CORRECTED **Reactor Operator Answer Key** - 1 RESUBMITTED

1. d
2. b
3. a
4. ~~c~~ Delete
5. b
6. b
7. a
8. b
9. d
10. a
11. b or d
12. a
13. d
14. d
15. c
16. b
17. a
18. a
19. b
20. a
21. ~~b~~ d
22. d
23. a
24. c
25. b
26. c
27. a
28. b
29. d
30. d
31. a
32. a
33. c
34. ~~d~~ Delete
35. c
36. ~~36 b d~~ DELETE
37. d
38. d
39. b
40. c
41. a
42. c
43. d
44. d
45. b
46. b
47. c
48. d
49. a
50. b

51. b
52. c
53. d
54. a
55. ~~a~~ Delete
56. b
57. b
58. d
59. ~~d~~ Delete
60. a
61. d
62. a
63. c
64. a
65. d
66. a
67. c
68. d
69. b
70. c
71. b
72. c
73. a
74. c
75. a
76. d
77. b
78. d
79. d
80. a
81. c
82. c
83. d
84. a
85. c
86. a
87. d
88. d
89. c
90. c
91. c
92. ~~a~~ Delete
93. c
94. b
95. b
96. b
97. b
98. c or a
99. b
100. ~~a~~ Delete

Corrected Reactor Operator Answer Key - 2 hour CERT

1. d
2. b
3. a
4. ~~c~~ Delete
5. b
6. b
7. a
8. b
9. d
10. a
11. b or d
12. a
13. d
14. d
15. c
16. b
17. a
18. a
19. b
20. a
21. ~~b~~ d
22. d
23. a
24. c
25. b
26. c
27. a
28. b
29. d
30. d
31. a
32. a
33. c
34. ~~d~~ Delete
35. c
36. ~~36. b d~~ DELETE
37. d
38. d
39. b
40. c
41. a
42. c
43. d
44. d
45. b
46. b
47. c
48. d
49. a
50. b

51. b
52. c
53. d
54. a
55. ~~a~~ Delete
56. b
57. b
58. d
59. ~~d~~ Delete
60. a
61. d
62. a
63. c
64. a
65. d
66. a
67. c
68. d
69. b
70. c
71. b
72. c
73. a
74. c
75. a
76. d
77. b
78. d
79. d
80. a
81. c
82. c
83. d
84. a
85. c
86. a
87. d
88. d
89. c
90. c
91. c
92. ~~a~~ Delete
93. c
94. ~~b~~ Delete
95. b
96. b
97. b
98. c or a
99. b
100. ~~a~~ Delete

CORRECTED Senior Reactor Operator Answer Key FOUR STATES

1. d
2. b
3. a
4. a
5. b
6. b
7. b
8. a
9. b
10. d
11. a
12. b or d
13. a
14. a
15. c
16. d
17. c
18. b
19. a
20. c
21. c
22. a
23. b
24. d
25. b
26. d
27. c
28. ~~b~~ d
29. d
30. a
31. c
32. b
33. a
34. b
35. d
36. d
37. d
38. a
39. b
40. a
41. c
42. c
~~43. b~~ ~~d~~ DELETE
44. d
45. c
46. a
47. d
48. c
49. d
50. b

Senior Reactor Operator Answer Key

51 . c
52 . d
53 . a
54 . b
55 . c
56 . b
57 . b
58 . d
59 . a
60 . ~~d~~ Delete
61 . a
62 . d
63 . a
64 . c
65 . c
66 . a
67 . c
68 . b
69 . c
70 . a
71 . b
72 . c
73 . b
74 . d
75 . b
76 . d
77 . a
78 . c
79 . b
80 . c
81 . d
82 . a
83 . d
84 . d
85 . d
86 . c
87 . c
88 . a
89 . c
90 . d
91 . b
92 . a
93 . b
94 . a
95 . d
96 . b
97 . c or a
98 . b
99 . b
00 . ~~a~~ Delete

CORRECTED **Senior Reactor Operator Answer Key** -2 FOUR LEFT

1. d
2. b
3. a
4. a
5. b
6. b
7. b
8. a
9. b
10. d
11. a
12. b or d
13. a
14. a
15. c
16. d
17. c
18. b
19. a
20. c
21. c
22. a
23. b
24. d
25. b
26. d
27. c
28. ~~b~~ d
29. d
30. a
31. c
32. b
33. a
34. b
35. d
36. d
37. d
38. a
39. b
40. a
41. c
42. c
~~43. b d~~ DELETE
44. d
45. c
46. a
47. d
48. c
49. d
50. b

51 . c

52 . d

53 . a

54 . b

55 . c

56 . b

57 . b

58 . d

59 . a

60 . ~~d~~ DeleTe

61 . a

62 . d

63 . a

64 . c

65 . c

66 . a

67 . c

68 . b

69 . c

70 . a

71 . b

72 . c

73 . b

74 . d

75 . b

76 . d

77 . a

78 . c

79 . b

80 . c

81 . d

82 . a

83 . d

84 . d

85 . d

86 . c

87 . c

88 . a

89 . c

90 . d

91 . b

92 . a

93 . ~~b~~ DeleTe

94 . a

95 . d

96 . b

97 . c oR a

98 . b

99 . b

00 . ~~a~~ DeleTe

Question Cross Reference

As Given Exam Key
3/18/02

KA		Record Number	Exam Level	RO	SRO
295001	AA1.02	1	B	1	1
295002	AK1.04	2	B	2	2
295003	AA1.03	3	B	3	3
295003	2.4.9	4	R	4	
295004	AA2.01	5	S		4
295004	AK3.03	6	B	5	5
295005	AA2.04	7	S		6
295006	2.1.28	8	B	6	7
295006	AK1.01	9	B	7	8
295007	AK2.05	10	B	8	9
295007	AK3.04	11	B	9	10
295008	AA1.01	12	B	10	11
295008	AK3.04	13	B	11	12
295009	AA2.01	14	B	12	13
295009	2.4.6	15	S		14
295010	AA1.02	16	S		15
295010	AA1.02	17	R	13	
295012	AK1.01	18	B	14	16
295013	AK2.01	19	B	15	17
295014	AK2.04	20	B	16	18
295014	AK3.01	21	B	17	19
295015	2.3.4	22	S		20
295015	AK3.01	23	S		21
295016	AA1.02	24	B	18	22
295017	AA2.01	25	S		23
295018	AA2.03	26	S		24
295019	AA1.02	27	B	19	25
295019	AA2.01	28	R	20	
295021	2.4.41	29	S		26
295022	2.4.48	30	S		27
295022	AK2.03	31	B	21	28
295023	AA1.02	32	B	22	29
295023	2.4.11	33	B	23	30
295024	EA1.10	34	B	24	31

KA		Record Number	Exam Level	RO	SRO
295024	2.1.6	35	B	25	32
295025	EA2.06	36	R	26	
295025	EK1.05	37	B	27	33
295026	EK1.02	38	B	28	34
295028	EK1.02	39	B	29	35
295030	EA2.04	40	S		36
295030	EK2.03	41	B	30	37
295031	EK2.13	42	B	31	38
295034	2.4.30	43	S		39
295036	EK2.01	44	B	32	40
295036	EK3.01	45	B	33	41
295038	EA2.03	46	R	34	
295038	EK1.02	47	B	35	42
500000	EK3.03	48	B	36	43
600000	2.4.25	49	R	37	
201001	A3.05	50	B	38	44
201002	2.4.21	51	R	39	
201002	K4.08	52	B	40	45
201003	K4.05	53	B	41	46
201006	2.1.12	54	S		47
201006	K3.01	55	B	42	48
202001	A4.04	56	R	43	
202001	K3.07	57	B	44	49
202002	K6.04	58	R	45	
203000	A4.07	59	B	46	50
203000	K1.14	60	B	47	51
204000	A2.14	61	B	48	52
206000	A1.06	62	B	49	53
206000	A3.07	63	R	50	
209001	K1.10	64	B	51	54
209001	K2.02	65	B	52	55
211000	2.4.10	66	R	53	
211000	K1.05	67	R	54	
212000	2.1.23	68	R	55	
212000	K5.02	69	B	56	56

KA		Record Number	Exam Level	RO	SRO
215001	K1.05	70	B	57	57
215004	A3.03	71	B	58	58
215004	2.2.6	72	S		59
215005	K3.05	73	B	59	60
215005	K5.05	74	B	60	61
216000	A2.08	75	B	61	62
216000	K2.01	76	B	62	63
217000	A2.01	77	S		64
217000	K4.05	78	B	63	65
219000	A3.01	79	R	64	
223001	K6.13	80	R	65	
223002	K3.16	81	B	66	66
223002	K4.01	82	B	67	67
226001	A1.06	83	R	68	
226001	A3.05	84	B	69	68
230000	K6.01	85	B	70	69
233000	2.1.7	86	S		70
234000	2.2.25	87	S		71
239002	A1.02	88	R	71	
239002	A1.05	89	B	72	72
245000	K5.02	90	R	73	
256000	A2.13	91	R	74	
256000	K4.06	92	R	75	
259002	2.4.32	93	S		73
261000	A4.07	94	B	76	74
261000	K6.03	95	B	77	75
262001	K2.01	96	R	78	
262001	K6.01	97	B	79	76
263000	A1.01	98	B	80	77
263000	K2.01	99	B	81	78
264000	2.1.11	100	S		79
268000	A1.01	101	B	82	80
271000	K1.02	102	B	83	81
272000	K6.03	103	B	84	82
290002	A2.02	104	R	85	

KA		Record Number	Exam Level	RO	SRO
290002	K3.03	105	R	86	
290003	K5.01	106	B	87	83
GENERIC	2.1.3	107	B	88	84
GENERIC	2.1.14	108	S		85
GENERIC	2.1.24	109	B	89	86
GENERIC	2.1.33	110	S		87
GENERIC	2.1.33	111	R	90	
GENERIC	2.1.34	112	S		88
GENERIC	2.2.22	113	R	91	
GENERIC	2.2.22	114	S		89
GENERIC	2.2.26	115	S		90
GENERIC	2.2.27	116	R	92	
GENERIC	2.2.27	117	S		91
GENERIC	2.2.30	118	R	93	
GENERIC	2.2.31	119	S		92
GENERIC	2.3.1	120	B	94	93
GENERIC	2.3.2	121	R	95	
GENERIC	2.3.4	122	S		94
GENERIC	2.3.9	123	R	96	
GENERIC	2.3.10	124	S		95
GENERIC	2.3.11	125	B	97	96
GENERIC	2.4.5	126	B	98	97
GENERIC	2.4.18	127	B	99	98
GENERIC	2.4.28	128	S		99
GENERIC	2.4.34	129	B	100	100

Given the following:

- The plant is operating at 100% power
- A transient caused by a short in the reactor recirculation control circuitry occurs

Immediately following the transient, the plant stabilizes with the following parameters:

- Reactor Power 50%
- "A" Recirc pump tripped
- "B" Recirc pump at 45% speed
- Loop "A" total jet pump flow is 10 Mlbmlhr
- Loop "B" total jet pump flow is 46 Mlbmlhr
- Total indicated core flow 36 Mlbm/hr

What is actual core flow, and how will the loss of the "A" Recirc pump affect the APRM Scram setpoint?

a. 36 Mlbm/hr. Setpoint unaffected

b. 36 Mlbm/hr. Setpoint needs to be adjusted

c. 56 Mlbm/hr. Setpoint unaffected

d. 56 Mlbm/hr. Setpoint needs to be adjusted

Answer d **Exam Level** B **Cognitive Level** Comprehension **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 2 295001A102

295001 Partial or Complete Loss of Forced Core Flow Circulation **Record Number** 1

AA1. Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION:

AA1.02 RPS 3.3 3.3

Explanation of Answer Below 48% running recirc loop speed, Jet pump loop flows are both positive and added together. Setpoints must be adjusted to single loop values within 4 hours

Reference Title

HC.OP-AB.ZZ-0300

HC.OP-DL.ZZ-0026 Attach 3V

TS 2.2.1 and 3.4.1

Learning Objectives

0AB300E003 (R) Discuss the operational implications of the abnormal indications/alarms for system operating parameters related to Reactor Power Oscillations, Abnormal Operating Procedure.

Material Required for Examination

Question Source: INPO Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: INPO EXAM BANK QID# 17049 Susquehanna 1 09/30/1999

The plant has been operating at full power for several days.

- Operators notice that, over the last several hours, Main Condenser Vacuum has risen from 3.2"HgA to 4.0"HgA.
- Over this same period, Offgas system flow has increased from 25 scfm to 38 scfm.
- There have been NO ALARMS associated with this problem.

Which one of the following would cause these indications?

a. Cooling tower outlet temperature increase

b. Reactor Feed Pump Turbine exhaust piping leak

c. Tube leak in #2A Feedwater Heater

d. Resin intrusion from the Condensate Demineralizers

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 2 295002K104

295002 Loss of Main Condenser Vacuum **Record Number** 2

AK1. Knowledge of the operational implications of the following concepts as they apply to LOSS OF MAIN CONDENSER VACUUM:

AK1.04 Increased offgas flow 3.0 3.3

Explanation of Answer Increase in air inleakage via the RFPT exhaust piping under vacuum into the main condenser will cause Offgas outlet flows to increase. Cooling tower outlet temp increase would degrade vacuum but not change outlet flow. 2A Heaters are internal to the main condenser so no change in outlet flow. Resin intrusion causes offgas radiation levels to increase

Reference Title

HC.OP-AB.ZZ-0001

Learning Objectives

0AB208E006 (R) Explain the bases for Subsequent Actions and the information contained in the Discussion Section of Main Condenser Low Vacuum, Abnormal Operating Procedure.

Material Required for Examination

Question Source: INPO Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: INPO BANK QID# 647 Duane Arnold 05/25/1999

Given the following:

- The plant is operating at 100 percent power
- A severe electrical transient results in a loss of all offsite power
- 2 control rods are at position "48"
- Reactor power is less than 1 percent

Which one of the following describes the equipment available to control reactor pressure and level?

a. HPCI and SRVs

b. HPCI and Main Steam Line Drains

c. Reactor Feed Pumps and SRVs

d. RCIC and Main Steam Line Drains

Answer a **Exam Level** B **Cognitive Level** Comprehension **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 1 295003A103

295003 Partial or Complete Loss of A.C. Power **Record Number** 3

AA1. Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER:

AA1.03 Systems necessary to assure safe plant shutdown 4.4 4.4

Explanation of Answer Although the reactor is not shutdown, power is less than 4%. The candidate must determine HPCI injection is allowed under EOP-101A PREFERRED ATWS INJECTION SYSTEMS TABLE 1 using EOP-322 as necessary. Loss of offsite power causes Group 1 isln. MSL Drains will close if the valves have power. RFPT oil pumps can be restored from EDG backed busses, but condensate pumps are tripped.

Reference Title

HC.OP-AB.ZZ-0135

EOP-101A

Learning Objectives

0AB135E006 (R) Explain the bases for Subsequent Actions and the information contained in the Discussion Section of Station Blackout/Loss Of Offsite Power Diesel Generator Malfunction, Abnormal Operating Procedure.

Material Required for Examination EOP Flowcharts without entry conditions

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following:

- The reactor is in Operational Condition 4
- "A" RHR Pump is in Shutdown Cooling at rated flow
- 10A404 4.16KV 1E Bus trips on bus differential overcurrent

Which one of the following describes the effect the bus loss will have on Shutdown Cooling?

a. The Shutdown Cooling common suction line isolates and CANNOT be reset

b. The AP228 Jockey pump trips causing Shutdown Cooling Loop "A" to lose keepfill

c. Both "A" and "B" Shutdown Cooling Loops lose ability to adjust flow

d. "B" Reactor Recirc Pump discharge valve automatically opens bypassing core flow

Answer c **Exam Level** R **Cognitive Level** Comprehension **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 1 295003G409

295003 Partial or Complete Loss of A.C. Power **Record Number** 4

2.4 Emergency Procedures and Plan

2.4.9 Knowledge of low power / shutdown implications in accident (e.g. LOCA or loss of RHR) mitigation strategies. 3.3 3.9

Explanation of Answer "A" RHR SDC Return valve F015A is powered from "D" Channel 1E 480VAC. Loss of D Bus fails this valve as is. Adjusting flow via RHR HX outlet valve and /or bypass valve is not proceduralized. AP228 provides keepfill to HPCI only. B RRP disch valve is controlled by NON 1E power.

Reference Title

HC.OP-SO.BC-0002

HC.OP-SO.SM-0001

Learning Objectives

000028E008 (R) Given a system which physically connects to or is required to support the operation of the RHR System or components therein, explain the function of the supporting system, IAW the RHR System Lesson Plan.

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Which one of the following conditions will automatically remove the 125 VDC battery charger from service per HC.OP-AB.ZZ-0150, 125VDC System Malfunction?

a. High output voltage

b. Equalize timer reaches zero

c. Blown fuse in the battery transfer switch

d. Low battery terminal voltage

Answer a **Exam Level** S **Cognitive Level** Memory **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 2 295004A201

295004 Partial or Complete Loss of D.C. Power **Record Number** 5

AA2. Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER:

AA2.01 Cause of partial or complete loss of D.C. power 3.2 3.6

Explanation of Answer

SRO UNIQUE - RO LEVEL QUESTION
The following battery charger malfunctions will shutdown the battery charger:
High Voltage Shutdown Relay
AC Input Breaker Open/Tripped
DC Output Breaker Open/Tripped
Loss of 120 VAC Supply Power

CORRECT - High Voltage Shutdown Relay.
INCORRECT - Low battery terminal voltage. This will generate a Battery Monitor Alarm not a charger trip.
INCORRECT - High Voltage Shutdown Relay. This will generate a Battery Monitor Alarm not a charger trip.
INCORRECT - Blown fuse in the battery transfer switch. This will generate a Battery Monitor Alarm, not a charger trip.

Reference Title

HC.OP-AB.ZZ-0150

Learning Objectives

0AB150E006 (R) Explain the bases for Subsequent Actions and the information contained in the Discussion Section of 125 VDC System Malfunction, Abnormal Operating Procedure.

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments: Concept used from Vision Bank QID# Q61703 for 24 VDC chargers

Given the following:

- The Reactor is in Operational Condition 4
- Plant startup operations are in progress
- The negative battery charger for the "A" ±24 VDC System is out of service
- The positive battery charger for the "B" ±24 VDC System is on an equalizing charge
- All other equipment is aligned for normal operation

Which one of the following will occur if these conditions remain for a prolonged period of time?

An RPS trip will occur due to:

a. A and C SRMs fail upscale because of low voltage to the drawers

b. A, C, E, and G IRMs fail upscale because of low voltage to the drawers

c. B and D LPRMs fail upscale because of high voltage to the detectors

d. B, D, and F APRMs fail upscale because of high voltage to the detectors

Answer b **Exam Level** B **Cognitive Level** Memory **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 2 295004K303

295004 Partial or Complete Loss of D.C. Power **Record Number** 6

AK3. Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER:

AK3.03 Reactor SCRAM: Plant-Specific 3.1 3.5

Explanation of Answer JUSTIFICATION:

The negative charger only charges the negative battery while the positive charger only charges the positive battery. Even with the positive charger operating in the Equalizer mode, the negative battery will be discharged resulting in the loss of the DC bus.

CORRECT - IRMs upscale (1/2 scram). The loss of the -24VDC from the A ± 24VDC System will cause IRM indications to rise (upscale). This will insert a 1/2 scram from RPS Channel A.
INCORRECT - SRMs upscale (Full Scram). SRM indications to lower (downscale)
INCORRECT - LPRMs upscale (Full Scram). LPRMs and APRMs are unaffected by the loss of -24VDC.
INCORRECT - APRMs upscale (1/2 scram). LPRMs and APRMs are unaffected by the loss of -24VDC.

Reference Title

HC.OP-AB.ZZ-0151, Sections 2.1, 4.5 & 5.1

H.C. Incident Report 86-067, CD-1826, PTS-1826

Learning Objectives

0AB151E003 (R) Discuss the operational implications of the abnormal indications/alarms for system operating parameters related to 24 VDC Malfunction, Abnormal Operating Procedure.

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: Vision Bank QID# Q61702

Given the following:

- The plant is operating at 29 percent power
- Overhead Annunciator C5-C2 TCV FAST CLOSURE & MSV TRIP BYP is ILLUMINATED

Then the Main Turbine Generator trips

- All Turbine Bypass valves responded full open
- Overhead Annunciator B3-E5 RPV PRESSURE HI is ILLUMINATED
- Overhead Annunciators C5-A2 & B2 for TCV FAST CLOSURE and MAIN STOP VALVE CLOSURE are ILLUMINATED
- Overhead Annunciators C3-A2, A3, A4, & A5 for REACTOR SCRAM TRIP LOGIC A1, A2, B1,& B2 are EXTINGUISHED

Which one of the following actions is required?

a. Lock the Reactor Mode Switch in Shutdown immediately

b. Reduce reactor pressure below the alarm point within 15 minutes

c. Reduce reactor power by at least 4 percent within 30 minutes

d. Commence a normal shutdown within one hour

Answer b **Exam Level** S **Cognitive Level** Comprehension **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 1 **SRO Group** 2 295005A204

295005 Main Turbine Generator Trip **Record Number** 7

AA2. Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP:

AA2.04 Reactor pressure 3.7 3.8

Explanation of Answer

RPV Pressure is above the Tech Spec LCO of 1020 psig but below the scram setpoint of 1037 psig. The required action is to lower RPV pressure below the TS 3.4.6.2 LCO value which coinsides with the alarm point (1020 psig) within 15 minutes.

Lock the MS in SD immediately - incorrect because RPS alarms are extinguished and all systems functioned properly. An ATWS does not exist.

Reduce power - incorrect because although reducing power will reduce RPV pressure, the time requirement of 15 minutes would not be met. Immediate operator action for AB-202 High RPV pressure is to reduce REACTOR POWER as necessary to clear the RPV PRESSURE HIGH alarm.

Commence a normal SD within one hour - incorrect. The TS action time is 15 minutes, not one hour.

Reference Title

HC.OP-AB.ZZ-0202

Tech Spec 3.4.6.2

10CFR55.43(5)

Learning Objectives

0AB138E004 Explain the reasons for how plant/system parameters respond when implementing, Turbine Generator Trip/Malfunction, Abnormal Operating Procedure.

000106E001 Given the following lists, summarize and explain both the initial response (goes up, down, stays the same) and the long term response of the parameters in List A to the plant transients in List B IAW the Student Handout.

List A
Reactor Power (APRM)
Reactor Power (Surface Heat Flux)
Reactor Pressure (Dome)

Reactor Indicated Water Level
Reactor Indicated Steam Flow
Reactor Actual Steam Flow
Reactor Feedwater Flow
Reactor Core Flow
Reactor Recirculation Loop Flow
SRV Flow

List B
Loss of Feedwater Heating
Feedwater Controller Failing to Maximum Demand
EHC Pressure Sensor Failing High
Generator Load Rejection with Bypass Valves Available
Generator Load Rejection without Bypass Valves Available
Turbine Trip with Bypass Valves Available
Turbine Trip without Bypass Valves Available
MSIV Closure
Loss of Condenser Vacuum
Loss of All Grid Connections
Loss of Feedwater Flow
Trip of One Recirculation Pump
Trip of Both Recirculation Pumps
Recirculation Flow Control Failure - Decreasing Flow
Seizure of One Recirculation Pump
Recirculation Flow Control Failure - Increasing flow

Material Required for Examination Tech Specs without Definitions, Safety Limits, and bases

Question Source: New

Question Modification Method:

Question Source Comments:

Which one of the following is the reason that the reactor operator must wait at least 10 seconds following a reactor scram before attempting a scram reset?

a. To allow reactor water level to recover above the scram setpoint

b. To allow all the control rods to insert fully

c. To allow the Scram Air header to repressurize

d. To allow the Scram Discharge Volume vent and drain valves to cycle

Answer b **Exam Level** B **Cognitive Level** Memory **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 1 **SRO Group** 1 295006G128

295006 SCRAM **Record Number** 8

2.1 Conduct of Operations

2.1.28 Knowledge of the purpose and function of major system components and controls. 3.2 3.3

Explanation of Answer 10 Second time delay is to allow all control rods time to insert full in.

Reference Title

Lesson Plan 0301-000.00H-000022-19

Learning Objectives

000022E007 From memory, state the purpose of the time delay after a scram, IAW the Lesson Plan.

Material Required for Examination

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: INPO EXAM BANK QID# 7112 Duane Arnold 1 06/03/1996

Hope Creek requires an Emergency Depressurization after performing steam cooling in EOP-101 "Reactor Control". All actions required by EOP-202, "Emergency Depressurization", have been taken but only 4 Safety Relief Valves (SRV) can be opened and no other means of depressurization is available.

Which one of the following describes the consequences of this failure?

a. Steam removal rate from the core is NOT adequate to ensure adequate decay heat removal exists.

b. Steam removal rate during a LOCA is NOT adequate to prevent exceeding the Drywell design pressure.

c. The pressure reduction rate will NOT allow low pressure injection systems to inject soon enough to recover level before the core becomes uncovered.

d. The pressure reduction rate will NOT allow low pressure injection systems to inject prior to reaching the Minimum Steam Cooling RPV Water Level.

Answer a **Exam Level** B **Cognitive Level** Memory **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 1 **SRO Group** 1 295006K101

295006 SCRAM **Record Number** 9

AK1. Knowledge of the operational implications of the following concepts as they apply to SCRAM:

AK1.01 Decay heat generation and removal. 3.7 3.9

Explanation of Answer Minimum Number of SRVs Required for Emergency Depressurization (MNSRED) at Hope Creek is 5 SRVs is sufficient to remove all decay heat from the core.

Reference Title

HC-EOP 202 Bases

HC.OP-EO.ZZ-LIMITS-CONV

Learning Objectives

000130E003 (R) Given any step of the procedure, describe the reason for performance of that step and/or expected system response to control manipulations prescribed by that step.

Material Required for Examination EOP Flowcharts without entry conditions

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: QID# 14157 Peach Bottom 2 03/26/2001

Given the following:

- The plant is in Operational Condition 3
- Main Condenser vacuum is broken
- RHR Loop "B" is in Shutdown Cooling
- Reactor level is stable at +35 inches
- Reactor pressure is 50 psig and lowering
- "D" SSW Pump has just tripped
- "B" SSW Pump will NOT start

Which one of the following describes the effect this will have on the plant? (Assume no operator action)

a. The RHR Shutdown Cooling Loop will isolate due to lowering reactor level

b. The RHR Shutdown Cooling Loop will isolate due to increasing reactor pressure

c. "B" RHR Pump Min-Flow valve will open due to lowering loop flow

d. "B" RHR Pump Min-Flow valve will open due to reaching pump shutoff head

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 1 **SRO Group** 1 295007K205

295007 High Reactor Pressure **Record Number** 10

AK2. Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following:

AK2.05 Shutdown cooling: Plant-Specific 2.9 3.1

Explanation of Answer Loss of cooling media to RHR HX will cause reactor pressure to increase until 82 psig setpoint for NSSSS SDC isolation

Reference Title

HC.OP-SO.SM-0001

Learning Objectives

000028E008 (R) Given a system which physically connects to or is required to support the operation of the RHR System or components therein, explain the function of the supporting system, IAW the RHR System Lesson Plan.

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Following a reactor scram and Main Steam Isolation Valve closure, reactor steam dome pressure reaches 1050 psig causing the "H" and "P" Safety Relief Valves (SRV) to open.

Which one of the following lists the operating setpoints for subsequent openings of the "P" SRV?

a. SRV "P" opens at 1017 psig and closes at 905 psig

b. SRV "P" opens at 1017 psig and closes at 935 psig

c. SRV "P" opens at 1047 psig and closes at 905 psig

d. SRV "P" opens at 1047 psig and closes at 935 psig

Answer d **Exam Level** B **Cognitive Level** Memory **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 1 **SRO Group** 1 295007K304

295007 High Reactor Pressure **Record Number** 11

AK3. Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE:

AK3.04 Safety/relief valve operation: Plant-Specific 4.0 4.1

Explanation of Answer SRV "P" opens at 1047 psig and closes at 935 psig

Reference Title

HC.OP-SO.SN-0001 Precautions 3.2.12

Learning Objectives

000046E003 (R) Concerning the safety relief valves; summarize, list or identify the following IAW the lesson plan.

a. The number and type of SRV's at Hope Creek.
b. Which SRV's have an ADS function.
c. Power supplies to the SRV solenoids.
d. Which SRV's can be operated remotely and the location from which each of these valves can be operated.
e. Purpose of the low-low set function and determine which SRV's are used for this function.
f. Determine the sequence of operation of the low-low set SRV's including arming setpoints, lift points and reclose setpoints.

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments: QID #8451 Hope Creek 02/28/1998

The plant is at 62% power, recovering from an inadvertent trip of the "B" Reactor Recirc pump. Shortly after the Recirc pump was started and power ascension commenced, annunciator C8-B5 "RPV LEVEL 7" is received. The NCO notes that actual level is 39" and rising.

At this time, the required operator action is to...

a. place the reactor vessel water level control system in manual.

b. verify Hydrogen Water Chemical Injection trip.

c. close the Main Steam Isolation Valves.

d. reduce reactor recirc flow to minimum.

Answer a **Exam Level** B **Cognitive Level** Memory **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 2 295008A101

295008 High Reactor Water Level **Record Number** 12

AA1. Ability to operate and/or monitor the following as they apply to HIGH REACTOR WATER LEVEL:

AA1.01 Reactor water level control: Plant-Specific 3.7 3.7

Explanation of Answer Immediate operator action from AB-200

Reference Title

HC.OP-AB.ZZ-0200

Learning Objectives

0AB200E002 (R) From memory, recall the Immediate Operator Actions for Reactor Level Control Malfunction, Abnormal Operating Procedure.

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: Vision QID# Q53987

A malfunction of the Digital Feedwater Level Controller has resulted in an INCREASING reactor water level. The Reactor Feedwater Pumps are automatically tripped on a high reactor water level signal to prevent:

a. feed pump damage due to increasing pump discharge flow rate and head.

b. main turbine damage due to water impingement on turbine blades.

c. reactor vessel damage due to completely filling and overpressurizing the vessel.

d. main steam line piping and hanger damage due to filling the main steam lines.

Answer b **Exam Level** B **Cognitive Level** Memory **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 2 295008K304

295008 High Reactor Water Level **Record Number** 13

AK3. Knowledge of the reasons for the following responses as they apply to HIGH REACTOR WATER LEVEL:

AK3.04 Reactor feed pump trip: Plant-Specific 3.3 3.5

Explanation of Answer Feedpumps are tripped to prevent reactor overfill and damage to the main turbine.

Reference Title

TC Bases 3/4.3.9

Learning Objectives

000002E008 (R) Given a list of reactor vessel pressure and/or level setpoints determine the automatic action that occurs IAW the Lesson Plan.

Material Required for Examination

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: QID #6574 Dresden 03/11/1996

Given the following:

- A plant start-up is in progress
- Reactor power is 1%
- Recirculation loop temperature is 300°F
- "RPV LEVEL 4" alarm is received

What is the actual RPV water level?

a. 24 inches

b. 27 inches

c. 30 inches

d. 33 inches

Answer a **Exam Level** B **Cognitive Level** Comprehension **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 1 **SRO Group** 1 295009A201

295009 Low Reactor Water Level **Record Number** 14

AA2. Ability to determine and/or interpret the following as they apply to LOW REACTOR WATER LEVEL:

AA2.01 Reactor water level 4.2 4.2

Explanation of Answer
Provide a copy of HC.OP-IO.ZZ-0003, Attachment 6 Narrow Range
CORRECT - 24 inches. Value obtained from the 250°F
INCORRECT - 27 inches. Value obtained from the 350°F lines.
INCORRECT - 30 inches. Value without temperature compensation.
INCORRECT - 33 inches. Value indicated level for actual level of 30" at 450 F.

Reference Title

HC.OP-IO.ZZ-0003, Attachment 6

Learning Objectives

00112CE005 (R) Interpret charts, graphs and tables contained within the STARTUP FROM COLD SHUTDOWN TO RATED POWER Integrated Operating Procedure to maintain plant operations within specified limits.

Material Required for Examination HC.OP-IO.ZZ-0003, Attachment 6 page 52

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: VISION BANK QID# Q56518

Given the following:

- A LOCA has occurred
- All rods are full in
- "A" and "B" RHR Pumps are NOT available
- HPCI AND RCIC are NOT available
- Reactor water level is -150 and steady
- Reactor Feedwater Pumps are flowing 12,000 gpm each
- Reactor pressure is 1000 psig
- Drywell pressure is 45 psig and rising at 10 psig per minute
- Suppression Chamber pressure is 45 psig and rising at 10 psig per minute

The EOP mitigation strategy for this event is:

a. Depressurize with SRVs; inject with sources internal to the containment

b. Depressurize with SRVs; inject with sources external to the containment

c. Inhibit ADS and remain at pressure to conserve inventory; inject with sources internal to the containment

d. Inhibit ADS and remain at pressure to conserve inventory; inject with sources external to the containment

Answer a **Exam Level** S **Cognitive Level** Application **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 1 **SRO Group** 1 295009G406

295009 Low Reactor Water Level **Record Number** 15

2.4 Emergency Procedures and Plan

2.4.6 Knowledge symptom based EOP mitigation strategies. 3.1 4.0

Explanation of Answer Conditions provided are symptoms of a Feedwater line break inside the drywell. Drywell pressure above PSP requires emergency depressurization. If Drywell pressure cannot be maintained below 65 psig, then terminate RPV injection from sources outside containment not required for adequate core cooling.

Reference Title

EOP 101 Step RC/L2

10CFR55.43(5)

Learning Objectives

00124AE006 (R) Given any step of the procedure, describe the reason for performance of that step and/or expected system response to control manipulation prescribed by that step.

Material Required for Examination EOP Flowcharts without entry conditions

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following:

- The plant is operating at 100 percent power
- Equipment Drain Sump leakage has remained constant at 2.0 gpm for 8 weeks.
- Floor Drain Sump leakage has risen steadily over several days from 1.2 g to 1.8 gpm.

At 0800 this day and hourly thereafter, operators obtained the following readings on the Floor Drain Sump:

0800	1.8
0900	2.1
1000	2.5
1100	2.7
1200	3.1
1300	3.2
1400	3.7
1500	3.9

Has a Technical Specification operational leakage limit for the Reactor Coolant System been exceeded and what is the bases for your answer?

a. No, because total leakage has remained less than 5 gpm

b. No, because unidentified leakage has remained at about 2 gpm

c. Yes, because unidentified leakage has increased by more than 2 gpm

d. Yes, because total leakage has increased to more than 5 gpm

Answer c **Exam Level** S **Cognitive Level** Application **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 1 **SRO Group** 1 295010A102

295010 High Drywell Pressure **Record Number** 16

AA1. Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE:

AA1.02 Drywell floor and equipment drain sumps 3.6 3.6

Explanation of Answer SRO UNIQUE - RO LEVEL QUESTION
Floor drain leakage is Unidentified leakage. 2 gpm or more increase in 24 hours is an entry into TS 3.4.3.2

Reference Title

TS 3.4.3.2

Learning Objectives

000033E007 (R) Given a copy of the Technical Specifications, choose those sections which are applicable to the Drywell Ventilation System IAW the Drywell Ventilation System Lesson Plan.

000221E006 Given a scenario of applicable operating conditions and access to Technical Specifications:

a. Identify those sections which are applicable to the Radiation Monitoring System IAW the Radiation Monitoring System Lesson Plan.
b. Evaluate RMS operability and determine required actions associated with Radiation Monitoring System inoperability.
c. Explain the bases for those Technical Specification items associated with the Radiation Monitoring System. (SRO only)

Material Required for Examination	Tech Specs without Definitions, Safety Limits, and bases

Question Source:	INPO Exam Bank	**Question Modification Method:**	Significantly Modified
Question Source Comments:	QID#628 Duane Arnold 05/25/1999		

Given the following:

- The reactor has scrammed due to rising Drywell pressure
- Drywell Floor Drain Sump Pumps have stopped running
- Drywell pressure continues to increase

Which one of the following describes the reason why the sump pumps have stopped?

a. The Drywell Leak Detection (DLD) Sump Monitoring goes offscale high

b. The Reactor Recirc Seal Staging flow is isolated

c. The sump pump suction screens are clogged

d. The Non-IE power source is shed

Answer d **Exam Level** R **Cognitive Level** Comprehension **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 1 **SRO Group** 1 295010A102

295010 High Drywell Pressure **Record Number** 17

AA1. Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE:

AA1.02 Drywell floor and equipment drain sumps 3.6 3.6

Explanation of Answer Drywell Floor Drain Sump pumps are powered from 10B252 and 262 MCC's which are shed on high drywell pressure.
- DLD goes offscale high- incorrect. This occurs but is not the reason the pumps stopped
- RR seal staging flow leakoff goes to the DW Equipment sump
- Suctions screens could clog from debris, but the pumps would continue to run

Reference Title

HC.OP-SO.SM-0001 Table SM-20

Learning Objectives

000086E011 (R) From memory test/identify the conditions/signals that will cause the Drywell Equipment and Floor Drain Containment Isolation valves to automatically close, IAW the Lesson Plan.

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- A LOCA has occurred
- Drywell temperature is 300F
- Drywell pressure is 3.0 psig

Which one of the following describes the plant response when one loop of Drywell Spray is initiated?

a. Reactor vessel level indications will be lost

b. SRV operation can no longer be assured

c. Running Drywell cooling fans will automatically trip

d. Drywell pressure will drop below the scram setpoint

Answer d **Exam Level** B **Cognitive Level** Comprehension **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 2 295012K101

295012 High Drywell Temperature **Record Number** 18

AK1. Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE:

AK1.01 Pressure/temperature relationship 3.3 3.5

Explanation of Answer Bases of Drywell Spray Initiation Limit states "DWSIL is the highest drywell temperature at which initiation of drywell sprays will not result in an evaporative cooling pressure drop to below either: The drywell-below-wetwell differential pressure capability, or - The high drywell pressure scram setpoint. Since the parameters given are in the UNSAFE region of the DWSIL curve, Drywell pressure will drop below the scram setpoint.
-Drywell sprays cool the areas surrounding the RPV Level instrumentation reference legs, improving reliablity
- Drywell cooling fans trip on 1.68 psig or manually before; not because sprays are initiated
- SRV operation is limited by DW temps above 340F, not sprays initiation

Reference Title

EOP Caution 1

Learning Objectives

00124AE006	(R) Given any step of the procedure, describe the reason for performance of that step and/or expected system response to control manipulation prescribed by that step.
000002E014	(R) Given changes in the following parameters, evaluate the affect on each RPV level indication IAW the Lesson Plan. a. Reactor Pressure b. Drywell Temperature c. Steam Flow

Material Required for Examination

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: QID# 8632 Hope Creek Unit 08/10/1998

Which one of the following describes the bases for Suppression Pool Cooling being required to be in service as a prerequisite to starting HPCI for surveillance testing?

- a. To ensure adequate thermal mixing of the water in the Suppression Pool to limit stress on the torus shell due to differential thermal expansion.
- b. To allow the maximum average Suppression Pool water temperature limit to be increased to 105°F.
- c. To extend the operating time for HPCI testing before the maximum average temperature limit is reached and testing is required to be stopped.
- d. To ensure that heat added to the Suppression Pool does NOT increase Suppression Chamber air space pressure to the point where the Suppression Chamber to Drywell vacuum breakers cycle.

Answer c **Exam Level** B **Cognitive Level** Memory **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 1 295013K201

295013 High Suppression Pool Temperature **Record Number** 19

AK2. Knowledge of the interrelations between HIGH SUPPRESSION POOL TEMPERATURE and the following:

AK2.01 Suppression pool cooling 3.6 3.7

Explanation of Answer Reason for prerequisite 2.1.9 of quarterly surveillance test

Reference Title

HC.OP-IS.BJ-0001 Section 2.1.9

Learning Objectives

000026E014 (R) Given plant problems/industry events associated with the HPCI system:
a. Discuss the root cause of the plant problem/industry event IAW the HPCI System Lesson Plan.
b. Discuss the HCGS design and/or procedural guidelines that mitigate/reduce the likelihood of the problem/industry event at HCGS IAW the HPCI System Lesson Plan.
c. Discuss the "lessons learned" from this problem/event IAW the HPCI System Lesson Plan.

Material Required for Examination

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: INPO BANK QID# 8883

Given the following:

- The plant is operating at 100% power
- Main Steam Isolation Valve AB-HV-F022A inadvertently closes

Which one of the following describes the response of the reactor?

Reactor power will:

a. drop initially due to a Reactor Recirc intermediate runback when RPV level reaches +30 inches. This increases the boiling boundary length which adds negative reactivity.

b. rise initially due to the reactor pressure rising. This causes a collapse of voids in the core which adds positive reactivity.

c. rise initially due to a rising core water level caused by rising reactor pressure. Power will return to a slightly lower level in response to Reactor Water Level Control and Turbine Control Valve movement.

d. drop initially due to the void boundary being pushed lower in the core. As the Turbine Control Valves respond to lower reactor pressure, power rises as the void boundary rises.

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 1 **SRO Group** 1 295014K204

295014 Inadvertent Reactivity Addition **Record Number** 20

AK2. Knowledge of the interrelations between INADVERTENT REACTIVITY ADDITION and the following:

AK2.04 Void concentration 3.2 3.3

Explanation of Answer When the MSIV closes, steam flow is reduced as 100 percent steam flow now passes through only 3 steamlines. Reactor pressure rises and voids collapse. Void collapse causes power to rise.

Reference Title

HC.OP-AB.ZZ-0202

Learning Objectives

000228E024 Given a reactor power change analyze that power change and predict how the various reactivity coefficients respond.

Material Required for Examination

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: INPO BANK QID# 16307 Grand Gulf 1 04/01/2000

Given the following:

- The plant is operating at 60% power
- Both Reactor Recirc Pump Speed Controllers are in AUTO (Master Manual)

Which one of the following would require the operator to immediately place the Reactor Mode Switch to Shutdown

a. SIC-R620 Master Speed Control Recirc Master Demand fails full upscale

b. SIC-R620 Master Speed Control Recirc Master Demand fails full downscale

c. SIC-R621A Reactor Recirc pump speed demand fails full upscale

d. SIC-R621A Reactor Recirc pump speed demand fails full downscale

Answer a **Exam Level** B **Cognitive Level** Memory **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 1 **SRO Group** 1 295014K301

295014 Inadvertent Reactivity Addition **Record Number** 21

AK3. Knowledge of the reasons for the following responses as they apply to INADVERTENT REACTIVITY ADDITION:

AK3.01 Reactor SCRAM 4.1 4.1

Explanation of Answer Immediate operator action for dual recirc pump runaway IAW HC.OP-AB.ZZ-0204

Reference Title

HC.OP-AB.ZZ-0204

Learning Objectives

0AB204E002 (R) From memory, recall the Immediate Operator Actions for Positive Reactivity Addition, Abnormal Operating Procedure.

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following:

- An ATWS with fuel damage has occurred
- The Emergency Duty Officer (EDO) decides that it is necessary to send someone into the Reactor Building (with Radiation Protection) to individually scram rods

What is the maximum allowable dose limit that the EDO may authorize for this evolution?

a. 5 REM

b. 10 REM

c. 25 REM

d. 75 REM

Answer c **Exam Level** S **Cognitive Level** Memory **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 1 **SRO Group** 1 295015G304

295015 Incomplete SCRAM **Record Number** 22

2.3 Radiological Controls

2.3.4 Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized. 2.5 3.1

Explanation of Answer The EDO may authorize 25 REM per person for emergency actions to mitigate the consequences of an accident.

Reference Title

NC.EP-EP.ZZ-0304 Sect 5.2

10CFR55.43(4)

Learning Objectives

Material Required for Examination

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: INPO EXAM BANK Question ID# 7986. Hatch 3/97 modified for Hope Creek. EP for Licensed Operators. Lesson Plan

Given the following:

- The reactor scrammed from 100 percent power
- Reactor power is on the Source Range Monitors
- 3 rods remain at position "48"
- Scram air header reads 0 psig
- The scram CANNOT be reset

IAW EOP Bases, which one of the following methods of achieving shutdown condition is best for these conditions?

a. Vent control rod over-piston areas to insert rods

b. De-energize scram solenoids to insert rods

c. Defeat Rod Worth Minimizer to insert rods

d. Initiate Standby Liquid Control to inject boron

Answer c **Exam Level** S **Cognitive Level** Comprehension **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 1 **SRO Group** 1 295015K301

295015 Incomplete SCRAM **Record Number** 23

AK3. Knowledge of the reasons for the following responses as they apply to INCOMPLETE SCRAM:

AK3.01 Bypassing rod insertion blocks 3.4 3.7

Explanation of Answer RC/Q-21 states "Drive control rods, defeat RWM interlocks if necessary. This method is best applied when only a few control rods cannot be inserted". The scram cannot be reset. Scram solenoids are already de-energized. Power is less than 4% so SBLC should not be used.

Reference Title

HC EOP Bases step RC/Q-21

10CFR55.43(2)

Learning Objectives

00124BE008 (R) Given any step of the procedure, explain the reason for performance of that step and/or evaluate the expected system response to control manipulations prescribed by that step.

Material Required for Examination EOP Flowcharts without entry conditions

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following:

- The plant was operating at 100% power
- Toxic gas concerns have required the Main Control Room to be evacuated
- The transfer of controls to the Remote Shutdown Panel have been completed

Which of the following systems are available for reactor vessel pressure control from the Remote Shutdown Panel?

a. SRV's F, H & M and RHR Shutdown Cooling

b. Turbine Bypass Valves and Reactor Core Isolation Cooling

c. Reactor Feed Pumps and Reactor Recirculation

d. High Pressure Coolant Injection and LO-LO SET SRVs

Answer a **Exam Level** B **Cognitive Level** Memory **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 1 295016A102

295016 Control Room Abandonment **Record Number** 24

AA1. Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT:

AA1.02 Reactor/turbine pressure regulating system 2.9 3.1

Explanation of Answer IO-8 initiates rpv cooldown with SRV's F,H, & M until SDC can be established. Subsequent actions of AB-130 Trip the main turbine and close the MSIVs. Recirc pumps are manually tripped and discharge valves closed. HPCI cannot be controlled from the RSP. LO LO SET SRVs are only controlled from the Control Room.

Reference Title

HC.OP-IO.ZZ-0008

HC.OP-AB.ZZ-0130

Learning Objectives

00112HE006 (R) Analyze plant conditions and parameters to determine if plant operation is in accordance with the SHUTDOWN FROM OUTSIDE THE CONTROL ROOM Integrated Operating Procedure, supporting System Operating Procedures and Technical Specifications.

Material Required for Examination

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: INPO EXAM BANK QID# 18592 Peach Bottom 2 09/19/1997

Given the following:

- A plant shutdown is in progress
- North Plant Vent RMS is in HIGH alarm
- South Plant Vent RMS is reading 4.5 e+2 uCi/sec
- FRVS Vent RMS is reading 6.5 e-2 uCi/sec
- FRVS is NOT in service

Which one of the following is the source of the high alarm?

a. Service Area Exhaust System

b. Solid Radwaste Exhaust System

c. Radwaste Area Exhaust System

d. Turbine Building Exhaust System

Answer b **Exam Level** S **Cognitive Level** Comprehension **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 1 295017A201

295017 High Off-Site Release Rate **Record Number** 25

AA2. Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE:

AA2.01 Off-site release rate: Plant-Specific 2.9 4.2

Explanation of Answer HC.OP-AB.ZZ-0126
Solid Radwaste Exhaust discharges to North Plant Vent Stack
Others discharge to the South Plant Vent

Reference Title

HC.OP-AB.ZZ-0126

10CFR55.43(4)

Learning Objectives

000114E003 (R) Discuss the operational implications of the abnormal indications/alarms for system operating parameters related to a given Abnormal Operating Procedure.

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments: VISION BANK QID# Q55943

Given the following:

- Marsh Grass intrusion has reduced the flow in Service Water Loops "A" & "B"
- The differential pressure across the "A" SSW Pump Strainer is being reduced to maximize strainer backwash operation

Per HC.OP-AB.ZZ-0122, Service Water System Malfunction, why should the discharge valve of "A" SSW Pump be closed for no more than two minutes during this evolution?

a. All SSW flow from SSW Loop "A" to RACS and SACS will be lost

b. Lubricating water flow will be lost to SSW Pump "A".

c. Spray Water Booster Pump "A" will remain stopped by interlock

d. Blockage problems could worsen on other SSW Pump strainers

Answer d **Exam Level** S **Cognitive Level** Memory **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 2 295018A203

295018 Partial or Complete Loss of Component Cooling Water **Record Number** 26

AA2. Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER:

AA2.03 Cause for partial or complete loss 3.2 3.5

Explanation of Answer

SRO UNIQUE - RO LEVEL QUESTION
CORRECT - Blockage problems could worsen on other SSW Pump strainers. Per Caution 4.6.4.D of HC.OP-AB.ZZ-0122, pump operation should be limited to 2 minutes with the discharge valve closed. Closing the discharge path on one pump may compound blockage problems on other pumps by increasing differential pressure for those strainers.
INCORRECT - Lubricating water flow will be lost to SSW Pump A. Lubricating water flow is supplied from the Lubrication Head Tanks upstream of the SSW Pump discharge valve.
INCORRECT - Spray Water Booster Pump A will remain stopped by interlock. The Spray Water Booster Pump is stopped when its own discharge valve is shut, not when the SSW Pump discharge valve is shut.
INCORRECT - All SSW flow from SSW Loop A to RACS and SACS will be lost. SSW Loop A flow can be maintained by operating the C SSW Pump while the SSW Pump A flow is stopped.

Reference Title

HC.OP-AB.ZZ-0122, Caution 4.6.4.D

Learning Objectives

0AB122E004 Explain the reasons for how plant/system parameters respond when implementing, Service Water System Malfunction, Abnormal Operating Procedure.

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments: Vision Bank QID# Q61195

Given the following:

- A leak on the Instrument Air header has resulted in lowering header pressure.
- The "INST AIR HEADER A PRESSURE LO" annunciator alarm is received.

Which one of the following valves automatically open to restore header pressure and at what pressure?

a. The Instrument Air Dryer 1A-F-104 outlet valve KB-HV-11416; 70 psig

b. The Instrument Air Dryer 1A-F-104 outlet valve KB-HV-11416; 85 psig

c. The Instrument Air Dryer 10-F-104 outlet valve KB-HV-7618; 70 psig

d. The Instrument Air Dryer 10-F-104 outlet valve KB-HV-7618; 85 psig

Answer b **Exam Level** B **Cognitive Level** Memory **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 2 295019A102

295019 Partial or Complete Loss of Instrument Air **Record Number** 27

AA1. Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR:

AA1.02 Instrument air system valves: Plant-Specific 3.3 3.1

Explanation of Answer Instrument Air Dryer AF-104 outlet valve will automatically open at 85 psig on lowering air pressure.

Reference Title

HC-OP.AB-ZZ-0131

Learning Objectives

0AB131E004 Explain the reasons for how plant/system parameters respond when implementing, Loss Of Instrument Air And/Or Service Air, Abnormal Operating Procedure.

Material Required for Examination

Question Source: INPO Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: INPO Exam Bank QID #6871 Dresden 2 07/28/1999

Given the following:

- The plant is operating at 50 percent power during a startup
- Overhead alarms received - "MSIV CLOSURE"
- All 4 Outboard MSIV's OPEN and CLOSED indication lights are illuminated

Which one of the following would cause the alarm condition?

a. Degrading Instrument Air header pressure

b. Degrading Instrument Gas header pressure

c. Loss of solenoid power to the MSIV 4-way "operator valves"

d. Loss of solenoid power to the MSIV "test valves"

Answer a **Exam Level** R **Cognitive Level** Comprehension **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 2 295019A201

295019 Partial or Complete Loss of Instrument Air **Record Number** 28

AA2. Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR:

AA2.01 Instrument air system pressure 3.5 3.6

Explanation of Answer Degrading air header pressure on the Reactor Bldg supply header which supplies the outboard MSIVs would cause all four to slowly close. Loss of solenoid power to the test valves will have no effect because they are normally deenergized. "4 way valves" are air powered.

Reference Title

HC.OP-AB.ZZ-0131 Attachment 1

Learning Objectives

000046E014 (R) Concerning the Main Steam Isolation Valves (MSIV's), summarize, list or identify the following IAW the lesson plan.

a. Assess the effect on a MSIV if loss of electric or loss of pneumatic supply occurs.
b. Determine the signals which will automatically close the MSIV's and when, if ever, certain isolations can be bypassed.

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Given the following:

- The reactor has been in COLD SHUTDOWN for two (2) days following power operation
- Reactor vessel water level is +30 inches
- Neither Reactor Recirculation pump is available
- Shutdown Cooling has isolated and the Shutdown Cooling suction valves CANNOT be opened
- The highest RPV metal temperature is 190°F and rising
- HC.OP-AB.ZZ-0142, Loss of Shutdown Cooling has been entered

Based on given information, which one of the following is the highest Reporting Requirement/ECG classification applicable?

a. 8 hour report

b. 4 hour report

c. Unusual Event

d. Alert

Answer d **Exam Level** S **Cognitive Level** Application **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 3 **SRO Group** 2 295021G441

295021 Loss of Shutdown Cooling **Record Number** 29

2.4 Emergency Procedures and Plan

2.4.41 Knowledge of the emergency action level thresholds and classifications. 2.3 4.1

Explanation of Answer ECG EAL 8.1.2 Inability to maintain the plant in Cold Shutdown

Reference Title

HC ECG EAL 8.1.2

10CFR55.43(5)

Learning Objectives

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments: EP Lesson Plan

Given the following

- The plant is operating at 100 percent power
- "A" CRD Pump is C/T for maintenance
- CRD SYSTEM TROUBLE overhead alarm C6-F2 comes in
- CRD Cooling Water flow drops to zero gpm

What actions are required and what is the bases for those actions?

a. Scram the reactor upon the receipt of the second accumulator trouble alarm based on demonstrated Shutdown Margin

b. Scram the reactor upon the receipt of the second accumulator trouble alarm based on average control rod scram times

c. Scram the reactor within 20 minutes based on adequate time to place a CRD pump back in service

d. Scram the reactor within 20 minutes based on the ability for charging header pressure alone to fully insert all control rods

Answer c **Exam Level** S **Cognitive Level** Memory **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 2 295022G448

295022 Loss of CRD Pumps **Record Number** 30

2.4 Emergency Procedures and Plan

2.4.48 Ability to interpret control room indications to verify the status and operation of system, and understand how operator action s and directives affect plant and system conditions. 3.5 3.8

Explanation of Answer Tech spec bases 3/4 1.3. The question is based on the TS bases for an operator manual scram time requirements on a loss of both CRD pumps.

Reference Title

Tech spec bases 3/4 1.3

10CFR55.43(2)

Learning Objectives

000006E033 (R) Given a scenario of applicable operating conditions and access to Technical Specifications complete each of the following IAW Technical Specifications:

a. Select those sections applicable to the CRDH System.
b. Evaluate CRDH System operability and determine required actions and time limits associated with inoperable components.
c. Explain the bases for those Technical Specification sections associated with the CRDH System. SRO ONLY

Material Required for Examination Tech Specs without Definitions, Safety Limits, and bases

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following:

- The plant is at 37% power
- Both CRD pumps are tripped on low suction pressure
- The Reactor Building Operator is swapping CRD suction filters
- CRD ACCUM TROUBLE Overhead Annuciator C6-D4 is clear

(Assume NO other operator actions)

Which one of the following describes the effect on gas pressure in the HCU Accumulators 2 minutes following the pump trip?

a. Stays the same because reactor pressure holds the charging water check valve closed

b. Stays the same because accumulator pressure holds the charging water check valve closed

c. Lowers because the reactor scrams

d. Lowers because the accumulator piston moves when charging water header pressure is lost

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 2 295022K203

295022 Loss of CRD Pumps **Record Number** 31

AK2. Knowledge of the interrelations between LOSS OF CRD PUMPS and the following:

AK2.03 Accumulator pressures. 3.4 3.4

Explanation of Answer Charging water check valve 115 maintains water volume on a loss of charging pressure from the CRD pumps. N2 gas pressure will remain the same as long as the check valve holds. If the check valve does not hold, the piston will stroke and N2 pressure will drop causing low accumulator pressure alarm.

Reference Title

HC.OP-IS.BF-0103 Purpose

Lesson Plan 00006

Learning Objectives

000006E017 (R) Given the appropriate procedure or access to the procedure, summarize the accumulator trouble alarms and their setpoints associated with each CRD HCU and how these problems may impact CRDH System Operation, IAW the Lesson Plan.

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Given the following:

- "A" Fuel Pool Cooling Pump is tagged for maintenance
- "B" Fuel Pool Cooling Pump trips

How does this affect the ability to monitor Fuel Pool temperature in the Control Room?

a. Temperature recorder TR-4683 is unaffected because it monitors Skimmer Surge Tank temperature

b. The High Temperature alarm to Fuel Pool System Trouble (D1-D5) is INVALID because it monitors Skimmer Surge Tank temperature

c. The High Temperature alarm to Fuel Pool System Trouble (D1-D5) is VALID because it monitors Fuel Pool Cooling Pump common discharge piping

d. Temperature recorder TR-4683 is INVALID because it monitors Fuel Pool Cooling Pump common discharge piping

Answer d **Exam Level** B **Cognitive Level** Comprehension **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 3 **SRO Group** 1 295023A102

295023 Refueling Accidents **Record Number** 32

AA1. Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS:

AA1.02 Fuel pool cooling and cleanup system 2.9 3.1

Explanation of Answer Both Temp alarm and Recorder TR-4683 monitor the same parameter in the common discharge piping of the FPCC pumps. With no flow, the piping will equalize with ambient air temperature, no longer valid indication of Fuel Pool Temperature.

Reference Title

HC.OP-AR.ZZ-0013 Attachment D5

Learning Objectives

000043E017 (R) Given a set of conditions and a drawing of the controls, instrumentation and/or alarms located in the Control Room, identify the status of the FPCCS or its components by evaluation of the controls/instrumentation/alarms, IAW the Fuel Pool Cooling and Cleanup System (FPCCS) Lesson Plan.

Material Required for Examination Drawing of alarm window D1-D5. Drawing of TR-4683

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following:

- Core offload is in progress
- A fuel bundle was removed from the reactor vessel, full up on the Fuel Hoist, and in the cattle chute heading for the Fuel Pool
- Fuel Pool Skimmer Surge Tank Level is lowering rapidly

Which one of the following describes the operator actions required?

a. Place the bundle into its original reactor core location

b. Place the bundle into the Fuel Prep Machine

c. Stop the bridge at its current location and leave the refueling floor

d. Stop the bridge at its current location and lower the bundle full down

Answer a **Exam Level** B **Cognitive Level** Memory **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 3 **SRO Group** 1 295023G411

295023 Refueling Accidents **Record Number** 33

2.4 Emergency Procedures and Plan

2.4.11 Knowledge of abnormal condition procedures. 3.4 3.6

Explanation of Answer Immediate operator action on loss of fuel pool inventory/cooling is to return the bundle to either the reactor vessel or the fuel pool. Tech Spec definition of Core Alterations allows continued movement of a component to a safe location.

Reference Title

HC.OP-AB.ZZ-0144

Learning Objectives

0AB144E002 (R) From memory, recall the Immediate Operator Actions for Loss Of Fuel Pool Inventory/Cooling, Abnormal Operating Procedure.

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following:

- The reactor at rated power
- Hope Creek experiences a Loss of Offsite Power event and a reactor scram
- Approximately 13 sec into the event, Drywell pressure is 1.9 psig

Which one of the following describes the operation of the LOCA and LOP sequencers?

a. The LOP sequencer program will be in control of restoring the loads.

b. The LOP sequencer will complete sequencing 2 minutes later, then the LOCA sequencer will start.

c. The LOCA sequencer program will be in control of restoring the loads.

d. The LOCA sequencer will complete sequencing 2 minutes later, then the LOP sequencer will start.

Answer c **Exam Level** B **Cognitive Level** Memory **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 1 **SRO Group** 1 295024A110

295024 High Drywell Pressure **Record Number** 34

EA1. Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE:

EA1.10 A.C. distribution 3.4 3.6

Explanation of Answer Justification
The LOCA Sequencer takes priority over the LOP Sequencer to ensure that all systems required to protect the core and prevent radioactive release are sequenced on when required and that unnecessary loads do not overload the diesels

Reference Title

HC.OP-SO.KJ-0001

Learning Objectives

000066E012 Summarize/identify the emergency load sequencer response for a LOP concurrent with a LOCA signal IAW Attachment 1 of the Lesson Plan.

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: VISION BANK QID# Q53753

Given the following:

- A small Reactor Coolant leak in the Drywell occurs
- Drywell Leak Detection System alarms
- Drywell pressure is rising

Which one of the following actions requires CRS authorization prior to performance?

a. Start an Emergency Diesel Generator following failure to start

b. Restore Primary Containment Instrument Gas following isolation

c. Maximize Drywell cooling prior to high Drywell pressure alarm

d. Terminate Drywell inerting if in progress

Answer b **Exam Level** B **Cognitive Level** Memory **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 1 **SRO Group** 1 295024G106

295024 High Drywell Pressure **Record Number** 35

2.1 Conduct of Operations

2.1.6 Ability to supervise and assume a management role during plant transients and upset conditions. 2.1 4.3

Explanation of Answer Restoring PCIG following isolation requires overriding LOCA signals. Overriding Tech Spec required isolation to Primary Containment Isolation Valves requires SRO authorization. This directed by the CRS through implementation of EOP-101 or 101A.
Start an EDG following failure - incorrect - Immediate operator action of AB-135
Maximize DW Cooling - incorrect - Immediate operator action of AB-201
Terminate DW inerting - incorrect - Immediate operator action of AB-201

Reference Title

HC.OP-EO.ZZ-0101 step RC/P-5

Learning Objectives

000113E079 State the three (3) conditions when a facility must evaluate proposed actions.

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: VISION BANK QID# Q57081

Given the following:

- The reactor is operating at 100% power
- A spurious Main Turbine trip occurs
- The reactor scrams with all rods going full in
- Turbine Bypass valves fail to operate properly resulting in a reactor pressure excursion up to 1100 psig

What is the impact on the Digital Feedwater Level Control System?
(Assume no operator action)

a. Operation of RFP Controllers in MANUAL is available after 55 seconds

b. Operation of all controllers is automatically restored in 12.5 minutes

c. Operation of RFP Controllers is available in MANUAL or AUTO until the RFP's trip

d. The Master Level Controller will stay at its original demand signal for 10 seconds

Answer c **Exam Level** R **Cognitive Level** Comprehension **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 1 **SRO Group** 1 295025A206

295025 High Reactor Pressure **Record Number** 36

EA2. Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE:

EA2.06 Reactor water level 3.7 3.8

Explanation of Answer Reactor high pressure causes RRCS initiation. ARI initiates but Feedwater Runback requires APRMS Inop or Not Downscale (ATWS). This question tests the operators ability to determine the post transient control of reactor level.

Reference Title

HC.OP-SO.AE-0001

Learning Objectives

000059E015 (R) From memory, describe the three possible RFP runback signals including conditions, setpoints and time delays if applicable, IAW the Feedwater Control System Lesson Plan.

Material Required for Examination

Question Source: INPO Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: INPO EXAM BANK QID# 2336 Perry 08/30/1997

Given the following:

- The plant was operating at 100% power
- A transient occurs
- RPV pressure reached 1330 psig before turning downward

WHICH ONE of the following states the required action(s) for RPV pressure reaching 1330 psig?

a. Prepare and submit a Safety Limit Violation Report within 30 days.

b. Restore to within limits within 15 minutes or be in COLD SHUTDOWN within the next 6 hours.

c. Restore to within limits within 1 hour or be in COLD SHUTDOWN within the next 12 hours.

d. Perform an engineering evaluation on the out-of-limits condition within 24 hours.

Answer	a	**Exam Level**	B	**Cognitive Level**	Application	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Emergency and Abnormal Plant Evolutions		**RO Group**	1	**SRO Group**	1		295025K105	
295025	High Reactor Pressure							**Record Number**	37

EK1.	Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE:		
EK1.05	Exceeding safety limits	4.4	4.7

Explanation of Answer Action for Safety Limit Violation is specified in TS Admin controls section 6.7.1.d

Reference Title

HC Tech Specs 6.7.1.d

Learning Objectives

000110E002 (R) Given Technical Specifications, determine the administrative and operational actions that must be performed if a Safety Limit is violated.

Material Required for Examination Tech Specs without Definitions, Safety Limits, and bases

Question Source: INPO Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: INPO BANK QID# 13957 Palo Verde 11/18/1996

Given the following:

- A LOCA has occurred
- Drywell temperature is 240°F
- Suppression Chamber pressure is 7.5 psig
- Suppression Pool temperature is 125 F and rising

Which one of the following describes the bases for initiating Suppression Chamber Spray at this pressure?

a. To prevent exceeding the negative design pressure of the primary containment.

b. To reduce primary containment pressure by condensing steam which may be present in the Suppression Chamber airspace.

c. To reduce accumulation of non-condensibles in the Suppression Chamber.

d. To prevent Drywell depressurization that exceeds the capacity of the Suppression Chamber to Drywell vacuum breakers.

Answer	b	**Exam Level**	B	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Emergency and Abnormal Plant Evolutions		**RO Group**	2	**SRO Group**	1			295026K102
295026	Suppression Pool High Water Temperature							**Record Number**	38

EK1. Knowledge of the operational implications of the following concepts as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE:

EK1.02 Steam condensation 3.5 3.8

Explanation of Answer Suppression chamber sprays are initiated below 9.5 psig to reduce primary containment pressure by condensing steam which may be present in the SC airspace.

Reference Title

HC EOP Bases step DW/P-5

Learning Objectives

00126AE009 (R) Given plant conditions and access to EOPs, select the value of the Suppression Chamber Spray Initiation Pressure and explain the basis for this limit IAW the Primary Containment Control - Drywell Lesson Plan.

Material Required for Examination

Question Source:	INPO Exam Bank	**Question Modification Method:**	Significantly Modified
Question Source Comments:	INPO EXAM BANK QID# 8062Hope Creek Unit 09/28/1997		

Given the following:

- The Reactor has scrammed
- A small break occurred on the RPV head vent line
- Drywell temperature is 330°F and rising
- Drywell sprays are NOT available

Emergency Depressurization is required to prevent exceeding which one of the following?

a. Readable range of Drywell temperature instrumentation

b. Maximum capacity of the Drywell Cooling system

c. Saturation temperature for the Drywell design pressure

d. Environmental qualification temperature of safety related equipment in the Drywell

Answer	d	**Exam Level**	B	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Emergency and Abnormal Plant Evolutions			**RO Group**	2	**SRO Group**	2		295028K102
295028	High Drywell Temperature							**Record Number**	39

EK1. Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE:

EK1.02 Equipment environmental qualification 2.9 3.1

Explanation of Answer IAW EOP 102 bases step DW/T-3, 340°F is the qualification limit for ADS as well as the Drywell design temp. ED before 340 so that ADS valves can be used.

Reference Title

EOP 102 bases step DW/T-3 and DW/T-5

Learning Objectives

00126AE007 (R) Given any step of the procedure, determine the reason for performance of that step and/or predict expected system response to control manipulations prescribed by that step IAW the Primary Containment Control - Drywell Lesson Plan.

Material Required for Examination EOP Flowcharts without entry conditions

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments:

Given the following:

- Reactor is scrammed
- Suppression Pool level is lowering

If Suppression Pool level reaches 49 inches, which one of the following would occur?

a. Reactor Building to Suppression Chamber Vacuum Breakers close if open

b. Reactor Building to Suppression Chamber Vacuum Breakers open if closed

c. Drywell to Suppression Chamber differential pressure increases

d. Drywell to Suppression Chamber differential pressure decreases

Answer d **Exam Level** S **Cognitive Level** Comprehension **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 1 295030A204

295030 Low Suppression Pool Water Level **Record Number** 40

EA2. Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL:

EA2.04 Drywell/ suppression chamber differential pressure: Mark-I&II 3.5 3.7

Explanation of Answer SRO UNIQUE - RO LEVEL QUESTION
At 55 inches in the SC, the Vent Header drain pipe uncovers causing differential pressure to equalize.

Reference Title

EOP- 102 step SP/L-5 bases

Learning Objectives

00125AE009	(R) Given any step of the procedure, determine the reason for performance of that step and/or predict expected system response to control manipulations prescribed by that step IAW the Primary Containment Control - Suppression Pool Lesson Plan.

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following:

- A plant shutdown is in progress
- "A" RHR is tagged for motor replacement
- "B" RHR is in Shutdown Cooling at 210°F
- Suppression Pool Level Low annunciator alarms
- The PO reports Suppression Pool level is lowering

Which one of the following makeup sources must be used?

a. Suppression Pool Makeup from HPCI using OP-EO.ZZ-0312

b. Suppression Pool Makeup from RCIC using OP-EO.ZZ-0313

c. Suppression Pool Makeup from Service Water using OP-EO.ZZ-0314

d. Suppression Pool Makeup from Core Spray using OP-EO.ZZ-0315

Answer	d	**Exam Level**	B	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Emergency and Abnormal Plant Evolutions			**RO Group**	2	**SRO Group**	1		295030K203
295030	Low Suppression Pool Water Level							**Record Number**	41

EK2. Knowledge of the interrelations between LOW SUPPRESSION POOL WATER LEVEL and the following:

EK2.03 LPCS 3.8 3.9

Explanation of Answer: The plant is in Op Cond 3. Core Spray must be used because it is the only source available for the given conditions. HPCI and RCIC do not have steam to run. SSW cannot be used due to B RHR is in SDC mode.

Reference Title

HC EOP step SP/L-4

Learning Objectives

00125AE009 (R) Given any step of the procedure, determine the reason for performance of that step and/or predict expected system response to control manipulations prescribed by that step IAW the Primary Containment Control - Suppression Pool Lesson Plan.

Material Required for Examination: EOP Flowcharts without entry conditions

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following:

- I&C surveillance testing has caused an inadvertent RPS scram signal
- RPS actuates but some rods remain out with power at 2%
- RPV level lowers until RCIC and HPCI initiate
- Operators commence recovering level with Feedwater
- RPV level was below Level 2 for 15 seconds
- The Main Turbine is still on-line

(Assume NO other operator actions)

Which one of the following describes the status of RRCS?

a. ARI valves are energized and RPT breakers are open

b. RPT breakers are closed and ARI valves are de-energized

c. Feed pumps have runback to minimum and RPT breakers are closed

d. ARI valves are energized and SLC pumps will initiate when 3.9 minute timer times out

Answer	a	**Exam Level**	B	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Emergency and Abnormal Plant Evolutions			**RO Group**	1	**SRO Group**	1		295031K213
295031	Reactor Low Water Level							**Record Number**	42

EK2. Knowledge of the interrelations between REACTOR LOW WATER LEVEL and the following:

EK2.13 ARI/RPT/ATWS: Plant-Specific 4.1 4.2

Explanation of Answer

SLC will not initiate, power < 4%
Feedpumps will not runback, no 1071 psig signal.
ARI will actuate, energizing the valves.
RPT breakers will be open, level was < -38 for >9 seconds.

Reference Title

HC.OP-SO.SA-0001

Learning Objectives

000024E007	(R) From memory, predict the sequence of events which occur within the Redundant Reactivity Control System upon: a. Automatic initiation in response to a high reactor vessel pressure condition with or without the APRM permissive, IAW the Lesson Plan. b. Automatic initiation in response to a low reactor vessel water level condition with or without the APRM permissive, IAW the Lesson Plan. c. Manual initiation with or without the APRM permissive, IAW the Lesson Plan.

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: Vision Bank QID# Q53742

Given the following:

- The plant is in Operational Condition 4 for vessel disassembly
- Due to mishandling of the Reactor Vessel head insulation package, all 3 channels of Refuel Floor Exhaust RMS unexpectedly alarm HIGH on the RM-11
- PCIS responds normally

Which one of the following is the highest Reporting Requirement/ECG classification (if any) applicable?

a. NOT Reportable

b. 8 hour report

c. Unusual Event

d. Alert

Answer	b	**Exam Level**	S	**Cognitive Level**	Application	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	2	**SRO Group**	2				295034G430
295034	Secondary Containment Ventilation High Radiation							**Record Number**	43

2.4	Emergency Procedures and Plan	
2.4.30	Knowledge of which events related to system operations/status should be reported to outside agencies.	2.2 3.6

Explanation of Answer RAL 11.3.3 Bases - Valid actuation of listed systems listed in the Tech Bases. The RFE RMS responded to valid Hi radiation conditions from the radiography. The actuations were not part of a pre-planned test. Therefore, the ESF is reportable.

Reference Title

ECG Section 11.3.3 bases

10CFR55.43(5)

Learning Objectives

Material Required for Examination	ECG and ECG Technical Bases ESF Actuation Flow chart page 2 and 3 of 4		
Question Source:	New	**Question Modification Method:**	
Question Source Comments:	EP Lesson Plan		

Given the following:

- The reactor is operating at 100% power
- Annunciator B1-B3 (RCIC PUMP ROOM FLOODED) alarms with the following alarm message presented on the CRIDS display: D2887 RCIC PUMP RM 4151-1 LSH 4151-1 HI
- An investigation reveals that Reactor Building Floor Drain Sump pumps have been running continuously for 10 minutes
- The Reactor Building Operator reports the leak is coming from the CST suction line

In addition to running the sump pumps, which of the following action(s), if any, is required by EOP 103/4?

I --- Isolate RCIC
II -- Immediately commence a normal reactor shutdown
III -- Runback reactor recirculation and manually scram the reactor
IV - Emergency depressurize the reactor

a. I - ONLY

b. II - ONLY

c. I and II

d. I, III, and IV

Answer	a	**Exam Level**	B	**Cognitive Level**	Application	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Emergency and Abnormal Plant Evolutions			**RO Group**	3	**SRO Group**	2		295036K201
295036	Secondary Containment High Sump/Area Water Level							**Record Number**	44

EK2.	Knowledge of the interrelations between SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL and the following:	
EK2.01	Secondary containment equipment and floor drain system	3.1 3.2

Explanation of Answer The source of the leak is RCIC suction from the CST. Step RB 14 applies since RCIC is not required to assure adequate core cooling, shutdown the reactor , protect primary containment integrity, or suppress a fire. Reactor coolant is not the source of the leak based on RBO report. RB-15 is answered NO. Only one area is affected therefore Step RB-22 is not reached.

Reference Title

EOP 103/4 step RB-12

Learning Objectives

000127E006	(R) Given any step in the procedure, describe the reason for performance of that step and/or expected system response to control manipulations prescribed by the step.

Material Required for Examination EOP Flowcharts without entry conditions

Question Source: New **Question Modification Method:**

Question Source Comments: Vision Bank QID# Q56139 concept used

HC.OP-EO.ZZ-103/4, "Reactor Building Control", requires an Emergency Depressurization of the RPV if the Maximum Safe Operating Limit is exceeded in 2 or more areas listed in Table 2 Column 2.

SELECT the BASES for this Emergency Depressurization of the RPV.

a. To reduce the maximum Iodine release allowable during a MSL break accident

b. To prevent release of fission products into the Reactor Building by preventing fuel damage

c. To reduce the driving head and, therefore, the flow of the unisolated leaking Primary System

d. To protect personnel from high temperature environments while operating equipment

Answer	c	**Exam Level**	B	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Emergency and Abnormal Plant Evolutions		**RO Group**	3	**SRO Group**	2			295036K301
295036	Secondary Containment High Sump/Area Water Level							**Record Number**	45

EK3. Knowledge of the reasons for the following responses as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL:

EK3.01 Emergency depressurization 2.6 2.8

Explanation of Answer EOP Bases states "RPV depressurization places the primary system in its lowest energy state, rejects heat to the suppression pool in preference to outside containment, and reduces the driving head and flow of the primary systems that are unisolated and discharging into the reactor building."

Reference Title

EOP 103 Bases Step RB-19

Learning Objectives

000127E003 (R) Define the term "Maximum Safe Operating Temperature".

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: Vision Bank QID# Q56112

Which one of the following gaseous radioactive release limits corresponds to the EOP-104 entry condition?

a. 500 mRem to the Thyroid CEDE

b. 5000 mRem to the Thyroid CEDE

c. 2 times 10CFR 20 Appendix B limits

d. 200 times 10CFR 20 Appendix B limits

Answer	d	**Exam Level**	R	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	2	**SRO Group**	1				295038A203
295038	High Off-Site Release Rate							**Record Number**	46

EA2.	Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE:		
EA2.03	Radiation levels	3.5	4.3

Explanation of Answer

JUSTIFICATION:
CORRECT - IAW ECG Section 6 and Lesson plan 0302-000.00H-000127, the alert value is 200 times the 10CFR20 Appendix B value

Reference Title

ECG Section 6.0

LP 0302-000.00H-000127

Learning Objectives

000127E002 Given a set of plant conditions, analyze and determine if entry conditions into HC.OP-EO.ZZ-0103/4 exists.

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following:

- A LOCA outside primary containment and the Reactor Building has occurred
- AB-203 Main Steam Line High Radiation actions have been completed
- All control rods are full in
- Fuel cladding damage has occurred
- Release rates are above General Emergency levels
- Reactor level is -60 inches and rising slowly
- Reactor pressure is 100 psig

Why is an Emergency Depressurization required?

a. To ensure primary containment integrity

b. To allow low pressure ECCS to inject

c. To reduce the release rates

d. To provide core steam cooling

Answer	c	**Exam Level**	B	**Cognitive Level**	Comprehension	**Facility**	Hope Creek
Exam Date:	03/12/2002						
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	2	**SRO Group**	1		295038K102
295038	High Off-Site Release Rate					**Record Number**	47

EK1. Knowledge of the operational implications of the following concepts as they apply to HIGH OFF-SITE RELEASE RATE:

EK1.02	Protection of the general public	4.2	4.4

Explanation of Answer EOP 104 Bases for step RR-6 states an ED is required if release rates are above GE levels to reduce the radioactivity release rate. RX pressure is already low enough for low press ECCS to inject. The primary containment is already somehow bypassed. The ED is not driven by adequate core cooling requirements. The RPV water level is above TAF.

Reference Title

EOP 103/4 step RR-6 through 8

Learning Objectives

000127E006	(R) Given any step in the procedure, describe the reason for performance of that step and/or expected system response to control manipulations prescribed by the step.

Material Required for Examination EOP Flowcharts without entry conditions

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: Vision Bank QID# Q56164

Given the following:

- A large break LOCA has occurred inside the Drywell
- Multiple equipment failures occurred
- Drywell pressure is 15 psig
- Steam cooling was required until water level was restored above TAF with Fire Water
- The Containment H2/O2 Analyzers were placed in service
- The High Hydrogen alarms are clear

Which one of the following actions is required IAW EOP-102?

a. Vent the Drywell because Hydrogen concentration is above 2%

b. Exit EOP-102 and enter SAG because Hydrogen concentration is above 2%

c. Vent the Suppression Chamber because Hydrogen concentration is below 2%

d. Place the Hydrogen Recombiners in service because Hydrogen concentration is below 2%

Answer	b	**Exam Level**	B	**Cognitive Level**	Application	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Emergency and Abnormal Plant Evolutions			**RO Group**	1	**SRO Group**	1		500000K303
500000	High Containment Hydrogen Concentration							**Record Number**	48

EK3. Knowledge of the reasons for the following responses as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS:

EK3.03 Operation of hydrogen and oxygen recombiners 3.0 3.5

Explanation of Answer: High H2 alarms come in at 2% Hydrogen. Since the H2 concentration is above 2%, EOP-102 step PC/H1 directs exit from EOP-102 and enter SAG

Reference Title

EOP-102 step PC/H1

Learning Objectives

00126AE004	Recall the reasons why the following are used for determining the entry condition and / or subsequent actions IAW the Primary Containment Control - Drywell Lesson Plan. a. Drywell Pressure b. Average Drywell Temperature c. H2 and O2 concentrations in the drywell

Material Required for Examination: EOP Flowcharts without entry conditions

Question Source: New **Question Modification Method:**

Question Source Comments:

During a fire in the Turbine Building, a Fire Department Liaison is assigned by the Operations Superintendent.

Who, by title, can be assigned this role and what is their duty?

a. Communicator #1. Advises the Fire Department on how to mitigate the fire.

b. Communicator #2. Advises the Operations Superintendent on what equipment needs to be removed from service.

c. Shift Technical Advisor. Advises the Fire Department on how to mitigate the fire.

d. Work Control Supervisor. Advises the Operations Superintendent on what equipment needs to be removed from service.

Answer	d	**Exam Level**	R	**Cognitive Level**	Memory	**Facility**	Hope Creek
Exam Date:	03/12/2002						
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	2	**SRO Group**	2		600000G425
600000	Plant Fire On Site					**Record Number**	49

2.4 Emergency Procedures and Plan

2.4.25 Knowledge of fire protection procedures. 2.9 3.4

Explanation of Answer The Work Control Supervisor or a qualified Equipment operator with no other emergency responsibilities, shall function as the station fire brigade liaison. The liaison shall make recommendations to the OS what equipment needs to be removed from service to mitigate the fire and/or stabilize the plant.

Reference Title

HC.FP-EO.ZZ-0001 note 2.13

NC.NA-AP.ZZ-0005 Attachment 9 Note 3

Learning Objectives

000113E011	a. Summarize the responsibilities of the following personnel: Operations Superintendent Control Room Supervisor/ Field Supervisor Shift Technical Advisor Licensed Operators [RO/PO]
000113E021	a. Determine the following: The level of licensing required for the OS, CRS, and RO/PO. Minimum shift manning requirements for all plant conditions. Normal shift staffing levels. When a person can serve a dual role as CRS/STA or OS/STA

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following:

- A plant startup is in progress.
- Reactor Power on range 4 of IRMs
- Reactor Level at + 46 inches
- Reactor Pressure at 0 psig
- Reactor Temperature at 180°F

The operating Control Rod Drive Pump trips. The Control Room Operator attempted to start the standby CRD Pump and the pump failed to start. Control Rod movement has been suspended.

Which one of the following describes the response of reactor water level and why? (ASSUME NO OPERATOR ACTION)

Reactor Water level will:

a. rise due to the reactor being at the point of adding heat.

b. remain stable due to water expansion from heating overcoming any losses to ambient.

c. remain stable due to water expansion from heating overcoming any losses to RWCU.

d. drop due to RWCU rejecting water for level control.

Answer	d	**Exam Level**	B	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	2				201001A305
201001	Control Rod Drive Hydraulic System							**Record Number**	50

A3. Ability to monitor automatic operations of the CONTROL ROD DRIVE HYDRAULIC SYSTEM including:

A3.05 Reactor water level 2.8 2.8

Explanation of Answer RWCU is normally balanced to reject the 69 gpm makeup from CRD. Without the CRD pump running, RWCU is rejecting at approximately the same rate. RPV level will lower.

Reference Title

HC.OP-IO.ZZ-0003

Learning Objectives

000006E028 From memory, determine why a method of reactor water level control must be available prior to placing the CRDH System in-service including the preferred method of level control, IAW the Lesson Plan.

Material Required for Examination

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: INPO EXAM BANK QID# 16316 Grand Gulf 1 04/01/2000

Given the following:

- The plant was manually scrammed due to prolonged loss of CRD
- A CRD pump has been restarted
- All surveillances are current
- All equipment is operable

Which one of the following PREVENTS control rod withdrawals?

a. Rod Worth Minimizer insert and withdraw errors will result in a control rod withdrawal block signal

b. The Reactor Mode Switch in "Shutdown" inserts a continuous control rod withdrawal block signal

c. The Reactor Mode Switch in "Shutdown" maintains a scram signal on RPS until reset by the operator

d. Rod Block Monitor "Downscale" inserts a control rod withdrawal signal until bypassed

Answer	b	**Exam Level**	R	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	2				201002G421
201002	Reactor Manual Control System							**Record Number**	51

2.4 Emergency Procedures and Plan

2.4.21 Knowledge of the parameters and logic used to assess the status of safety functions including: 3.7 4.3
1. Reactivity control
2. Core cooling and heat removal
3. Reactor coolant system integrity
4. Containment conditions
5. Radioactivity release control.

Explanation of Answer MS in Refuel, one rod withdrawn, second rod selected causes a rod out motion block.

Reference Title

HC.OP-SO.KE-0001

Learning Objectives

000007E008 (R) From memory, explain the interrelationships between the Reactor Manual Control System and the following, IAW the Reactor Manual Control System Lesson Plan:
a. Rod Worth Minimizer
b. Neutron Monitoring System
c. Rod Block Monitor System
d. Mode Switch
e. Refueling System
1) Refueling Bridge
2) Refueling Grapple/Hoists
f. 120 VAC Uninterruptible Power Supply

Material Required for Examination

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: INPO BANK QID#1599 Peach Bottom 2 09/19/1997

The Control Room operator is moving control rods when a ROD DRIFT annunciator is received.

Which one of the following controls caused this annunciator?

a. An odd reed switch is passed while settling from Insert of the control rod one notch using the INSERT PB

b. An even reed switch is passed while settling from Withdrawal of the control rod one notch using the WITHDRAW PB

c. An odd reed switch is passed while settling from Insert of the control rod two notches using the CONTINUOUS INSERT PB

d. An even reed switch is passed while settling from Withdrawal of the control rod two notches using the CONTINUOUS WITHDRAW PB

Answer	c	**Exam Level**	B	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	2				201002K408
201002	Reactor Manual Control System							**Record Number**	52

K4. Knowledge of REACTOR MANUAL CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following:

K4.08	Continuous In rod insertion	3.2	3.2

Explanation of Answer Rod settle function is bypassed during Continuous Insert. A rod settling from continuous insert does not have a rod motion command. Any rod motion detected from odd or even reed switches without a rod motion command causes the Rod Drift alarm.

Reference Title

HC.OP-AR.ZZ-0011 Attachment E3

Learning Objectives

000007E003 (R) Given a labeled diagram/drawing of, or access to, the Reactor Manual Control System controls/indication bezel, summarize the following IAW the Reactor Manual Control System Lesson Plan:

a. The function of each indicator.
b. The condition which will cause the indicator to light or extinguish.
c. The effect of each control on the Reactor Manual Control System.
d. The conditions or permissives required for the control switches to perform their intended function.

Material Required for Examination

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: INPO BANK QID# 17255 Columbia Gen Sta 03/08/1999

Control rod 30-31 is being inserted from position 12 to position 08.

The RO notes that during rod motion the following occur:

- Control Rod 30-31 position indicates "XX" on the 4-Rod display
- Control Rod 30-31 position indicates "XX" on CRIDS
- RPIS Status DATA FAULT light on 10C651 is lit

WHICH ONE of the following describes the status of rod 30-31?

a. Reed switch has failed

b. Scrammed

c. Uncoupled

d. Disarmed

Answer	a	**Exam Level**	B	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems	**RO Group**	2	**SRO Group**	3				201003K405
201003	Control Rod and Drive Mechanism							**Record Number**	53

K4.	Knowledge of CONTROL ROD AND DRIVE MECHANISM design feature(s) and/or interlocks which provide for the following:		
K4.05	Rod position indication	3.2	3.3

Explanation of Answer Control rod indication with a failed open reed switch is XX.

Reference Title

HC.OP-SO.SF-0001 Attachment 1 page 31

Learning Objectives

000007E004 (R) Given plant conditions and a drawing of the controls, instrumentation and/or alarms located in the Control Room, assess the status of the Reactor Manual Control System IAW the Reactor Manual Control System Lesson Plan.

Material Required for Examination

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: INPO EXAM BANK QID# 12303 Limerick 01/20/1998

Given the following:

- An entire startup was performed with an inoperable RWM on 1/12/2001
- With the RWM still inoperable, the reactor scrams on 12/25/2001
- Today's date is 1/6/2002

What RWM requirements must be met to withdraw control rods per Technical Specifications?

a. Startup is NOT allowed until 01/12/2002

b. The RWM must be restored to operability within 8 hours of withdrawal of the first rod

c. Startup may commence as soon as one licensed operator and one technically qualified member of the technical staff are present at the reactor control console until the first twelve control rods are fully withdrawn

d. Startup may commence as soon as one licensed operator and one technically qualified member of the technical staff are present at the reactor control console until power is above 10%

Answer	d	**Exam Level**	S	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems	**RO Group**	2	**SRO Group**	2				201006G112
201006	Rod Worth Minimizer System (RWM) (Plant Specific)							**Record Number**	54

2.1 Conduct of Operations

2.1.12 Ability to apply technical specifications for a system. 2.9 4.0

Explanation of Answer TS 3.1.4.1 is limited by calendar year not rolling year. Startup may commence since this is the first startup of the new year. Once >10 percent power, the LCO is no longer applicable.

Reference Title

HC TS 3.1.4.1

10CFR55.43(2)

Learning Objectives

000009E009 (R) Given plant conditions and access to Technical Specifications:
a. Select those sections, which are applicable to the Rod Worth Minimizer IAW HCGS Technical Specifications.
b. Evaluate Rod Worth Minimizer operability and determine required actions based upon system inoperability IAW HCGS Technical Specifications.
c. Explain the bases for those Technical Specifications associated with the Rod Worth Minimizer IAW HCGS Technical Specifications. (SRO ONLY).

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: Vision Bank QID# Q56372

Given the following:

- A reactor startup is in progress
- Reactor power is 42% after completion of RWM Group Step 500
- The total steam flow signal output from Digital Feed fails to the equivalent of 15% power

Which one of the following describes how control rod motion is effected by the Rod Worth Minimizer (RWM)?

a. The RWM will NOT allow any control rod insertion or withdrawal.

b. The RWM will allow all normal control rod motion until actual power is less than the LPSP.

c. The RWM will apply rod blocks in accordance with the loaded rod sequence.

d. The RWM will allow continued control rod movement only by single notch increments.

Answer	c	**Exam Level**	B	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002

Tier:	Plant Systems	**RO Group**	2	**SRO Group**	2	201006K301
201006	Rod Worth Minimizer System (RWM) (Plant Specific)			**Record Number**		55

K3. Knowledge of the effect that a loss or malfunction of the ROD WORTH MINIMIZER SYSTEM (RWM) will have on following:

K3.01 Reactor manual control system: P-Spec(Not-BWR6) 3.2 3.5

Explanation of Answer

"The RWM will not allow control rod withdrawals if any control rod is withdrawn past its withdraw limit."
The following distractors are incorrect as follows:
"The RWM will allow all normal control rod motion until actual power is less than the LPSP."
Total steam flow from Digital Feed is the signal used by the RWM to determine the LPSP, not actual power.

"The RWM will NOT allow any control rod insertion or withdrawal."
The RWM will allow movements as long as they meet the required sequence programmed into the computer.

"The RWM will allow continued control rod movement only by single notch increments."
The RWM will allow movements as long as they meet the required sequence programmed into the computer.

Reference Title

HC.OP-SO.SF-0003

Learning Objectives

000009E004 (R) Given plant conditions, summarize the interrelationship(s) between the Rod Worth Minimizer and any of the following IAW the RWM Lesson Plan.
a. Rod Position Information System (RPIS)
b. Reactor Manual Control System (RMCS)
c. Feedwater Level Control System
d. Process Computer
e. 120 VAC

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments:	Vision Bank QID# Q53252

Given the following:

- The reactor is operating at 80% power
- Core flow was 68.0 Mlbm/hour
- The "A" Recirculation Pump tripped
- Reactor power stabilized at 57%
- Total core flow stabilized at 43.0 Mlbm/hour
- No operator actions have been taken

Based on plant conditions, which one of the following operator actions are required?
(AB-300 Attachment-1 is attached)

a. Reduce power by single rod scrams

b. Reduce power by lowering recirculation flow

c. Raise flow by restarting the "A" Recirculation Pump

d. Raise flow by raising the speed of the "B" Recirculation Pump

Answer	d	**Exam Level**	R	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems	**RO Group**	2	**SRO Group**	2				202001A404
202001	Recirculation System							**Record Number**	56

A4. Ability to manually operate and/or monitor in the control room:

A4.04	System flow	3.7	3.7

Explanation of Answer: Options are to raise recirc flow to exit or insert rods. Inserting rods by single rod scrams is not allowed.

Reference Title

HC.OP-AB.ZZ-0300

Learning Objectives

0AB300E005	(R) Interpret and apply charts, graphs and tables contained within the Reactor Power Oscillations, Abnormal Operating Procedure.

Material Required for Examination: HC.OP-AB.ZZ-0300 Attachment 1

Question Source: INPO Exam Bank

Question Modification Method: Significantly Modified

Question Source Comments: INPO BANK QID# 15785 Vermont Yankee 01/22/1999

Given the following:

- The plant was operating at 100 percent power
- RWCU pump "A" is C/T for maintenance
- The plant scrammed following a dual recirc pump trip
- RPV level is stable at +30 inches
- RPV Pressure is stable at 920 psig

Based on plant conditions, which one of the following is required?

a. Trip CRD pumps

b. Trip RWCU pumps

c. Increase CRD cooling water flow

d. Reduce RWCU flow from the Recirc Loops

Answer	d	**Exam Level**	B	**Cognitive Level**	Application	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems			**RO Group**	2	**SRO Group**	2		202001K307
202001	Recirculation System							**Record Number**	57

K3. Knowledge of the effect that a loss or malfunction of the RECIRCULATION SYSTEM will have on following:

K3.07 Vessel bottom head drain temperature 2.9 2.9

Explanation of Answer If 1 RWCU pump is running, maximize bottom head drain flow to prevent thermal stratification. This is accomplished by throttling down flow to the recirc loops using valves HV-F102

Reference Title
HC.OP-AB.ZZ-0000 Step S-18
HC.OP-SO.BG-0001 Step 5.5.3.B

Learning Objectives	
000123E004	(R) Given any step of the procedure, determine the reason for performance of that step and/or evaluate expected system response to control manipulations prescribed by that step.
000021E013	Given any system that interrelates with the RWCU System, explain the purpose of that interface IAW the RWCU System Lesson Plan.

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following:

- The plant is operating at 100 percent power
- Total Feedwater Flow signal from Digital Feed is lost

Which one of the following describes the effect of the loss on the plant?

a. Recirc Pump Scoop Tube Lockup

b. Recirc Pump Speed Limiter Full runback

c. Reactor Scram on Low RPV level

d. Reactor Feed Pumps Speed Limited to 2500 RPM

Answer	b	**Exam Level**	R	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				202002K604
202002	Recirculation Flow Control System							**Record Number**	58

K6. Knowledge of the effect that a loss or malfunction of the following will have on the RECIRCULATION FLOW CONTROL SYSTEM:

K6.04 Feedwater flow inputs: BWR-3, 4, 5,6 — 3.5 3.5

Explanation of Answer: Loss of FW flow signal causes RR runback full due to FW flow <20%

Reference Title

HC.OP-SO.BB-0002

HC.OP-SO.AE-0001

Learning Objectives

000019E013	(R) From memory, explain the purpose of each recirc pump runback and list signals which will generate each runback IAW the Lesson Plan.

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following:

- A LOCA concurrent with a partial Station Blackout has occurred
- "A" LPCI is being injected into the RPV
- Reactor Pressure is steady at 100 psig
- Reactor Bldg Temperature is steady at 105°F
- Drywell Temperature is increasing slowly at 285°F
- Fuel Zone indicators are reading -168 inches and steady

Based on the above current conditions, adequate core cooling is ....

a. assured, since actual RPV level is -150"

b. assured, since actual RPV level is -159"

c. NOT assured, since actual RPV level is -170"

d. NOT assured, since actual level is -173"

Answer b **Exam Level** B **Cognitive Level** Application **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Plant Systems **RO Group** 1 **SRO Group** 1 203000A407

203000 RHR/LPCI: Injection Mode (Plant Specific) **Record Number** 59

A4. Ability to manually operate and/or monitor in the control room:

A4.07 Reactor water level 4.5 4.5

Explanation of Answer

Uncompensated level is -168".
-RB Temp Correction: 105° - 75° = 30°
-DW Temp Correction: 285° - 135° = 150°
-TAF curves shift upwards 6" for a 30° F increase in RX Bldg temp
-TAF curves shift down 3" for a 150°F increase in Drywell Temp
-The resulting TAF curve at 100 psig is shifted upwards 3".
-The TAF Curve at 100 psig is -173".
-Shifting upwards 3" places the Curve at -170".
-Indicated level of -168" is 2" above the TAF Curve.
Actual compensated level is therefore 2" above TAF or -159" and therefore Adequate Core Cooling is assured.

Reference Title

Station Aid OPA-92-039

Learning Objectives

00124AE006 (R) Given any step of the procedure, describe the reason for performance of that step and/or expected system response to control manipulation prescribed by that step.

Material Required for Examination Station Aid OPA-92-039

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: vision Bank QID # Q56158

Given the following:

- "B" RHR pump is running in Shutdown Cooling (SDC)
- I&C error initiates "B" LPCI Initiation Logic on High Drywell Pressure signal

Which one of the following describes the "B" RHR Pump, SDC Discharge Valve F015B, and LPCI Injection Valve F017B response?

a. "B" RHR pump trips, F015B closes and F017B opens

b. "B" RHR pump trips, F015B closes and F017B remains closed

c. "B" RHR pump remains running, F015B remains open and F017B opens

d. "B" RHR pump remains running, F015B remains open and F017B remains closed

Answer	c	**Exam Level**	B	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems			**RO Group**	1	**SRO Group**	1		203000K114
203000	RHR/LPCI: Injection Mode (Plant Specific)							**Record Number**	60

K1. Knowledge of the physical connections and/or cause- effect relationships between RHR/LPCI: INJECTION MODE and the following:

K1.14 Shutdown cooling system: Plant-Specific 3.6 3.7

Explanation of Answer: F015B closes on Low RPV level or Hi RPV pressure. Neither is present so F015B stays open. F017B opens on High Drywell Pressure or Low RPV pressure. Since High DW pressure initiation is given, F017B opens. There is no signal present to trip LPCI pump B so B RHR pump remains running.

Reference Title

HC.OP-SO.BC-0001

HC.OP-SO.BC-0002

Learning Objectives

000028E014 Given a copy/mimic of the RHR System controls on 10C650A, predict proper RHR System response during the LPCI mode of operation to include the following, IAW the RHR System Lesson Plan:

a. From memory, state the two automatic initiation signals and setpoints for LPCI initiation, IAW the RHR System Lesson Plan.
b. Determine the pump starting sequence for the LPCI pumps with and without off-site power available, IAW the RHR System Lesson Plan.
c. Determine the actions required to override the LPCI initiation and stop the LPCI pump, IAW the RHR System Lesson Plan.
d. Determine the actions required to override the LPCI initiation and close the LPCI injection valve HV-F017, IAW the RHR System Lesson Plan.
e. Determine the operator actions required to initiate suppression pool cooling during LPCI mode of operation, IAW the RHR System Lesson Plan.
f. Determine the operator actions required to initiate Torus/containment spray during LPCI mode of operation, IAW the RHR System Lesson Plan.

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following:

- The reactor is operating in STARTUP
- RCS temp is 190°F
- RWCU blowdown operation to the Liquid Radwaste System at 60 gpm
- The operator fully opens Blowdown Line Restricting Orifice Bypass Valve (HV-F031)

Which one of the following describes the operational effect of this high bypass flow and how does the operator adjust for the change?

a. The Regenerative Heat Exchanger (RHX) RWCU outlet temperature will lower. Lower RACS flow IAW OP-SO.BG-0001

b. The Regenerative Heat Exchanger (RHX) RWCU outlet temperature will lower. Raise RACS flow IAW OP-SO.BG-0001

c. The Non-Regenerative Heat Exchanger (NRHX) RACS outlet temperature will rise. Lower RACS flow IAW OP-SO.BG-0001

d. The Non-Regenerative Heat Exchanger (NRHX) RACS outlet temperature will rise. Raise RACS flow IAW OP-SO.BG-0001

Answer d **Exam Level** B **Cognitive Level** Comprehension **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Plant Systems **RO Group** 2 **SRO Group** 2 204000A214

204000 Reactor Water Cleanup System **Record Number** 61

A2. Ability to (a) predict the impacts of the following on the REACTOR WATER CLEANUP SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

A2.14 System high temperature 3.2 3.2

Explanation of Answer With increased blowdown flow, less RWCU return flow through the Regen HX causes temp to increase at the RWCU inlet to the NRHX. For the same RACS cooling flow, RACS outlet temp will increase. Step 5.9.4 throttles open 1-ED-V035 to increase RACS flow.

Reference Title

HC.OP-SO.BG-0001 step 5.9.4

Learning Objectives

000021E012	(R) From memory, summarize the effects of RWCU System blowdown operation on the RHX and NRHX's IAW the RWCU System Lesson Plan.

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: Vision Bank QID# Q56399

Given the following:

- The High Pressure Coolant Injection System (HPCI) is operating in Pressure Control alignment
- The HPCI flow controller is in "Automatic"
- HPCI turbine speed is 2450 rpm

Which one of the following describes the response of HPCI turbine speed and system flow if the operator throttles the HPCI Test Bypass To CST Isolation Valve (F008) in the "open" direction for the given conditions?

(Compare the conditions after they stabilize to before the valve was throttled.)

a. -- HPCI turbine speed lowers
-- System flow returns to original value

b. -- HPCI turbine speed lowers
-- System flow goes down

c. -- HPCI turbine speed raises
-- System flow returns to original value

d. -- HPCI turbine speed raises
-- System flow goes up

Answer	a	**Exam Level**	B	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems			**RO Group**	1	**SRO Group**	1		206000A106
206000	High Pressure Coolant Injection System							**Record Number**	62

A1. Ability to predict and/or monitor changes in parameters associated with operating the HIGH PRESSURE COOLANT INJECTION SYSTEM (HPCI) controls including:

A1.06 System flow: BWR-2, 3, 4 — 3.8 3.7

Explanation of Answer: With flow controller in Auto opening, the CST valve reduces the resistance to flow (flow increases for that turbine speed), flow controller reduces turbine speed to return to flow setpoint.

Reference Title

HC.OP-SO.BJ-0001

Learning Objectives

000026E016	(R) Given plant conditions and a drawing of the controls, instrumentation and/or alarms located in the main control room, assess the status of the HPCI System by evaluation of the controls/instrumentation/alarms, IAW the HPCI System Lesson Plan.

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following:

- I&C is performing testing on HPCI TURBINE EXHAUST DIAPHRAGM RUPTURE transmitter PISH-N655A
- A ZERO psig signal is set on the calibration device
- The following Alarms/ Status lights from the testing are received in the Control Room:
    - HPCI SYSTEM OUT OF SERVICE - LIT
    - IN TEST STATUS on Logic Channel "A" - LIT
    - TRIP UNIT IN CAL OR GROSS FAIL on Logic Channel "A" - LIT
    - HPCI TURBINE EXHAUST DIAPHRAGM RUPTURED - Extinguished

With this configuration, how will HPCI respond to an actual HPCI Initiation with a subsequent diaphragm rupture?

a. "A" channel isolation valves only will isolate. HPCI Turbine will NOT trip.

b. "C" channel isolation valves only will isolate. HPCI Turbine will trip.

c. "A" and "C" channel isolation valves will isolate. HPCI Turbine will trip.

d. "A" and "C" channel isolation valves will isolate. HPCI Turbine will NOT trip.

Answer	b	**Exam Level**	R	**Cognitive Level**	Application	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				206000A307
206000	High Pressure Coolant Injection System							**Record Number**	63

A3. Ability to monitor automatic operations of the HIGH PRESSURE COOLANT INJECTION SYSTEM (HPCI) including:

A3.07 Lights and alarms: BWR-2, 3, 4 — 3.9 3.8

Explanation of Answer: "C" Channel transmitters PISH -655C & G will still respond properly to a valid diaphragm rupture. HPCI Turbine will trip. The Channel "A" Logic will not trip due to the 655A transmitter is in test with a zero psig signal. 2 of 2 transmitters are required per logic channel to actuate.

Reference Title

HC.OP-SO.BJ-0001

Learning Objectives

000026E016	(R) Given plant conditions and a drawing of the controls, instrumentation and/or alarms located in the main control room, assess the status of the HPCI System by evaluation of the controls/instrumentation/alarms, IAW the HPCI System Lesson Plan.

Material Required for Examination: J-0650 of HPCI status lights and overhead alarm windows

Question Source: New **Question Modification Method:**

Question Source Comments: INPO BANK QID# 17084 Susquehanna 1 09/30/1999 Concept Used

Given the following:

- The plant is operating at 100% reactor power
- A small instrument line break LOCA occurs
- Drywell pressure is 2.5 psig increasing
- RPV water level reaches -30 inches and is rising
- Drywell pressure trip unit N694F to Core Spray (1 of 2) has failed to trip

Which one of the following describes the response of the Emergency Diesel Generators?

a. All Emergency Diesel Generators start and load onto their respective busses

b. A, C, & D Emergency Diesel Generators start but DO NOT load onto their respective busses

c. A, B, & D Emergency Diesel Generators start and load onto their respective busses

d. All Emergency Diesel Generators start but DO NOT load onto their respective busses

Answer	b	**Exam Level**	B	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				209001K110
209001	Low Pressure Core Spray System							**Record Number**	64

K1. Knowledge of the physical connections and/or cause- effect relationships between LOW PRESSURE CORE SPRAY SYSTEM and the following:

K1.10 Emergency generator 3.7 3.8

Explanation of Answer Both Hi DW Pressure trip units N694B & F must trip to cause an initiation signal to B Core Spray Loop and its respective EDG. RPV level did not reach -129" necessary to cause Level 1 trip. The EDG does not load because LOP is not present.

Reference Title
HC.OP-SO.BE-0001
HC.OP-SO.KJ-0001

Learning Objectives	
000027E009	(R) From memory, summarize/identify the two Core Spray System initiation signals which will also cause an automatic start of the emergency diesel generators, IAW the Core Spray System Lesson Plan.

Material Required for Examination

Question Source:	Facility Exam Bank	**Question Modification Method:**	Direct From Source
Question Source Comments:	Vision Bank QID# Q56486		

Given the following:

- A Loss of Offsite Power occurs followed by a LOCA
- "B" EDG fails to start
- Drywell pressure is 5.7 psig
- Reactor pressure is 440 psig decreasing

Which one of the following describes the effect on the "D" Core Spray Pump and Injection Valve BE-HV-F005B?

a. "D" Core Spray Pump will NOT start but F005B opens

b. "D" Core Spray Pump will NOT start and F005B will NOT open

c. "D" Core Spray Pump starts but F005B will NOT open

d. "D" Core Spray Pump starts and F005B opens

Answer	c	**Exam Level**	B	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				209001K202
209001	Low Pressure Core Spray System							**Record Number**	65

K2. Knowledge of electrical power supplies to the following:

K2.02 Valve power 2.5 2.7

Explanation of Answer Correct answer based on F005B is a B channel valve which will not open in response to the LOCA.

Reference Title

HC.OP-SO.BE-0001

P&ID M-52

Learning Objectives

000027E004 (R) From memory, summarize/identify the sequence of events following receipt of an automatic or manual Core Spray System initiation signal, IAW the Core Spray System Lesson Plan.

000027E005 (R) For a given set of plant conditions, from memory, summarize/identify the interrelationship between the Core Spray System and any of the following, IAW the Core Spray System Lesson Plan:
a. Residual Heat Removal (RHR) System
b. Torus Compartment
c. 4160 VAC Class 1E Distribution System
d. 480 VAC Class 1E Distribution System
e. 125 VDC Class 1E Distribution System
f. Nuclear Boiler
g. Liquid Radwaste System
h. Condensate Storage and Transfer System
i. Primary Containment Instrument Gas (PCIG) System
j. High Pressure Coolant Injection (HPCI) System
k. Condensate Storage Tank
l. Automatic Depressurization System (ADS)
m. Emergency Diesel Generators (EDGs)
n. Nuclear Boiler Instrumentation System
o. Standby Liquid Control (SLC) System

000027E012 (R) Given a labeled diagram/drawing of the Core Spray System controls/indication bezel, IAW the Core Spray System Lesson Plan:
a. Explain the function of each indicator.
b. Assess plant conditions that will cause the indicators to light or extinguish.
c. Determine the effect of each control switch on the Core Spray System.
d. Assess plant conditions or permissives required for the control switches to perform their intended functions.

Material Required for Examination

Question Source:	Facility Exam Bank	**Question Modification Method:**	Editorially Modified
Question Source Comments:	Vision Bank QID# Q56228		

Given the following:

- A plant shutdown is in progress IAW HC.OP-IO.ZZ-0004
- Both Standby Liquid Control pumps are inoperable
- A scram condition is reached and the reactor fails to scram

When will the SLC/RRCS INITIATION FAILURE Overhead Alarm occur?

(Assume RPV level stabilizes at -50 inches and reactor power remains at 8%.)

a. When the RRCS POTENTIAL ATWS alarm occurs

b. When the RRCS CONFIRMED ATWS alarm occurs

c. 30 seconds after the RRCS POTENTIAL ATWS alarm

d. 30 seconds after the RRCS CONFIRMED ATWS alarm

Answer	d	**Exam Level**	R	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems			**RO Group**	1	**SRO Group**	1		211000G410

211000	Standby Liquid Control System	**Record Number**	66
2.4	Emergency Procedures and Plan		
2.4.10	Knowledge of annunciator response procedures.	3.0	3.1

Explanation of Answer 30 seconds after the RRCS CONFIRMED ATWS alarm occurs.-Correct- IAW OHA C1-F1

Reference Title

HC.OP-AR.ZZ-0008 Attachment F1

Learning Objectives

000024E005	(R) Given a set of conditions and a drawing of the controls, instrumentation, and/or alarms located in the main control room, determine the status of the Redundant Reactivity Control System by evaluation of the controls/instrumentation/alarms, IAW the Lesson Plan.

Material Required for Examination

Question Source:	Facility Exam Bank	**Question Modification Method:**	Editorially Modified
Question Source Comments:	Vision Bank QID# Q56844		

Given the following:

- Reactor power is 90%
- HC.OP-IS.BH-0001, Standby Liquid Control Pump AP208 In-service Test, will be performed to check flow rates during power operation.

How is the automatic Reactor Water Cleanup system isolation avoided during this test?

a. The Standby Liquid Control pump is started with the local control switch.

b. The RWCU system must be shutdown and the appropriate isolation valves closed.

c. The breakers for the appropriate RWCU isolation valves must be opened.

d. The fuses for the SLC squib valve firing circuitry must be removed.

Answer	a	**Exam Level**	R	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				211000K105

211000 Standby Liquid Control System — **Record Number** 67

K1. Knowledge of the physical connections and/or cause- effect relationships between STANDBY LIQUID CONTROL SYSTEM and the following:

K1.05 RWCU 3.4 3.6

Explanation of Answer Starting the Standby Liquid Control pump with the local control switch bypasses the RWCU isolation signal.

Reference Title

HC.OP-SO.BH-0001

HC.OP-SO.BG-0001

Learning Objectives

000023E004 Given plant conditions, summarize/identify the interrelationship between the following Systems and the Standby Liquid Control System I.A.W. the Lesson Plan.
a. 480V 1E AC Distribution
b. Core Spray
c. Service Compressed Air
d. Demineralized Water Makeup, Storage & Transfer System
e. Redundant Reactivity Control System
f. Reactor Water Cleanup System
g. Standby Diesel Generator
h. Nuclear Steam Supply Shutoff System
i. Heat Trace

Material Required for Examination

Question Source: Facility Exam Bank — **Question Modification Method:** Editorially Modified

Question Source Comments: Vision Bank QID# Q56772

Which one of the following describes when the Reactor Mode Switch Shutdown position scram may be bypassed?

a. When moving the mode switch from REFUEL to SHUTDOWN

b. When moving the mode switch from SHUTDOWN to REFUEL

c. When testing the "One Rod Out Interlock"

d. When a control rod blade is being uncoupled

Answer	a	**Exam Level**	R	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002

Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1	212000G123

212000	Reactor Protection System	**Record Number**	68

2.1	Conduct of Operations	
2.1.23	Ability to perform specific system and integrated plant procedures during different modes of plant operation.	3.9 4.0

Explanation of Answer: The Reactor Mode Switch Shutdown position scram may be bypassed to move the MS from refuel to Shutdown when all control rods are fully inserted or the reactor is defueled.

Reference Title

HC.OP-SO.SB-0001 Prereq 2.6.2

Learning Objectives

000022E004	(R) From memory, identify the parameters which initiate a Reactor Scram, list the scram initiation setpoints for each identified parameter, and determine when the parameter is bypassed, IAW the Lesson Plan.

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

With the plant operating at rated power, the power supply fuse to a backup scram valve fails creating an open in the supply circuit.

Which one of the following identifies the response of the associated backup scram valve and scram response due to this failure?

a. Valve repositions to trip position but NO scram occurs

b. Valve CANNOT reposition but redundant valves can effect scram if an RPS trip occurs

c. Valve CANNOT reposition and NO scram can occur even if an RPS trip occurs

d. Valve repositions to trip position and a full scram occurs

Answer	b	**Exam Level**	B	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				212000K502
212000	Reactor Protection System							**Record Number**	69

K5. Knowledge of the operational implications of the following concepts as they apply to REACTOR PROTECTION SYSTEM:

K5.02 Specific logic arrangements 3.3 3.4

Explanation of Answer Backup Scram valves are normally de-energized, energize to operate solenoid valves therefore the valve cannot reposition. In conjunction with the valve piping arrangement, the other valve will complete the scram function if a full scram signal is received.

Reference Title

HC.OP-SO.SB-0001

Learning Objectives

000022E009 (R) Given plant conditions, evaluate the response of RPS to an electrical failure, IAW the Lesson Plan.

Material Required for Examination

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: INPO EXAM BANK QID# 11769 Nine Mile Point 1 01/20/1998

A TIP System trace is being taken when an I&C Technician error causes actuation of the NSSSS Channel "A" manual isolation switch.

Which one of the following describes the TIP system response?

a. The TIP Shear Valve automatically fires to cut the detector cable and seal the guide tube.

b. The TIP detectors will automatically withdraw to their "in-shield" position and the TIP Guide Tube Ball Valves automatically close.

c. The TIP Guide Tube Ball Valve automatically closes, cutting the detector cable and sealing the guide tube.

d. No automatic actions occur when only one NSSSS channel manual isolation switch is actuated.

Answer	b	**Exam Level**	B	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems			**RO Group**	3	**SRO Group**	3		215001K105
215001	Traversing In-Core Probe							**Record Number**	70

K1. Knowledge of the physical connections and/or cause- effect relationships between TRAVERSING IN-CORE PROBE and the following:

K1.05 Primary containment isolation system: (Not-BWR1) 3.3 3.4

Explanation of Answer

JUSTIFICATION:
The TIP detectors not in the "in-shield" position will automatically withdraw to their "in-shield" position and the TIP Guide Tube Ball Valves automatically close. Correct
The TIP Shear Valve automatically fires to cut the detector cable and seal the guide tube.
Incorrect - the Shear Valves must be manually initiated.
The TIP Guide Tube Ball Valve automatically closes, cutting the detector cable and sealing the guide tube.
Incorrect - the Ball Valve will not close with the cable inside the valve.
No automatic actions occur when only one NSSSS channel manual isolation switch is actuated.
Incorrect - manual initiation of NSSSS Channel "A" will cause isolation of affected systems, including TIP.

Reference Title

HC.OP-SO.SM-0001 Table SM-017

Learning Objectives

000018E006 (R) From memory explain the response of the TIP System following the receipt of an isolation signal from the Nuclear Steam Supply Shutoff System, IAW the Lesson Plan.

Material Required for Examination

Question Source:	Facility Exam Bank	**Question Modification Method:**	Direct From Source
Question Source Comments:	Vision Bank QID# Q53710		

Given the following:

- The reactor is in Operational Condition 5
- The RPS shorting links are removed
- SRM "A" fails upscale

Which one of the following describes the resulting automatic action?

a. Rod block only

b. 1/2 scram RPS-A only

c. Rod block and 1/2 scram RPS-A only

d. Full scram

Answer	d	**Exam Level**	B	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				215004A303
215004	Source Range Monitor (SRM) System							**Record Number**	71

A3. Ability to monitor automatic operations of the SOURCE RANGE MONITOR (SRM) SYSTEM including:

A3.03 RPS status 3.6 3.5

Explanation of Answer Installation of the Shorting links enables the SRM Hi-Hi rps scrams and changes the coincidence to 1 of 18 taken once

Reference Title

HC.OP-SO.SB-0001

Learning Objectives

000022E014	Given labeled diagrams/drawings of the RPS trip logics, explain the coincidence requirements necessary to generate a reactor scram.

Material Required for Examination

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: INPO BANK QID# 17225 Columbia Gen Sta 03/08/1999

Given the following:

- An I&C Technician is in the middle of SRM "A" Channel Functional Test
- The next section of his procedure contains several discrepancies

Which one of the following changes is PROHIBITED as an "On The Spot Change" to the procedure?

a. Increasing the trip setpoint tolerance to reduce nuisance alarms

b. Minor alterations to a step to clarify that step

c. Changing a step which returns the "B" SRM Mode Switch to the original position

d. Adding a supervisory review signoff

Answer	a	**Exam Level**	S	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems			**RO Group**	1	**SRO Group**	1		215004G206
215004	Source Range Monitor (SRM) System							**Record Number**	72

2.2 Equipment Control

2.2.6 Knowledge of the process for making changes in procedures as described in the safety analysis report. 2.3 3.3

Explanation of Answer Increasing the tolerance of the trip setpoint is a change of intent because it is not being performed to align with Technical Specifications. Clarifying a step is permitted under Attachment 1. Changing the level of oversight is permitted IF the change results in increased oversight. "B" SRM Mode switch is a typo error because the Tech is performing "A" SRM channel testing.

Reference Title

NC.NA-AP.ZZ-0001 Attachment 1 and Form 1

10CFR55.43(3)

Learning Objectives

000113E002 Describe what requirements must be satisfied to make an On-the-Spot change, and the required approval signatures.

Material Required for Examination

Question Source: INPO Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: INPO EXAM BANK QID# 355

Given the following:

- Local Power Range Monitor (LPRM) detector 32-33-C has just failed downscale
- Subsequently, Control Rod 30-31 is selected

Which one of the following describes the effect of the failure on the associated APRM and RBM channels?

The LPRM input:

a. will be automatically bypassed and removed from the APRM only. The APRM and RBM readings will be lower than actual power.

b. will be automatically bypassed and removed from both the APRM and RBM. The APRM and RBM readings will remain the same.

c. will be automatically bypassed and removed from the APRM only. The APRM reading will remain the same and the RBM reading will be lower than actual power.

d. will be automatically bypassed and removed from the RBM only. The APRM and the RBM readings will be lower than actual power.

Answer d **Exam Level** B **Cognitive Level** Comprehension **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Plant Systems **RO Group** 1 **SRO Group** 1 215005K305

215005 Average Power Range Monitor/Local Power Range Monitor System **Record Number** 73

K3. Knowledge of the effect that a loss or malfunction of the APRM/LPRM will have on following:

K3.05 Reactor power indication 3.8 3.8

Explanation of Answer The LPRM must be manually bypassed to remove from the APRM averaging circuit. The LPRM is automatically bypassed in the RBM Count Circuit if the detector is reading <4%. Since the LPRM is still feeding the APRM avg, the indicated avg will be lower. Since the control rod is selected after the LPRM fails downscale so the gain change circuit will null to the now lower APRM reference signal.

Reference Title

HC.OP-SO.SF-0002

Learning Objectives

000017E008 Given the applicable drawing, determine how the Rod Block Monitor (RBM) System interrelates with the following systems:
a. Local Power Range Monitoring (LPRM) System
b. Average Power Range Monitoring (APRM) System
c. Recirculation Flow Units
d. 120 VAC Instrument Power System
e. 120 VAC Un-interruptible Power Supply System
f. Reactor Manual Control System (RMCS)
IAW the Rod Block Monitor (RBM) System Lesson Plan

Material Required for Examination

Question Source: INPO Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: QID# 12556 Limerick 11/10/1995

Given the following:

The plant is operating at full power when the hold down assembly fails on Jet Pumps #1 & #2. This allows the Jet Pump nozzle assembly (Rams Head) to separate from the "B" Recirc Loop piping inside the RPV.

- Annunciators APRM/RBM FLOW REF OFF NORMAL and ROD OUT MOTION BLOCK are also received
- At 10C650, Recirc Pump Discharge Flow indicators are found to be reading 47,000 gpm for "A" Recirc and 54,000 gpm for "B" Recirc

Which one of the following describes how the APRM Flow Units will respond in this situation?

a. Upscale trips from all four (4) Flow units

b. Compare trips from only two (2) Flow units

c. Compare trips from all four (4) Flow units

d. Upscale trips from only two (2) Flow units

Answer	a	**Exam Level**	B	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems			**RO Group**	1	**SRO Group**	1		215005K505
215005	Average Power Range Monitor/Local Power Range Monitor System							**Record Number**	74
K5.	Knowledge of the operational implications of the following concepts as they apply to APRM/LPRM:								
K5.05	Core flow effects on APRM trip setpoints								3.6 3.6

Explanation of Answer: Recirc Flow units are part of the APRMS. This topic was chosen because of the KA similarity to Question #1. Summed flows from both loops will be 101 Kgpm or >111% rated flow upscale setpoint. Rated flow is 45,200 gpm per loop. Comparator trips will not be in because each flow unit sums flow through both loops. They will read high but the same values between channels.

Reference Title

HC.OP-SO.SE-0001 Table SE-001

Learning Objectives

000016E002	(R) Given a labeled diagram of, or access to, the APRMS/Flow Unit controls located on control room panels 10C608/10C651: a. Explain the function of each indicator, IAW the Student Handout. b. Assess the plant conditions that cause each indicator to light or extinguish, IAW the Student Handout. c. Predict the effect of each control switch on the APRMS/Flow Units, IAW the Student Handout. d. Select the conditions or permissives required for the control switches to perform their intended function, IAW the Student Handout.
000016E005	Given a basic diagram of the recirc flow units and an APRM Block Diagram evaluate how the flow signal is developed for use in determining flow biased setpoints, IAW the Student Handout.

Material Required for Examination

Question Source: INPO Exam Bank | **Question Modification Method:** Editorially Modified

Question Source Comments: INPO BANK QID# 675 Duane Arnold 1 05/25/1999

Given the following:

- A Small Break LOCA occurred
- Drywell temp is 450°F and rising
- RPV pressure is 275 psig
- RPV level indication is lost
- 28 control rods are full out
- Suppression Chamber pressure is 10 psig

What is action is required to assure adequate core cooling?

a. Enter HC.OP-EO.ZZ-0206, open SRVs until RPV pressure is below 60 psig

b. Enter HC.OP-EO.ZZ-0206, open at least 5 SRVs

c. Enter HC.OP-EO.ZZ-0206A, open SRVs until RPV pressure is below 275 psig

d. Enter HC.OP-EO.ZZ-0206A, open at least 5 SRVs

Answer	d	**Exam Level**	B	**Cognitive Level**	Application	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems			**RO Group**	1	**SRO Group**	1		216000A208
216000	Nuclear Boiler Instrumentation							**Record Number**	75

A2. Ability to (a) predict the impacts of the following on the NUCLEAR BOILER INSTRUMENTATION; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

A2.08 Elevated containment temperature 3.2 3.4

Explanation of Answer 28 rods are out so EOP -206A is appropriate. 5 SRVs are required to assure adequate flow to assure adequate core cooling.

Reference Title

EOP-206A step RF-5

Learning Objectives

000134E008	(R) Given any step of the procedure, describe the reason for performance of that step and/or expected system response to control manipulations prescribed by that step.

Material Required for Examination EOP Flowcharts without entry conditions

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following:

- 120 VAC UPS TROUBLE Annunciator D3-E3 alarms
- "B" channel ECCS Rosemount Trip Units lose power
- "B" channel analog RPV level indications fail downscale

Which one of the following 120 VAC inverter malfunctions would cause this loss?

a. 1BD-481

b. 1BD-483

c. 1BD-491

d. 1BD-492

Answer	a	**Exam Level**	B	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002

Tier: Plant Systems **RO Group** 1 **SRO Group** 1 216000K201

216000 Nuclear Boiler Instrumentation **Record Number** 76

K2. Knowledge of electrical power supplies to the following:

K2.01 Analog trip system: Plant-Specific 2.8 2.8

Explanation of Answer: AB-136 Caution 4.9 states: "The 1(A-D)D481 Inverters power the ECCS Analog Trip Units and the 1(A-D)D484 Inverters power the Bailey 1E and Non 1E Logic Cabinets." 1BD483 Inverter powers the overhead annunciators which would prevent all overhead alarms from coming in. 1BD492 feeds the BOP Computer. 1BD491 is the B channel essential lighting inverter.

Reference Title

HC.OP-AB.ZZ-0136 Caution 4.9

Learning Objectives

000066E018 From memory, summarize/identify the systems/components supplied by the Uninterruptable Power Supplies System, IAW Attachment 2 of the Lesson Plan.

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following:

- RPV level dropped until RCIC reached an automatic initiation setpoint
- RCIC failed to automatically initiate
- When armed and pressed, RCIC fails to initiate

IAW HC.OP-AB.ZZ-0001 Transient Plant Conditions, which one of the following actions is taken FIRST to manually inject with RCIC?

a. Adjust FIC-R600 RCIC FLOW setpoint to zero %

b. Press and hold the FC-HV-F045 RCIC Steam Supply OPEN PB

c. Ensure OP-219 RCIC VACUUM PUMP is running

d. Ensure BD-HV-F046 Lube Oil Cooling is open

Answer	c	**Exam Level**	S	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				217000A201
217000	Reactor Core Isolation Cooling System (RCIC)							**Record Number**	77

A2. Ability to (a) predict the impacts of the following on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

A2.01 System initiation signal 3.8 3.7

Explanation of Answer HC.OP-AB.ZZ-0001 Attachment 6 step B.1. answer c. is first, followed by d. , b. then a.

Reference Title

HC.OP-AB.ZZ-0001 Attachment 6

10CFR55.43(5)

Learning Objectives

000030E022 (R) Given RCIC turbine control system failures, evaluate and determine the effect on the RCIC system, IAW the RCIC System Lesson Plan.

000030E023 (R) Given any of the following and appropriate control room reference material, evaluate and determine the effect on the RCIC system, IAW the RCIC System Lesson Plan:

a. A given valve opening or closure
b. Loss of DC or AC power supply
c. Inadequate system flow
d. An oil system malfunction
e. Failure of the RCIC Gland Seal Condenser Vacuum Pump
f. Loss of room cooling
g. Rupture disc failure on the RCIC exhaust
h. Steam line break
i. Low condensate storage tank level

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following:

- A reactor scram occurred during a startup
- RCIC started and tripped
- RCIC Turbine Exhaust piping has ruptured
- Reactor pressure is 50 psig and lowering
- A small steam leak in the Drywell is causing Drywell pressure to rise

Which one of the following valves will automatically close if Drywell pressure reaches HI-HI?

a. BD-HV-F031 Torus Suction Isolation Valve

b. BD-HV-F013 Pump Discharge to Feedwater Isolation Valve

c. FC-HV-F062 Vacuum Breaker Isolation Valve

d. FC-HV-F059 Exhaust Line Isolation Valve

Answer	c	**Exam Level**	B	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems			**RO Group**	1	**SRO Group**	1		217000K405

217000	Reactor Core Isolation Cooling System (RCIC)	**Record Number**	78

K4. Knowledge of REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) design feature(s) and/or interlocks which provide for the following:

K4.05	Prevents radioactivity release to auxiliary/reactor building	3.2	3.5

Explanation of Answer RCIC Exhaust vacuum breaker isln valve will close to isolate the piping breach

Reference Title

HC.OP-SO.BD-0001

Learning Objectives

000030E023 (R) Given any of the following and appropriate control room reference material, evaluate and determine the effect on the RCIC system, IAW the RCIC System Lesson Plan:

a. A given valve opening or closure
b. Loss of DC or AC power supply
c. Inadequate system flow
d. An oil system malfunction
e. Failure of the RCIC Gland Seal Condenser Vacuum Pump
f. Loss of room cooling
g. Rupture disc failure on the RCIC exhaust
h. Steam line break
i. Low condensate storage tank level

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following:

- "A" RHR Pump is running in Suppression Pool Cooling mode
- A complete loss of offsite power occurs
- All Emergency Diesel Generators have automatically started and aligned to their respective busses

Which one of the following describes the response of the "A" RHR Test Return Valve BC-HV-F024A?

a. Remains open until CLOSE PB is pressed

b. Remains open until AUTO CLOSE OVERRIDE PB is pressed

c. Receives close signal 5 seconds after bus reenergized

d. Receives close signal 10 seconds after bus reenergized

Answer	a	**Exam Level**	R	**Cognitive Level**	Comprehension	**Facility**	Hope Creek
Exam Date:	03/12/2002						
Tier:	Plant Systems	**RO Group**	2	**SRO Group**	2		219000A301

219000 RHR/LPCI: Torus/Suppression Pool Cooling Mode — **Record Number** 79

A3. Ability to monitor automatic operations of the RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE including:

A3.01 Valve operation 3.3 3.3

Explanation of Answer: On LOP , F024A will remain as is. Once power is restored, valve will remain open because there is no LOCA signal. 5 and 10 second delays are for pump start for normal or emergency power.

Reference Title

HC.OP-SO.BC-0001

Learning Objectives

000028E012 Given a set of conditions and a drawing of the controls, instrumentation and/or alarms located in the main control room, assess the status of the Residual Heat Removal System or its components by evaluation of the controls/instrumentation/alarms IAW the RHR System Lesson Plan.

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments: INPO EXAM BANK QID# 12246 Limerick 01/20/1998 Concept Used

Given the following:

- A LOCA has occurred
- The CRS directs the Suppression Chamber to be vented IAW HC.OP-EO.ZZ-0318 Containment Venting
- Instrument air header pressure is 0 psig

Which one of the following describes how the Hard Torus Vent path valves/dampers are operated IAW HC.OP-EO.ZZ-0318 under these conditions?

a. PCIG opens the inboard damper; the outboard valve is motor operated

b. PCIG opens the inboard damper; the outboard valve is manually operated

c. The inboard damper is motor operated; the outboard valve is motor operated

d. The inboard damper is manually operated; the outboard valve is manually operated

Answer	d	**Exam Level**	R	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems			**RO Group**	1	**SRO Group**	1		223001K613
223001	Primary Containment System and Auxiliaries							**Record Number**	80

K6. Knowledge of the effect that a loss or malfunction of the following will have on the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES:

K6.13 Applicable plant air system/ nitrogen make-up system. 3.2 3.4

Explanation of Answer The inboard damper is normally operated with Instrument Air. Since IA is zero psig, the only way to vent is using Hydraulic Manual operators on both valves.

Reference Title

HC.OP-EO.ZZ-0318

Learning Objectives

000158E004	From memory, describe any/all flow paths established by the performance of each of the 300 series Emergency Operating procedures.

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

The plant is in Cold Shutdown with Shutdown Cooling in service. A single transmitter fails causing a loss of Shutdown Cooling.

Which one of the following caused the trip?

a. N078B RPV Pressure transmitter fails upscale

b. N080A RPV Level transmitter fails upscale

c. N080A RPV Level transmitter fails downscale

d. N078B RPV Pressure transmitter fails downscale

Answer	a	**Exam Level**	B	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				223002K316

223002 Primary Containment Isolation System/Nuclear Steam Supply Shut-Off **Record Number** 81

K3. Knowledge of the effect that a loss or malfunction of the PCIS/NSSSS will have on following:

K3.16 Shutdown cooling system/RHR 3.2 3.3

Explanation of Answer

Justification: N080A RPV Level transmitter fails upscale-Incorrect- failure mode would be downscale
N080A RPV Level transmitter fails downscale-Incorrect- correct mode of failure but requires two detectors per channel to fail
N078B RPV Pressure transmitter fails downscale-Incorrect- wrong failure mode needs to see high pressure not low
N078B RPV Pressure transmitter fails upscale-Correct -Pressure transmitter upscale is single coincidence isolation

Reference Title

HC.OP-SO.SM-0001

Learning Objectives

000045E010 Given a malfunction of the NSSSS, which either isolates or fails to isolate a plant system, evaluate and explain the effects, if any, of that malfunction on each of the following IAW the NSSSS Lesson Plan.
a. Reactor Water Level
b. Fuel Cladding Temperatures
c. Inplant/Offsite Radiological Concerns
d. Reactor Pressure

000045E014 (R) Given a specific parameter, which initiates NSSSS, isolation signals, identify all valves isolated by that parameter and the setpoint at which the isolation signal is generated IAW the NSSSS Lesson Plan.

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments: VISION BANK QID# Q56211

Given the following:

- The plant is operating at 100 percent power
- Main Steam Line (MSL) "A" Flow Transmitter PDT- N086A fails low

Which one of the following describes how Main Steam Lines will be isolated if an actual high flow in the "A" MSL occurs?
(LIMIT YOUR RESPONSE TO MAIN STEAM LINE FLOW INSTRUMENTATION ONLY)

a. "A" and "C" NSSSS logic will trip closing Inboard MSIVs only

b. "A" and "D" NSSSS logic will trip closing Outboard MSIVs only

c. "B" and "C" NSSSS logic will trip closing Inboard and Outboard MSIVs

d. "B" and "D" NSSSS logic will trip closing Inboard and Outboard MSIVs

Answer	c	**Exam Level**	B	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				223002K401
223002	Primary Containment Isolation System/Nuclear Steam Supply Shut-Off							**Record Number**	82

K4. Knowledge of PCIS/NSSSS design feature(s) and/or interlocks which provide for the following:

K4.01 Redundancy 3.0 3.2

Explanation of Answer MSL Flow transmitter failed low in A MSL will prevent A NSSSS logic from tripping. B, C, and D NSSSS flow transmitters on the A MSL will trip in response to an actual high flow but only C and B or C and D can make the MSIVs go closed

Reference Title

HC.OP-SO.SM-0001

HC Tech Specs 3.3.2

Learning Objectives

000045E005 Given a labeled diagram/drawing of NSSSS controls, identify/explain each of the following IAW the NSSSS Lesson Plan.
a. The function of each indicator.
b. The condition which will cause the indicator to light or extinguish.
c. The effect of each control on the NSSSS.

Material Required for Examination P&ID M-41 Sheet 1

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following:

- Drywell pressure is 13 psig
- "A" RHR pump is running
- The CRS orders Drywell Spray initiated on the "A" RHR loop
- The associated Drywell Spray Containment Isolation Valves are opened

Which one of the following describes actions required to establish desired RHR flow?

a. Throttle BC-HV-F048A to obtain 540 gpm flow on FI-4461A

b. Throttle BC-HV-F048A to obtain 10,470 gpm flow on FR-R608A

c. Throttle BC-HV-F003A to obtain 540 gpm flow on FI-4461A

d. Throttle BC-HV-F0003A to obtain 10,470 gpm flow on FR-R608A

Answer	d	**Exam Level**	R	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems	**RO Group**	2	**SRO Group**	1				226001A106
226001	RHR/LPCI: Containment Spray System Mode							**Record Number**	83

A1. Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE controls including:

A1.06 System flow 3.2 3.2

Explanation of Answer

IAW HC.OP-SO.BC-0001 and HC.OP-AB.ZZ-0001 Drywell Spray is throttled to maintain 10,470 gpm loop flow on FR-R608A in the control room.
FI-4461 is Suppression Chamber Spray Flow indication
F048A valve is fully closed.
P&ID M-51-0 Sheet 2 is used by the student to determine which flow indicator monitors Drywell Spray

Reference Title

HC.OP-SO.BC-0001

M-51 Sheet 2

Learning Objectives

000028E011 Given a labeled drawing of, or access to the Residual Heat Removal System controls/indication on 10C650:
a. Explain the function of each indicator IAW the RHR System Lesson Plan.
b. Assess plant conditions which will cause the indicators to light or extinguish IAW the RHR System Lesson Plan.
c. Determine the effect of each control on the RHR System IAW the RHR System Lesson Plan.
d. Assess plant conditions or permissives required for the control switches/pushbuttons to perform their intended functions IAW the RHR System Lesson Plan.

Material Required for Examination P&ID M-51 Sheet 2

Question Source: New **Question Modification Method:**

Question Source Comments:

In response to a steam leak in the Drywell, the "B" loop of RHR was placed in Suppression Chamber Spray and Suppression Pool Cooling.
The "A" loop of RHR was placed in Drywell Spray IAW EOP-102.

Select the automatic system response as Drywell pressure lowers below 1.68 psig. Assume no other operator action.

a. Drywell and Suppression Chamber sprays isolating.

b. Drywell and Suppression Chamber sprays continuing.

c. Drywell spray continuing and Suppression Chamber spray isolating.

d. Drywell spray isolating and Suppression Chamber spray continuing.

Answer	b	**Exam Level**	B	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems	**RO Group**	2	**SRO Group**	1				226001A305

226001	RHR/LPCI: Containment Spray System Mode	**Record Number**	84
A3.	Ability to monitor automatic operations of the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE including:		
A3.05	Containment pressure		4.0 4.0

Explanation of Answer Drywell spray valves need 1.68 psig permissive to open however once started open the valves will stay open. There is no interlock to close the valves on low Drywell pressure.

Reference Title

HC.OP-SO.BC-0001

Learning Objectives

000028E011 Given a labeled drawing of, or access to the Residual Heat Removal System controls/indication on 10C650:

a. Explain the function of each indicator IAW the RHR System Lesson Plan.
b. Assess plant conditions which will cause the indicators to light or extinguish IAW the RHR System Lesson Plan.
c. Determine the effect of each control on the RHR System IAW the RHR System Lesson Plan.
d. Assess plant conditions or permissives required for the control switches/pushbuttons to perform their intended functions IAW the RHR System Lesson Plan.

Material Required for Examination

Question Source: INPO Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: INPO BANK QID# 18032 Pilgrim 1 10/16/1998

Given the following:

- Suppression Chamber pressure is elevated
- The CRS orders Suppression Chamber Sprays placed in service
- While opening the "B" RHR Suppression Chamber Spray Valve F027B, indications are as follows:
    - Yellow OVLD/PWR FAIL light is FLASHING
    - Green CLSD light is EXTINGUISHED
    - Red OPEN light is LIT
    - White OVERRIDDEN light is LIT

Which one of the following describes the valve status?

a. The valve breaker is tripped open. The valve is open with spray flow.

b. The valve breaker is tripped open. The valve is closed.

c. The valve overloads have tripped. The valve is open with spray flow.

d. The valve overloads have tripped. The valve is closed.

Answer	c	**Exam Level**	B	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems	**RO Group**	2	**SRO Group**	2				230000K601
230000	RHR/LPCI: Torus/Suppression Pool Spray Mode							**Record Number**	85

K6. Knowledge of the effect that a loss or malfunction of the following will have on the RHR/LPCI: TORUS/SUPPRESSION POOL SPRAY MODE:

K6.01 A.C. electrical 3.3 3.4

Explanation of Answer Valve motor overloads have tripped causing the yellow flashing light. Red OPEN light Lit means the MOV still has power, therefore the breaker is not tripped.

Reference Title

HC.OP-AR.ZZ-0005 Attachment B1

Learning Objectives

000028E011 Given a labeled drawing of, or access to the Residual Heat Removal System controls/indication on 10C650:

a. Explain the function of each indicator IAW the RHR System Lesson Plan.
b. Assess plant conditions which will cause the indicators to light or extinguish IAW the RHR System Lesson Plan.
c. Determine the effect of each control on the RHR System IAW the RHR System Lesson Plan.
d. Assess plant conditions or permissives required for the control switches/pushbuttons to perform their intended functions IAW the RHR System Lesson Plan.

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following:

- The plant is in Operational Condition 2 with a reactor startup in progress
- One Fuel Pool Cooling Pump, Heat Exchanger and demin are in service
- Fuel Pool inventory is slowly lowering
- Digital alarms and leak detection monitors do NOT identify the source of the leakage
- ALL sump pumps appear to be operating normally
- CST level is stable
- HC.OP-AB.ZZ-0144, Loss of Fuel Pool Inventory/Cooling is entered

Which one of the following actions is required IAW HC.OP-AB.ZZ-0144 Attachment 1.

a. Isolate FPCC Heat Exchanger

b. Enter the Drywell and check for leakage

c. Check Torus Level and verify RHR alignment

d. Isolate RWCU Non-Regenerative Heat Exchanger

Answer	a	**Exam Level**	S	**Cognitive Level**	Application	**Facility** Hope Creek	**Exam Date:** 03/12/2002
Tier:	Plant Systems	**RO Group**	3	**SRO Group**	3		233000G107
233000	Fuel Pool Cooling and Clean-up					**Record Number**	86
2.1	Conduct of Operations						
2.1.7	Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.						3.7 4.4

Explanation of Answer

Justification

CORRECT - Isolate FPCC Heat Exchanger. Following the flow chart of Attachment 1 of HC.OP-AB.ZZ-0144: All conditions enter the first two decision blocks on excessive sump pump operations. The stem stipulates that all sump pumps are operating normally. The third decision block is in regards to the CST level. The stem stipulates that the CST level is normal. The fourth decision block is in regards to RPV head status. The stem stipulates OC2; hence the RPV head is installed. The fifth decision block asks whether FPCC or RHR FPCC Assist is in service. The stem stipulates that FPCC is in service. Therefore, the action is to check for increasing SACS Head Tank Levels - isolate FPCC Hx.

Reference Title

HC.OP-AB.ZZ-0144 Attachment 1

Learning Objectives

0AB144E005	(R) Interpret and apply charts, graphs and tables contained within the Loss Of Fuel Pool Inventory/Cooling, Abnormal Operating Procedure.

Material Required for Examination HC.OP-AB.ZZ-0144 Attachment 1

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: Vision BANK QID# Q61347

Which one of the following describes the bases for the Refueling Platform Main Grapple weight limit interlocks?

a. Prevents release of activity in excess of that contained in a single fuel assembly

b. Prevents damage to core internals from excessive lifting force

c. Prevents damage to hoist safety brake from excessive speed

d. Prevents engaging more than one fuel assembly or control rod blade guide

Answer	b	**Exam Level**	S	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002

Tier:	Plant Systems	**RO Group**	3	**SRO Group**	2	234000G225

234000	Fuel Handling Equipment	**Record Number**	87

2.2	Equipment Control

2.2.25	Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.	2.5	3.7

Explanation of Answer	Tech Spec bases 3/4.9.6

Reference Title

Tech Spec bases 3/4.9.6

10CFR55.43(7)

Learning Objectives

000226E012	(R) Given a scenario of applicable operating conditions and access to Technical Specifications: a. Choose those sections which are applicable to the refueling platform and associated equipment IAW HCGS Technical Specifications. b. Evaluate Refuel Platform operability and determine required actions based upon system operability IAW HCGS Technical Specifications. c. Explain the basis for those Tech Spec items associated with the refuel platform IAW HCGS Technical Specifications. (SRO only)

Material Required for Examination	Tech Specs without Definitions, Safety Limits, and bases

Question Source:	New	**Question Modification Method:**	
Question Source Comments:			

Given the following:

- The plant is operating at 100 percent power
- SRV "B" has inadvertently opened
- Operators attempt to close the SRV

IAW HC.OP-AB.ZZ-0121 "FAILED OPEN SRV/RELIEF VALVE", which one of the following is a positive indication that the SRV has CLOSED?

The "B" SRV...

a. "SV ENRGZ" light extinguishes.

b. Acoustic Monitor green light illuminates.

c. associated power fuse is pulled.

d. tailpipe temperature stabilizes.

Answer	b	**Exam Level**	R	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002

Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1	239002A102
239002	Relief/Safety Valves				**Record Number**	88

A1. Ability to predict and/or monitor changes in parameters associated with operating the RELIEF/SAFETY VALVES controls including:

A1.02 Acoustical monitor noise: Plant-Specific 3.7 3.8

Explanation of Answer Acoustic Monitor green close light on is used to verify the SRV is closed

Reference Title

HC.OP-AB.ZZ-0121 step 4.5

Learning Objectives

0AB121E001	Recognize abnormal indications/alarms and/or procedural requirements for implementing, Failed Open Safety Relief Valve, Abnormal Operating Procedure.
0AB121E004	Explain the reasons for how plant/system parameters respond when implementing, Failed Open Safety Relief Valve, Abnormal Operating Procedure.
0AB121E006	(R) Explain the bases for Subsequent Actions and the information contained in the Discussion Section of Failed Open Safety Relief Valve, Abnormal Operating Procedure.

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

The following plant conditions exist at T = 0:

- Reactor water level is -130 inches
- Reactor pressure is 900 psig
- Drywell pressure is 1.2 psig
- All ECCS pumps are running
- MSIV's are closed

Based on plant conditions, which one of the following describes the response of ADS?

a. ADS will initiate at T = 105 seconds

b. ADS will initiate at T = 300 seconds

c. ADS will initiate at T = 405 seconds

d. ADS will NOT initiate until Drywell pressure increases above 1.68 psig

Answer	c	**Exam Level**	B	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems			**RO Group**	1	**SRO Group**	1		239002A105
239002	Relief/Safety Valves							**Record Number**	89

A1. Ability to predict and/or monitor changes in parameters associated with operating the RELIEF/SAFETY VALVES controls including:

A1.05 Reactor water level 3.7 3.8

Explanation of Answer

Justification: IAW HC.OP-SO.SN-0001

·ADS will NOT initiate until Drywell pressure increases above 1.68 psig.-Incorrect- it will initiate based on the DW Press Bypass Timer

·ADS will initiate at T = 105 seconds.-Incorrect- With the plant conditions as stated, the ADS HIGH DRYWELL PRESSURE BYPASS TIMER will have initiated. This timer is 5 minutes (or 300 seconds). Once this timer is timed out the ADS initiating timer starts. This has a time of 105 seconds. The total time required to reach initiation is 405 seconds

·ADS will initiate at T = 300 seconds. Incorrect- With the plant conditions as stated, the ADS HIGH DRYWELL PRESSURE BYPASS TIMER will have initiated. This timer is 5 minutes (or 300 seconds). Once this timer is timed out the ADS initiating timer starts. This has a time of 105 seconds. The total time required to reach initiation is 405 seconds

·ADS will initiate at T = 405 seconds.-Correct-

Reference Title

HC.OP-SO.SN-0001 section 3.3.1

Learning Objectives

000029E010 (R) From memory, evaluate the interrelationship between the Automatic Depressurization System and the following, IAW the Automatic Depressurization System Lesson Plan:

a. Residual Heat Removal (RHR) and Core Spray Systems
b. Deleted
c. Primary Containment Instrument Gas (PCIG) System
d. 125 VDC Class 1E Distribution System
e. 120 VAC Uninterruptible Power Supply (UPS) Instrumentation

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments:	Vision Bank QID# Q53701

Given the following:

- With power at 22%, a loss of Stator Cooling occurred
- All automatic actions occurred as designed
- The turbine did NOT trip
- HC.OP-AB.ZZ-0138 MAIN TURBINE TRIP/MALFUNCTION has been entered
- There is no time estimate for restoration of Stator Cooling

The decision if and when to trip the Main Turbine is based upon:

a. stator cooling water conductivity prior to the start of the transient.

b. the rate of increase of stator temperatures after the runback is complete.

c. the current plant location on the power to flow map.

d. final main generator field (amps) after the runback has gone to completion.

Answer	a	**Exam Level**	R	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems	**RO Group**	2	**SRO Group**	2				245000K502
245000	Main Turbine Generator and Auxiliary Systems							**Record Number**	90

K5. Knowledge of the operational implications of the following concepts as they apply to MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS:

K5.02	Turbine operation and limitations	2.8	3.1

Explanation of Answer Conductivity readings are not valid following loss of system flow. The conductivity reading prior to the event is key.

Reference Title

HC.OP-AB.ZZ-0138 Step 4.4.8

Learning Objectives

0AB138E006	(R) Explain the bases for Subsequent Actions and the information contained in the Discussion Section of Turbine Generator Trip/Malfunction, Abnormal Operating Procedure.

Material Required for Examination

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: INPO Exam Bank QID# 14179 Peach Bottom 2 03/26/2001

Given the following:

- The plant is operating at 100 percent power
- Instrument air is lost to the following valves:
  - AD-LV-1657-1 Condensate Makeup
  - AD-LV-1657-2 Condensate Reject
  - AD-FV-1677 SCP Suction Reject Bypass

IAW HC.OP-AB.ZZ-0131 "LOSS OF INSTRUMENT AIR AND/OR SERVICE AIR", which one of the following describes the Condensate System response and operator "Contingency Action" necessary to mitigate the event?

a. Condensate Reject Valve fails open; Close Condensate Makeup Bypass Valve to restore Hotwell level

b. Condensate Reject Valve fails open; Open Condensate Makeup Isolation Valve to restore Hotwell level

c. Condensate Makeup Valve fails closed; Open Condensate Makeup Bypass Valve to restore Hotwell level

d. Condensate Makeup Valve fails closed; Close Condensate Makeup Isolation Valve to restore Hotwell level

Answer	c	**Exam Level**	R	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems	**RO Group**	2	**SRO Group**	3				256000A213
256000	Reactor Condensate System							**Record Number**	91

A2. Ability to (a) predict the impacts of the following on the REACTOR CONDENSATE SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

A2.13 Loss of applicable plant air systems 2.9 3.0

Explanation of Answer Contingency action of AB-131. Condensate Makeup valve LV-1657 fails closed. Makeup bypass valve v091 is opened to raise level

Reference Title

HC.OP-AB.ZZ-0131 Attachment 1

M-05 sheet 3

Learning Objectives

0AB131E004	Explain the reasons for how plant/system parameters respond when implementing, Loss Of Instrument Air And/Or Service Air, Abnormal Operating Procedure.

Material Required for Examination P&ID M-05 Sheet 3

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following:

- The plant is at 70% power
- The Main Turbine trips causing Hi Hi levels in the 1A, 2A heaters and 2A drain cooler

Which one of the following describes the valves that isolate for the 1A, 2A heaters and 2A drain cooler?

a. Condensate side inlet and outlet valves

b. The extraction steam isolation valves

c. The High Level Dump valves

d. The Startup and Operating vent valves

Answer	a	**Exam Level**	R	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems	**RO Group**	2	**SRO Group**	3				256000K406
256000	Reactor Condensate System							**Record Number**	92

K4. Knowledge of REACTOR CONDENSATE SYSTEM design feature(s) and/or interlocks which provide for the following:

K4.06 Control of extraction steam 2.8 2.8

Explanation of Answer

JUSTIFICATION:

Condensate system inlet and outlet valves for the 1A, 2A heaters and 2A drain cooler.
Correct - IAW HC.OP-SO.AF-0001, Section 3.2, Limitations.

The extraction steam isolation valve for the 1A, 2A heaters and 2A drain cooler.
Incorrect - there is no extraction steam isolation valve for the 1A, 2A heaters and 2A drain cooler.

The level control valves for the 1A, 2A heaters and 2A drain cooler.
Incorrect - there are no level control valves for feedwater heater 1A; only the normal level control valve for feedwater heater 2A (and drain cooler 2A)--going to feedwater heater 1A--would close

Condensate inlet valve for the 1A, 2A feedwater heaters, allowing condensate flow through the 2A drain cooler.
Incorrect - the condensate inlet and outlet valves close, isolating flow through feedwater heaters 1A and 2A and the drain cooler; they are in series.

Reference Title

HC.OP-SO.AF-0001 Section 3.2.4

Learning Objectives

0AB118E004	Explain the reasons for how plant/system parameters respond when implementing, Loss Of Feedwater Heaters, Abnormal Operating Procedure.
000055E008	(R) From memory, determine the automatic system response associated with the following abnormal conditions for all feedwater heaters, IAW the lesson plan. a. Heater high level b. Heater trip c. Main turbine trip

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments:	Vision QID# Q53238

Given the following:

- The plant is operating at 90 percent power
- Power ascension in progress
- 1BD483 120 VAC inverter output is lost

In addition to entering HC.OP-AB.ZZ-0136 "Loss of 120 VAC Inverter", which other operating procedure must be entered for this condition and why?

a. HC.OP-EO.ZZ-0101 "RPV Control" to stabilize reactor pressure

b. HC.OP-AB.ZZ-0143 "Loss of Overhead Annunciators / Loss of CRIDS" to stabilize RPV Level

c. HC.OP-EO.ZZ-0101A "ATWS RPV Control" to respond to failure to scram

d. HC.OP-AB.ZZ-0153 "Optic Isolator Panel Malfunction" to respond to single Recirc Pump trip

Answer	b	**Exam Level**	S	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems		**RO Group**	1	**SRO Group**	1			259002G432
259002	Reactor Water Level Control System							**Record Number**	93

2.4 Emergency Procedures and Plan

2.4.32 Knowledge of operator response to loss of all annunciators. 3.3 3.5

Explanation of Answer Loss of BD 483 inverter causes loss of overhead annunciators and trip of B RFPT. AB-143 directs operator to stabilize level if 1BD483 inverter is lost. The reactor should not scram.

Reference Title

HC.OP-AB.ZZ-0143

10CFR55.43(5)

Learning Objectives

0AB143E004 Explain the reasons for how plant/system parameters respond when implementing, Loss Of The Overhead Annunciators/Loss of CRIDS, Abnormal Operating Procedure.

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following:

- The plant is in Operational Condition 5
- All RBVS and RBVE fans are running
- FRVS is in a normal standby configuration
- "B" and "D" Emergency Diesel Generators are tagged out for maintenance

A radiological incident on the Refuel Floor causes Refuel Floor Exhaust Radiation to reach 3.1x10-3 uCi/ml.

Which one of the following describes total FRVS recirculation flow one minute after this event? (Assume no operator actions)

a. 0 cfm

b. 90,000 cfm

c. 120,000 cfm

d. 180,000 cfm

Answer	d	**Exam Level**	B	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems		**RO Group**	1	**SRO Group**	1			261000A407
261000	Standby Gas Treatment System							**Record Number**	94
A4.	Ability to manually operate and/or monitor in the control room:								
A4.07	System flow								3.1 3.2

Explanation of Answer: RFE HI HI Start of FRVS is 2.0 E-2 uCi/ml. The value given is above this setpoint. Refuel Floor Exhaust HIHI starts all 6 fans at 30,000 scfm each, since normal power is available.

Reference Title

HC.OP-SO.GU-0001 section 3.1.1

Learning Objectives

000042E006	(R) Given plant conditions, distinguish between the automatic starts and stops associated with the Filtration Recirculation Ventilation System (FRVS) Recirc Fans, IAW the Lesson Plan.

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: Vision QID# Q60662

Given the following:

- The plant is operating at 100 percent power
- A Loss of Offsite power occurs
- Drywell pressure is 5 psig and rising
- "A" Emergency Diesel Generator fails to start

Which one of the following describes the effect on FRVS after 3 minutes?
(Assume NO operator action)

a. Only 3 Recirc Fans and one Vent Fan start

b. Only 4 Recirc Fans and one Vent Fan start

c. Only 3 Recirc Fans and NO Vent Fan start

d. Only 4 Recirc Fans and NO Vent Fan start

Answer	b	**Exam Level**	B	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				261000K603
261000	Standby Gas Treatment System							**Record Number**	95

K6. Knowledge of the effect that a loss or malfunction of the following will have on the STANDBY GAS TREATMENT SYSTEM:

K6.03 Emergency diesel generator system 3.0 3.1

Explanation of Answer A EDG powers A and E FRVS Recirc and A Vent fan. The B,C, D and F Recirc Fans will start as well as the B Vent Fan after Low Flow from A Fans starts the Auto fan.

Reference Title

HC.OP-SO.GU-0001

Learning Objectives

000040E012	(R) Given a set of conditions and a drawing of the controls, instrumentation and/or alarms located in the main control room, identify the status of the Secondary Containment by evaluation of the controls/ instrumentation/alarms IAW the Secondary Containment Lesson Plan.

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following:

- A station blackout has occurred
- All 500 KV Lines to Hope Creek are de-energized

IAW HC.OP-AB.ZZ-0135, which one of the following 500 KV lines is the first to be re-energized to restore power to Hope Creek 13 KV ring bus?

a. Red Lion 5015 Line

b. Deans 5021 Line

c. New Freedom 5023 Line

d. Salem 5037 Line

Answer	d	**Exam Level**	R	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems	**RO Group**	2	**SRO Group**	1				262001K201
262001	A.C. Electrical Distribution							**Record Number**	96
K2.	Knowledge of electrical power supplies to the following:								
K2.01	Off-site sources of power								3.3 3.6

Explanation of Answer AB-135 power restoration strategy is to restore power via the Salem 5037 line and the Salem Gas Turbine

Reference Title

HC.OP-AB.ZZ-0135

Learning Objectives

000065E015	(R) Given plant problems/industry events associated with the Main Power System: a. Discuss the root cause of the plant problem/industry event IAW the Main Power System Lesson Plan. b. Discuss the HCGS design and /or procedural guidelines that mitigate/reduce the likelihood of the problem/event IAW the Main Power System Lesson Plan. c. Discuss the "lesson learned" from this problem/event IAW the Main Power System Lesson Plan.
0AB135E006	(R) Explain the bases for Subsequent Actions and the information contained in the Discussion Section of Station Blackout/Loss Of Offsite Power Diesel Generator Malfunction, Abnormal Operating Procedure.

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following:

- The plant is operating at 100 percent power
- Several overhead annunciators alarm including:
  - 4.16KV FDR TO USS XFMR BRKR MALF
  - 4.16KV SYS INCOMING BRKR MALF
- Yellow INOP control bezels are flashing on 10A401 "A" 1E 4.16KV bus circuit breakers but the equipment does NOT change state
- Reactor power, pressure, and level remain stable

Which one of the following caused the alarms?

a. Loss of power to the Optical Isolator Cabinets

b. Loss of inverter power to the "A" Channel 1E Bailey Cabinets

c. Loss of AC power to the 10A401 bus

d. Loss of DC control power to the 10A401 bus

Answer d **Exam Level** B **Cognitive Level** Comprehension **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Plant Systems **RO Group** 2 **SRO Group** 1 262001K601

262001 A.C. Electrical Distribution **Record Number** 97

K6. Knowledge of the effect that a loss or malfunction of the following will have on the A.C. ELECTRICAL DISTRIBUTION:

K6.01 D.C. power 3.1 3.4

Explanation of Answer A loss of DC control power prevents the breakers from tripping and causes flashing INOP alarm on bezels. A loss of AC power would cause equipment to trip, specifically A RFPT would trip which would cause RPV level to change. Loss of inverter power to Bailey would cause all control room breaker bezels to go dark. Loss of power to the optical Isolation cabinets would not cause flashing INOP bezels.

Reference Title

HC.OP-AR.ZZ-0016

Learning Objectives

000066E026	Given a set of conditions and a drawing of the controls, instrumentation, and/or alarms located in the main control room, assess the status of the 1E AC Power Distribution by evaluation of the controls/instrumentation/alarms IAW the Lesson Plan.
000066E027	Given the loss of a portion of the DC distribution system, evaluate the affect on the 1E AC distribution system IAW the Lesson Plan.

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following:

- The plant is operating at 100 percent power
- HPCI 250 VDC battery has just completed deep discharge rate surveillance testing
- The HPCI Battery charger has been returned to service and associated fuse transfer switch closed
- Overhead annunciator 250 VDC TROUBLE alarm remains ILLUMINATED

Which one of the following is recommended by HC.OP-AB.ZZ-0149 250 VDC MALFUNCTION prior to declaring the HPCI 250 VDC system operable?

a. Perform the Maintenance Weekly Battery Surveillance

b. Place the battery charger timer to the ZERO position

c. Verify the battery charger voltage is less than 268 volts

d. Verify charging current is less than 5 amps

Answer	a	**Exam Level**	B	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems			**RO Group**	2	**SRO Group**	2		263000A101
263000	D.C. Electrical Distribution							**Record Number**	98

A1. Ability to predict and/or monitor changes in parameters associated with operating the D.C. ELECTRICAL DISTRIBUTION controls including:

A1.01 Battery charging/discharging rate 2.5 2.8

Explanation of Answer: AB-149 recommends performing Maint surv HC.MD-ST.PJ-0001 250 Volt Weekly Battery Surveillance to verify battery operability following the battery discharge event. This is based on OE9182 - Battery Inoperable When Exiting LCO where the battery was declared operable before charging restored battery to operable category limits.

Reference Title

HC.OP-AB.ZZ-0149

Learning Objectives

0AB149E001	Recognize abnormal indications/alarms and/or procedural requirements for implementing, 250 VDC System Malfunction, Abnormal Operating Procedure.
0AB149E006	(R) Explain the bases for Subsequent Actions and the information contained in the Discussion Section of 250 VDC System Malfunction, Abnormal Operating Procedure.

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Which one of the following describes the effect on the Class 1E AC Power Distribution System by a loss of Channel "A" 125VDC CLASS 1E Panel 1AD417?

a. Loss of switchgear 10B430 Normal Control Power

b. Loss of switchgear 10B440 Normal Control Power

c. Loss of switchgear 10B450 Alternate Control Power

d. Loss of switchgear 10B460 Alternate Control Power

Answer	c	**Exam Level**	B	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems	**RO Group**	2	**SRO Group**	2				263000K201
263000	D.C. Electrical Distribution							**Record Number**	99

K2. Knowledge of electrical power supplies to the following:

K2.01 Major D.C. loads 3.1 3.4

Explanation of Answer correct answer. 10B450 is "A" Channel switchgear Alternate control power is fed from 1AD417

Reference Title

HC.OP-SO.PK-0001

Learning Objectives

000069E019 (R) Given a D.C. electrical load and access to control room reference material, determine the power supply to the load IAW the DC Distribution System Lesson Plan.

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: Vision Bank QID# Q56195

Given the following:

- The plant is operating at 100 percent power
- "D" SACS Pump is inoperable for scheduled maintenance
- "B" Emergency Diesel Generator (EDG) becomes inoperable

Which one of the following actions is required within one hour?

a. Cross-tie the "D" EDG to the "A" SACS Loop IAW HC.OP-SO.EG-0001

b. Perform AC Power Distribution Lineup - Weekly IAW HC.OP-ST.ZZ-0001

c. Perform "B" SACS Pump In-service Test - Quarterly IAW HC.OP-IS.EG-0002

d. Perform "A" EDG Operability Surveillance Test - Monthly IAW HC.OP-ST.KJ-0001

Answer	b	**Exam Level**	S	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems			**RO Group**	1	**SRO Group**	1		264000G111
264000	Emergency Generators (Diesel/Jet)							**Record Number**	100

2.1 Conduct of Operations

2.1.11 Knowledge of less than one hour technical specification action statements for systems. 3.0 3.8

Explanation of Answer Tech spec 3.8.1.1 action b. requires surveillance requirement 4.8.1.1.1.a. within one hour and at least once per 8 hours thereafter. This surveillance requirement is contained within HC.OP-ST.ZZ-0001

Reference Title

Tech spec 3.8.1.1.1

10CFR55.43(2)

Learning Objectives

000068E030 (R) Given a scenario of applicable conditions and access to technical specifications:
a. Choose those sections which are applicable to the Emergency Diesel Generators, IAW HCGS Technical Specifications.
b. Assess Emergency Diesel Generator operability and determine required actions associated with Diesel Generator inoperability, IAW HCGS Technical Specifications.
c. Explain the basis for those technical specification items associated with the Diesel Generators, IAW HCGS Technical Specifications. (SRO ONLY)

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following:

- A discharge of the Equipment Drain Sample Tank is in progress to the River
- The Liquid Radwaste Discharge Isolation Valve to the Cooling Tower Blowdown automatically closes

Which one of the following conditions would cause this termination?
(Assume no operator action)

a. Cooling Tower Blowdown weir flow rate HI setpoint is reached

b. Liquid Radwaste Effluent sample flow rate HI setpoint is reached

c. Liquid Radwaste Effluent radiation HI setpoint is reached

d. The Cooling Tower Blowdown RMS radiation HI setpoint is reached

Answer	c	**Exam Level**	B	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems		**RO Group**	3	**SRO Group**	3			268000A101
268000	Radwaste							**Record Number**	101

A1. Ability to predict and/or monitor changes in parameters associated with operating the RADWASTE controls including:

A1.01 Radiation level 2.7 3.1

Explanation of Answer Of choices given, only Radwaste Effluent Radiation HI setpoint will cause release isolation and termination. Other answer choices cause alarms but not isolation.

Reference Title

HC.OP-AR.SP-0001 Attachment 5

Learning Objectives

000086E005 (R) From memory list/identify the five conditions that will cause a liquid release to be automatically terminated, IAW the Lesson Plan.

Material Required for Examination

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: INPO EXAM BANK QID# 16367 Grand Gulf 04/01/2000

The Off-Gas Pre-Treatment High Radiation alarm on the RM-11 has just annunciated. In addition to a fuel element failure, which one of the following could cause the high offgas pre-treatment radiation condition?

a. Fire in the offgas holdup pipe

b. Low offgas recombiner temperatures

c. Increased Main Condenser air in-leakage

d. Condensate demineralizer resin intrusion

Answer	d	**Exam Level**	B	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems			**RO Group**	2	**SRO Group**	2		271000K102
271000	Offgas System							**Record Number**	102

K1. Knowledge of the physical connections and/or cause- effect relationships between OFFGAS SYSTEM and the following:

K1.02	Process radiation monitoring system	3.1	3.3

Explanation of Answer Condensate demin resin intrusion into the RPV will cause increased rad levels at the Pre-Treatment Rad monitors. Others affect offgas flows and temperatures. Fire in the holdup pipe is downstream of offgas pretreatment RMS.

Reference Title

HC.OP-AB.ZZ-0100

Learning Objectives

0AB100E006	(R) Explain the bases for Subsequent Actions and the information contained in the Discussion Section of High Reactor Coolant Activity, Abnormal Operating Procedure.

Material Required for Examination

Question Source:	INPO Exam Bank	**Question Modification Method:**	Significantly Modified
Question Source Comments:	INPO EXAM BANK QID# 6550 Dresden 03/11/1996		

Given the following:

- A plant startup is in progress
- The 'A' RPS Motor-Generator Voltage Regulator fails causing generator output voltage to decrease to approximately 100VAC

Which one of the following describes the effect of this condition on the Main Steam Line (MSL) Radiation Monitors?

a. Power is lost to MSL Radiation Monitors RE-N006A and RE- N006C, resulting in an INOP trip

b. Power is lost to MSL Radiation Monitors RE-N006A and RE-N006C, resulting in a HI-HI RAD trip

c. The reduced voltage causes a DOWNSCALE trip of MSL Radiation Monitors RE-N006A and RE-N006C

d. The reduced voltage causes radiation levels for MSL Radiation Monitors RE-N006A and RE-N006B to indicate lower than actual

Answer	a	**Exam Level**	B	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems	**RO Group**	2	**SRO Group**	2				272000K603
272000	Radiation Monitoring System							**Record Number**	103

K6. Knowledge of the effect that a loss or malfunction of the following will have on the RADIATION MONITORING SYSTEM:

K6.03	A.C. power	2.8 3.0

Explanation of Answer

Power is lost to MSL Radiation Monitors RE-N006A and RE-N006C, resulting in an INOP trip.
Correct - when 'A' RPS MG output is less than 108 VAC, the EPA Breakers on the MG output to the 'A' RPS Bus trip on undervoltage, causing a loss of the 'A' RPS bus. This results in an INOP trip of MSL Rad Monitors RE-N006A & C since they are powered from RPS Bus 'A'.

Power is lost to MSL Radiation Monitors RE-N006A and RE-N006C, resulting in a HI-HI RAD trip.
Incorrect - an INOP trip occurs on a loss of power to the MSL Rad Monitors.

The reduced voltage causes a DOWNSCALE trip of MSL Radiation Monitors RE-N006A and RE-N006C.
Incorrect - any voltage reduction would be momentary due to the UV trip of the EPA Breakers; an INOP trip occurs on a loss of power to the MSL Rad Monitors.

The reduced voltage causes radiation levels for MSL Radiation Monitors RE-N006A and RE-N006B to indicate lower than actual.
Incorrect - any voltage reduction would be momentary due to the UV trip of the EPA Breakers; an INOP trip occurs on a loss of power to the MSL Rad Monitors.

Reference Title

HC.OP-SO.SB-0001

HC.OP-SO.SP-0001

Learning Objectives

000221E002

(R) Regarding the main steam line Radiation Monitoring System:
a. From memory, explain the setpoints/conditions associated with a high-high radiation or inoperative trip IAW the Radiation Monitoring System Lesson Plan.
b. Given normal Control Room references, determine the automatic plant actuations/trips which occur as a result of a high-high radiation or inoperative trip IAW the Radiation Monitoring System Lesson Plan.
c. From memory, evaluate the effect of a loss of RPS power IAW the Radiation Monitoring System Lesson Plan.

Material Required for Examination	
Question Source: Facility Exam Bank	**Question Modification Method:** Editorially Modified
Question Source Comments:	VISION BANK QID# Q56950

Given the following:

- The plant is in Operational Condition 2 with a startup in progress
- Reactor pressure is 300 psig
- The lowest Reactor Vessel Metal Temperature thermocouple is reading 150°F

Which one of the following actions is required?

(Use Technical Specification Figure 3.4.6.1-3 provided)

a. Hold reactor pressure at current value for at least 30 minutes

b. Raise reactor pressure at least 20 psig within the next 30 minutes

c. Lower reactor pressure at least 20 psig within the next 30 minutes

d. Raise reactor metal temperature a maximum of 20°F within the next 30 minutes

Answer c **Exam Level** R **Cognitive Level** Application **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Plant Systems **RO Group** 3 **SRO Group** 3 290002A202

290002 Reactor Vessel Internals **Record Number** 104

A2. 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:

A2.02 Overpressurization transient 3.6 3.9

Explanation of Answer Given conditions place the lowest metal temp to the left of the curve in figure 3.4.6.1-3. Lowering reactor pressure moves the operating plot to the right side of the curve. Metal temp must move 37 DegF to get to the right side of the curve

Reference Title

Tech spec 3.4.6.1 figure 3.4.6.1-3

Learning Objectives

00112CE006	(R) Analyze plant conditions and parameters to determine if plant operation is in accordance with the STARTUP FROM COLD SHUTDOWN TO RATED POWER Integrated Operating Procedure, supporting System Operating Procedures and Technical Specifications.

Material Required for Examination Tech spec 3.4.6.1 figure 3.4.6.1-3

Question Source: New **Question Modification Method:**

Question Source Comments:

The plant is operating at 100 percent power

Which one of the following describes the effect on the plant if a piece of foreign material blocked a fuel support piece flow orifice?

a. Core thermal power would decrease

b. Steam quality exiting the reactor vessel will decrease

c. Jet pump net positive suction head would increase

d. Indicated reactor water level will fluctuate

Answer	a	**Exam Level**	R	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems	**RO Group**	3	**SRO Group**	3				290002K303
290002	Reactor Vessel Internals							**Record Number**	105

K3. Knowledge of the effect that a loss or malfunction of the REACTOR VESSEL INTERNALS will have on following:

K3.03 Reactor power 3.3 3.4

Explanation of Answer Low reactor coolant flow past a bundle will drastically increase voids in the channel. Reactor power will decrease.

Reference Title

LP 0301-000.00H-000001-12

Learning Objectives

000228E024	Given a reactor power change analyze that power change and predict how the various reactivity coefficients respond.
000001E008	(R) From memory, explain the reason for core orificing and how this is accomplished, IAW the Lesson Plan.
000001E009	(R) Given plant problems/industry events associated with the Reactor Vessel and Internals: a. Discuss the root cause of the plant problem/industry event IAW the plant/industry event. b. Discuss the HCGS design and/or procedural guidelines that mitigate/reduce the likelihood of the problem/industry event at HCGS IAW the plant/ industry event. c. Discuss the "lessons learned" from this problem/event IAW the plant/industry event.

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments: New

Given the following:

- The plant is operating at 100% power
- "A" Control Room HVAC train and Chilled Water system is running
- A light haze with an acrid odor is noticed in the Main Control Room
- No alarms are received that could explain the origin of the haze and odor
- HC.OP-AB.ZZ-0129, High Radiation, Smoke or Toxic Gases in the Control Room Air Supply is entered

Based on plant conditions, which one of the following is an immediate action IAW HC.OP-AB.ZZ-0129?

a. Verify that the Control Room Supply Ventilation has automatically isolated

b. Verify that the "A" Control Room Emergency Filter Unit automatically started

c. Press the CONTROL ROOM EMER FILTER UNIT A and B OA pushbuttons

d. Press the CONTROL ROOM EMER FILTER UNIT A and B RECIRC MODE pushbuttons

Answer	d	**Exam Level**	B	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Plant Systems	**RO Group**	2	**SRO Group**	2				290003K501
290003	Control Room HVAC							**Record Number**	106

K5. Knowledge of the operational implications of the following concepts as they apply to CONTROL ROOM HVAC

K5.01 Airborne contamination (e.g., radiological, toxic gas, smoke) control 3.2 3.5

Explanation of Answer

Press the CONTROL ROOM EMER FILTER UNIT A and B RECIRC MODE pushbuttons. For a toxic gas in the Control Room Supply, isolate Control Room Ventilation and place CREF in the Recirc Mode.
INCORRECT - Press the CONTROL ROOM EMER FILTER UNIT A and B OA pushbuttons. CREF must be in the Recirc Mode for a toxic gas event.
INCORRECT - Verify that the Control Room Supply Ventilation has automatically isolated. Toxic gas will not automatically isolate Control Room Ventilation. Only high rad.
INCORRECT -Verify that the "A" Control Room Emergency Filter Unit automatically started. Does not automatically start on toxic gas, only high rads.

Reference Title

HC.OP-AB.ZZ-0129

Learning Objectives

0AB129E002 (R) From memory, recall the Immediate Operator Actions for High Radiation, Smoke or Toxic Gases in the Contorl Room Air Supply, Abnormal Operating Procedure.

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: VISION BANK QID# Q61261

Given the following:

- The Oncoming Day Shift Reactor Operator (RO) is returning to shift after 4 days vacation
- Today is March 18, 2002

Which one of the following identifies the date of the earliest Control Room Narrative Log the RO is required to review PRIOR to assuming the watch today?

a. March 10, 2002

b. March 13, 2002

c. March 14, 2002

d. March 15, 2002

Answer	d	**Exam Level**	B	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Generic Knowledge and Abilities		**RO Group**	1	**SRO Group**	1			294001G103
GENERIC								**Record Number**	107

2.1 Conduct of Operations

2.1.3 Knowledge of shift turnover practices. 3.0 3.4

Explanation of Answer: The RO must review 72 hours prior to assuming watch because it is shorter than the 4 days of vacation. The RO must then review the previous 5 days after turnover. Distractors are based on different combinations of these requirements.

Reference Title

SH.OP-AP.ZZ-0107 Section 5.3.1

Learning Objectives

000113E100 Summarize six items covered in a minimum Shift Turnover.

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: INPO BANK QID # 15027 Salem 2/22/99

Given the following:

- The plant is in Operational Condition 5
- Core offload is in progress
- A spent fuel bundle is full up on the main hoist over the core
- The refuel bridge spotter notices the fuel bundle has unlatched and fallen free into the vessel

What operator action is required?

a. Determine South Plant Vent RMS release rate

b. Determine the location of the dropped bundle and inform the Reactor Engineer

c. Re-establish Secondary Containment within 1 hour

d. Evacuate all unnecessary personnel from the Reactor Building

Answer	d	**Exam Level**	S	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Generic Knowledge and Abilities			**RO Group**	1	**SRO Group**	1		294001G114
GENERIC								**Record Number**	108

2.1	Conduct of Operations		
2.1.14	Knowledge of system status criteria which require the notification of plant personnel.	2.5	3.3

Explanation of Answer

Subsequent operator action 4.2 of AB-101
Justification: IAW HC.OP-AB.ZZ-0101 section 4.0
- Evacuate all unnecessary personnel from the Reactor Building.-Correct- IAW HC.OP-AB.ZZ-0101 step 4.2
- Determine South Plant Vent RMS release rate-Incorrect- if Rx Bldg or RF Floor rad levels are rising the FRVS system is placed in service which does not exhaust though the South Plant vent see step 4.5
- Re-establish Secondary Containment within 1 hour- Incorrect- Secondary containment is required to be in place during all fuel moves there is no time limit if lost, actions are to suspend irradiated fuel moves , CORE ALTERATIONS and operations with the potential for draining the vessel. See step 4.3 -Determine the location of the dropped bundle and inform the SRO-Incorrect- there are no actions to determine the location

Reference Title

HC.OP-AB.ZZ-0101 section 4.0

10CFR55.43(5)(7)

Learning Objectives

0AB101E006	(R) Explain the bases for Subsequent Actions and the information contained in the Discussion Section of Irradiated Fuel Damage, Abnormal Operating Procedure.

Material Required for Examination

Question Source:	Facility Exam Bank	**Question Modification Method:**	Editorially Modified
Question Source Comments:	Vision QID# Q60871		

Using provided copy of P&ID M-51 Sheet 2, determine the computer point ID for "A" RHR Heat Exchanger Outlet Temperature.

a. A2020

b. A2380

c. A2381

d. A3132

Answer	c	**Exam Level**	B	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Generic Knowledge and Abilities			**RO Group**	1	**SRO Group**	1		294001G124
GENERIC								**Record Number**	109

2.1 Conduct of Operations

2.1.24 Ability to obtain and interpret station electrical and mechanical drawings. 2.8 3.1

Explanation of Answer

A2020 HX Outlet Conductivity - Incorrect
A2380 HX Inlet Temperature - Incorrect
A2381 HX Outlet Temperature - Correct
A3132 HX Outlet Flow - Incorrect

Reference Title

M-51 Sheet 2

Learning Objectives

000028E012 Given a set of conditions and a drawing of the controls, instrumentation and/or alarms located in the main control room, assess the status of the Residual Heat Removal System or its components by evaluation of the controls/instrumentation/alarms IAW the RHR System Lesson Plan.

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following:

- The reactor is operating at 75% power following a transient
- The reactor engineer reports that the MAXIMUM FRACTION OF LIMITING CRITICAL POWER RATIO (MFLCPR) is 1.001

Which one of the following describes the Technical Specifications required action(s)?

a. The reactor must be in HOT SHUTDOWN within two hours and the NRC notified within one hour.

b. The reactor must be in STARTUP within 6 hours, HOT SHUTDOWN within the following 6 hours, and COLD SHUTDOWN within the subsequent 24 hours.

c. Corrective action be initiated within 15 minutes and the MCPR restored to within the limit within two hours or reduce thermal power to less than 25% of rated within the next four hours.

d. An immediate reactor scram by placing the Reactor Mode Switch in the SHUTDOWN position.

Answer	c	**Exam Level**	S	**Cognitive Level**	Application	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Generic Knowledge and Abilities	**RO Group**	1	**SRO Group**	1				294001G133
GENERIC								**Record Number**	110

2.1 Conduct of Operations

2.1.33 Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications. 3.4 4.0

Explanation of Answer

TECH SPEC 2.1.2 | TECH SPEC 6.7.1

Justification:
The reactor to be in HOT SHUTDOWN within two hours and the NRC operations center notified as soon as possible and in all cases within one hour-Incorrect- exceed SL 2.1.2 MCPR limit of >1.10 MCPR, requires Hot shutdown within 2 hours, 6.7.1.a requires 1 hour notification
No operator action since reactor pressure is greater than 785 psig and core flow is greater than 10% of rated flow.-Incorrect- SL 2.1.2 exceed
Corrective action be initiated within 15 minutes and the MCPR restored to within the limit within two hours or reduce thermal power to less than 25% of rated within the next four hours-Correct- A MFLCPR Value of 1.001 indicates the CPR in the core is slightly exceeding the LCO Limit but below the SL. This is the action for MCPR Thermal Limit exceeding Tech Spec limit. The Safety Limit is not violated.
An immediate reactor scram by placing the Reactor Mode Switch in the SHUTDOWN position.-Incorrect- Hot shutdown in 4 hrs does not requires immediate MSS to Shutdown

Reference Title

TS 2.1.2

10CFR55.43(2)

Learning Objectives

000110E008 (R) Given specific plant operating conditions and a copy of the Hope Creek Generating Station Technical Specifications, evaluate plant/system operability and determine required actions (if any) to be taken. (SRO Only)

Material Required for Examination Tech Specs without Definitions, Safety Limits, and bases

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments:	Vision Bank QID# Q54864

Given the following:

- The plant is operating at 90 percent power
- The #1 Main Turbine Stop Valve has slowly drifted closed
- All Turbine Bypass valves responded full open
- Reactor steam dome pressure stabilizes at 1025 psig
- All other equipment functions properly

Which one of the following actions is required by Technical Specifications?

a. Re-open the Turbine Stop Valve within one hour

b. Reduce reactor thermal power by at least 25 percent within 15 minutes

c. Reduce reactor steam dome pressure by at least 6 psig within 15 minutes

d. Determine MCPR is less than or equal to the EOC-RPT inoperable limit within one hour

Answer c **Exam Level** R **Cognitive Level** Application **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Generic Knowledge and Abilities **RO Group** 1 **SRO Group** 1 294001G133

GENERIC **Record Number** 111

2.1 Conduct of Operations

2.1.33 Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications. 3.4 4.0

Explanation of Answer Reactor steam dome pressure is above the LCO limit of 1020 psig. Reduce pressure to less than 1020 psig within 15 minutes or be in at least hot shutdown within 12 hours

Reference Title

HC Tech Specs 3.4.6.2

Learning Objectives

000051E017 (R) Given a scenario of applicable operating conditions and access to Technical Specifications:
a. Select those sections applicable to the EHC Control Logic System, IAW HCGS Technical Specifications.
b. Evaluate EHC Control Logic System operability and determine required actions based upon system inoperability, IAW HCGS Technical Specifications.
c. Explain the bases for those Technical Specification sections associated with the EHC Logic System, IAW HCGS Technical Specifications. (SRO ONLY)

Material Required for Examination Tech Specs without Definitions, Safety Limits, and bases

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following:

- A plant condition has resulted in a reactor power reduction
- Reactor power is now stable at 50% after the transient
- Chemistry reports that DOSE EQUIVALENT I-131 is 3.0 microcuries/gram

Which one of the following describes the bases that allows plant operation to continue for 48 hours IAW Technical Specifications?

a. To allow for possible Iodine spiking phenomenon

b. To allow for stable Reactor Coolant chemistry sample data

c. To allow for decay of short lived isotopes

d. To allow reasonable time to verify the initial sample results

Answer a **Exam Level** S **Cognitive Level** Memory **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Generic Knowledge and Abilities **RO Group** 1 **SRO Group** 1 294001G134

GENERIC **Record Number** 112

2.1 Conduct of Operations

2.1.34 Ability to maintain primary and secondary plant chemistry within allowable limits. 2.3 2.9

Explanation of Answer TS BASES 3.4.5 Allows for up to 48 hours with a limit of 4 microcuries/gram to accommodate possible iodine spiking which may occur following changes in Thermal Power

Reference Title

Tech Spec bases 3/4.4.5

10CFR55.43(2)

Learning Objectives

000220E006 (R) Given a copy of the Chemistry Daily Summary, a scenario of applicable operating conditions and access to Technical Specifications:
a. Identify the sections, which are applicable to Chemistry Control IAW Technical Specifications. (SRO / STA only)
b. Evaluate the status of the applicable LCOs and summarize the actions required IAW Technical specifications.
c. Explain the bases for those Technical specification sections associated with Chemistry Control IAW Technical Specifications.

Material Required for Examination

Question Source: INPO Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: INPO BANK QID# 1573 Palo Verde 03/24/1997

Given the following:

- Reactor power is 40%
- ALL Turbine Bypass Valves fail OPEN
- The MSIVs FAIL to automatically close

Which one of the following combinations of reactor power and reactor pressure would indicate that a Safety Limit violation occurred?

a. Reactor power is 10% and RPV pressure is 750 psig

b. Reactor power is 20% and RPV pressure is 770 psig

c. Reactor power is 30% and RPV pressure is 775 psig

d. Reactor power is 35% and RPV pressure is 810 psig

Answer	c	**Exam Level**	R	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Generic Knowledge and Abilities	**RO Group**	1	**SRO Group**	1				294001G222
GENERIC								**Record Number**	113

2.2 Equipment Control

2.2.22 Knowledge of limiting conditions for operations and safety limits. 3.4 4.1

Explanation of Answer: Reactor thermal power is greater than 25% with reactor pressure less than 800 psia

Reference Title

Tech Spec 2.1.1

Learning Objectives

000110E001 (R) From memory, state the four (4) Safety Limits in terms of conditions.

Material Required for Examination

Question Source: INPO Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments: INPO BANK QID# 6303 Dresden 2 09/26/1998

The reactor is operating at 100% power. During an operability check of the RCIC system it is discovered that the flow controller FIC-600 on 10C650B will NOT regulate RCIC flow in automatic, however, manual control does function properly.

Based on plant conditions, which one of the following actions is required?

a. No action is required since RCIC flow can be manually controlled

b. Restore the controller to operable status within 7 days, or be in at least hot shutdown within the next 12 hours, and have steam dome pressure less than 150 psig in the following 24 hours

c. Restore the controller to operable status within 14 days, or be in at least hot shutdown within the next 12 hours and have steam dome pressure less than 150 psig in the following 24 hours

d. Restore the controller to operable status within 14 days, or be in at least hot shutdown within the next 12 hours and have steam dome pressure less than 100 psig in the following 24 hours

Answer	c	**Exam Level**	S	**Cognitive Level**	Application	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Generic Knowledge and Abilities	**RO Group**	1	**SRO Group**	1				294001G222
GENERIC								**Record Number**	114

2.2 Equipment Control

2.2.22 Knowledge of limiting conditions for operations and safety limits. 3.4 4.1

Explanation of Answer

Justification:
T/S [amendment 126] 3.7.4 action with RCIC operable is a 14 day LCO with HOT SHUTDOWN within the next 12 hrs and reactor steam dome pressure is 150 psig within the following 24 hrs.

Reference Title

T/S [amendment 126] 3.7.4 action

10CFR55.43(2)

Learning Objectives

000030E013 (R) Given plant conditions and access to Technical Specifications:
a. Select those sections which are applicable to the RCIC System, IAW HCGS Technical Specifications.
b. Evaluate RCIC System operability and determine required actions based upon system inoperability, IAW HCGS Technical Specifications. (SRO Only)
c. Explain the bases for those Technical Specification items associated with the RCIC System, IAW HCGS Technical Specifications.

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments: Vision Exam Bank QID# Q54171

Given the following:

- A complete core offload was completed at the beginning of the refueling outage
- Fuel reload is ready to commence IAW "Fuel Handling Control" Core Alteration Forms. [HC.OP-FR.ZZ-0001]
- All SRM's are fully inserted with the following count rates:
    - "A" – 5 cps
    - "B" – 2 cps
    - "C" – 6 cps
    - "D" – 1 cps

Based on these conditions, which of the following actions is required IAW plant procedures?

a. Spiral Reload may commence with no restrictions as long as any two SRM's are reading > 3 cps

b. A Movable SRM detector must be hooked up to the normal SRM channel instrumentation and be placed in either "B" or "D" quadrant, indicating > 3 CPS prior to Spiral fuel reload commencement

c. Spiral fuel reload may commence in "A" and "C" quadrants only, until either "B" or "D" quadrant SRM is reading > 3 cps at which time complete reload may be commenced

d. Spiral fuel reload may commence up to the first 16 bundles, at which time all four SRM's must read > 3 cps to perform a complete reload

Answer	d	**Exam Level**	S	**Cognitive Level**	Application	**Facility**	Hope Creek	**Exam Date:**	03/12/2002

Tier:	Generic Knowledge and Abilities	**RO Group**	1	**SRO Group**	1	294001G226
GENERIC				**Record Number**		115

2.2	Equipment Control		
2.2.26	Knowledge of refueling administrative requirements.	2.5	3.7

Explanation of Answer: Justification: IAW HC.OP-IO.ZZ-0009 step 5.2.10 directs to verify SRM counts > 3CPS after first 16 bundles when performing Spiral reload. This is to ensure compliance T.S.3.9.2.e.

Reference Title
T.S.3.9.2.e.
HC.OP-IO.ZZ-0009 step 5.2.10
10CFR55.43(6)

Learning Objectives	
00112IE006	(R) Analyze plant conditions and parameters to determine if plant operation is in accordance with the REFUELING OPERATIONS Integrated Operating Procedure, supporting System Operating Procedures and Technical Specifications

Material Required for Examination: Tech Specs without Definitions, Safety Limits, and bases

Question Source: Facility Exam Bank | **Question Modification Method:** Editorially Modified

Question Source Comments: Vision Exam Bank QID# Q58929

Given the following:

- Tech Spec compliance has been verified IAW "Refueling Operations". [HC.OP-IO.ZZ-0009]
- Multiple Control Rod Drive Mechanisms are being removed IAW Technical Specification 3.9.10.2
- Spiral Fuel offload is in progress per directions of Reactor Engineers and Fuel Handling Control Core Alteration forms. [HC.RE-FR.ZZ-0001]
- 14 Fuel Assemblies are remaining in the Vessel

Which one of the following conditions would require a formal declaration of Suspension of Core Alterations as described in plant procedures?

a. Spent Fuel Storage Area Radiation Monitor in alarm while transporting LPRMS through the Cattle Shute

b. All SRMs indicate between 2.1 & 2.6 cps

c. Mode Switch position change from Shutdown to Refuel for Rod Speed adjustments per system operating procedure

d. Refueling Bridge Platform surveillance identifies Frame Mounted hoist up travel stops are out of Technical Specification tolerance

Answer	a	**Exam Level**	R	**Cognitive Level**	Application	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Generic Knowledge and Abilities	**RO Group**	1	**SRO Group**	1				294001G227
GENERIC								**Record Number**	116

2.2 Equipment Control

2.2.27 Knowledge of the refueling process. 2.6 3.5

Explanation of Answer

Justification
HC.OP-IO.ZZ-0009, directs use of NC.NA-AP.ZZ-0049, for direction on formal suspension of fuel handling activities, adverse radiological conditions are one of the criteria.
Additionally, Refuel Radiation Area Alarms is an entry condition for HC.OP-AB.ZZ-0101 "Irradiated Fuel Damage" which directs suspension of all refueling operations.
Other choices are all within the Allowable Technical Specification boundaries for Core Alterations.

Reference Title
NC.NA-AP.ZZ-0049

Learning Objectives	
00112IE004	(R) Apply Precautions, Limitations and Notes while executing the REFUELING OPERATIONS Integrated Operating Procedure

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: Vision Bank QID# Q58930

Given the following:

- The plant has just completed a shutdown for refueling
- Vessel disassembly has commenced
- The I&C department has determined that IRM "A" and SRM "B" have bad detectors and are inoperable

Which one of the following actions must be completed prior to full core offload?

a. Shutdown margin must be demonstrated

b. SRM B will have to be replaced so offload can occur in that quadrant

c. Both instruments must be replaced before any core alterations can begin

d. IRM A must be restored in order to meet the minimum operable channel requirements

Answer	b	**Exam Level**	S	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Generic Knowledge and Abilities		**RO Group**	1	**SRO Group**	1			294001G227
GENERIC							**Record Number**		117

2.2	Equipment Control		
2.2.27	Knowledge of the refueling process.	2.6	3.5

Explanation of Answer Core alterations may only be conducted in a quadrant with an operable SRM detector

Reference Title

Tech Spec 3.9.2

NC.NA-AP.ZZ-0049

10CFR55.43(6)

Learning Objectives

000113E071 a. State the responsibilities of the following personnel:
Refueling SRO.(SRO ONLY)
Refueling Bridge Operator
Control Room refuel Monitor

Material Required for Examination Tech Specs without Definitions, Safety Limits, and bases

Question Source: INPO Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: INPO EXAM BANK QID# 16846 Quad Cities 03/19/1998

Given the following:

- The Core has been off-loaded to the Fuel Pool per NC.NA-AP.ZZ-0049, Conduct of Fuel Handling
- Five control rods are to be replaced with new ABB rods
- All plant conditions have been met for the control rod replacement

In addition to the Refuel Bridge operator and RP Technician, which one of the following must be part of the minimum crew compliment required for the replacement of the control rods?

a. Refueling SRO - Only required to be on site

b. Refueling SRO - Required on the Refuel Floor

c. Reactor Engineer - Required on the Refuel Floor

d. Reactor Engineer - Only required to be on site

Answer	c	**Exam Level**	R	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Generic Knowledge and Abilities	**RO Group**	1	**SRO Group**	1				294001G230
GENERIC								**Record Number**	118

2.2	Equipment Control		
2.2.30	Knowledge of new and spent fuel movement procedures.	2.6	3.5

Explanation of Answer

JUSTIFICATION:
With no fuel in the vessel, no component manipulation within the vessel is considered a Core Alteration IAW Technical Specifications 1.7, and NC.NA-AP.ZZ-0049 sections 5.1.2.A & 7.1
NC.NA-AP.ZZ-0049 stipulates that the minimum crew for non-core alteration fuel handling activities includes the Fuel Crane Operator, Radiation Protection Technician, Reactor Engineer and Spotter. The Reactor Engineer may fulfill the duties of the spotter; hence the minimum permissible crew is three.

Reference Title

NC.NA-AP.ZZ-0049

Learning Objectives

000113E073	State the minimum fuel handling crew requirement for non-core alteration non irradiated fuel handling.

Material Required for Examination

Question Source: Facility Exam Bank | **Question Modification Method:** Direct From Source

Question Source Comments: Vision Exam Bank QID# Q61057

Given the following:

- The plant is in Operational Condition 5
- You are the oncoming Refueling SRO
- The offgoing SRO briefs you of their activities

Which one of the following would constitute a violation of Refuel SRO duties while core alterations are IN PROGRESS?

a. Picking up a fuel bundle after Control Room communications are lost

b. 5 hours of continuous fuel moves

c. Control rod blade removal from an unloaded fuel cell

d. Fuel movement with Fuel Pool water level 1 inch below wave scuppers

Answer a **Exam Level** S **Cognitive Level** Memory **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Generic Knowledge and Abilities **RO Group** 1 **SRO Group** 1 294001G231

GENERIC **Record Number** 119

2.2 Equipment Control

2.2.31 Knowledge of SRO fuel handling responsibilities. 1.6 3.8

Explanation of Answer Continuous communications must be established with the main control room. Core Alts must be suspended if continuous comms lost.

Reference Title

NC.NA-AP.ZZ-0049

10CFR55.43(7)

Learning Objectives

000113E071 a. State the responsibilities of the following personnel:
Refueling SRO.(SRO ONLY)
Refueling Bridge Operator
Control Room refuel Monitor

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments: VISION BANK QID# Q56513

An operator has the following exposure history this year until today:

Deep Dose Equivalent (DDE)	210 mrem
Committed Effective Dose Equivalent (CEDE)	45 mrem
Shallow Dose Equivalent (SDE)	33 mrem

Today, the operator was required to make two entries into the Drywell at 5 percent reactor power:

Entry 1: Gamma dose: 52 mrem; Neutron dose: 24 mrem
Entry 2: Gamma dose: 124 mrem; Neutron dose: 54 mrem

How much radiation exposure is available to the operator without extension if he has to make additional entries?

His available Non-Emergency margin for the year is...

a. 1488 mrem

b. 1521 mrem

c. 1599 mrem

d. 1712 mrem

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Generic Knowledge and Abilities **RO Group** 1 **SRO Group** 1 294001G301

GENERIC **Record Number** 120

2.3 Radiological Controls

2.3.1 Knowledge of 10 CFR 20 and related facility radiation control requirements. 2.6 3.0

Explanation of Answer CORRECT ANSWER. Gamma and neutron dose are summed for DDE. DDE and CEDE are summed together to obtain TEDE. The Dose limit without extension is 2000 mrem/year TEDE

Reference Title

NC.NA-AP.ZZ-0024

Learning Objectives

000113E059 a. Identify the personnel responsible for approval of the following dose extension:
Yearly Dose Extension
Declared Pregnant Women Dose Extension
Lifetime Dose Extension

Material Required for Examination

Question Source: INPO Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: INPO EXAM BANK QUESTION ID #3324. Braidwood 1 09/14/1998

Given the following:

- A LPCI manual injection valve with remote indication requires an Independent Verification (IV)
- The valve is located 8 feet above the grating
- The valve is located in a 90 mrem/hr radiation area
- The temperature in the area is 90 F
- It is estimated that an individual will take 10 minutes to conduct the IV locally

Based on these conditions, which one of the following describes when the "Hands On" IV requirement can be waived?

a. For climbing on equipment concerns

b. For ALARA concerns

c. For heat stress concerns

d. For fall protection concerns

Answer	b	**Exam Level**	R	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Generic Knowledge and Abilities			**RO Group**	1	**SRO Group**	1		294001G302
GENERIC								**Record Number**	121

2.3 Radiological Controls

2.3.2 Knowledge of facility ALARA program. 2.5 2.9

Explanation of Answer IAW NC.NA-AP.ZZ-0005, Attachment 6, section 1.4 areas in which Independent Verification would receive in excess of 10 mrem dose can be verified by alternate means such as status/position indicators. Distrators are based on safety concerns which would make the job longer or more difficult but are not allowed to waive Hands On IV.

Reference Title

NC.NA-AP.ZZ-0005 Section 1.4

Learning Objectives

000113E015 Determine the requirements for Independent Verification

Material Required for Examination

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: INPO EXAM BANK QID# 11424 LaSalle 1 04/21/1997

Given the following:

- The plant is in Operational Condition 4 for a short outage
- During a Drywell inspection, the operator notices some radiation barricade ropes in the area of RWCU Isolation valve BG-HV-F001
- A radiation sign on the ropes reads "Caution; High Radiation Area, RWP Required For Entry" and indicates a MAXIMUM radiation level of 1.10 Rem/hr inside the ropes

Which one of the following additional posting requirements and /or controls are required for this area according to Technical Specifications?

a. The area requires a flashing light in the immediate area as a warning device

b. The area is required to be fenced off and the Drywell Airlock shall be kept locked with the keys kept under the administrative control of the Operations Superintendent

c. The area should be posted as a Very High Radiation Area with continuous electronic surveillance used to control access

d. The area requires a closed circuit TV monitor be installed to give radiation protection personnel continuous monitoring capability

Answer a **Exam Level** S **Cognitive Level** Application **Facility** Hope Creek **Exam Date:** 03/12/2002

Tier: Generic Knowledge and Abilities **RO Group** 1 **SRO Group** 1 294001G304

GENERIC **Record Number** 122

2.3 Radiological Controls

2.3.4 Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized. 2.5 3.1

Explanation of Answer TS 6.12 requires the area roped off, conspicuously posted and a flashing warning light.

Reference Title

Tech Specs 6.12.2

10CFR55.43(4)

Learning Objectives

000113E057 a. State the definition of the following terms:
Contaminated Area
High Radiation Area
Locked High Radiation Area
Radiation Area
Restricted Area
Very High Radiation Area
Airborne Radioactivity Area
Declared Pregnant Woman (DPW)
Total Effective Dose Equivalent (TEDE)

Material Required for Examination Tech Specs without Definitions, Safety Limits, and bases

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: INPO EXAM BANK QID# 5484 Salem Unit 07/08/1996

Given the following:

- The plant is in Operational Condition 3 - Hot Shutdown, going to Cold Shutdown
- The reason for shutdown was excessive unidentified RCS leakage
- Reactor pressure is 920 psig
- Drywell Oxygen concentration is 2.5%
- Primary Containment Release permit has been obtained

Which one of the following is required prior to purging the Primary Containment?

a. A Drywell walkdown must be completed

b. A Valve Open Time permit must be initiated

c. The plant must be in Operational Condition 4 - Cold Shutdown

d. Primary Containment Airlock Operability Test must be performed

Answer	b	**Exam Level**	R	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Generic Knowledge and Abilities	**RO Group**	1	**SRO Group**	1				294001G309
GENERIC								**Record Number**	123

2.3 Radiological Controls

2.3.9 Knowledge of the process for performing a containment purge. 2.5 3.4

Explanation of Answer: correct answer. A Valve open time permit must be prepared to track # of hours that purge valves are open.

Reference Title

HC.OP-SO-GS.0001

HC.OP-AP.ZZ-0104

Learning Objectives

000032E015 (R) Given a scenario of applicable operating conditions and access to Technical Specifications:
a. Select those sections which are applicable to the Containment Inerting and Purge System IAW the Lesson Plan.
b. Evaluate Containment Inerting and Purge System operability and determine required actions based upon system inoperability IAW the Lesson Plan. (SRO Only)
c. Explain the bases for those Technical Specification sections associated with the Containment Inerting a Purge System IAW the Lesson Plan.

Material Required for Examination: Tech Specs without Definitions, Safety Limits, and bases

Question Source: Facility Exam Bank | **Question Modification Method:** Editorially Modified

Question Source Comments: Vision Bank QID# Q55956

Given the following:

Off Gas Radiation 9RX612 and 9RX622 parameters indicate yellow on the RM-11 terminal
Chemistry has been directed to commence sampling

Based on plant conditions, power level should be lowered ...

a. until the GAS RADW CHAR TRTMT PNL 00C367 alarm is clear.

b. to maintain Main Steam Line Rad Monitors less than the HIGH alarm setpoint.

c. until North Plant Vent activity less than the HIGH alarm setpoint.

d. to maintain Off Gas activity less than the RM-11 ALERT alarm setpoint

Answer	d	**Exam Level**	S	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Generic Knowledge and Abilities	**RO Group**	1	**SRO Group**	1				294001G310
GENERIC								**Record Number**	124

2.3 Radiological Controls

2.3.10 Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure. 2.9 3.3

Explanation of Answer

Justification:
Subsequent operator actions for Offgas system High radiation. Reduce reactor power as necessary to maintain the Offgas activity less than the alert alarm setpoint. Correct
Reduce Power to maintain Main Steam Line Rad Monitors less than the high alarm setpoint.-Incorrect-Main Steam Rad not listed in this procedure
Reduce power until the GAS RADW CHAR TRTMT PNL 00367 alarm is clear-Incorrect-RM-11 is the entry not the Charcoal treatment.
Reduce Power to maintain North Plant Vent activity less than the high alarm setpoint. Incorrect-Off Gas RM-11 is the entry not the NPV.

Reference Title

HC.OP-AB.ZZ-0127 rev 5, Section 4.1

10CFR55.43(4)

Learning Objectives

0AB127E006 (R) Explain the bases for Subsequent Actions and the information contained in the Discussion Section of Off-Gas System-High Radiation, Abnormal Operating Procedure.

Material Required for Examination

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: Vision Bank QID# Q62046

Given the following conditions:

- The plant has been operating at 100% power for several weeks
- Main Steam Line (MSL) radiation levels have been averaging 120 mRem but are now slowly trending upwards
- Chemistry reports the higher radiation levels are due to failed fuel
- HC.OP-AB.ZZ-0203, Main Steam Line High Radiation is entered

Based on plant conditions, which one of the following Immediate Operator Actions are required?

a. Place additional Condensate Demineralizers in service if possible

b. Reduce reactor power to maintain MSL radiation levels less than 180 mRem

c. Direct Reactor Water Cleanup flow to the main condenser to reduce coolant activity

d. Scram the reactor and close the Main Steam Isolation Valves when MSL levels reach 180 mRem

Answer	b	Exam Level	B	Cognitive Level	Memory	Facility	Hope Creek	Exam Date:	03/12/2002
Tier:	Generic Knowledge and Abilities		RO Group	1	SRO Group	1			294001G311
GENERIC								Record Number	125

2.3 Radiological Controls

2.3.11 Ability to control radiation releases. 2.7 3.2

Explanation of Answer

JUSTIFICATION:
The immediate actions of HC.OP-AB.ZZ-0203 include:
Ensure all appropriate automatic actions have occurred
Reduce reactor power to clear the MN STM LINE RADIATION HI alarm (1.5X)
Trip the H2O2 Injection System if Radmonitors reach 2.0 and notify Chemistry to verify the system is Shutdown
If a valid MAIN STEAM LINE HI HI Radiation Condition exists, then SCRAM and shut the MSIVs and Drains

CORRECT - Reduce reactor power to maintain MSL radiation levels less than 180 mrem. 180 mr is 1.5X the normal average value of 120 mr stated in the stem. Reducing power is IOA 2 above.
INCORRECT - Direct Reactor Water Cleanup flow to the main condenser to reduce coolant activity. This will not have an appreciable affect on coolant activity unless the RWCU demin is out of service of exhausted neither of which were stated in the stem.
INCORRECT - Scram the reactor and close the Main Steam Isolation Valves when MSL levels are greater than 120 mrem. This is not performed until the MSL HI HI Rad alarm is in at 3xNormal.
INCORRECT - Place additional Condensate Demineralizers in service if possible. While this could be done, it is not directed by AB-203.

Reference Title

HC.OP-AB.ZZ-0203, Section 3.2

Learning Objectives

0AB203E002	(R) From memory, recall the Immediate Operator Actions for Main Steam Line High Radiation, Abnormal Operating Procedure.

Material Required for Examination

Question Source:	Facility Exam Bank	**Question Modification Method:**	Editorially Modified
Question Source Comments:	Vision Bank QID# Q61774		

Which one of the following describes organizational grouping of Abnormal Operating Procedures (ABs) IAW SH.OP-AP.ZZ-0102 "Use of Procedures".

a. 100 series are operational transient procedures

b. 200 series address component failures

c. 300 series apply at all times

d. 000 series address fire and medical emergencies

Answer	c	**Exam Level**	B	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Generic Knowledge and Abilities	**RO Group**	1	**SRO Group**	1				294001G405
GENERIC								**Record Number**	126

2.4 Emergency Procedures and Plan

2.4.5 Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions. 2.9 3.6

Explanation of Answer Justification
IAW SH.OP-AP.ZZ-0102, section 5.5.2

Reference Title

SH.OP-AP.ZZ-0102, section 5.5.2

Learning Objectives

000113E005 a. Summarize the guidelines for the use of the following types of procedures:
Abnormal Operating Procedures
Emergency Operating Procedures
Alarm Response procedures

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments: VISION BANK QID# Q57004

HC.OP-EO.ZZ-0206A is being implemented during an ATWS event.

Which one of the following describes why RCIC injection must be terminated prior to opening SRVs?

a. RCIC is injecting cold water

b. RCIC Turbine damage may occur

c. The Boron concentration will be diluted

d. RPV pressure may NOT be sufficient to drive the RCIC Turbine

Answer	b	**Exam Level**	B	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Generic Knowledge and Abilities	**RO Group**	1	**SRO Group**	1				294001G418
GENERIC								**Record Number**	127

2.4	Emergency Procedures and Plan		
2.4.18	Knowledge of the specific bases for EOPs.	2.7	3.6

Explanation of Answer Reference: HC.OP-EO.ZZ-0206A, Step RF-15 Bases

Reference Title

HC.OP-EO.ZZ-0206A, Step RF-15 Bases

Learning Objectives

000134E008	(R) Given any step of the procedure, describe the reason for performance of that step and/or expected system response to control manipulations prescribed by that step.

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: Vision Exam Bank QID # Q62115

Given the following:

- The Control Room receives a telephoned bomb threat
- The caller states that an explosive device is attached to a hydrogen trailer at the Hydrogen Water Chemical Injection offloading station in the yard south of the power block
- Security is implementing Contingency Procedures
- Security officers confirm the presence of a suspicious device
- No other suspicious activity is observed at this time

Which one of the following describes the time requirement in which the NRC must be notified?

a. Within 15 minutes

b. Within 1 hour

c. Within 4 hours

d. Within 24 hours

Answer	b	**Exam Level**	S	**Cognitive Level**	Comprehension	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Generic Knowledge and Abilities		**RO Group**	1	**SRO Group**	1		294001G428	
GENERIC							**Record Number**	128	

2.4 Emergency Procedures and Plan

2.4.28 Knowledge of procedures relating to emergency response to sabotage. 2.3 3.3

Explanation of Answer The event requires Unusual Event declaration IAW ECG Section 9.1.1. NRC notification is required within I one hour.

Reference Title

HC ECG Section 9.1.1

10CFR55.43(1)

Learning Objectives

Material Required for Examination ECG without Usage Section 1

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: INPO EXAM BANK QID# 1978 Palisades 1 06/14/1999

Which one of the following describes how a scram is verified in accordance with HC.OP-IO.ZZ-0008 Shutdown from Outside the Control Room?

a. HCU nitrogen pressure verified to be less than 800 psig at each HCU

b. Reactor vessel pressure verified less than 920 psig

c. RPS power distribution circuit breakers verified to be open

d. Scram air header pressure verified to be less than 100 psig

Answer	a	**Exam Level**	B	**Cognitive Level**	Memory	**Facility**	Hope Creek	**Exam Date:**	03/12/2002
Tier:	Generic Knowledge and Abilities	**RO Group**	1	**SRO Group**	1				294001G434
GENERIC								**Record Number**	129

2.4 Emergency Procedures and Plan

2.4.34 Knowledge of RO tasks performed outside the main control room during emergency operations including system geography and system implications. 3.8 3.6

Explanation of Answer: The scram is verified outside the control room via HCU Accumulator pressures < 800 psig at each HCU

Reference Title

HC.OP-IO.ZZ-0008

Learning Objectives

00112HE004 (R) Apply Precautions, Limitations and Notes while executing the SHUTDOWN FROM OUTSIDE THE CONTROL ROOM Integrated Operating Procedure.

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments: Vision Exam Bank QID # Q54018