	RO
	Tier #	-
	Group #	-
	K/A #	204000 K1.05
	Importance Rating	2.7

Knowledge of the physical connections and/or cause-effect relationships between reactor water cleanup system and the following: Plant air systems.

Proposed Question:

The Reactor Water Cleanup (WCS) System is operating with some flow being rejected to the Main Condenser when a complete loss of instrument air occurs. Which one of the following describes the effect on the WCS system?

a. WCS filter supply and return valves remain as is. WCS continues to operate.

b. WCS filter demineralizer inlet and outlet isolation valves and the reject flow control valve close.

c. WCS containment isolation valves fail closed. WCS pumps trip if no action is taken within 15 minutes.

d. WCS system will continue to operate in the reject mode but the return to the feedwater system will isolate.

Proposed Answer: b.

Explanation (Justification of Distractors):

a. Filter demin valves fail closed.
c. Containment isolation valves are not air operated.
d. The rejct mode is isolated because FV-135 closed.

Technical Reference(s): N2-SOP-19, Section 5.0

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-204-2-01, EO-5, EO-8 O2-OPS-001-279-2-00, EO-8	
Question Source:	Bank # Modified Bank # New	Q13207
Question History:	Previous NRC Exam Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	1
10CFR Part 55 Content:	55.41.4 55.41.7 55.43.5 55.45.7	

Comments:

67-2

Question # RO 68 SRO 71

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	2	2
	Group #	2	2
	K/A #	214000 K4.01	214000 K4.01
	Importance Rating	3.0	3.1

Knowledge of the Rod Position Information System design feature(s) and/or interlocks which provide for the following: reed switch locations.

Proposed Question:

An individual rod scram has been performed on control rod 30-31 using the SRI test switches. When the control rod is selected the four-rod display indicates two blank windows for control rod 30-31.

Which one of the following is the reason for the blank indication for control rod 30-31?

a. The rod is bypassed in the RPIS cabinet.

b. CRDM magnet for the rod is past position "00".

c. An odd reed switch position is actuated for the rod.

d. A substitute rod position is entered in RWM for position "00".

Proposed Answer: b.

Explanation (Justification of Distractors):
a. "XX" would be indicated.
c. "- -" would be indicated.
d. The substituted position numerical value is indicated.

Technical Reference(s): N2-OP-96, Section B.3.0, H.2.0.

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-201-2-02, EO-3	
Question Source:	Bank #	Q8284
	Modified Bank #	
	New	
Question History:	Previous NRC Exam	
	Previous Test / Quiz	10/99 System exam
Question Cognitive Level:	Memory of Fundamental Knowledge	1
	Comprehension or Analysis	
10CFR Part 55 Content:	55.41.7	

Comments:

68-

Question # RO 69

Examination Outline Cross-Reference	Level	RO
	Tier #	2
	Group #	2
	K/A #	215002 A3.05
	Importance Rating	3.2

Ability to monitor automatic operations of the Rod Block Monitor System including: Back panel meters and indicating lights: BWR-3,4,5.

Proposed Question:

A power ascension is in progress. Annunciator 603204, RBM UPSCALE/INOPERABLE alarms.

The following indications are observed on the RBM A NUMAC.

- RBM FLUX 89%
- APRM FLUX 68%
- FLOW 77%

- LPRMS IN RBM AVERAGE 3
- MINIMUM LPRMS ALLOWED 4
- SETUP RANGE PERMITTED HIGH

Which one of the following describes the required operator response?

a. Inform the CRS that RBM A is inoperable.

b. Reduce power to below the alarm setpoint.

c. Select another rod and then reselect the affected rod.

d. At 2CEC*PNL603, depress the PUSH TO SET UP pushbutton.

Proposed Answer: a.

Explanation (Justification of Distractors):

b. Power is below the rod block set point.

c. This will reinitiate the nulling sequence but will not correct the inoperable LPRM inputs.

d. RBM is already at the HIGH setpoint.

Technical Reference(s): N2-ARP-01, Rev 00, 603204

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-215-2-06, EO-4c, EO-5	
Question Source:	Bank # Modified Bank # New	 New
Question History:	Previous NRC Exam Previous Test / Quiz	New New
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 2
10CFR Part 55 Content:	55.41.7 55.45.7	

Comments:

69-2

Question # RO 70

Examination Outline Cross-Reference	Level	RO
	Tier #	2
	Group #	2
	K/A #	219000 K3.01
	Importance Rating	3.9

Knowledge of the effect that a loss or malfunction of the RHR/LPCI: Torus / Suppression Pool cooling mode will have on the following: Suppression pool temperature control.

Proposed Question:

The plant is at 100% power. The "A" RHR loop is out of service.

- One SRV opens and CANNOT be closed
- The mode switch is placed to SHUTDOWN
- All control rods do NOT insert.
- Reactor power is 35%

Which one of the following describes the limit that will be challenged FIRST if the reactor CANNOT be scrammed and "B" RHR loop CANNOT be started when required?

a. SRV Tail Pipe level Limit

b. RPV Saturation Temperature

c. Pressure Suppression Pressure

d. Heat Capacity Temperature Limit

Proposed Answer: d.

Explanation (Justification of Distractors):

a. Cool down rate is not challenged until blowdown is required. After exceeding HCTL.

b. Not challenged until after the blowdown since there is no LOCA to challenge primary containment parameters.

c. There is sufficient injection to maintain RPV level in the EOP specified bands including after terminate and prevent.

Technical Reference(s):	NMP2 EOP Technical Bases, N2-EOP-C5 N2-EOP-C5, Rev 8

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-006-344-2-17, EO-3	
Question Source:	Bank # Modified Bank # New	 New
Question History:	Previous NRC Exam Previous Test / Quiz	New New
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 2
10CFR Part 55 Content:	55.41.7 55.45.7	

Comments:

Question # RO 71 SRO 82

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	2	2
	Group #	2	1
	K/A #	239001 K2-01	239001 K2.01
	Importance Rating	3.2	3.3

Knowledge of electrical power supplies to the following: Main steam isolation valve solenoids.

Proposed Question:

While operating at 50% power, the 2VBS-UPS3A output to its loads is lost. Which one of the following describes the final position of the MSIVs ten (10) seconds following the power loss?

	Inboard MSIVs	Outboard MSIVs
a.	Closed	Open
b.	Open	Closed
c.	Closed	Closed
d.	Open	Open

Proposed Answer: d.

Explanation (Justification of Distractors):
No MSIVs close because only half of the pilot valve solenoids are deenergized.

Technical Reference(s): N2-OP-1, Rev 10, Section B

Proposed references to be provided to applicants during the examination:

None.

Learning Objective: O2-OPS-001-239-2-00, EO-3, EO-5, EO-8

Question Source:	Bank #	Q8264
	Modified Bank #	
	New	
Question History:	Previous NRC Exam	
	Previous Test / Quiz	Week 14 exam
Question Cognitive Level:	Memory of Fundamental Knowledge	1
	Comprehension or Analysis	
10CFR Part 55 Content:	55.41.5	
	55.45.5	

Comments:

Question # RO 72

Examination Outline Cross-Reference	Level	RO
	Tier #	2
	Group #	2
	K/A #	239001 2.2.2
	Importance Rating	4.0

Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.

Proposed Question:

The plant is at 75% power with quarterly MSIV Functional Testing in progress. Inboard MSIV, MSS*AOV6A, will be tested first. Operator actions are as follows:

- MSS*AOV6A Close/Auto/Test Control switch is positioned to TEST. The MSIV remains OPEN and NO half scram occurs.
- Then MSS*AOV6A TRIP TEST pushbutton is depressed and is held in the depressed state for one (1) minute.

Which one of the following describes the plant response?

a. A full reactor scram occurs.

b. The remaining MSIVs close.

c. Reactor power stabilizes at a higher power.

d. Reactor pressure stabilizes at a lower pressure.

Proposed Answer: c.

Explanation (Justification of Distractors):

a. Half scram occurs on RPS B

b. Only the affected MSIV closes. Reactor power rises and will be stable at a higher power level.

d. Reactor pressure may rise but will not lower. Steam flows will change affecting reactor power.

Technical Reference(s):	N2-OSP-MSS-Q002, Section 4, Section 8.2 N2-OP-1, Section B.	

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-239-2-00, EO-8	
Question Source:	Bank # Modified Bank # New	 New
Question History:	Previous NRC Exam Previous Test / Quiz	New New
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 2
10CFR Part 55 Content:	55.45.2	

Comments:

72-2

Question # RO 73 SRO 72

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	2	2
	Group #	2	2
	K/A #	245000 K5.02	245000 K5.02
	Importance Rating	2.8	3.1

Knowledge of the operational implications of the following concepts as they apply to Main Turbine Generator and Auxiliary Systems: Turbine operation and limitations.

Proposed Question:

In response to a lowering main condenser vacuum, reactor power is being reduced. Current conditions are:

- Main condenser vacuum is 24.3 inches Hg and slowly lowering
- 603112, RPS A CONT & STOP V CLOSURE BYPASSED, is ON
- 603412, RPS B CONT & STOP V CLOSURE BYPASSED, is ON

Which one of the following describes when the Main Turbine should be tripped?

a. Immediately

b. After manual reactor scram

c. Condenser Vacuum Low alarms

d. When vacuum lowers to 22.1 inches HG

Proposed Answer: a.

Explanation (Justification of Distractors):

b. Reactor scram is not required below 30% reactor power.

c. The existing turbine limits require that the turbine be removed from service. Lowering power will not preserve the integrity of the main turbine.

d. The existing turbine limits require that the turbine be removed from service. This is a subsequent action assuming that main generator load is above 30%.

Technical Reference(s):	N2-OP-21, Rev 07, Section D.3 N2-SOP-09, Rev 00, Section 4.2

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-006-245-2-01, EO-5, EO-8	
Question Source:	Bank # Modified Bank # New	 New
Question History:	Previous NRC Exam Previous Test / Quiz	New New
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	1
10CFR Part 55 Content:	55.41.5 55.45.3	

Comments:

Question # RO 74

Examination Outline Cross-Reference	Level	RO
	Tier #	2
	Group #	2
	K/A #	256000 A4.10
	Importance Rating	3.2

Ability to manually operate and/or monitor in the control room: Feedwater temperature.

Proposed Question:

With the plant operating at 70% reactor power, 2FWS-MOV102, 6th POINT HEATERS BYPASS VALVE, inadvertently opens. The reactor operator is able to close the valve within 30 seconds of opening.

Which one of the following describes the effect on feedwater temperature including why?

a. Lowers then returns to normal because feedwater heating is restored.

b. Lowers and remains lower until the reactor operator unisolates extraction steam.

c. Remains the same because the extraction steam to the feedwater heaters remains in service.

d. Remains the same because the 6th point heaters bypass inlet and outlet MOVs are overridden closed.

Proposed Answer: a.

Explanation (Justification of Distractors):

b. Extraction steam remains in service and will return FW temperature to normal once flow through the high-pressure heaters is reestablished.

c. Although extraction steam remains in service, flow through the high-pressure heaters is lost and all flow bypasses the heaters causing FW temperature to lower.

d. These valves do not exist on the bypass line. They are present on the feedwater heater string inlet and outlet lines.

Technical Reference(s):	N2-OP-8, Section D. N2-SOP-8, Section 4.2, Section 5	

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:

Question Source:	Bank # Modified Bank # New	 New
Question History:	Previous NRC Exam Previous Test / Quiz	New New
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 2
10CFR Part 55 Content:	55.41.7 55.45.5 55.45.8	

Comments:

74-2

Question # RO 75 SRO 63

		RO	SRO
Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	2	2
	Group #	2	1
	K/A #	262001 A2.03	262001 A2.03
	Importance Rating	3.9	4.3

Ability to (a) predict the impacts of the following on the AC Electrical Distribution System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of offsite power.

Proposed Question:

The plant is operating at 100% power when a <u>complete</u> loss of offsite power occurs. Emergency Diesel Generator response is as follows:

- Division I EDG failed to start and CANNOT be started
- Division II EDG failed to start and CANNOT be started
- Division III EDG started and energized its bus

Which one of the following describes immediate operator actions for these conditions?

a. Enter N2-EOP-RPV, RPV Control, and N2-EOP-PC, Primary Containment Control.

b. Depress the EMERGENCY STOP PUSHBUTTON for EDG III. Enter N2-SOP-01, Station Blackout.

c. Start one service water pump in each division 2SWP*P1(A,C,E) and 2SWP*P1(B,D,F).

d. Close service water valves to the reactor and turbine buildings 2SWP*V23 and 2SWP*V17 and align service water to "A" RHS heat exchanger.

Proposed Answer: b.
EDG III is shutdown because there is no service water available. The unit is in a station blackout requiring entry into N2-SOP-02, Station Blackout.

Explanation (Justification of Distractors):

a. Required but not immediate operator actions.
c. IOA if both the DIV I and DIV II EDG energize their respective busses.
d. IOA if 2ENS*SWG103 power is lost and the EDG energizes the bus.

Technical Reference(s): N2-SOP-02, Section 3

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-006-SOP-01-29, TO-1, EO-2	
Question Source:	Bank #	
	Modified Bank #	
	New	New
Question History:	Previous NRC Exam	New
	Previous Test / Quiz	New
Question Cognitive Level:	Memory of Fundamental Knowledge	1
	Comprehension or Analysis	
10CFR Part 55 Content:	55.41.5	
	55.45.6	

Comments:

75-2

Question # RO 76 SRO 74

		RO	SRO
Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	2	2
	Group #	2	2
	K/A #	262002 K6.01	262002 K6.01
	Importance Rating	2.7	2.9

Knowledge of the effect that a loss or malfunction of the following will have on the Uninterruptable Power Supply (A.C./D.C.): A.C. electrical power.

Proposed Question:

2VBB-UPS1A is aligned to the inverter with the TRANSFER CONTROL SWITCH positioned to AUTO RESTART when the following occurs:

- 2NJS-US3 to 2VBB-TRS1 becomes deenergized and **remains deenergized**.
- 2VBB-TRS1 transfer to the alternate AC source is complete after 10 seconds.

Which one of the following describes where 2VBB-UPS1A loads are automatically powered from after this event?

a. 2NJS-US4 through the inverter.

b. 2NJS-US6 bypassing the inverter.

c. 2NJS-US5 bypassing the inverter.

d. 2BYS-SWG001A through the inverter.

Proposed Answer: a.

Explanation (Justification of Distractors):

b. UPS1A transfers to the alternate AC supply through 2VBB-TRS1 although a slow transfer. Some UPS transfer to the maintenance supply on any loss of power. 2NJS-US6 is the maintenance supply for UPS1B.

c. UPS1A transfers to the alternate AC supply through 2VBB-TRS1 although a slow transfer. Some UPS transfer to the maintenance supply on any loss of power. 2NJS-US5 is the maintenance supply for UPS1A.

d. UPS1A transfers to the alternate AC supply through 2VBB-TRS1 although a slow transfer. 2BYS-SWG001A is the battery supply.

Technical Reference(s): N2-OP-71D, Section B

Proposed references to be provided to applicants during the examination:

None

Learning Objective: O2-OPS-001-262-2-03, EO-4.b, EO-6.e

Question Source:	Bank #	Q15782
	Modified Bank #	
	New	
Question History:	Previous NRC Exam	
	Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge	
	Comprehension or Analysis	2
10CFR Part 55 Content:	55.41.7	
	55.45.7	

Comments:

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Question # RO 77 SRO 75

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	2	2
	Group #	2	2
	K/A #	263000 K3.03	263000 K3.03
	Importance Rating	3.4	3.8

Knowledge of the effect that a loss or malfunction of the D.C. electrical distribution will have on the following: Systems with D.C. components.

Proposed Question:

The plant is operating at 100% power. One of the operating service water pumps is 2SWP*P1A. A ground fault results in the loss of 2BYS*SWG002A.

Which one of the following describes the effect of the power loss on the "A" Service Water Pump, 2SWP*P1A?

a. Trips and CANNOT be restarted until the ground fault is corrected.

b. Continues to run, but all trips and automatic functions are lost.

c. Trips and CANNOT be restarted until Division I DC power is restored.

d. Continues to run, but all protection <u>except</u> for the overcurrent and low suction pressure trips is lost.

Proposed Answer: b.

Explanation (Justification of Distractors):

a. Ground fault was on DC bus. Pump does not trip.
c. Ground fault was on DC bus. Pump does not trip.
d. All protection is lost.

Technical Reference(s): N2-SOP-04, Rev 00

Proposed references to be provided to applicants during the examination:

None

Learning Objective:	O2-OPS-001-263-2-01, EO-5, EO-8	
Question Source:	Bank #	Q8438
	Modified Bank #	
	New	
Question History:	Previous NRC Exam	
	Previous Test / Quiz	Week 13 exam
Question Cognitive Level:	Memory of Fundamental Knowledge	1
	Comprehension or Analysis	
10CFR Part 55 Content:	55.41.7	
	55.45.4	

Comments:

Question # RO 78 SRO 76

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	2	2
	Group #	2	2
	K/A #	271000 A3.02	271000 A3.02
	Importance Rating	2.9	2.8

Ability to monitor automatic operations of the Offgas system including: System flows.

Proposed Question:

During power operation, fuel failures have caused the following conditions:

- Process Radiation Monitor **2OFG-RU13A** has exceeded its **High** Setpoint
- Process Radiation Monitor **2OFG-RU13B** has exceeded its **Alert** Setpoint

Which one of the following describes the expected Offgas System flow indications on 2CEC*PNL851?

	Train "A" Flow (SCFM)	Train "B" Flow (SCFM)
a.	0	36
b.	36	0
c.	18	18
d.	0	0

Proposed Answer: d.

Explanation (Justification of Distractors):

a. b. c. There should NOT be any flow through either Offgas Trains because RU13A **or** B monitor reaching a High setpoint should Close the Offgas Outlet Valve, AOV-103, and Trip the Offgas Pumps.

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Technical Reference(s): N2-ARP-01, Rev 00, Ann 851253
02-OPS-001-271-2-01, OFFGAS SYSTEM, Fig 4

Proposed references to be provided to applicants during the examination:

None.

Learning Objective: O2-OPS-001-271-2-01, EO-8

Question Source:

Bank #		
Modified Bank #		
New	NEW	

Question History:

Previous NRC Exam	NEW
Previous Test / Quiz	NEW

Question Cognitive Level:

Memory of Fundamental Knowledge	1
Comprehension or Analysis	

10CFR Part 55 Content: 55.41.7
55.45.7

Comments:

78-

Question # RO 79

Examination Outline Cross-Reference	Level	RO
	Tier #	2
	Group #	2
	K/A #	272000 2.4.46
	Importance Rating	3.5
Ability to verify that alarms are consistent with the plant conditions.		

Proposed Question:

The plant is operating at 80% power. The following alarm is received:

- 851256, STACK EFFLUENT RAD MON ACTIVATED

Which one of the following describes how this alarm is verified consistent with plant conditions?

a. Review the status of the DRMS monitors.

b. Verify the reactor building isolates and Standby Gas starts.

c. Compare recorder readings on 2CEC-PNL882 to the posted aid.

d. Verify the Offgas discharge valve to the stack (AOV103) closes.

Proposed Answer: c.

Explanation (Justification of Distractors):
a. No DRMS indication for these monitors.
b. No isolation/actuation signal is generated for secondary containment.
d. No isolation signal is generated for Offgas.

Technical Reference(s): N2-ARP-01, Rev 00, 851256

Proposed references to be provided to applicants during the examination:

None.

Learning Objective: N2-OPS-001-272-2-01, EO-4, EO-5

Question Source:	Bank #	
	Modified Bank #	
	New	New
Question History:	Previous NRC Exam	New
	Previous Test / Quiz	New
Question Cognitive Level:	Memory of Fundamental Knowledge	
	Comprehension or Analysis	2
10CFR Part 55 Content:	55.43.5	
	55.45.3	
	55.45.12	

Comments:

79-

Question # RO 80 SRO 77

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	2	2
	Group #	2	2
	K/A #	286000 A2.06	286000 A2.06
	Importance Rating	3.1	3.2

Ability to (a) predict the impacts of the following on the fire protection system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low fire main pressure: plant-specific.

Proposed Question:

Which one of the following describes the pumps that automatically start if the fire protection header pressure lowers from 130 psig to 88 psig?

Assume all components function at their design setpoints.

a. Only the lead and lag pressure maintenance pumps.

b. Only the motor-driven fire pump and diesel driven fire pumps.

c. Only the lead and lag pressure maintenance and the motor-driven fire pumps start.

d. Only the lag pressure maintenance, motor-driven, and diesel driven fire pumps start.

Proposed Answer: c.

Explanation (Justification of Distractors):

a. The motor-driven fire pump auto starts at 90 psig.

b. The lead pressure maintenance pump starts at 120 psig and the lag pressure maintenance pump starts at 110 psig. Diesel driven fire pump does NOT start until 85 psig.

d. The lead pump auto starts. The motor-driven fire pump auto starts at 90 psig. The diesel-driven fire pump does NOT start until 85 psig.

Technical Reference(s):	N2-OP-43, Rev 05, Section B N2-ARP-01, Rev 00, 849238	

Proposed references to be provided to applicants during the examination:

None

Learning Objective:	O2-OPS-001-286-2-01, EO-4b, EO-4c, EO-8	
Question Source:	Bank # Modified Bank # New	New
Question History:	Previous NRC Exam Previous Test / Quiz	New New
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 1
10CFR Part 55 Content:	55.41.5 55.45.6	

Comments:

80-2

Question # RO 81

Examination Outline Cross-Reference	Level	RO
	Tier #	2
	Group #	2
	K/A #	290001 K1.02
	Importance Rating	3.4

Knowledge of the physical connections and/or cause-effect relationships between SECONDARY CONTAINMENT and the following: Primary containment system: Plant-Specific

Proposed Question:

Which one of the following systems when subjected to a single failure could cause a leak path from the primary containment to the secondary containment when the unit is operating at power?

a. MSS, Main Steam System

b. ICS, Reactor Core Isolation Cooling

c. WCS, Reactor Water Cleanup System

d. CCP, Reactor Building Closed Loop Cooling

Proposed Answer: b.

Explanation (Justification of Distractors):

a. MSS does not interface with the secondary containment

c. WCS does not interface directly with the primary containment, rather the RPV.

d. CCP does not interface directly with the primary containment. A CCP leak in the drywell would leak into the drywell. Also, two failures to get to secondary containment.

Technical Reference(s): N2-OP-35, Section B, Section D

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-217-2-00, EO-8 O2-OPS-001-221-2-00, EO-4 O2-OPS-001-288-2-03, EO-5	
Question Source:	Bank # Modified Bank # New	 New
Question History:	Previous NRC Exam Previous Test / Quiz	New New
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 2
10CFR Part 55 Content:	55.41.4 55.41.8	

Comments:

81-2

Question # RO 82

Examination Outline Cross-Reference	Level	RO
	Tier #	2
	Group #	2
	K/A #	290003 A1.05
	Importance Rating	3.2

Ability to predict and/or monitor changes in parameters associated with operating the CONTROL ROOM HVAC controls including: Radiation monitoring (control room)

Proposed Question:

Following a LOCA, the following conditions exist:

- Drywell pressure is 2.3 psig and slowly rising
- RPV water level is 140 inches and stable
- 2HVC*RE18B and 2HVC*RE18D are in alarm
- 2HVC*RE18A and 2HVC*RE18C are rising but have NOT alarmed

Which one of the following describes the HVC Special Filter Train (SFT) response and the required operator actions?

a. Verify the "B" SFT has automatically started. Place the control switch for the non-running SFT to Normal-After-Stop to keep it off.

b. Verify both the "A" and "B" SFT have automatically started. Ensure that one SFT is secured within 20 minutes of actuation.

c. Verify the "B" SFT has automatically started. Place the control switch for the "A" SFT to start. Within 8 hours, isolate the more contaminated east or west outside air intake path.

d. Verify both the "A" and "B" SFT have automatically started. Ensure that the more contaminated east or west outside outside air intake path is automatically closed.

Proposed Answer: b.

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Explanation (Justification of Distractors):

a. This would be true if LOCA signal was not present

c. Both trains start on a LOCA signal. The action to isolate the more contaminated intake path is a valid action after 8 hours.

d. The action to secure the more contaminated air intake path is an 8-hour requirement. This is a manually initiated required action.

Technical Reference(s): N2-OP-53A, Rev 08, Section D.19.0, Section H.1.0

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-288-2-02, 00, EO-4b, EO-4c, EO-6.0		
Question Source:	Bank #		
	Modified Bank #		
	New	New	
Question History:	Previous NRC Exam	New	
	Previous Test / Quiz	New	
Question Cognitive Level:	Memory of Fundamental Knowledge		
	Comprehension or Analysis		2
10CFR Part 55 Content:	55.41.5		
	55.45.5		

Comments:

Question # RO 83 SRO 79

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	2	2
	Group #	2	2
	K/A #	300000 K5.13	300000 K5.13
	Importance Rating	2.9	2.9

Knowledge of the operational implications of the following concepts as they apply to the instrument air system: Filters.

Proposed Question:

An Auxiliary Operator reports that the in-service Instrument Air System (IAS) prefilter d/p is 9.2 psid for 2IAS-FLT2A. The operator is directed to change the in-service filter to 2IAS-FLT2B.

The operator reports that 2IAS-V218, PREFILTER 2B INLET, will NOT open.

Assuming NO additional operator actions are taken, which one of the following describes the plant response?

a. When pressure downstream of the prefilter reaches 70 psig, the MSIVs close.

b. After airflow through the filter lowers to zero, the reactor scrams on RPV low water level.

c. When prefilter d/p reaches 10 psid, 2IAS-V298, AIR DRYERS 1A & 1B BYPASS, opens to maintain IAS pressure.

d. After airflow through the filter lowers to zero, 2IAS-AOV171, INSTR/SERV AIR CROSSTIE, closes and maintains IAS pressure.

Proposed Answer: b.

Explanation (Justification of Distractors):

a. The MSIVs will be maintained open by the accumulators. The reactor will scram on low RPV level when the feedwater minimum flow valves fail open on loss of air.

c. This is a manual valve.

d. AOV 171 will automatically close, but will not correct the problem since the problem is upstream of the valve.

Technical Reference(s): N2-SOP-19, Rev 00, Section 2.0

Proposed references to be provided to applicants during the examination:

None

Learning Objective:

Question Source:	Bank # Modified Bank # New	 New
Question History:	Previous NRC Exam Previous Test / Quiz	New New
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	1
10CFR Part 55 Content:	55.41.5 55.45.3	

Comments:

Question # RO 84

Examination Outline Cross-Reference	Level	RO
	Tier #	2
	Group #	3
	K/A #	215001 K4.01
	Importance Rating	3.4

Knowledge of Traversing In-Core Probe design feature(s) and/or interlocks which provide for the following: Primary containment isolation: Mark I&II (Not-BWR1)

Proposed Question:

Reactor Engineering is running TIP traces using the automatic mode. Four (4) TIPs are stowed in their shield chambers.

One TIP is out of its shield chamber and running into the core but has NOT reached the CORE TOP LIMIT. The low speed switch on the running TIP control panel is in the off position.

Which one of the following describes the automatic response of the TIP system to a PCIS isolation signal?

a. When a **group 3** signal is received, the TIP will retract in fast to the indexer, shift to slow, and the ball valve will close when it reaches the shield chamber.

b. When a **group 2** signal is received, the shear valve will fire and the ball valve will close leaving the TIP trapped in its guide tube and isolated from the secondary containment.

c. When a **group 2** signal is received, the TIP will stop where it is. When a confirmatory **group 3** isolation signal is received, the TIP retracts at fast speed until stowed, then the ball valve closes.

d. When a **group 3** signal is received, the TIP shifts to fast and when the CORE TOP LIMIT is reached, it will reverse and retract. The ball valve closes after a **group 2** signal is received and the TIP is stowed.

Proposed Answer: a.

Explanation (Justification of Distractors):
The TIP system only responds to a group 3 isolation signal.

Technical Reference(s):	N2-OP-83, Rev 03, Attachment 2

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-215-2-01, EO-7.0	
Question Source:	Bank #	Q8130
	Modified Bank #	
	New	
Question History:	Previous NRC Exam	
	Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge	1
	Comprehension or Analysis	
10CFR Part 55 Content:	55.41.7	

Comments:

Question # RO 85

Examination Outline Cross-Reference	Level	RO
	Tier #	2
	Group #	3
	K/A #	233000 K1.15
	Importance Rating	2.9

Knowledge of the physical connections and/or cause-effect relationships between FUEL POOL COOLING AND CLEAN-UP and the following: Storage pools.

Proposed Question:

With the plant operating at power, which one of the following describes the method utilized to maintain the desired SFC pool level?

a. A low fuel pool level signal causes pneumatic valves to open to provide makeup flow from the Makeup Water.

b. A low fuel pool level signal causes motor operated valves to open to provide makeup flow from the Condensate Transfer System.

c. A low skimmer surge tank level causes pneumatic valves to open to provide makeup flow from the Condensate Transfer System.

d. A low skimmer surge tank level causes motor operated valves to open to provide makeup flow from the Makeup Water System.

Proposed Answer: c.

Explanation (Justification of Distractors):
The normal method is automatic. When SFC skimmer surge tank level lowers pneumatic valves open to raise fuel pool level using makeup from the Condensate Transfer System.

Technical Reference(s): N2-ARP-01, Rev 00, 873317, 875117

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-233-2-00, EO-4a. EO-4b, EO-5	
Question Source:	Bank #	Q8833
	Modified Bank #	
	New	
Question History:	Previous NRC Exam	
	Previous Test / Quiz	Week 13 exam
Question Cognitive Level:	Memory of Fundamental Knowledge	1
	Comprehension or Analysis	
10CFR Part 55 Content:	55.41.7	

Comments:

Question # RO 86

Examination Outline Cross-Reference	Level	RO
	Tier #	2
	Group #	3
	K/A #	234000 A1.01
	Importance Rating	3.1

Ability to predict and/or monitor changes in parameters associated with operating the fuel handling equipment controls including: Spent fuel pool level.

Proposed Question:

Following a complete core offload, LPRMs are being changed out. After disconnecting the LPRM below vessel and installing the water seal cap, the water seal cap drain valve is left full open. The LPRM is removed from the reactor core.

It takes thirty (30) minutes before the water seal cap drain valve is closed. Which one of the following describes the effect on spent fuel pool level?

a. Lowers and continues to lower until the drain valve is closed.

b. Lowers and returns to normal after automatic makeup is initiated.

c. Remains the same because the LPRM guide tube is always dry.

d. Remains the same since the check valve in the drain line seats.

Proposed Answer: b.

Explanation (Justification of Distractors):
The flow from the reactor vessel to the drywell floor drains will be sufficient to cause the reactor cavity level and fuel pool level to lower.

a. Skimmer surge tank automatic makeup will restore level.
c. The SRM and IRM guide tubes are dry tubes. The LPRM guide tube is flooded.
d. There is no check valve installed when performing this evolution.

Technical Reference(s): N2-FHP-13

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-234-2-00, EO-8	
Question Source:	Bank # Modified Bank # New	 New
Question History:	Previous NRC Exam Previous Test / Quiz	New New
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 2
10CFR Part 55 Content:	55.41.5 55.45.5	

Comments:

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Question # RO 87 SRO 83

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	2	2
	Group #	3	3
	K/A #	290002 A2.04	290002 A2.04
	Importance Rating	3.7	4.1

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: Excessive heatup/cooldown rate.

Proposed Question:

A unit shutdown is in progress. After placing the B RHR loop into shutdown cooling per N2-OP-31, Residual Heat Removal System, it is determined that the cooldown rate is being exceeded.

Which one of the following describes the required operator action to reduce the cooldown rate?

a. Throttle open RHS*MOV40B, SDC B RETURN THROTTLE.

b. Throttle open RHS*MOV104, RHS B TO REACTOR HEAD SPRAY.

c. Throttle open RHS*MOV8B, HEAT EXCHANGER B INLET BYP VLV THROTTLE.

d. Throttle open SWP*MOV33B, HEAT EXCHANGER 1B SVCE WTR OUTLET VLV.

Proposed Answer: c.

Explanation (Justification of Distractors):
a.b.c. Raise cooldown rate.

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Technical Reference(s): N2-OP-31, Rev 13, F.6.20, F.6.26, F.6.27

Proposed references to be provided to applicants during the examination:

None.

Learning Objective: O2-OPS-001-205-2-00, EO-4a, EO-4b, EO-5

Question Source:	Bank #		
	Modified Bank #		
	New	New	
Question History:	Previous NRC Exam	New	
	Previous Test / Quiz	New	
Question Cognitive Level:	Memory of Fundamental Knowledge		1
	Comprehension or Analysis		
10CFR Part 55 Content:	55.41.5		
	55.45.6		

Comments:

87-2

Question #		RO 88	SRO 85
Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	-	-
	Group #	-	-
	K/A #	Generic 2.1.17	Generic 2.1.17
	Importance Rating	3.5	3.6

Ability to make accurate, clear and concise verbal reports.

Proposed Question:

The CRS has directed the ATC RO to report reactor water level. Which one of the following describes an acceptable communication by the ATC RO?

With NO repeat-back by the ATC RO and NO confirmation by the CRS, the ATC RO reports...

a. "approaching level 8."

b. "no change since last report."

c. "reactor water level is 200 inches."

d. "rising slowly at 2 inches per minute."

Proposed Answer: c.

Explanation (Justification of Distractors):

Directions for immediate reports do not require a repeat back provided the parameter is reported with the value. The CRS could request a parameter and the ATC RO could repeat back the request and upon confirmation then only report a value which is the case in the incorrect responses.

Technical Reference(s): Operations Manual, Section 3.4.2

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-276-2-00, EO 11		
Question Source:	Bank #		
	Modified Bank #		
	New	New	
Question History:	Previous NRC Exam	New	
	Previous Test / Quiz	New	
Question Cognitive Level:	Memory of Fundamental Knowledge		1
	Comprehension or Analysis		
10CFR Part 55 Content:	55.45.12		
	55.45.13		

Comments:

Question #		RO 89	SRO 86
Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	-	-
	Group #	-	-
	K/A #	Generic 2.1.16	Generic 2.1.16
	Importance Rating	2.9	2.8

Ability to operate plant phone, paging system, and two-way radio.

Proposed Question:

In response to a fire, the CSO positions the MERGE/UNIT 1 & 2 ISOLATE switch on the Main Page Party / Public Address System Control Console to MERGE, and then sounds the fire alarm. After the fire alarm terminates, the CSO announces the fire location.

Regarding the following areas (Unit 2, Unit 1, NMP Admin Bldg, and outside), which one of the following describes where the alarm and announcement are heard?

a. The alarm and announcement are only heard in Unit 2 and Unit 1.

b. The alarm and announcement are only heard in Unit 2, Unit 1, and outside areas.

c. The alarm and announcement are heard in Unit 2, Unit 1 NMP Admin Bldg and outside.

d. The alarm is only heard in Unit 2. The announcement is heard in Unit 2, Unit 1, and outside areas.

Proposed Answer: c.

Explanation (Justification of Distractors):
The alarm and announcement are heard in each area identified. Placing the Merge/Unit 1 & 2 Isolate switch to MERGE merges Unit 1 and Unit 2. Sounding the alarm automatically merges all system page lines.

Technical Reference(s): N2-OP-76, Rev 02, B.1.1

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-285-2-01, # 4a, #4b	
Question Source:	Bank # Modified Bank # New	 New
Question History:	Previous NRC Exam Previous Test / Quiz	New New
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	1
10CFR Part 55 Content:	55.41.10 55.45.12	

Comments:

Question # RO 90 SRO 87

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	-	-
	Group #	-	-
	K/A #	Generic 2.1.4	Generic 2.1.4
	Importance Rating	4.3	4.2

Ability to execute procedural steps.

Proposed Question:

The "A" RHR loop is being placed into suppression pool cooling to support RCIC Surveillance testing. After opening SWP*MOV90A, HEAT EXCHANGER 1A SVCE WTR INLET VLV, the Control Room E operator reports that the procedure cannot be continued as written.

Which one of the following is required to complete suppression pool cooling?

a. Stop and note the deficiency, complete the procedure for pool cooling, and then initiate a procedure change.

b. Place RHR back in standby. The procedure shall be changed using the procedure change process prior to placing RHR in pool cooling.

c. Stop action, make a pen and ink correction to the procedure, then complete the procedure for pool cooling. Initiate a procedure change after pool cooling is established.

d. Discontinue actions and leave all components operated in their current condition. The procedure shall be changed using the procedure change process prior to placing RHR in pool cooling.

Proposed Answer: b.

Explanation (Justification of Distractors):

a. A procedure change must be made first.
c. Pen and ink changes are initiated through the procedure change process.
d. Must be returned to a safe condition.

Technical Reference(s): NIP-PRO-01, Rev 06, Section 3.3.4

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-276-2-00, # 11		
Question Source:	Bank #		
	Modified Bank #		
	New	New	
Question History:	Previous NRC Exam	New	
	Previous Test / Quiz	New	
Question Cognitive Level:	Memory of Fundamental Knowledge		1
	Comprehension or Analysis		
10CFR Part 55 Content:	55.41.10		
	55.43.5		
	55.45.12		

Comments:

Question # RO 91

Examination Outline Cross-Reference	Level	RO
	Tier #	-
	Group #	-
	K/A #	Generic 2.2.30
	Importance Rating	3.5

Knowledge of RO duties in the control room during fuel handling such as alarms from fuel handling area.

Proposed Question:

The reactor core is being offloaded. Conditions are as follows:

- Reactor Mode Switch is in REFUEL position
- All control rods are fully inserted into the reactor core

Step 213 just unlatched in the fuel pool. The Main Hoist has NOT been raised.

Step 214 removal of a fuel assembly from the reactor, will be performed next.

Which one of the following describes when Annunciator 603442, CONTROL ROD OUT BLOCK is expected to alarm during the performance of **step 214**?

a. The Main Hoist is raised to the Normal-Up position in the spent fuel pool.

b. The Main Hoist has been lowered from the Normal-Up position over the reactor core location.

c. The fuel assembly has been grappled but Main Hoist raise motion has NOT been commanded.

d. The fuel assembly has been grappled and raised from its seated position in the reactor core.

Proposed Answer: d.

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Explanation (Justification of Distractors):

a. Rod Block is received when the refueling bridge is over the reactor core and the Main Hoist is loaded.

b. Rod Block is received when the refueling bridge is over the reactor core and the Main Hoist is loaded.

c. Rod Block is received when the refueling bridge is over the reactor core and the Main Hoist is loaded.

Technical Reference(s): N2-ARP-01, Rev 00, 603442, e.4
Tech Spec 3.9.2

Proposed references to be provided to applicants during the examination:

None

Learning Objective:	O2-OPS-001-234-2-01, EO-4c, EO-7b	
Question Source:	Bank # Modified Bank # New	 New
Question History:	Previous NRC Exam Previous Test / Quiz	New New
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 2
10CFR Part 55 Content:	55.45.12	

Comments:

91-2

Question # RO 92

Examination Outline Cross-Reference	Level	RO
	Tier #	-
	Group #	-
	K/A #	Generic 2.2.23
	Importance Rating	2.5
Ability to track limiting conditions for operations.		

Proposed Question:

A Limiting Condition for Operation (LCO) on RHR loop A is entered to support surveillance testing during the shift. The testing is completed and RHR Loop A is restored to OPERABLE status prior to the end of the shift.

Which one of the following describes where the short term LCO is required to be tracked?

a. CSO Log

b. SSS Log

c. Operability Log

d. Equipment Status Log

Proposed Answer: b.

Explanation (Justification of Distractors):

a. LCOs are not racked in the CSO log.

c. Operability Checklist is used for determining inoperability, not for tracking LCOs. Operability log is not used.

d. Equipment Status Log entry is only made for LCOs that have a duration longer than 1 shift.

Technical Reference(s): Conduct of Operations Manual, Section 3.7.5

Proposed references to be provided to applicants during the examination:

None.

Learning Objective: 03-OPS-006-343-3-01, EO-6

Question Source:	Bank #	
	Modified Bank #	
	New	New
Question History:	Previous NRC Exam	New
	Previous Test / Quiz	New
Question Cognitive Level:	Memory of Fundamental Knowledge	1
	Comprehension or Analysis	
10CFR Part 55 Content:	55.43.2	
	55.45.13	

Comments:

92-2

Question # RO 93

Examination Outline Cross-Reference	Level	RO
	Tier #	
	Group #	
	K/A #	Generic 2.2.1
	Importance Rating	3.7

Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.

Proposed Question:

The following conditions occur during a startup and heatup:

- The reactor is critical on range 5 of the IRMs
- Reactor period is 120 seconds and lowering
- Reactor coolant temperature is 180°F and rising
- As reactor coolant temperature continues to rise reactor period shortens

With CRS concurrence, which one of the following actions should be taken?

a. Immediately insert control rods to make the reactor subcritical.

b. Position control rods as necessary to maintain the temperature below 212°F.

c. Bypass the RWM and insert control rods assigned to the cram array to position 00.

d. When reactor period is 60 seconds insert the last withdrawn control rod to position 00.

Proposed Answer: a.

Explanation (Justification of Distractors):
Control room operators shall take conservative action when any unexpected situation occurs with respect to core reactivity or any other core abnormality as follows: Take immediate action to stabilize the reactor.
b., d., are not conservative for this situation.
c. Not permitted for this situation. Deviation from the sequence also requires SSS and Reactor Engineer permission.

Technical Reference(s): GAP-OPS-05, Rev 00

Proposed references to be provided to applicants during the examination:

None.

Learning Objective: 03-OPS-006-343-3-01, EO-2, EO-5

Question Source:	Bank #		
	Modified Bank #		
	New	New	
Question History:	Previous NRC Exam	New	
	Previous Test / Quiz	New	
Question Cognitive Level:	Memory of Fundamental Knowledge		
	Comprehension or Analysis		3
10CFR Part 55 Content:	55.45.1		

Comments:

93-2

Question # RO 94 SRO 94

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	-	-
	Group #	-	-
	K/A #	Generic 2.3.11	Generic 2.3.11
	Importance Rating	2.7	3.2
Ability to control radiation releases.			

Proposed Question:

The plant is at 100% power. Irradiated fuel is being arranged in the fuel pool to support receipt of new fuel when annunciator 851254, PROCESS AIRBORNE RADN MON ACTIVATED, is received.

DRMS indicates "red" for the following:

- 2HVR-CAB14A-1, HVR ABOVE REFUEL FLR
- 2HVR-CAB14B-1, HVR ABOVE REFUEL FLR

Which one of the following describes the required operator action(s)?

a. Manually isolate the above refuel floor ventilation dampers. Start GTS and unit cooler 2HVR*UC413B.

b. Manually isolate the above <u>and</u> below refuel floor ventilation dampers. Start GTS and unit cooler 2HVR*UC413B.

c. Verify the above refuel floor ventilation dampers are isolated and both GTS are operating. Start unit cooler 2HVR*UC413B.

d. Verify the above <u>and</u> below refuel floor ventilation dampers are isolated, both GTS and unit cooler 2HVR*UC413B are operating.

Proposed Answer: d.

Explanation (Justification of Distractors):

a. The above and below refuel floor ventilation dampers automatically close, and GTS and unit cooler 2HVR*413A automatically start.
b. The above and below refuel floor ventilation dampers automatically close, and GTS and unit cooler 2HVR*UC413A automatically start.
c. The below refuel floor ventilation dampers automatically close and unit cooler 2HVR*UC413A automatically starts.

Technical Reference(s): N2-ARP-01, Rev 00, annunciator 851254

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-288-2-03, # 4c, #7d, #7e	
Question Source:	Bank # Modified Bank # New	 New
Question History:	Previous NRC Exam Previous Test / Quiz	New New
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 2
10CFR Part 55 Content:	55.45.9 55.45.10	

Comments:

Question # RO 95 SRO 95

		RO	SRO
Examination Outline Cross-Reference	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	Generic 2.3.9	Generic 2.3.9
	Importance Rating	2.5	3.4

Knowledge of the process for performing a containment purge.

Proposed Question:

Following an accident, it is necessary to purge the suppression chamber with nitrogen using EOP-6, Attachment 25, Containment Purging. Suppression pool level is 203 feet.

Which one of the following describes how the gasses in the drywell atmosphere are vented to the main stack when performing this procedure?

a. After nitrogen is aligned to the suppression chamber and the suppression chamber is being vented, the drywell purge outlet valves are opened.

b. After nitrogen is aligned to the suppression chamber and the suppression chamber is being vented, nitrogen is aligned to the drywell and the drywell purge outlet valves are opened.

c. When drywell pressure is at least 5 psig higher than suppression chamber pressure, the drywell purge outlet valves and then the suppression chamber purge outlet valves are opened.

d. After the drywell purge inlet and suppression chamber purge outlet valves are open, drywell pressure rises and gasses vent through the downcomers to the suppression chamber.

Proposed Answer: d.

Explanation (Justification of Distractors):
When purging the containment from the suppression chamber, N2 is added to the drywell and the suppression chamber is vented to the main stack through GTS. As N2 is added to the drywell, dryell pressure rises until it is sufficient to overcome the hydrostatic head of water in the downcoomers (approximately 5 psig). The drywell atmosphere will then vent to the suppression chamber and out the stack. When performing this procedure, no N2 is added to the suppression chamber and the drywell purge outlet valves are not opened.

Technical Reference(s): N2-EOP-6, Attachment 25, Section 3.3

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-223-2-03, EO-4a, EO-4b	
Question Source:	Bank #	
	Modified Bank #	
	New	New
Question History:	Previous NRC Exam	New
	Previous Test / Quiz	New
Question Cognitive Level:	Memory of Fundamental Knowledge	1
	Comprehension or Analysis	
10CFR Part 55 Content:	55.41.9	
	55.43.4	
	55.45.10	

Comments:

95-2

Question # RO 96 SRO 96

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	Generic 2.3.1	Generic 2.3.1
	Importance Rating	2.6	3.0

Knowledge of 10CFR: 20 and related facility radiation control requirements.

Proposed Question:

An auxiliary operator receives 28 mrem while performing an on-shift evolution.

Which one of the following describes the required action to permit performance of the evolution again on this shift?

a. Read the operators TLD to confirm the dose.

b. Survey the area and verify radiological postings.

c. Assess the task for ways to reduce radiation exposure.

d. Obtain permission from the General Supervisor Operations.

Proposed Answer: c. The required action is to perform a self-assessment to identify improvements.

Explanation (Justification of Distractors):

a. This is not a requirement for this exposure. The daily limit is 50 mrem.
b. This is not a requirement for this exposure. The daily limit is 50 mrem.
d. The only requirement is to perform a self-assessment to identify improvements.

Technical Reference(s): Operations Manual, Section 3.2.4

Proposed references to be provided to applicants during the examination:

None.

Learning Objective: 03-OPS-006-343-3-01, EO-4, EO-5

Question Source:	Bank #		
	Modified Bank #		
	New	New	
Question History:	Previous NRC Exam	New	
	Previous Test / Quiz	New	
Question Cognitive Level:	Memory of Fundamental Knowledge		1
	Comprehension or Analysis		
10CFR Part 55 Content:	55.41.12		
	55.43.3		
	55.45.9		
	55.45.10		

Comments:

96-2

Question # RO 97

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	-	-
	Group #	-	-
	K/A #	Generic 2.3.2	-
			-
	Importance Rating	2.5	-
Knowledge of facility ALARA program.			

Proposed Question:

Which one of the following meets the requirements for the SSS to waive the independent verification of a markup?

a. Markup is in a **high radiation area** and can only be applied by a licensed reactor operator.

b. Markup is in a **high radiation area** and can be applied by either an auxiliary operator <u>or</u> a licensed reactor operator.

c. Markup is in a **radiation area** with an expected exposure of ≥15 mrem and can only be applied by a licensed reactor operator.

d. Markup is in a **radiation area** with an expected exposure of ≥15 mrem and can be applied by either an auxiliary operator <u>or</u> a licensed reactor operator.

Proposed Answer: a.

Explanation (Justification of Distractors):
For high radiation areas, the SSS may waive the independent verification provided a licensed reactor operator applies the markup. Waivers of independent verification are not permitted in a radiation area.

Technical Reference(s): GAP-OPS-02, Rev 10, Section 3.4.2

Proposed references to be provided to applicants during the examination:

None.

Learning Objective: 03-OPS-006-343-3-01, EO-7

Question Source:	Bank #		
	Modified Bank #		
	New	New	
Question History:	Previous NRC Exam	New	
	Previous Test / Quiz	New	
Question Cognitive Level:	Memory of Fundamental Knowledge		1
	Comprehension or Analysis		
10CFR Part 55 Content:	55.41.12		
	55.43.4		
	55.45.9		
	55.45.10		

Comments:

97-2

Question # RO 98 SRO 98

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #	Generic	Generic
	Group #	-	-
	K/A #	2.4.32	2.4.32
	Importance Rating	3.3	3.5
Knowledge of operator response to loss of all annunciators.			

Proposed Question:

Following annunciator testing of control room panel 2CEC*PNL603, its annunciators are locked in the fast flash mode. Which one of the following describes the immediate operator actions?

a. Station a licensed operator to continuously monitor all control room panels.

b. Notify the Fire Chief to initiate increased monitoring of the Fire System status.

c. Station a licensed operator to continuously monitor the affected control room panels.

d. Direct a licensed operator to start a new set of rounds as another operator is completing the rounds in affected areas of the plant.

Proposed Answer: c.

Explanation (Justification of Distractors):

a. Monitoring is only required for the panels which have lost annunciators.
b. This action is only required if annunciators are lost for 2CEC-PNL849.
d. This is a subsequent operator action.

Technical Reference(s): N2-SOP-91, Rev 00, 3.0

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-006-SOP-2-01-29, # 2	
Question Source:	Bank # Modified Bank # New	 New
Question History:	Previous NRC Exam Previous Test / Quiz	New New
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	1
10CFR Part 55 Content:	55.41.10 55.43.5 55.45.13	

Comments:

Question # RO 99

Examination Outline Cross-Reference	Level	RO
	Tier #	-
	Group #	-
	K/A #	Generic 2.4.19
	Importance Rating	2.7
Knowledge of EOP layout / symbols / and icons.		

Proposed Question:

While performing N2-EOP-6, Attachment 14, Alternate Control Rod Insertions, the following step is encountered.

3.3.3 Defeat RPS interlocks as follows:

T

- Install EOP Jumper #21 from fuse C72A-F14A to terminal B on relay C72A-K12E in 2CEC*PNL609, Bay A (Control Room) . . . . (__)

Which one of the following describes what is indicated by the "T" that is in the left margin adjacent to this step?

a. Indicates a temporary alteration.

b. Indicates the use of a jumper for the alteration.

c. Indicates the time that the alteration is made must be recorded.

d. Indicates the TSC must authorize performance of the alteration.

Proposed Answer: b.

Explanation (Justification of Distractors):

A "T" within a circle notation in the left margin adjacent to the step or note indicates that a tool or material is required for performance.

Technical Reference(s): N2-EOP-6, Rev 05, Precaution and Limitation 9.0

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-006-344-2-22, EO-4.0		
Question Source:	Bank # Modified Bank # New	 New	
Question History:	Previous NRC Exam Previous Test / Quiz	New New	
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis		1
10CFR Part 55 Content:	55.41.10 55.45.13		

Comments:

99-2

Question # RO 100 SRO 100

Examination Outline Cross-Reference	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	Generic 2.4.21	Generic 2.4.21
	Importance Rating	3.7	4.3

Knowledge of the parameters and logic used to assess the status of safety functions including:
1. Reactivity control
2. Core cooling and heat removal
3. Reactor coolant system integrity
4. Containment conditions
5. Radioactivity release control.

Proposed Question:

The Safety Parameter Display System (SPDS) is selected to indicate SAFETY FUNCTION STATUS.

Which one of the following describes how an operator is alerted that drywell pressure is at 2.0 psig?

a. Only the Level 2 Safety Status Indicator for CONTAINMENT INTEGRITY is **red**.

b. Only the Level 2 Safety Status Indicator for CONTAINMENT INTEGRITY is **yellow**.

c. The parameter and the Level 2 Safety Status Indicator for CONTAINMENT INTEGRITY are **red**.

d. The parameter and the Level 2 Safety Status Indicator for CONTAINMENT INTEGRITY are **yellow**.

Proposed Answer: d.

Explanation (Justification of Distractors):
a. The parameter is abnormal and will only be yellow.
b. The parameter is also yellow.
c. The parameter is abnormal and will only be yellow.

Technical Reference(s): N2-OP-91B, Rev 02, Section B.2, B.3, Attachment 5.

Proposed references to be provided to applicants during the examination:

None.

Learning Objective: 02-OPS-001-226-2-02, EO-4c, EO-9

Question Source:	Bank #		
	Modified Bank #		
	New	New	
Question History:	Previous NRC Exam	New	
	Previous Test / Quiz	New	
Question Cognitive Level:	Memory of Fundamental Knowledge		1
	Comprehension or Analysis		
10CFR Part 55 Content:	55.41.4		
	55.43.5		
	55.45.12		

Comments:

SRO ONLY

Question # SRO 2

Examination Outline Cross-Reference	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295003 2.2.22
	Importance Rating	4.1
Knowledge of limiting conditions for operations and safety limits		

Proposed Question:

The plant is at 70% power and continuing to raise power following a refueling outage. The following events occur:

Day 1, 1100 hours - A fault occurs on the 115KV Offsite Line 6. LCO entered on one offsite circuit inoperable.

Day 1, 1400 hours - 2ENS*SWG103 was transferred to the Auxiliary Boiler Service Transformer, 2ABS-X1.

Day 1, 2200 hours - Diagnosis of the fault determined it was caused by the wrong contacts (undersized) installed in Motor Operated Disconnect 2YUL-MDS2.

Day 2, 0100 hours - Further investigation reveals that these same contacts where installed in 115KV Offsite Line 5 Motor Operated Disconnect 2YUL-MDS1.

Which one of the following Technical Specifications actions is required?

a. Restore Line 6 to service by Day 4, 1100 hours, then remove Line 5 from service, then enter the LCO for one offsite circuit inoperable.

b. Remove Line 5 from service, then enter the LCO for two offsite circuits inoperable, restore Line 6 within 24 hours of removing Line 5 from service.

c. Declare Line 5 inoperable at Day 2, 0100 hours, restore Line 6 by Day 3, 0100 hours, then remove Line 5 from service and restore Line 5 by Day 4, 1100.

d. Enter the LCO for two offsite circuits inoperable at Day 2, 0100 hours, Return Line 5 to operability by Day 3, 1100 hours, and Line 6 by Day 8, 0100 hours.

Proposed Answer: c. Line 5 must be declared inoperable per the guidance in Sect 3.8 of the Conduct of Ops this places the plant in a 24 hour LCO until one line can be restored.

Explanation (Justification of Distractors):

a. Line 5 must be declared inoperable for a common mode failure, so the situation becomes a 24 hour LCO.
b. Can't remove Line 5 from service with Line 6 already out, and Line 5 is considered inoperable at Day 2, 0100.
d. Line 5 and/or Line 6 must be returned to operable by Day 4,1100 provided one is restored to operable by Day 3, 0100.

Technical Reference(s): Technical Specifications 3.8.1
Conduct of Operations Manual, Section 3.8

Proposed references to be provided to applicants during the examination:

Technical Specifications 3.8.1

Learning Objective: 02-OPS-001-262-2-01, EO-10.0

Question Source: Bank #
Modified Bank #
New New

Question History: Previous NRC Exam
Previous Test / Quiz

Question Cognitive Level: Memory of Fundamental Knowledge
Comprehension or Analysis 2

10CFR Part 55 Content: 43.2 / 45.2

Comments:
SRO Only: Technical Specification application.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 AC SOURCES

AC SOURCES - OPERATING

LIMITING CONDITIONS FOR OPERATION

3.8.1.1 As a minimum, the following AC electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Three separate and independent diesel generators, each with:

1. Separate day fuel tanks containing a minimum of 403 gallons of fuel for EDG*1 (Division I) and EDG*3 (Division II), and 282 gallons for EDG*2 (HPCS-Division III)

2. A separate fuel storage system containing a minimum of 47,547 gallons of fuel for EDG*1 (Division I) and EDG*3 (Division II), and 35,342 gallons for EDG*2 (HPCS-Division III), and

3. Two fuel oil transfer pumps.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one offsite circuit of the above required AC electrical power sources inoperable, demonstrate the OPERABILITY of the remaining AC sources by performing Surveillance Requirements 4.8.1.1.1 within 1 hour and at least once every 8 hours thereafter. Restore the offsite circuit to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With either diesel generator EDG*1 or EDG*3 inoperable, demonstrate the OPERABILITY of the above required AC offsite sources by performing Surveillance Requirement 4.8.1.1.1 within 1 hour and at least once every 8 hours thereafter. If the diesel generator became inoperable from any cause other than preplanned maintenance or testing, within 24 hours, for each OPERABLE diesel generator separately, either verify that the cause of the diesel generator being inoperable does not impact the OPERABILITY of the OPERABLE diesel generator or perform Surveillance Requirement 4.8.1.1.2.a.4*. Restore the inoperable diesel generator to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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* This is required to be completed regardless of when the inoperable diesel generator is restored to OPERABLE status. The provisions of Specification 3.0.2 are not applicable.

AC SOURCES

AC SOURCES - OPERATING

LIMITING CONDITIONS FOR OPERATION

3.8.1.1 (Continued)

ACTION:

c. With one offsite circuit of the above required AC sources and diesel generator EDG*1 or EDG*3 of the above required AC electrical power sources inoperable, demonstrate the OPERABILITY of the remaining AC sources by performing Surveillance Requirement 4.8.1.1.1 within 1 hour and at least once every 8 hours thereafter. If a diesel generator became inoperable from any cause other than preplanned maintenance or testing, within 8 hours, for each OPERABLE diesel generator separately, either verify that the cause of the diesel generator being inoperable does not impact the OPERABILITY of the OPERABLE diesel generator or perform Surveillance Requirement 4.8.1.1.2.a.4*. Restore at least one of the inoperable AC sources to OPERABLE status within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Restore at least two offsite circuits and diesel generators EDG*1 and EDG*3 to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With diesel generator EDG*2 of the above required AC electrical power sources inoperable, demonstrate the OPERABILITY of the offsite AC sources by performing Surveillance Requirement 4.8.1.1.1 within 1 hour and at least once every 8 hours thereafter. If the diesel generator becomes inoperable as a result of any cause other than preplanned maintenance or testing, within 24 hours, for each OPERABLE diesel generator separately, either verify that the cause of the diesel generator being inoperable does not impact the OPERABILITY of the OPERABLE diesel generator or perform Surveillance Requirement 4.8.1.1.2.a.4*. Restore diesel generator EDG*2 to OPERABLE status within 72 hours or declare the HPCS inoperable and take the ACTION required by Specification 3.5.1.

* This is required to be completed regardless of when the inoperable diesel generator is restored to OPERABLE status. The provisions of Specification 3.0.2 are not applicable.

LIMITING CONDITIONS FOR OPERATION

3.8.1.1 (Continued)

ACTION:

e. With diesel generator EDG*1 or EDG*3 of the above required AC electrical power sources inoperable, in addition to taking ACTION b or c, as applicable, verify within 2 hours that all required redundant systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE; otherwise, either declare inoperable the redundant systems, subsystems, trains, components and devices served by the inoperable diesel generator and take the ACTION required by the associated specification(s) for both divisional systems, subsystems, trains, components or devices inoperable or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

f. With both of the above required offsite circuits inoperable, restore at least one of the above required offsite circuits to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours. With only one offsite circuit restored to OPERABLE status, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

g. With diesel generators EDG*1 and EDG*3 of the above required AC electrical power sources inoperable, demonstrate the OPERABILITY of the remaining AC sources by performing Surveillance Requirement 4.8.1.1.1 within 1 hour and at least once every 8 hours thereafter and, within 8 hours, either verify that the cause(s) of diesel generators EDG*1 and EDG*3 being inoperable do not impact the OPERABILITY of diesel generator EDG*2 or perform Surveillance Requirement 4.8.1.1.2.a.4* for diesel generator EDG*2. Restore at least one of the inoperable diesel generators EDG*1 and EDG*3 to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Restore both diesel generators EDG*1 and EDG*3 to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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* This is required to be completed regardless of when the inoperable diesel generator is restored to OPERABLE status. The provisions of Specification 3.0.2 are not applicable.

LIMITING CONDITIONS FOR OPERATION

3.8.1.1 (Continued)

ACTION:

h. With one offsite circuit of the above-required AC electrical power sources inoperable and diesel generator EDG*2 inoperable, apply the requirements of ACTIONS a and d specified above.

i. With either diesel generator EDG*1 or EDG*3 inoperable and diesel generator EDG*2 inoperable, apply the requirements of ACTIONS b, d, and e specified above.

j. With one or more diesel fuel storage tank(s) containing less than the minimum quantity of fuel oil but greater than or equal to 40,755 gallons of fuel for EDG*1 and EDG*3, or greater than or equal to 30,293 gallons for EDG*2, restore fuel oil to required levels within 48 hours or declare the affected diesel generator(s) inoperable.

k. With one or more diesel generator(s) with new diesel fuel oil properties not within limits, restore stored fuel oil properties to required limits within 30 days or declare the affected diesel generator(s) inoperable.

l. With one or more diesel generator(s) with stored fuel total particulates not within limits, restore stored fuel total particulates to required limits within 7 days or declare the affected diesel generator(s) inoperable.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be determined OPERABLE at least once every 7 days by verifying correct breaker alignments and indicated power availability.

4.8.1.1.2 Each of the above required diesel generators shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8.1.1.2-1 on a STAGGERED TEST BASIS by:

   1. Verifying the fuel level in the day fuel tank.

   2. Verifying the fuel level in the fuel storage tank.

SURVEILLANCE REQUIREMENTS

4.8.1.1.2.a (Continued)

3. Verifying each fuel transfer pump starts and transfers fuel from the storage system to the day fuel tank.

4. Verifying that on a start from ambient conditions:

   a) That diesel engines EDG*1 and EDG*3 accelerate to at least 600 rpm in less than or equal to 10 seconds.* The generator voltage and frequency shall be 4160 ± 416 volts and 60 ± 3.0 Hz within 10 seconds and 4160 ± 416 volts and 60 ± 1.2 Hz within 13 seconds after the start signal.

   b) That diesel engine EDG*2 accelerates to at least 870 rpm and at least 3750 volts in less than or equal to 10 seconds.* The generator voltage and frequency shall be 4160 ± 416 volts and 60 ± 1.2 Hz within 15 seconds after the start signal.

   c) Each diesel generator shall be started for this test by using one of the following signals:

      1) Manual.

      2) Simulated loss of offsite power by itself.

      3) Simulated loss of offsite power in conjunction with an ESF actuation test signal.

      4) An ESF actuation test signal by itself.

5. Verifying that after the diesel generator is synchronized, it is loaded to greater than or equal to 4400 KW for diesel generators EDG*1 and EDG*3 and greater than or equal to 2600 KW for diesel generator EDG*2 in less than or equal to 90 seconds* and operates with these loads for at least 60 minutes.

6. Verifying the diesel generator is aligned to provide standby power to the associated emergency buses.

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* All diesel generator starts for the purpose of this surveillance test may be preceded by an engine prelube period. Further, all surveillance tests, with the exception of once per 184 days, may also be preceded by warmup procedures and may also include gradual loading as recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

SURVEILLANCE REQUIREMENTS

4.8.1.1.2.a (Continued)

7. Verifying the pressure in diesel generator air start receivers for EDG*1 and EDG*3 to be greater than or equal to 225 psig.

8. Verifying the pressure in diesel generator air start receivers for EDG*2 to be greater than or equal to 190 psig.

b. By removing accumulated water:

1. From the day tank at least once per 31 days and after each occasion when the diesel is operated for more than 1 hour, and

2. From the storage tank at least once per 31 days.

c. By verifying fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.

d. Deleted.

SURVEILLANCE REQUIREMENTS

4.8.1.1.2 (Continued)

e. At least once per 18 months,* during shutdown, by:

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service.

2. Verifying the diesel generator capability to reject a load of greater than or equal to 1125 kW for diesel generator EDG*1, greater than or equal to 750 kW for diesel generator EDG*3, and greater than or equal to 2433 kW for diesel generator EDG*2 while maintaining engine speed increase less than or equal to 75% of the difference between nominal speed and the overspeed trip setpoint or 15% of nominal, whichever is less.

3. Verifying the diesel generator capability to reject a load of 4400 kW for diesel generators EDG*1 and EDG*3 and 2600 kW for diesel generator EDG*2 without tripping.** The generator voltage shall not exceed 4576 volts for EDG*1 and EDG*3, and 5824 volts for EDG*2 during and following the load rejection.

4. Simulating a loss of offsite power by itself, and:

    a) For Divisions I and II:

        1) Verifying deenergization of the emergency buses and load shedding from the emergency buses.

        2) Verifying the diesel generator starts*** on the autostart signal, energizes the emergency buses with permanently connected loads within 13 seconds†, energizes the auto-connected (shutdown) loads through the load timers and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency buses shall be maintained at 4160 ± 416 volts and 60 ± 1.2 Hz during this test.

* For any start of a diesel, the diesel must be operated with a load in accordance with the manufacturer's recommendations.

** Momentary transients due to changing bus loads shall not invalidate the test.

*** All diesel generator starts for the purpose of this surveillance test may be preceded by an engine prelube period. Further, all surveillance tests, with the exception of once per 184 days, may also be preceded by warmup procedures and may also include gradual loading as recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

† From initiation of loss of offsite power.

ELECTRICAL POWER SYSTEMS

AC SOURCES

AC SOURCES - OPERATING

SURVEILLANCE REQUIREMENTS

4.8.1.1.2.e.4 (Continued)

b) For Division III:

1) Verifying deenergization of the emergency bus.

2) Verifying the diesel generator starts* on the autostart signal, energizes the emergency bus with the permanently connected loads within 13 seconds** and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency bus shall be maintained at 4160 ± 416 volts and 60 ± 1.2 Hz during this test.

5. Verifying that on an ECCS actuation test signal, without loss of offsite power:

a) That diesel generators EDG*1 and EDG*3 start* on the autostart signal and operate on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be 4160 ± 416 volts and 60 ± 3.0 Hz within 10 seconds and 4160 ± 416 volts and 60 ± 1.2 Hz within 13 seconds after the autostart signal; the steady state generator voltage and frequency shall be maintained within these limits during this test.

b) That diesel generator EDG*2 starts* on the autostart signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be 4160 ± 416 volts and 60 ± 1.2 Hz within 15 seconds after the autostart signal; the steady state generator voltage and frequency shall be maintained within these limits during this test.

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* All diesel generator starts for the purpose of this surveillance test may be preceded by an engine prelube period. Furthermore all surveillance tests, with the exception of once per 184 days, may also be preceded by warmup procedures and may also include gradual loading as recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

** From initiation of loss of offsite power.

SURVEILLANCE REQUIREMENTS

4.8.1.1.2.e (Continued)

6. Simulating a loss of offsite power in conjunction with an ECCS actuation test signal, and:

   a) For Divisions I and II:

      1) Verifying deenergization of the emergency buses and loads shedding from the emergency buses.

      2) Verifying the diesel generator starts* on the autostart signal, energizes the emergency buses with permanently connected loads within 10 seconds, energizes the auto-connected (shutdown) loads through the load timers, and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady-state voltage and frequency of the emergency buses shall be maintained at 4160 ± 416 volts and 60 ± 1.2 Hz during this test.

   b) For Division III:

      1) Verifying deenergization of the emergency bus.

      2) Verifying the diesel generator starts* on the autostart signal, energizes the emergency bus with the permanently connected loads and the auto-connected emergency loads within 10 seconds and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady-state voltage and frequency of the emergency bus shall be maintained at 4160 ± 416 volts and 60 ± 1.2 Hz during this test.

7. Verifying that all automatic diesel generator trips are automatically bypassed upon loss of voltage on the emergency bus concurrent with an ECCS actuation signal except engine overspeed trip and generator differential trip.

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* All diesel generator starts for the purpose of this surveillance test may be preceded by an engine prelube period. Furthermore, all surveillance tests, with the exception of once per 184 days, may also be preceded by warmup procedures and may also include gradual loading as recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

4.8.1.1.2.e (Continued)

8. Verify the diesel generator operates for at least 24 hours.†

   a) For Divisions I and II:

      During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 4840 kW*. During the remaining 22 hours of this test, the diesel generator shall be loaded to greater than or equal to 4400 kW*. The generator voltage and frequency shall be 4160 ± 416 volts and 60 ± 3.0 Hz within 10 seconds and 4160 ± 416 volts and 60 ± 1.2 Hz within 13 seconds after the start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test.

   b) For Division III:

      During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 2860 kW*. During the remaining 22 hours of this test, the diesel generator shall be loaded to greater than or equal to 2600 kW*. The generator voltage and frequency shall be 4160 ± 416 volts and 60 ± 1.2 Hz within 15 seconds after the start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test.

9. Verifying that the autoconnected loads to each diesel generator do not exceed the 2000-hour rating of 4750 kW for diesel generators EDG*1 and EDG*3 and 2850 kW for diesel generator EDG*2.

10. Verifying the diesel generator's capability to:

   a) Manually synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

   b) Transfer its loads to the offsite power source, and

   c) Be restored to its standby status.

11. Verifying that with the diesel generator operating in a test mode and connected to its bus, a simulated ECCS actuation signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizes the emergency loads with offsite power.

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* Momentary transients due to changing bus loads shall not invalidate the test.

† This test may be performed during power operation provided that the other two diesel generators are operable. Should either of the two diesel generators become inoperable, the test will be aborted.

SURVEILLANCE REQUIREMENTS

4.8.1.1.2.e (Continued)

12. Verifying that the automatic load timer relays are OPERABLE with the interval between each load block within ± 10% of its design interval for diesel generators EDG*1 and EDG*3.

13. Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

    a) For Divisions I and II, turning gear engaged and emergency stop.

    b) For Division III, engine in the maintenance mode and diesel generator lockout.

f. At least once per 18 months verify each diesel generator starts and accelerates to at least 600 RPM within 10 seconds for EDG*1 and EDG*3, and 870 RPM within 10 seconds for EDG*2. The generator voltage and frequency for EDG*1 and EDG*3 shall be 4160 ± 416 volts and 60 ± 3.0 Hz within 10 seconds and 4160 ± 416 volts and 60 ± 1.2 Hz within 13 seconds after the start signal. The generator voltage and frequency for EDG*2 shall be 4160 ± 416 volts and 60 ± 1.2 Hz within 15 seconds after the start signal. This test shall be performed within 5 minutes of shutting down the diesel generator after the diesel generator has operated for at least 2 hours at 4400 kW or more for EDG*1 and EDG*3 and 2600 kW or more for EDG*2. For any start of a diesel, the diesel must be loaded in accordance with manufacturer's recommendations. Momentary transients due to changing bus loads shall not invalidate this test.

g. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting all three diesel generators simultaneously, during shutdown, and verifying that all diesel generators EDG*1 and EDG*3 accelerate to at least 600 rpm and EDG*2 accelerates to at least 870 rpm in less than or equal to 10 seconds.

h. At least once per 10 years by:

    1. Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution, and

    2. Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code in accordance with ASME Code Section XI Article IWD-5000.

4.8.1.1.3 All diesel generator failures, valid or non-valid, shall be reported to the Commission pursuant to Specification 6.9.2, within 30 days. Reports of diesel generator failures shall include the information recommended in Position C.3.b of RG 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests, on a per nuclear unit basis, is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Position C.3.b of RG 1.108, Revision 1, August 1977.

TABLE 4.8.1.1.2-1

DIESEL GENERATOR TEST SCHEDULE

NUMBER OF FAILURES IN LAST 20 VALID TESTS*	NUMBER OF FAILURES IN LAST 100 VALID TESTS*	TEST FREQUENCY
≤ 1	≤ 4	At least once per 31 days
≥ 2**	≥ 5	At least once per 7 days

* Criteria for determining number of failures and number of valid tests shall be in accordance with Position C.2.e of RG 1.108, but determined on a per diesel generator basis.

For the purposes of determining the required test frequency, the previous test failure count may be reduced to zero if a complete diesel overhaul to like-new condition is completed, provided that the overhaul, including appropriate postmaintenance operation and testing, is specifically approved by the manufacturer <u>and</u> if acceptable reliability has been demonstrated. The reliability criterion shall be the successful completion of 14 consecutive tests in a single series. Ten of these tests shall be in accordance with the routine Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 and four tests in accordance with the 184-day testing requirement of Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5. If this criterion is not satisfied during the first series of tests, any alternate criterion to be used to transvalue the failure count to zero requires NRC approval.

** The associated test frequency shall be maintained until seven consecutive failure-free demands have been performed <u>and</u> the number of failures in the last 20 valid demands has been reduced to 1.

AC SOURCES - SHUTDOWN

LIMITING CONDITIONS FOR OPERATION

3.8.1.2 As a minimum, the following AC electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. Diesel generator EDG*1 or EDG*3, and diesel generator EDG*2 when the HPCS system is required to be OPERABLE, with each diesel generator having:

   1. Separate day fuel tanks containing a minimum of 403 gallons of fuel for EDG*1 (Division I) and EDG*3 (Division II) and 282 gallons for EDG*2 (HPCS-Division III).

   2. A separate fuel storage system containing a minimum of 47,547 gallons of fuel for EDG*1 (Division I) and EDG*3 (Division II) and 35,342 gallons of fuel for EDG*2 (HPCS-Division III).

   3. Two fuel oil transfer pumps.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and *.

ACTION:

a. With less than the the above required AC electrical power sources OPERABLE, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment, operations with a potential for draining the reactor vessel and crane operations over the spent fuel storage pool when fuel assemblies are stored therein. In addition, in OPERATIONAL CONDITION 5, with the water level less than 22 feet 3 inches above the reactor pressure vessel flange, immediately initiate corrective action to restore the required power sources to OPERABLE status as soon as practical.

b. With diesel generator EDG*2 of the above required AC electrical power sources inoperable, restore the inoperable diesel generator to OPERABLE status within 72 hours or declare the HPCS system inoperable and take the ACTION required by Specification 3.5.2 and 3.5.3

c. The provisions of Specification 3.0.3 are not applicable.

d. With one or more diesel fuel storage tank(s) containing less than the minimum quantity of fuel oil but greater than or equal to 40,755 gallons of fuel for EDG*1 and EDG*3, or greater than or equal to 30,293 gallons for EDG*2, restore fuel oil to required levels within 48 hours or declare the affected diesel generator(s) inoperable.

---

* When handling irradiated fuel in the secondary containment.

AC SOURCES - SHUTDOWN

LIMITING CONDITIONS FOR OPERATION

3.8.1.2 (Continued)

e. With one or more diesel generator(s) with new diesel fuel oil properties not within limits, restore stored fuel oil properties to required limits within 30 days or declare the affected diesel generator(s) inoperable.

f. With one or more diesel generator(s) with stored fuel total particulates not within limits, restore stored fuel total particulates to required limits within 7 days or declare the affected diesel generator(s) inoperable.

SURVEILLANCE REQUIREMENTS

4.8.1.2 At least the above required AC electrical power sources shall be demonstrated OPERABLE per Surveillance Requirements 4.8.1.1.1, 4.8.1.1.2, and 4.8.1.1.3, except for the requirement of 4.8.1.1.2.a.5.

Question # SRO 4

Examination Outline Cross-Reference	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295006 AK2.07
	Importance Rating	4.1
Knowledge of the interrelations between SCRAM and the following: reactor pressure control.		

Proposed Question:

The plant is operating at 100% power when a feedwater control malfunction causes an RPV high level that trips the reactor feedwater pumps.

Which one of the following describes how reactor pressure is controlled for the next few minutes? **ASSUME NO OPERATOR ACTIONS**

a. **No** SRVs open, turbine bypass valves fully open then throttle to maintain reactor pressure.

b. Several SRVs open then sequentially close, one or two SRVs remain open to control reactor pressure.

c. Several SRVs open then close as turbine bypass valves fully open then throttle to maintain reactor pressure.

d. **No** SRVs open, turbine bypass valves fully open and remain open to control reactor pressure.

Proposed Answer: c. TCVs Close , SRVs open initially open then close as BPV control pressure.

Explanation (Justification of Distractors):
See justification for proposed answer above.

S2-1

Technical Reference(s):

USAR, 15.2.3.2
N2-SOP-21, TURBINE TRIP

Proposed references to be provided to applicants during the examination:

None

Learning Objective: 02-OPS-001-245-2-01, EO-4a, EO-5, EO-8, EO-10

Question Source:	Bank #	
	Modified Bank #	
	New	NEW
Question History:	Previous NRC Exam	
	Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge	
	Comprehension or Analysis	1
10CFR Part 55 Content:	41.7/ 45.8	

Comments:

Question # SRO 6

Examination Outline Cross-Reference	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295007 AA1.04
	Importance Rating	3.9

Ability to operate and/or monitor the following as they apply to High Reactor Pressure: Safety/relief valve operation: Plant-Specific.

Proposed Question:

The plant is operating at full power when all MSIVs close. All control rods fully insert into the reactor. Reactor pressure rises to 1128 psig.

Assume that the Safety Relief Valves (SRVs) function at their design set point (± 0.0 psig).

Which one of the following describes how many SRVs will open?

a. Two (2)

b. Six (6)

c. Ten (10)

d. Fourteen (14)

Proposed Answer: c.

Explanation (Justification of Distractors):

1128 causes SRV Groups 1,2 & 3 to lift, 10 SRVs

SRV GROUP	NO. of SRVs	SETPOINT
1	2	1103 psig
2	4	1113 psig
3	4	1123 psig
4	4	1133 psig
5	4	1143 psig

Technical Reference(s):	N2-OP-34, NUCLEAR BOILER, AUTOMATIC DEPRESSURIZATION AND SAFETY RELIEF VALVES, Section B.2.0

Proposed references to be provided to applicants during the examination:

N/A

Learning Objective:	02-OPS-001-239-2-00, H. EO-1.0	
Question Source:	Bank #	New
	Modified Bank #	New
	New	New
Question History:	Previous NRC Exam	New
	Previous Test / Quiz	New
Question Cognitive Level:	Memory of Fundamental Knowledge	
	Comprehension or Analysis	2
10CFR Part 55 Content:	41.7 / 45.6	

Comments:

53-2

Question # SRO 13

Examination Outline Cross-Reference	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295016 2.4.11
	Importance Rating	3.6
Knowledge of abnormal condition procedure.		

Proposed Question:

With the plant operating at 100% power, a Control Room evacuation becomes necessary. Following the evacuation the Control Room E operator implements the immediate actions of N2-SOP-78, CONTROL ROOM EVACUATION, to lineup Reactor Core Isolation Cooling (RCIC). The Control Room E operator reports that RCIC injection can **NOT** be established.

Which one of the following should be directed to maintain reactor water level?

a. Locally start HPCS and feed the RPV as necessary.

b. Locally start the second CRD Pump and maximize injection.

c. Lower RPV pressure with the SRVs and establish makeup with the RHR System.

d. Take local control of Turbine Bypass Valves and lower pressure and establish injection with the feedwater system.

Proposed Answer: c.

Explanation (Justification of Distractors):

a. Not a valid option
b. Not a valid option
d. Not a valid option

Technical Reference(s): N2-SOP-78, sect. 3.5

Proposed references to be provided to applicants during the examination:

None

Learning Objective:	O2-OPS-001-296-2-00, TO 3
Question Source:	Bank # Modified Bank # Q15831 New
Question History:	Previous NRC Exam Previous Test / Quiz
Question Cognitive Level:	Memory of Fundamental Knowledge 1 Comprehension or Analysis
10CFR Part 55 Content:	55.41 10 55.43.5 55.45.13

Comments:
SRO Only: N2-SOP-78 requires entry into N2-OP-78 to establish level control using the LPCI Pseudo Mode. This evolution is performed concurrent with N2-OP-78 sections for "Reactor Pressure Control" and "Reactor Pressure Vessel Cooldown". These actions are directed and controlled by the SSS from the Remote Shutdown Panel because of the multiple tasks being performed concurrently.

QUESTION 2116. (Point value: 1.00 , T.R.A.I.N. Q15831)

Following a control room evacuation, the Control Room E quickly implements the immediate actions of N2-SOP-78, "Control Room Evacuation", to assess the ability to lineup RCIC injection from the Remote Shutdown Room.

Which one of the following states the required actions and associated time requirements if RCIC injection cannot be established?

a. Locally start the High Pressure Core Spray system and within 5 minutes establish injection.

b. Locally start the Control Rod Hydraulics system pumps and within 5 minutes establish injection with CRD pumps.

c. Blowdown the RPV pressure using SRV's within 9 minutes and establish RHR injection.

d. Blowdown the RPV pressure using SRV's within 9 minutes and establish Condensate Booster pump injection.

Answer:

C

KA/Setting:
NMPC KA #: 0.00
Setting : C1

Related Training:
O2 -OPS -001-296-2-00 Rev 0

Related Items:
NMPC, LP, N2, O2-OPS-001-296-2-00, na,
NMPC, PROC, N2, N2-SOP-78, na,
NRC, NUREG, NA, 1123, na, 295016 K3.02

Question # SRO 14

Examination Outline Cross-Reference	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295017 AA2.01
	Importance Rating	4.2

Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: off-site release rate: plant specific.

Proposed Question:

The plant is operating at 100% power when a drain line breaks off the "C" Main Steam line immediately upstream of the Turbine Control Valves. Operators in the Control Room manually initiate a Group 1 isolation but the isolation fails to stop the leak. The following conditions exist:

- An **Site Area Emergency** has been declared
- Turbine building full of steam and evacuated
- Measured release rates have **NOT** risen above normal
- Several turbine building ARMs have alarmed
- Wind direction is **from 295°**
- Wind speed is **3 mph**

Which one of the following methods should be used to make an initial off-site dose assessment?

a. Direct RP to calculate off-site dose rates based on the stack release.

b. Send RP to South East side of the site boundary to take dose rate readings.

c. Dispatch on-site monitoring teams to monitor the North side of the site fence.

d. Request a monitoring team be sent to the first population center located 115° from the site.

Proposed Answer: b.

Explanation (Justification of Distracters):

a. The dose rate must be calculated after the ground release is determined
b. The release is unmonitored and outside the secondary containment
d. The release is outside the secondary containment

Technical Reference(s): EPIP-EPP-08

Proposed references to be provided to applicants during the examination:

EPIP-EPP-08, Attachment 1 and Table1.1

Learning Objective:

Question Source:	Bank # Modified Bank # New	 new
Question History:	Previous NRC Exam Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 3
10CFR Part 55 Content:	41.10 / 43.4 / 43.5 / 45.13	

Comments:
SRO Only: Site Emergency Director responsibilities and application of the emergency plan.

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION

EMERGENCY PLAN IMPLEMENTING PROCEDURE

EPIP-EPP-08

REVISION 09

OFF-SITE DOSE ASSESSMENT AND PROTECTIVE ACTION RECOMMENDATION

TECHNICAL SPECIFICATION REQUIRED

Approved by: R. G. Smith	R G Smith Plant Manager - Unit 1	11/16/98 Date
Approved by: N. C. Paleologos	Nick Paleologos Plant Manager - Unit 2	11-16-98 Date

PERIODIC REVIEW, 03/10/99, NO CHANGE

Effective Date: 12/18/98

PERIODIC REVIEW DUE DATE MARCH 2000

S5-3

LIST OF EFFECTIVE PAGES

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5		30			
6		31			
7		32			
8	.51367	33			
9	.51609	34			
10					
11					
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15	.51608				
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TABLE OF CONTENTS

1.0 PURPOSE

To provide the methods for determining meteorology data, release rates, dose assessment and protective actions during accident conditions at Nine Mile Point.

2.0 PRIMARY RESPONSIBILITIES

2.1 The **Station Shift Supervisor/Site Emergency Director (SSS/SED)**:

2.1.1 Ensures meteorological data acquisition, release rate determination, and dose assessment are performed during the initial stages of an emergency to support development of Protective Action Recommendations (PARs)

2.1.2 Approves PARs and ensures their timely issue to the State and County

2.2 The **Corporate Emergency Director (CED)** approves PARs prior to their transmittal to the State and County, following EOF activation.

2.3 The **Radiation Assessment Manager (RAM)** is responsible to the SED for managing the onsite radiological monitoring and assessment aspects of the station during an emergency, following TSC activation.

2.4 **Chemistry Technicians** perform release rate assessments, obtain meteorological data, and develop PARs, prior to EOF activation.

2.5 The **Offsite Dose Assessment Manager (ODAM)** manages the offsite dose aspects of an emergency in order to assess the radiological consequences to the public, following EOF activation.

2.6 The **Radiological Assessment Staff** is responsible to the ODAM for obtaining meteorological data, determining source term, performing dose assessment, and developing PARs, following EOF activation.

3.0 PROCEDURE

3.1 Dose Assessment and Protective Action from the Control Room

* *

CAUTION

Calculation involving the determination of release rates and/or protection action shall be self-checked for accuracy.

* *

3.1.1 Chemistry Technician Actions

a. Consult the SSS/SED on plant conditions and possible release paths. If a General Emergency has been declared, assist SSS/SED in making Protective Action Recommendations based on plant conditions using Attachment 1.

b. Access EDAMS computer using Attachment 2

c. Obtain meteorological data using Attachment 3.

d. Assess effluent monitor readings and conditions.

e. Determine release rate using Attachment 4. Combine multiple release points as follows:

   1. Sum all release points from the same elevation (ground or elevated).

   2. Calculate the total release rate from combined ground and elevated sources using the workspace on Attachment 1.

f. Use Attachment 1 flowchart and advise SSS/SED of any PARs recommended by the flowchart.

g. IF an unmonitored atmospheric release is suspected or known to be in progress, then assist the SSS/SED in the following actions:

   1. Advise the SSS/SED to expedite the dispatch of Radiation Protection (RP) Technician. Request assistance of the unaffected Unit or J.A. Fitzpatrick if needed.

   2. The RP Technician should be dispatched to potential plume centerline (wind direction (degrees) ± 180° = plume centerline), as close to the site boundary as practicable. See Attachment 1, Figure 1.4 for Site boundary location.

3.1.1.g (Cont)

3. IF readings indicate > 1 rem/hr based on field survey perform the actions indicated in Attachment 1.

h. Assist Communications Aide in completing the meteorological data and release rate sections of the Part 1 Notification Fact Sheet.

i. Continue to monitor meteorological data, changes in effluent conditions or conditions that might lead to abnormal radiological effluents.

j. When requested, turn over all duties to the EOF.

3.1.2 <u>SSS Actions</u>

a. Verify that the Chemistry Technician is performing dose assessment and protective action development in a timely fashion and in accordance with Attachment 1.

b. Assess any release rates provided by the Chemistry Technician against the Emergency Action Levels (EAL).

c. Review AND approve PARs recorded on the Notification Fact Sheet Part 1, as required. Use ERPA map in Attachment 1 if desired.

3.2 Dose Assessment and Protective Actions from the EOF

3.2.1 <u>Offsite Dose Assessment Manager (ODAM) Actions</u>

NOTE: IF at any time the initiating conditions listed in Attachment 1 are met, THEN perform the actions listed in that attachment.

a. Perform actions as indicated in EPIP-EPP-23.

b. Verify Environmental Survey Sample Team Coordinator has been assigned and is:

1. Preparing for the dispatch of downwind survey teams.

2. Is aware of meteorological advisor status.

c. Perform or have performed the following:

1. Obtain meteorology data using Attachment 3 of this procedure.

2. Obtain effluent monitor readings and calculate release rate using Attachment 4 of this procedure.

3.2.1.c (Cont)

3. Perform dose assessment calculation using Attachment 5 of this procedure.

d. Determine PARs using Attachment 5 of this procedure.

e. Interface with State and County representatives in the EOF.

1. Keep State/County representatives informed of confirmed data and results.

f. Complete Part 2 Notification Fact Sheet when ANY of the following conditions exist or are met:

1. Rad release that exceeds Tech Specification limits.

2. Significant changes in meteorological OR rad release conditions.

3. Every 30 minutes.

g. With each significant change in meteorological, actual release rate, and dose assessment data, OR every 30 minutes.

h. Constantly reassess effluent monitors (release rate) and meteorological data for changes. Perform new dose assessment as needed. Develop new PARs and/or verify the adequacy of PARs already made.

i. As Downwind Survey Team (DST) becomes available, utilize it to verify release rates. If these refined release rates differ significantly from those calculated from effluent monitor readings, reperform dose assessment using refined release rates.

j. Provide data for the Part 1 Notification Fact Sheet as requested.

k. Provide CED with pertinent information as needed.

1. Changing radiological conditions that may lead to PARs.

2. Protective actions for site staff.

ℓ. Maintain Chronological Release Rate Log (see Attachment 5.1).

3.2.2 EOF Dose Assessment Staff

a. IF at any time the initiating conditions listed in Attachment 1 are met, THEN perform the actions listed in that attachment.

b. Perform actions as indicated in EPIP-EPP-23.

c. Perform any actions as requested by the ODAM, including:

- Obtaining meteorological data (Attachment 3)
- Obtaining release rate data (Attachment 4)
- Performing dose assessment and protective action recommendations (Attachment 5)

4.0 DEFINITIONS

4.1 **CDE_T.** Committed dose equivalent to the thyroid for the child.

4.2 **EDAMS.** Emergency Dose Assessment Modeling System. A PC-based computer program that calculates release rates, doses and protective actions, and obtains meteorological data for emergencies.

4.3 **MMS.** Meteorological Monitoring System. Consists of the dedicated computer, main, backup and inland towers and software. Stores and edits site meteorological data.

4.4 **RADDOSE.** A subprogram of EDAMS, it performs the dose assessment functions during emergencies.

4.5 **SHELTERING.** A protective action whose benefit is to bring the public to a heightened state of awareness. No dose reduction is assumed for sheltering.

4.6 **TEDE.** Total Effective Dose Equivalent.

5.0 REFERENCES/COMMITMENTS

5.1 Technical Specifications

None

5.2 Licensee Documentation

5.2.1 NMP Unit 1 FSAR, Section XV

a. Table XV-32

b. Table XV-28

5.2.1 (Cont)

c. Table XV-29

d. Table XV-23

e. Table XV-29d

f. Section 1.3.1

g. Section 2.1

5.2.2 NMP Unit 2 USAR, Section 15

a. Table 15.6-15b

b. Table 15.4-12

c. Table 15.7-11

d. Table 15.6-8

e. Table 15.7-4

f. Table 15.6-3

g. Table 16.6-19

5.2.3 SEP, NMPC Nine Mile Point Nuclear Station Site Emergency Plan

5.2.4 NMPC Correspondence 96-MET-001 (Backup Tower Wind Speed Correction Factor)

5.2.5 NMP Correspondence 96-MET-002 (Main Tower Wind Speed Correction Factor)

5.2.6 NMP Correspondence 96-MET-004 (Backup Tower Wind Direction Concerns)

5.2.7 NMP Correspondence 96-MET-003 (Discussion at DER C-95-0693)

5.2.8 NMP Correspondence 96-MET-005 (Main Tower 30' Sigma Theta Concern)

5.2.9 NMP Correspondence 97-MET-002 (Main Tower Wind Obstructions)

5.3 <u>Standards, Regulations, and Codes</u>

NUREG-0654, FEMA-REP-1, Rev 1, Supp 3, Criteria for Protective Action Recommendations for Severe Accidents

6.0 RECORDS REVIEW AND DISPOSITION

6.1 The following records generated by this procedure shall be maintained by Records Management for the Permanent Plant File in accordance with NIP-RMG-01, Records Management:

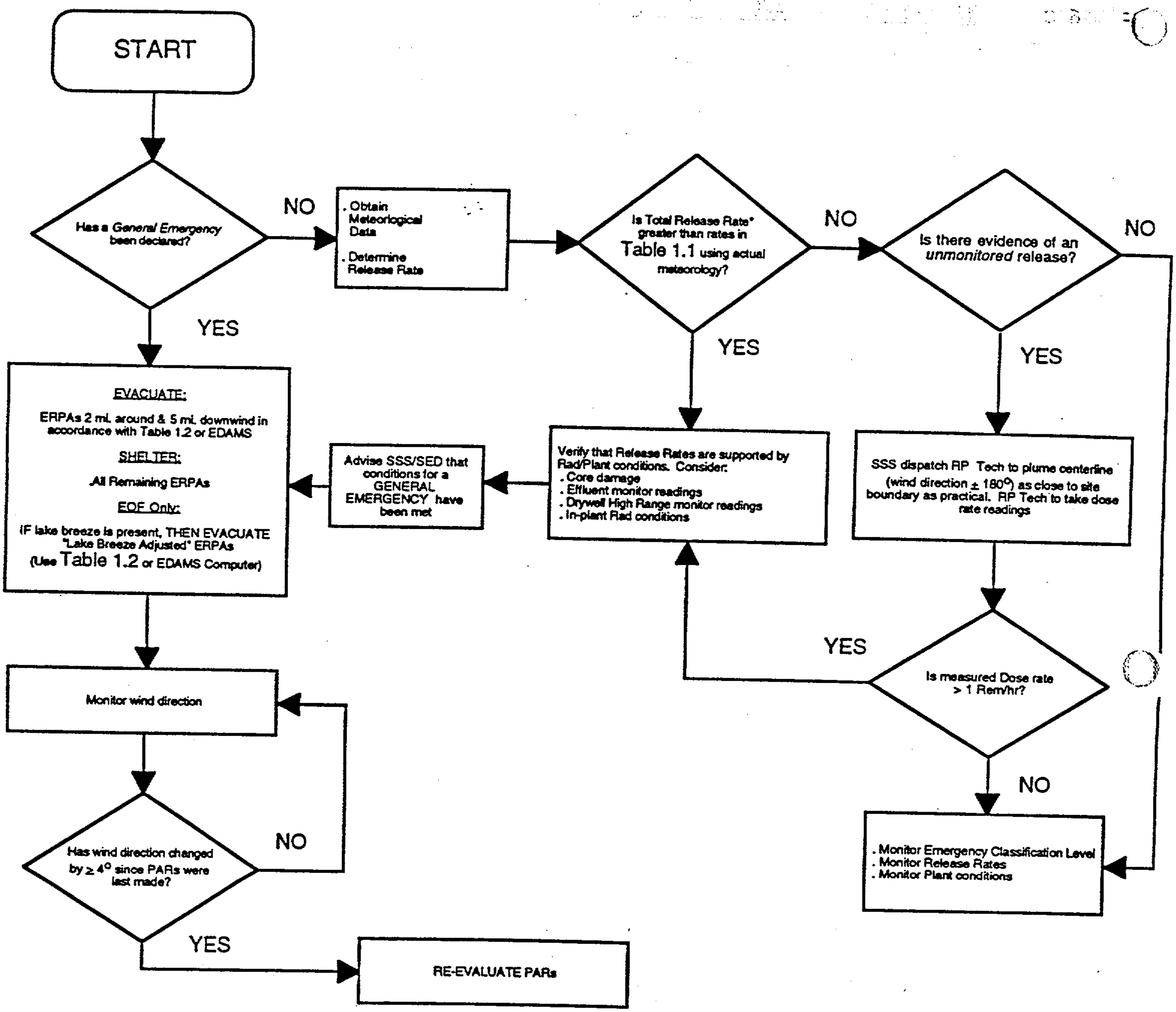
NOTE: For records generated due to an actual declared emergency only.

- Attachment 1, Initial Dose Assessment and Protective Actions
- Attachment 4, Release Rate Determination
- Figure 5.1, Chronological Release Rate Log
- Figure 5.2, EDAMS Data Entry Form

6.2 The following records generated by this procedure are not required for retention in the Permanent Plant File:

NOTE: For records generated NOT due to an actual declared emergency only.

- Attachment 1, Initial Dose Assessment and Protective Actions
- Attachment 4, Release Rate Determination
- Figure 5.1, Chronological Release Rate Log
- Figure 5.2, EDAMS Data Entry Form



*Use this formula if release has a ground AND elevated source:

$$\left[\frac{\text{Ground Release Rate (Ci/s)}}{\text{Table 1.1 Ground Release Rate (Ci/s)}}\right] + \left[\frac{\text{Elevated Release Rate (Ci/s)}}{\text{Table 1.1 Elevated release rate (Ci/s)}}\right] = \text{IF} \geq 1, \text{ A General Emergency Exists}$$

$$\left[\frac{\quad}{\quad}\right] + \left[\frac{\quad}{\quad}\right] =$$

TABLE 1.1 - GENERAL EMERGENCY RELEASE RATES

Ground Release (Ci/s)				
Wind Speed (mi/h)	Stability Class			
	A	B/C	D	E/F/G
0-3	1333	213	119	38
4-6	3226	286	143	48
7-9	5556	526	250	83
10-13	7692	769	357	117
14-17	10753	1075	500	164
18-21	13514	1389	667	213
>21	16393	1667	833	256

Elevated Release (Ci/s)				
Wind Speed (mi/h)	Stability Class			
	A	B/C	D	E/F/G
0-3	2041	1124	3030	769
4-6	3703	909	769	769
7-9	5882	1515	1075	1250
10-13	7692	2083	1388	1724
14-17	11494	2857	1818	2273
18-21	14286	3704	2273	2778
>21	17241	4348	2632	3226

TABLE 1.2 - AFFECTED ERPAs

Wind Direction From	2 Mile Around and 5 Mile Downwind	EOF Only beyond this line!	10 Mile Radius	Lake Breeze Adjusted (5 Mile Radius)
214 to 222	1, 2, 3, 26, 27		14, 29	
223 to 233	1, 2, 3, 26, 27		14, 29	4, 7
234 to 240	1, 2, 3, 7, 26, 27		14, 15, 29	4
241 to 254	1, 2, 3, 4, 7, 26, 27		14, 15, 29	9
255 to 262	1, 2, 3, 4, 7, 26, 27		14, 15, 16, 17, 29	9
263 to 278	1, 2, 3, 4, 7, 9, 26, 27		8, 14, 15, 16, 17, 29	5
279 to 292	1, 2, 3, 4, 5, 7, 9, 26, 27		8, 14, 15, 16, 17, 18, 29	10
293 to 305	1, 2, 3, 4, 5, 7, 9, 10, 26, 27		8, 14, 15, 16, 17, 18, 29	
306 to 311	1, 2, 3, 4, 5, 7, 9, 10, 26, 27		8, 14, 15, 16, 17, 18, 19, 20, 29	
312 to 332	1, 2, 3, 4, 5, 7, 9, 10, 26, 27		8, 14, 15, 16, 17, 18, 19, 20	6, 11
333 to 340	1, 2, 3, 4, 5, 9, 10, 11, 26, 27		8, 15, 16, 17, 18, 19, 20, 21, 25	6, 7, 12
341 to 349	1, 2, 3, 4, 5, 9, 10, 11, 26, 27		8, 17, 18, 19, 20, 21, 24, 25,	6, 7, 12
350 to 356	1, 2, 3, 5, 6, 9, 10, 11, 26, 27		8, 13, 18, 19, 20, 21, 22, 24, 25, 12	4, 7
357 to 0 0 to 12	1, 2, 3, 5, 6, 9, 10, 11, 26, 27		13, 18, 19, 20, 21, 22, 23, 24, 25, 12	4
13 to 20	1, 2, 3, 5, 6, 10, 11, 26, 27		13, 18, 19, 20, 21, 22, 23, 24, 25, 12	4, 9
21 to 51	1, 2, 3, 5, 6, 10, 11, 26, 27		13, 19, 20, 21, 22, 23, 24, 25, 28, 12	9
52 to 56	1, 2, 3, 5, 6, 11, 26, 27		13, 19, 20, 21, 22, 23, 24, 28, 12	10
57 to 61	1, 2, 3, 5, 6, 11, 26, 27		13, 19, 21, 22, 23, 24, 28, 12	10
62 to 70	1, 2, 3, 6, 11, 26, 27		13, 19, 21, 22, 23, 24, 28, 12	10
71 to 89	1, 2, 3, 6, 26, 27		13, 21, 22, 23, 24, 28, 12	11
90 to 95	1, 2, 3, 6, 26, 27		28	5, 11, 12
96 to 114	1, 2, 3, 26, 27		28	6, 12
115 to 146	1, 2, 3, 26, 27		28	
147 to 213	1, 2, 3, 26, 27		28, 29	

TABLE 1.3 - EPA 400 Protective Action Guidelines (EPA PAGs)

PAR	TEDE (rem)	CDE_T (rem)
Evacuate	> 1	> 5

FIGURE 1.4 - Site Boundary Map

Nine Mile Point
Site Boundary Map

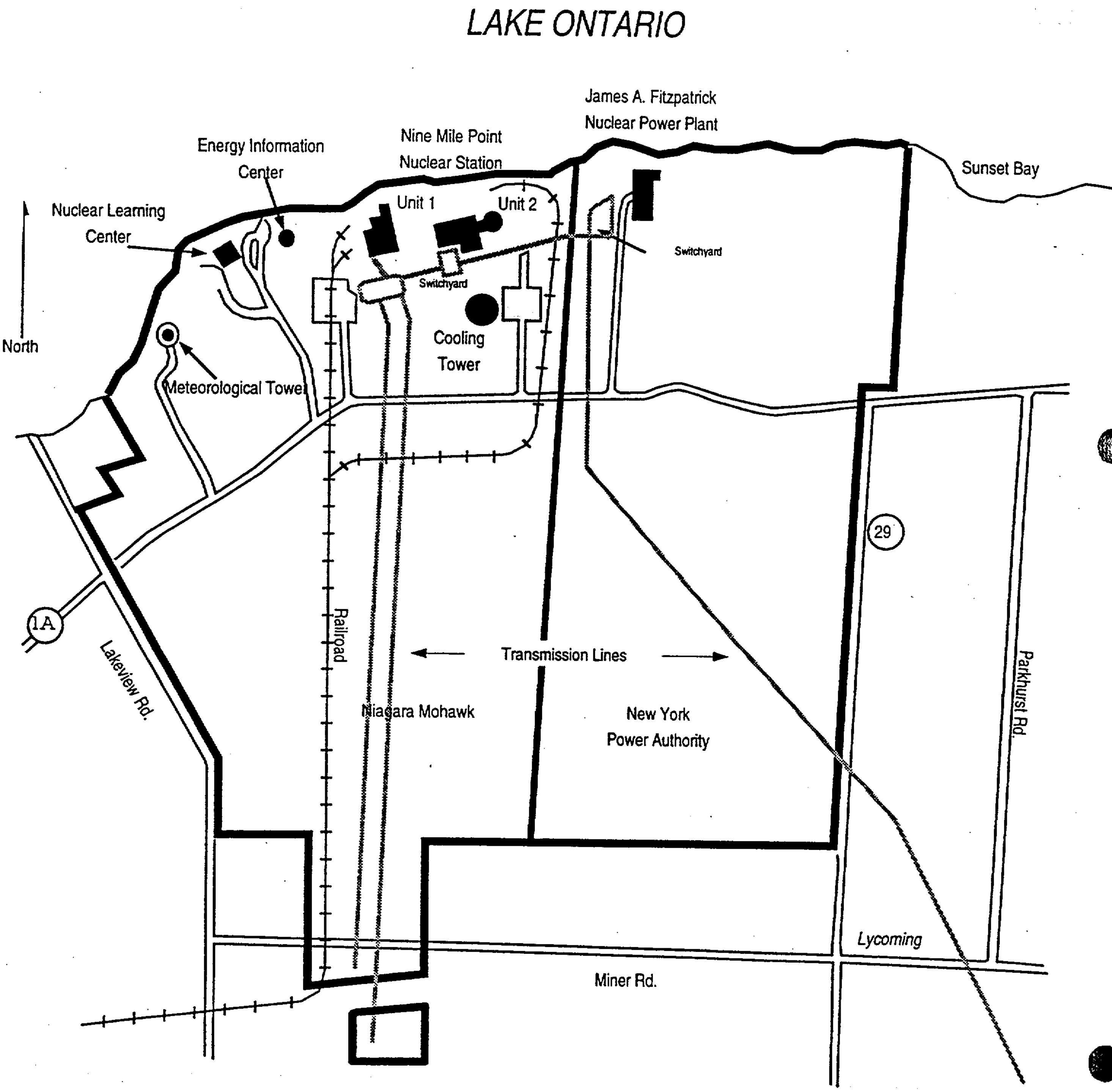
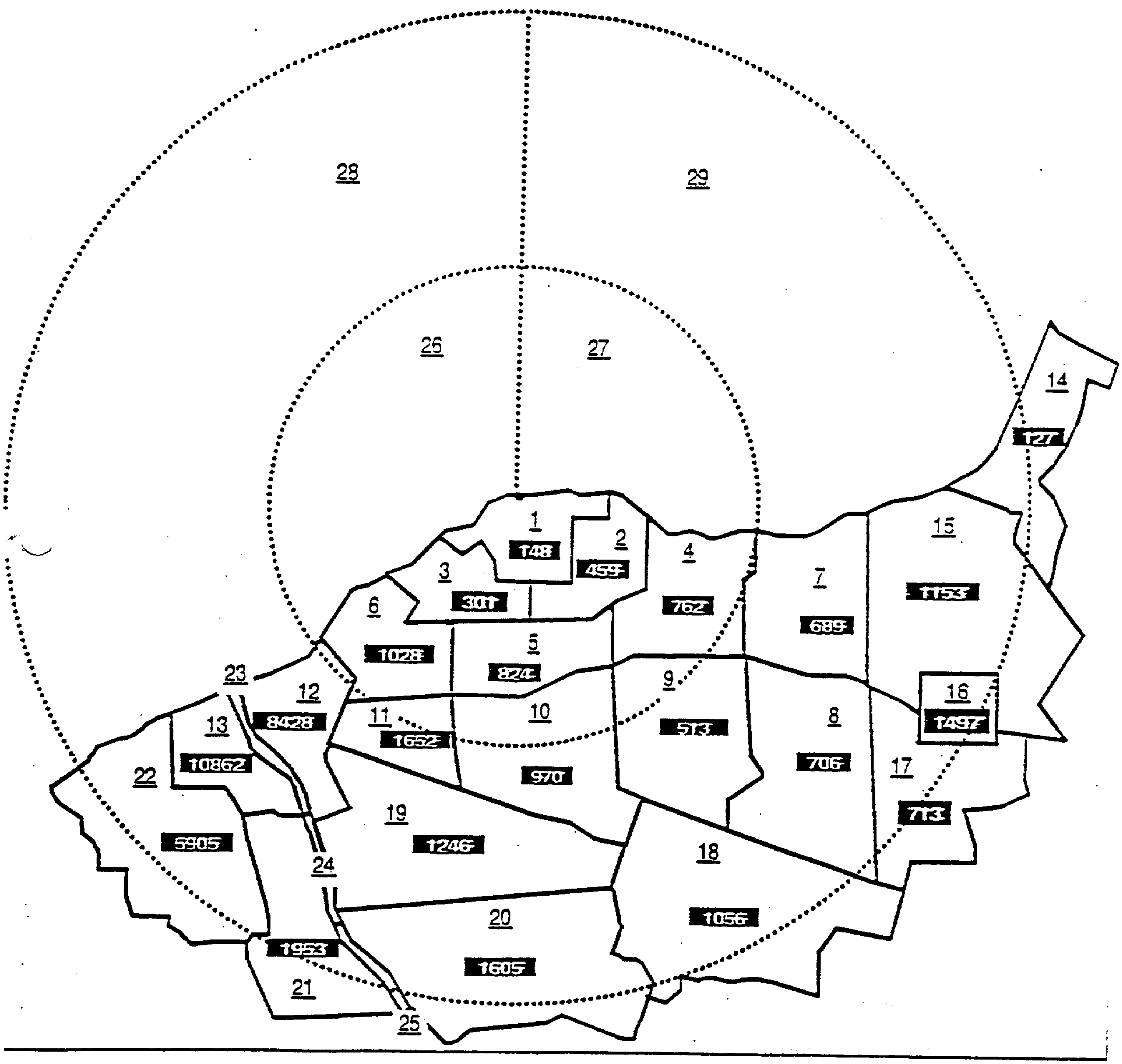


FIGURE 1.5 - ERPA Map



LEGEND 1 ERPA Number 148 ERPA Population

ATTACHMENT 2: USE OF THE EDAMS COMPUTER

1.0 CONTROL ROOM EDAMS

1.1 Turn the system on: Turn on the power to the EDAMS computer, monitor, and printer. After the computer boots:

a. Select the "EDAMS" icon.
b. Select the "Login" icon.
c. Select "Direct Connect to Met Data".
d. Once Login is successful/complete, select "OK".
e. Select appropriate icon.

1.2 Computer or Connect Problems

a. If "Direct Connect to Met Data" fails, select "Automatic Dial-In to Met Data".

b. If "Automatic Dial-In to Met Data" fails, select "Manual Dial-In to Met Data".

c. If at any time problems are experienced with the computer, depress the eject button on the front of the computer. This will eject the laptop computer. Continue this procedure with the laptop.

d. If the laptop should fail, have the Chemistry Tech from the unaffected Unit go to the unaffected Control Room and bring the EDAMS laptop back to the affected Control Room and continue with this procedure.

NOTE: In this case, meteorological data will have to be obtained manually.

2.0 EOF EDAMS

2.1 Turn the system on: Turn on the power to the EDAMS computer, monitor, and printer. After the computer boots:

a. Select the "EDAMS" icon.
b. Select the "Login" icon.
c. Select "Automatic Dial-In to Met Data".
d. Once Login is successful/complete, click on "OK".
e. Select the appropriate icon.

2.2 Computer or Dial-In Problems

a. If "Automatic Dial-In to Met Data" fails, select "Manual Dial-In to Met Data".

b. If at any time problems are experienced with the computer, use the duplicate EDAMS computer in the EOF.

3.0 EDAMS DOSE MODEL LIMITATIONS

3.1 A calculational limitation of the dose assessment model occurs when an extreme wind (direction) shift takes place. The model may not calculate doses in sectors that the plume skips over entirely within a single 15 minute calculation step.

3.2 EDAMS only allows the operation of one application at a time.

3.3 Dose rates and deposition rates reported by the model are the maximum for the sector, not necessarily the dose rate or deposition rate at the center of the sector. This avoids the situation of a narrow (stable) plume slipping between receptor points and being missed.

3.4 Deposition data reported is not intended for an environmental evaluation; its intent is to indicate areas of potentially high ground level concentrations.

1.0 OBTAINING METEOROLOGICAL DATA

The methods of obtaining meteorological data are listed below in the order that they should be used.

- EDAMS (see Section 3.0 of this Attachment)
- Strip Chart Recorder (see Section 4.0 of this Attachment)
- Manual input from alternate sources (see Section 5.0 of this Attachment)

2.0 USE OF METEOROLOGICAL DATA: GENERAL CONDITIONS

NOTE: Wind speed measurements at both the main and backup towers may on occasion be less than actual observed winds. When using the main tower winds and the wind direction is between 0° and 100° or when using the backup tower and the wind direction is between 220° and 270°, caution should be exercised when estimating plume arrival time, its likely that the plume will arrive sooner than what the wind speed would indicate. Additionally the actual dose may be less than forecast by EDAMS.

2.1 Hierarchy of NMP meteorological data sources is shown in Table 3.1 below.

NOTE: Heights of meteorological instrumentation is approximate.

2.2 If substitute data is to be used, consult the Meteorological Advisor (if available).

2.3 If using 200', 100', or 30' sigma theta stability and the wind is blowing from a direction listed below, substitute to the next source per table 3.1's Stability section.

Main Tower Sigma Theta Stability	Wind Direction
200'	030° to 096°
100'	030° to 077°
30'	035° to 076°

2.4 If using the JAF Backup sigma theta stability (for either ground or elevated level release), the following adjustments should be made:

JAF Backup Tower Wind Direction (From)	JAF Backup Sigma Theta Stability Adjustment
232° to 246° or 270° to 281°	Add one stability class such that: A → B B → C C → D D → E E → F F or G → G
247° to 269°	Add two stability classes such that: A → C B → D C → E D → F E,F, or G → G

2.5 If no release is in progress, or release path is unknown, use elevated data (200' main tower), or substitute as outlined in Table 3.1.

2.6 The Meteorological Advisor may use any source (Sodar, other towers, characterization tables, etc.) or skills of the trade to satisfy the need for meteorological data.

TABLE 3.1: HIERARCHY FOR USE OF NMP METEOROLOGICAL DATA SOURCES

Parameter	Hierarchy	Elevated Release	Ground Release
Wind Speed & Direction	Primary	200' Main	30' Main
	Substitutes	100' Main	100' Main
		JAF Backup	JAF Backup
		30' Main	200' Main
		Inland	Inland
Stability	Primary	200' ΔT	100' ΔT
	Substitutes	100' ΔT	200' ΔT
		200' Sigma Theta (1)	30' Sigma Theta (1)
		100' Sigma Theta (1)	100' Sigma Theta (1)
		JAF Backup Sigma Theta (2)	JAF Backup Sigma Theta (2)
		30' Sigma Theta (1)	200' Sigma Theta (1)
		Inland Sigma Theta	Inland Sigma Theta

(1) If using the 30', 100', or 200' Sigma Theta stability, AND the wind is from a direction listed in Step 2.3, THEN substitute the next source of data in accordance with Table 3.1 of this attachment.

(2) If using the JAF Backup Sigma Theta stability, AND the wind is from a direction listed in Step 2.4, THEN substitute the next source of data in accordance with Table 3.1 of this attachment.

2.7 Refer to Figure 3.2 to determine if lake breeze is a possibility (EOF only).

2.8 Refer to Figure 3.3 to determine if land breeze is a possibility (EOF only).

3.0 EDAMS

3.1 To obtain meteorological data for the Notification Fact Sheet Part 1, select "Emergency Meteorological Report" from the EDAMS main menu.

3.2 Select "Print Met Data" to print the data.

4.0 STRIP CHART RECORDER

* *

CAUTION

Do not use the LED readouts associated with the strip chart recorders.

* *

NOTE: Use this data only if the method described in Section 3.0 of this Attachment is unavailable.

4.1 Strip chart meteorological data can be found in both of the Control Rooms, and in the TSC. Utilize Table 3.1 to determine source of data.

NOTE: ΔT cannot be obtained in the TSC. Utilize $\sigma\theta$ in determining stability.

4.2 Figures 3.4 and 3.5 show sample strip chart traces showing ambient air temperature, ΔT, $\sigma\theta$, wind speed and direction data.

4.3 Observe the values of the vertical temperature difference, ΔT, from the primary meteorological tower over the last 15 minute period. The preferred reading is the 30'-200' Delta temperature reading, for an elevated (stack) release.

4.4 Compare the values of ΔT to the Stability Classification Chart (Table 3.6) and select the appropriate stability class and record.

4.5 If values of ΔT are not available, then observe the values of $\sigma\theta$, directly from the primary or backup meteorological tower recorders, over the last 15 minute period.

4.6 Compare these values of $\sigma\theta$ to Table 3.6. Using the chart, select the appropriate stability class and record.

4.7 If both data are available, use the ΔT at 30'-200' elevation for elevated releases: use $\sigma\theta$ at the 30' elevation for ground (vent) releases.

4.8 If values for ΔT and $\sigma\theta$ are not available, then observe the wind direction trace over the last 15 minute period. Determine $\sigma\theta$ by dividing the horizontal deviation of the wind direction trace over the last 15 minutes by 6. To make reading of the strip charts easier, you may want to advance the chart.

5.0 MANUAL INPUT FROM ALTERNATE SOURCES

NOTE: Use this data only if the methods described in Section 3 and 4 of Attachment 3 are unavailable.

* *

CAUTION

Data obtained by the methods described below will not be site-specific and will likely introduce errors into dose assessments. The Meteorological advisor shall be consulted regarding the use of all substitute data. If the Meteorological advisor is not available, use the data as obtained.

* *

5.1 National Weather Service

a. Telephone the National Weather Service (NWS) in Buffalo at 800-462-7751 or 716-565-9001.

b. Request the current wind speed and direction, stability class and temperature.

c. Use this data as follows:

1. Wind speed (NWS) = elevated and ground wind speed

2. Wind Direction (NWS) = elevated and ground wind direction

3. Stability Class (NWS) = stability class

5.1.c (Cont)

4. Temperature (NWS) = ambient temperature

5.2 Other Sources

a. Other sources of meteorological data that may be utilized are as follows:

1. SODAR

2. Other (non-NWS) meteorology towers

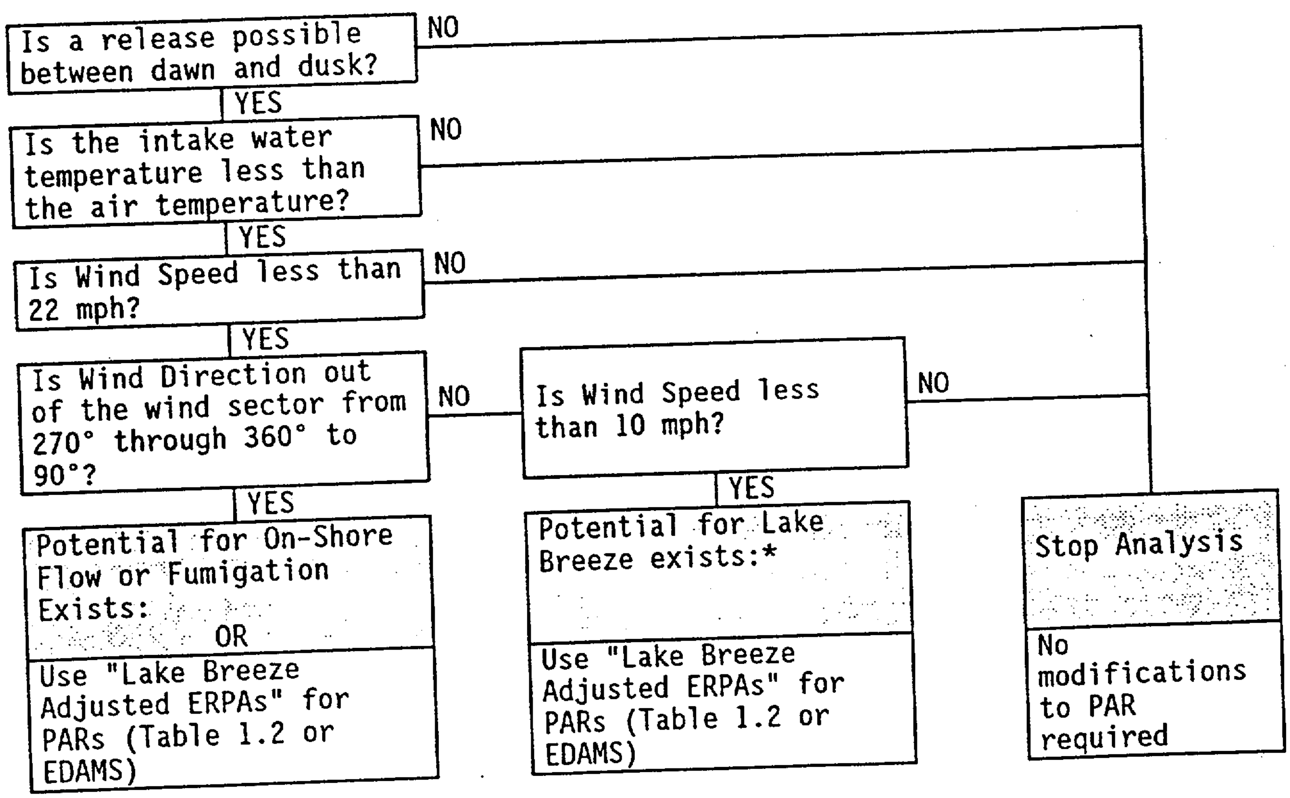
3. Commercial weather services

FIGURE 3.2
Lake Breeze/On-Shore Flow and Fumigation Flow Chart

1. Obtain meteorological data per Section 1.0 of this Attachment.

2. Obtain lake intake water temperature from Unit 1 or 2 process computer or from Control Rooms.

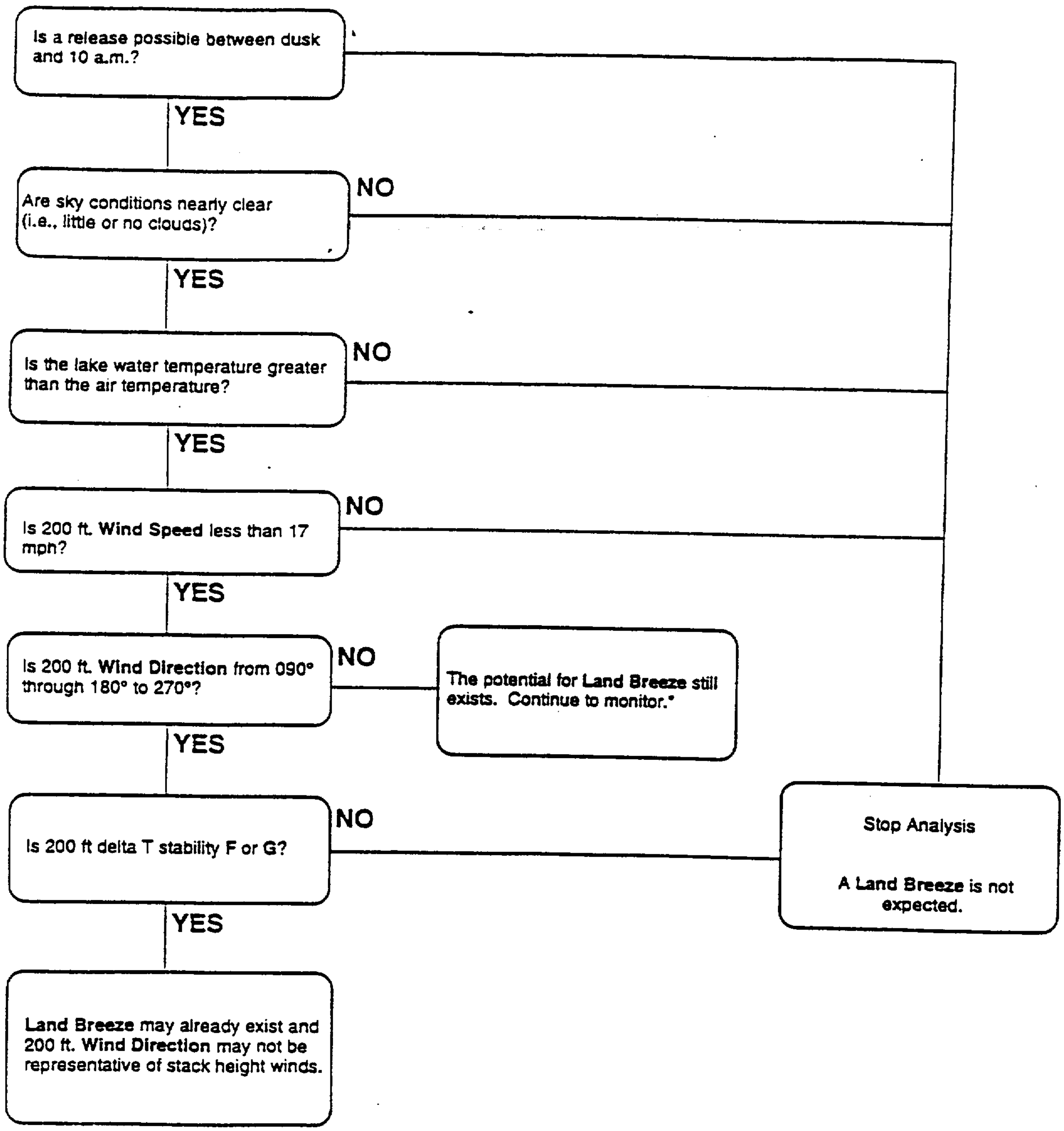
3. Follow the flow chart answering the appropriate questions.



* Also note that there is a potential for a sudden shift in wind direction to 245° through to 65° if the lake breeze has not already formed.

FIGURE 3.3
LAND BREEZE FLOW CHART

1. Obtain Meteorological Data.
2. Obtain lake temperature.
3. Follow the flow chart answering the appropriate questions.



***NOTE**: There is a potential for a shift in Wind Direction to 090° through 180° to 270° at the weather tower.

FIGURE 3.4
SAMPLE AIR TEMPERATURE AND STABILITY CLASS TRACE - CONTROL ROOM

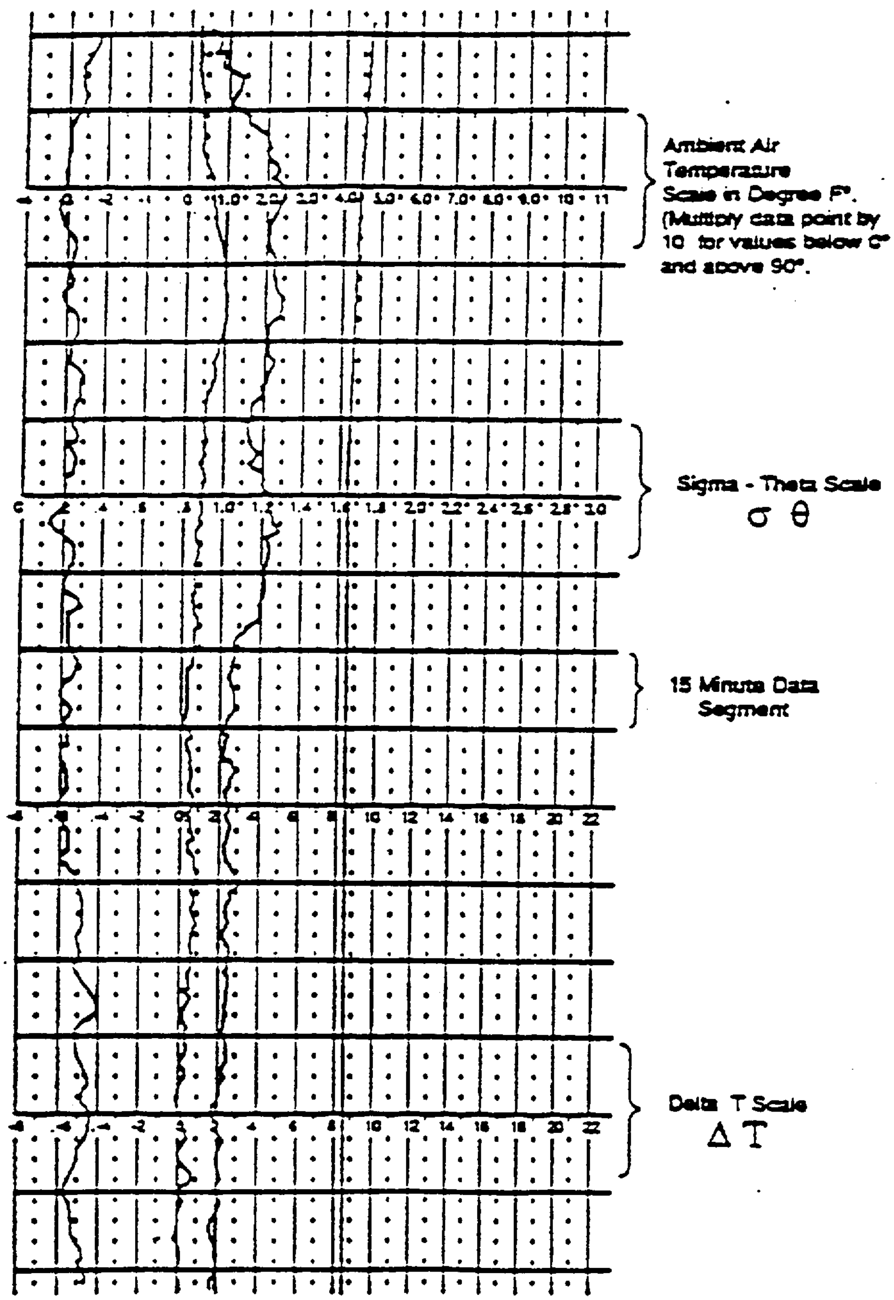
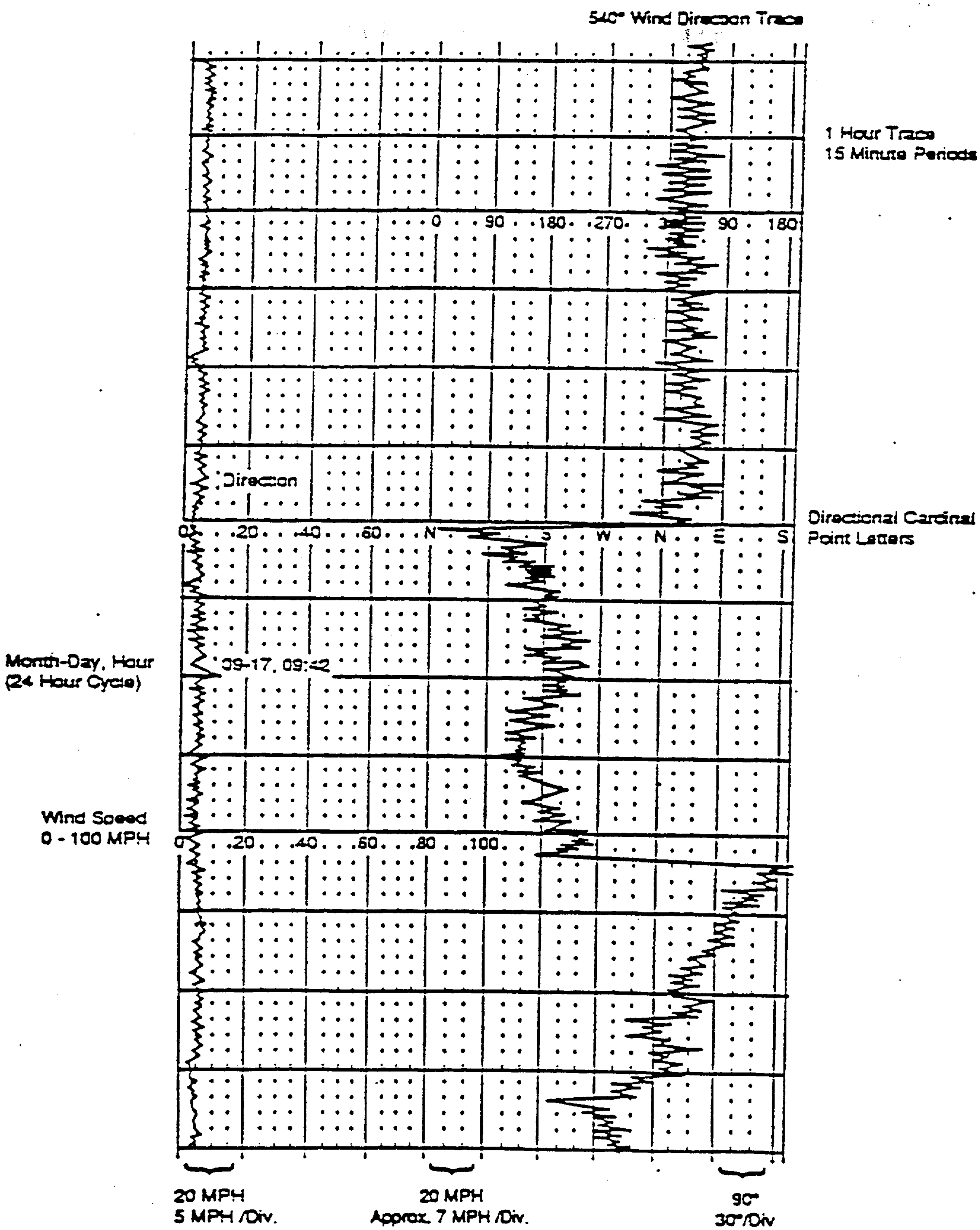


FIGURE 3.5
SAMPLE WIND SPEED AND WIND DIRECTION TRACE - CONTROL ROOM/TSC



1.0 METHOD

a. Access the EDAMS Computer using Attachment 2 of this procedure.

b. Select the "EDAMS (PARs and Release Rate Calcs)" Icon.

c. IF Unit 1 was selected, go to Section 2.0 of this Attachment.

d. IF Unit 2 was selected, go to Section 3.0 of this Attachment.

2.0 UNIT 1 METHODS

2.1 OGESMS

a. Select monitor (7, 8, 10a or 10b)

NOTE: Monitor 7 = indicator 112-07A
Monitor 8 = indicator 112-08A
Monitor 10a = indicator RN10A
Monitor 10b = indicator RN10B

b. Enter time that reading was obtained (using 24 hour format)

c. Enter monitor reading (cpm for monitors 7 or 8, cps for monitors 10a or 10b). Use J panel readings or the following computer points:

- monitor 7, use E334
- monitor 8, use E335
- monitor 10a, use E488
- monitor 10b, use E489

d. Enter process computer calibration factor. If unavailable, use default values below:

- 4.4E-8 for 7 or 8
- 4.4E-7 for 10a or 10b

e. Enter Stack Flow (kcfm). Use computer point C320 or calculate from Table 4.1.

f. Hit the "F9" key.

g. Print results.

2.2 RAGEMS

a. Enter the time that the reading was obtained (24 hour format).

b. Enter the monitor reading (cps). Use J panel reading or computer point C321.

c. Enter calibration factor (use posted value).

d. Enter dilution factor as follows:

= 1 if 6 liter chamber is used
= 1E3 if 30 cc chamber is used
= 2E5 if 30 cc chamber plus first stage dilution is used. Use Total Dilution Ratio (TDR) x1000 as the dilution factor, if TDR is known.
= 4E7 if 30 cc chamber plus first and second stage dilution is used. Use TDR x1000 as the dilution factor, if TDR is known.

TABLE 3.6 - STABILITY CLASSIFICATION CHART

STABILITY CLASSIFICATION	NMPC TURBULENCE CLASS	PASQUILL CAT.	TEMP CHANGE WITH HEIGHT, °F/70ft[1]	σ_θ DEGREES RANGE OF VALUES[2]	TEMP CHANGE WITH HEIGHT, °F/168ft[3]
Extremely Unstable	I	A	$\Delta T/\Delta Z \le -0.73$	$22.5 \le \sigma\theta <$	$\Delta T/\Delta Z \le -1.75$
Moderately Unstable	II	B	$-0.73 < \Delta T/\Delta Z \le -0.65$	$17.5 \le \sigma\theta < 22.5$	$-1.75 < \Delta T/\Delta Z \le -1.57$
Slightly Unstable	II	C	$-0.65 < \Delta T/\Delta Z \le -0.58$	$12.5 \le \sigma\theta < 17.5$	$-1.57 < \Delta T/\Delta Z \le -1.38$
Neutral	III	D	$-0.58 < \Delta T/\Delta Z \le -0.19$	$7.5 \le \sigma\theta < 12.5$	$-1.38 < \Delta T/\Delta Z \le -0.46$
Slightly Stable	IV	E	$-0.19 < \Delta T/\Delta Z \le 0.58$	$3.8 \le \sigma\theta < 7.5$	$-0.46 < \Delta T/\Delta Z \le 1.38$
Moderately Stable	IV	F	$0.58 < \Delta T/\Delta Z \le 1.53$	$2.1 \le \sigma\theta < 3.8$	$1.38 < \Delta T/\Delta Z \le 3.69$
Extremely Stable	IV	G	$1.53 < \Delta T/\Delta Z$	$\sigma\theta < 2.1$	$3.69 < \Delta T/\Delta Z$

(1) Adjusted to correspond to the ΔT measured between the 30-foot and 100-foot levels.

(2) Note on symbol convention "$3.8 \le \sigma\theta < 7.5$" means that $\sigma\theta$ is greater than or equal to 3.8 degrees but less than 7.5 degrees.

(3) Adjusted to correspond to the ΔT measured between the 30-foot and 200-foot levels.

ATMOSPHERIC STABILITY CHARACTERIZATION

A. (I) Mid-afternoon only, with clear skies or skies with very few thin clouds; late spring to early fall, winds usually are below 6 miles per hour.

B. (II) Late morning to mid-afternoon only, with clear or partly cloudy skies; mid spring to mid-fall, winds are usually below 9 miles per hour.

C. (II) Late morning to late afternoon only, with partly cloudy skies; spring through fall, winds are usually below 11 miles per hour.

D. (III) All daytime, with overcast or partly cloudy skies or early morning and late afternoon with clear or partly cloudy skies, all night time with overcast skies or partly cloudy year around, winds are moderate to high (greater than 6 miles per hour).

E. (IV) Typically night time only, with thin overcast or partly cloudy skies, all year around, winds less than 10 miles per hour.

F. (IV) Typically night time only, with clear to partly cloudy skies, all year around, winds less than 7 miles per hour.

G. (IV) Typically night time only, with clear skies or very few thin clouds all year around, winds less than 5 miles per hour.

2.2 (Cont)

e. Enter Total Stack Flow (kcfm). Use computer point C320 or calculate from Table 4.1.

f. Hit the "F9" key.

g. Print results.

2.3 <u>Stack Teletector</u>

a. Enter the time that the reading was obtained (24-hour format).

b. Enter the monitor reading (mrem/hr).

c. Enter the calibration factor. If unavailable, use default value of 0.5.

d. Enter Total Stack Flow (kcfm). Use computer point C320 or calculate from Table 4.1.

e. Hit the "F9" key.

f. Print the results.

2.4 <u>Grab Sample (Noble Gas)</u>

* *

<u>CAUTION</u>

In using grab samples to determine release rate, the results may be invalid if significant changes in source terms have occurred since the sample was taken.

* *

a. Enter the time that the reading was obtained (24-hour format).

b. Enter total Noble Gas concentration (μCi/cc).

c. Enter Total Stack Flow (kcfm). Use computer point C320 or calculate from Table 4.1.

d. Hit the "F9" Key.

e. Print the results.

2.5 <u>Back Calculation</u>

<u>NOTE</u>: Use back calculation of downwind survey team data to determine release rate when no other method is available, AND to verify calculated release rates.

a. Enter the time that the reading was obtained (24-hour format).

b. Enter the wind speed (mi/hr). Use the method described in Attachment 3.

c. Enter "E" for elevated/stack or "G" for ground/vent release.

2.5 (Cont)

d. Enter the stability class (A-G).

e. Enter the three foot closed window reading from the ion chamber (mrem/hr). If readings are in CPM, then convert using 3500 CPM = 1 mrem/hr.

f. Enter the downwind distance that the above reading was obtained.

g. Hit the "F9" key.

h. Print the results.

2.6 <u>FSAR</u>

NOTE: Input from the Control Room or TSC staff is necessary to select the FSAR accident type that most closely describes the conditions being experienced.

a. Select the accident being experienced or projected (Use Attachment 5, Table 5.1).

b. Print results.

2.7 <u>Containment High Range Monitor</u>

NOTE: This method is only valid if the monitor is able to "see" the release. Therefore, consult Operations personnel on the validity of monitor readings.

a. Enter the monitor ID or number.

b. Enter the time that the reading was obtained (24-hour format).

c. Enter the date that the reading was obtained.

d. Enter the time of reactor shutdown (24-hour format).

e. Enter the date that the reactor was shutdown.

f. Enter the monitor reading (rem/hr). Use computer point E467 or E468.

g. Enter the expected flow rate (kcfm) to the environment. Consult with Operations personnel if needed.

h. Hit the "F9" key.

i. Print results.

2.8 <u>For liquid releases, consult N1-CSP-M204</u>

3.0 UNIT 2 METHODS

3.1 GEMS

a. Enter the time that the reading was obtained (24-hour format).

b. Enter "S" if this is a stack reading or "V" if it is a vent reading.

c. Enter monitor reading (μCi/s). Use GEMS readings from SPDS display or the 882 panel. If offscale, use GEMS computer.

d. Hit the "F9" key.

e. Print results.

3.2 Grab Sample (Noble Gas)

* *

CAUTION

In using grab samples to determine release rate, the results may be invalid if significant changes in source terms have occurred since sample was taken.

* *

a. Enter the time that the reading was obtained (24-hour format)

b. Enter total Noble Gas reading (μCi/cc).

c. Enter total stack or vent flow (kcfm). Calculate from Figure 4.2 or 4.3.

d. Hit the "F9" Key.

e. Print the results.

3.3 Back Calculation

Use Section 2.5 of this Attachment.

3.4 USAR

Use Section 2.6 of this Attachment.

3.5 Containment High Range Monitor

Use Section 2.7 of this Attachment. Monitor readings are available on the DRMS system (RMS1a,b,c or d), the SPDS display or the 880 panel.

3.6 For liquid releases, consult N2-CSP-LWS-M203

TABLE 4.1
FLOW RATES CORRESPONDING TO FAN CONFIGURATIONS FOR UNIT 1

Drywell Vent, Purge, and Fill Line (10.00 KCFM)	KCFM
Turbine Building High Speed Fans (170.00 KCFM)	KCFM
Turbine Building Low Speed Fans (120.00 KCFM)	KCFM
Reactor Building High Speed Fans (70.00 KCFM)	KCFM
Reactor Building Low Speed Fans (35.00 KCFM)	KCFM
Waste Building (8.00 KCFM)	KCFM
Waste Building Extension (5.30 KCFM)	KCFM
Offgas Building (6.00 KCFM)	KCFM
Reactor Building Emergency Vent. (1.60 KCFM)	KCFM
RSSB Extension (10.25 KCFM)	KCFM
Total Stack Flow	KCFM

TABLE 4.2
FLOW RATES CORRESPONDING TO FAN CONFIGURATIONS FOR UNIT 2 STACK

CST Room 1 Fan (2200 SCFM)	Stack Substructure 1 Fan (1400 SCFM)	Turbine Building 1 Fan (40,000 SCFM)	Turbine Building 2 Fans (80,000 SCFM)	Standby Gas Treatment (4,000 SCFM)	Nominal SCFM	Nominal cm^3/sec
				X	4,000	1.89 E6
		X			40,000	1.89 E7
		X		X	44,035	2.08 E7
			X		80,000	3.78 E7
			X	X	84,000	3.96 E7
	X				1400	6.61 E5
X					2200	1.04 E6

TABLE 4.3
FLOW RATES CORRESPONDING TO FAN CONFIGURATIONS FOR UNIT 2 VENT

TB 250' & 306' Decon Rm 1 Fan (3300 SCFM)	Radwaste Liner 1 Fan (800 SCFM)	Radwaste Tanks 1 Fan (4910 SCFM)	Radwaste Building 1 Fan (47,800 SCFM)	Radwaste Building 2 Fans (95,600 SCFM)	Aux Boiler (23,000 SCFM)	Refueling Floor Above (70,000 SCFM)	Refueling Floor Below (70,000 SCFM)	Nominal SCFM	Nominal cm^3/sec
			X					47,800	2.256 E7
				X				95,600	4.512 E7
			X		X			70,800	3.341 E7
				X	X			118,600	5.597 E7
			X			X	X	187,800	8.864 E7
				X		X	X	235,600	1.112 E8
			X		X	X	X	210,800	9.948 E7
				X	X	X	X	258,600	1.22 E8
		X						4910	2.317 E6
	X							800	3.775 E5
X								3300	1.557 E6

1.0 DOSE ASSESSMENT

1.1 General Considerations

1.1.1 The dose assessment program is called RADDOSE.

1.1.2 Meteorological data is automatically sent to RADDOSE by the Meteorological Monitoring System (MMS). The user can use this data or manually input data.

1.1.3 Source term and release rate determination is identical to that described in Attachment 4.

1.2 Dose Assessment Procedure

NOTE: The dose assessment model has many capabilities beyond those used in this procedure. Use the "EDAMS Operators Manual" (available in the EOF) for further reference.

1.2.1 Log on to EDAMS computer using Attachment 2.

1.2.2 Select the affected Unit "Dose Assessment Model."

1.2.3 Utilize "EDAMS Data Entry Form", Figure 5.2, or equivalent.

1.2.4 Select "Begin New Incident" at the options.

1.2.5 Select "Yes" to erase all previous data when prompted.

1.2.6 Enter the following at the Accident Scenario Definition screen:

a. Reactor Trip Date. This is the date that the reactor scrammed or was manually tripped. IF the reactor is not shut down, enter tomorrow's date.

b. Reactor Trip Time (24-hour format). This is the time that the reactor scrammed or was manually tripped.

c. Release Date. This is the date that the release to the atmosphere began, or is projected to begin.

d. Release Time (24-hour format). This is the time that release to atmosphere began or is projected to begin.

e. Enter the lake temperature (deg F). If unknown, hit "Enter" and historical data will be entered.

f. Enter the initials of the user (two or three initials).

g. Verify entries, make any necessary changes, and select accept to continue.

1.2.7 Select "Enter/Edit Source Term Data" from the EDAMS main menu.

NOTES: 1. Use Attachment 4 to obtain the information needed to complete this section.

2. The preferred source of release rate data is the actual isotopic distribution, if available.

a. Select "Accident Type" by choosing the accident that most suits the current conditions. Use Table 5.1 in making the choice.

b. Select "Yes" for elevated releases OR "No" for ground releases when asked, "Is this release Elevated?".

NOTE: "Elevated" releases are releases from the stack. "Ground" releases are from any other release point.

c. Select the "Method" used to determine the release rate by selecting the highlighted cell or by hitting the "F2" key and selecting. Enter the "Flowrate" and "Monitor Reading" if required.

d. Select the Iodine release rate "Method" by selecting the highlighted cell or by hitting the "F2" key. Enter the "Monitor Reading" and "Release Rate" if required.

e. Up to three Accident Types (and therefore three release paths) can be entered. To enter additional release paths, repeat Steps a - d above. When all applicable accident types have been entered, proceed to the next step.

f. Upon completion of this screen, verify data and make any necessary changes before selecting "Accept".

1.2.8 The user will be queried only for the meteorological data required. Enter meteorological data as required:

a. Select "Enter/Edit Meteorological Data", Elevated or Ground as appropriate.

b. If the MMS is available, the data will be automatically displayed for the current time step.

1. Select "Requery MMS".

2. Select "Accept" as necessary.

c. If the MMS is unavailable, enter met data obtained from alternate sources, as outlined in Attachment 3 of this procedure and select "Accept".

1.2.9 Select "Perform Calculations" from the EDAMS main menu and "Verify source term data before calculating "as prompted.

* *

CAUTION

Any calculations performed on actual data shall be verified. The ODAM may act as the checker for calculations performed by the Rad Assessment Staff.

* *

a. The map of the 10 mile Emergency Planning Zone (EPZ) will appear with centerline dose rates when the calculation is complete.

b. Select "Continue" to go to the output menu.

c. Select "Continue Calculations" from the output menu.

d. Select "Perform Forecast" from the RADDOSE main menu.

e. Verify meteorology and source term data as required.

f. Enter "Forecast Period" (i.e. - release duration). Use 4 hours as a default value.

g. Select "OK".

h. After the forecast map appears select "Continue" to go to the output menu.

i. Select "Go to Report Menu".

j. Select "Print 10-Mile ERPA Map".

k. Select "Print Complete Dose/Dose Rate Report".

ℓ. Attach results of Step 1.2.9.j and k to EDAMS Data Entry Form, Attachment 5.2 or equivalent.

m. Verify that any results are supported by radiological and plant conditions. Consider:

- Core damage
- Drywell high range monitor readings
- Effluent monitor readings
- Inplant radiological conditions
- Containment hydrogen monitor readings

n. Document the verification of the calculation using the signature lines on Figure 5.2 or equivalent.

2.0 REFINED PROTECTIVE ACTIONS

2.1 These actions are initiated for the purpose of verifying the adequacy of PARs made using Attachment 1 of this procedure **OR** to develop PARs using projected doses obtained from Attachment 5, Step 1.2.9 of this procedure.

2.2 In determining PARs based on dose assessment, carefully consider factors such as release duration and Evacuation Travel Time Estimates (ETTE). (For example, puff releases may yield doses in excess of Protective Action Guidelines for an evacuation, but the plume will pass before an evacuation could be completed). ETTEs are available in the EOF.

NOTE: County and State PARs take many factors into account that NMP procedures do not (i.e. - road conditions, special population needs, evacuation scenarios, and shelter vs evacuation doses). Therefore, differences in PARs may occur. The ODAM must account for differences in PARs, when those differences exist. This can be accomplished via consultation with County and State representatives in the EOF as to the assumptions used in their dose calculations and PAR development.

2.3 Obtain dose projection for each ERPA.

2.3.1 PARs are listed on the 10 mile ERPA map obtained per Attachment 5, Step 1.2.9. j.

2.3.2 The following criteria are used in determining the PAR for each ERPA.

PAR	TEDE (rem)	CDE_T (rem)
Evacuate	> 1	> 5

2.3.3 Record the PAR for each ERPA on the Part 1 Notification Form and give to the CED for approval.

2.3.4 PARs that have been made previously must be accounted for when PARs are revised. For example, if a PAR to evacuate an ERPA was previously made to the State/County and that PAR does not appear on a revised map from 1.2.9.j, that PAR must still be included on the revised recommendation to the State/County.

2.3.5 If projected doses exceed values listed in Attachment 5 Step 2.3.2 for distances greater than 10 miles, PARs shall be made using convenient geographic boundaries (such as townships).

TABLE 5.1 - FSAR/USAR ACCIDENT TYPE

Accident Type	Noble Gas Release Rate (Ci/s)	Iodine Release Rate (Ci/s)	Analyzed Release Point
Unit 1:			
DBA Loss of Coolant	5.50E+0	4.53E-3	Elevated
Control Rod Drop	2.51E+1	6.03E-5	Elevated
Refueling Accident	3.78E-2	3.84E-5	Elevated
Steam Line Break	6.36E+0	4.86E+1	Ground
Loss of Coolant (Realistic)	1.79E-3	1.00E-6	Elevated
Unit 2:			
DBA Loss of Coolant	1.03E+1	2.03E-1	Elevated
Control Rod Drop	4.22E-2	4.70E-4	Ground
Refueling Accident	1.77E+1	1.65E-1	Ground
Steam Line Break	3.64E+0	1.22E+2	Ground
Rad Gas Waste System Leak	4.06E+0	0.00	Ground
Instrument Line Failure	0.00	2.17E-2	Ground
Fuel Cask Drop	2.06E+0	2.68E-3	Ground
Loss of Coolant (Realistic)	1.05E-2	2.38E-5	Elevated

NIAGARA MOHAWK

CHRONOLOGICAL RELEASE RATE LOG

Date:	Release Form:	Survey Location:	Completed By:

Effluent Monitor Data				Environmental Sampling Data								Release Log			
													Assigned Time Interval		
Time of Monitor Reading	Monitor System	Release Rate (Ci/sec)	Duration of Release	Survey Time	Location	Gamma Dose Rate (mR/hr)	Distance (Mi)	Wind Speed (mph)	*** Transit Time (min)	**** Est. Time of Release from Site	Release Rate (Ci/sec)	Assigned Release Rate (Ci/sec)	Start	Stop	ODAM Review

Notes: *** Transit Time (min) = (Distance/Wind Speed) x 60 min/hr **** Est. Time of Release = Survey Time - Transit Time

☐ "What If"
☐ Actual Data (Checker Required!)

Rx Trip: Date: ________ Time: ________
☐ Unit 1 ☐ Unit 2

Release:
Release Duration (Hr): ____________

Accident Type	Release Point *(Circle One)*
Containment DBA	Elev/Grd
Control Rod Drop	Elev/Grd
Refueling Accident	Elev/Grd
Steam Line Break	Elev/Grd
Loss of Coolant	Elev/Grd
Rad Gas Waste System	Elev/Grd
Inst. Line Failure	Elev/Grd
Fuel Cask Drop	Elev/Grd
Severe Accident	Elev/Grd

Flow Rate ____________

Method ____________

Monitor Reading ____________

Iodine Method ____________

Iodine Monitor ____________

Met Data: ☐ Automatic ☐ Manual *(Below)* Lake Temp ____°F or Default

	Elevated	Ground
Wind Speed (mi/hr)		
Wind Direction (from - degrees)		
Stability (A-G)		
Temperature (°F)		
Precipitation (in/15 min)		

Attach:
- Map from color printer
- "Complete Dose/Dose Rate" report

Misc: ____

____

____

Calculations Performed By: ____________________

Calculations Verified By: ____________________

Question # SRO 18

Examination Outline Cross-Reference	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295025 EA2.04
	Importance Rating	3.9

Ability to determine and interpret the following as they apply to HIGH REACTOR PRESSURE: suppression pool level.

Proposed Question:

The plant is operating at 100% power when an improper valve lineup drains CST water into the suppression pool. As suppression pool level rises the crew enters N2-SOP-101C, REACTOR SCRAM. During the scram a Group 1 isolation occurs. The following conditions exist:

- RPV pressure is **1042 psig** and rising
- RPV level is **223 inches**
- Suppression Pool Level is **210 feet** and rising

Which one of the following actions should be taken?

a. Immediately open all 7 ADS valves and blowdown.

b. Open an SRV to lower pressure to less than 870 psig.

c. Position turbine bypass valves to depressurize the reactor.

d. Place RHS in steam condensing and RCIC in full flow test lineup.

Proposed Answer: a. PC control 16/5

Explanation (Justification of Distractors):

b. SRVs will NOT restore SP level or RPV pressure to within the SRV tailpipe level limit
c. This would take time while pressure and SP level are rising.
d. RCIC is NOT available because of high RPV level.

Technical Reference(s): PC Control SPL leg, step 16/5

Proposed references to be provided to applicants during the examination:

EOPs without entry conditions

Learning Objective:	N2-OPS-006-344-2-04, EO-2.0
Question Source:	Bank # Modified Bank # New NEW
Question History:	Previous NRC Exam Previous Test / Quiz
Question Cognitive Level:	Memory of Fundamental Knowledge 2 Comprehension or Analysis
10CFR Part 55 Content:	41.10 / 43.5 / 45.13

Comments:

Question # SRO 19

Examination Outline Cross-Reference	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295026 2.2.12
	Importance Rating	3.4
Knowledge of surveillance procedures.		

Proposed Question:

The plant is operating at 50% power with the following conditions:

- RHR A is in suppression pool cooling
- RCIC is operating in the CST to CST mode for a surveillance
- During the surveillance, suppression pool temperature reaches 96°F

Which one of the following describes the CRS consideration for entry into and execution of N2-EOP-PC, PRIMARY CONTAINMENT CONTROL.

a. Technical Specifications allow modification of the EOP entry condition to 105°F while performing this test.

b. Surveillance allows 4 hours to reduce suppression pool temperature below the EOP entry condition upon completion.

c. As soon as suppression pool temperature exceeds the EOP entry condition, the EOP must be entered and the actions performed.

d. EOP actions are deferred for 24 hours after the test if suppression pool temperature can be reduced below the EOP entry condition.

Proposed Answer: c.

Explanation (Justification of Distractors):
There are no provisions in Technical Specifications or other plant procedures that exempt entry into the EOPs under these conditions.

a. Tech Spec limit is raised when any surveillance test is being performed that adds heat to the Suppression Pool.
b. Surveillance procedure does NOT alter EOP entry conditions.
d. Tech Specs allows this for Tech Spec limit, not EOP entry.

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Technical Reference(s):	Technical Specifications, Section 3.6.2.1 NIP-PRO-01, Rev 06, Section 3.2.1		

Proposed references to be provided to applicants during the examination:

None.

Learning Objective: O2-OPS-006-344-2-04, EO-1

Question Source:	Bank # Modified Bank # New	 New	
Question History:	Previous NRC Exam Previous Test / Quiz	New New	
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis		1
10CFR Part 55 Content:	55.41.10 55.43.2 55.43.5 55.45.13		

Comments:
SRO Only: Technical Specification application. Control of plant testing and requirements associated with the testing.

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Question # SRO 20

Examination Outline Cross-Reference	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295026 EK1.01
	Importance Rating	3.4

Knowledge of operational implications of the following concepts as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE and the following: Pump NPSH.

Proposed Question:

The following conditions exist after a LOCA:

- RPV Level +24 inches and rising
- RPV Pressure 29 psig
- Drywell Pressure 12 psig
- Suppression Pool Water Temperature 255°F
- Suppression Pool Pressure 9 psig
- Suppression Pool Level 194 ft
- RHR "B" LPCI Flow 4500 gpm

With regard to the "B" RHR Loop which one of the following is required?

a. Raise RHR pump flow rate to 7000 gpm to restore reactor water level.

b. Monitor RHR pump flow rate and do **NOT** allow flow to exceed 6000 gpm.

c. Enter Attachment 3 of EOP-6, EOP SUPPORT PROCEDURE, and throttle RHR flow to less than 2000 gpm.

d. Shutdown the RHR Pump and establish injection from sources **NOT** taking a suction from the Suppression Pool.

Proposed Answer: b. The high pressure in the containment ensures NPSH for this temperature; 194 ft = 6.5 psig + 9 psig = 15.5 psig

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Explanation (Justification of Distractors):

a. There is a restriction on RHR flow.
c. RHR flow does not require throttling
d. There is no justification to shutdown the pump.

Technical Reference(s): N2-EOP-6, Attachment 29

Proposed references to be provided to applicants during the examination:

N2-EOP-6, Attachment 29

Learning Objective:

Question Source:	Bank # Modified Bank # New	 NEW
Question History:	Previous NRC Exam Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 3
10CFR Part 55 Content:	55.41.8 55.41.10 55.43.5	

Comments:
SRO Only: ECCS pump vortex limits and NPSH determination.

1.0 PURPOSE

1.1 To provide instruction for determining Heat Capacity Temperature Limit(HCTL), Pump Net Positive Suction Head (NPSH) and Vortex Limits for Emergency Core Cooling Systems (ECCS) and Reactor Core Isolation Cooling (RCIC). This attachment also provides direction for determining when the bounding HCTL analysis and limits of a Station Blackout have been exceeded or are no longer valid.

1.2 Applicability

1.2.1 Used to support the NMP2 in determining HCTL by use of an alternate curve when suppression pool level is below elevation 195 ft.

1.2.2 Used to support the NMP2 EOPs when a caution exists in the applicable procedure warning of system damage if suppression pool level is below elevation 195 ft. Containment overpressure is calculated to allow accurate use of the pump NPSH Limit curves by taking credit for the pressure existing in the suppression chamber.

1.2.3 Used to support the NMP2 EOPs when a Station Blackout has occurred.

2.0 TOOLS AND MATERIALS

None

3.0 PROCEDURE

3.1 Determining HCTL when suppression pool level is below El. 195 ft.

NOTE: When suppression pool level is below El. 197 ft., suppression pool temperature may be only be determined using temperature recorder E21-R601 *(2CEC*PNL601)* **OR** as read locally using a hand held pyrometer on the pump suction line. Each method requires a pump running with suction on the suppression pool.

3.1.1 Obtain Suppression Pool Temperature . . . . (__)

3.1.2 Obtain RPV Pressure as read in the Control Room . . . (__)

3.1.3 Using figure 29-2, determine HCTL **AND** refer to EOPs for required actions . . . . (__)

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3.2 Determining pump NPSH limits.

3.2.1 Obtain Suppression Chamber Pressure, as read in the Control Room, as Suppression Chamber Overpressure . . (__)

NOTE: IF Suppression Chamber Overpressure falls between the curves on the pump NPSH limit figure, **THEN** the lower curve is used.

3.2.2 Obtain system flow for pump in question, as read in the Control Room . . . . (__)

3.2.3 Using the appropriate figure for the pump in question (Figures 29-3 through 29-6) determine if the pump NPSH limit has been exceeded . . . . (__)

3.2.4 IF a pump NPSH Limit curve is exceeded, perform the following:

N/A, No pump NPSH Limits were exceeded . . . . (____)

3.2.5 Record Suppression Pool Water Level:

__________ Feet . . . . (__)

3.2.6 Using the value of Suppression Pool Water Level obtained in Step 3.2.1 **AND** Figure 29-1, Hydrostatic Head Pressure Determination, determine the Hydrostatic Head Pressure due to the height of water in the Suppression Pool **AND** record below:

__________ psig . . . . (__)

3.2.7 Record Suppression Chamber Pressure:

_________ psig . . . . (__)

3.2.8 Calculate Suppression Chamber Overpressure as follows:

________psig +	________psig =	________psig . . . . (__)
Suppression Chamber Pressure	Hydrostatic Head Pressure	Suppression Chamber Overpressure

3.2.9 Use the calculated value of Suppression Chamber Overpressure with the pump NPSH Limit curves (Figures 29-3 through 29-6 . . . . (__)

3.2.10 **IF** an NPSH limit is still being exceeded **AND** throttling of the pumps discharge flow will **NOT** prevent meeting the flowchart objective, **THEN** throttle the pump discharge flow using:

- Attachment 3 of this procedure for ECCS pumps . (__)
- Attachment 4 of this procedure for RCIC . . . . (__)

N/A, No NPSH Limits have been exceeded . . . . (____)

3.3 Determining pump Vortex limits.

3.3.1 Obtain Suppression Pool Level, as read in the Control Room . . . . (__)

3.3.2 Obtain system flow rate for the pump in question as read in the Control Room . . . . (__)

3.3.3 Using the appropriate figure for the pump in question (Figures 29-3 through 29-6) determine if the pump Vortex limit has been exceeded . . . . (__)

3.3.4 **IF** the Vortex limit is being exceeded **AND** throttling of the pumps discharge flow will **NOT** prevent meeting the flowchart objective, **THEN** throttle the pump discharge flow using:

- Attachment 3 of this procedure for ECCS pumps . (__)
- Attachment 4 of this procedure for RCIC . . . . (__)

N/A, No Vortex Limits have been exceeded . . . (____)

3.4 RCIC operation during Station Blackout

NOTES:
1. This section is applicable only when a Station Blackout exists as described in the Station Blackout Bases.
2. The Station Blackout analysis assumes no other malfunctions or accidents exist other than those that resulted in the Station Blackout condition.
3. During a Station Blackout, the Heat Capacity Temperature Limit is 186°F with RPV pressure less than or equal to 200 psig. RPV Blowdown for exceeding HCTL is not required unless this limit is exceeded.

3.4.1 **IF** other malfunctions **OR** accidents develop that result in conditions other than those assumed in the Station Blackout Bases, **THEN** exit this sub-section and observe the Heat Capacity Temperature Limit as shown on the NMP2 EOPs . . . . (__)

N/A, No conflict exists with SBO Bases . . . . (____)

3.4.2 Continue to monitor RPV Pressure and Suppression Pool Temperature as read in the Control Room . . . . (__)

4.0 RESTORATION

None

FIGURE 29-1
SUPPRESSION CHAMBER OVERPRESSURE

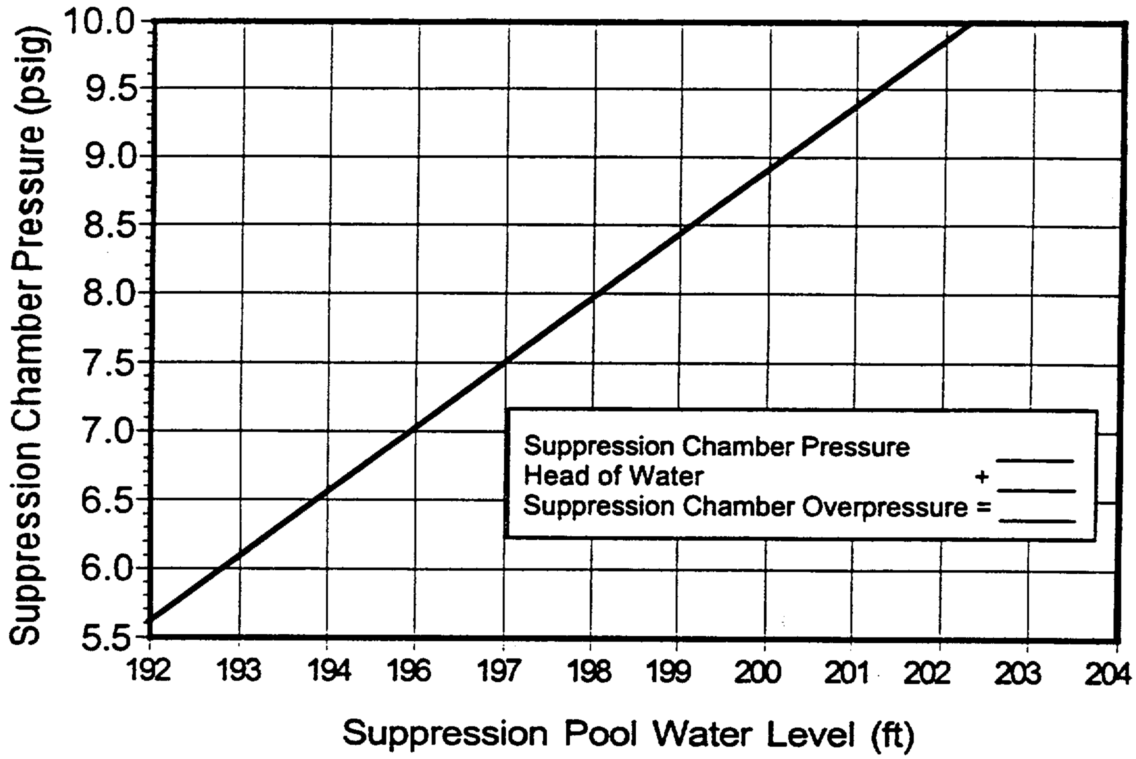


FIGURE 29-2
ALTERNATE HCTL CURVE

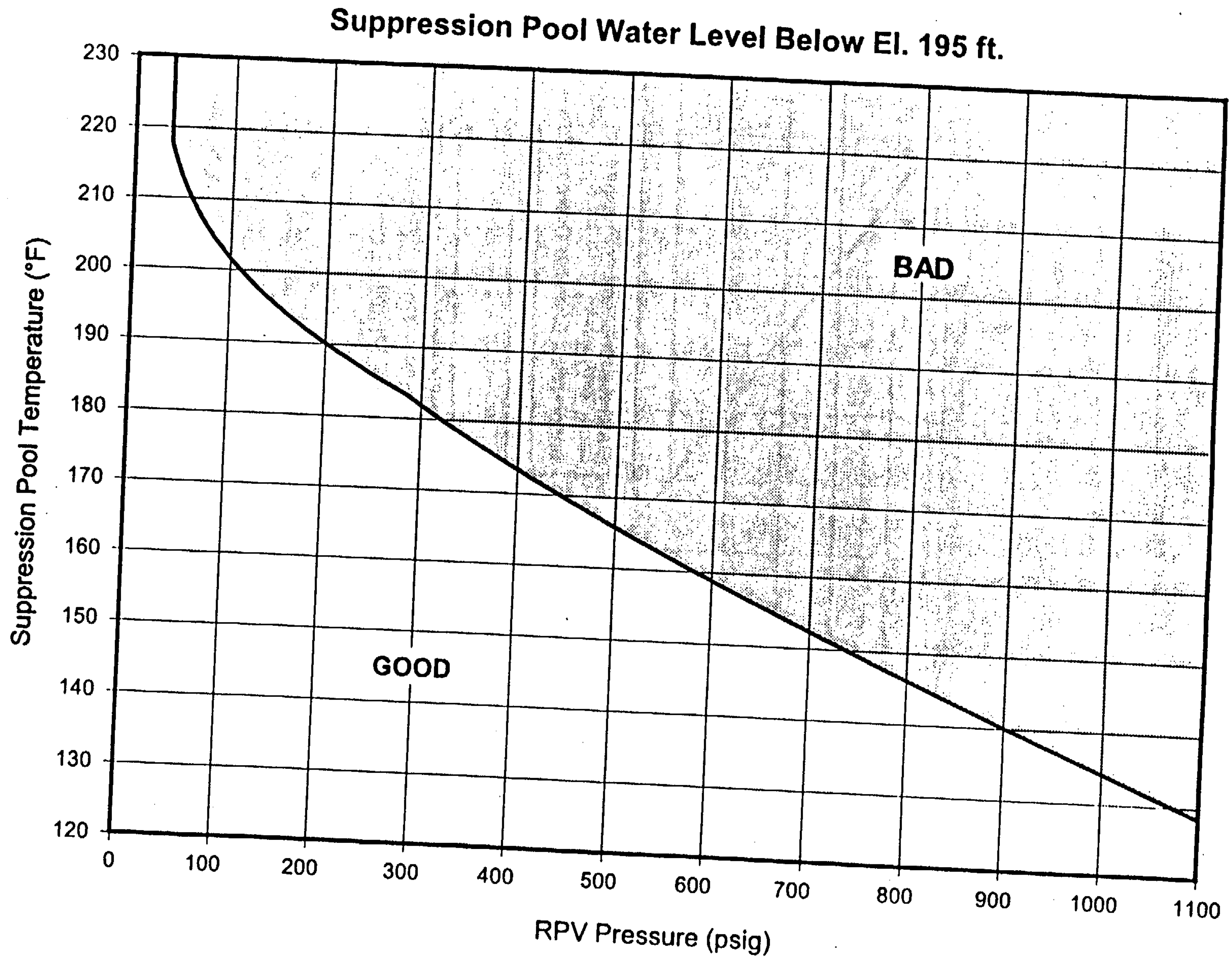
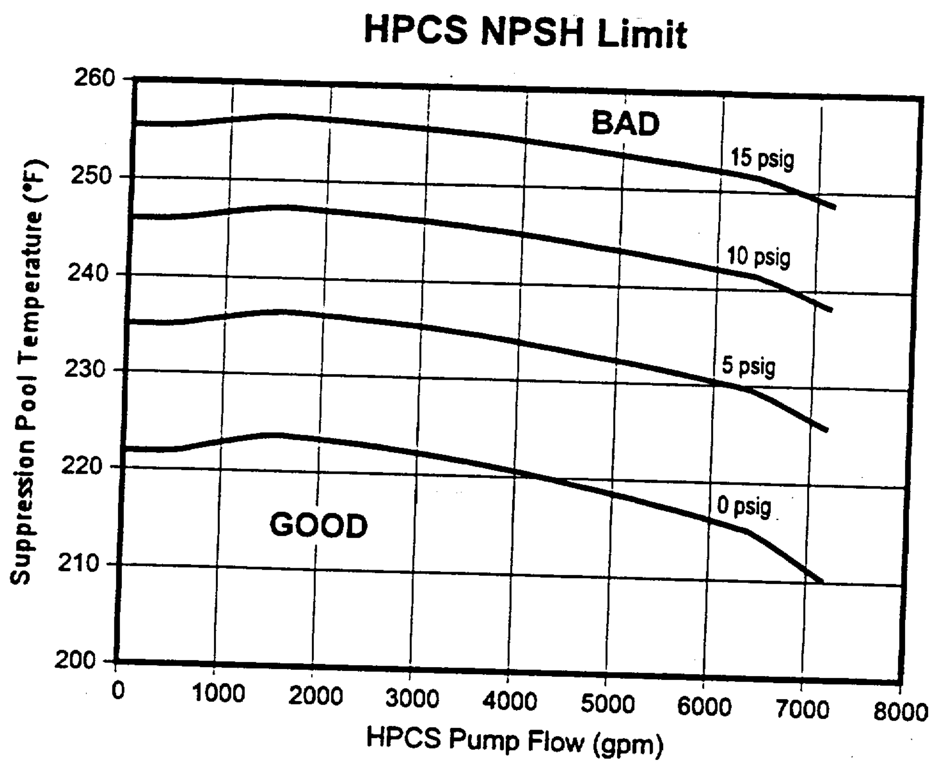


FIGURE 29-3
HPCS NPSH AND VORTEX LIMITS



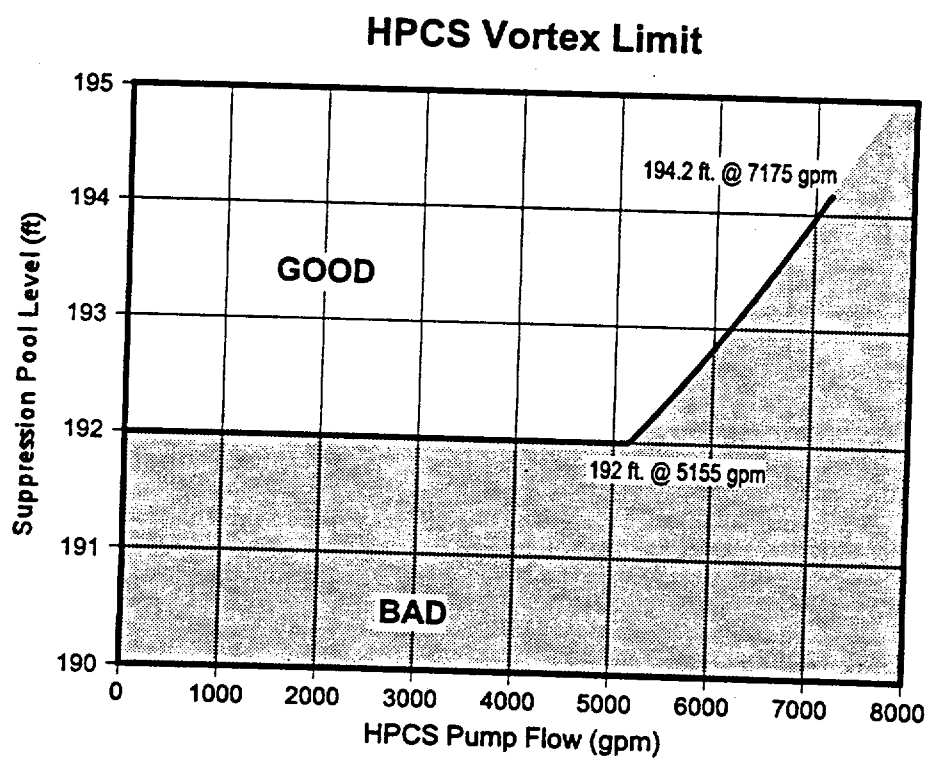
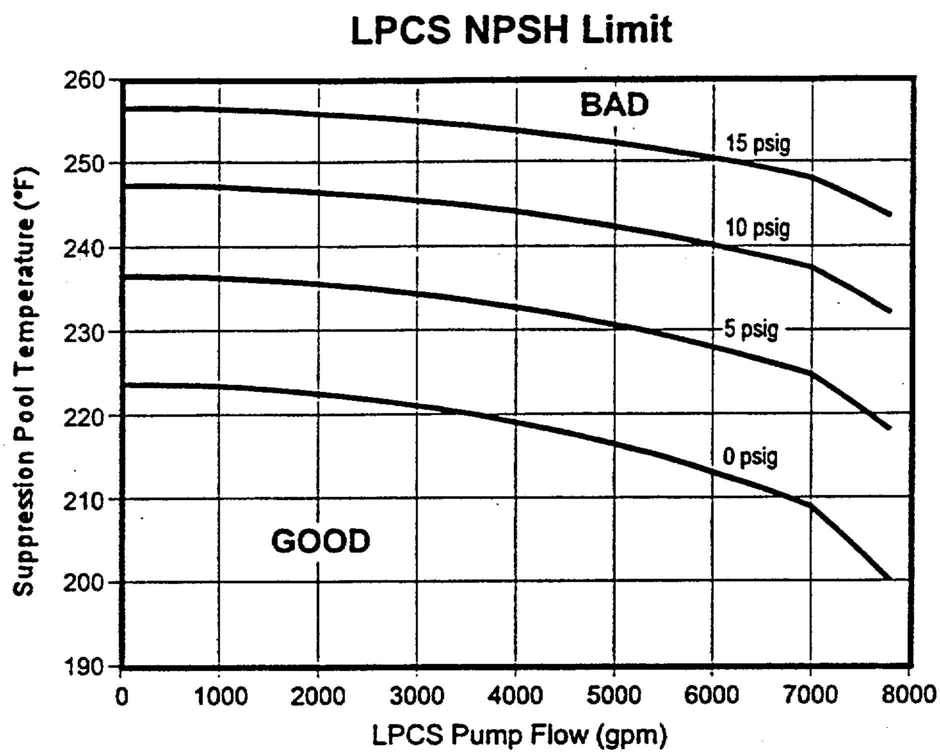


FIGURE 29-4
LPCS NPSH AND VORTEX LIMITS



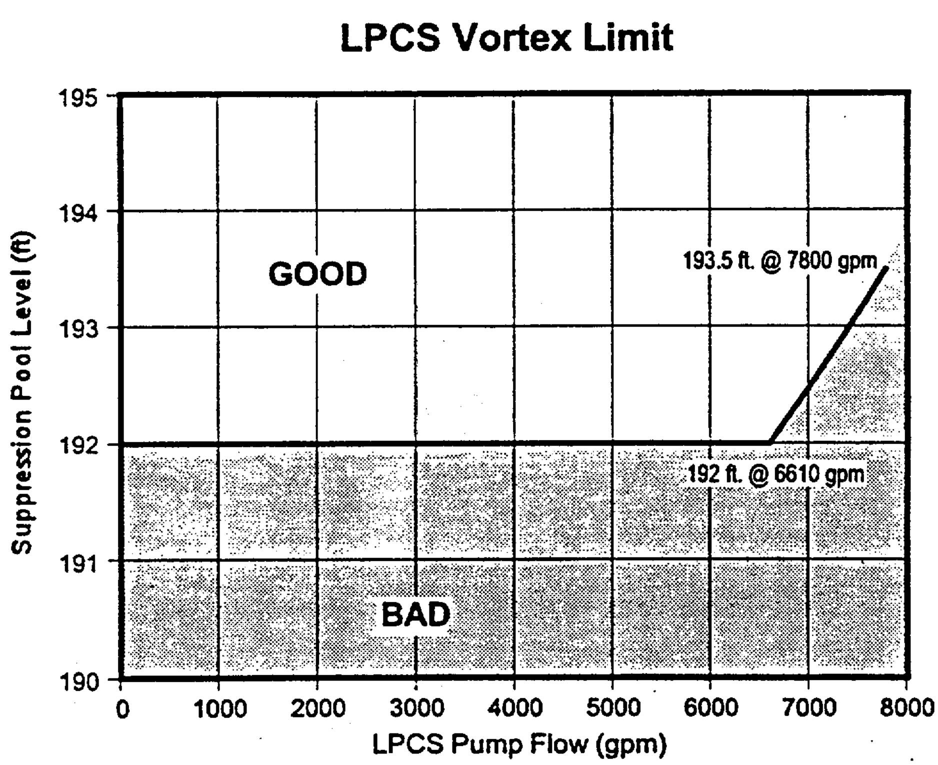
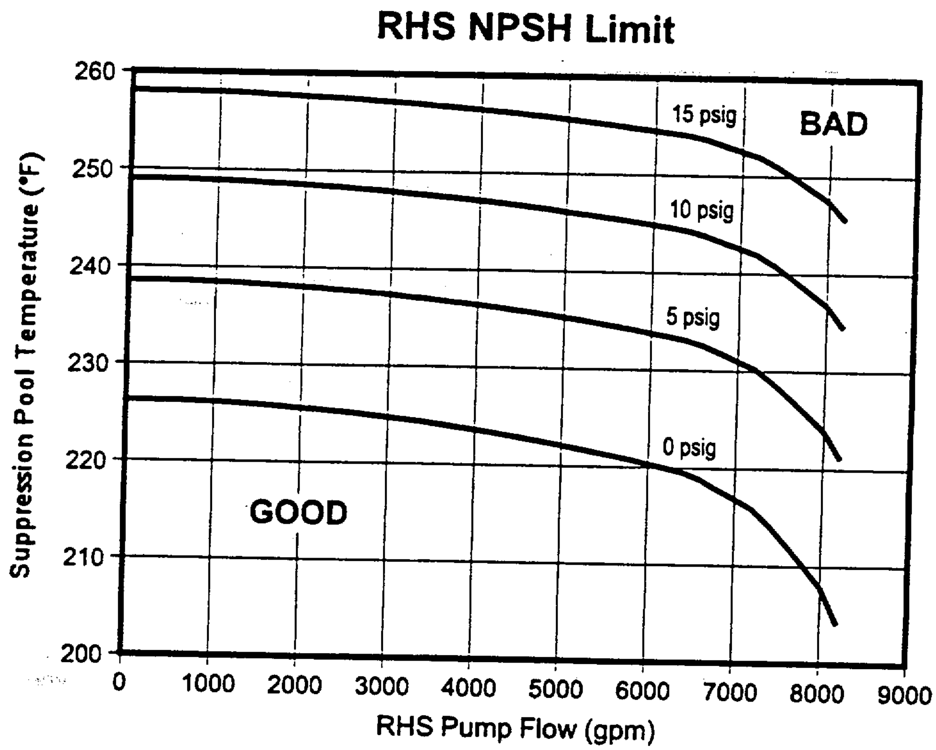


FIGURE 29-5
RHS NPSH AND VORTEX LIMITS



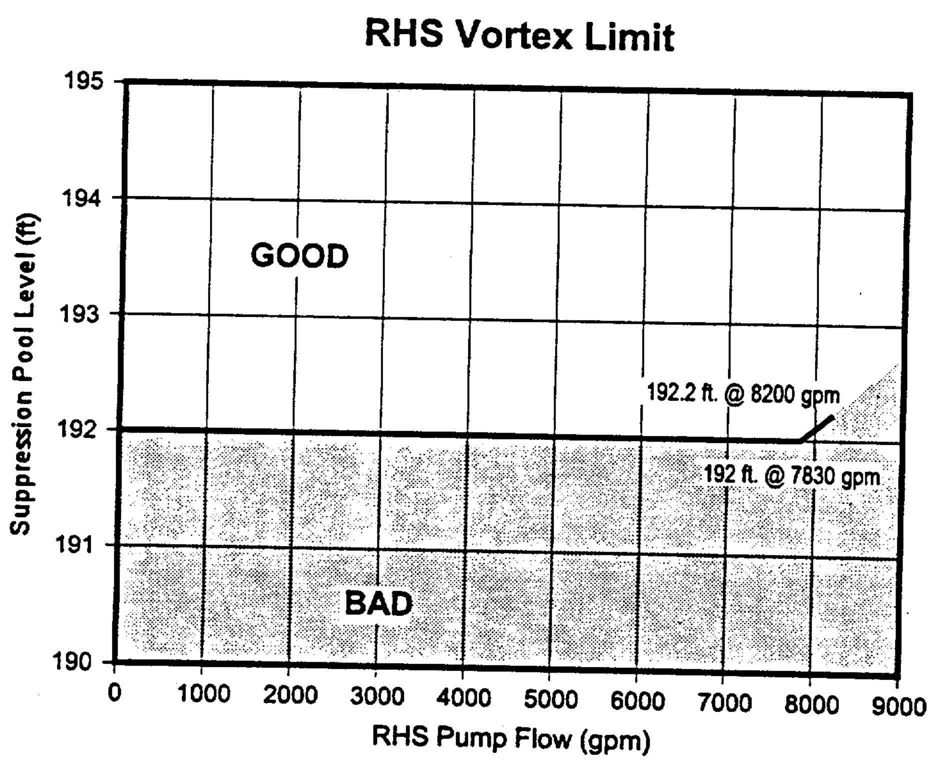
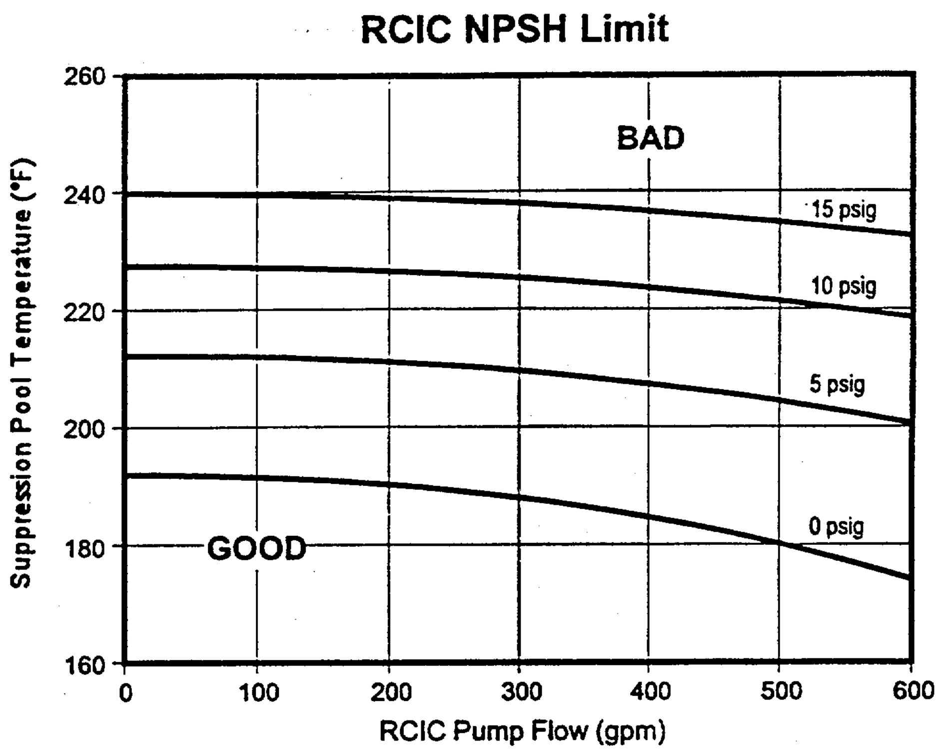
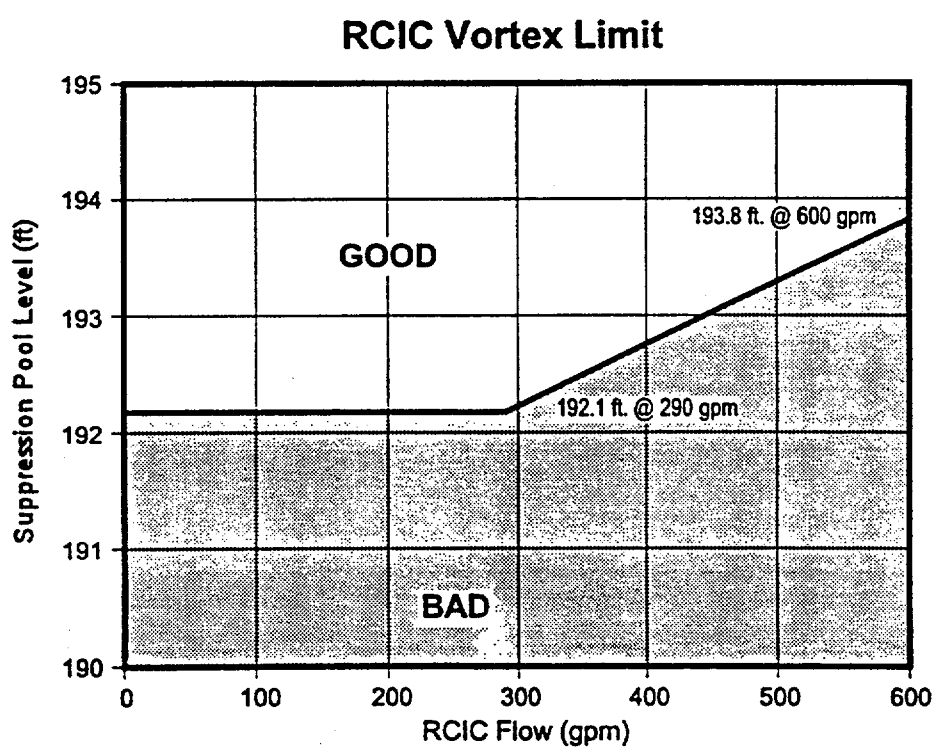


FIGURE 29-6
RCIC NPSH AND VORTEX LIMITS





test will be conducted to ensure that the frequencies used for portable communications systems will not affect the actuation of protective relays. If repeaters are used, they will be suitably protected from exposure to fire damage.

9A.3.6 Fire Detection and Suppression

9A.3.6.1 Fire Detection

As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 9A.3-18 shall be operable whenever safety-related equipment protected by the fire detection instrument is required to be operable.

Action

1. With any, but not more than one-half the total in any, fire zone, Function N*, fire detection instruments shown in Table 9A.3-18 inoperable, restore the inoperable Function N* instrument(s) to operable status within 14 days, or within 1 hr establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour.

2. With more than one-half the Function N* fire detection instruments in any fire zone shown in Table 9A.3-18 inoperable, or with any Functions S* and X* instruments shown in Table 9A.3-18 inoperable, or with any two or more adjacent instruments shown in Table 9A.3-18 inoperable, within 1 hr establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour.

Surveillance Requirements

The above-required fire detection instruments shall be demonstrated operable at least annually in accordance with the following test methodology.

At least 10 percent of the installed detectors and a minimum of one detector in each detection loop will be tested by initiating an alarm at the detector in its installed location (channel functional test). Should a detector fail to alarm under the simulated fire condition, it will be corrected per procedure, and an additional 20 percent, minimum of two, detectors in the affected loop will be tested. Should a failure to alarm occur in this expanded sample population, the failure will be corrected per procedure, and all remaining detectors in the affected loop

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* These letters are found in the alphanumeric fire zone designation and are explained in the footnote of Table 9A.3-18.

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will be tested and corrected as necessary. This testing scheme will be cycled through all detectors in a detection loop until all detectors in the loop have been tested. The cycle will then be repeated. Where the above outlined testing cannot be accomplished within the required time frame due to accessibility or safety concerns during plant operation, the testing shall be performed during each cold shutdown exceeding 24 hr unless performed within the previous 12 months.

The supervised circuits associated with the detector alarms of each of the above-required fire detection instruments are continuously monitored by their associated fire alarm panel.

The nonsupervised circuits associated with detector alarms between the instruments and the control room shall be demonstrated operable at least once per 31 days.

9A.3.6.1.1 Fire Detection for Safety-Related Equipment

Class A, supervised fire detection is provided for areas that contain or present a fire exposure to safety-related equipment. Areas containing negligible fire exposure from cable loading and/or transient combustibles, as determined by the Unit 2 Fire Protection Engineer, do not require Class A supervised fire detection. Class B supervision is provided for actuation portions of the suppression system circuits.

9A.3.6.1.2 Fire Detection (NFPA-72D and NFPA-72E)

The fire detection and signaling systems comply with the requirements of NFPA-72D and NFPA-72E, except that position switches for steel-bodied valves indicate the off-normal position when the stem of the valve has moved 2/5 the distance from the normal position. Location and placement of detectors is in accordance with the guidelines of NFPA-72E, except where special conditions do not permit. In those cases, detectors are located using criteria developed by qualified Fire Protection Engineers, based on engineering judgment, as permitted by NFPA-72E. Refer to Table 1.9-1, Note 57 (Difference 9), and Table 9.5-3 for interpretation and specific deviations from NFPA-72D and NFPA-72E.

9A.3.6.1.3 Testing of Detectors

Preoperational and periodic testing of detectors will not affect the actuation of protective relays in other plant systems. Unit 2 does not use pulsed-line type detectors.

9A.3.6.1.4 Audible and Visual Alarms

Fire detection systems give audible and visual annunciation in the control room and local audible alarms, and annunciate at the local panels.

9A.3.6.1.5 Unique Alarms

Fire alarms are distinctive and unique from other plant system alarms.

9A.3.6.1.6 Power Supplies for the Fire Detection System

The Unit 2 fire detection system is fed through the two stub buses through an automatic transfer switch. The primary source which feeds the two stub buses is the unit generator, and the secondary sources include either of the two offsite power sources, or the diesel generators. In case of the loss of the primary source, the transfer switch automatically connects the system to the secondary source. This arrangement satisfies the intent of the requirements of NFPA-72D, Section 2220. For further details on the Unit 2 electrical system, refer to Section 8.3.

9A.3.6.1.7 Design Bases for Fire Detection Instrumentation

Operability of the detection instrumentation addressed in Section 9A.3.6.1 ensures that adequate warning capability is available for prompt detection of fires and that fire suppression systems that are actuated by fire detectors will discharge extinguishing agent in a timely manner. Prompt detection and suppression of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

Fire detectors that are used to actuate fire suppression systems represent a more critically important component of a plant's fire protection program than detectors that are installed solely for early fire warning and notification. Consequently, the minimum number of operable fire detectors must be greater.

The loss of detection capability for fire suppression systems, actuated by fire detectors, represents a significant degradation of fire protection for any area. As a result, the establishment of a fire watch patrol must be initiated at an earlier stage than would be warranted for the loss of detectors that provide only early fire warning. The establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to operability.

9A.3.6.2 Fire Protection Water Supply System

The requirements can be found in Sections 9A.3.6.2.6 and 9A.3.6.2.7.

9A.3.6.2.1 Yard Fire Main Loop

An underground yard fire main loop is designed to meet the anticipated water requirements. The installation of the fire

TABLE 9A.3-18

FIRE DETECTION INSTRUMENTATION

Instrument Location			Total Number of Instruments**		
Fire Zone*	Room or Area	Elev.	Heat	Ionization	Photo-Electric
Reactor Building/Auxiliary Bays					
201SW	CCP Ht Exch & LPCS Pump Room	175'-0"	NA	16	NA
202SW	RHR Pump A Room	175'-0"	NA	7	NA
203SW	RHR Ht Exch A Room	175'-0"	NA	6	NA
204SW	RCIC Pump Room	175'-0"	6	NA	NA
205NZ	HPCS Pump Room	175'-0"	NA	7	NA
206SW	RHR Ht Exch B Room	175'-0"	NA	8	NA
207SW	RHR Pump B Room	175'-0"	NA	7	NA
208SW	RHR Pump C Room	175'-0"	NA	11	NA
212SW	Gen Area North	175'-0" 196'-0"	13	34	NA
213SW	Gen Area South	175'-0" 196'-0"	20	35	NA
211SW	N Aux Bay Above Pump Rooms	198'-0"	NA	22	NA
214SW	S Aux Bay Above Pump Rooms	198'-0"	NA	22	NA
221SW	N Aux Bay Above Pump Rooms	215'-0"	NA	28	NA
222SW	Gen Area 0°-180°	215'-0"	NA	39	NA
223SW	Gen Area 180°-360°	215'-0"	NA	39	NA
224SW	S Aux Bay Above Pump Rooms	215'-0"	NA	25	NA
231SW	N Aux Bay Elect MCC Area	240'-0"	NA	30	NA
232SW	Gen Area 0°-180°	240'-0"	5	32	NA
234NZ	Primary Containment (Outage Only)	261'-0" 289'-0"	NA NA	NA NA	7 4

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TABLE 9A.3-18 (Cont'd.)

Instrument Location			Total Number of Instruments**		
Fire Zone*	Room or Area	Elev.	Heat	Ionization	Photo-Electric
Reactor Building/Auxiliary Bays (cont'd.)					
238SW	Gen Area 180°-360°	240'-0"	1	32	NA
239SW	S Aux Bay Elect MCC Area	240'-0"	NA	29	NA
243SW	Gen Area 0°-180°	261'-0"	5	38	NA
245SW	Gen Area 180°-360°	261'-0"	2	37	NA
252SW	Gen Area 0°-180°	289'-0"	NA	39	NA
253XL	Elect Load Center Room	289'-0"	NA	6	NA
255SW	Gen Area 180°-360°	289'-0"	4	33	NA
261SW	Pipe Chase 0°-180°	306'-6"	14	NA	NA
262SW	Gen Area 180°-360° & Pipe Chase 180°-360°	306'-6"	NA	26	NA
271SW	Gen Area 0°-90°	328'-10"	NA	19	NA
272SW	Gen Area 270°-360°	328'-10"	NA	19	NA
273SW	Gen Area 90°-180°	328'-10"	NA	15	NA
274SW	Gen Area 180°-270°	328'-10"	NA	19	NA
281NZ	Gen Area 0°-360°	353'-10"	NA	84	NA
Control Building					
305NW	Div I Riser Area	214'-0"	NA	4	NA
306NW	Div I Cable Area	214'-0"	NA	13	NA
307NZ	24V Battery Room	214'-0"	NA	1	NA
308NZ	24V Battery Room	214'-0"	NA	1	NA
309NW	Div II Cable Chase	214'-0"	NA	5	NA
311NZ	Computer Battery Room	214'-0"	NA	3	NA
312NZ	Div II Cable Area	214'-0"	NA	9	NA

TABLE 9A.3-18 (Cont'd.)

Instrument Location			Total Number of Instruments**		
Fire Zone*	Room or Area	Elev.	Heat	Ionization	Photo-Electric
Control Building (cont'd.)					
321NW	Div I Riser Area	237'-0"	NA	4	NA
322NW	Div I Cable Area	237'-0"	NA	14	NA
323NW	Div II Cable Area	237'-0"	NA	15	NA
324NW	Div II Riser Area	237'-0"	NA	4	NA
325NW	Div I Cable Area	237'-0"	NA	5	NA
326NW	Div II Cable Area	237'-0"	NA	5	NA
327NW	Div III Cable Area	237'-0"	NA	6	NA
331NW	Corridor	261'-0"	NA	20	NA
332NW	Div I Cable Chase	261'-0"	NA	5	NA
333XL	Div I Switchgear Room	261'-0"	NA	7	NA
334NZ	Div I Battery Room	261'-0"	NA	4	NA
335NZ	Div II Battery Room	261'-0"	NA	4	NA
336XL	Div II Switchgear Room	261'-0"	NA	7	NA
337NW	Div II and III Cable Chase	261'-0"	NA	5	NA
338NZ	Remote Shutdown Room B	261'-0"	NA	2	NA
339NZ	HPCS Battery Room	261'-0"	NA	1	NA
340NZ	Div I Chiller Room	261'-0"	NA	2	NA
341NZ	Div II Chiller Room	261'-0"	NA	2	NA
342XL	Div III Switchgear Room	261'-0"	NA	4	NA
343NZ	Remote Shutdown Room A	261'-0"	NA	2	NA
351NZ	Instrument Room and Corridor	288'-6"	NA	17	NA
352NW	Div I Cable Chase	288'-6"	NA	4	NA

TABLE 9A.3-18 (Cont'd.)

Instrument Location			Total Number of Instruments**		
Fire Zone*	Room or Area	Elev.	Heat	Ionization	Photo-Electric
Control Building (cont'd.)					
353SG	Relay Room	288'-6"	50	106	NA
354SG	Relay Room	288'-6"	50	120	NA
356NZ	Relay Room	288'-6"	NA	14	NA
357XG	Computer Room	288'-6"	NA	8	NA
358XG	Computer Room	288'-6"	NA	4	NA
359NW	Div II and III Cable Chase	288'-6"	NA	5	NA
360NZ	HVAC Equipment Room	288'-6"	NA	11	NA
362SG	Relay Room	288'-6"	40	72	NA
371NW	Div I Cable Chase	306'-0"	NA	4	NA
373NZ	Control Room	306'-0"	NA	25	NA
374SG	Control Room	306'-0"	43	68	NA
375SG	Control Room	306'-0"	44	75	NA
376XG	Control Room	306'-0"	NA	11	NA
377NW	Div II and III Cable Chase	306'-0"	NA	3	NA
378NZ	HVAC Equipment Room	306'-0"	NA	9	NA
380NZ	Lunch Room, Work Release Room, Ladies' and Mens Toilet Rooms, and Corridor	306'-0"	NA	13	NA
381SG	Control Room	306'-0"	62	88	NA
Diesel Generator Building					
401NZ	Div I, II and III Control Room	261'-0"	NA	NA	9
402SW	Div I D/G Room	261'-0"	NA	NA	7
403SW	Div II D/G Room	261'-0"	NA	NA	7

TABLE 9A.3-18 (Cont'd.)

Instrument Location			Total Number of Instruments**		
Fire Zone*	Room or Area	Elev.	Heat	Ionization	Photo-Electric
Diesel Generator Building (cont'd.)					
404SW	HPCS D/G Room	261'-0"	NA	NA	7
Electrical Tunnels					
301NW	140° Tunnel	215'-0"	NA	23	NA
302NW	35° Tunnel	215'-0"	NA	15	NA
303NW	315° Tunnel	215'-0"	NA	3	NA
304NW	230° Tunnel	215'-0"	NA	12	NA
236NZ	Div I HVAC Room	237'-0"	NA	8	NA
237NZ	Div II HVAC Room	237'-0"	NA	9	NA
Service Water Pump Bays					
806NZ	Div I Pump Bay	244'-0"	NA	6	NA
807NZ	Div II Pump Bay	244'-0"	NA	6	NA
Fire Pump Rooms					
804NW	Diesel Engine Fire Pump Room	261'-0"	NA	NA	8
805NZ	Elect Motor Fire Pump Room	261'-0"	NA	2	NA
Standby Gas Treatment Rooms					
247NZ	Div I GTS Room	261'-0"	NA	7	NA
248NZ	Div II GTS Room	261'-0"	NA	9	NA
Steam Tunnel					
256NZ	Main Steam Tunnel	240'-0"	9	NA	NA
Pipe Tunnels					
361NZ	Pipe Tunnel	245'-0"	NA	10	NA
362NZ	Pipe Tunnel	239'-0"	NA	32	NA

TABLE 9A.3-18 (Cont'd.)

Instrument Location			Total Number of Instruments**		
Fire Zone*	Room or Area	Elev.	Heat	Ionization	Photo-Electric
Pipe Tunnels (cont'd.)					
363NZ	Pipe Tunnel	244'-0"	7	36	NA
Screenwell					
802NZ	Service Intake and Discharge	241'-0"	NA	15	NA

* The first letter in the alphanumeric fire zone designation denotes: S, actuation of fire suppression; N, no actuation of fire suppression; and X, actuation of fire suppression (Halon and CO_2 only) provided one detector is tripped in each of two loops. The second letter denotes: W-water, L-low pressure CO_2; G-Halon, Z-nothing; and F-foam.

** In the case for a fire zone which contains two fire detection loops (denoted by an X in the fire zone designation), the number listed is the total number of detectors in both loops.

Question # SRO 23

Examination Outline Cross-Reference	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295037 2.4.8
	Importance Rating	3.7

Knowledge of how the event-based emergency/abnormal operating procedures are used in conjunction with the symptom-bases EOPs.

Proposed Question:

The plant is operating at 100% power when a transient occurs. The following annunciators are received:

- 603110, RPS A AUTO TRIP
- 603410, RPS B AUTO TRIP
- 603402, RPS B NMS TRIP
- 603102, RPS A NMS TRIP

Immediately following receipt of these annunciators, the following parameters are reported to the CRS:

- Reactor Power 38% and stable
- RPV Level 166 inches, lowering slowly
- Reactor Pressure 1085 psig, rising slowly

No operator actions have been taken to this point. Which one of the following is the **first** action to be directed by the Control Room Supervisor (CRS)?

a. Arm and depress both Manual Scram pushbuttons on either side of 2CEC*PNL603.

b. Manually inhibit ADS and override the opening of CSH*MOV107, HPCS INJECTION VALVE.

c. Place the Reactor Mode switch in SHUTDOWN; verify RPS pilot scram valve solenoid lights are OFF.

d. Initiate RRCS by Arming and depressing DIVISION I AND II CHANNEL A and B MANUAL INITIATION pushbuttons.

Proposed Answer: c.

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Explanation (Justification of Distractors):

a. 2nd step in SOP 101C after placing Mode Switch in SHUTDOWN
b. Done after M/S
d. Done after M/S

Technical Reference(s):

N2-SOP-101C

Proposed references to be provided to applicants during the examination:

EOPs without entry conditions

Learning Objective:	N2-OPS-006-SOP-2-01, EO-1.0, EO-2.0
Question Source:	Bank # Modified Bank # New NEW
Question History:	Previous NRC Exam Previous Test / Quiz
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis 2
10CFR Part 55 Content:	41.10 / 43.5 / 45.13

Comments:

Question # SRO 25

Examination Outline Cross-Reference	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295038 EK2.05
	Importance Rating	4.7

† Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: Site emergency plan.

Proposed Question:

A primary system leak has occurred in the secondary containment, with a subsequent failure of the secondary containment. All attempts to isolate the leak have failed.

One point at the site boundary has a projected dose rate of **1015 mr/hr**. Which one of the following actions is required?

a. Perform an orderly shutdown and monitor the off-site release.

b. Verify a reactor scram occurred and perform an RPV Blowdown.

c. Initiate a reactor scram and open turbine bypass valves in anticipation of RPV Depressurization.

d. Isolate the affected area of the secondary containment and perform an orderly shutdown.

Proposed Answer: b.

Explanation (Justification of Distractors):

This release rate is above the General Emergency level which requires a reactor scram and blowdown in Radioactivity Release Control Procedure.

Technical Reference(s): EPIP-EPP-02
N2-EOP-SC/RR

Proposed references to be provided to applicants during the examination:

EPIP-EPP-02, Attachment 1, and all EOPs with the entry conditions blanked out.

Learning Objective:	03-OPS-006-350-3-01, EO-3 02-OPS-006-344-2-12, EO-2	
Question Source:	Bank # Modified Bank # New	 NEW
Question History:	Previous NRC Exam Previous Test / Quiz	
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 2
10CFR Part 55 Content:	41.7 / 43.5 / 45.8	

Comments:
SRO Only: Site Emergency Director responsibilities and application of the emergency plan.

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Question # SRO 35

Examination Outline Cross-Reference	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295020 AA2.02
	Importance Rating	3.4

Ability to determine and/or interpret the following as they apply to inadvertent containment isolation: Drywell containment temperature.

Proposed Question:

The plant is operating at 60% power when an inadvertent group 8 isolation occurs.

Which one of the following describes how to determine containment temperature is below 150°F?

a. Only method is to monitor SPDS indication.

b. Align the Post Accident Sampling system to the drywell.

c. Monitor back panel recorders, process computer, or SPDS.

d. Only method is to monitor the containment high temperature alarm.

Proposed Answer: c.

Explanation (Justification of Distractors):

a. Recorders and SPDS can also be used.
b. Not permitted by procedure. Has no temperature monitoring ability.
d. Recorders, SPDS, and process computer indications are available.

Technical Reference(s):	N2-OP-82, Rev 04, Section B	

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-001-223-2-06, EO-4a, EO-4b, EO-5	
Question Source:	Bank #	
	Modified Bank #	
	New	New
Question History:	Previous NRC Exam	New
	Previous Test / Quiz	New
Question Cognitive Level:	Memory of Fundamental Knowledge	
	Comprehension or Analysis	1
10CFR Part 55 Content:	55.43.5	
	55.45.13	

Comments:

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Question # SRO 38

Examination Outline Cross-Reference	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295029 EK3.01
	Importance Rating	3.9

Knowledge of the reasons for the following responses as they apply to HIGH SUPPRESSION POOL WATER LEVEL: Emergency depressurization.

Proposed Question:

Which one of the following is the basis for maintaining the suppression pool water level within the safe region of the SRV Tail Pipe Level Limit (N2-EOP-PC, Figure N)?

a. Maintain the capability to vent the suppression chamber.

b. Prevent damage to the ECCS suction strainers or their supports.

c. Prevent damage to the relief valve tailpipes, T-quenchers or their supports.

d. Maintain the suppression chamber-to-drywell vacuum breakers uncovered.

Proposed Answer: c.

Explanation (Justification of Distractors):
a. Not a concern until suppression pool water level is higher.
b. Not a concern associated with suppression pool water level.
d. Not a concern until suppression pool water level is higher.

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Technical Reference(s): NMP2 EOP Basis Document, Section E

Proposed references to be provided to applicants during the examination:

None.

Learning Objective: O2-OPS-006-344-2-04, EO-3

Question Source:	Bank #	
	Modified Bank #	
	New	New
Question History:	Previous NRC Exam	New
	Previous Test / Quiz	New
Question Cognitive Level:	Memory of Fundamental Knowledge	1
	Comprehension or Analysis	
10CFR Part 55 Content:	55.41.5	
	55.43.5	
	55.45.6	

Comments:

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Question # SRO 39

Examination Outline Cross-Reference	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295033 EA1.01
	Importance Rating	4.0

Ability to operate and/or monitor the following as they apply to high secondary containment area radiation levels: Area radiation monitoring system.

Proposed Question:

Which one of the following Radiation Monitoring events requires that you assume the role as Station Emergency Director (SED)?

a. Lowering fuel pool water level causes an automatic containment isolation.

b. A coolant leak at one HCU causes the local ARM to indicate yellow on DRMS.

c. The TIP equipment area ARM goes offscale high when placing a TIP in the transfer cask.

d. During removal, the "hot end" of an LPRM causes the reactor cavity ARM to indicate upscale before it can be lowered.

Proposed Answer: a.

Explanation (Justification of Distractors):

b. This is below the alarm setpoint. Emergency classification is required when value is 100 times the DRMS alarm setpoint.

c. Must be sustained high – returns to normal. Must be an uncontrolled process.

d. Must be sustained high – returns to normal when the LPRM is lowered. Must be an uncontrolled process.

Technical Reference(s): EPIP-EPP-02, Attachment 1, Rev 8

Proposed references to be provided to applicants during the examination:

EPIP-EPP-02, Attachment 1, Rev 8

Learning Objective: 03-OPS-006-350-3-01, EO-2

Question Source:	Bank #	
	Modified Bank #	
	New	New
Question History:	Previous NRC Exam	New
	Previous Test / Quiz	New
Question Cognitive Level:	Memory of Fundamental Knowledge	
	Comprehension or Analysis	2
10CFR Part 55 Content:	55.41.7	
	55.43.4	
	55.43.5	
	55.45.6	

Comments:
SRO Only: Site Emergency Director responsibilities.

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Question # SRO 43

Examination Outline Cross-Reference	Level	SRO
	Tier #	1
	Group #	2
	K/A #	600000 AK2.01
	Importance Rating	2.7

Knowledge of the interrelations between plant fire on site and the following: Sensors/detectors and valves.

Proposed Question:

The plant is at 100% power. It is determined that two (2) ionization detectors in fire zone 333XL, DIV 1 Switchgear Room, are inoperable at 0800 on 12/1/99. One (1) detector is inoperable in each loop of detection.

Which one of the following describes the required actions in accordance with Section 9.A of the USAR?

a. A fire watch must be stationed by 0900 and must be stationed until both detectors are operable.

b. A fire watch must be established by 0900 but may be secured when one of the detectors is operable.

c. If both detectors are **NOT** operable by 0800 on 12/15/99, then a fire watch must be stationed within the next hour.

d. If both detectors are **NOT** operable by 0800 on 12/15/99, then a unit shutdown must be commenced within the next hour.

Proposed Answer: a.

Explanation (Justification of Distractors):

b. This is an action for Function N* fire detection instruments. The inoperable detection is Function X*.

c. Securing the fire watch is not permitted until both detectors are operable.

d. This is an action for Function N* fire detection instruments. The inoperable detection is Function X*.

Technical Reference(s):	USAR, Rev 10, Section 9A.3.6.1 USAR, Rev 10, Table 9A.3-18

Proposed references to be provided to applicants during the examination:

USAR, Rev 10, Section 9A.3.6.1
USAR, Rev 10, Table 9A.3-18

Learning Objective:	O2-OPS-001-286-2-01, EO-7d, EO-11	
Question Source:	Bank # Modified Bank # New	 New
Question History:	Previous NRC Exam Previous Test / Quiz	New New
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	 2
10CFR Part 55 Content:	55.41.5 55.43.1 55.45.5	

Comments:
SRO Only: Use and application of the fire protection system requirements in the USAR for inoperable equipment.

Question # SRO 57

Examination Outline Cross-Reference	Level	SRO
	Tier #	2
	Group #	1
	K/A #	223002 A2.01
	Importance Rating	3.5

Ability to (a) predict the impacts of the following on the primary containment isolation system / nuclear steam supply shut-off; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A.C. electrical distribution failures.

Proposed Question:

The plant is operating at 100% power when Electrical Protection Assembly (EPA) 2VBS*ACB2A trips. The following group isolations are received:

- Groups 2, 3, 4, 5, 8, and 9

Which one of the following describes the required Technical Specification actions in response to Reactor Coolant System leakage detection?

a. Enter T.S. 3.0.3 and start a power reduction within 1 hour.

b. Be in MODE 3 within 12 hours and MODE 4 within 36 hours.

c. Determine drywell leak rate by other means until group 9 is reset.

d. Analyze grab samples of drywell every 12 hours until group 8 is reset.

Proposed Answer: a. There is no T.S. condition for both the particulate and gaseous monitors being inoperable this requires entry into T.S. 3.0.3

Explanation (Justification of Distractors):

b. With both the particulate or gaseous monitor is inoperable, there is no action specified requiring entry into T.S. 3.0.3

c. This action must continue until the group 8 is reset.

d. This is correct if the particulate or gaseous monitor is inoperable, but both are inoperable.

Technical Reference(s): Technical Specification 3.4.3, 3.4.3.1/4.4.3.1

Proposed references to be provided to applicants during the examination:

Technical Specification 3.4.3, 3.4.3.1/4.4.3.1

Learning Objective: O2-OPS-001-233-2-02, EO-11

Question Source:	Bank # Modified Bank # New	New	
Question History:	Previous NRC Exam Previous Test / Quiz	New New	
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis		2
10CFR Part 55 Content:	55.41.5 55.43.2 55.45.6		

Comments: SRO only: Technical Specification application

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REACTOR COOLANT SYSTEM

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITIONS FOR OPERATION

3.4.3.1 The following reactor coolant system leakage detection systems shall be OPERABLE:

a. The primary containment airborne particulate radioactivity monitoring system,

b. The primary containment airborne gaseous radioactivity monitoring system,

c. The drywell floor drain tank fill rate monitoring system, and

d. Drywell equipment drain tank fill rate monitoring system.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the primary containment airborne particulate radioactivity monitoring system or the primary containment airborne gaseous radioactivity monitoring system inoperable, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 12 hours; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the drywell equipment drain tank fill rate monitoring system inoperable, operation may continue for up to 30 days provided that the drywell equipment drain tank fill rate is determined via alternate methods; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With the drywell floor drain tank fill rate monitoring system inoperable, operation may continue for up to 30 days provided that the drywell floor drain tank fill rate is determined via alternate methods; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With both drywell floor drain and the drywell equipment drain tank fill rate monitoring systems inoperable, restore either system to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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REACTOR COOLANT SYSTEM

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

SURVEILLANCE REQUIREMENTS

4.4.3.1 The reactor coolant system leakage detection systems shall be demonstrated OPERABLE by:

a. Primary containment atmosphere particulate and gaseous monitoring systems-performance of a CHANNEL CHECK at least once per 12 hours, a SOURCE CHECK at least once per 31 days, a CHANNEL FUNCTIONAL TEST at least once per 184 days and a CHANNEL CALIBRATION at least once per 18 months.

b. Primary containment sump flow monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION TEST at least once per 18 months.

Question # SRO 58

Examination Outline Cross-Reference	Level	SRO
	Tier #	2
	Group #	1
	K/A #	226001 A1.05
	Importance Rating	3.4

Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: Containment spray mode controls including: system lineup.

Proposed Question:

A steam leak inside the Drywell is in progress. The following conditions exist:

- Drywell Pressure is 2.5 psig
- The "B" Loop of Residual Heat Removal (RHR) is placed in operation
- Drywell Spray Valves RHS*MOV25B and RHS*MOV15B are stroking open

Which one of the following describes the response of the Drywell Spray Valves if drywell pressure lowers to 1.0 psig <u>before</u> the valves are full open?

a. Stroke full open and then close.

b. Stroke full open and remain full open.

c. Stop stroking at an intermediate position.

d. Reverse direction at an intermediate position and close.

Proposed Answer: c.

Explanation (Justification of Distractors):

a. Stop at the current position.
b. Stop at the current position.
d. Stop at the current position.

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Technical Reference(s): USAR Section 7.3, Figure 7.3-6

Proposed references to be provided to applicants during the examination:

None.

Learning Objective: O2-OPS-001-205-2-00, EO-4b, EO-4c, EO-9

Question Source:	Bank #	Q15822
	Modified Bank #	
	New	
Question History:	Previous NRC Exam	July 1996 NRC
	Previous Test / Quiz	Week 16 exam
Question Cognitive Level:	Memory of Fundamental Knowledge	1
	Comprehension or Analysis	
10CFR Part 55 Content:	55.41.5	
	55.43.	
	55.45.5	

Comments: SRO only: This question requires systems knowledge beyond that of an RO to insure that the facilities procedures are adhered to and that the limits in its license and amendments are NOT exceeded.

Question # SRO 59

Examination Outline Cross-Reference	Level	SRO
	Tier #	2
	Group #	1
	K/A #	239002 A4.06
	Importance Rating	3.4

Ability to manually operate and/or monitor in the control room: Reactor water level.

Proposed Question:

During the execution of EOP-C5, Failure to Scram, which one of the following is a concern when Safety Relief Valves (SRVs) are used for pressure control?

a. Inadequate core cooling.

b. Reactor power transients.

c. Loss of preferred injection sources.

d. Inaccurate reactor pressure indication.

Proposed Answer: b.

Explanation (Justification of Distractors):

a. & c. Some injection sources could be affected if the level transient causes a group isolation, but all injection will not be lost.

d. Pressure indication may change, but will be accurate for the reactor pressure being sensed.

Technical Reference(s):	NMP2 EOP Technical Bases, N2-EOP-C5, Step P-5 Stabilize Pressure

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-006-344-2-17, EO-2, EO-3

Question Source:	Bank #	
	Modified Bank #	
	New	New
Question History:	Previous NRC Exam	New
	Previous Test / Quiz	New
Question Cognitive Level:	Memory of Fundamental Knowledge	
	Comprehension or Analysis	2
10CFR Part 55 Content:	55.41.7	
	55.43.5	
	55.45.5, 45.7, 45.8	

Comments: SRO only: CRS (SRO) must understand the benefits and consequences of tasks directed when operating the facility.

Question # SRO 61

Examination Outline Cross-Reference	Level	SRO
	Tier #	2
	Group #	1
	K/A #	259002 2.1.6
	Importance Rating	4.3

Ability to supervise and assume a management role during plant transients and upset conditions.

Proposed Question:

Following a LOCA reactor water level CANNOT be maintained above the top of active fuel (TAF) using preferred injection systems. Actions are in progress to lineup Alternate Injection Systems for injection. Conditions are:

- RPV Pressure 22 psig
- Drywell Pressure 12 psig
- Drywell Temperature 278°F
- Suppression Chamber Pressure 9 psig
- Suppression Pool Level 204 feet
- Suppression Pool Temperature 131°F

Which one of the following describes the required action?

a. Enter the RPV Flooding EOP.

b. Enter the Steam Cooling EOP.

c. Enter the RPV Blowdown EOP.

d. Continue in the RPV Control EOP.

Proposed Answer: a.

Explanation (Justification of Distractors):
b. Not required for these conditions.
c. The blowdown for RPV flooding is executed in the RPV flooding EOP.
d. RPV level instrumentation is lost. RPV flooding is entered and RPV Control is exited.

Technical Reference(s):	N2-EOP-RPV, Rev 8, Override Statement NMP2 EOP Technical Bases, Introduction, Step Execution

Proposed references to be provided to applicants during the examination:

RPV Control EOP.

Learning Objective:	O2-OPS-006-344-2-01, EO-1, EO-2

Question Source:	Bank #		
	Modified Bank #		
	New	New	
Question History:	Previous NRC Exam	New	
	Previous Test / Quiz	New	
Question Cognitive Level:	Memory of Fundamental Knowledge		
	Comprehension or Analysis		3
10CFR Part 55 Content:	55.43.5		
	55.45.12		
	55.45.13		

Comments: SRO only question to determine the ability to use the RPV Saturation Curve while in the EOPS and that when the conditions of an override statement are met, the specified conditions must be executed.

Question # SRO 66

Examination Outline Cross-Reference	Level	SRO
	Tier #	2
	Group #	1
	K/A #	290001 2.4.16
	Importance Rating	4.0

Knowledge of EOP implementation hierarchy and coordination with other support procedures.

Proposed Question:

While executing N2-EOP-RPV, RPV Control, the following annunciators are received:

- 601341, RCIC EQUIP ROOM TEMPERATURE HI-HI
- 601531, ADS A SYSTEM LOGIC TIMER INTIATED
- 601728, HPCS PUMP 1 AUTO TRIP/OUT OF SER

Which one of the following describes the importance of these alarms while executing RPV Control?

a. Re-enter RPV Control at the beginning of the procedure.

b. Enter N2-EOP-SC, Secondary Containment Control, and execute it concurrently.

c. Anticipate RPV Blowdown, rapidly depressurize the RPV using Main Turbine Bypass Valves.

d. When it is known that NO EOP "override" or "wait" steps are challenged, then take the actions of the ARP.

Proposed Answer: b.

Explanation (Justification of Distractors):

a. There is not an entry condition for RPV Control and this re-entry is not required.

c. These annunciators do NOT require RPV depressurization so alternate depressurization is not required.

d. When an entry condition exists, the applicable EOP is entered and executed concurrently.

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Technical Reference(s): NMP2 EOP Technical Bases, Introduction

Proposed references to be provided to applicants during the examination:

None.

Learning Objective: O2-OPS-006-344-2-22, EO-1

Question Source:	Bank #		
	Modified Bank #		
	New	New	
Question History:	Previous NRC Exam	New	
	Previous Test / Quiz	New	
Question Cognitive Level:	Memory of Fundamental Knowledge		
	Comprehension or Analysis		2
10CFR Part 55 Content:	55.41.5		
	55.43.5		
	55.45.3		

Comments: SRO only: Ability to direct actions within the EOPs

Question # SRO 69

Examination Outline Cross-Reference	Level	SRO
	Tier #	2
	Group #	2
	K/A #	204000 2.4.49
	Importance Rating	3.8

Ability to interpret control room indications to verify status and operation of system, and understand how operator actions and directives effect plant and system condition.

Proposed Question:

During a reactor startup the Reactor Water Cleanup System (WCS) is operating in Flow Rejection (Blowdown Mode). The following WCS lineup exists:

- Reactor Pressure is 890 psig
- WCS-P1A and WCS-P1B are in service
- Four (4) WCS Filter Demineralizers are in service.
- 2WCS-MOV128, REJECT RESTRICTING ORIFICE BYPASS VLV is Closed
- 2WCS-FV135, REJECT FLOW CONTROL MANUAL CONTROL is 100% Open
- WCS-MOV200, CLEANUP RETURN ISOL VLV THROTTLE is Open
- Reject Flowrate is 90 gpm
- Non-Regenerative Heat Exchanger Outlet Temperature is 128°F

Which one of the following will occur if the Control Switch for 2WCS-MOV128 is placed and held in the open position?

a. Pressure will rise downstream of 2WCS-FV135 initiating an isolation of 2WCS-FV135 and Low Flow trip of both WCS Pumps.

b. A rising Non-Regenerative Heat Exchanger outlet temperature isolates the Cleanup Outboard Suction Isolation Valve, 2WCS-MOV112, and this causes a trip of both WCS Pumps.

c. A High Delta Flow condition causes an isolation of the Cleanup Outboard Suction Isolation Valve, 2WCS-MOV112, and a subsequent low flow trip of the WCS Pumps in about 14 minutes.

d. A rapid pressure reduction upstream of 2WCS-FV135, initiates an isolation of Cleanup Outboard and Inboard Suction Isolation Valves, 2WCS-MOV112, and 2WCS-MOV102, trip of both WCS Pumps on loss of suction flow path.

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Proposed Answer: b.

Explanation (Justification of Distractors):

a. Pressure may rise but not sufficient to cause an isolation of FV135, and FV125 isolating will NOT trip the WCS Pumps.
c. A high delta flow condition should NOT exist and if one did, the suction isolation valve would trip the pumps NOT low flow.
d. Pressure will NOT lower above FV135.

Technical Reference(s): N2-OP-37, S ect. H.3.7

Proposed references to be provided to applicants during the examination:

None.

Learning Objective: O2-OPS-001-204-2-01, EO-6

Question Source:	Bank #	
	Modified Bank #	
	New	NEW
Question History:	Previous NRC Exam	NEW
	Previous Test / Quiz	NEW
Question Cognitive Level:	Memory of Fundamental Knowledge	
	Comprehension or Analysis	2
10CFR Part 55 Content:	55.41.10	
	55.43.2	
	55.45.6	

Comments: SRO only: Level of knowledge above RO required for control of plant evolutions.

S 20-2

Question # SRO 70

Examination Outline Cross-Reference	Level	SRO
	Tier #	2
	Group #	2
	K/A #	205000 A4.07
	Importance Rating	3.7

Ability to manually operate and/or monitor in the control room: Reactor temperatures (moderator, vessel, flange).

Proposed Question:

The unit is in MODE 5 for a refueling outage. The reactor cavity is flooded and the fuel pool gates are removed.

- Shutdown cooling is lost and CANNOT be restarted
- A reactor recirculation pump CANNOT be started
- Reactor water cleanup is operating

Within one (1) hour, the Alternate Decay Heat (ADH) system is aligned and placed in operation. Which one of the following describes how to determine reactor coolant temperature?

a. RHR HX 1A Inlet temperature

b. Recirc Loop B suction temperature

c. RPV bottom head drain temperature

d. Spent Fuel Pool Cooling HX inlet temperature

Proposed Answer: c.

Explanation (Justification of Distractors):

a. No flow to provide accurate indication.
b. No flow to provide accurate indication.
d. Used when performing alternate SDC using SFC

Technical Reference(s): N2-OP-115, Rev 02, Section F.1

Proposed references to be provided to applicants during the examination:

None.

Learning Objective: O2-OPS-001-231-2-00, EP-4a, EO-4b, EO-5

Question Source:	Bank #	
	Modified Bank #	
	New	New
Question History:	Previous NRC Exam	New
	Previous Test / Quiz	New
Question Cognitive Level:	Memory of Fundamental Knowledge	2
	Comprehension or Analysis	
10CFR Part 55 Content:	55.41.7	
	55.43.5	
	55.45.5, 45.7	

Comments: SRO only: The occurrence or absence of stratification in the reactor vessel must be assessed by the SRO based on current system alignment and time.

S21-2

Question # SRO 78

Examination Outline Cross-Reference	Level	SRO
	Tier #	2
	Group #	2
	K/A #	290003 K1.04
	Importance Rating	3.3

Knowledge of the physical connections and/or cause-effect relationships between control room HVAC and the following: Nuclear steam supply shutoff system (NSSSS/PCIS): Plant-specific.

Proposed Question:

One (1) hour following a large break loss of coolant accident, the Control Room "E" operator reports that NO manual alignment changes have been made to the Control Building Ventilation System. Which one of the following describes the concern with the Control Room environment?

a. Pressure will become negative.

b. Temperature will rise above 90°F.

c. Humidity will be higher than expected.

d. Dose rate will be higher than expected.

Proposed Answer: d.

Explanation (Justification of Distractors):

a. Pressure will be positive because both special filter trains automatically started.
b. Temperature will be about room temperature because the HVC chillers continue to run.
c. Humidity will be lower because of the heaters that are operating in the inlet to the special filter trains.

Technical Reference(s): N2-OP-53A, Rev 08, Section D.19

Proposed references to be provided to applicants during the examination:

None.

Learning Objective: O2-OPS-001-288-2-02, EO-6

Question Source:	Bank #	
	Modified Bank #	
	New	New
Question History:	Previous NRC Exam	New
	Previous Test / Quiz	New
Question Cognitive Level:	Memory of Fundamental Knowledge	1
	Comprehension or Analysis	
10CFR Part 55 Content:	55.43.4	

Comments: SRO only: Understand the benefits and consequences of operator actions and failure to perform actions – plant is in a condition challenging the design bases.

S22-2

Question # SRO 81

Examination Outline Cross-Reference	Level	SRO
	Tier #	-
	Group #	-
	K/A #	Generic 2.1.4
	Importance Rating	3.3

Knowledge of system status criteria which require the notification of plant personnel.

Proposed Question:

In accordance with the Operations Manual, which one of the following Spent Fuel Pool Cooling and Cleanup (SFC) related events requires notifying the General Supervisor Operations (GSO) when operating in MODE 1?

a. After a trip of 2SFC*P1A, **NEITHER** SFC pump can be started.

b. A fire alarm is activated in the vicinity of a SFC pump due to a faulty detector.

c. 2SFC*P1A is **NOT** returned to service within the scheduled time after maintenance.

d. The SFC lineup will be changed from the "Cooling Only" Mode to "Filter/Demin Subsystem" Mode.

Proposed Answer: a.

Explanation (Justification of Distractors):
Unexpected entry into the SOPs requires GSO notification. N2-SOP-38 is entered if neither SFC pump is operating and cannot be started.

b. If the emergency plan is entered due to a fire, the GSO is notified. GSO is not required to be notified for false fire alarms.
c. There is not requirement to notify the GSO.
d. This is a lineup approved in N2-OP-38 and does not require notification prior to performance.

Technical Reference(s): Operations Manual, Section 3.6

Proposed references to be provided to applicants during the examination:

None.

Learning Objective: O2-OPS-006-343-3-01, EO-2, EO-5

Question Source:	Bank #		
	Modified Bank #		
	New	New	
Question History:	Previous NRC Exam	New	
	Previous Test / Quiz	New	
Question Cognitive Level:	Memory of Fundamental Knowledge		1
	Comprehension or Analysis		
10CFR Part 55 Content:	55.41.10		
	55.43.2		

Comments: SRO only: SSS/CRS roles and responsibilities during plant operation.

S23-2

Question # SRO 84

Examination Outline Cross-Reference	Level	SRO
	Tier #	-
	Group #	-
	K/A #	Generic 2.1.4
	Importance Rating	3.4
Knowledge of shift staffing requirements.		

Proposed Question:

The unit is in MODE 3. The CRS is designated to assume the role of the STA if the Site Emergency Plan is activated.

In the absence of the SSS from the control room, which one of the following describes who may be designated to assume the control room command function?

a. The on-shift CRS if the absence will be less than 10 minutes.

b. Any individual with an active SRO license including the on-shift CRS.

c. Any individual with an active SRO license other than the on-shift CRS.

d. Only an individual with an active SRO license who is qualified as SSS.

Proposed Answer: c.

Explanation (Justification of Distractors):

a. The CRS can only fill the control room command function when not in the STA function.
b. This would include the CRS who is not permitted to fulfill the function because of being designated to fill the STA function.
d. There is no requirement to be qualified as SSS.

S24-1

Technical Reference(s): Technical Specification Table 6.2.2-1, Note (d)

Proposed references to be provided to applicants during the examination:

None.

Learning Objective: O2-OPS-006-343-3-01, EO-4, EO-5

Question Source:	Bank #	
	Modified Bank #	
	New	New
Question History:	Previous NRC Exam	New
	Previous Test / Quiz	New
Question Cognitive Level:	Memory of Fundamental Knowledge	1
	Comprehension or Analysis	
10CFR Part 55 Content:	55.41.10	
	55.43.1	
	55.43.2	

Comments: SRO only: Command and control function.

Question # SRO 88

Examination Outline Cross-Reference	Level	SRO
	Tier #	-
	Group #	-
	K/A #	Generic 2.1.12
	Importance Rating	4.0
Ability to apply Technical Specifications for a system.		

Proposed Question:

The unit is operating at 80% power. Which one of the following limiting conditions from the Service Water (SW) Technical Specifications has the most restrictive requirement on continued plant operation?

At 0800 on 12/6/99...

a. SW Loop "A" is declared inoperable.

b. SW Pump "A" is declared inoperable and SW Pump "B" is declared inoperable 24 hours later.

c. SW supply header discharge temperature is discovered at 83°F and is lowered to 80°F within 4 hours.

d. Intake tunnel water temperature is discovered at 38°F. Each intake structure has seven (7) operable heaters that are in operation.

Proposed Answer: d. Must be in HOT SHUTDOWN within 12 hours.

Explanation (Justification of Distractors):

a. 72 hours are permitted to restore to OPERABLE prior to a unit shutdown.
b. 8 days are permitted to restore to OPERABLE prior to a unit shutdown.
c. The LCO statement continued to be met because the temperature was lowered within 8 hours.

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Technical Reference(s): Technical Specification 3.7.1.1

Proposed references to be provided to applicants during the examination:

Technical Specifications, Section 3.7.1.1

Learning Objective:	O2-OPS-001-276-2-00, # 11		
Question Source:	Bank #		
	Modified Bank #		
	New	New	
Question History:	Previous NRC Exam	New	
	Previous Test / Quiz	New	
Question Cognitive Level:	Memory of Fundamental Knowledge		
	Comprehension or Analysis		2
10CFR Part 55 Content:	55.43.2		
	55.43.5		
	55.45.3		

Comments: SRO only: Technical Specification application

Check to ensure that TS and OP have not changed.

525-2

3/4.7 PLANT SYSTEMS

3/4.7.1 PLANT SERVICE WATER SYSTEM

PLANT SERVICE WATER SYSTEM - OPERATING

LIMITING CONDITIONS FOR OPERATION

3.7.1.1 Two independent plant service water system loops shall be OPERABLE with one loop in operation. Each loop shall be comprised of:

a. Two plant service water pumps capable of taking suction from Lake Ontario and transferring the water to the associated safety related equipment.

b. Service water supply header discharge water temperature of 81°F or less.

The intake deicing heater system shall be OPERABLE and in operation when intake tunnel water temperature is less than 39°F; Division I shall have 7 heaters in operation in each intake structure and Division II shall have 7 heaters in operation in each intake structure.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3

ACTION:

a. With one less than the required number of OPERABLE plant service water pumps in one loop, restore the inoperable pump to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one less than the required number of OPERABLE plant service water pumps in each loop, restore at least one inoperable pump to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. Within two less than the required number of OPERABLE plant service water pumps in one loop or with one plant service water loop otherwise inoperable, restore at least one pump to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With two less than the required number of OPERABLE plant service water pumps in one loop and one less than the required number of plant service water pumps in the other loop, restore at least one of the two inoperable pumps in the same loop to OPERABLE status within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. With two plant service water system loops OPERABLE and the service water supply header discharge water temperature continuously exceeding 81°F for any 8 hour period, within one hour initiate action to be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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LIMITING CONDITIONS FOR OPERATION

3.7.1.1 (Continued)

ACTION:

f. With less than the required Division I and Division II heaters OPERABLE within one hour initiate action to be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.1.1 The plant service water system shall be demonstrated OPERABLE.

a. By verifying the plant service water supply header discharge water temperature to be less than or equal to 81°F.

1. At least once per 24 hours, and

2. At least once per 4 hours when the last recorded water temperature is greater than or equal to 75°F, and

3. At least once per 2 hours when the last recorded water temperature is greater than or equal to 79°F.

b. At least once per 12 hours by verifying the water level at the service water pump intake is greater than or equal to elevation 233.1 feet.

c. At least once per 31 days by verifying that each valve - manual, power-operated, or automatic, servicing safety-related equipment that is not locked, sealed or otherwise secured in position - is in its correct position.

d. At least once per 18 months during shutdown, by verifying:

1. After a simulated test signal, each automatic valve servicing nonsafety-related equipment actuates to its isolation position.

2. After a simulated test signal, each service water system cross connect and pump discharge valve actuates automatically to its isolation position.

3. For each service water pump, after a simulated test signal, the pump starts automatically and the associated pump discharge valve opens automatically, in order to supply flow to the system safety-related components.

SURVEILLANCE REQUIREMENTS

4.7.1.1.1 (Continued)

e. At least once per 18 months:

1. Perform a LOGIC SYSTEM FUNCTIONAL TEST of the service water pump starting logic.

2. Verify each pump runs and maintains service water pump discharge pressure equal to or greater than 80 psig with a pump flow equal to or greater than 6500 gpm.

4.7.1.1.2 The Intake Deicing Heater System shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying the intake tunnel water temperature is greater than or equal to 39°F, or

b. At least once per 7 days by verifying that the current of the heater feeder cables at the motor control centers is 10 amps* or more (total for three phases) at ≥ 518 volts per divisional heater in each intake structure.

c. At least once per 18 months by verifying the resistance is ≥ 28 ohms for each feeder cable and associated heater elements in the intake deicing heater systems.

* For 7 heater elements in operation.

Question # SRO 89

Examination Outline Cross-Reference	Level	SRO
	Tier #	Generic
	Group #	-
	K/A #	2.2.26
	Importance Rating	3.7
Knowledge of refueling administrative requirements.		

Proposed Question:

Preparations have been made to start a full core offload in the next hour (at 0800 on 12/3/99). All requirements of N2-FHP-13.1, Complete Core Offload, are satisfied with the following exception:

- SRM B was declared inoperable at 0700 on 12/3/99.

Which one of the following describes when the core offload can be started and the restrictions that apply?

a. The core offload can be started as scheduled, but must be stopped at the completion of **sequence step 33**.

b. The core offload can be started as scheduled, but must be stopped at the completion of **sequence step 64**.

c. The core offload CANNOT be started until **after** SRM B operability is demonstrated by the SRM Channel Functional Test.

d. The core offload CANNOT be started until **after** SRM B is operable and all "prior to fuel movement" checks are performed again.

Proposed Answer: a.

Explanation (Justification of Distractors):

b. Proceeding beyond step 33 with SRM B inop is a procedure/Tech Spec violation because steps 34 –37 are in the same quadrant as SRM B.
c. The core offload can be started provided it is stopped prior to completing sequence step 34.
d. The core offload can be started provided it is stopped prior to completing sequence step 34.

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Technical Reference(s): N2-FHP-13.1, Rev 04, 4.2.11, 5.16.4, and 5.16.5
Tech Spec 3.9.2

Attach a copy of N2-FHP-13.1

Proposed references to be provided to applicants during the examination:

N2-FHP-13.1
Tech Spec Section 3.9

Learning Objective: O2-OPS-001-234-2-01, EO-7a, EO-8, EO-11

Question Source: Bank #
Modified Bank #
New New

Question History: Previous NRC Exam New
Previous Test / Quiz New

Question Cognitive Level: Memory of Fundamental Knowledge
Comprehension or Analysis 2

10CFR Part 55 Content: 55.43.2
55.43.5

Comments: SRO only: Technical specification application. Control of plant evolutions.

S 26-2

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION UNIT 2

FUEL HANDLING PROCEDURE

N2-FHP-13.1

REVISION 04

COMPLETE CORE OFFLOAD

TECHNICAL SPECIFICATION REQUIRED

Approved by: R. G. Smith — Manager Operations - Unit 2 — Date 9/13/96

SATELLITE MASTER COPY

****

PROCEDURE IS A SPECIAL EVOLUTION
REQUIREMENTS OF GAP-SAT-03 APPLICABLE

****

PERIODIC REVIEW, 09/17/98, NO CHANGE

Effective Date: 09/20/96

NOT TO BE USED AFTER SEPTEMBER 2000
SUBJECT TO PERIODIC REVIEW

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TABLE OF CONTENTS

1.0 PURPOSE

To perform Core Alterations in order to completely offload the core during a refueling outage.

1.1 Discussion

1.1.1 This procedure is used whenever fuel is moved for the purpose of completely offloading the core; therefore, there is no set frequency for this procedure.

1.1.2 This procedure will provide general guidelines to ensure that all prerequisites for commencing Core Alterations are met.

1.1.3 This procedure provides guidelines to ensure the continued operability of the Source Range Monitors and for bypassing the SRM Downscale Rod Block which will be generated as fuel is removed from around the SRM detectors.

1.1.4 This procedure will provide criteria for halting fuel movement and the steps necessary to be performed prior to the resumption of Core Alterations (see Step 4.1.3, Suspension of Core Alterations).

1.1.5 The specific order and locations for the fuel moves will be provided by Reactor Engineering on Fuel Movement Instructions which will be attached to, and become Appendix 1 to, this procedure. The core Offload begins on a peripheral cell with a subsequent spiral inward to a cell adjacent to an SRM. The four bundles next to each SRM will be left in the core to maintain required SRM count rate. They will be removed after all the remaining fuel is Offloaded.

1.1.6 This procedure provides guidelines to defeat SRM period alarms AFTER all fuel has been removed from the reactor vessel.

2.0 REFERENCES AND COMMITMENTS

2.1 Technical Specifications

2.1.1 Reactivity Control Systems

- Sections 3/4.1.1 Shutdown Margin

2.1.2 Reactor Protection System Instrumentation

- Tables 3.3.1-1.1.a/ 4.3.1.1-1.1.a IRM Neutron Flux - High

- Tables 3.3.1-1.1.b/ 4.3.1.1-1.1.b IRM Inoperative

2.1.2 (Cont)

- Tables 3.3.1-1.11/ 4.3.1.1-1.11 Reactor Mode Switch - Shutdown
- Tables 3.3.1-1.12/ 4.3.1.1-1.12 Manual Scram

2.1.3 <u>Isolation Activation Instrumentation</u>

- Tables 3.3.2-1.1.a.2/ 4.3.2.1-1.1.a.2 Reactor Vessel Water Level - Level 2
- Tables 3.3.2-1.3.a/ 4.3.2.1-1.3.a Reactor Building Above the Refuel Floor Exhaust Radiation - High
- Tables 3.3.2-1.3.b/ 4.3.2.1-1.3.b Reactor Building Below the Refuel Floor Exhaust Radiation - High

2.1.4 <u>ECCS Actuation Instrumentation</u>

- Tables 3.3.3-1.D.1/ 4.3.3.1-1.D.1 4.16-kV Emergency Bus Undervoltage-Loss of Voltage (Divisions I & II)
- Tables 3.3.3-1.D.2/ 4.3.3.1-1.D.2 4.16-kV Emergency Bus Undervoltage-Degraded Voltage (Divisions I & II)

2.1.5 <u>Control Rod Block Instrumentation</u>

- Tables 3/4.3.6-1.2 SRM
- Tables 3/4.3.6-1.3 IRM
- Tables 3/4.3.6-1.6.b Reactor Mode Switch (Refuel Mode)

2.1.6 <u>Radiation Monitoring Instrumentation</u>

- Tables 3/4.3.7.1-1.1 Main Control Room Ventilation Radiation Monitors

2.1.7 <u>ECCS - Shutdown</u>

- Sections 3/4.5.2 ECCS Shutdown

2.1.8 Secondary Containment

• Sections 3/4.6.5.1	Secondary Containment Integrity
• Sections 3/4.6.1.2	Primary Containment Leakage N2-ODP-OPS-0113
• Sections 3/4.6.5.2	Secondary Containment Automatic Isolation Dampers
• Sections 3/4.6.5.3	Standby Gas Treatment System

2.1.9 Plant Systems

• Sections 3/4.7.1.2	Plant Service Water System - Shutdown
• Sections 3/4.7.3	Control Room Outdoor Air Special Filter Train System

2.1.10 Electrical Power Systems

• Sections 3/4.8.1.2	AC Sources - Shutdown
• Sections 3/4.8.2.2	DC Sources - Shutdown
• Sections 3/4.8.3.2	On Site Power Distribution Systems - Shutdown

2.1.11 Refueling

• Sections 3/4.9	Refueling Operations

2.1.12 Administrative Controls

• Section 6.2.2.f	Organization - Unit Staff

2.2 Licensee Documentation

- Nine Mile Point 2 Operating License, NPF-69, Docket No. 50-410
- Nine Mile Point 2 USAR Section 9.1, Refueling Equipment
- Nine Mile Point 2 USAR, Chapter 15E, Complete Core Offload/Reload Procedure Guidelines

2.3 Policies, Programs, and Procedures

- NDD-NFM, Nuclear Fuel Management
- GAP-HSC-02, System Cleanness Controls
- GAP-HSC-03, Foreign Material Exclusion Control
- GAP-FPP-02, Control of Hot Work
- GAP-RPP-02, Radiation Work Permits

2.3 (Cont)

- GAP-SAT-03, Control of Special Evolutions
- NIP-SNM-01, Special Nuclear Material Control
- N2-ODP-NFM-0101, Refueling Operations
- N2-FHP-3, Refueling Manual
- N2-FHP-10, LPRM Removal and Installations
- N2-FHP-21, Control Rod Uncoupling, Removal, and Installation
- N2-OP-38, Spent Fuel Pool Cooling and Cleanup System
- N2-OP-39, Fuel Handling and Reactor Service Equipment
- N2-OP-84, Reactor Building Crane
- N2-OP-92, Neutron Monitoring
- N2-SOP-39, Refuel Floor Events
- N2-OSP-LOG-S004/5, Shift Checks - Mode 4 And Mode 5
- N2-OSP-LOG-M001, Monthly Checks
- N2-PM-S001, Refueling Platform And Grapple Inspection
- N2-OPS-FNR-@001, Refueling Platform Cutoff And Interlock Operability Test
- N2-OSP-RMC-W@002, Reactor Mode Switch Functional Test Of Refuel Interlocks
- N2-OSP-NMS-@002, Source Range Monitor Check During Core Offload/Reload
- N2-ISP-NMS-W@008, Source Range Monitor And Rod Block Trip Channel Functional Test
- N2-MSP-MHR-W001, Reactor Building Polar Crane Interlock and Travel Restriction Test
- N2-ISP-NMS-W@009, Intermediate Range Monitor Channel Functional Test
- S-RAP-RPP-0801, High Radiation Area Monitoring And Control
- N2-REP-8, Core Post-Alteration Inspection And Verification

4.1.5 Fuel Loading Error (FLE)

Fuel Loading Error - The placement of a fuel bundle in the core in a location other than that specified by the Fuel Movement Instructions. This includes partial insertion of a fuel bundle into the reactor core.

4.1.6 SFP-Special Setdown Area

Spent Fuel Pool locations in rows 1A or 1B immediately adjacent to the fuel transfer canal.

4.2 Fuel Movement

4.2.1 Administrative controls in N2-ODP-NFM-0101, Refueling Operations shall be followed.

4.2.2 All fuel movements shall be in accordance with the Fuel Movement Instructions which shall be approved by the Reactor Engineering Supervisor or designee. These Instructions will be independently prepared and verified by other members of the Reactor Engineering Department.

4.2.3 The Fuel Movement Instructions should be prepared following the cell unloading sequence specified in Attachment 3, Core Offload Sequence Map. The four bundles next to each SRM should be left in the core and removed AFTER all the remaining fuel has been Offloaded.

4.2.4 The detailed fuel moves shall be written such that no more than three fuel assemblies shall occupy a control cell, unless they represent the initial or final configuration, or a configuration less reactive than the final configuration as indicated on the Fuel Movement Instructions.

(C1) 4.2.5 If visual contact is lost, immediately stop all movement of fuel or equipment within the Reactor Vessel area. Do not resume movement until visual contact is restored.

4.2.6 Fuel assemblies shall be stored in designated storage cells only.

4.2.7 Fuel assemblies shall not be placed in aisles or moved through aisles adjacent to and at the same level as the storage racks.

4.2.8 Movement of leads in excess of 1000 lbs shall be made in accordance with T.S. 3/4.9.7.

4.2.9 Per NMP2 Operating License, WHEN NOT in the reactor vessel, no more than three fuel assemblies shall be allowed outside of their shipping containers or storage racks in the New Fuel Vault or Spent Fuel Storage Facility. These three assemblies shall maintain a minimum edge-to-edge spacing of twelve (12) inches from the shipping container array and approved storage rack locations.

4.0 PRECAUTIONS AND LIMITATIONS

4.1 Terms and Definitions

The following definitions apply to the performance of this procedure:

4.1.1 Core Alterations

CORE ALTERATION shall be the movement of any fuel, 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.

Suspension of CORE ALTERATIONS shall not preclude completion of movement to a safe position.

4.1.2 Flux Trap

A Flux Trap is the condition where an empty fuel cell, or group of empty cells, is completely surrounded by fuel.

4.1.3 Suspension Of Core Alterations

NOTE: Routine activities which may halt refueling operations such as surveillance tests, preventive maintenance, shift turnover, or break, are not considered a Suspension Of Core Alterations.

Suspension Of Core Alterations shall be the termination of Core Alterations as a result of:

a. Action Statements in Technical Specification Limiting Conditions of Operation which state: Suspend Core Alterations

or

b. Refueling equipment malfunctions which require maintenance action and are expected to take longer than eight hours

4.1.4 Fuel Movement Discrepancy

A Fuel Bundle/Blade Guide with a location/orientation different than that specified by the Fuel Movement Instructions. This includes any upward motion of a Fuel Bundle/Blade Guide following Grapple engagement out of the sequence specified in the Fuel Movement Instructions.

4.1.5 Fuel Loading Error (FLE)

Fuel Loading Error - The placement of a fuel bundle in the core in a location other than that specified by the Fuel Movement Instructions. This includes partial insertion of a fuel bundle into the reactor core.

4.1.6 SFP-Special Setdown Area

Spent Fuel Pool locations in rows 1A or 1B immediately adjacent to the fuel transfer canal.

4.2 Fuel Movement

4.2.1 Administrative controls in N2-ODP-NFM-0101, Refueling Operations shall be followed.

4.2.2 All fuel movements shall be in accordance with the Fuel Movement Instructions which shall be approved by the Reactor Engineering Supervisor or designee. These Instructions will be independently prepared and verified by other members of the Reactor Engineering Department.

4.2.3 The Fuel Movement Instructions should be prepared following the cell unloading sequence specified in Attachment 3, Core Offload Sequence Map. The four bundles next to each SRM should be left in the core and removed AFTER all the remaining fuel has been Offloaded.

4.2.4 The detailed fuel moves shall be written such that no more than three fuel assemblies shall occupy a control cell, unless they represent the initial or final configuration, or a configuration less reactive than the final configuration as indicated on the Fuel Movement Instructions.

(C1) 4.2.5 If visual contact is lost, immediately stop all movement of fuel or equipment within the Reactor Vessel area. Do not resume movement until visual contact is restored.

4.2.6 Fuel assemblies shall be stored in designated storage cells only.

4.2.7 Fuel assemblies shall not be placed in aisles or moved through aisles adjacent to and at the same level as the storage racks.

4.2.8 Movement of leads in excess of 1000 lbs shall be made in accordance with T.S. 3/4.9.7.

4.2.9 Per NMP2 Operating License, WHEN NOT in the reactor vessel, no more than three fuel assemblies shall be allowed outside of their shipping containers or storage racks in the New Fuel Vault or Spent Fuel Storage Facility. These three assemblies shall maintain a minimum edge-to-edge spacing of twelve (12) inches from the shipping container array and approved storage rack locations.

4.2.10 Fuel movement shall be immediately stopped and the SSS notified if any of the conditions in Attachment 5 are encountered. In addition to stopping fuel movement, the SSS shall determine if the condition constitutes a Suspension of Core Alterations.

4.2.11 If for any reason a single cell or multiple cell locations cannot be unloaded (i.e. Inoperable SRM), fuel offload may continue provided the following additional requirements are strictly adhered to:

- The spiral offload pattern of Attachment 3 is followed in the region being offloaded.
- Proposed movement does not result in the removal of imbedded fuel cells (i.e. creation of flux traps).
- T.S. 3/4.9.2 is met.
- Proposed movement shall be reviewed by the Reactor Engineering Supervisor, or designated qualified alternate, and the SSS prior to implementation.
- Proposed movement shall be documented in the Remarks Section of this procedure with SSS and Reactor Engineering Supervisor, or designated qualified alternate, signatures of concurrence.

4.2.12 If required, changes to the Fuel Movement instructions shall be marked on the Control Room copy of the instructions and shall be initialed and dated by the SSS and the Reactor Engineering Supervisor, or designated qualified alternate. The Refuel Floor working copy shall also be updated.

4.2.13 Following a Suspension Of Core Alterations, Attachment 2, Checklist For Resuming Core Offload, SHALL be completed prior to the resumption of fuel movement.

4.2.14 An Inoperable SRM may be bypassed (Joystick at PNL-603) AND fuel movement may continue provided the requirements of Precaution 4.2.11 and 4.2.12 are met.

4.2.15 Fuel movement shall be stopped in the event of a Fuel Movement Discrepancy (FMD). Fuel movement may resume once Attachment 6 and 7 are complete.

4.2.16 Fuel Movement into Spent Fuel Pool is restricted until after 96 hours after Reactor Shutdown (all rods inserted).

(C9) 4.2.17 Single Blade Guide(s) must be placed in the core with the spacer buttons oriented towards the center of the cell. This will prevent control rod interference with fuel bundle loading in the adjacent bundle location.

4.2.18 Extra caution should be exercised when moving a fuel bundle adjacent to an SRM or IRM dry tube to prevent damage to the dry tube.

4.5.12 No control rod withdrawal will be allowed during fuel movement. Fuel movement shall be stopped for the performance of N2-OSP-RMC-W@002, as required.

4.5.13 Replacement of a bent channel fastener shall be performed per N2-FHP-16. Required changes to Fuel Movement Instruction shall be made in accordance with Precautions 4.2.11 and 4.2.12.

(C5) 4.5.14 Core offload is considered a special evolution and, as such, is controlled by GAP-SAT-03.

4.5.15 The Fuel Handler Spotter shall sign off a working copy of the Fuel Movement Instructions on the refuel platform to document completion of fuel or blade guide movement.

4.6 <u>Spent Fuel Pool Cooling System (SFC) Requirements</u>

(C7)
4.6.1 For a normal full core offload the SFC system must comply with single failure criteria which requires a Safety Related (SR) system with SR power capable of performing it's SR function. The system must be able to withstand a single active failure assuming no off-site power. Thus both loops of SFC are required to be available. Full core offload can not begin unless Engineering has performed cycle specific heat load analysis that demonstrate SFC single failure conformance.

4.6.2 Prior to the performance of this procedure, cycle specific heat load calculations must be performed by Engineering to determine applicable restrictions for Spent Fuel Pool Cooling (SFC) to ensure compliance with SFC Design Basis in USAR Section 9.1.3.
This would include limits on decay time, bundle offload rates and Service water temperatures. This restrictions shall be incorporated in a N2-OSP-LOG-@001 prior to fuel offload.

4.6.3 During a Divisional Bus and/or Divisional Service Water outage, the SFC Single Failure requirement is relaxed per GL 91-18. The requirement is to have 2 100% decay heat removal systems for pool cooling available. One of which will be an SFC loop with Safety Related cooling water available. The normal lineup will be one ADH train operating with SFC in backup or viceversa, or both operating.

4.6.4 Isolation of the Spent Fuel Pool from the Reactor Cavity is restricted until after cycle specific calculations are performed by Engineering that demonstrate single failure conformance.

4.6.5 Permissible Spent Fuel Pool Cooling and Alternate Decay Heat Removal System alignments throughout the outage shall follow NIP-OUT-01 requirements.

4.6.6 If during refueling an event occurs that requires the SFP gates to be installed, the SFC single failure criteria does not apply. This condition is bounded by the Emergency Core offload in the USAR and the same requirements apply.

4.4.6 Applicable radiological precautions shall be observed. Radiation Protection shall be contacted for guidance, as required.

4.4.7 ALARA practices shall be observed to minimize personnel exposure and the spread of contamination.

(C8) 4.4.8 To avoid extremely high radiation doses in the upper areas of the Drywell, highly activated or potentially activated components shall not be placed or stored on the Bulkhead area of the reactor cavity unless specifically approved by the Rad Protection Manager or designee.

4.5 General

4.5.1 The controller of this procedure shall be a Licensed Reactor Operator. This operator shall maintain direct communication with the Fuel Handling Team.

(C2) 4.5.2 Reactor Engineering personnel shall sign off a working copy of the Fuel Movement Instructions on the Refuel Platform to document independent verification.

4.5.3 The Controller will sign off the Control Rooms copy of the Fuel Movement Instructions when notified by the Fuel Handling Team that a step has been completed and confirm the next planned movement step with the Fuel Handling Team.

4.5.4 The use of clear plastic of any kind is prohibited on the Refuel Floor unless approved by reactor engineering personnel.

4.5.5 The SSS shall be notified immediately when a step cannot be completed as stated or if acceptance criteria are not met.

4.5.6 Prior to initialing any step in this procedure, all individuals shall place their initials, signatures, and printed names on Attachment 1, Signature and Initial Log.

4.5.7 N/A or N/R may be used where the procedure specifically allows it.

4.5.8 N/A or N/R may be used to eliminate steps when only a portion of the procedure is performed, such as Post-Maintenance Testing, retest to verify questionable data, or other testing. Document the reason for using N/A or N/R in Section 9, Remarks.

4.5.9 Secondary Containment Integrity shall be maintained while handling irradiated fuel, during core alterations, and during activities that could potentially drain the reactor.

4.5.10 Total drive flow (RHR Shutdown Cooling and Reactor Recirculation Pump) through the jet pumps shall be <5700 gpm when any in-core instrumentation is not fully surrounded by fuel assemblies or blade guides in the Reactor Vessel to preclude damage from flow induced vibration and prevent blade guides from lifting.

4.5.11 If an SRM count rate falls below 3 cps during the core offload prior to the removal of the four fuel assemblies immediately surrounding that SRM, that SRM shall be demonstrated operable by a signal to noise ratio check per N2-OSP-NMS-@002.

4.5.12 No control rod withdrawal will be allowed during fuel movement. Fuel movement shall be stopped for the performance of N2-OSP-RMC-W@002, as required.

4.5.13 Replacement of a bent channel fastener shall be performed per N2-FHP-16. Required changes to Fuel Movement Instruction shall be made in accordance with Precautions 4.2.11 and 4.2.12.

(C5) 4.5.14 Core offload is considered a special evolution and, as such, is controlled by GAP-SAT-03.

4.5.15 The Fuel Handler Spotter shall sign off a working copy of the Fuel Movement Instructions on the refuel platform to document completion of fuel or blade guide movement.

4.6 <u>Spent Fuel Pool Cooling System (SFC) Requirements</u>

(C7)
4.6.1 For a normal full core offload the SFC system must comply with single failure criteria which requires a Safety Related (SR) system with SR power capable of performing it's SR function. The system must be able to withstand a single active failure assuming no off-site power. Thus both loops of SFC are required to be available. Full core offload can not begin unless Engineering has performed cycle specific heat load analysis that demonstrate SFC single failure conformance.

4.6.2 Prior to the performance of this procedure, cycle specific heat load calculations must be performed by Engineering to determine applicable restrictions for Spent Fuel Pool Cooling (SFC) to ensure compliance with SFC Design Basis in USAR Section 9.1.3.
This would include limits on decay time, bundle offload rates and Service water temperatures. This restrictions shall be incorporated in a N2-OSP-LOG-@001 prior to fuel offload.

4.6.3 During a Divisional Bus and/or Divisional Service Water outage, the SFC Single Failure requirement is relaxed per GL 91-18. The requirement is to have 2 100% decay heat removal systems for pool cooling available. One of which will be an SFC loop with Safety Related cooling water available. The normal lineup will be one ADH train operating with SFC in backup or viceversa, or both operating.

4.6.4 Isolation of the Spent Fuel Pool from the Reactor Cavity is restricted until after cycle specific calculations are performed by Engineering that demonstrate single failure conformance.

4.6.5 Permissible Spent Fuel Pool Cooling and Alternate Decay Heat Removal System alignments throughout the outage shall follow NIP-OUT-01 requirements.

4.6.6 If during refueling an event occurs that requires the SFP gates to be installed, the SFC single failure criteria does not apply. This condition is bounded by the Emergency Core offload in the USAR and the same requirements apply.

Initials/Date

5.0 PREREQUISITES

NOTE: The steps in Section 5.0 may be signed off in any order as long as any time limits given in a step are met.

5.1 Obtain a copy of the Fuel Movement Instructions, approved by the Reactor Engineering Supervisor. ____/____

5.2 Verify that all personnel assigned to perform fuel movements have completed fuel movement OJT/OJE. ____/____

5.3 Verify that a sufficient number of double blade guides, as determined by the Reactor Engineering Supervisor, are assembled AND stored in the Spent Fuel Pool. ____/____

5.4 Verify that Secondary Containment is established AND operable per Technical Specification 3/4.6.5. ____/____

5.5 Verify a Refuel Floor Area Radiation Monitor, 2RMS-RE111 (RMS111) OR 2RMS-RE112 (RMS112), is operable per N2-OSP-LOG-M001 (USAR SEC. 7.5.3). ____/____

5.6 Verify Refueling Platform And Grapple Inspection is satisfactory per N2-PM-S001. ____/____

5.7 Notify QA of the expected start time of this procedure so they can provide monitoring of fuel movement activities, as determined by QA supervision.

____________ / ______ / ______
Person Notified Date Time ____/____

5.8 Review Equipment Status Log for any conditions that would prevent the movement of Irradiated Fuel in the Reactor Building OR Core Alterations. ____/____

5.9 Verify that NO other testing is in progress that affects this procedure. ____/____

5.10 Radiation Protection shall:

a. Obtain a Specific Radiation Work Permit AND record RWP Number below:

____
RWP Number ____/____

Initials/Date

5.11 (Cont)

Technical Specification	Description	Initials/Date
• 3/4.6.5.2	Secondary Containment Automatic Isolation Dampers	/
• 3/4.6.5.3	Standby Gas Treatment System	/
Plant Systems		
• 3/4.7.1.2	Plant Service Water System - Shutdown	/
• 3/4.7.3	Control Room Outdoor Air Special Filter Train System	/
Electrical Power Systems		
• 3/4.8.1.2	AC Sources - Shutdown	/
• 3/4.8.2.2	DC Sources - Shutdown	/
• 3/4.8.3.2	Onsite Power Distribution Systems-Shutdown	/

5.12 The General Supervisor - Operations, OR his/her alternate, has verified compliance with ALL Technical Specification requirements for the movement of Irradiated Fuel in the Reactor Building OR Core Alterations.

General Supervisor - Operations (OR Alternate) ____/________

5.13 The plant is in Operational Condition 5 <u>Refueling</u>. ____/________

5.14 Verify the status of the Refuel Floor as follows:

5.14.1 Reactor Vessel head removed and stored on Refuel Floor. ____/________

5.14.2 Separator AND Dryer removed and stored in the Reactor Internals Storage Pit. ____/________

5.14.3 Main Steam Line plugs installed. ____/________

5.14.4 Fuel Transfer Shield Bridge installed. ____/________

5.14.5 Underwater lights installed AND operable. ____/________

5.14.6 Decon Platform is over Reactor Internals Storage Pit. ____/________

5.14.7 Spent Fuel Pool gates removed. ____/________

Initials/Date

5.11 (Cont)

Technical Specification	Description	Initials/Date
• Tables 3.3.2-1.3.b/ 4.3.2.1-1.3.b	Reactor Building Below the Refuel Floor Exhaust Radiation - High	___/___

ECCS Actuation Instrumentation

• Tables 3.3.3-1.D.1/ 4.3.3.1-1.D.1	4.16-kV Emergency Bus Undervoltage-Loss of Voltage (Divisions I & II)	___/___
• Tables 3.3.3-1.D.2/ 4.3.3.1-1.D.2	4.16-kV Emergency Bus Undervoltage-Degraded Voltage (Divisions I & II)	___/___

Control Rod Block Instrumentation

• Tables 3/4.3.6-1.2	SRM	___/___
• Tables 3/4.3.6-1.3	IRM	___/___
• Tables 3/4.3.6-1.6.b	Reactor Mode Switch (Refuel Mode)	___/___

Radiation Monitoring Instrumentation

• Tables 3/4.3.7.1-1.1	Main Control Room Ventilation Radiation Monitors	___/___

ECCS - Shutdown

• Sections 3/4.5.2	ECCS - Shutdown	___/___

Secondary Containment

• 3/4.6.5.1	Secondary Containment Integrity	___/___
• 3/4.6.1.2	Primary Containment Leakage N2-ODP-OPS-0113	___/___

Initials/Date

5.11 (Cont)

Technical Specification	Description	Initials/Date
• 3/4.6.5.2	Secondary Containment Automatic Isolation Dampers	___/___
• 3/4.6.5.3	Standby Gas Treatment System	___/___
Plant Systems		
• 3/4.7.1.2	Plant Service Water System - Shutdown	___/___
• 3/4.7.3	Control Room Outdoor Air Special Filter Train System	___/___
Electrical Power Systems		
• 3/4.8.1.2	AC Sources - Shutdown	___/___
• 3/4.8.2.2	DC Sources - Shutdown	___/___
• 3/4.8.3.2	Onsite Power Distribution Systems-Shutdown	___/___

5.12 The General Supervisor - Operations, OR his/her alternate, has verified compliance with ALL Technical Specification requirements for the movement of Irradiated Fuel in the Reactor Building OR Core Alterations.

General Supervisor - Operations (OR Alternate) ___/___

5.13 The plant is in Operational Condition 5 Refueling. ___/___

5.14 Verify the status of the Refuel Floor as follows:

5.14.1 Reactor Vessel head removed and stored on Refuel Floor. ___/___

5.14.2 Separator AND Dryer removed and stored in the Reactor Internals Storage Pit. ___/___

5.14.3 Main Steam Line plugs installed. ___/___

5.14.4 Fuel Transfer Shield Bridge installed. ___/___

5.14.5 Underwater lights installed AND operable. ___/___

5.14.6 Decon Platform is over Reactor Internals Storage Pit. ___/___

5.14.7 Spent Fuel Pool gates removed. ___/___

Initials/Date

5.15 The Manager Radiation Protection Unit 2, OR his/her alternate, has verified that adequate radiological equipment AND procedural controls (including Refuel Area Radiation Monitors AND portable monitor(s) below the Fuel Transfer Shield Bridge) are in place OR are available per S-RAP-RPP-0801.

Manager Radiation Protection Unit 2 (OR Alternate) ____/________

5.16 Verify completion of the following checks AND surveillance:

5.16.1 The reactor has been subcritical for at least 24 hours. (T/S 4.9.4)

Reactor subcritical:__________/__________ ____/________
Date Time

5.16.2 The Refuel Bridge has been demonstrated operable per N2-OSP-FNR-@001 within seven days prior to commencement of fuel movement. (T/S 4.9.6)

Completion:__________/__________ ____/________
Date Time

5.16.3 Within 24 hours prior to commencement of fuel movement:

a. The Reactor Mode Switch Refuel Position interlocks have been demonstrated operable per N2-OSP-RMC-W@002 (T/S 4.9.1.2)

Completion:__________/__________ ____/________
Date Time

b. A Channel Functional Test for ALL operable SRMs, has been completed per N2-ISP-NMS-W@008 (T/S 4.9.2.b.1)

Completion:__________/__________ ____/________
Date Time

5.16.4 A Channel Check on ALL operable SRMs has been performed per N2-OSP-LOG-S004/5 prior to commencement of fuel movement. (T/S 4.9.2.a & 4.9.2.c)

Completion:__________/__________ ____/________
Date Time

5.16.5 The required SRMs are operable in accordance with Technical Specification 3.9.2 AND are NOT bypassed. ____/________

Initials/Date

6.0 PROCEDURE

6.1 Radiation Protection Drywell Restriction

NOTES: 1. For transfers of fuel between the Spent Fuel Pool and Reactor Vessel, a radiation monitor will be located under (Drywell side) the Fuel Transfer Shielding Bridge (Japanese Bridge) to warn of a fuel drop accident. This monitor will alarm at a setting determined by the Radiation Protection Supervisor.

2. No one is allowed access to elevations above 288' in the Drywell while fuel transfers are being made except by special permission by the Radiation Protection Supervision AND the SSS.

6.1.1 Notify Radiation Protection to control access above 288' elevation in the Drywell per S-RAP-RPP-0801.

____________ / ______ / ______ ____ / ______
Person Notified Time Date

6.1.2 Start N2-OSP-LOG-@001 for Cycle specific SFC restrictions as applicable. ____ / ______

6.2 Complete Core Offload

NOTES: 1. The controller of this procedure shall be a Licensed Reactor Operator. This Operator shall maintain direct communication with the Fuel Handling Team.

(C2) 2. Reactor Engineering personnel shall sign off a working copy of the Fuel Movement Instructions on the Refuel Platform to document independent verification.

3. The Controller will sign off the Control Room's copy of the Fuel Movement Instructions when notified by the Fuel Handling Team that a step has been completed and confirm the next planned movement step with the Fuel Handling Team.

4. If ALL the fuel assemblies are to be removed from a single fuel cell, a double blade guide or two single blade guides must be installed to provide control rod support and alignment. The blade guide(s) is (are) to be installed after the first two fuel assemblies have been removed. The first two fuel assemblies removed shall be diagonally opposed.

Initials/Date

PLANT IMPACT:
- **COMPLETE CORE OFFLOAD**
- **CORE ALTERATIONS PERFORMED**
- **BYPASSING OF SRM DOWNSCALE ROD BLOCKS**
- **REMOVAL OF K2 RELAYS FROM SRM CIRCUITS**
- **INSTALLATION OF JUMPERS ACROSS K2 RELAY CONTACTS**
- **BYPASSING OF SRM PERIOD ALARMS AFTER ALL FUEL HAS BEEN OFFLOADED**
- **REMOVAL OF K5 RELAYS FROM SRM CIRCUITS**
- **INSTALLATION OF JUMPERS ACROSS K5 RELAY CONTACTS**
- **SECONDARY CONTAINMENT INTEGRITY IS REQUIRED FOR CORE ALTERATIONS**

5.20 Discuss plant impact with the SSS. Obtain SSS permission to perform procedure and acknowledgement that temporary alterations are to be used. ____/______ SSS

5.21 Discuss Plant Impact with the CSO. Notify the CSO that procedure is to be performed and that temporary alterations are to be used. ____/______ CSO

5.22 Record complete core offload start date AND time:

______/______
Date Time

____/______

Initials/Date

6.0 PROCEDURE

6.1 Radiation Protection Drywell Restriction

NOTES:

1. For transfers of fuel between the Spent Fuel Pool and Reactor Vessel, a radiation monitor will be located under (Drywell side) the Fuel Transfer Shielding Bridge (Japanese Bridge) to warn of a fuel drop accident. This monitor will alarm at a setting determined by the Radiation Protection Supervisor.

2. No one is allowed access to elevations above 288' in the Drywell while fuel transfers are being made except by special permission by the Radiation Protection Supervision AND the SSS.

6.1.1 Notify Radiation Protection to control access above 288' elevation in the Drywell per S-RAP-RPP-0801.

_______________ / _______ / _______ ____ / ______

Person Notified Time Date

6.1.2 Start N2-OSP-LOG-@001 for Cycle specific SFC restrictions as applicable. ____ / ______

6.2 Complete Core Offload

NOTES:

1. The controller of this procedure shall be a Licensed Reactor Operator. This Operator shall maintain direct communication with the Fuel Handling Team.

2. (C2) Reactor Engineering personnel shall sign off a working copy of the Fuel Movement Instructions on the Refuel Platform to document independent verification.

3. The Controller will sign off the Control Room's copy of the Fuel Movement Instructions when notified by the Fuel Handling Team that a step has been completed and confirm the next planned movement step with the Fuel Handling Team.

4. If ALL the fuel assemblies are to be removed from a single fuel cell, a double blade guide or two single blade guides must be installed to provide control rod support and alignment. The blade guide(s) is (are) to be installed after the first two fuel assemblies have been removed. The first two fuel assemblies removed shall be diagonally opposed.

6.2 (Cont)

NOTES: (Cont)

5. If core alterations are suspended, Attachment 2 MUST be completed prior to resuming core alterations.

6. An Inoperable SRM may be bypassed (Joystick at PNL-603) AND Fuel Movement may continue provided the requirements of Precaution 4.2.11 and 4.2.12 are met.

7. ALL movement of nuclear fuel will be done as directed by Appendix 1, Fuel Movement Instructions.

8. Refueling bridge, mast and grapple operation is controlled in N2-OP-39.

9. As fuel is removed near the SRMs, their count rates will decrease. The SRM count rate will eventually fall below the downscale setpoint and cause a control rod block.

10. During complete core spiral offloading, the SRM count rate need not be maintained when the four fuel assemblies immediately surrounding an SRM are removed.

11. No visual indication is required when all but four assemblies have been removed from the core.

12. Fuel movement must be stopped in the event of a Fuel Movement Discrepancy or Fuel Loading Error. Fuel movement may resume once Attachment 6, Fuel Movement Discrepancy/Fuel Loading Error Evaluation, and, Attachment 7, Fuel Movement Discrepancy/Fuel Loading Error Evaluation Form (FMDEF) are completed.

13. Fuel movement must be stopped if any of the conditions in Attachment 5 are encountered. (See step 4.2.10)

14. Steps 6.2.3 to 6.2.8 and N2-OP-39 describe the actions to be taken to move each fuel assembly and blade guide. The signoffs for these steps will be recorded on Appendix 1, Fuel Movement Instructions.

15. N2-OSP-LOG-@001 should be used shiftly to verify compliance with the limitations on Spent Fuel movement into Spent Fuel Pool.

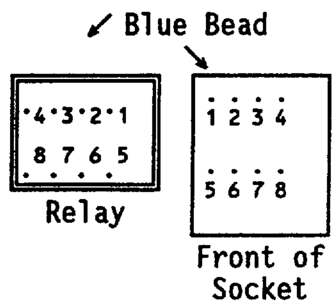
Initials/Date

6.3.3 Remove K2 relay for SRM A in Auxiliary Trip Unit C51A-Z2A at 2CEC*PNL606. ____/____

____/____
Ind. Verifier

6.3.4 Place jumpers across the K2 relay contacts:

a. Contact 3 to 5 ____/____



b. Contact 4 to 6 ____/____

Ref: GE dwg 195B9206 sh. 2

____/____
Ind. Verifier

6.3.5 Verify the following alarms and indicators are clear:

a. 603215 "SRM DOWNSCALE" ____/____

b. 603442 "CONTROL ROD OUT BLOCK" ____/____

c. SRM A Downscale white indication light on P603. ____/____

6.3.6 (C6) Make an ESL entry that SRM A downscale Rod Block is bypassed and that TS 4.0.4 is applicable to TS 3.3.6-1 Note f. ____/____
SSS

6.3.7 Place holdout on SRM drawer A. ____/____

6.3.8 Place K2 relay for SRM A in an envelope with a holdout reference tag attached AND store the relay in the Control Room red markup drawer under system #92. ____/____

Initials/Date

6.2.6 IF any SRM count rate falls below 3 cps prior to the four fuel assemblies immediately surrounding it being removed, THEN demonstrate operability of that SRM per N2-OSP-NMS-@002. IF N2-OSP-NMS-@002 is performed for an SRM, document the demonstration of operability by checking under the "Performed" column below. (IF N2-OSP-NMS-@002 is NOT needed for an SRM, check N/A.)

SRM	N2-OSP-NMS-@002 Performed Sat	N/A	
A	( )	( )	____/______
B	( )	( )	____/______
C	( )	( )	____/______
D	( )	( )	____/______

6.2.7 Repeat Steps 6.2.3 through 6.2.6 as needed to sequentially move ALL the fuel assemblies and blade guides per Appendix 1, Fuel Movement Instructions.

6.2.8 AFTER ALL fuel assemblies have been removed from the reactor vessel, DEFEAT SRM Period Alarms per Attachment 4. ____/______

6.3 Bypassing SRM A Downscale Rod Block

NOTE: During core offloading, at least one SRM will be continuously monitoring the fueled region. The SRM count rate need not be maintained when the four fuel assemblies immediately surrounding an SRM are removed.

PLANT IMPACT: ASSOCIATED SRM DOWNSCALE ROD BLOCK AND DOWNSCALE ANNUNCIATOR WILL BE DISABLED.

6.3.1 Obtain SSS permission to install the jumpers. ____/______ SSS

6.3.2 Inform the CSO that relay C51A-K2A will be pulled AND jumpers will be placed in SRM A Downscale Rod Block circuit. ____/______

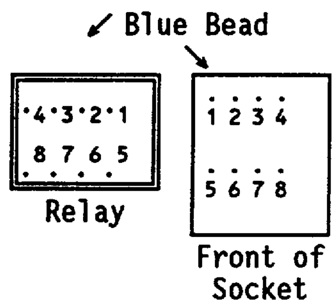
Initials/Date

6.3.3 Remove K2 relay for SRM A in Auxiliary Trip Unit C51A-Z2A at 2CEC*PNL606. ____/____

____/____
Ind. Verifier

6.3.4 Place jumpers across the K2 relay contacts:

a. Contact 3 to 5 ____/____



b. Contact 4 to 6 ____/____

Ref: GE dwg 195B9206 sh. 2

____/____
Ind. Verifier

6.3.5 Verify the following alarms and indicators are clear:

a. 603215 "SRM DOWNSCALE" ____/____

b. 603442 "CONTROL ROD OUT BLOCK" ____/____

c. SRM A Downscale white indication light on P603. ____/____

6.3.6 (C6) Make an ESL entry that SRM A downscale Rod Block is bypassed and that TS 4.0.4 is applicable to TS 3.3.6-1 Note f. ____/____
SSS

6.3.7 Place holdout on SRM drawer A. ____/____

6.3.8 Place K2 relay for SRM A in an envelope with a holdout reference tag attached AND store the relay in the Control Room red markup drawer under system #92. ____/____

Initials/Date

6.4 Bypassing SRM B Downscale Rod Block

NOTE: During core offloading, at least one SRM will be continuously monitoring the fueled region. The SRM count rate need not be maintained when the four fuel assemblies immediately surrounding an SRM are removed.

PLANT IMPACT: **ASSOCIATED SRM DOWNSCALE ROD BLOCK AND DOWNSCALE ANNUNCIATOR WILL BE DISABLED.**

6.4.1 Obtain SSS permission to install the jumpers. ____/____ SSS

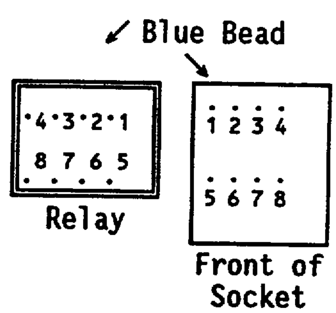
6.4.2 Inform the CSO that relay C51A-K2B will be pulled AND jumpers will be placed in SRM B Downscale Rod Block circuit. ____/____

6.4.3 Remove K2 relay for SRM B in Auxiliary Trip Unit C51A-Z2B at 2CEC*PNL633. ____/____

____/____ Ind. Verifier

6.4.4 Place jumpers across the K2 relay contacts:

a. Contact 3 to 5 ____/____



b. Contact 4 to 6 ____/____

Ref: GE dwg 195B9206 sh. 2

____/____ Ind. Verifier

6.4.5 Verify the following alarms and indicators are clear:

a. 603215 "SRM DOWNSCALE" ____/____

b. 603442 "CONTROL ROD OUT BLOCK" ____/____

c. SRM B Downscale white indication light on P603. ____/____

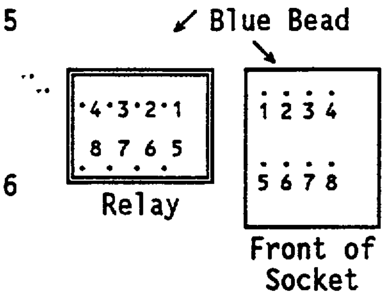
6.4.6 (C6) Make an ESL entry that SRM B downscale Rod Block is bypassed and that TS 4.0.4 is applicable to TS 3.3.6-1, Note f. ____/____ SSS

Initials/Date

6.6.4 Place jumpers across the K2 relay contacts:

a. Contact 3 to 5 ____/____

b. Contact 4 to 6



____/____

Ref: GE dwg 195B9206 sh. 2 ____/____
Ind. Verifier

6.6.5 Verify the following alarms and indicators are clear:

a. 603215 "SRM DOWNSCALE" ____/____

b. 603442 "CONTROL ROD OUT BLOCK" ____/____

c. SRM D Downscale white indication light on P603. ____/____

6.6.6 (C6) Make an ESL entry that SRM D downscale Rod block is bypassed and that TS 4.0.4 is applicable to TS 3. ____/____
SSS

6.6.7 Place holdout on SRM drawer D. ____/____

6.6.8 Place K2 relay for SRM D in an envelope with a holdout reference tag attached AND store the relay in the Control Room red markup drawer under system #92. ____/____

7.0 <u>RETURN TO NORMAL</u>

7.1 Shutdown the Refuel Bridge per N2-OP-39, Section G.1.0 OR G.3.0, OR as directed by the Refuel Floor SRO. ____/____

7.2 IF Attachment 2, Checklist For Resuming Core Alterations, was used, verify that all copies completed are attached to this procedure. ____/____

N/A, Attachment 2 was NOT used . . . . (__)

7.3 IF no Suspension Of Core Alterations occurred during the performance of this procedure, mark Attachment 2, Checklist For Resuming Core Alterations, N/A. ____/____

N/A, Attachment 2 was used . . . . (__)

Initials/Date

6.5.5 Verify the following alarms and indicators are clear:

a. 603215 "SRM DOWNSCALE" ___/___

b. 603442 "CONTROL ROD OUT BLOCK" ___/___

c. SRM C Downscale white indication light on P603. ___/___

6.5.6 (C6) Make an ESL entry that SRM C downscale Rod Block is bypassed and that TS 4.0.4 is applicable to TS 3.3.6-1, Note f. ___/___ SSS

6.5.7 Place holdout on SRM drawer C. ___/___

6.5.8 Place K2 relay for SRM C in an envelope with a holdout reference tag attached AND store the relay in the Control Room red markup drawer under system #92. ___/___

6.6 Bypassing SRM D Downscale Rod Block

NOTE: During core offloading, at least one SRM will be continuously monitoring the fueled region. The SRM count rate need not be maintained when the four fuel assemblies immediately surrounding an SRM are removed.

PLANT IMPACT: ASSOCIATED SRM DOWNSCALE ROD BLOCK AND DOWNSCALE ANNUNCIATOR WILL BE DISABLED.

6.6.1 Obtain SSS permission to install the jumpers. ___/___ SSS

6.6.2 Inform the CSO that relay C51A-K2D will be pulled AND jumpers will be placed in SRM D Downscale Rod Block circuit. ___/___

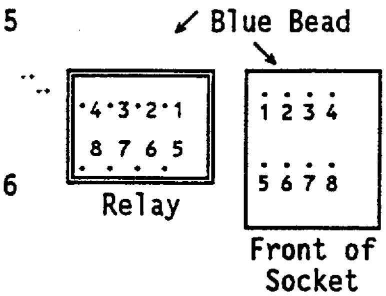
6.6.3 Remove K2 relay for SRM D in Auxiliary Trip Unit C51A-Z2D at 2CEC*PNL633. ___/___

___/___ Ind. Verifier

Initials/Date

6.6.4 Place jumpers across the K2 relay contacts:

a. Contact 3 to 5 ____/____



b. Contact 4 to 6 ____/____

Ref: GE dwg 195B9206 sh. 2 ____/____
Ind. Verifier

6.6.5 Verify the following alarms and indicators are clear:

a. 603215 "SRM DOWNSCALE" ____/____

b. 603442 "CONTROL ROD OUT BLOCK" ____/____

c. SRM D Downscale white indication light on P603. ____/____

6.6.6 (C6) Make an ESL entry that SRM D downscale Rod block is bypassed and that TS 4.0.4 is applicable to TS 3. ____/____
SSS

6.6.7 Place holdout on SRM drawer D. ____/____

6.6.8 Place K2 relay for SRM D in an envelope with a holdout reference tag attached AND store the relay in the Control Room red markup drawer under system #92. ____/____

7.0 RETURN TO NORMAL

7.1 Shutdown the Refuel Bridge per N2-OP-39, Section G.1.0 OR G.3.0, OR as directed by the Refuel Floor SRO. ____/____

7.2 IF Attachment 2, Checklist For Resuming Core Alterations, was used, verify that all copies completed are attached to this procedure. ____/____

N/A, Attachment 2 was NOT used . . . . (__)

7.3 IF no Suspension Of Core Alterations occurred during the performance of this procedure, mark Attachment 2, Checklist For Resuming Core Alterations, N/A. ____/____

N/A, Attachment 2 was used . . . . (__)

Initials/Date

7.4 Attach the Refuel Floor working copy of Appendix 1, Fuel Movement Instructions, to this procedure to provide documentation of independent verification initials for the fuel movements. ____/____

7.5 Review both copies of Appendix 1, Fuel Movement Instructions, to ensure all fuel movements have been initialed OR appropriate remarks made. ____/____

7.6 Verify ALL test personnel who have initialed steps in this procedure, including attachments have filled out Attachment 1, Signature And Initial Sheet. ____/____

7.7 Notify the CSO that this procedure is completed AND of any devices to be left out of normal such as SRM Downscale Rod Blocks left bypassed AND SRM Period Alarms defeated. ____/____
CSO

7.8 Notify the SSS that this procedure is completed AND of any devices to be left out of normal such as SRM Downscale Rod Blocks left bypassed AND SRM Period Alarms defeated. ____/____
SSS

7.9 Record procedure stop date AND time:

____/____
Date Time
____/____

8.0 ACCEPTANCE CRITERIA

All fuel has been Offloaded from the Reactor Vessel to the Spent Fuel Pool in accordance with Appendix 1, Fuel Movement Instructions, AND Section 7.0 has been completed satisfactorily. ____/____

9.0 RECORD REVIEW AND DISPOSITION

9.1 Record

This procedure shall be maintained by Nuclear Records Management for the Permanent Plant File in accordance with NIP-RMG-01, Identification, Maintenance, Storage, and Transfer of Nuclear Division Records.

ATTACHMENT 2
CHECKLIST FOR RESUMING CORE ALTERATIONS (OFFLOAD)

NOTE: Following a Suspension of Core Alterations, perform the following surveillances and verifications within the required time intervals prior to resumption of fuel movement.

		Initials/Date/Time
I.	Within 24 Hours	
•	N2-OSP-RMC-W@002: Applicable sections as required for Post-Maintenance Testing following maintenance affecting Refuel Position Interlock operability. (T/S 4.9.1.2 & 4.9.1.3)	____/____/____
II.	Within 12 Hours	
•	N2-OSP-LOG-S004/S005 completed satisfactorily.	____/____/____
•	Verify T/S 3/4.9.2 is met.	____/____/____
III.	Within 1 Hour	
•	Verify direct communication demonstrated between the Control Room AND the Refuel Floor. (T/S 4.9.5)	____/____/____
•	Verify ALL control rods fully inserted. (T/S 4.9.3)	____/____/____
•	Verify Reactor Vessel water level > 22'3" over top of the Reactor Vessel flange by observing water flowing over the Reactor Cavity weirs. (T/S 4.9.8)	____/____/____
•	Verify that the Reactor Mode Switch is locked in the REFUEL position. (T/S 4.9.1.1)	____/____/____
•	Verify that N2-PM-S01, Refueling Platform and Grapple Inspection, has been performed within the required time interval.	____/____/____
•	Verify that areas of the Drywell, as determined by Radiation Protection Manager and SSS, have been evacuated by ALL personnel. (Precaution 4.4.2)	____/____/____
•	Verify that secondary containment integrity is met per T/S 3/4.6.5.1 and N2-ODP-OPS-0113.	____/____/____
IV.	Record the date AND time of resumption of Core Alterations here AND in the SSS log.	____/____/____

______/______
Date Time

____/____/____
SSS

ATTACHMENT 1
SIGNATURES AND INITIALS SHEET

Sheet __ of __

PRINTED NAME	SIGNATURE	INITIALS

ATTACHMENT 2
CHECKLIST FOR RESUMING CORE ALTERATIONS (OFFLOAD)

NOTE: Following a Suspension of Core Alterations, perform the following surveillances and verifications within the required time intervals prior to resumption of fuel movement.

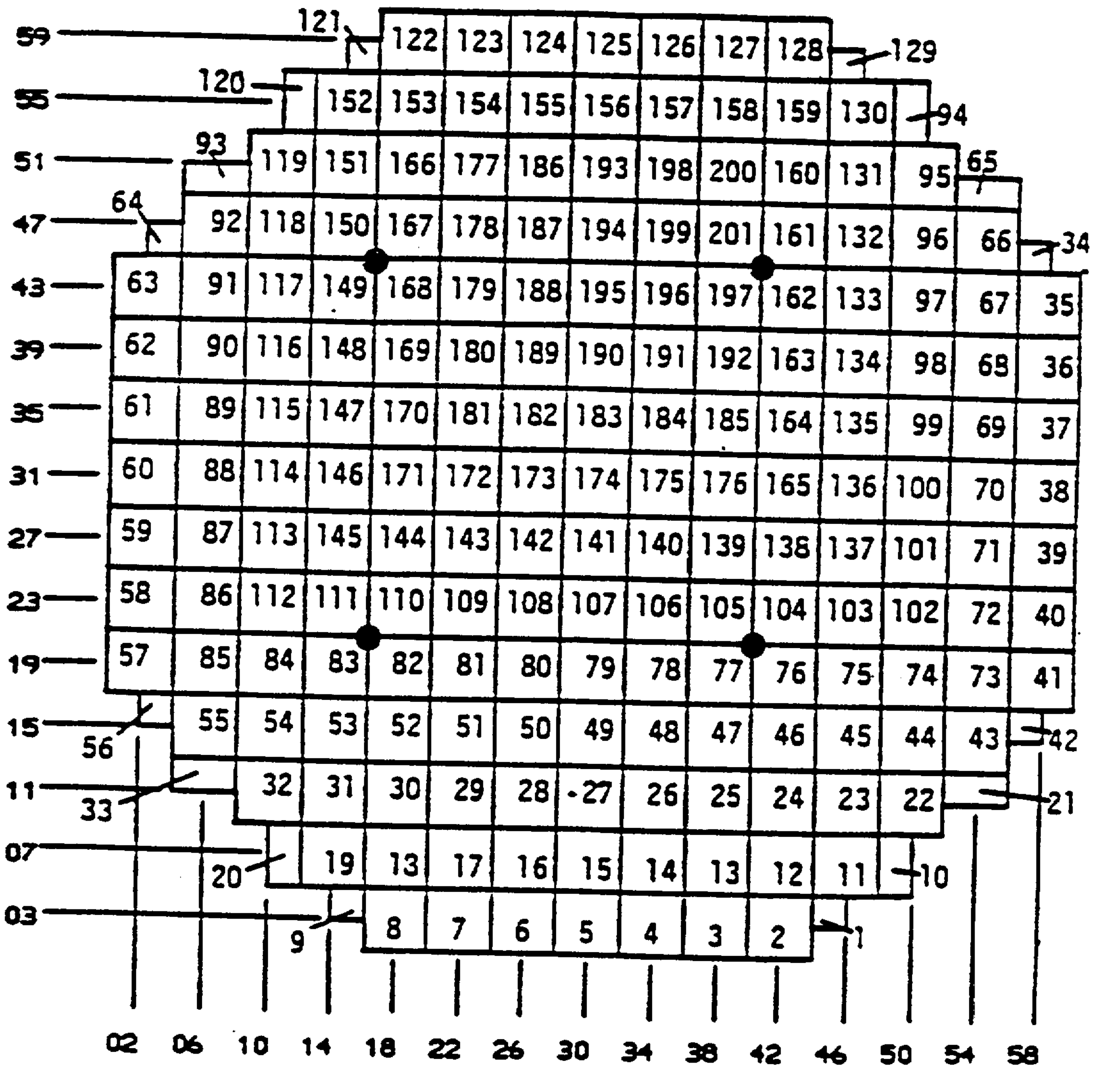
		Initials/Date/Time
I.	Within 24 Hours	
•	N2-OSP-RMC-W@002: Applicable sections as required for Post-Maintenance Testing following maintenance affecting Refuel Position Interlock operability. (T/S 4.9.1.2 & 4.9.1.3)	____/____/____
II.	Within 12 Hours	
•	N2-OSP-LOG-S004/S005 completed satisfactorily.	____/____/____
•	Verify T/S 3/4.9.2 is met.	____/____/____
III.	Within 1 Hour	
•	Verify direct communication demonstrated between the Control Room AND the Refuel Floor. (T/S 4.9.5)	____/____/____
•	Verify ALL control rods fully inserted. (T/S 4.9.3)	____/____/____
•	Verify Reactor Vessel water level > 22'3" over top of the Reactor Vessel flange by observing water flowing over the Reactor Cavity weirs. (T/S 4.9.8)	____/____/____
•	Verify that the Reactor Mode Switch is locked in the REFUEL position. (T/S 4.9.1.1)	____/____/____
•	Verify that N2-PM-S01, Refueling Platform and Grapple Inspection, has been performed within the required time interval.	____/____/____
•	Verify that areas of the Drywell, as determined by Radiation Protection Manager and SSS, have been evacuated by ALL personnel. (Precaution 4.4.2)	____/____/____
•	Verify that secondary containment integrity is met per T/S 3/4.6.5.1 and N2-ODP-OPS-0113.	____/____/____
IV.	Record the date AND time of resumption of Core Alterations here AND in the SSS log.	____/____/____

____/____
Date Time

____/____/____
SSS

ATTACHMENT 3
CORE OFF LOAD SEQUENCE MAP

SRM	Coordinate
A	16-45
B	40-45
C	40-21
D	16-21



● = SRM Detector Location

ATTACHMENT 5
CRITERIA FOR STOPPING FUEL MOVEMENT

1.0 Circumstances requiring stopping fuel movement include, but are not limited to, the following:

a. Inadvertent Criticality.

b. Loss of secondary containment

c. All control rods not fully inserted. (Such as during the performance of N2-OSP-RMC-W@002)

d. Less than the required operable number of SRM channels.

e. Loss of communication between the Control Room and the Refuel Floor.

f. Reactor Mode Switch not in SHUTDOWN or REFUEL position.

g. Anytime SRM operability is in question.

h. Loss of minimum required offsite power supplies.

i. Reactor cavity water level less than 22 feet 3 inches above the top of RPV flange.

j. At any time the SSS or SRO/LSRO deems appropriate.

k. Loss of Visual contact with Fuel or equipment.

ℓ. Loss of multiple control rod position indications

m. Refuel Floor area high radiation alarm.

n. Any fuel assembly damage.

ATTACHMENT 4 (Con't)

Initials

1.5 (Cont)

c. Place a holdout on SRM drawer B. ________

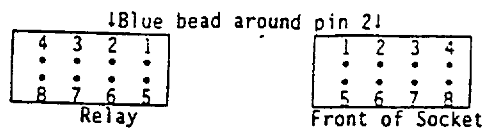
d. Place K5 relay for SRM B in an envelope with a holdout reference tag attached AND store the relay in the Control Room red markup drawer under system #92. ________

1.6 Remove relay K5C to defeat the SRM C Period Alarm as follows:

a. Remove K5 relay for SRM C in Auxiliary Trip Unit C51A-Z2C at 2CEC*PNL606. ________

____
Ind. Verif.

b. Install a jumper between pins 3 and 5 of the relay base for relay K5. ________



____
Ind. Verif.

c. Place a holdout on SRM drawer C. ________

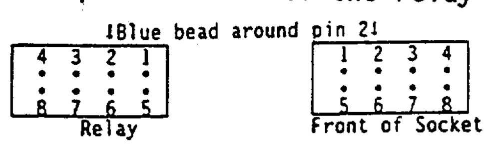
d. Place K5 relay for SRM C in an envelope with a holdout reference tag attached AND store the relay in the Control Room red markup drawer under system #92. ________

1.7 Remove relay K5D to defeat the SRM D Period Alarm as follows:

a. Remove K5 relay for SRM D in Auxiliary Trip Unit C51A-Z2D at 2CEC*PNL606. ________

____
Ind. Verif.

b. Install a jumper between pins 3 and 5 of the relay base for relay K5. ________



____
Ind. Verif.

c. Place a holdout on SRM drawer D. ________

d. Place K5 relay for SRM D in an envelope with a holdout reference tag attached AND store the relay in the Control Room red markup drawer under system #92. ________

1.8 SSS notified that SRM Period Alarms have been defeated. ________
SSS

ATTACHMENT 5
CRITERIA FOR STOPPING FUEL MOVEMENT

1.0 Circumstances requiring stopping fuel movement include, but are not limited to, the following:

a. Inadvertent Criticality.

b. Loss of secondary containment

c. All control rods not fully inserted. (Such as during the performance of N2-OSP-RMC-W@002)

d. Less than the required operable number of SRM channels.

e. Loss of communication between the Control Room and the Refuel Floor.

f. Reactor Mode Switch not in SHUTDOWN or REFUEL position.

g. Anytime SRM operability is in question.

h. Loss of minimum required offsite power supplies.

i. Reactor cavity water level less than 22 feet 3 inches above the top of RPV flange.

j. At any time the SSS or SRO/LSRO deems appropriate.

k. Loss of Visual contact with Fuel or equipment.

ℓ. Loss of multiple control rod position indications

m. Refuel Floor area high radiation alarm.

n. Any fuel assembly damage.

NOTE: A separate attachment must be completed for each Fuel Movement Discrepancy/Fuel Loading Error. Only the applicable sections need to be performed as required.

1.0 Fuel Loading Error

N/A, Section 1.0 is not applicable . . . . (___)

1.1 IF possible record Z digital display indicating how far the Fuel Bundle was moved into the core. . . . . (___)

____
Z digital display

1.2 Notify Procedure Controller of event AND to monitor SRM countrates. . (___)

1.3 IF an inadvertent criticality occurs, refer to N2-SOP-39 for guidance. (___)

1.4 Notify the following:

- SSS . . . . (___)
- Special Evolution Senior Manager (SESM) . . . . (___)
- Reactor Engineer (RE) . . . . (___)
- Refuel Floor Coordinator (RFC) . . . . (___)

1.5 IF Main Grapple is engaged AND SRM countrates are NOT increasing, place Fuel Bundle in SFP-Special Setdown Area . . . . (___)

1.6 Initiate a DER AND a Fuel Movement Discrepancy Fuel Loading Error Evolution Form (FMDEF) . . . . (___)

1.7 Document the location and orientation of the improperly placed Fuel Bundle in Comments Section of FMIs. . . . . (___)

1.8 PRIOR to resuming Fuel Movement verify that as a minimum the following have been completed:

a. SDM with the misloaded Fuel Bundle has been analyzed and/or has been determined to be bounded by current analysis. . . . . (___)

b. The FMDEF has been completed and all corrective actions have been approved. . . . . (___)

b. IF Blade Guide above Control Rod Tip AND the Control Rod is NOT properly supported to prevent tilting, perform the following:

1. Notify the following:
   - SSS . . . . (___)
   - Special Evolution Senior Manager (SESM) . . . . . (___)
   - Reactor Engineer (RE) . . . . (___)
   - Refuel Floor Coordinator (RFC) . . . . (___)

2. Visually determine Control Rod position using binoculars, telescope, or video camera as necessary . . . . (___)

NOTE: The following steps provide direction to restore Control Rod Support to limit damage to the Control Rod as well as core components.

3. IF Control Rod is positioned to allow Blade Guide entry perform the following:

   a. Verify that a blade guide can be inserted without damage to core components . . . . (___)

   b. Ensure refueling platform is correctly positioned (___)

   c. Slowly LOWER Main Grapple . . . . (___)

   d. Monitor Load Cell indication for weight DECREASE . (___)

   e. WHEN Load Cell indication decreases by 50 pounds (excluding expected Grapple weight transfers) or Blade Guide is seated STOP . . . . (___)

   f. Initiate a FMDEF (Attachment 7) AND a DER . . . . (___)

4. IF Control Rod is positioned such that a Blade Guide can NOT be lowered to provide support, perform the following:

   a. Place Blade Guide in SFP . . . . (___)

   b. Initiate a FMDEF (Attachment 7) AND a DER . . . . (___)

NOTES (Cont)

3. Ensure refueling platform is correctly positioned . . . . (___)

4. Slowly LOWER Grapple . . . . (___)

5. Monitor Load Cell indication for weight DECREASE . . . . (___)

6. WHEN Load Cell indication decreases by 50 pounds (excluding expected Grapple weight transfers) or Blade Guide is seated STOP . . (___)

7. Document Single Blade Guide movement in Comments Section of applicable step of the FMIs . . . . (___)

c. IF Control Rod position does not allow a single blade guide entry, stop fuel movement activities AND contact fuel vendor for assistance. . . . . . (___)

4. Initiate a FMDEF (Attachment 7) AND a DER . . . . . . (___)

c. IF Fuel Bundle below Control Rod TIP, perform the following:

1. Place Fuel Bundle back into original Core location . . (___)

2. Initiate a FMDEF (Attachment 7) and if required by SRO/LSRO a DER . . . . (___)

2.1.2 For a Blade Guide perform the following:

a. Determine if Blade Guide is above Control Rod Tip as follows: . . . . (___)

1. Z digital display

<u>AND</u>

2. Visual observation

b. IF Blade Guide above Control Rod Tip AND the Control Rod is NOT properly supported to prevent tilting, perform the following:

1. Notify the following:
   - SSS . . . . (___)
   - Special Evolution Senior Manager (SESM) . . . . . (___)
   - Reactor Engineer (RE) . . . . (___)
   - Refuel Floor Coordinator (RFC) . . . . (___)

2. Visually determine Control Rod position using binoculars, telescope, or video camera as necessary . . . . (___)

NOTE: The following steps provide direction to restore Control Rod Support to limit damage to the Control Rod as well as core components.

3. IF Control Rod is positioned to allow Blade Guide entry perform the following:

   a. Verify that a blade guide can be inserted without damage to core components . . . . (___)

   b. Ensure refueling platform is correctly positioned (___)

   c. Slowly LOWER Main Grapple . . . . (___)

   d. Monitor Load Cell indication for weight DECREASE . (___)

   e. WHEN Load Cell indication decreases by 50 pounds (excluding expected Grapple weight transfers) or Blade Guide is seated STOP . . . . (___)

   f. Initiate a FMDEF (Attachment 7) AND a DER . . . . (___)

4. IF Control Rod is positioned such that a Blade Guide can NOT be lowered to provide support, perform the following:

   a. Place Blade Guide in SFP . . . . (___)

   b. Initiate a FMDEF (Attachment 7) AND a DER . . . . (___)

c. IF Blade Guide is below the Control Rod Tip, perform the following.

1. Slowly LOWER Grapple . . . . (___)

2. Monitor Load Cell indication for weight decrease . . . (___)

3. WHEN Load Cell indication decreases by 50 pounds or Blade Guide is seated STOP . . . . (___)

4. Initiate a FMDEF . . . . (___)

2.2 Grapple Engaged Out of Sequence From SFP

N/A, Subsection 2.2 is not applicable . . . . (___)

2.2.1 For a Fuel Bundle perform the following:

a. Determine if Fuel Bundle has been withdrawn ABOVE TOP of SFP Rack: . . . . (___)

1. Z digital display

AND

2. Visual Inspection

b. IF Fuel Bundle ABOVE TOP of SFP Rack, place Fuel Bundle into SFP-Special Setdown Area . . . . (___)

c. IF Fuel Bundle BELOW TOP of SFP Rack, place Fuel Bundle back into original SFP location . . . . (___)

d. Initiate a FMDEF (Attachment 7) . . . . (___)

2.2.2 For a Blade Guide perform the following:

a. Determine if Blade Guide is ABOVE TOP of SFP Rack as follows: . . . . (___)

1. Z digital display

AND

2. Visual observation

b. IF Blade Guide is ABOVE TOP of SFP Rack, place Blade Guide in SFP-Special Setdown Area . . . . (___)

APPENDIX 1
FUEL MOVEMENT INSTRUCTIONS (FMIs)

To be provided by Reactor Engineering prior to fuel movement and attached after this page during the performance of the Return to Normal section of this procedure.

ATTACHMENT 7

FUEL MOVEMENT DISCREPANCY/FUEL LOADING ERROR EVALUATION FORM (FMDEF)

Date/time of discrepancy: ______/______ Refuel Floor SRO/LSRO: ______________

Discrepancy(ies) Identified:

Immediate Corrective Actions (SRO/LSRO):

Analysis and Final Disposition (Reactor Engineer):

Preparer: Reactor Engineer (Initials/Date): ______/______

Approval: SRO/LSRO (Initials/Date): ______/______

Concurrence:
Special Evolution Senior Manager (Initials/Date): ______/______

APPENDIX 1
FUEL MOVEMENT INSTRUCTIONS (FMIs)

To be provided by Reactor Engineering prior to fuel movement and attached after this page during the performance of the Return to Normal section of this procedure.

3/4.9 REFUELING OPERATIONS

3/4.9.1 REACTOR MODE SWITCH

LIMITING CONDITIONS FOR OPERATION

3.9.1 The reactor mode switch shall be OPERABLE and locked in the Shutdown or Refuel position. When the reactor mode switch is locked in the Refuel position:

a. A control rod shall not be withdrawn unless the Refuel position one-rod-out interlock is OPERABLE.

b. CORE ALTERATIONS shall not be performed using equipment associated with a Refuel position interlock unless at least the following associated Refuel position interlocks are OPERABLE for such equipment.

   1. All rods in.
   2. Refuel platform position.
   3. Refuel platform hoists fuel-loaded.
   4. Fuel grapple position.
   5. Service platform hoist fuel-loaded.

APPLICABILITY: OPERATIONAL CONDITION 5* #

ACTION:

a. With the reactor mode switch not locked in the Shutdown or Refuel position as specified, suspend CORE ALTERATIONS and lock the reactor mode switch in the Shutdown or Refuel position.

b. With the one-rod-out interlock inoperable, lock the reactor mode switch in the Shutdown position.

c. With any of the above required Refuel position equipment interlocks inoperable, suspend CORE ALTERATIONS with equipment associated with the inoperable Refuel position equipment interlock.

---

* See Special Test Exceptions 3.10.1 and 3.10.3.

The reactor shall be maintained in OPERATIONAL CONDITION 5 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

S 22-4

REACTOR MODE SWITCH

SURVEILLANCE REQUIREMENTS

4.9.1.1 The reactor mode switch shall be verified to be locked in the Shutdown or Refuel position as specified:

a. Within 2 hours before:

   1. Beginning CORE ALTERATIONS, and
   2. Resuming CORE ALTERATIONS when the reactor mode switch has been unlocked.

b. At least once per 12 hours.

4.9.1.2 Each of the above required reactor mode switch Refuel position interlocks* shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST within 24 hours before the start of and at least once per 7 days during control rod withdrawal or CORE ALTERATIONS, as applicable.

4.9.1.3 Each of the above required reactor mode switch Refuel position interlocks* that is affected shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST before resuming control rod withdrawal or CORE ALTERATIONS, as applicable, following repair, maintenance or replacement of any component that could affect the Refuel position interlock.

* The reactor mode switch may be placed in the Run or Startup/Hot Standby position to test the switch interlock functions provided that all control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITIONS FOR OPERATION

3.9.2 At least 2 source range monitor* (SRM) channels shall be OPERABLE and inserted to the normal operating level with:

a. Continuous visual indication of the required count rate in the control room,**

b. Audible annunciation in the control room,

c. One of the required SRM detectors located in the quadrant where CORE ALTERATIONS are being performed and the other required SRM detector located in an adjacent quadrant, and

d. Unless adequate shutdown margin has been demonstrated per Specification 3.1.1 and the "one rod out" interlock is OPERABLE per Specification 3.9.1, the shorting links shall be removed from the RPS circuitry prior to and any time one control rod is withdrawn.***

APPLICABILITY: OPERATIONAL CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS and insert all insertable control rods.

SURVEILLANCE REQUIREMENTS

4.9.2 Each of the above required SRM channels shall be demonstrated OPERABLE by:

a. At least once per 12 hours:

1. Performing a CHANNEL CHECK,
2. Verifying the detectors are inserted to the normal operating level, and
3. During CORE ALTERATIONS, verifying that the detector of an OPERABLE SRM channel is located in the core quadrant where CORE ALTERATIONS are being performed and another is located in an adjacent quadrant.

---

* The use of special movable detectors during CORE ALTERATIONS in place of the normal SRM nuclear detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

** During complete core spiral offload and reload, only one of the required SRM channels must have continuous visual indication in the control room. No visual indication is required until after the first four fuel bundles have been placed in the core, and no visual indication is required when all but four bundles have been removed from the core.

*** Not required for control rods removed per Specification 3.9.10.1 and 3.9.10.2.

SURVEILLANCE REQUIREMENTS

4.9.2 (Continued)

b. Performing a CHANNEL FUNCTIONAL TEST:

   1. Within 24 hours before the start of CORE ALTERATIONS, and
   2. At least once per 7 days.

c. Verifying that the channel count rate is at least 3 cps*

   1. Before control rod withdrawal,

   2. Before and at least once per 12 hours during CORE ALTERATIONS, and

   3. At least once per 24 hours,

Except that:

   1. During complete core spiral offloading, the SRM count rate need not be maintained when the fuel assemblies around the SRM are removed.

   2. Prior to and during complete core spiral reloading, the required count rate may be achieved by:

      a) Use of a portable external source, or

      b) Loading up to 4 fuel assemblies in cells containing inserted control rods around an SRM.

d. Verifying, within 8 hours before and at least once per 12 hours during the time any control rod is withdrawn that the shorting links have been removed from the RPS circuitry, unless adequate shutdown margin has been demonstrated per Specification 3.1.1 and the "one rod out" interlock is OPERABLE per Specification 3.9.1.

* The count rate may be less than 3 cps if the following conditions are met: (1) the signal-to-noise ratio is greater than or equal to 5, and (2) the signal is greater than 1.3 cps.

3/4.9.3 CONTROL ROD POSITION

LIMITING CONDITIONS FOR OPERATION

3.9.3 All control rods shall be fully inserted.*

APPLICABILITY: OPERATING CONDITION 5 when loading fuel assemblies into the core.

ACTION:

With one or more control rods not fully inserted, suspend loading fuel assemblies into the core.

SURVEILLANCE REQUIREMENTS

4.9.3 All control rods shall be verified to be fully inserted at least once per 12 hours during loading of fuel assemblies into the core.

* Except control rods removed per Specification 3.9.10.1 or 3.9.10.2 or with one control rod withdrawn under control of reactor mode switch Refuel position one-rod-out interlock.

REFUELING OPERATIONS

3/4.9.4 DECAY TIME

LIMITING CONDITIONS FOR OPERATION

3.9.4 The reactor shall be subcritical for at least 24 hours.

APPLICABILITY: OPERATIONAL CONDITION 5, during movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 24 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.4 The reactor shall be determined to have been subcritical for at least 24 hours by verification of the date and time of subcriticality before movement of irradiated fuel in the reactor pressure vessel.

REFUELING OPERATIONS

3/4.9.5 COMMUNICATIONS

LIMITING CONDITIONS FOR OPERATION

3.9.5 Direct communication shall be maintained between the control room and refueling floor personnel.

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.

ACTION:

When direct communication between the control room and refueling floor personnel cannot be maintained, immediately suspend CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communication between the control room and refueling floor personnel shall be demonstrated within 1 hour before the start of and at least once per 12 hours during CORE ALTERATIONS.

3/4.9.6 REFUELING PLATFORM

LIMITING CONDITIONS FOR OPERATION

3.9.6 The refueling platform shall be OPERABLE and used for handling fuel assemblies or control rods within the reactor pressure vessel.

APPLICABILITY: During handling of fuel assemblies or control rods within the reactor pressure vessel.

ACTION:

With the requirements for refueling platform OPERABILITY not satisfied, suspend use of any inoperable refueling platform equipment from operations involving the handling of control rods and fuel assemblies within the reactor pressure vessel after placing the load in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.6 Each refueling platform crane or hoist used for handling of control rods or fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 7 days before the start of such operations with that crane or hoist by:

a. Demonstrating operation of the overload cutoff on the main hoist when the load exceeds 1600 +100/-0 pounds.

b. Demonstrating operation of the overload cutoff on the frame mounted and monorail mounted auxiliary hoists when the load exceeds 1000 ± 50 pounds.

c. Demonstrating operation of the main and auxiliary hoist uptravel stops when the grapple is lower than or equal to 7 feet 3 3/4 inches below the platform tracks.

d. Demonstrating operation of the downtravel mechanical cutoff on the main hoist when grapple hook down travel reaches 4 inches below fuel assembly handle.

e. Demonstrating operation of the slack cable cutoff on the main hoist when the load is less than 50 ± 10 pounds.

f. Demonstrating operation of the loaded interlock on the main hoist when the load exceeds 700 +50/-0 pounds.

g. Demonstrating operation of the redundant loaded interlock on the main hoist when the load exceeds 700 +50/-0 pounds.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL

LIMITING CONDITIONS FOR OPERATION

3.9.7 Loads in excess of 1000 pounds shall be prohibited from travel over fuel assemblies in the spent fuel storage pool racks unless handled by a single failure-proof handling system.

APPLICABILITY: With fuel assemblies in the spent fuel storage pool racks.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7.1 Crane interlocks that prevent crane travel over fuel assemblies in the spent fuel storage pool racks shall be demonstrated OPERABLE within 7 days before and at least once per 7 days during crane operation.

4.9.7.2 The single failure-proof lifting devices shall be visually inspected and verified OPERABLE within 7 days prior to and at least once per 7 days during polar crane operation over the spent fuel pool.

REFUELING OPERATIONS

3/4.9.8 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.8 At least 22 feet 3 inches of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During handling of fuel assemblies or control rods within the reactor pressure vessel while in OPERATIONAL CONDITION 5 when the fuel assemblies being handled are irradiated or the fuel assemblies seated within the reactor vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving handling of fuel assemblies or control rods within the reactor pressure vessel after placing all fuel assemblies and control rods in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.8 The reactor vessel water level shall be determined to be at least its minimum required depth within 2 hours before the start of and at least once per 24 hours during handling of fuel assemblies or control rods within the reactor pressure vessel.

3/4.9.9 WATER LEVEL - SPENT FUEL STORAGE POOL

LIMITING CONDITIONS FOR OPERATION

3.9.9 At least 22 feet 3 inches of water shall be maintained over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel storage pool.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel storage pool area after placing the fuel assemblies and crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The water level in the spent fuel storage pool shall be determined to be at least at its minimum required depth at least once per 7 days.

REFUELING OPERATIONS

3/4.9.10 CONTROL ROD REMOVAL

SINGLE CONTROL ROD REMOVAL

LIMITING CONDITIONS FOR OPERATION

3.9.10.1 One control rod and/or the associated control rod drive mechanism may be removed from the core and/or reactor pressure vessel provided that at least the following requirements are satisfied until a control rod and associated control rod drive mechanism are reinstalled and the control rod is fully inserted in the core.

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Table 1.2 and Specification 3.9.1.

b. The source range monitors (SRM) are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied, except that the control rod selected to be removed

   1. May be assumed to be the highest worth control rod required to be assumed to be fully withdrawn by the SHUTDOWN MARGIN test, and

   2. Need not be assumed to be immovable or untrippable.

d. All other control rods in a 5 x 5 array centered on the control rod being removed are inserted and electrically or hydraulically disarmed or the four fuel assemblies surrounding the control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

e. All other control rods are inserted.

APPLICABILITY: OPERATIONAL CONDITIONS 4 and 5.

ACTION:

With the requirements of the above specification not satisfied, suspend removal of the control rod and/or associated control rod drive mechanism from the core and/or reactor pressure vessel and initiate action to satisfy the above requirements.

CONTROL ROD REMOVAL

SINGLE CONTROL ROD REMOVAL

SURVEILLANCE REQUIREMENTS

4.9.10.1 Within 4 hours before the start of removal of a control rod and/or the associated control rod drive mechanism from the core and/or reactor pressure vessel and at least once every 24 hours thereafter until a control rod and associated control rod drive mechanism are reinstalled and the control rod is inserted in the core, verify that:

a. The reactor mode switch is OPERABLE per Surveillance Requirement 4.3.1.1 or 4.9.1.2, as applicable, and locked in the Shutdown position or in the Refuel position with the "one rod out" Refuel position interlock OPERABLE per Specification 3.9.1.

b. The SRM channels are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied per Specification 3.9.10.1.c.

d. All other control rods in a 5 x 5 array centered on the control rod being removed are inserted and electrically or hydraulically disarmed or the four fuel assemblies surrounding the control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

e. All other control rods are inserted.

REFUELING OPERATIONS

CONTROL ROD REMOVAL

MULTIPLE CONTROL ROD REMOVAL

LIMITING CONDITIONS FOR OPERATION

3.9.10.2 Any number of control rods and/or control rod drive mechanisms may be removed from the core and/or reactor pressure vessel provided that at least the following requirements are satisfied until all control rods and control rod drive mechanisms are reinstalled and all control rods are inserted in the core.

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Specification 3.9.1, except that the Refuel position "one-rod-out" interlock may be bypassed, as required, for those control rods and/or control rod drive mechanisms to be removed, after the fuel assemblies have been removed as specified below.

b. The source range monitors (SRMs) are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.

d. All other control rods are either inserted or have the surrounding four fuel assemblies removed from the core cell.

e. The four fuel assemblies surrounding each control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

f. All fuel loading operations have been suspended.*

APPLICABILITY: OPERATIONAL CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, suspend removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and initiate action to satisfy the above requirements.

* Except during complete core spiral reload where the shorting links shall be removed and dedicated procedures shall be strictly followed.

MULTIPLE CONTROL ROD REMOVAL

SURVEILLANCE REQUIREMENTS

4.9.10.2.1 Within 4 hours before the start of removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and at least once per 24 hours thereafter until all control rods and control rod drive mechanisms are reinstalled and all control rods are inserted in the core, verify that:

a. The reactor mode switch is OPERABLE per Surveillance Requirement 4.3.1.1 or 4.9.1.2, as applicable, and locked in the Shutdown position or in the Refuel position per Specification 3.9.1

b. The SRM channels are OPERABLE per Specification 3.9.2

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.

d. All other control rods are either inserted or have the surrounding four fuel assemblies removed from the core cell. .

e. The four fuel assemblies surrounding each control rod and/or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

f. All fuel loading operations have been suspended.*

4.9.10.2.2 Following replacement of all control rods and/or control rod drive mechanisms removed in accordance with this specification, perform a functional test of the "one-rod-out" Refuel position interlock, if this function had been bypassed.

* Except during complete core spiral reload where the shorting links shall be removed and dedicated procedures shall be strictly followed.

REFUELING OPERATIONS

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITIONS FOR OPERATION

3.9.11.1 At least one shutdown cooling mode loop of the residual heat removal (RHR) system shall be OPERABLE and in operation* with at least:

a. One OPERABLE RHR pump, and
b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is greater than or equal to 22 feet 3 inches above the top of the reactor pressure vessel flange.

ACTION:

a. With no RHR shutdown cooling mode loop OPERABLE, within 1 hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method of decay heat removal. Otherwise, suspend all operations involving an increase in the reactor decay heat load and establish SECONDARY CONTAINMENT INTEGRITY within 4 hours.

b. With no RHR shutdown cooling mode loop in operation, within 1 hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature at least once per hour.

SURVEILLANCE REQUIREMENTS

4.9.11.1 At least one shutdown cooling mode loop of the residual heat removal system or alternate method shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

* The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period.

LOW WATER LEVEL

LIMITING CONDITIONS FOR OPERATION

3.9.11.2 Two shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE and at least one loop shall be in operation,* with each loop consisting of at least:

a. One OPERABLE RHR pump, and
b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is less than 22 feet 3 inches above the top of the reactor pressure vessel flange.

ACTION:

a. With less than the above required shutdown cooling mode loops of the RHR system OPERABLE, within 1 hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternative method capable of decay heat removal for each inoperable RHR shutdown cooling mode loop.

b. With no RHR shutdown cooling mode loop in operation, within 1 hour establish reactor coolant circulation by an alternative method and monitor reactor coolant temperature at least once per hour.

SURVEILLANCE REQUIREMENTS

4.9.11.2 At least one shutdown cooling mode loop of the residual heat removal system, or alternate method shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

* The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period.

Question # SRO 90

Examination Outline Cross-Reference	Level	SRO
	Tier #	-
	Group #	-
	K/A #	Generic 2.2.17
	Importance Rating	3.5

Knowledge of the process for managing maintenance activities during power operations.

Proposed Question:

The unit is operating at 100% power. Switchyard maintenance is in progress and will be complete in 24 hours. Maintenance also desires to work on any one of the following this shift: **EDG1, RCIC, LPCS, Div. I Battery.**

To comply with GAP-PSH-03, Control of On-Line Work Activities, which one of the maintenance activities above could be approved to work this shift without introducing a higher than usual risk?

a. Removal of EDG1 from service.

b. Removal of RCIC from service.

c. Removal of LPCS from service.

d. Removal of Div. I Battery from service.

Proposed Answer: c.

Explanation (Justification of Distractors):
If plant activities introduce a higher than usual risk of an initiating event such as a loss of offsite power, SSCs that perform key safety functions such as diesel generators, RCIC, and batteries, should not be removed from service. 2BYS*PNL204B would make the Div II diesel generator inoperable.

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Technical Reference(s): GAP-PSH-03, Rev 02, 3.2.5 and 3.3.1.d.

Proposed references to be provided to applicants during the examination:

None.

Learning Objective: O2-OPS-006-343-3-01, EO-3, EO-5

Question Source:	Bank #	
	Modified Bank #	
	New	New
Question History:	Previous NRC Exam	New
	Previous Test / Quiz	New
Question Cognitive Level:	Memory of Fundamental Knowledge	
	Comprehension or Analysis	2
10CFR Part 55 Content:	55.43.5	
	55.45.13	

Comments: SRO only: Control of maintenance activities.

Question # SRO 91

Examination Outline Cross-Reference	Level	SRO
	Tier #	-
	Group #	-
	K/A #	2.2.6
	Importance Rating	3.3
Knowledge of the process for making changes in procedures as described in the safety analysis report.		

Proposed Question:

A Type 1 change to N2-OP-101A, Plant Startup, that does NOT alter the intent of the procedure is requested. Which one of the following satisfies the approval requirements to implement the temporary change?

a. Only the CRS or SSS approve the change.

b. The CSO **and** the SSS approve the change.

c. Any two members of the management staff approve the change.

d. The CRS **and** a member of management staff approve the change.

Proposed Answer: d.

Explanation (Justification of Distractors):
The intent of the procedure is NOT altered. The change is approved by two members of the unit management staff, at least one of whom holds a Senior Operator license on the unit affected.

The change is documented, reviewed, and approved within 14 days of implementation by the branch manager or higher levels of management. Not required to answer the questions because it asks the approvals to implement the change.

a. Another member of the management staff must also approve the change.
b. The CSO is not a member of the management staff.
c. One of the members of the management staff must be a SRO

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Technical Reference(s): Tech Spec 6.8.3

Proposed references to be provided to applicants during the examination:

None

Learning Objective: O2-OPS-006-343-3-01, EO-5, EO-6

Question Source:	Bank # Modified Bank # New	 New
Question History:	Previous NRC Exam Previous Test / Quiz	New New
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	1
10CFR Part 55 Content:	55.43.3 55.45.12	

Comments: SRO only: Change process for procedures described in the USAR. Approval requirements for a temporary procedure change, which is an application of Technical Specification Administrative Requirements.

S28-2

Question # SRO 92

Examination Outline Cross-Reference	Level	SRO
	Tier #	-
	Group #	-
	K/A #	Generic 2.2.23
	Importance Rating	3.8
Ability to track limiting conditions for operations.		

Proposed Question:

A short term Limiting Condition for Operation (LCO) on RHR loop A is entered to support surveillance testing during the shift. The testing is completed and RHR Loop A is restored to OPERABLE status prior to the end of the shift.

Which one of the following describes where the short term LCO is tracked including the information that is required to be entered?

a. Only the date and time of action statement entry are entered in the SSS log.

b. Only the date and time of action statement entry are entered into the SSS log and the ESL log.

c. The date and time of action statement entry and the actions taken are only entered in the SSS log.

d. The date and time of action statement entry and the actions taken are entered into the SSS log and the ESL log.

Proposed Answer: c.

Explanation (Justification of Distractors):

a. The T.S. actions must also be entered.
b. Equipment Status Log entry is only made for LCOs that have a duration longer than 1 shift. The T.S. actions must also be entered.
d. Equipment Status Log entry is only made for LCOs that have a duration longer than 1 shift.

529-1

Technical Reference(s):	Conduct of Operations Manual, Section 3.7.5 GAP-OPS-01, Rev 11, Section 3.10.3

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-006-343-3-01, EO-3, EO-4, EO-5,	
Question Source:	Bank # Modified Bank # New	 New
Question History:	Previous NRC Exam Previous Test / Quiz	New New
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	1
10CFR Part 55 Content:	55.43.2 55.45.13	

Comments: SRO only: LCO tracking/logging

Question # SRO 93

Examination Outline Cross-Reference	Level	SRO
	Tier #	-
	Group #	-
	K/A #	Generic 2.3.4
	Importance Rating	3.1

Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.

Proposed Question:

A station operator has an accumulated TEDE of 3800 mrem for the year. Because of dose projections during the assigned outage work, the individual is expected to receive an additional TEDE of 300 mrem.

In accordance with S-RAP-RPP-0703, Authorization to Exceed Administrative Dose Limits, which one of the following describes the **final** authorization required for the worker to receive the expected dose?

a. Plant Manager

b. Outage Manager

c. Unit ALARA Manager

d. Site Vice President – Nuclear

Proposed Answer: a.

Explanation (Justification of Distractors):
The plant manager provides the final review and approval of requests to increase individual dose limits.

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Technical Reference(s): S-RAP-RPP-0703

Proposed references to be provided to applicants during the examination:

None.

Learning Objective: O2-OPS-006-343-3-01, EO-6

Question Source:	Bank #	
	Modified Bank #	
	New	New
Question History:	Previous NRC Exam	New
	Previous Test / Quiz	New
Question Cognitive Level:	Memory of Fundamental Knowledge	1
	Comprehension or Analysis	
10CFR Part 55 Content:	55.43.4	
	55.45.10	

Comments: SRO only: Authorization to exceed station administrative dose requirements.

S30-2

Question # SRO 97

Examination Outline Cross-Reference	Level	SRO
	Tier #	-
	Group #	-
	K/A #	Generic 2.4.1
	Importance Rating	4.6
Knowledge of EOP entry conditions and immediate action steps.		

Proposed Question:

A unit startup is in progress. The first Reactor Feedwater Pump was just placed into service when the "A" CRD pump trips. The initial attempt to start the "B" CRD pump is unsuccessful.

Which one of the following conditions requires that the reactor be scrammed per N2-SOP-101C, Reactor Scram?

a. Neither CRD pump can be started within 20 minutes.

b. Seal cooling cannot be aligned to the WCS pumps from CCP.

c. Reference leg backfill is secured to RPV instrumentation for greater than 20 minutes.

d. Accumulator pressure is verified at 930 psig for a control rod at position 04.

Proposed Answer: d.

Explanation (Justification of Distractors):

a. This requires a scram provided that more than one accumulator is inoperable.

b. This requires that the running WCS pump is tripped and not a reactror scram.

c. Instrumentation operability becomes a concern when secured for 26 hours. The corrective action is to flush the reference legs.

Technical Reference(s): N2-SOP-30, Rev 00, Section 3.0

Proposed references to be provided to applicants during the examination:

None.

Learning Objective:	O2-OPS-006-SOP-2-01-29, TO-13, EO-2	
Question Source:	Bank #	
	Modified Bank #	
	New	New
Question History:	Previous NRC Exam	New
	Previous Test / Quiz	New
Question Cognitive Level:	Memory of Fundamental Knowledge	
	Comprehension or Analysis	2
10CFR Part 55 Content:	55.41.10	
	55.43.5	
	55.45.13	

Comments: SRO only: SRO actions to direct a plant scram based on Technical Specifications requirements.

531-2

Question # SRO 99

Examination Outline Cross-Reference	Level	SRO
	Tier #	-
	Group #	-
	K/A #	Generic 2.4.19
	Importance Rating	3.7
Knowledge of EOP layout / symbols / and icons.		

Proposed Question:

Which one of the following symbols is used to identify a "conditional step" in N2-EOP-PC, Primary Containment Control?

a. Diamond.

b. Octagon.

c. Rectangle with single lines.

d. Rounded rectangle with double lines.

Proposed Answer: a.

Explanation (Justification of Distractors):

b. Hold points are formatted as octagons.

c. Contingency actions, contingency steps, and decision tables are formatted as rectangles.

d. Overrides are formatted as rounded rectangles with double lines.

S32-1

Technical Reference(s):	N2-ODP-PRO-0301, Rev 02, Section 4 NMP2-EOP-Basis Document, Section B

Proposed references to be provided to applicants during the examination:

ALL EOPs with the entry conditions blacked out.

Learning Objective:	O2-OPS-006-344-2-04, # 2	
Question Source:	Bank # Modified Bank # New	 New
Question History:	Previous NRC Exam Previous Test / Quiz	New New
Question Cognitive Level:	Memory of Fundamental Knowledge Comprehension or Analysis	1
10CFR Part 55 Content:	55.41.10 55.45.13	

Comments: SRO only: EOP use and execution – understanding symbols used on the EOP flowcharts. The RO question is different and evaluates symbols used in N2-EOP-6 Attachments which the ROs use. This question is for the flowcharts which the SRO uses.