ATTACHMENT (2)

CLEAN COPIES OF SCANTRON ANSWER SHEETS

Nine Mile Point Nuclear Station, LLC
March 21, 2007

FORM NO. 888-E

SCANTRON

NAME		
SUBJECT	TEST NO.	
DATE	HOUR	

TEST RECORD	
PART 1	
PART 2	
TOTAL	

SUBJECTIVE SCORE INSTRUCTOR USE ONLY

PART 1

Question	Key
1	A
2	C
3	A
4	C
5	B
6	B
7	C
8	C
9	A
10	B
11	A
12	C
13	C
14	A
15	C
16	A
17	C
18	D
19	D
20	D
21	D
22	C
23	C
24	D
25	D
26	B
27	B
28	A
29	B
30	B
31	B
32	D
33	D
34	C
35	C
36	B
37	D
38	C
39	A
40	C
41	C
42	C
43	D
44	B
45	D
46	A
47	A
48	B
49	D
50	B

FEED THIS DIRECTION

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

Applicant Information

Name:	
Date: March 16, 2007	Facility/Unit: Nine Mile Point / Unit 1
Region: I	Reactor Type: GE
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination, you must achieve a final grade of at least 80.00 percent. Examination papers will be collected 6 hours after the examination begins.

Applicant Certification

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

____
Applicant's Signature

Results

Examination Value	75 Points
Applicant's Score	____ Points
Applicant's Grade	____ Percent

SUBJECTIVE SCORE INSTRUCTOR USE ONLY

100	90	80	70	60
50	40	30	20	10
9	8	7	6	5
4	3	2	1	0

IMPORTANT

- USE NO. 2 PENCIL ONLY
- MAKE DARK MARKS
- ERASE COMPLETELY TO CHANGE
- EXAMPLE: A B D E

TO USE SUBJECTIVE SCORE FEATURE:
- Mark total possible subjective points
- Only one mark per line on key
- 163 points maximum

EXAMPLE OF STUDENT SCORE

PART 2

NAME			
SUBJECT		TEST NO.	
DATE		HOUR	

TEST RECORD	
PART 1	
PART 2	
TOTAL	

	(T)	(F)			KEY
	1	2	3	4	5
51	A	B	C	D	E
52	A	B	C	D	E
53	A	B	C	D	E
54	A	B	C	D	E
55	A	B	C	D	E
56	A	B	C	D	E
57	A	B	C	D	E
58	A	B	C	D	E
59	A	B	C	D	E
60	A	B	C	D	E
61	A	B	C	D	E
62	A	B	C	D	E
63	A	B	C	D	E
64	A	B	C	D	E
65	A	B	C	D	E
66	A	B	C	D	E
67	A	B	C	D	E
68	A	B	C	D	E
69	A	B	C	D	E
70	A	B	C	D	E
71	A	B	C	D	E
72	A	B	C	D	E
73	A	B	C	D	E
74	A	B	C	D	E
75	A	B	C	D	E
76	A	B	C	D	E
77	A	B	C	D	E
78	A	B	C	D	E
79	A	B	C	D	E
80	A	B	C	D	E
81	A	B	C	D	E
82	A	B	C	D	E
83	A	B	C	D	E
84	A	B	C	D	E
85	A	B	C	D	E
86	A	B	C	D	E
87	A	B	C	D	E
88	A	B	C	D	E
89	A	B	C	D	E
90	A	B	C	D	E
91	A	B	C	D	E
92	A	B	C	D	E
93	A	B	C	D	E
94	A	B	C	D	E
95	A	B	C	D	E
96	A	B	C	D	E
97	A	B	C	D	E
98	A	B	C	D	E
99	A	B	C	D	E
100	A	B	C	D	E

FEED THIS DIRECTION

EXAMINATION ANSWER KEY
(ANSWERS ONLY)

NRC Reactor Operator Written Examination Nine Mile Point Unit 1 2007

1		**ID: NRC 2006 RO 1**	**Points: 1.00**
	Answer: A		
2		**ID: NRC 2006 RO 2**	**Points: 1.00**
	Answer: C		
3		**ID: NRC 2006 RO 3**	**Points: 1.00**
	Answer: A		
4		**ID: NRC 2006 RO 4**	**Points: 1.00**
	Answer: C		
5		**ID: NRC 2006 RO 5**	**Points: 1.00**
	Answer: B		
6		**ID: NRC 2006 RO 6**	**Points: 1.00**
	Answer: B		
7		**ID: NRC 2006 RO 7**	**Points: 1.00**
	Answer: C		
8		**ID: NRC 2006 RO 8**	**Points: 1.00**
	Answer: C		
9		**ID: NRC 2006 RO 9**	**Points: 1.00**
	Answer: A		
10		**ID: NRC 2006 RO 10**	**Points: 1.00**
	Answer: B		

11			**ID: NRC 2006 RO 11**	**Points: 1.00**
	Answer:	A		
12			**ID: NRC 2006 RO 12**	**Points: 1.00**
	Answer:	C		
13			**ID: NRC 2006 RO 13**	**Points: 1.00**
	Answer:	C		
14			**ID: NRC 2006 RO 14**	**Points: 1.00**
	Answer:	A		
15			**ID: NRC 2006 RO 15**	**Points: 1.00**
	Answer:	C		
16			**ID: NRC 2006 RO 16**	**Points: 1.00**
	Answer:	A		
17			**ID: NRC 2006 RO 17**	**Points: 1.00**
	Answer:	C		
18			**ID: NRC 2006 RO 18**	**Points: 1.00**
	Answer:	D		
19			**ID: NRC 2006 RO 19**	**Points: 1.00**
	Answer:	D		
20			**ID: NRC 2006 RO 20**	**Points: 1.00**
	Answer:	D		

EXAMINATION ANSWER KEY
(ANSWERS ONLY)

NRC Reactor Operator Written Examination Nine Mile Point Unit 1 2007

21 **ID: NRC 2006 RO 21** **Points: 1.00**

Answer: D

22 **ID: NRC 2006 RO 22** **Points: 1.00**

Answer: C

23 **ID: NRC 2006 RO 23** **Points: 1.00**

Answer: C

24 **ID: NRC 2006 RO 24** **Points: 1.00**

Answer: D

25 **ID: NRC 2006 RO 25** **Points: 1.00**

Answer: D

26 **ID: NRC 2006 RO 26** **Points: 1.00**

Answer: B

27 **ID: NRC 2006 RO 27** **Points: 1.00**

Answer: B

28 **ID: NRC 2006 RO 28** **Points: 1.00**

Answer: A

29 **ID: NRC 2006 RO 29** **Points: 1.00**

Answer: B

30 **ID: NRC 2006 RO 30** **Points: 1.00**

Answer: B

EXAMINATION ANSWER KEY
(ANSWERS ONLY)

NRC Reactor Operator Written Examination Nine Mile Point Unit 1 2007

31		**ID: NRC 2006 RO 31**	**Points: 1.00**
	Answer: B		
32		**ID: NRC 2006 RO 32**	**Points: 1.00**
	Answer: D		
33		**ID: NRC RO 33 REV 1**	**Points: 1.00**
	Answer: D		
34		**ID: NRC 2006 RO 34**	**Points: 1.00**
	Answer: C		
35		**ID: NRC 2006 RO 35**	**Points: 1.00**
	Answer: C		
36		**ID: NRC 2006 RO 36**	**Points: 1.00**
	Answer: B		
37		**ID: NRC 2006 RO 37**	**Points: 1.00**
	Answer: D		
38		**ID: NRC RO 38**	**Points: 1.00**
	Answer: C		
39		**ID: NRC 2006 RO 39**	**Points: 1.00**
	Answer: A		
40		**ID: NRC 2006 RO 40**	**Points: 1.00**
	Answer: C		

EXAMINATION ANSWER KEY
(ANSWERS ONLY)

NRC Reactor Operator Written Examination Nine Mile Point Unit 1 2007

41		**ID: NRC 2006 RO 41**	**Points: 1.00**
	Answer: C		
42		**ID: NRC 2006 RO 42**	**Points: 1.00**
	Answer: C		
43		**ID: NRC 2006 RO 43**	**Points: 1.00**
	Answer: D		
44		**ID: NRC 2006 RO 44**	**Points: 1.00**
	Answer: B		
45		**ID: NRC 2006 RO 45**	**Points: 1.00**
	Answer: D		
46		**ID: NRC 2006 RO 46**	**Points: 1.00**
	Answer: A		
47		**ID: NRC 2006 RO 47**	**Points: 1.00**
	Answer: A		
48		**ID: NRC 2006 RO 48**	**Points: 1.00**
	Answer: B		
49		**ID: NRC 2006 RO 49**	**Points: 1.00**
	Answer: D		
50		**ID: NRC 2006 RO 50**	**Points: 1.00**
	Answer: B		

EXAMINATION ANSWER KEY
(ANSWERS ONLY)

NRC Reactor Operator Written Examination Nine Mile Point Unit 1 2007

51	**ID: NRC 2006 RO 51**	**Points: 1.00**
Answer: D		
52	**ID: NRC 2006 RO 52**	**Points: 1.00**
Answer: D		
53	**ID: NRC 2006 RO 53**	**Points: 1.00**
Answer: C		
54	**ID: NRC 2006 RO 54**	**Points: 1.00**
Answer: A		
55	**ID: NRC 2006 RO 55**	**Points: 1.00**
Answer: D		
56	**ID: NRC 2006 RO 56**	**Points: 1.00**
Answer: B		
57	**ID: NRC 2006 RO 57**	**Points: 1.00**
Answer: D		
58	**ID: NRC 2006 RO 58**	**Points: 1.00**
Answer: B		
59	**ID: NRC 2006 RO 59**	**Points: 1.00**
Answer: C		
60	**ID: NRC 2006 RO 60**	**Points: 1.00**
Answer: A		

EXAMINATION ANSWER KEY
(ANSWERS ONLY)

NRC Reactor Operator Written Examination Nine Mile Point Unit 1 2007

61		**ID: NRC 2006 RO 61**	**Points: 1.00**
	Answer: A		
62		**ID: NRC 2006 RO 62**	**Points: 1.00**
	Answer: B		
63		**ID: NRC 2006 RO 63**	**Points: 1.00**
	Answer: C		
64		**ID: NRC 2006 RO 64**	**Points: 1.00**
	Answer: D		
65		**ID: NRC 2006 RO 65**	**Points: 1.00**
	Answer: A		
66		**ID: NRC 2006 RO 66**	**Points: 1.00**
	Answer: B		
67		**ID: NRC 2006 RO 67**	**Points: 1.00**
	Answer: D		
68		**ID: NRC 2006 RO 68**	**Points: 1.00**
	Answer: C		
69		**ID: NRC 2006 RO 69**	**Points: 1.00**
	Answer: C		
70		**ID: NRC 2006 RO 70**	**Points: 1.00**
	Answer: C		

EXAMINATION ANSWER KEY (ANSWERS ONLY)

NRC Reactor Operator Written Examination Nine Mile Point Unit 1 2007

71		**ID: NRC 2006 RO 71**	**Points: 1.00**
	Answer: A		
72		**ID: NRC 2006 RO 72**	**Points: 1.00**
	Answer: D		
73		**ID: NRC 2006 RO 73**	**Points: 1.00**
	Answer: A		
74		**ID: NRC 2006 RO 74**	**Points: 1.00**
	Answer: D		
75		**ID: NRC 2006 RO 75**	**Points: 1.00**
	Answer: D		

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

Applicant Information

Name:	
Date: March 16, 2007	Facility/Unit: Nine Mile Point / Unit 1
Region: I	Reactor Type: GE
Start Time:	Finish Time:

Instructions

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Applicant Certification

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

Applicant's Signature

Results

Examination Value	75 Points
Applicant's Score	____ Points
Applicant's Grade	____ Percent

EXAMINATION ANSWER KEY

Reactor Operator Written Examination Nine Mile Point Unit 1 2007

1 **USERID: NRC 2006 RO 1** **SYSID: 21226** **Points: 1.00**

The plant is in a refueling outage with the following:

- Shift Briefing is in progress

Which one of the following identifies the Operator At The Controls (OATC) responsibilities per the Operations Manual OM.2, Conduct of Operations?

A. Must continue panel walkdowns even during brief and should know status of control room alarms for in-service systems only.

B. Must continue panel walkdowns even during brief and should know status of control room alarms for ALL safety related systems.

C. Panel walkdowns are NOT required during brief and should know status of control room alarms for in-service systems only.

D. Panel walkdowns are NOT required during brief and should know status of control room alarms for ALL safety related systems.

Answer: A

Answer Explanation: A. is correct. Per OM.2.1.2 h. during shift briefings, it is expected that OATC maintains full board awareness and continues panel walkdowns. If needed, the OATC may be briefed separately. Step j states, that during outages, the control room operators should know the status of control room annunciators associated with in-service systems only.
B. is incorrect because the operators are not required or expected to know the status of ALL safety related systems, during an outage, but only those that are in-service.
C. and D are incorrect because panel walkdowns are not suspended during shift briefing. They are still required to be performed by the OATC.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

EXAMINATION ANSWER KEY

Reactor Operator Written Examination Nine Mile Point Unit 1 2007

Question 1 Details

Question Type:	Multiple Choice
Topic:	NRC RO 1
System ID:	21226
User ID:	NRC 2006 RO 1
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 27, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Technical References: Operations Manual OM.2.2.1.2 h and j. **Reference Provided: NONE** Enabling Objective: O3-OPS-006-343-3-51 EO-1.2

Question 1 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

MANUAL
- OPERATIONS MANUAL Rev. NA

NUREG 1123 KA Catalog Rev. 2
- G2.1.2 3/4 Knowledge of operator responsibilities during all modes of plant operation

Level of Difficulty
- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level
- 1 Memory/Fundamental Knowledge

10CFR55
- 41(b)(10)

Question Source
- New

2 **USERID: NRC 2006** **SYSID: 21227** **Points: 1.00**
RO 2

The plant is in a Refueling Outage with the following:

- Valve lineup of the Backfill System is being performed
- Valve 28. 1-43, BV-REF LEG FILL, has a YELLOW Clearance tag attached to the valve
- Clearance section is held by the Mechanical Maintenance Department
- Clearance indicates the valve is CLOSED and operating the valve is prohibited
- Normal position per the valve lineup indicates that the valve should be open

Which one of the following describes the actions required to complete this valve lineup, per N1-VLU-01?

A. Update the Clearance Section Tag Restoration Lineup Sheet then reposition the valve.

B. Indicate the Clearance Section number on the valve lineup and reposition the valve.

C. Leave the valve in its present position and indicate the Clearance Section Number in the INITIALS/DATE column.

D. Leave the valve in its present position, notify the CSO and WEC of the discrepancy and note the discrepancy.

Answer: C

Answer Explanation:	Answer:	C. indicate Clearance Section Number in INITIALS/DATE is **correct** - Per N1-VLU-01, Section 5.1.4, the clearance section tag is not to be removed and the valve is not to be manipulated unless the specific conditions for manipulation of the valve are satisfied.
	Distractor:	D. is incorrect. Per Section 5.1.2 of N1-VLU-01, valves that are out of expected position with a clearance section tag installed are NOT discrepancies and their is no requirement to notify the Work Execution Center (WEC).
	Distractor:	A and B are incorrect. Operation of the valve is prohibited. GAP-OPS-02 step 3.5.4 requires that to manipulate the component, it cannot be prohibited by the clearance section. Repositioning would also require CSO permission.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 2 Details

Question Type:	Multiple Choice
Topic:	NRC RO 2
System ID:	21227
User ID:	NRC 2006 RO 2
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Dec 19, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	21068
Comments:	**Reference Provided: NONE** Enabling Objective: O3-OPS-006-VLU-3-01 EO-1.5

Question 2 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2
- G2.1.29 3.4/3.3 Knowledge of how to conduct and verify valve lineups

Level of Difficulty
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level
- 1 Memory/Fundamental Knowledge

10CFR55
- 41(b)(10)

Question Source
- Bank

The plant is operating at 100% power with the following:

- Process Computer is taken out of service for troubleshooting at 0100
- Computer is expected to be out of service until 1600

Which one of the following administrative compensatory actions is required?

A. An additional Non-Licensed Operator must be added to shift compliment to comply with Tech Specs.

B. An additional Shift Technical Advisor must be added to shift compliment to comply Tech Specs.

C. An additional Reactor Operator must be added to shift compliment to comply with the Core Operating Limits Report.

D. A Reactor Engineer must remain in the control room to comply with the Core Operating Limits Report.

Answer: A

Answer Explanation: A. is correct. Per N1-OP-42 Precaution and Limitation D.2.0, GAP-OPS-01, Section 3.2.1 Table and Tech Spec Table 6.2-1, 6.2-2. After eight hours without the PPC, a 3rd NLO must be added to the crew composition.

B. is incorrect. Tech Specs does not require and additional STA. Since the Process Computer is used to provide information related to thermal limits and core thermal power calculations, it is conceivable to select an STA or Reactor Engineer as possible positions to fill to increase shift. An additional NLO is required.

C. and D. are incorrect. Core Operating Limits Report does not require additional staffing with the process computer out of service. Since the Process Computer is used to provide information related to thermal limits and core thermal power calculations, it is conceivable to select an STA or Reactor Engineer as possible positions to fill to increase shift, based on items that may be contained in the Core Operating Limits Report.

Question 1 Details

Associated objective(s):

1. LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question Type:	Multiple Choice
Topic:	NRC RO 3
System ID:	21228
User ID:	NRC 2006 RO 3
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference Number:	LC1 05-01
Num Field 1:	17151
Num Field 2:	
Text Field:	
Comments:	Question used for 2002 annual requal exam, Created for requal exam,Old SysID in secret room: 16488, Old USER ID: NONE , 1/12/2004

Reference Provided: None

Enabling Objective: 283000 RBO-9 and O1-OPS-001-283-1-01 EO-1.6

Justification for use as RO question: Training program contains learning objectives (O1-OPS-001-283-1-01 EO-1.6 and 283000 RBO-9) for operators that include knowledge and application of system procedure related precautions and limitations. Operation with the computer unavailable is a precaution and limitation contained as D.2.0 of N1-OP-42, Process Computer. The objectives are "from memory" for initial operator training. The specific requirements are contained in Tech Specs, supporting the KA statement knowledge requirement.

Question 1 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

PROC

- GAP-OPS-01 Rev. NA
- N1-OP-42 Rev.

TECHSPEC

- 6.2.2, Rev. NA

NUREG 1123 KA Catalog Rev. 2

- G2.1.33 3.4/4 Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 1 Memory/Fundamental Knowledge

10CFR55

- 43(b)(2)

Question Source

- Bank

4 **USERID: NRC 2006 RO 4** **SYSID: 21229** **Points: 1.00**

A plant startup is in progress following an outage with the following:

- All IRM's are on range 7 or greater
- ROD BLOCK (F3-4-4) is alarming
- RWM ERROR/SELECT ERROR is indicated
- Attempts to restore the rod worth minimizer to operation are unsuccessful

Which one of the following actions is required by N1-OP-37, Rod Worth Minimizer?

A. Terminate rod movement until the RWM becomes operable.

B. Terminate the startup and place the plant in a less restricting condition.

C. Bypass the RWM, station a Human RWM, and continue with the startup.

D. Bypass the RWM and continue startup with no additional compensatory actions.

Answer: C

Answer Explanation: C. is correct. N1-OP-37 H.2.0 for INOP RWM contains the following caution:

It is a violation of Technical Specification 3.1.1.b.(3)(b) to move rods below 10% power unless the computer RWM or designated Human RWM is present and monitoring.

A and B are incorrect because terminating the startup or rode movement is not required. Compensatory action of the designated human RWM allows continued plant startup.

D is incorrect because in order to continue the startup, the human RWM compensatory action must be initiated.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 4 Details

Question Type:	Multiple Choice
Topic:	NRC RO 4
System ID:	21229
User ID:	NRC 2006 RO 4
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 27, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	16303
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE** Enabling Objective: O1-OPS-001-201-1-03 EO-1.7

Question 4 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

PROC
- N1-OP-37 Rev. NA
- N1-ST-V3 Rev. NA

TECHSPEC
- 3.1.1(3) Rev. NA

NUREG 1123 KA Catalog Rev. 2
- G2.2.1 3.7/3.6 Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity

Level of Difficulty
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level
- 1 Memory/Fundamental Knowledge

10CFR55
- 41(b)(5)

Question Source
- Bank

5 **USERID: NRC 2006** **SYSID: 21230** **Points: 1.00**
RO 5

A Temporary Change Package has just been used to perform the following:

- Annunciator H3-1-4, FEEDWATER HEATER 111 - 115 LEVEL HIGH, has 3 of its inputs defeated
- The work has been documented in the Defeated Annunciator Log
- There are no previous entries for this annunciator in the Defeated Annunciator Log

Which one of the following identifies the color of the sticker to be attached to the annunciator window, and describes the information to be included on that sticker, per GAP-OPS-01, Administration of Operations?

	STICKER COLOR	INFORMATION TO INCLUDE
A.	Yellow	Defeated Annunciator Log Sequence Number
B.	Yellow	Temporary Change Package number
C.	Red	Defeated Annunciator Log Sequence Number
D.	Red	Temporary Change Package number

Answer: B

Answer Explanation: B is correct - Per GAP-OPS-01, 3.11.7, attach a yellow sticker whenever < all of the inputs are defeated. There are 5 inputs to this window. Information to include on the sticker includes: authorizing package ID (the TCP #, in this case), and affected computer points (when practicable).

A is incorrect - No such Log number exists.

C and D are incorrect- Red sticker is required only when defeating all of the inputs to the given annunciator.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 5 Details

Question Type:	Multiple Choice
Topic:	NRC RO 5
System ID:	21230
User ID:	NRC 2006 RO 5
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Dec 06, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	NRC 2002 RO 3
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE** Enabling Objective: O3-OPS-006-343-3-39 EO-1.13

Question 5 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2
- G2.2.11 2.5/3.4 Knowledge of the process for controlling temporary changes

Level of Difficulty
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level
- 1 Memory/Fundamental Knowledge

10CFR55
- 41(b)(7)

Question Source
- Bank

6 **USERID: NRC 2006 RO 6** **SYSID: 21231** **Points: 1.00**

Reactor power is 10% during plant startup with the following:

- ERV surveillance is in progress
- Heat is being added to the Torus during ERV cycling
- Torus Water Temperature is being logged every 5 minutes
- Torus Water Temperature is 85.1°F and rising

Which one of the following describes the requirement regarding ERV testing?

A. Test can continue, however torus water temperature must now be logged every minute.

B. Test can continue because torus water temperature is still below applicable LCO limit.

C. Test must be stopped because torus water temperature is above applicable LCO limit.

D. Test must be stopped because an Emergency Operating Procedure entry condition is now met.

Answer: B

Answer Explanation:

Answer: B is correct. Per TS 3.3.2., during test of relief valves which add heat to the torus bulk temperature shall not exceed 10°F above the normal limit. The normal limit without testing is 85°F.(spec b.) There is no stopping requirement per N1-ST-C2 other than previously stated.

Distractor: A is incorrect. Even though test can continue, it is not because temperature logging requirements have changed. When testing occurs that adds heat to the Torus, SR 3.3.2.c requires logging every 5 minutes. There is no requirement to log temperature every minute.

Distractor: C is incorrect. Per TS 3.6.2.1, with testing in progress that adds heat to the Torus, the applicable LCO limit for for torus water temperature is 95°F. The normal limit without testing is 85°F. If the 85°F limit is applied (incorrectly) the testing must be stopped.

Distractor: D is incorrect. EOP-4 entry condition is 85°F. EOP entry is required but the test is not required to be stopped just because the EOP is entered. The high temperature is expected because of the surveillance test.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 6 Details

Question Type:	Multiple Choice
Topic:	NRC RO 6
System ID:	21231
User ID:	NRC 2006 RO 6
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Dec 06, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE** Enabling Objective: O1-OPS-001-239-1-01 EO-1.11

Question 6 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

TECHSPEC
- 3.3.2 Rev. NA

NUREG 1123 KA Catalog Rev. 2
- G2.2.25 2.5/3.7 Knowledge of bases in technical specifications for limiting conditions for operations and safety limits

Level of Difficulty
- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 41(b)(10)

Question Source
- New

7 **USERID: NRC 2006 RO 7** **SYSID: 26974** **Points: 1.00**

The plant is shutdown with the following:

- A 31 year old AO is entering the Drywell for a job
- General area radiation level is 3 rem/hr
- The AOs TEDE for the year is 1710 mr
- The AOs lifetime exposure is 25 Rem
- No dose extension is obtained
- It takes 45 minutes to complete the job

Which one of the following states the limit(s) exceeded, if any, in order to complete the job?

A. None

B. NRC only

C. Administrative only

D. NRC and Administrative

Answer: C

Answer Explanation: C. is correct. The administrative dose control limit without any extensions is 2000 mr TEDE. The total dose the worker will have received after completing the job will be 1710+3000(45/60)=3960 mr. A worker can receive up to 4000 mr TEDE with all of the approved extensions. The NRC yearly limit is 5000 mr.

A. is incorrect. The administrative dose limit without any extensions is 2000 mr. The worker will have 3960 mr when the job is completed.

B. and D. are incorrect. The NRC yearly limit is 5000 mr.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 7 Details

Question Type:	Multiple Choice
Topic:	NRC RO 7 REV 1
System ID:	26974
User ID:	NRC 2006 RO 7
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	3.00
Time to Complete:	5
Point Value:	1.00
Date Changed:	Jan 31, 2007
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	O1-OPS-006-APS-1-01 thru 18 EP-1.5, Explain the requirements of the procedure. **Reference Provided: NONE** Enabling Objective: O3-OPS-006-343-3-40 EO-3.5

Question 7 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2
- G2.3.1 2.6/3 Knowledge of 10 CFR 20 and related facility radiation control requirements

Level of Difficulty
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level
- 1 Memory/Fundamental Knowledge

10CFR55
- 41(b)(12)

Question Source
- New

PROCEDURES (WWG PROJECTS)
- GAP-RPP-07 , INTERNAL AND EXTERNAL DOSIMETRY PROGRAM

8 **USERID: NRC 2006 RO 8** **SYSID: 21233** **Points: 1.00**

The plant is operating at 75% power when the following alarm is received:

- H1-1-7, OFF GAS HIGH RADIATION alarm
- Chemistry confirms the off gas high radiation condition

Which one of the following describes the required operator action(s), per N1-SOP-25.2, FUEL FAILURE OR HIGH ACTIVITY IN RX COOLANT OR OFFGAS?

A. Enter N1-EOP-6, Radioactivity Release Control, and verify the turbine building roof vents are closed.

B. Enter N1-EOP-5, Secondary Containment Control, and verify ventilation system isolations and actuations.

C. Perform an Emergency Power Reduction without entering the restricted zone and commence a normal shutdown per N1-OP-43C.

D. Lower recirculation flow to 38 x 10^6 lbm/hr, then scram the reactor and place both Mechanical Vacuum Pumps in Parallel per N1-OP-25.

Answer: C

Answer Explanation: C. is correct. Per ARP and N1-SOP-25.2, an emergency power reduction is directed, and then continue with an orderly plant shutdown per N1-OP-43C

A. is incorrect. EOP-5 is only entered if release rate reaches alert levels. No alarm is given to indicate high release rates.
B. is incorrect. EOP-5 is only entered if a Reactor Building ARM alarms. No alarm is given to indicate high radiation levels in the Reactor Building.
D. is incorrect. This is not an appropriate action for this condition. Offgas radiation levels may indicate fuel failure. Inserting a scram may make the leak worse and cause additional fuel failure.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 8 Details

Question Type:	Multiple Choice
Topic:	NRC RO 8
System ID:	21233
User ID:	NRC 2006 RO 8
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 30, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE** Enabling Objective: O1-OPS-001-271-1-01 EO-1.7

Question 8 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

PROC

- N1-SOP-25.2 Rev. NA
- N1-ARP-H1-1-7, Rev. NA

NUREG 1123 KA Catalog Rev. 2

- G2.3.11 2.7/3.2 Ability to control radiation releases

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 1 Memory/Fundamental Knowledge

10CFR55

- 41(b)(5)

Question Source

- Bank

9 **USERID: NRC 2006 RO 9** **SYSID: 21234** **Points: 1.00**

While executing an SOP an entry condition into the Emergency Operating Procedures (EOPs) is met. The appropriate EOPs have been entered.

Which one of the following describes the EOP and SOP use for this condition?

A. Execute all EOP legs concurrently and the remaining steps of the SOP when the plant is stable.

B. Execute all EOP legs concurrently and exit the SOP when the EOPs are entered.

C. Execute only the EOP leg for the plant parameters being threatened and exit the SOP when the EOPs are entered.

D. Execute only the EOP leg for the plant parameters being threatened and the remaining steps of the SOP when the plant is stable.

Answer: A

Answer Explanation: A is correct. All legs of the EOPs entered are executed concurrently and the SOPs are not exited when entering the EOPs.

B is incorrect. The SOPs are not exited when the EOPs are exited

C is incorrect. All legs of any EOP entered are executed concurrently. The SOPs are not exited when the EOPs are exited.

D. is incorrect. All legs of any EOP entered are executed concurrently.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 9 Details

Question Type:	Multiple Choice
Topic:	NRC RO 9
System ID:	21234
User ID:	NRC 2006 RO 9
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 30, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE** Enabling Objective: O1-OPS-006-344-1-01 EO-1.1

Question 9 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2
- G2.4.13 3.3/3.9 Knowledge of crew roles and responsibilities during EOP flowchart use

Level of Difficulty
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level
- 1 Memory/Fundamental Knowledge

10CFR55
- 41(b)(10)

Question Source
- New

10 **USERID: NRC 2006 RO 10** **SYSID: 21235** **Points: 1.00**

The plant is in STARTUP with the following:

- RPV pressure is 925 psig and steady
- CRD pump 11 is in service
- Annunciator F3-2-2, CONTROL ROD DRIVE PUMP 11 SUCT PRESS LOW, is in alarm
- Annunciator F3-1-2, CONTROL ROD DRIVE PUMP 11 TRIP-VIB, is in alarm
- Annunciator F3-1-5, CRD CHARGING WTR PRESSURE HI/LO, is in alarm
- Control rod 22-19 has received an accumulator trouble alarm

Which one of the following operator actions is required to be performed and the reason for these actions, per N1-SOP-5.1 LOSS OF CONTROL ROD DRIVE?

A. Start a CRD pump to restore cooling water flow to prevent CRD mechanism high temperature alarms

B. Start a CRD pump to restore charging water pressure to prevent a manual reactor scram

C. Immediately scram the reactor due to low charging water pressure to ensure rod insertion times are not exceeded

D. Immediately scram the reactor due to possible CRD mechanism failure to prevent an unanalyzed rod pattern

Answer: B

Answer Explanation: B. is correct. The alarms that were received tell the operator that CRD pump 11 has tripped and are entry conditions for N1-SOP-5.1, LOSS OF CONTROL ROD DRIVE. Conditions provided are that reactor pressure is greater than 900 psig. Therefore, N1-SOP-5.1 directs the operator to restart at least one CRD pump within 20 minutes. This is done to restore charging water pressure.

A. is incorrect. N1-SOP-5.1 directs the operator to start a CRD pump to restore charging water pressure not to restore cooling water flow

C. and D. are incorrect. N1-SOP-5.1 only directs the operator to immediately scram the reactor if reactor pressure is < 900 psig and at rated power reactor pressure is ~1025 psig.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 10 Details

Question Type:	Multiple Choice
Topic:	NRC RO 10
System ID:	21235
User ID:	NRC 2006 RO 10
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Dec 06, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE** Enabling Objective: O1-OPS-001-201-1-01 EO-1.7

Question 10 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2
- G2.4.48 3.5/3.8 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

Level of Difficulty
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 43(b)(5)

Question Source
- New

11 **USERID: NRC 2006 RO 11** **SYSID: 21236** **Points: 1.00**

The plant is in cold shutdown following a 365 day continuous run with the following:

- One Shutdown Cooling Pump is running
- Reactor water temperature is being maintained steady
- All reactor recirculation pumps are secured
- All five recirculation pump discharge and discharge bypass valves closed
- All five recirculation pump suction valves are open
- Reactor recirculation pump 14 discharge bypass valve inadvertently opens

Which one of the following describes the consequences of the event?

A. Heatup and subsequent pressurization of RPV.

B. High temperature isolation of SDC system.

C. Excessive cooldown rate of RPV bottom head.

D. Damage to the affected RRP from reverse rotation.

Answer: A

Answer Explanation: A is correct: To prevent thermal stratification and short cycling while operating SDC with no recirc flow, all five recirc loops must have flow through them stopped by keeping either the suction valve or discharge and discharge bypass valves closed. If a bypass valve is opened, flow will be diverted away from the core region and will flow through the open valve. Decay heat will cause core temperature to rise. The short cycled flowpath of the SDC system will cause all temperatures in that path to continue to lower. The flowpath would be: suction on loop 14 inlet, through SDC system, discharge of loop 15 outlet, lower vessel plenum, loop 14 discharge, through the open valve.
B is incorrect: High temperature is a pump trip. Loop temperature is likely to lower due to flow bypassing the core.
C is incorrect: Cooler water in bottom head may result in a lower temperature, but an excessive cooldown rate will not occur, with the plant already in cold shutdown.
D is incorrect: Reverse rotation from reverse flow will not damage the RRP.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 11 Details

Question Type:	Multiple Choice
Topic:	NRC RO 11
System ID:	21236
User ID:	NRC 2006 RO 11
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Jan 31, 2007
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE** Enabling Objective: O1-OPS-001-205-1-01 EO-1.5

Question 11 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

PROC
- N1-OP-4 Rev. NA

NUREG 1123 KA Catalog Rev. 2
- 205000 K6.03 3.1/3.2 Knowledge of the effect that a loss or malfunction of the following will have on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE): Recirculation system

Level of Difficulty
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 41(b)(7)

Question Source
- New

12 **USERID: NRC 2006 RO 12** **SYSID: 21237** **Points: 1.00**

The plant is operating at 100% power with the following:

- FWP 13 and 12 are operating
- FWP 11 is in standby
- THEN, a Main Turbine trip AND FWP 12 trip occurs
- RPV level drops to 40 inches before recovering
- Total Feedwater Flow is 5.5 Mlbm/hr

Which one of the following describes the response of FWP 11 during the transient?

A. Does not start because total feedwater flow is too high.

B. Starts but does not inject because total feedwater flow is too high.

C. Starts and immediately begins to inject until water level is 65 inches.

D. Starts and subsequently trips on low feedwater pump suction pressure.

Answer: C

Answer Explanation: C is correct. For HPCI, both Feedwater Pumps will start on a Turbine trip or RPV low level. However, if total feedwater flow is greater than 4.5 X 10^6 lbm/hr, FCV 11 will be biased closed if already open, or maintained closed until total flow drops below this value. The logic to close FCV 11 will be bypassed if 12 Feedwater Pump is locked out or not running. If 12 Feedwater Pump is tripped, FCV 11 would function as required and be controlled at its HPCI level control setpoint of 65 inches.

A. is incorrect because both Feedwater pumps receive start signals, independent of the total feed flow value.

B. is incorrect because with the 12 Feedwater pump tripped, the bias is removed from FCV 11 and it opens to restore level. This would be correct Feedwater pump 12 is running. The bias would be applied to FCV11 and it would be closed until flow dropped below 4.5 Mlbm/hr.

D. is incorrect because the system design is intended to prevent the trip of the Feedwater pumps on low suction pressure during the transient.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 12 Details

Question Type:	Multiple Choice
Topic:	NRC RO 12
System ID:	21237
User ID:	NRC 2006 RO 12
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 30, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	25803 Modified
Comments:	**Reference Provided: NONE** Enabling Objective: O1-OPS-001-259-1-01 EO-1.4.c In accordance with appropriate NMP documents, describe: Feedwater System and High Pressure Coolant Injection interlocks and setpoints.

Question 12 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2

- 206000 K3.01 4/4 Knowledge of the effect that a loss or malfunction of HIGH PRESSURE COOLANT INJECTION SYSTEM will have the following: Reactor water level control: BWR-2,3,4

Level of Difficulty

- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level

- 2 Higher Cognitive Level comprehensive

10CFR55

- 41(b)(7)

Question Source

- Modified

13 **USERID: NRC 2006 RO 13** **SYSID: 26972** **Points: 1.00**

The plant is experiencing a transient with the following:

- Reactor is manually scrammed
- FWP 13 clutch is dis-engaged
- FWP 11 and 12 are running
- FWPs are NOT at runout flows
- RPV water level is 45 inches and rising

Which one of the following describes the relative flow rates for FWP 11 and 12?

A. Only FWP 12 is injecting with FWP 11 running on minimum flow.

B. Both are injecting but FWP 11 flow is higher than FWP 12 flow.

C. Both are injecting but FWP 12 flow is higher than FWP 11 flow.

D. Both are injecting with FWP 12 flow equal to FWP 11 flow.

Answer: C

Answer Explanation: C is correct. Both FWP 11 and 12 start in HPCI mode. Since level error signal between FWP 12 setpoint (72 inches) and actual level (45 inches) is greater than the level error signal between FWP 11 setpoint (65 inches) and actual level (45 inches), FWP 12 flow control valve is open wider than FWP 11 flow control valve. FWP 12 is injecting at a higher rate than FWP 11. As level approaches the controlling setpoint, the valves throttle closed, with FWP 11 going to minimum flow before FWP 12, because its setpoint is lower.

A is incorrect because both FWPs are injecting. With FWP 13 de-clutched, all FW flow is from both pumps in HPCI mode.
B is incorrect. FWP 11 flow is lower than FWP 12, not greater.
D is incorrect. The pumps do not inject at equal flow rates because of the two different setpoints for the flow control valves. FWP 11 valve controls to 65 inch setpoint and FWP 12 valve controls to 72 inch setpoint.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 13 Details

Question Type:	Multiple Choice
Topic:	NRC RO 13 REV 1
System ID:	26972
User ID:	NRC 2006 RO 13
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Jan 30, 2007
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	N1-259001 RBO-7 Technical reference N1-259000 Feedwater and HPCI KA match because understanding valve operation is required to flow characteristic of the pumps. **Reference Provided: NONE**

Question 13 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2
- 206000 A4.04 3.7/3.7 Ability to manually operate and/or monitor in the control room: Major system valves: BWR-2,3.4

Level of Difficulty
- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 41(b)(7)

14 **USERID: NRC 2006 RO 14** **SYSID: 21239** **Points: 1.00**

The plant is operating at 100% power with the following:

- BOTH Emergency Condenser (EC) Loops are in standby
- THEN a loss of power to Powerboard PB161B occurs

Which one of the following is an effect of the power loss on EC Loop 12 valves?

A. Inside Steam Isolation Valve cannot close on high steam flow.

B. Outside Steam Isolation Valve cannot close on high steam flow.

C. Condensate Return Valve cannot open from Remote Shutdown panel.

D. Condensate Return Valve opens initiating heat removal from the vessel.

Answer: A

Answer Explanation: A. is correct. Inside Loop 12 IV 39-10 is powered from AC PB161B. On a loss of power, the valve will not close on an isolation signal.
B. is incorrect because it is DC powered
C. is incorrect because AC power for this valve solenoid is from RPS, not PB161B.
D. is incorrect because the valve does not receive AC power from PB161. The valve does fail open on loss of DC power.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 14 Details

Question Type:	Multiple Choice
Topic:	NRC RO 14
System ID:	21239
User ID:	NRC 2006 RO 14
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 30, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE**
	Enabling Objective: O1-OPS-001-207-1-01 EO-1.5

Question 14 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

DRW
- C-18017-C Rev. NA
- C-19409-C Rev. na Item Sheet 1B

NUREG 1123 KA Catalog Rev. 2
- 207000 K6.07 3.0*/3.2 Knowledge of the effect that a loss or malfunction of the following will have on the ISOLATION (EMERGENCY) CONDENSER: A.C. power: BWR-2,3

Level of Difficulty
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level
- 1 Memory/Fundamental Knowledge

10CFR55
- 41(b)(7)

Question Source
- New

15 **USERID: NRC 2006** **SYSID: 21240** **Points: 1.00**

RO 15

The plant is experiencing a transient with the following:

- CRS orders RPV pressure band of 800 to 1000 psig
- Emergency Condenser (EC) 11 is manually initiated
- Alarm K1-1-3, EMER COND CONDEN RET ISOL VALVE 11 OPEN alarms

Which one of the following additional annunciators actuating over the next several minutes indicates that EC 11 is functioning properly?

A. K1-1-2, EMER COND VENT 11 RAD MONITOR at 5 mr/hr

B. K1-2-3, EMER COND 111-112 LEVEL HIGH-LOW at 7.8 feet

C. K1-3-3, EMER COND 111-112 SHELL TEMP HIGH at 150°F

D. K1-4-2, EMER COND 111-112 INLET STEAM PRESS LOW at 500 psig

Answer: C

Answer Explanation: C. is correct. When EC initiates, the steam flow through the tubes transfers heat to the water on the shell side. Shell side water temperature will rise and K1-3-3, EMER COND 111-112 SHELL TEMP HIGH alarms at 150°F.
A. is incorrect. Vent Rad Monitor reading may rise but it is not expected to reach the alarm setpoint of 5 mr/hr. This is indicative of a tube bundle leak.

B. is incorrect. Shell water level will lower as water in the shell evaporates. A low level alarm is expected at 6 feet, not a high level at 7.8 feet.
D. is incorrect. With pressure band of 800 to 1000 psig, a low steam pressure will not occur. This may be indicative of an isolation or steam line leak.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 15 Details

Question Type:	Multiple Choice
Topic:	NRC RO 15
System ID:	21240
User ID:	NRC 2006 RO 15
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 30, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE** Enabling Objective: O1-OPS-001-207-1-01 EO-1.2

Question 15 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

PROC
- N1-OP-13 Rev. NA

NUREG 1123 KA Catalog Rev. 2
- 207000 A4.07 4.2*/4.3 Ability to manually operate and/or monitor in the control room: Manually initiate the isolation condenser: BWR-2,3

Level of Difficulty
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 41(b)(7)

Question Source
- New

16 **USERID: NRC 2006** **SYSID: 21241** **Points: 1.00**
RO 16

The plant is operating at 100% power with the following:

- N1-ST-M6 CORE SPRAY KEEP FILL SYSTEM VERIFICATION TEST is being performed for Loop 11
- An operator is stationed at valve 40-26 VENT-CS LOOP 11 KEEP FILL
- Valve 40-30 CORE SPRAY VENT ISOLATION VALVE 111 is OPEN
- Valve 40-32 CORE SPRAY VENT ISOLATION VALVE 112 is OPEN
- Flow is observed from the vent line to the equipment drain tank funnel
- THEN Drywell Pressure rises to 4.0 psig

Which one of the following is required to stop vent flow to the funnel?

A. Automatic closing of valves 40-30 and 40-32.

B. Manually closing valves 40-30 and 40-32 from K Panel.

C. Locking the breaker in OFF for valve 40-30 at PB161B.

D. Directing operator stationed at 40-26 to manually close it.

Answer: A

Answer Explanation: A. is correct. Valves 40-30 and 40-32 automatically close as the system realigns for injection on the high drywell pressure initiation.
B. is incorrect. Manually closing the vent valves is not required because they receive an automatic closure.
C. is incorrect. Locking breakers in OFF is not required to stop flow. This is a normal step in the surveillance to return the configuration to normal after the surveillance is complete.
D is incorrect. The operator stationed at 40-26 is stationed to close the valve if directed, due to Appendix R concerns, not to close it on a LOCA.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 16 Details

Question Type:	Multiple Choice
Topic:	NRC RO 16
System ID:	21241
User ID:	NRC 2006 RO 16
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 30, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE** Enabling Objective: O1-OPS-001-209-1-01 EO-1.7f

Question 16 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2

- 209001 K5.05 2.5/2.5 Knowledge of the operational implications of the following concepts as they apply to LOW PRESSURE CORE SPRAY SYSTEM: System venting

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 1 Memory/Fundamental Knowledge

10CFR55

- 41(b)(5)

Question Source

- New

17 **USERID: NRC 2006 RO 17** **SYSID: 21242** **Points: 1.00**

The plant is operating at 100% power with the following:

- Loss of 120 VAC RPS Bus 12 occurs
- Conditions require a manual scram be inserted
- A Failure to Scram occurs
- Liquid Poison (LP) injection is required
- THEN LP keylock switch on K panel is placed to SYS 12

Which one of the following describes the status of the LP system as a result of positioning the switch to SYS 12?

A. No pumps are running and no squib explosive valve fired.

B. No pumps are running with only squib explosive valve 11 fired.

C. LP Pump 12 is injecting through only squib explosive valve 11.

D. LP Pump 12 is injecting through squib explosive valves 11 and 12.

Answer: C

Answer Explanation: C is correct. Starting LP pump 12 will close contacts in the firing circuits of both squib valves but since there is no power to RPS bus 12 the System 12 squib valve will not fire. Liquid poison pump 12 will start and run and inject through 11 valve only.
A is incorrect because LP Pump 12 still starts since it still has power from PB17B. Explosive valve 11 fires because placing keylock switch to either SYS 11 or SYS 12 closes contacts in both valve firing circuits.
B. is incorrect because LP Pump 12 does start.
D. is incorrect because explosive valve 12 firing circuit is deenergized with loss of RPS Bus 12. It does not open.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 17 Details

Question Type:	Multiple Choice
Topic:	NRC RO 17
System ID:	21242
User ID:	NRC 2006 RO 17
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 30, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE**

Enabling Objective: O1-OPS-001-211-1-01 EO-1.8 Describe the impact of component malfunctions on the Liquid Poison System.

Drawing: C-19439-C Sheet 3.

Question 17 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

DRW
- C-19439-C Rev. NA

NUREG 1123 KA Catalog Rev. 2
- 211000 K2.02 3.1*/3.2* Knowledge of electrical power supplies to the following: Explosive valves

Level of Difficulty
- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 41(b)(7)

Question Source
- New

18 **USERID: NRC 2006 RO 18** **SYSID: 21243** **Points: 1.00**

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

- A gross rupture occurs on YARWAY LEVEL COLUMN 11 VARIABLE LEG

Which one of the following describes the resulting condition of RPS Channel 11 and 12 scram functions?

	RPS CH 11	RPS CH 12
A.	Reset	Reset
B.	Reset	Tripped
C.	Tripped	Reset
D.	Tripped	Tripped

Answer: D

Answer Explanation: D is correct. Yarway Level Column 11 has 36-03A and 36-03B connected to it. These Rosemount H/L transmitters provide input to trip RPS Channels at 53 inches. These are arranged such that 36-03A inputs to RPS Channel 11-1 and 36-03B inputs to RPS Channel 12-1. Both RPS Channels 11 and 12 will trip resulting in a reactor scram.

A, B and C are incorrect because a trip of both RPS Channels will occur.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 18 Details

Question Type:	Multiple Choice
Topic:	NRC RO 18
System ID:	21243
User ID:	NRC 2006 RO 18
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 30, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: C-18015-C** Enabling Objective: O1-OPS-001-212-1-01 EO-1.5.b, O1-OPS-001-216-1-01 EO-1.8 Describe the impact of component malfunctions on the Reactor Vessel Instrumentation System.

Question 18 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2
- 212000 K6.03 3.5/3.7 Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR PROTECTION SYSTEM: Nuclear boiler instrumentation

Level of Difficulty
- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 41(b)(7)

Question Source
- Bank

19 **USERID: NRC 2006 RO 19** **SYSID: 21244** **Points: 1.00**

A plant startup is in progress with the following:

- Group 1 white lights for RPS Channel 12 on BOTH Panels F and M are NOT LIT
- All light bulbs have been verified working
- All other group lights are LIT
- IRMs are on Range 5 and 6
- THEN IRM Channel 16 fails upscale

Which one of the following identifies the required operator actions?

A. Insert a manual scram and execute SOP-1.

B. Confirm automatic scram and execute SOP-1.

C. Bypass IRM 16 and reset RPS Channel 11.

D. Bypass IRM 16 and reset RPS Channel 12.

Answer: D

Answer Explanation: D. is correct. IRM 16 provides trip input into RPS Channel 12. With Group 1 RPS Channel 12 lights out, and IRM 16 above its trip setpoint, the result will be only a trip of RPS Channel 12.
A. is incorrect. Manual scram is not required. If the Group 1 lights were NOT LIT on Channel 11 and IRM 16 tripped Channel 12, then some rod motion will occur and a manual scram is required.
B. is incorrect. IRM 16 provides trip input into RPS Channel 12. With Group 1 RPS Channel 12 lights out, and IRM 16 above its trip setpoint, the result will be only a trip of RPS Channel 12. If Group 1 lights were NOT LIT on Channel 11 and Channel 12 trip, then rod motion may result in an automatic scram.
C. is incorrect. IRM 16 provides input into RPS Channel 12. RPS Channel 11 will not trip and does not require reset.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 19 Details

Question Type:	Multiple Choice
Topic:	NRC RO 19
System ID:	21244
User ID:	NRC 2006 RO 19
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 30, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	Modified Bank SYSID 21036
Comments:	**References provided: NONE** Enabling Objective: O1-OPS-001-212-1-01, EO-1.4.b In accordance with appropriate NMP documents, describe: Reactor Protection System operations/lineups Question derived from bank SYSID 12388

Question 19 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2
- 215003 A1.03 3.6/3.7 Ability to predict and/or monitor changes in parameters associated with operating the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM controls including: RPS status

Level of Difficulty
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 41(b)(5)

Question Source
- Modified

20 **USERID: NRC 2006 RO 20** **SYSID: 21245** **Points: 1.00**

A plant startup is in progress with the following:

- Reactor Mode Switch is about to be transferred to RUN
- F3-4-5 RPS REACTOR PRESS LOW BYPASS alarm is clear
- F1-4-7 RPS CH 11 REACTOR PRESS LOW alarm is clear
- F4-4-2 RPS CH 12 REACTOR PRESS LOW alarm is clear
- THEN an equipment failure results in RPV pressure dropping below 850 psig

Which one of the following describes the response of the MSIVs and the Reactor Protection System?

	MSIVs	RPS
A.	Remain open	No scram will occur
B.	Remain open	A scram will occur
C.	Isolate	No scram will occur
D.	Isolate	A scram will occur

Answer: D

Answer Explanation: D. is correct because with IRM's on range 10, the MSIV Isolation on low pressure is enabled, which will result in reactor scram on MSIV position. With F3-4-5 clear, the scram is in effect, IRMs must be on Range 10 with Mode Switch in Startup, for this alarm to be clear). F1-4-7 and F4-4-2 are clear when RPV pressure is above 865 psig.

A. is incorrect because a MSIV Isolation will occur and a reactor scram will occur.

B. is incorrect because a MSIV Isolation will occur.

C. is incorrect because a reactor scram will occur.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 20 Details

Question Type:	Multiple Choice
Topic:	NRC RO 20
System ID:	21245
User ID:	NRC 2006 RO 20
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Dec 14, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	12412
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE** Enabling Objective: O1-OPS-001-212-1-01 EO-1.5

Question 20 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2
- G2.1.32 3.4/3.8 Ability to explain and apply system limits and precautions
- 215003 IRM

Level of Difficulty
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 41(b)(10)

Question Source
- New

21 **USERID: NRC 2006 RO 21** **SYSID: 21246** **Points: 1.00**

The plant is in a refuel outage, with the following:

- Reactor Mode Switch is in REFUEL
- REFUEL INSTRUMENT TRIP BYPASS 11 is in NON-COINCIDENT
- REFUEL INSTRUMENT TRIP BYPASS 12 is in NON-COINCIDENT
- A loss of 24 VDC Nuclear Instrumentation Bus 12 occurs

Which one of the following describes the effect of the power loss on the RPS trip channels and the SRMs?

A. Only RPS Channel 11 trips. SRMs 11 and 12 deenergize.

B. Only RPS Channel 12 trips. SRMs 13 and 14 deenergize.

C. Both RPS Channels 11 and 12 trip. SRMs 11 and 12 deenergize.

D. Both RPS Channels 11 and 12 trip. SRMs 13 and 14 deenergize.

Answer: D

Answer Explanation: O1-OPS-001-215-1-02 - EO-1.4

D. is correct. Loss of 24/48 VDC power from 24 V Instrument Bus 12 results in power loss to SRM 13 and 14. IRM 15-18 also experience the power loss. Since RPS is in NON-COINCIDENT, both RPS channels manual scram relays are deenergized causing a full scram.

A. and B. are incorrect because BOTH RPS channel 11 and 12 trip in NON-COINCIDENT mode.

A would be correct in COINCIDENT logic with Bus 11 loss.

B would be correct in COINCIDENT logic with Bus 12 loss.

C. is incorrect. SRM 11 and 12 are powered from Instrument Bus 11 and are not affected. This would be correct Bus 11 loss in NON-COINCIDENT logic.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

EXAMINATION ANSWER KEY

Reactor Operator Written Examination Nine Mile Point Unit 1 2007

Question 21 Details

Question Type:	Multiple Choice
Topic:	NRC RO 21
System ID:	21246
User ID:	NRC 2006 RO 21
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Dec 14, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: None** Enabling Objective: O1-OPS-001-215-1-02 - EO-1.4 Drawings C-19866-C Sheet 3, C-22024-C Sheet 2

Question 21 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

DRW
- C-22024-C Rev. NA
- C-19866-C Rev. NA

PROC
- N1-OP-47B, H.2.0, Rev. NA

NUREG 1123 KA Catalog Rev. 2
- 215004 K6.02 3.1/3.3 Knowledge of the effect that a loss or malfunction of the following will have on the SOURCE RANGE MONITOR (SRM) SYSTEM: 24/48 volt D.C. power

Level of Difficulty
- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 41(b)(7)

Question Source
- New

22 **USERID: NRC 2006 RO 22** **SYSID: 21247** **Points: 1.00**

The plant is operating at 65 % power with the following:

- Rod line is 100%
- All recirc pumps are controlled by the master controller
- APRM FLOW UNIT 12 signal drifts down to 30% and remains steady

Which one of the following is the plant response?

A. No rod block or half scram.

B. Rod block and no half scram.

C. Half scram and rod block.

D. Reactor scram and rod block.

Answer: C

Answer Explanation: C. is correct. The power /flow operating map for 5 loop operation shows the flow unit reading to be above the rod block and scram setpoints. Actual plant power remains unchanged.

APRM 12 flow unit provides the flow input to ARPM 15-18 only these APRMs will generate a trip signal in RPS Channel 12. Comparator rod block trip also occurs if difference between flowchart 11 and 12 exceeds 6.8%.

A. and B. are incorrect because a half scram on RPS Channel 12 occurs.

D. is incorrect. RPS Channel 11 stays reset, so a full scram does not occur.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

EXAMINATION ANSWER KEY

Reactor Operator Written Examination Nine Mile Point Unit 1 2007

Question 22 Details

Question Type:	Multiple Choice
Topic:	NRC RO 22
System ID:	21247
User ID:	NRC 2006 RO 22
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Dec 14, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	Direct 22508
Comments:	**Reference Provided: Power Flow Operating Maps F-45683-C** Enabling Objective: O1-OPS-001-215-1-02 EO-1.4.c In accordance with appropriate NMP documents, describe: Neutron Monitoring System interlocks and setpoints For Initial Training the 5 Loop Power to Flow Map is to be provided. This is NOT a "from memory" question, because of the varying scram and rodblock setpoints.

Question 22 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

PROC
- N1-OP-38C Rev. NA

NUREG 1123 KA Catalog Rev. 2
- 215005 K4.07 3.7/3.7 Knowledge of AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM design feature(s) and/or interlocks which provide for the following: Flow biased trip setpoints

Level of Difficulty
- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 41(b)(7)

Question Source
- Bank

23 **USERID: NRC 2006 RO 23** **SYSID: 21248** **Points: 1.00**

The plant experiences a scram from 100% power with the following:

RPV pressure is being maintained 800 to 1000 psig with ECs

- t = 0 sec RPV water level drops to -11 inches
- t = 10 sec Drywell pressure is 3.6 psig
- t = 20 sec RPV water level rises to 15 inches due to injection
- t = 30 sec RPV water level drops to -11 inches again and remains at that level
- NO operator actions are taken for ADS system

Which one of the following indicates when the ADS valves start to open?

A. t= 111 seconds

B. t= 121 seconds

C. t= 141 seconds

D. t= 146 seconds

Answer: C

Answer Explanation: C is correct. ADS logic automatically initiates to open 3 primary ERVs with RPV water level below -10 inches AND DW pressure exceeds 3.5 psig after 111 second timer times out. The lo-lo-lo level signal does NOT seal in, so if level rises above -10 inches, the timer resets. When level drops below -10 inches, the second time (t= 30 seconds) the ADS timer will restart and after timing out 111 seconds later, at t= 141 seconds, the 3 primary ERVs will open and pressure will be rapidly reduced.

A. is incorrect, because the ERVs will still be closed at 111 seconds. Plausible if the logic initiated valve opening on lo-lo-lo water level only and the level signal did seal in, which are incorrect.

B. is incorrect, because the ERVs will still be closed at 121 seconds. Plausible if the logic initiated valve opening on lo-lo-lo water level with high drywell pressure and the level signal did seal in. The timer resets when level rises above -10 inches.

D. is incorrect, because the ERVs have already opened at 141 seconds. Plausible if the primary ERVs did not open after the 111 second timer timed out and the primary ERVs remained closed. The backup timer at 115 seconds would open the backup ERvs.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 23 Details

Question Type:	Multiple Choice
Topic:	NRC RO 23
System ID:	21248
User ID:	NRC 2006 RO 23
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Jan 16, 2007
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Technical Reference: N1-218000 **Reference Provided: NONE** Enabling Objective: O1-OPS-001-218-1-01 EO-1.4

Question 23 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2
- 218000 A3.08 4.2*/4.3* Ability to monitor automatic operations of the AUTOMATIC DEPRESSURIZATION SYSTEM including: Reactor pressure

Level of Difficulty
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 41(b)(7)

Question Source
- New

24 **USERID: NRC 2006 RO 24** **SYSID: 21249** **Points: 1.00**

The plant is operating at 100% power with the following:

- All Reactor Building Ventilation system components are operable.
- THEN a loss of 120 VAC RPS Bus 11 power occurs.

Which one of the following is the status of the Reactor Building Emergency Ventilation System (RBEVS) 11 and 12 trains immediately after the power loss?

	RBEVS TRAIN 11	RBEVS TRAIN 12
A.	Standby	Standby
B.	Standby	Starts
C.	Starts	Standby
D.	Starts	Starts

Answer: D

Answer Explanation: D. is correct. Both trains of RBEVS auto start. There are two initiation logic channels (11 and 12) arranged in a one out of two logic to trip and isolate normal ventilation and start both RBEVS fans. The logic circuits are 120 VAC RPS powered and are normally energized circuits which deenergize to cause the function to occur. Actuation occurs when either channel trips or becomes de-energized.

A. is incorrect because loss of power will start both trains.

B. is incorrect because loss of power will also start 11 train.

C. is incorrect because loss of power will also start 12 train.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 24 Details

Question Type:	Multiple Choice
Topic:	NRC RO 24
System ID:	21249
User ID:	NRC 2006 RO 24
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Dec 19, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE**

Enabling Objective: O1-OPS-001-261-1-01 EO-1.5

Question 24 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

DRW
- C-19859-C SHEET 15 Rev. NA

PROC
- N1-SOP-40.1, Rev. NA

NUREG 1123 KA Catalog Rev. 2
- 223002 K3.11 2.8/2.9 Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF will have on following: Plant ventilation

Level of Difficulty
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level
- 1 Memory/Fundamental Knowledge

10CFR55
- 41(b)(7)

Question Source
- New

The plant is operating at 95% reactor power with the following:

0100	Control Rod 38-11 drifts out
0150	Offgas radiation level is 1850 mr/hr
0200	F1-2-7, MAIN STEAM RAD MONITOR CH 11 HI/LO is in alarm
0205	F4-2-2, MAIN STEAM RAD MONITOR CH 12 HI/LO is in alarm
0205	N1-SOP-25.2, FUEL FAILURE/HIGH ACTIVITY, is being implemented
0210	Main Steam Line radiation is 2750 mr/hr
0220	Condenser vacuum is 27 inches and lowering slowly

Which one of the following is the required plant condition and the required operator action(s)?

	Condition	Operator Action
A.	Offgas Isolated	Start the Mechanical Vacuum Pumps
B.	Offgas Isolated	Scram when vacuum reaches 22.1 inches
C.	MSIVs closed	Insert N1-EOP-1 Attachment 2 jumpers
D.	MSIVs closed	Control RPV Pressure using the ECs

Answer: D

Answer Explanation: This is a KA match because the candidate has to determine that a Vessel Isolation has been performed from N1-SOP-25.2. The impact of the vessel isolation is an MSIV closure.

D. is correct. With 2750 mr/hr radiation levels in the main steam lines N1-SOP-25.2 directs a reactor scram and a manual vessel isolation. The manual vessel isolation isolates the TBVs and the TCVs. In addition, a manual vessel isolation isolates the RWCU system. Therefore, RPV pressure control must be transferred to a alternate pressure control system like the Emergency Condensers.

A. is incorrect. An Offgas isolation does occur but the mechanical vacuum pumps will not be started with a high Main Steam Line radiation level.

B. is incorrect. An Offgas isolation does occur. However, N1-SOP-25.1 directs that a reactor scram should be inserted **BEFORE** condenser vacuum reaches 22.1 inches.

C. is incorrect. The MSIVs do close. However, N1-EOP-1 attachment 2 jumpers are only authorized to be inserted from N1-EOP-3.

Question 1 Details

Associated objective(s):

1. LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question Type:	Multiple Choice
Topic:	NRC RO 25 REV 1
System ID:	27156
User ID:	NRC 2006 RO 25
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	3.00
Time to Complete:	3
Point Value:	1.00
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	LP N1-223000 RBO-10

Reference Provided to Candidate: None

Question 1 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2

- 223002 A2.09 3.6/3.7 Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal condition or operations: System initiation

Level of Difficulty

- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level

- 2 Higher Cognitive Level comprehensive

10CFR55

- 41(b)(7)

Question Source

- New

PROCEDURES (WWG PROJECTS)

- N1-EOP-2, RPV CONTROL
- N1-SOP-25.2, FUEL FAILURE OR HIGH ACTIVITY IN RX COOLANT OR OFF GAS

26 **USERID: NRC 2006 RO 26** **SYSID: 26973** **Points: 1.00**

The plant is operating at 100% power with the following:

- Main Condenser vacuum is 25.1 inches Hg and lowering
- Loss of Battery Board 12 occurs due to fault

Which one of the following is the immediate consequence of these events?

A. MPR control from E Panel will not function.

B. ERV 121, 122 and 123 controls from F Panel will not function.

C. Bypass valve opening jack motor control from A Panel will not function.

D. EC Loop 12 condensate return valve control on K Panel will not function.

Answer: B

Answer Explanation: B. is correct. BB 12 supplies power to ERV 121, 122 and 123 solenoids. These are the backup ADS valves. Power is NOT transferable to BB 11, per N1-OP-47A, Attachment 5.

A. is incorrect. MPR control from E Panel still functions but would not function if BB 11 is lost.
C. is incorrect. Bypass opening jack motor control from A Panel is powered from BB 11, so function is not lost. E panel will also still function.
D. is incorrect. EC Loop 12 Condensate Return Valve 39-06 will still function if BB 12 is lost, but position indication red and green lights will be out.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 26 Details

Question Type:	Multiple Choice
Topic:	NRC RO 26 REV 1
System ID:	26973
User ID:	NRC 2006 RO 26
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Jan 17, 2007
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Technical References: N1-218000 ADS N1-SOP-47A N1-OP-47A **Reference Provided: NONE** Enabling Objective: O1-OPS-001-218-1-01 RBO 8

Question 26 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

PROC
- N1-OP-47A Rev. NA
- N1-SOP-47A, Rev. NA

NUREG 1123 KA Catalog Rev. 2
- 239002 K2.01 2.8*/3.2* Knowledge of electrical power supplies to the following: SRV solenoids

Level of Difficulty
- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level
- 1 Memory/Fundamental Knowledge

10CFR55
- 41(b)(7)

Question Source
- New

27 **USERID: NRC 2006** **SYSID: 21252** **Points: 1.00**
RO 27

The plant is operating at 100% power with the following:

- The BLUE LIGHT above the control switches for ERV 112 and 113 are ON
- The BLUE LIGHT above the control switches for ERV 111, 121, 122 and 123 are OFF
- All ERVs are verified closed

Assuming ERVs open at their design set point ($\pm$ 0 psig), which one of the following is the LOWEST reactor pressure at which an ERV opens during an overpressure event?

A. 1090 psig

B. 1095 psig

C. 1100 psig

D. 1105 psig

Answer: B

Answer Explanation: B. is correct. ERV 111,121, 122, and 123 BLUE light is OFF, so none of the valves that are set to lift at 1090 psig are available to operate. The next valve with a BLUE light on is ERV 112, with a lift setpoint of 1095 psig. Therefore, 1095 psig is correct based on ERV 112.

ERV 111 and 122 open at 1090 psig
ERV 112 and 121 open at 1095 psig
ERV 113 and 123 open at 1100 psig
1105 psig is an arbitrary value based on balance with other answers and also would be indicative that NO ERVs open and safety valves actuate.

Explanation of ERV Indication on F Panel
ERV Indication is located on F Panel and contains the following indication:

- Red and Green lights indicate the position of the pilot valve solenoid.
- The normally lit blue light above each ERV control switch indicates that DC Control Power is available to the ERV pilot valve solenoid. The Blue Light will de-energize for two reasons: (1) If the ERV receives an initiating signal or (2) If DC Control Power to the ERV pilot valve solenoid circuit is lost. ERV 111, 121, 122 and 123 do not have DC control power and will NOT function.
- RED status monitoring lights to the right of each ERV Control Switch provides additional indication that an ERV has opened. A high alarm condition, as monitored by the acoustic monitors, will cause the RED light to illuminate. The light will remain illuminated until the alarm is reset at the System Panel.

A. is incorrect because neither 111 or 122 have blue light lit, which are 1090 psig

C. and D. are incorrect because ERV 112 will lift at 1095 psig setpoint.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 27 Details

Question Type:	Multiple Choice
Topic:	NRC RO 27
System ID:	21252
User ID:	NRC 2006 RO 27
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Feb 16, 2007
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	Modified 21044
Comments:	**References Provided: NONE** Enabling Objective: O1-OPS-001-239-1-01 EO-1.8

Question 27 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2

- 239002 K3.02 4.2*/4.4 Knowledge of effect that a loss or malfunction of the RELIEF/SAFETY VALVES will have on the following: Reactor over pressurization

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 1 Memory/Fundamental Knowledge

10CFR55

- 41(b)(7)

Question Source

- Modified

28 **USERID: NRC 2006** **SYSID: 21253** **Points: 1.00**
RO 28

The plant is at 100% power with the following:

- FWP 11 Flow Control Valve is in service and in MAN
- FWP 13 Flow Control Valve is in BAL
- FWLC system is in three-element control
- FWP 11 flow indication begins to slowly lower due to a transmitter malfunction

Which one of the following describes how the FWP 13 Flow Control Valve responds?

A. Open further causing total feed flow and level to rise.

B. Open further to keep total feed flow and level the same.

C. Remain at the same position causing total feed flow and level to lower.

D. Remain at the same position to keep total feed flow and level the same.

Answer: A

Answer Explanation: Explanation:
A is correct. FWLC uses pump discharge flow transmitters 29-53, 29-54 and 29-113A to develop total FW flow signal for use by the FWLC system. As FWP 11 discharge flow lowers a "steam flow > feed flow" error signal is generated. FWLC will send a signal to raise FW flow by opening FWP 13 valve, raising FW flow and RPV level, since actual flow was sufficient to maintain level steady.

B. is incorrect because level is rising. This would be correct if actual FWP 11 flow was lowering. Then the FWP13 valve would open to maintain current flow and level.

C and D are incorrect because FWP 13 valve is opening. This could be correct, if FWP13 valve was in MAN, instead of BAL.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 28 Details

Question Type:	Multiple Choice
Topic:	NRC RO 28
System ID:	21253
User ID:	NRC 2006 RO 28
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 30, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	Modified 12566
Comments:	Drawings: C-23076-C Sheet 1, C-18005-C Sheet 1

Reference Provided: NONE

Enabling Objective: O1-OPS-001-259-1-02 EO-1.8 Describe the impact of component malfunctions on the Feedwater Level Control System.

Question 28 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

DRW
- C-18005-C Rev. NA
- C-23076-C Rev. NA

PROC
- N1-OP-16 Rev. NA

NUREG 1123 KA Catalog Rev. 2
- 259002 K1.04 3.5/3.6 Knowledge of the physical connections and/or cause-effect relationships between REACTOR WATER LEVEL CONTROL SYSTEM and the following: Reactor feedwater flow

Level of Difficulty
- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 41(b)(2)

Question Source
- Modified

29 **USERID: NRC 2006 RO 29** **SYSID: 21254** **Points: 1.00**

The plant is operating at 100% power with the following:

- Both RBEVS trains are operable and in standby
- THEN a reactor scram occurs
- Power Boards 11 AND 12 fail to fast transfer and power cannot be restored

Which one of the following describes the RBEVS system flow **AND** required actions, if any, to place the ventilation system in the proper post-transient lineup?

A. Initially remains at zero. No action is required to realign the system, so flow remains at zero.

B. Initially remains at zero. One fan is manually started and flow rises to 1600 scfm.

C. Initially rises to 3200 scfm. No action is required to realign the system, so flow remains at 3200 scfm.

D. Initially rises to 3200 scfm. One fan is manually stopped and flow lowers to 1600 scfm.

Answer: B

Answer Explanation: B. is correct. RB Normal ventilation supply and exhaust fans trip when power is lost to PB 11 and 12. RB differential pressure, which is normally maintained at a negative value, will drop to zero. RBEVS does not automatically start, so initially flow will remain at zero. After operator action is taken to establish RB differential pressure, one RBEVS train will be manually started and system flow will be at 1600 scfm.

A. is incorrect because one RBEVS fan is required to be started to restore reactor building differential pressure to a negative value.

C. is incorrect. Initial flow does not initially rise to 3200 scfm. This would be correct if both RBEVS fans automatically started, which is the correct system response on a valid initiation signal. If two trains start, one is required to be shutdown within 30 minutes.

D. is incorrect because both fans do not auto start, but correctly identifies the manual action to stop one fan on an automatic initiation.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 29 Details

Question Type:	Multiple Choice
Topic:	NRC RO 29
System ID:	21254
User ID:	NRC 2006 RO 29
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 30, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE** Enabling Objective: O1-OPS-001-261-1-01 RBO 10

Question 29 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

PROC
- N1-OP-10 Rev. NA

NUREG 1123 KA Catalog Rev. 2
- 261000 A1.01 2.9/3.1 Ability to predict and/or monitor changes in parameters associated with operating the STANDBY GAS TREATMENT SYSTEM controls including: System flow

Level of Difficulty
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 41(b)(5)

Question Source
- New

30 **USERID: NRC 2006** **SYSID: 26975** **Points: 1.00**
RO 30

The plant is experiencing a transient from 100% power with the following:

- A Loss of Offsite Power occurs
- PB 11 and PB 12 are deenergized
- EDG 102 and EDG 103 start and load, as designed
- Drywell Pressure is 4.0 psig
- BOTH Lo-Lo-Lo F Panel Level instruments are pegged high
- PB 16A-B is cross-tied
- PB 17A-B is cross-tied
- EDG102 load is 1800 KW
- EDG103 load is 2400 KW

Which one of the following actions is required to be taken per the SOP's?

A. Stop Core Spray Topping Pump 122

B. Stop Control Rod Drive Pumps 11 and 12

C. Deenergize PB17A and leave PB16A energized

D. Place Drywell Cooling Fans 11, 12 and 13 in PTL

Answer: B

Answer Explanation: B is correct. Per SOP-33A.1 for PB11 and 12 de-energized and BOTH EDGs running, leg B is entered. Override step directs CRD Pump 11 AND 12 secured.

A is incorrect. Stopping CSTP 122 is a step directed for these conditions (Lo-Lo-Lo level indicators both upscale) in leg C of the SOP as part of an OR step. Leg C is entered if only EDG 103 was running. Therefore this is not the correct leg of the SOP to be executing, since both EDG are running. Since CSTP 122 is powered from PB103, but is NOT directed to be stopped when implementing the correct leg of the SOP (leg B).

C is incorrect. The SOP override directs tripping R1042 PB16A-B tie and R1052 PB17A-B tie breaker, if either EDG is overloaded. The step structure uses lettered steps, not bulleted or conditional steps to allow tripping the tie breaker for only the overloaded EDG, therefore maintaining PB16A does not comply with the procedure.

D is incorrect. Placing the DW Fans in PTL is not correct because leg B is the correct leg to implement under these conditions, NOT leg C, where this action is directed from.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 30 Details

Question Type:	Multiple Choice
Topic:	NRC RO 30 REV 1
System ID:	26975
User ID:	NRC 2006 RO 30
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Jan 17, 2007
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	N1-262000 RBO-11, AC MAIN POWER N1-SOP-33A.1 **Reference Provided to Candidate: N1-SOP-33A.1 Flowchart**

Question 30 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

PROC
- N1-SOP-33A.1, Rev. NA

NUREG 1123 KA Catalog Rev. 2
- 262001 A2.03 3.9/4.3* Ability to (a) predict the impacts of the following A.C. ELECTRICAL DISTRIBUTION; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of off-site power

Level of Difficulty
- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 41(b)(5)

Question Source
- New

31 **USERID: NRC 2006** **RO 31** **SYSID: 21256** **Points: 1.00**

The plant is operating at 100% power with the following:

t = 0 sec RPS Channel 11 is tripped due to surveillance testing
t = 30 sec Supply breaker to in-service UPS172A trips at Power Board 17B

Which one of the following is the effect on reactor power and why?

A. Power remains at 100% because UPS loads transfer to bypass transformer.

B. Power remains at 100% because UPS input supply transfers to DC input.

C. Power drops rapidly to 0% because UPS loads momentarily deenergize during power transfer.

D. Power drops rapidly to 0% because UPS loads are deenergized until manually restored.

Answer: B

Answer Explanation: B is correct. With only the normal AC power lost, the UPS load will continue to be supplied by the inverter output from backup 125 VDC power from BB 12. UPS172A and B supply power to RPS channel 12. No loss of RPS 12 occurs and the reactor will remain at 100% power.

A. is incorrect. With the AC power loss, the UPS transfers supply to DC power. The bypass transformer power is also lost, since they are supplied from same AC source.

C. is incorrect. No loss of power to RPS Channel 12 occurs, so the reactor does not scram. With RPS Channel 11 already tripped, a loss of RPS 12 power would cause a full scram and power rapidly dropping to 0%.

D. is incorrect. No loss of power to RPS Channel 12 occurs, so the reactor does not scram. Power transfer is automatic, with no manual restoration required.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 31 Details

Question Type:	Multiple Choice
Topic:	NRC RO 31
System ID:	21256
User ID:	NRC 2006 RO 31
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Dec 18, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Technical References: N1-212000 N1-OP-40

Reference Provided: NONE

Enabling Objective: O1-OPS-001-212-1-01 RBO 11

Question 31 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

PROC
- N1-OP-40 Rev. NA

NUREG 1123 KA Catalog Rev. 2
- 262002 K3.14 2.8/3.1 Knowledge of the effect that a loss or malfunction of the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) will have on following: Rx power: Plant-Specific

Level of Difficulty
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 41(b)(7)

Question Source
- New

32 **USERID: NRC 2006 RO 32** **SYSID: 21257** **Points: 1.00**

The plant is operating at 100% power with the following:

- Static Battery Chargers (SBC) 161A and 171A are in service
- Powerboard 16B is inadvertently deenergized then re-energized

TWO minutes after Powerboard 16B is re-energized, and with no additional operator action, which one of the following identifies the Battery Board 11 and 12 voltages?

	Battery Board 11	Battery Board 12
A.	125 VDC lowering	125 VDC lowering
B.	125 VDC lowering	135 VDC steady
C.	135 VDC steady	125 VDC lowering
D.	135 VDC steady	135 VDC steady

Answer: D

Answer Explanation: D. is correct – 100 seconds after AC power is restored to SBC161A (from powerboard 16B) the SBC will align itself to the Battery 11 and return voltage to the normal float voltage of 135VDC

A. and B. are incorrect – Battery 11 voltage will return to 135 VDC.

C. is incorrect – Battery 12 is unaffected as it's SBC is powered from PB-17A(B).

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 32 Details

Question Type:	Multiple Choice
Topic:	NRC RO 32
System ID:	21257
User ID:	NRC 2006 RO 32
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 30, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE** Enabling Objective: O1-OPS-001-263-1-01 EO-1.4.c In accordance with appropriate NMP documents, describe: DC Electrical Distribution System interlocks and setpoints.

Question 32 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

PROC

- N1-OP-47A Rev. NA

NUREG 1123 KA Catalog Rev. 2

- 263000 A3.01 Ability to monitor automatic operations of the D.C. ELECTRICAL DISTRIBUTION including: Meters,Dials,Recorders,Alarms & Indicating Lights, Rev. NA

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 2 Higher Cognitive Level comprehensive

10CFR55

- 41(b)(7)

Question Source

- Bank

33 **USERID: NRC RO 33 REV 1** **SYSID: 27158** **Points: 1.00**

The plant is operating at rated power with the following:

t = 0000	Core Spray Pump 111 and the Core Spray Topping Pump 111 are in PTL
t = 0100	A loss of offsite power occurs
t = 0130	The plant experiences an DBA LOCA

Which one of the following describes the status of the Containment Spray Pumps 40 second after the LOCA with no operator action?

	Pump 111 & 112	Pump 121 & 122
A.	Not Running	Not Running
B.	Not Running	Running
C.	Running	Not Running
D.	Running	Running

Answer: D

Answer Explanation: D. is correct. With Core Spray Pump 111 and Core Spray Topping Pump 111 in PTL the PB 102 Containment Spray Pumps will both be started 30 seconds after the LOCA. The Containment Spray Pumps would not start until 55 and 60 seconds if a Core Spray or Core Spray Topping Pump failed to start. However, this logic does not apply if the pumps are in PTL. With the Core Spray Pump 111 and Core Spray Topping Pump 111 in PTL the load sequencing logic for the Emergency Diesel Generators does not delay starting the Containment Spray Pumps because the Core Spray and Core Spray Topping Pumps are unable to start. The PB 103 pumps are unaffected by Core Spray Pump 111 and Core Spray Topping Pump 111 therefore they will start both be started 30 seconds after the LOCA.

A. is incorrect. This answer would be chosen if Core Spray Pump 111 and Core Spray Topping Pump 111 affected both PB 102 and PB 103 Containment Spray Pumps.

B. is incorrect. This answer would be chosen if the Core Spray Pump 111 and Core Spray Topping Pump 111 affected the PB 102 Containment Spray Pumps.

C. is incorrect. This answer would be chosen if the Core Spray Pump 111 and Core Spray Topping Pump 111 affected the PB 103 Containment Spray Pumps.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 33 Details

Question Type:	Multiple Choice
Topic:	NRC RO 33 REV 1
System ID:	27158
User ID:	NRC RO 33 REV 1
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	3.00
Time to Complete:	3
Point Value:	1.00
Date Changed:	Feb 26, 2007
Cross Reference Number:	LC 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**References Provided: NONE** 264000 RBO-8

Question 33 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2
- 264000 K4.05 3.2/3.5 Load shedding and sequencing

Level of Difficulty
- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level
- 1 Memory/Fundamental Knowledge

10CFR55
- 41(b)(7)

Question Source
- New

34 **USERID: NRC 2006 RO 34** **SYSID: 21259** **Points: 1.00**

Given the following conditions for Diesel Generator (DG) 102:

- A leak in the common header downstream of the DG 102 Air Compressors occurs
- Air Receiver Tank pressures are lowering
- THEN an automatic start signal is received
- 5 seconds after start signal the engine speed reaches 175 rpm
- Annunciator A4-3-5 DIESEL GENERATOR START-RUN OFF NORMAL alarms

Which one of the following describes the automatic response of the DG and the manual operator actions required to attempt a restart of the DG?

A. The DG shuts down immediately and locks out. The operator must correct the start failure condition and place the engine control switch to EM STOP.

B. The DG shuts down immediately and locks out. The operator must correct the start failure condition and depress the local 48x, RESET and FAST STOP pushbuttons.

C. The DG attempts a second start. If that start fails, the operator must correct the start failure condition and depress the local 48x, RESET and FAST STOP pushbuttons.

D. The DG attempts a second start. If that start fails, the operator must correct the start failure condition and place the engine control switch to EM STOP.

Answer: C

Answer Explanation: C. is correct. If the diesel engine does not attain 200 rpm in five seconds the diesel fast stops. The diesel then attempts a second start. If the diesel does not attain 750 rpm in 2 minutes the diesel will shutdown and the local 48x and ALARM/FAST STOP pushbuttons must be depressed to attempt starting the diesel.

A. is incorrect. The DG will fast stop and attempt a second start.

B. is incorrect. The DG will fast stop and attempt a second start.

D. is incorrect. In order to reset the relays the ALARM/FAST STOP pushbuttons must be depressed. Without pushing these buttons the DG will not start.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 34 Details

Question Type:	Multiple Choice
Topic:	NRC RO 34
System ID:	21259
User ID:	NRC 2006 RO 34
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Jan 16, 2007
Cross Reference Number:	LC1 05-01
Num Field 1:	1027
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE** Enabling Objective: O1-OPS-001-264-1-01 EO-1.5

Question 34 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2
- G2.4.50 3.3/3 Ability to verify system alarm setpoints and operate controls identified in the alarm response manual
- 264000 EDGs

Level of Difficulty
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 41(b)(5)

Question Source
- Bank

35 **USERID: NRC 2006 RO 35** **SYSID: 21260** **Points: 1.00**

The plant is operating at 100% power with the following:

- 11 Instrument Air Compressor (IAC) is the BACKUP (red flagged) and running
- 12 Instrument Air Compressor is the STANDBY (Green flagged) and NOT running
- THEN L1-2-7, "INST AIR COMP 11-12 DISCH AIR-COOL WTR TEMP HI alarms
- Comp point A191 IA CMPR 11 DIS TEMP HIGH indicates air temperature is 325°F
- Air header pressure lowered to 97 psig and is now steady

Which one of the following describes the condition of IAC 11 and 12?

	IAC 11	IAC 12
A.	Running	Off
B.	Running	Automatically started
C.	Tripped	Off
D.	Tripped	Automatically started

Answer: C

Answer Explanation:

C. is correct. Running compressor IAC 11 trips when air temperature exceeds 320°F. ARP says verify start of standby air compressor. Op Tech describes the BACKUP compressor as the red flagged and runs with pressure below 98 psig. This is IAC 11. The STANDBY compressor is green flagged and start when pressure drops to 93 psig. Therefore IAC 11 is tripped and IAC 12 is still off.

A., B. and D. are incorrect because they identify the incorrect lineup.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 35 Details

Question Type:	Multiple Choice
Topic:	NRC RO 35
System ID:	21260
User ID:	NRC 2006 RO 35
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Dec 18, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE** Enabling Objective: O1-OPS-001-278-1-01 EO-1.4

Question 35 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

PROC
- N1-ARP-L1-2-7, Rev. NA

NUREG 1123 KA Catalog Rev. 2
- 300000 A3.02 2.9/2.7 Ability to monitor automatic operations of the INSTRUMENT AIR SYSTEM including: Air temperature

Level of Difficulty
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level
- 1 Memory/Fundamental Knowledge

10CFR55
- 41(b)(7)

Question Source
- Bank

36 **USERID: NRC 2006** **SYSID: 21261** **Points: 1.00**
RO 36

The plant is operating at 100% power, with the following:

- H1-4-2, R BUILDING SW PRESS/SERV W PUMP HDR PRESS LOW is in alarm
- N1-SOP-18.1 SW Failure / Low INTAKE LEVEL has been entered
- Neither Service Water Pump can be started

Which one of the following is the required action and the effect on plant operations?

A. Start an ESW Pump. Power operations can continue because all cooling water is restored.

B. Start an ESW Pump. A plant shutdown or scram is required due to partial loss of cooling water.

C. Lineup fire water to RBCLC heat exchanger. Power operations can continue because all cooling water is restored.

D. Lineup fire water to RBCLC heat exchanger. A plant shutdown or scram is required due to partial loss of cooling water.

Answer: B

Answer Explanation: B. is correct. Per N1-SOP-18.1 an ESW pump must be started. Since ESW can only supply RBCLC HXs and not the TBCLC HXs, a plant shutdown or scram will be required due to loss of cooling to TBCLC loads.
A is incorrect. Since ESW will not supply TBCLC loads, all cooling water is NOT restored. Power operations cannot be continued, since TBCLC loads no longer have a heat sink.
C. & D. are incorrect. Fire water is only lined up if ESW pumps cannot be started.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 36 Details

Question Type:	Multiple Choice
Topic:	NRC RO 36
System ID:	21261
User ID:	NRC 2006 RO 36
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Feb 16, 2007
Cross Reference Number:	LC1 05-01
Num Field 1:	17894
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE**
	Enabling Objective: O1-OPS-001-208-1-01 EO-1.7G

Question 36 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

PROC

- N1-SOP-18.1, Rev. NA

NUREG 1123 KA Catalog Rev. 2

- 400000 K1.01 3.2/3.3 Knowledge of the physical connections and / or cause-effect relationships between CCWS and the following: Service water system

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 1 Memory/Fundamental Knowledge

10CFR55

- 41(b)(4)

Question Source

- New

37 **USERID: NRC 2006** **SYSID: 21262** **Points: 1.00**

RO 37

Given the following conditions:

- The IRMs are on range 8
- Control rod 22-51 was just moved from position 36 to position 38
- The RMCS timer sent a signal to the RWM
- The rod position information was NOT applied to the computer with the signal from the timer

Which one of the following describes how the Reactor Manual Control System will respond?

A. Rods can be inserted or withdrawn. RWM generates an error.

B. Rods can be inserted but not withdrawn. RWM generates a withdraw block.

C. Rods cannot be inserted but can be withdrawn. RWM generates an insert block.

D. Rods cannot be inserted or withdrawn. RWM generates insert and withdraw blocks.

Answer: D

Answer Explanation: D. is correct. A program abort will occur in the event that an RPIS input to the RWM program fails in any way, this message appears. When a program abort condition exists the RWM will go off-line and all rod blocks (insert and withdraw) will be applied regardless of the power level.

A. is incorrect. Both an insert and a withdraw block will be present.

B. is incorrect. A insert block is present.

C. is incorrect. A withdraw block is present.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 37 Details

Question Type:	Multiple Choice
Topic:	NRC RO 37
System ID:	21262
User ID:	NRC 2006 RO 37
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Dec 18, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	12312
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE**
	Enabling Objective: O1-OPS-001-201-1-02 EO-1.4c

Question 37 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

PROC
- N1-OP-37 Rev. NA

NUREG 1123 KA Catalog Rev. 2
- 201002 K1.05 3.4/3.5 Knowledge of the physical connections and/or cause-effect relationships between REACTOR MANUAL CONTROL SYSTEM and the following: Rod worth minimizer: Plant-Specific

Level of Difficulty
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level
- 1 Memory/Fundamental Knowledge

10CFR55
- 41(b)(6)

Question Source
- Bank

38 **USERID: NRC RO 38** **SYSID: 26978** **Points: 1.00**

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

- Control Rod 30-31 is at position 48
- Computer printout indicates a high temperature condition for Control Rod 30-31
- SM has determined that CRD cooling is required per N1-OP-5

Which one of the following describes the required operator actions to clear the high temperature alarm?

A. Initiate repeated drive signals

B. Open CRD Cooling water PCV

C. Insert the control rod one notch

D. Verify no air leaks affecting CRD

Answer: C

Answer Explanation: C. is correct. Per N1-OP-5 H.12.0 for a single control rod mechanism high temperature alarm with the control rod at position 48 the operator is required to insert the control rod to position 46 to clear the high temperature alarm.

A. is incorrect. A caution statement in the procedure reads as follows: Correcting CRD temperature alarms by applying drive signals, will only temporarily cool the drive and will place undesirable thermal stresses on the CRD.

B. is incorrect. The CRD Cooling Water PCV is downstream of the Cooling water header. Opening this valve will lower cooling water dp and result in less cooling water flow.

D. is incorrect. Verifying there are no air leaks that affect the CRD system is done to ensure the CRD FCV is not closing. If this were the case there would be multiple control rod mechanism high temperature alarms. N1-OP-5 only directs this action if there are multiple control rod mechanism high temperature alarms.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 38 Details

Question Type:	Multiple Choice
Topic:	NRC RO 38 REV 1
System ID:	26978
User ID:	NRC RO 38
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	2.00
Time to Complete:	3
Point Value:	1.00
Date Changed:	Feb 22, 2007
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	201003 RBO 10

REFERENCE PROVIDED: NONE

Question 38 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2
- 201003 A2.06 3/3.1 Loss of CRD cooling water flow

Level of Difficulty
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level
- 1 Memory/Fundamental Knowledge

10CFR55
- 41(b)(5)

Question Source
- New

PROCEDURES (WWG PROJECTS)
- N1-OP-5, CONTROL ROD DRIVE SYSTEM

39 **USERID: NRC 2006** **SYSID: 21264** **Points: 1.00**
RO 39

The plant is operating at 90% power with the following:

- Five Recirc Pumps are running
- All RRP control M/A stations at F Panel are in BAL and nulled
- THEN the RRP 12 tachometer signal to its speed control circuitry fails low

Which one of the following is the effect on RRP 12 pump speed and the action required to control pump speed?

A. Rises to maximum. Take local manual control to reduce speed.

B. Rises to maximum. Place F panel controller in manual to reduce speed.

C. Lowers to minimum. Take local manual control to raise speed.

D. Lowers to minimum. Place F panel controller in manual to raise speed.

Answer: A

Answer Explanation: A. is correct. Speed controller senses pump speed at zero or minimum due to loss of speed feedback signal. RRP speed will increase to maximum. Speed control is in local manual only, because the individual MA station will not effect pump speed, even in manual.

B. is incorrect because the F panel MA station will not control pump speed, even in manual with the speed signal failed.

C. and D. are incorrect because pump speed will rise with the loss of speed feedback signal.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

EXAMINATION ANSWER KEY

Reactor Operator Written Examination Nine Mile Point Unit 1 2007

Question 39 Details

Question Type:	Multiple Choice
Topic:	NRC RO 39
System ID:	21264
User ID:	NRC 2006 RO 39
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 30, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE** Enabling Objective: N1-202001 RBO-11 Technical Reference(s): N1-202001 Reactor Recirculation System

Question 39 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2

- 202002 A1.01 3.2/3.2 Ability to predict and/or monitor changes in parameters associated with operating the RECIRCULATION FLOW CONTROL SYSTEM controls including: Recirculation pump speed: BWR-2,3,4,5,6

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 2 Higher Cognitive Level comprehensive

10CFR55

- 41(b)(5)

Question Source

- New

40 **USERID: NRC 2006 RO 40** **SYSID: 21265** **Points: 1.00**

The plant is operating at 100% power with the following:

- A TIP trace is in progress with the probe at the Core Top Limit
- A loss of 120/208 VAC Distribution Panel 167C occurs
- THEN a valid automatic containment isolation signal is received

Which one of the following describes the operator action required to complete the containment isolation per N1-OP-39, Traversing Incore Probe (TIP)?

A. Manually retract the TIP probe locally.

B. Manually close the ball valve locally.

C. Fire the associated TIP squib valve.

D. Initiate a manual containment isolation.

Answer: C

Answer Explanation: C is correct: Distribution Panel 167C powers the TIP Drive Mechanisms. Without power to the drive unit, the probe will not retract, in any electrical mode, for any reason. To complete the containment isolation, operators will have to fire the squib valve per N1-OP-39, Section H.1.0.

A is incorrect: There is no NMP1 procedure that allows for manually retracting the probe, locally, under these conditions.

D is incorrect: Nothing in the stem conditions indicates a fault with the processing of the automatic containment isolation signal; nor does N1-SOP-40.2 (Vessel/Containment Isolation) direct that the TIP ball valves be "verified" closed by depressing the manual Containment Isolation pd's at the E Panel.

B is incorrect: Manually closing the ball valve, a would not be possible with the TIP cable running through the valve.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 40 Details

Question Type:	Multiple Choice
Topic:	NRC RO 40
System ID:	21265
User ID:	NRC 2006 RO 40
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Dec 18, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	17656
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE** Enabling Objective: O1-OPS-001-215-1-01 EO-1.5

Question 40 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

OTHER_REFS

- (SYSTEM) SDBD-306, Rev. NA

PROC

- N1-OP-39 Rev. NA

NUREG 1123 KA Catalog Rev. 2

- G2.4.49 4/4 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls
- 215001 Traversing In-core Probe

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 2 Higher Cognitive Level comprehensive

10CFR55

- 41(b)(10)

Question Source

- Bank

41 **USERID: NRC 2006 RO 41** **SYSID: 21266** **Points: 1.00**

Containment Spray Loop 121 is cooling the Torus when a tube rupture in its heat exchanger occurs.

Which one of the following describes the initial response of Containment Spray Pump 121 discharge pressure and Raw Water Pump 121 flow?

	SPRAY PUMP DISCHARGE PRESSURE	RAW WATER PUMP FLOW
A.	Lowers	Rises
B.	Lowers	Lowers
C.	Rises	Rises
D.	Rises	Lowers

Answer: C

Answer Explanation: C. is correct because Raw Water pressure is > Cont Spray pressure. Rupture results in a higher Cont Spray pressure and increased Raw Water flow (additional flow path made available). Both of these parameters are monitored in the control room.

A., B. and D. are incorrect for reasons as already described for justifying answer choice C.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 41 Details

Question Type:	Multiple Choice
Topic:	NRC RO 41
System ID:	21266
User ID:	NRC 2006 RO 41
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 30, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	17972
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE** Enabling Objective: O1-OPS-001 226-1-01 EO-1.8

Question 41 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

PROC
- N1-OP-14 Rev. NA

NUREG 1123 KA Catalog Rev. 2
- 219000 A1.03 2.9/2.9 Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE controls including: System pressure

Level of Difficulty
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 41(b)(5)

Question Source
- Bank

42 **USERID: NRC 2006** **SYSID: 21267** **Points: 1.00**
RO 42

The plant is operating at 100% power with the following:

t = 0 sec	Loss of Battery Board 12 occurs
t = 30 sec	Loss of Offsite Power occurs
t = 1:00 min	Large reactor coolant rupture occurs inside the Drywell
t = 1:00 min	Reactor automatically scrams
t = 1:30 min	Containment Spray pumps receive an automatic start signal
t = 1:30 min	Containment Spray pumps are NOT locked out
	NO actions have been taken per N1-SOP-47.1, Loss of DC due to the timing of the above events

Which one of the following describes the effect of the Containment Spray Pump response, if any, on the Emergency Diesel Generator kilowatt loading?

	EDG102	EDG103
A.	Remain at 0	Remain at 0
B.	Remain at 0	Rises from initial load value
C.	Rises from initial load value	Remain at 0
D.	Rises from initial load value	Rises from initial load value

Answer: C

Answer Explanation: C is correct. With loss of BB 12, EDG103 will not automatically start and its associated Containment Spray Pumps breaker control power is lost. EDG103 powered pumps via PB103 (121 and 122) will not start and since the EDG did not start, EDG103 load "remains at 0". EDG102 powered pumps via PB102 (111 and 112) will auto start and since EDG102 is started and has some initial kw load on it, its load "rises from initial load value". Since no actions per SOP-47A.1 are taken, no assumptions can be made regarding transfer of any DC control power to alternate sources (i.e., EDG control power or other transferable loads from BB 11).

A. is incorrect since EDG102 will load, but would be correct if EDGs did not start and load and Containment Spray pumps were starting on their offsite power sources.

B. and D. are incorrect since EDG103 will not start and load with BB 12 deenergized, but would be correct if BB 11 had been lost.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 42 Details

Question Type:	Multiple Choice
Topic:	NRC RO 42
System ID:	21267
User ID:	NRC 2006 RO 42
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Dec 18, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE** Enabling Objective: O1-OPS-001-226-1-01 EO-1.5

Question 42 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

PROC
- N1-SOP-47A.1 Rev. NA

NUREG 1123 KA Catalog Rev. 2
- 226001 A1.10 3/3.2 Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE controls including: Emergency generator loading

Level of Difficulty
- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 41(b)(7)

Question Source
- New

EXAMINATION ANSWER KEY

Reactor Operator Written Examination Nine Mile Point Unit 1 2007

43 **USERID: NRC 2006 RO 43** **SYSID: 21268** **Points: 1.00**

The plant is in an outage with the following:

- The core is offloaded and the cavity is drained
- An inadvertent valve closure causes a loss of Instrument Air to the Spent Fuel Pool Cooling (SFP) system

Which one of the following identifies the effect on Spent Fuel Pool Temperature and identifies an action to mitigate?

	Effect on temperature	Action to mitigate
A.	Lowers	Remove the operating SFP pump from service.
B.	Lowers	Throttle closed SFP heat exchanger RBCLC outlet valve.
C.	Rises	Throttle open SFP heat exchanger RBCLC outlet valve.
D.	Rises	Feed and Bleed using available makeup source and the condenser.

Answer: D

Answer Explanation: D. is correct – fuel pool cooling pumps trip on a loss of air which will cause pool temperature to rise, feed and bleed is available and directed from N1-SOP-6.1, LOSS OF SFP/RX CAVITY LEVEL/DECAY HEAT REMOVAL.

A. is incorrect – pool temperature will rise, if air system effect on the RBCLC supply to SFP heat exchanger is not understood (fails as is) and pump trip is not recognized could be selected

B. is incorrect – pool temp will rise.

C. is incorrect – temp will rise but without instrument air and the SFP pumps tripped, the normal method of temperature control will be ineffective.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

EXAMINATION ANSWER KEY

Reactor Operator Written Examination Nine Mile Point Unit 1 2007

Question 43 Details

Question Type:	Multiple Choice
Topic:	NRC RO 43
System ID:	21268
User ID:	NRC 2006 RO 43
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Dec 18, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	18025
Num Field 2:	
Text Field:	
Comments:	Technical References: N1-SOP-6.1 N1-SOP-20.1

Reference Provided: NONE

Enabling Objective: N1-233000 RBO 8

Question 43 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

PROC

- N1-SOP-6.1, Rev. NA
- N1-SOP-20.1, Rev. NA

NUREG 1123 KA Catalog Rev. 2

- 233000 A2.05 2.5/2.5 Ability to (a) predict the impacts of the following on the FUEL POOL COOLING AND CLEAN-UP; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Valve closures

Level of Difficulty

- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level

- 2 Higher Cognitive Level comprehensive

10CFR55

- 41(b)(7)

Question Source

- Bank

44 **USERID: NRC 2006** **RO 44** **SYSID: 21269** **Points: 1.00**

The plant is operating at 100% power with the following:

- Annunciator H3-2-4 FW HTR 121-125 LEVEL HIGH alarms
- Computer point A035 FW HTR 124 LVL HIGH is indicated
- FW HTR 124 level control valve is closed
- Feedwater temperature entering the vessel is 361°F and stable

Which one of the following identifies the actions required by plant procedures?

A. Confirm Extraction Steam supply to heater is isolated.

B. Take manual control of Feedwater Heater level control valve.

C. Take manual control of Moisture Separator Drain tank level control valve.

D. Lower power to restore feedwater temperature above minimum allowable value.

Answer: B

Answer Explanation: B. is correct. When FW HTR level reaches the high setpoint H3-2-4 alarm actuates. With the level control valve malfunctioning (closed, when level is high), the correct action is to take manual control per N1-OP-16 section H.

A. is incorrect because the extraction steam supply to the heater does not isolate until high-high level (H3-2-5) is received.

C. is incorrect because this action is not directed. Moisture Separator drain tanks do drain to the 124 feedwater heater (plausible).

D. is incorrect because FW temperature entering the vessel is stable and is already above the minimum allowable value in SOP-16.1. Given temperature is above 350°F, which is minimum for 100%. There is also no indication provided that 3°F difference in FW temperatures exist.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 44 Details

Question Type:	Multiple Choice
Topic:	NRC RO 44
System ID:	21269
User ID:	NRC 2006 RO 44
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Dec 18, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Technical References: N1-260000 Feedwater Heating and Extraction Steam N1-ARP-H3-2-4 N1-ARP-H3-2-5 **Reference Provided: NONE** Enabling Objective: N1-260000 RBO-11

Question 44 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

PROC

- N1-ARP-H3 Rev. Item 2-4, 2-5

NUREG 1123 KA Catalog Rev. 2

- 256000 A3.07 2.9/2.9 Ability to monitor automatic operations of the REACTOR CONDENSATE SYSTEM including: Feedwater heater level

Level of Difficulty

- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level

- 2 Higher Cognitive Level comprehensive

10CFR55

- 41(b)(7)

Question Source

- New

45 **USERID: NRC 2006 RO 45** **SYSID: 21270** **Points: 1.00**

The plant is operating at 100% power with the following:

- FWP 11 and FWP 13 are running
- FWBP 13 is tagged out for maintenance
- THEN A Loss of all Offsite Power occurs
- Turbine trips on load reject
- All systems operate as designed

Which one of the following HPCI components automatically start upon power restoration?

	Condensate Pump	FW Booster Pump	FW Pump
A.	11	11	11
B.	11	12	11
C.	13	12	12
D.	13	11	12

Answer: D

Answer Explanation: D. is correct. The preferred HPCI components are Condensate Pump 13, FW Booster Pump 13 and FW Pump 12. With FW Booster pump 13 tagged out for maintenance (FW Booster Pump 13 in PTL), FW Booster Pump 11 will start.

A. is incorrect. Condensate pump 11 and FW Pump 11 will not start when power is restored.

C. is incorrect. When FW Booster Pump 13 is in PTL, HPCI logic will start 11 FW Booster Pump not 12 FW Booster Pump.

B. is incorrect. None of these pumps start on power restoration. However, under normal low power operating conditions this is an allowable pump configuration.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 45 Details

Question Type:	Multiple Choice
Topic:	NRC RO 45
System ID:	21270
User ID:	NRC 2006 RO 45
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 30, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Technical Reference: N1-259001 N1-OP-16 **Reference Provided: NONE** Enabling Objective: N1-259001 RBO 8

Question 45 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

PROC

- N1-OP-16 Rev. NA

NUREG 1123 KA Catalog Rev. 2

- 259001 K1.07 2.9/2.9 Knowledge of the physical connections and/or cause-effect relationship between REACTOR FEEDWATER SYSTEM and the following: A.C. electrical power

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 2 Higher Cognitive Level comprehensive

10CFR55

- 41(b)(7)

Question Source

- New

46 **USERID: NRC 2006 RO 46** **SYSID: 21271** **Points: 1.00**

The plant is operating at 100% power with the following:

- Hydrogen Water Chemistry (HWC) is out of service,
- Recombiner 11 is in service
- THEN L2-4-3, OFF-GAS H2 MONITOR SYSTEM 11 OFF-NORMAL, alarms
- Computer point indicates Offgas System 11 Hydrogen is reading 1.8% and slowly rising

Which one of the following describes the required operator actions?

A. Reduce load to 300 MWe then place Recombiner 12 in service.

B. Commence a normal reactor shutdown if hydrogen reaches 2%.

C. Perform a normal reactor shutdown before hydrogen reaches 4%.

D. Manually scram the reactor if hydrogen exceeds 4%.

Answer: A

Answer Explanation: A. is correct. If hydrogen remains 1.5% then the standby recombiner must be placed in service per N1-OP-25 Section F.9.0. N1-OP-25 Section F.9.0 requires generator load to be reduced to 300 MWe prior to placing the standby recombiner in service.

B. and C. are incorrect. A normal orderly shutdown is only required if Hydrogen concentration exceeds 4%.

D. is incorrect. If Hydrogen Concentration exceeds 4% a normal orderly shutdown is required not a reactor scram.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 46 Details

Question Type:	Multiple Choice
Topic:	NRC RO 46
System ID:	21271
User ID:	NRC 2006 RO 46
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Jan 17, 2007
Cross Reference Number:	LC1 05-01
Num Field 1:	16303
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: N1-OP-25 Section F only; ARP L2-3-1 through 4-8 (17 pages).** Enabling Objective: O1-OPS-001-271-1-01 EO-1.7

Question 46 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

PROC

- N1-ARP-L2 Rev. NA
- N1-OP-25 Rev. NA

NUREG 1123 KA Catalog Rev. 2

- 271000 K5.06 2.7/2.7 Knowledge of the operational implications of the following concepts as they apply to OFFGAS SYSTEM: Catalytic recombination

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 2 Higher Cognitive Level comprehensive

10CFR55

- 41(b)(7)

Question Source

- Bank

47 **USERID: NRC 2006 RO 47** **SYSID: 21272** **Points: 1.00**

The plant is in a refueling outage ready to start the core offload.

- A new fuel assembly was just placed in the fuel preparation machine.
- Before the fuel preparation machine is lowered, the refueling floor SRO observes that spent fuel pool level is lowering.

Which one of the following states the **INITIAL** point at which evacuation of ALL personnel from the refuel floor is required, per N1-SOP-6.1, LOSS OF SFP/RX CANTILEVER/DECAY HEAT REMOVAL?

A. When the radiation monitor on the Refueling Bridge alarms.

B. When the assembly in the fuel prep machine becomes uncovered.

C. When the fuel pool water level lowers to the FSAR limit of 24 feet.

D. When the fuel pool water level is approaching the top of the fuel pool racks.

Answer: A

Answer Explanation: A. is correct. While executing the steps of N1-SOP-6.1 if an irradiated fuel bundle has been uncovered or if the Refueling Bridge high radiation alarm sounds, then the 340' elevation of the reactor building must be evacuated. Until either of these conditions are present, the only requirement is to evacuate only unnecessary personnel, not all personnel.

B. is incorrect. Uncovering a new fuel assembly will not provide any radiological hazard and does not require evacuation of all personnel. If the fuel assembly in the fuel preparation machine were an irradiated fuel assembly, then this response would be correct.

C. is incorrect. The conditions requiring a complete evacuation of the refuel floor are independent of the spent fuel pool level, although level will determine the shielding available and ultimately the conditions for evacuation.

D. is incorrect. Although irradiated fuel may eventually become uncovered, the refueling bridge high radiation alarm will have already sounded. Level given is "approaching" the top of the fuel bundles, but not actually "at" the top.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 47 Details

Question Type:	Multiple Choice
Topic:	NRC RO 47
System ID:	21272
User ID:	NRC 2006 RO 47
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 30, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	21111
Num Field 2:	
Text Field:	
Comments:	Technical References: N1-SOP-6.1, Override Step **Reference Provided: NONE** Enabling Objective: O1-OPS-001-233-1-01 EO-8

Question 47 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

PROC

- N1-SOP-6.1, Rev. NA

NUREG 1123 KA Catalog Rev. 2

- 272000 A2.09 3.1/3.3 Ability to (d) predict the impacts of the following on the RADIATION MONITORING SYSTEM: and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low fuel pool level

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 1 Memory/Fundamental Knowledge

10CFR55

- 41(b)(5)

Question Source

- Bank

48 **USERID: NRC 2006** **SYSID: 21273** **Points: 1.00**
RO 48

The plant is experiencing a transient with the following:

t = 0 min	Drywell Pressure rises to 4.2 psig
t = 1 min	Reactor Building Emergency Ventilation (RBEVS) is in standby
t = 1 min	Control Room Emergency Ventilation (CREVS) is in standby

Which one of the following actions is required, if any, to align these plant ventilation systems to the correct configuration for the transient?

A. No action is required for RBEVS. No action required for CREVS.

B. No action is required for RBEVS. Start a CREVS train.

C. Trip RB normal ventilation and manually start an RBEVS train. Start a CREVS train.

D. Trip RB normal ventilation and manually start an RBEVS train. No action is required for CREVS.

Answer: B

Answer Explanation: B. is correct. CREVS receives an automatic initiation signal from Core Spray Logic. The LOCA signal is High DW pressure (>3.5 psig) or RPV Lo-Lo water level (5 inches). If CREVS is still in standby when an initiation signal is received, it must be manually started. RBEVS does not receive a LOCA start signal on high drywell pressure. RBEVS starts on RB Vent radiation monitor above 5 mr/hr. The RBEVS requires no action, since it has responded properly to these conditions.

A. is incorrect because action is required for CREVS, since it has failed to respond correctly to the high DW pressure signal.

C. is incorrect because RBEVS requires no action, since it has responded properly to these conditions

D. is incorrect because RBEVS requires no action, since it has responded properly to these conditions and CREVS does require action.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 48 Details

Question Type:	Multiple Choice
Topic:	NRC RO 48
System ID:	21273
User ID:	NRC 2006 RO 48
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Dec 18, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Technical References: O1-OPS-001-288-1-03

Reference Provided: NONE

Enabling Objective: O1-OPS-001-288-1-03 EO-1.7

Question 48 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2

- 288000 A2.03 3.5/3.7 Ability to (a) predict the impacts of the following on the PLANT VENTILATION SYSTEMS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of coolant accident: Plant-Specific

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 2 Higher Cognitive Level comprehensive

10CFR55

- 41(b)(5)

Question Source

- New

49 **USERID: NRC 2006 RO 49** **SYSID: 21274** **Points: 1.00**

The plant is operating at power with the following:

0800 All Reactor Recirc Pumps (RRPs) are running
0800 Total Core Flow is 50 Mlbm/hr
0805 Reactor Recirc Pump (RRP) 11 trips
0805 Post-RRP trip flow is as follows (Mlbm/hr):

RRP 11	4
RRP 12	10
RRP 13	11
RRP 14	10
RRP 15	11

0805 NO operator action is taken for the tripped pump

Which one of the following is the approximate value for actual core flow?

A. 46 Mlbm/hr

B. 42 Mlbm/hr

C. 40 Mlbm/hr

D. 38 Mlbm/hr

Answer: D

Answer Explanation: D. is correct. For the tripped pump, the 4 Mlbm/hr indicated flow is actually reverse flow. The Total Core Flow senses this as forward flow, so would indicate 46 Mlbm/hr. Since operator must manually determine flow under this condition, two times the indicated flow must be subtracted. 46-8=38 Mlbm/hr.

A. is incorrect. This would be the total of all pump flows, assuming the flow for the tripped pump is still forward flow.

B. is incorrect. But this value incorrectly subtracts only the 4 Mlbm/hr that is indicated.

C. is incorrect. But assumes that 10 Mlbm is lost from initial core flow of 50 Mlbm.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

EXAMINATION ANSWER KEY

Reactor Operator Written Examination Nine Mile Point Unit 1 2007

Question 49 Details

Question Type:	Multiple Choice
Topic:	NRC RO 49
System ID:	21274
User ID:	NRC 2006 RO 49
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Dec 18, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE** Enabling Objective: N1-202001 RBO-11, Reactor Recirculation System

Question 49 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2

- 295001 AA2.06 3.2/3.3 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Nuclear boiler instrumentation

Level of Difficulty

- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level

- 2 Higher Cognitive Level comprehensive

10CFR55

- 41(b)(10)

Question Source

- New

50 **USERID: NRC 2006** **SYSID: 21275** **Points: 1.00**
RO 50

The plant is operating at 100% when an ATWS occurs. The following conditions exist:

- CRD Pump 11 is operating
- CRD Pump 12 is in standby
- Liquid Poison (LP) Pump 11 is injecting
- THEN A loss of offsite power occurs
- BOTH Diesel Generators start and power their respective Power Boards.

Which one of the following describes the effect of these conditions on the LP Pump 11 AND CRD Pump operation?

	LP Pump 11 Status	CRD Pumps Running
A.	Running	Only CRD Pump 11
B.	Running	BOTH CRD Pumps
C.	Tripped	Only CRD Pump 11
D.	Tripped	BOTH CRD Pumps

Answer: B

Answer Explanation: B is correct. 86-16 trips on the loss of power to PB102 to initiate load shedding to reduce the loading on the diesels when they start and energize their busses. Liquid Poison pump 11 breaker is one of the loads that is not effected by the trip of the 86 device. When power is restored to the power board, the LP 11 pump will restart. CRD Pump 12 receives a start signal from the 86-17, which trips on loss of power to PB103.

A. is incorrect. CRD Pump 12 is now running because it receives a start signal from 86-17 actuation.

C. is incorrect. Resetting the 86-16 is not necessary for the LP pump to restart. LP Pump does not receive a trip signal when the 86-16 is actuated. Also, CRD Pump 12 is now running because it receives a start signal from 86-17 actuation.

D. is incorrect. Resetting the 86-16 is not necessary for the LP pump to restart. LP Pump does not receive a trip signal when the 86-16 is actuated.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 50 Details

Question Type:	Multiple Choice
Topic:	NRC RO 50
System ID:	21275
User ID:	NRC 2006 RO 50
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 30, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE** Enabling Objective: O1-OPS-001-262-1-02 EO-1.4.b For each system interrelated with the Liquid Poison System: Predict the effects on the system of a loss of the interrelating system.

Question 50 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2

- 295003 AK1.02 3.1/3.4 Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: Load shedding

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 1 Memory/Fundamental Knowledge

10CFR55

- 41(b)(8)

Question Source

- New

51 **USERID: NRC 2006 RO 51** **SYSID: 21276** **Points: 1.00**

The plant experienced a scram from 15% power with the following:

- MSIVs are closed
- RPV Pressure is 1000 psig and rising slowly
- The plant is being brought to Cold Shutdown
- THEN a loss of Battery Board 11 occurs

Which one of the following describes the impact of the power loss on the ability to use Emergency Condenser and ERV systems from the control room to perform plant cooldown?

A. Neither EC is available. ERVs must be used.

B. EC 12 is available but EC 11 is NOT. ERVs can be used.

C. Both ECs are available. ERVs CANNOT be used.

D. Both ECs are available. ERVs can be used.

Answer: D

Answer Explanation: D. is correct. Condensate Return Valves have two DC solenoids (one from BB 11 and the other from BB12), which both must deenergize for the Condensate Return to open and place an EC in service. Since the loss of BB 11 will deenergize one of EC 12's Condensate return valve AND one of EC 11s solenoids, Both ECs can still be placed in service using control switch. Three ERVs also still have power, so they are also still available for use.

A. and B. are incorrect because the power loss does NOT disable or prevent either EC from performing its function to cooldown the plant

C. is incorrect, because 3 ERVs are powered from BB 11 and 3 are powered from BB 12. ERVs CAN still be used to cooldown.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 51 Details

Question Type:	Multiple Choice
Topic:	NRC RO 51
System ID:	21276
User ID:	NRC 2006 RO 51
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 30, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE** Enabling Objective: N1-207000 RBO 8

Question 51 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

PROC

- N1-SOP-47A.1 Rev. NA

NUREG 1123 KA Catalog Rev. 2

- 295004 AA1.02 3.8/4.1 Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: Systems necessary to assure safe plant shutdown

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 2 Higher Cognitive Level comprehensive

10CFR55

- 41(b)(7)

Question Source

- New

52 **USERID: NRC 2006 RO 52** **SYSID: 21277** **Points: 1.00**

The plant is operating at 35% power with the following:

- A Main Generator Lockout trip occurs

Which one of the following describes the effect on reactor power as indicated on the APRMs following the transient and why?

A. Lowers due to rapid control rod insertion.

B. Lowers due to reduction in total core flow.

C. Rises due to stopping all main steam line flow.

D. Rises due to reduction in feedwater temperature.

Answer: D

Answer Explanation: D. is correct. Feedwater temperature lowers due to closure of Non Return Valves on turbine trip. The feedwater temperature reduction causes reactor power to rise slowly.

A. is incorrect because the reactor will not scram on the generator/turbine trip, since power is below 45%.

B. is incorrect because core flow does not change during the transient. No electrical power is lost to any busses.

C. is incorrect because all main steam line flow is not stopped, there is no MSIV closure for this transient. Turbine bypass valve capacity is sufficient to accommodate the steam flow demand with power below 45%. Some pressure change may be seen due to valve response time.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 52 Details

Question Type:	Multiple Choice
Topic:	NRC RO 52
System ID:	21277
User ID:	NRC 2006 RO 52
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Jan 31, 2007
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE** Enabling Objective: N1-245000 RBO-11, Main Turbine N1-26000 RBO-8, Feedwater Heaters & Extraction Steam

Question 52 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2

- 295005 AA2.06 2.6/2.7 Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP: Feedwater temperature

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 2 Higher Cognitive Level comprehensive

10CFR55

- 41(b)(10)

Question Source

- New

EXAMINATION ANSWER KEY

Reactor Operator Written Examination Nine Mile Point Unit 1 2007

53 **USERID: NRC 2006 RO 53** **SYSID: 21278** **Points: 1.00**

The plant experiences a transient from 100% power with the following:

t = 0 sec:	Mode Switch is placed in SHUTDOWN
t = 0 sec:	RPS Group 1-4 white lights for both RPS Channel 11 and 12 are extinguished
t = 25 sec:	All Control Rods remain at pre-scram positions
t = 45 sec:	RPS Manual Scram pushbuttons are depressed with no effect
t = 1:00 min:	Feedwater injection is terminated
t = 1:30 min:	RPV water level drops to 53 inches
t = 2:00 min:	Reactor power is 25%
t = 3:00 min:	Generator MWe are 0

Which one of the following identifies when the Main Turbine receives its trip signal?

A. 5 sec

B. 50 sec

C. 1:35 min

D. 3:00 min

Answer: C

Answer Explanation: C. is correct. RPS, by design, initiates a Turbine trip five seconds after the automatic scram is received and the logic channels trip. In this case, with no rod motion, plant conditions such as level and pressure are stable and will not exceed setpoints. An automatic scram signal will not be sensed by RPS until injection is terminated and level drops below 53 inches, generating an auto scram.

A. is incorrect because placing the Mode Switch in SHUTDOWN initiates a scram through the manual scram portion of the scram logic.

B. is incorrect because pushing the manual scram buttons initiates a scram through the manual scram portion of the scram logic.

D. is incorrect because the turbine will already be tripped from the automatic tripping of RPS channels, prior to any reverse power condition.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 53 Details

Question Type:	Multiple Choice
Topic:	NRC RO 53
System ID:	21278
User ID:	NRC 2006 RO 53
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 30, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Technical References: N1-248000 Main Turbine Controls **Reference Provided: NONE** Enabling Objective: N1-248000 RBO-11

Question 53 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2

- 295006 AK2.04 3.6/3.7 Knowledge of the interrelations between SCRAM and the following: Turbine trip logic: Plant-Specific

Level of Difficulty

- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level

- 2 Higher Cognitive Level comprehensive

10CFR55

- 41(b)(7)

Question Source

- New

54 **USERID: NRC 2006 RO 54** **SYSID: 21279** **Points: 1.00**

The following plant conditions exist following a reactor scram:

- Control Room Evacuation is in progress
- Reactor pressure is 1090 psig and lowering slowly
- Reactor water level is 100 inches and lowering slowly
- Both ECs have automatically initiated before establishing control at RSP11

Per SOP-21.2, Control Room Evacuation, which one of the following actions can be taken to control the cool down rate from RSP 11?

A. Place CHANNEL 11 CONTROL TRANSFER switch in EMER and then cycle EC Steam Supply IV (39-09R).

B. Place CHANNEL 11 CONTROL TRANSFER switch in EMER and then cycle EC Condensate Return IV (39-05).

C. Place EMERGENCY COOLING ISOLATION BYPASS switch in BYPASS and then cycle EC Steam Supply IV (39-09R).

D. Place EMERGENCY COOLING ISOLATION BYPASS switch in BYPASS and then cycle EC Condensate Return IV (39-05).

Answer: A

Answer Explanation: A. is correct. Since the EC auto initiated (>1080 psig for 12 seconds) before control was taken at RSP 11, operation of the EC Condensate Return valve (the desired means) is unavailable, because the automatic initiation has actuated the DC solenoids. When operating the condensate return from the RSP, the AC solenoids are used to reposition the valve.

B is incorrect because the only method to reduce cool down rate is to throttle on the steam supply. To operate the EC from the RSP the CHANNEL 11 CONTROL TRANSFER switch is placed in EMER.

C and D are incorrect because the EMERGENCY COOLING ISOLATION BYPASS switch is not operated unless isolation occurs. There is no isolation condition present. In addition placing these switches in BYPASS will not transfer control of the valves to the RSP.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question Type:	Multiple Choice
Topic:	NRC RO 54
System ID:	21279
User ID:	NRC 2006 RO 54
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**References Provided: None**

Note: Basis for KA statement match. Given that both ECs are in operation and that RPV cooldown is excessive (about 300°F/hr), prompt action (immediate operation of system components) related to operation of ECs is warranted. The actions identified in the question are contained in an override step of the SOP. Training material O1-OPS-001-207-1-01, EO-1.7.g requires the candidate to describe, from memory, basic operational actions from SOP's. The candidate must also have an understanding of how the EC's are operated from the RSP, following an automatic initiation.

Enabling Objective: O1-OPS-001-207-1-01, EO-1.7

Question 1 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

PROC
- N1-OP-13 Rev. NA
- N1-SOP-21.2, Rev. NA

NUREG 1123 KA Catalog Rev. 2
- G2.4.49 4/4 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls
- 295016 Control Room Abandonment

Level of Difficulty
- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 41(b)(7)

Question Source
- Bank

55 **USERID: NRC 2006 RO 55** **SYSID: 26976** **Points: 1.00**

The plant is operating at 60% power with the following:

- Core Flow is 27 Mlbm/hr
- Operating TBCLC pump trips
- Standby TBCLC pump CANNOT be started
- CRS intends to direct initiation of a manual scram

Which one of the following actions is to be taken before the scram per N1-SOP-24.1, TBCLC Failure?

A. Reduce power by lowering Recirc Flow.

B. Trip all running Recirc Motor Generator sets.

C. Manually initiate both Emergency Condensers.

D. Transfer house loads to Offsite power supply.

Answer: D

Answer Explanation: D is correct. Prior to the scram, house loads are transferred to the reserve power transformers.

A is incorrect. reducing recirc flow from the initial conditions provided (60% power and 27 Mlbm/hr flow) will result in entry into the restricted zone. This is not in compliance with SOP direction.

B is incorrect. Tripping motor generator sets is an action that is taken after the scram.

C is incorrect. Initiating Emergency Condensers is an action taken after the scram.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 55 Details

Question Type:	Multiple Choice
Topic:	NRC RO 55 REV 1
System ID:	26976
User ID:	NRC 2006 RO 55
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Jan 17, 2007
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	O1-OPS-001-274-1-01 EO-1.7.f **Reference Provided to Candidate: Power to Flow Maps**

Question 55 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

LP
- O1-OPS-001-274-1-01 Rev. na

PROC
- N1-SOP-24.1, Rev. NA

NUREG 1123 KA Catalog Rev. 2
- 295018 AK2.02 3.4/3.6 Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER and the following: Plant operations

Level of Difficulty
- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level
- 1 Memory/Fundamental Knowledge

10CFR55
- 41(b)(7)

Question Source
- New

56 **USERID: NRC 2006 RO 56** **SYSID: 21281** **Points: 1.00**

The plant is operating at 100% power with the following:

- Instrument Air Header pressure is lowering
- Annunciator F3-3-2, CRD CONTROL AIR PRESSURE HI-LO, is in alarm
- F panel chart recorder indicates CRD air header pressure is 60 psig
- CRS directs a manual reactor scram be inserted

Which one of the following is the reason for initiating the manual reactor scram?

A. Scram Discharge Volume may isolate.

B. Control Rods may begin to drift into the core.

C. CRD flow control valve has failed to its minimum position.

D. Cooling water flow is lost to all control rod drive mechanisms.

Answer: B

Answer Explanation: B. is correct. With air header pressure below 60 psig, N1-SOP-20.1 and ARP F3-3-2 direct a manual scram, because scram inlet and outlet valves will start opening, scramming some control rods.
A. is incorrect. The SDV vents and drain close on loss of air, but manual scram is not required. Tech Specs allows 24 hours to restore.
C. is incorrect. CRD system FCV does fail closed, but manual scram is not required based on low system flow.
D. is incorrect. Cooling flow is lost when the FCV closes, but elevated drive temperatures does not require a manual scram.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

EXAMINATION ANSWER KEY

Reactor Operator Written Examination Nine Mile Point Unit 1 2007

Question 56 Details

Question Type:	Multiple Choice
Topic:	NRC RO 56
System ID:	21281
User ID:	NRC 2006 RO 56
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 30, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Technical References: N1-SOP-20.1 N1-ARP-F3-3-2

Reference Provided: NONE

Enabling Objective: N1-201000 RBO 8

Question 56 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

PROC

- N1-SOP-19 Rev. NA

NUREG 1123 KA Catalog Rev. 2

- 295019 AK2.01 3.8/3.9 Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following: CRD hydraulics

Level of Difficulty

- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level

- 2 Higher Cognitive Level comprehensive

10CFR55

- 41(b)(7)

Question Source

- Bank

57 **USERID: NRC 2006 RO 57** **SYSID: 21282** **Points: 1.00**

The plant is shutdown and Refueling is in progress with the following:

- Shutdown Cooling (SDC) loops 11 and 12 are in service with forced recirculation secured
- An equipment malfunction results in a loss of instrument air

Which one of the following is the effect, if any, on SDC flow through the core and RBCLC flow through the SDC heat exchangers?

	SDC Flow through Core	RBCLC Flow through SDC HX
A.	Remains the same	Remains the same
B.	Remains the same	Drops to zero
C.	Drops to zero	Remains the same
D.	Drops to zero	Drops to zero

Answer: D

Answer Explanation: D is correct. The instrument air system provides motive force to operate the electropneumatic flow control valves and the minimum flow control valves. It provides motive force for RBCLC inlet to SDC. On a loss of air pressure, the flow control valves close and the minimum flow control valves open. Also, the RBCLC blocking valve fails closed. Therefore both flows will drop to zero.

A is incorrect because both valves will close resulting in no flow. Flow does not remain the same.

B is incorrect because SDC flow drops to zero.

C is incorrect because RBCLC flow drops to zero.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 57 Details

Question Type:	Multiple Choice
Topic:	NRC RO 57
System ID:	21282
User ID:	NRC 2006 RO 57
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 30, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE** Enabling Objectives: N1-205000 RBO-8 O1-OPS-001-205-1-01 EO-1.5.b For each system interrelated with the Shutdown Cooling system: Predict the effects on the system of a loss of the interrelating system.

Question 57 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2

- 295021 AA2.02 3.4/3.4 Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: RHR/shutdown cooling system flow

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 1 Memory/Fundamental Knowledge

10CFR55

- 41(b)(10)

Question Source

- Bank

58 **USERID: NRC 2006** **SYSID: 21283** **Points: 1.00**
RO 58

The plant is in a refueling outage with the following:

- A fuel bundle being moved from its core location to the Spent Fuel Pool has just been dropped from the grapple
- The dropped bundle is now in the transfer canal
- No control room alarms have been received

Which one of the following identifies the actions required by N1-SOP-34, Dropped Fuel Assembly?

A. Start an RBEVS train and maintain CREVS system in standby.

B. Manually start an RBEVS train and start the CREVS system.

C. Notify the Shift Manager to initiate a protected area evacuation.

D. Direct personnel remaining in the area to stand clear of the canal.

Answer: B

Answer Explanation: B. is correct. N1-SOP-34 requires both the RBEVS and CREVS systems be verified initiated, if a spent fuel bundle is dropped.

A. is incorrect because starting a CREVS train is also required, if a spent fuel bundle is dropped.

C. is incorrect because a protected area evacuation is not required. Evacuation of the Refuel Floor and Drywell are required, if a spent fuel bundle is dropped.

D. is incorrect. The bundle dropped is not a new fuel bundle. Dropping a spent fuel bundle requires the refuel floor to be evacuated. There should be no personnel remaining on the floor, since an evacuation is required.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 58 Details

Question Type:	Multiple Choice
Topic:	NRC RO 58
System ID:	21283
User ID:	NRC 2006 RO 58
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Jan 25, 2007
Cross Reference Number:	LC1 05-01
Num Field 1:	16320
Num Field 2:	
Text Field:	
Comments:	Technical References: N1-234000 N1-SOP-34

Reference Provided: NONE

Enabling Objective: N1-234000 RBO 10
Objective requires that initial candidates demonstrate the ability to restate in their own words the general sequence of operational actions. Additionally, given a set of conditions the candidate is expected to assess conditions to determine the required configuration. These are basic operator actions for a dropped fuel bundle.

Question 58 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

PROC

- N1-SOP-34, Rev. NA

NUREG 1123 KA Catalog Rev. 2

- 295023 AA1.07 3.6/3.6 Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS: Standby gas treatment/FRVS

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 1 Memory/Fundamental Knowledge

10CFR55

- 41(b)(7)

Question Source

- Bank

59 **USERID: NRC 2006** **SYSID: 21284** **Points: 1.00**
RO 59

The plant is operating at 100% power with the following:

- N1-ST-Q1C CS 112 PUMP AND VALVE OPERABILITY TEST is in progress
- Valve 40-12 CORE SPRAY DISCHARGE IV 11 (OUTSIDE) is closed
- Valve 40-06 CORE SPRAY LOOP 11 TEST VALVE TO TORUS is open
- Core Spray 112 AND Core Spray Topping Pump 112 are running

THEN the following events occur:

- PB103 deenergizes and EDG103 fails to start
- Drywell Pressure rises to 13.0 psig
- RPV pressure is 355 psig and lowering
- Core Spray EOP jumpers are NOT installed

Which one of the following describes the status of Core Spray System 11?

A. NOT injecting to the RPV and system flow to the torus maintained through only the minimum flow relief valves.

B. NOT injecting to the RPV with system flow to the torus maintained through the open 40-06 test valve.

C. IS injecting to the RPV with flow to the torus isolated by the closed 40-06 test valve.

D. IS injecting to the RPV with flow to the torus maintained through the open 40-06 test valve.

Answer: C

Answer Explanation: C is correct. Core Spray initiates on high drywell pressure above 3.5 psig. When in a test lineup the system realigns for injection by closing the 40-06 test valve when initiation occurs, isolating flow to the torus. Since RPV pressure is below 365 psig, the 40-12 outside isolation opens. Inside IV 40-10 and 40-11 also open below 365 psig. Injection occurs with 40-06 isolated. Effect of PB103 loss is that only one system pump and topping pump from PB102 start. Also, since the 40-06 and 40-12 are powered from Powerboard 167, which can be energized by either PB102 or PB103. With PB103 deenergized, PB167 will still be energized from PB102. This maintains power to the outside and test IVs, making them single failure proof for a single diesel failure during operability testing.
A. is incorrect because the system is designed to realign and inject, even with loss of an EDG. Flow through min flow relief valves would occur if pressure were above 365 psig and injection valves were closed.
B. is incorrect because the loop is injecting. These conditions would occur if the loss of power prevented the test valve from closing and the outside IV from opening.
D. is incorrect because the 40-06 closes to stop flow to the torus. If the test valve remained open due to the power loss and the outside IV repositioned, this condition can result.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 59 Details

Question Type:	Multiple Choice
Topic:	NRC RO 59
System ID:	21284
User ID:	NRC 2006 RO 59
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 30, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Technical References: N1-209001 SDBD-201

Reference Provided: NONE

Enabling Objective: N1-209001 RBO 5

Question 59 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

OTHER_REFS

- (SYSTEM) SDBD-201 Rev. na

NUREG 1123 KA Catalog Rev. 2

- 295024 EK2.03 3.8/3.8 Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following: LPCS: Plant-Specific

Level of Difficulty

- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level

- 2 Higher Cognitive Level comprehensive

10CFR55

- 41(b)(7)

Question Source

- New

60 **USERID: NRC 2006** **SYSID: 21285** **Points: 1.00**
RO 60

The plant is experiencing a transient with the following:

- Drywell Pressure is 3.0 psig and steady
- RPV water level is 20 inches and rising
- RPV pressure peaked at 1150 psig and is now lowering
- All RRPs are tripped

Which one of the following is the reason for the automatic RRP trip?

A. Rapidly reduces reactor power during an ATWS.

B. Rapidly reduces RPV water level during an ATWS.

C. Prevents excessive RPV inventory loss during a LOCA.

D. Prevents motor damage when sprays are initiated during a LOCA.

Answer: A

Answer Explanation: A. is correct. ATWS-RPT is initiated when RPV pressure exceeds 1135 psig. The trip initiates a trip of RRP MG field breakers and drive motor breakers to reduce Recirc flow to establish natural circulation conditions. This results in increasing voids, adding negative reactivity and lower power to mitigate the effects of the ATWS.
B. is incorrect. RRP trip is not initiated to reduce water level. Reducing water level during an ATWS is a manual action to drop power level during an ATWS.

C and D are incorrect. The RRPs will trip on high drywell pressure of 3.5 psig, which is indicative of a LOCA. Since DWP provided is below the setpoint, the RRP trip was not caused by the LOCA signal.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 60 Details

Question Type:	Multiple Choice
Topic:	NRC RO 60
System ID:	21285
User ID:	NRC 2006 RO 60
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 30, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE** Enabling Objective: O1-OPS-001-212-1-01 EO-1.2

Question 60 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2
- 295025 EK3.02 3.9/4.1 Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE: Recirculation pump trip: Plant-Specific

Level of Difficulty
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 41(b)(5)

Question Source
- New

61 **USERID: NRC 2006 RO 61** **SYSID: 21286** **Points: 1.00**

Given the following:

- At 12:15, Torus Temperature reaches 85.10°F
- At 12:19, Torus Temperature reaches 86.10°F
- All four Containment Spray loops are available

Which one of the following is the **latest time** that Torus Cooling shall be placed in service?

A. No later than 12:30

B. No later than 12:34

C. No later than 12:45

D. No later than 12:49

Answer: A

Answer Explanation: a. is correct - per N1-EOP-1, Att. 16, step 2.1, Torus cooling shall be in service within 15 minutes of Torus temperature equal to or greater than 85F.

b., c., d. incorrect - uses greater than 85F and/or substitutes 30 min for 15 min

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 61 Details

Question Type:	Multiple Choice
Topic:	NRC RO 61
System ID:	21286
User ID:	NRC 2006 RO 61
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Dec 18, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE** Enabling Objective: O1-OPS-006-344-1-23 EO-1.5, Given a specific step in the EOPs, justify why the action is appropriate.

Question 61 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2

- 295026 EA1.01 4.1/4.1 Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Suppression pool cooling

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 1 Memory/Fundamental Knowledge

10CFR55

- 41(b)(7)

Question Source

- Bank

62 **USERID: NRC 2006 RO 62** **SYSID: 21287** **Points: 1.00**

The plant has scrammed with the following:

- Drywell pressure is 13 psig and rising slowly
- Drywell temperature is 298°F and rising slowly
- CRS directs Containment Spray initiated

Which one of the following is the basis for initiating containment spray, based on current parameters?

A. Pressure suppression function is about to be lost.

B. ADS qualification temperature is about to be exceeded.

C. Drywell design temperature limit is about to be exceeded.

D. Primary Containment Pressure Limit is about to be exceeded.

Answer: B

Answer Explanation: B. is correct. N1-EOP-4 gives direction to initiate containment spray before reaching 300 °F. ADS environmental qualification temperature is 301°F and is the basis for initiating spray before 300 °F.

A. and D are incorrect. Pressure suppression capability is not threatened with DWP at 13 psig. Containment spray is not initiated based on this value. Primary Containment Pressure limit is significantly above 13 psig, so the limit is not the bases for directed spray.

C. is incorrect. The DWT design limit is 310 °F and spray is initiated before 300 °F based on the ADS qual temperature.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

EXAMINATION ANSWER KEY

Reactor Operator Written Examination Nine Mile Point Unit 1 2007

Question 62 Details

Question Type:	Multiple Choice
Topic:	NRC RO 62
System ID:	21287
User ID:	NRC 2006 RO 62
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Dec 18, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: None** Enabling Objective: O1-OPS-006-344-1-04 EO-1.2

Question 62 Cross References (table item links)

NUREG 1123 KA Catalog Rev. 2

- 295028 EK1.02 2.9/3.1 Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE: Equipment environmental qualification

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 1 Memory/Fundamental Knowledge

10CFR55

- 41(b)(8)

Question Source

- New

63 **USERID: NRC 2006 RO 63** **SYSID: 21288** **Points: 1.00**

The plant is experiencing a torus leak with the following:

- Reactor is manually scrammed
- Torus water level is 8.5 feet and dropping
- Torus water makeup is in progress
- RPV pressure is being controlled with Alternate Pressure Control systems between 600 to 800 psig
- Torus water temperature is 110°F and rising slowly
- An RPV Blowdown is required

Which one of the following describes the reason for the blowdown?

A. Downcomer openings are about to be uncovered.

B. Torus water temperature is about to exceed HCTL.

C. ERV discharge tailpipe holes are about to be uncovered.

D. Core Spray pump NPSH limits are about to be exceeded.

Answer: C

Answer Explanation: C. is correct per the EOP bases. Below 8 feet the ERV Y quencher holes begin to uncover. This can lead to pressurization of the suppression chamber air space.

Distractor: A. is incorrect because downcomers will uncover when level drops to about 6 feet. The ERV Y quenchers uncover first.

Distractor: B is incorrect because HCTL is not about to be exceeded. With pressure between 600 to 800 psig, HCTL limit is about 120°F. Temperature approaching 110°F does require entry into EOP-2 and a manual scram.

Distractor: D is incorrect because Core Spray NPSH limits are not being exceeded and a blowdown is not required based on NPSH.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 63 Details

Question Type:	Multiple Choice
Topic:	NRC RO 63
System ID:	21288
User ID:	NRC 2006 RO 63
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 30, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE** Enabling Objective: O1-OPS-006-344-1-04 EO-1.3

Question 63 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

OTHER_REFS

- NER 1M-095, Rev. NA

NUREG 1123 KA Catalog Rev. 2

- 295030 EK3.01 3.8/4.1 Knowledge of the reasons for the following responses as they apply to LOW SUPPRESSION POOL WATER LEVEL: Emergency depressurization

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 1 Memory/Fundamental Knowledge

10CFR55

- 41(b)(5)

Question Source

- New

64 **USERID: NRC 2006 RO 64** **SYSID: 21289** **Points: 1.00**

A reactor scram has just occurred with the following:

- Drywell Pressure is 4.0 psig and steady
- Torus water level is 11.0 feet and steady
- ALL Condensate Pumps have tripped off and cannot be started
- ALL Core Spray Pumps have started
- Reactor water level is -85 inches and slowly lowering
- Reactor pressure is being maintained 800-1000 psig
- Appropriate EOP actions are being implemented

Which one of the following is required to be performed?

A. Wait until water level drops to -109 inches, then enter RPV Blowdown.

B. Wait until water level drops to -121 inches, then enter RPV Blowdown.

C. Initiate ECs and confirm automatic opening of three ERVs at F Panel.

D. Initiate ECs and manually open three ERVs now using F Panel switches.

Answer: D

Answer Explanation: D is correct. Core Spray pumps are running but not yet injecting because RPV pressure is above 365 psig. Per EOP-2 steps L-10 through L-16, EOP-8 is entered based on current level and the ERVs are opened now, based on current water level. This is consistent with TMG step 1.2.2.5 (Ops Manual Tab 7)

A is incorrect. L-13 directs blowdown BEFORE -109 inches. This is NOT a WAIT block.

B is incorrect. -121 inches is the level at which a blowdown is performed if implementing Steam Cooling EOP, which is not the case, since Core Spray is running.

C is incorrect because if the ADS Timers had initiated, ADS Bypass manual action is accomplished previously in step L-3. Step L-5 also directs bypass, regardless of the status of ADS timer initiation.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 64 Details

Question Type:	Multiple Choice
Topic:	NRC RO 64
System ID:	21289
User ID:	NRC 2006 RO 64
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 30, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: EOP 2 & 8 flow charts** Enabling Objective: N1-218000 RBO-12

Question 64 Cross References (table item links)

MANUAL

- OPERATIONS MANUAL Rev. NA

OTHER_REFS

- NER 1M-095, Rev. NA

PROC

- N1-EOP-2 Rev. NA
- N1-EOP-8 Rev. NA

NUREG 1123 KA Catalog Rev. 2

- 295031 EA1.07 3.7*/3.7* Ability to operate and/or monitor the following as they apply to REACTOR LOW WATER LEVEL: Safety/relief valves

Level of Difficulty

- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level

- 2 Higher Cognitive Level comprehensive

10CFR55

- 41(b)(7)

Question Source

- New

65 **USERID: NRC 2006** **SYSID: 21290** **Points: 1.00**

RO 65

The plant is experiencing a transient with the following:

- MSIVs are closed on high steam flow
- APRM power is oscillating between 30% and 40%
- ERVs are cycling
- Both Emergency Condensers are in service
- An RPV pressure band has not yet been directed
- CRS orders manual operation of the ERVs to stop the valves from cycling

Which one of the following describes a reason for taking immediate manual control of the ERVs to stop them from cycling?

A. Prevent automatic operation to stabilize pressure and suppress the power transients.

B. Prevent turbine bypass valves from closing to minimize the heat rejected to the torus.

C. Allows for equalizing torus water heatup by using a specific valve opening sequence.

D. Allows for pressure stabilization so that one emergency condenser can be secured.

Answer: A

Answer Explanation: A is correct - Per EOP Bases, page 139, ERV cycling is undesirable because it exerts dynamic loading on components, complicates level control, increases risk of a stuck open ERV and **the consequent level and pressure oscillations can result in significant power transients.** EOP-3 pressure leg steps P-1 and P-2 direct initiation of EC and manually opening ERVs to drop pressure to 965 psig. With MSIVs closed the idea is to reduce pressure to stay below the ERV lift setpoints, which stops the ERVs from cycling.
B is incorrect - This would be the reason for choosing the 965 psig value IF the MSIVs were open. Given that the MSIVs are closed the 100% steam flow turbine bypass capacity is irrelevant. This distractor is plausible because it is a common misconception that the reason is the same, regardless of MSIV position. This is based on past candidate performance for similar questions.
C is incorrect – When ERVs are cycling and before a pressure band is assigned, the immediate action to stop the ERVS from cycling (P-2) does not restrict operator from using any specific sequence to equalize torus water temperature. Later, in step P-4, a specific sequence is provided, to equalize temperatures.
D. is incorrect – Step P-2 directs EC operation to be initiated. With power at 30-40%, operation of BOTH EC is appropriate as pressure control mechanisms. Removing an EC from operation with these conditions would result in more heat and energy rejection through the ERVs, which would cause more ERV cycling.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

EXAMINATION ANSWER KEY

Reactor Operator Written Examination Nine Mile Point Unit 1 2007

Question 65 Details

Question Type:	Multiple Choice
Topic:	NRC RO 65
System ID:	21290
User ID:	NRC 2006 RO 65
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Dec 19, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Technical Reference(s): N1-EOP-3 EOP Bases NER-1M-095 **Reference Provided: N1-EOP-3 Flowchart** Enabling Objective: O1-OPS-006-344-1-03 EO-1.3

Question 65 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

PROC

- N1-EOP-3 Rev. NA

NUREG 1123 KA Catalog Rev. 2

- 295037 EK1.01 4.1*/4.3* Knowledge of the operational implications of the following concepts as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Reactor pressure effects on reactor power

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 1 Memory/Fundamental Knowledge

10CFR55

- 41(b)(8)

Question Source

- New

66 **USERID: NRC 2006** **SYSID: 21291** **Points: 1.00**
RO 66

The plant is operating at 100% power when a valid scram signal is received with the following:

- RPS remains energized
- All rods fail to insert
- Scram fuses are now pulled
- Scram Pilot Solenoid Valve lights are deenergized
- Scram air header pressure is 0 psig
- All rods remain at their same positions

Without re-installing the RPS fuses, which one of the following methods would insert the rods?

A. Driving rods using RMCS.

B. Venting the over-piston areas.

C. Initiating repeated manual scrams.

D. Operating individual rod scram switches.

Answer: B

Answer Explanation: EO-1.2

B is correct – this is the only available method to insert the hydraulically locked rods without re-pressurizing the scram air header. **A/C/D** incorrect – all require that fuses be reinstalled to re-pressurize.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

EXAMINATION ANSWER KEY

Reactor Operator Written Examination Nine Mile Point Unit 1 2007

Question 66 Details

Question Type:	Multiple Choice
Topic:	NRC RO 66
System ID:	21291
User ID:	NRC 2006 RO 66
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 30, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE** Enabling Objective: O1-OPS-006-344-1-11 EO-1.2

Question 66 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

PROC
- N1-EOP-3.1 Rev. NA

NUREG 1123 KA Catalog Rev. 2
- 295037 EK3.07 4.2/4.3* Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Various alternate methods of control rod insertion: Plant-Specific

Level of Difficulty
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 41(b)(5)

Question Source
- Bank

67 **USERID: NRC 2006 RO 67** **SYSID: 21292** **Points: 1.00**

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

- A plant transient is in progress
- STACK GAS MONITORS HIGH RADIATION is in alarm.
- A report has been received that the turbine building ventilation is shutdown
- SM declares an ALERT condition based on off-site release rates

Which one of the following operator actions should be performed, if any, and why?

A. Leave the turbine building ventilation system shutdown because no procedural guidance is provided.

B. Leave the turbine building ventilation system shutdown to minimize the radiological release from the turbine building.

C. Restart the turbine building ventilation system to prevent transferring air between the reactor building and the turbine building.

D. Restart the turbine building ventilation system to direct any radioactivity release through an elevated, monitored path.

Answer: D

Answer Explanation: D. is correct because per the basis document for N1-EOP-6 the turbine building ventilation is restarted to prevent an unmonitored ground release. N1-EOP-6 is entered when the shift manager declares an ALERT condition based on off-site release rates.

A. is incorrect because procedural guidance is given in N1-EOP-6 to start turbine building ventilation. N1-EOP-6 is entered when the shift manager declares an ALERT condition based on off-site release rates.

B. is incorrect because the turbine building is not air tight, not restarting the turbine would not only restrict personnel access, but result in an unmonitored, ground release of radioactivity.

C. is incorrect because starting the turbine building ventilation would not prevent air from transferring to the reactor building.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 67 Details

Question Type:	Multiple Choice
Topic:	NRC RO 67
System ID:	21292
User ID:	NRC 2006 RO 67
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Dec 19, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE** Enabling Objective: O1-OPS-006-344-1-06 EO-1.3

Question 67 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

OTHER_REFS
- (BOOK) NMP1 EOP/SAP BASIS DOCUMENT NER-1M-095, Rev, NA

PROC
- N1-EOP-6 Rev. NA

NUREG 1123 KA Catalog Rev. 2
- 295038 EA1.06 3.5/3.6 Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RELEASE RATE: Plant ventilation

Level of Difficulty
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 41(b)(7)

Question Source
- New

68 **USERID: NRC 2006 RO 68** **SYSID: 21293** **Points: 1.00**

The Fire Detection System senses a fire in Hazard C-2123, Power Board Room 102. The following Alarm Detection Zones are received at the Main Fire Control Panel:

- DX-2123A
- DX-2123B
- All automatic fire suppression systems on MFP-2 are in automatic

Which one of the following describes the response of the Fire Protection system?

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

B. Deluge system actuated and the motor-driven fire pump is running.

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

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

Answer: C

Answer Explanation: C. is the correct answer. PB102 is protected by CO2 and operation is correctly described in this choice.

A. is incorrect this area has CO2 fire suppression. Foam suppression systems are for the Main Turbine island and lube oil areas.

B. is incorrect this area has CO2 fire suppression. Water suppression system are throughout the plant but not in electrical areas.

D. is incorrect this area has CO2 fire suppression. Halon suppression systems are in electronic equipment areas such as the auxiliary control room, EC isolation valve room, Security Alarm Station, Security CPU/Equipment Room, RSSB Control Room, and RSSB Electrical Equipment Room.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 68 Details

Question Type:	Multiple Choice
Topic:	NRC RO 68
System ID:	21293
User ID:	NRC 2006 RO 68
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Jan 25, 2007
Cross Reference Number:	LC1 05-01
Num Field 1:	26475
Num Field 2:	
Text Field:	
Comments:	N1-OP-21C, Section b **Reference Provided: NONE** Enabling Objective:O1-OPS-001-286-1-03 EO-1.2

Question 68 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

PROC
- N1-OP-21C Rev. NA

NUREG 1123 KA Catalog Rev. 2
- 600000 AK2.01 2.6/2.7 Knowledge of the interrelations between PLANT FIRE ON SITE and the following: Sensors, detectors and valves

Level of Difficulty
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level
- 1 Memory/Fundamental Knowledge

10CFR55
- 41(b)(5)

Question Source
- Bank

69 **USERID: NRC 2006 RO 69** **SYSID: 21294** **Points: 1.00**

The plant is operating at 100% power with the following:

- A reactor scram occurs
- RPV Water Level lowers to 30 inches before recovering
- NOW RPV water level is 85 inches and rising slowly

Which one of the following actions is now required, per N1-SOP-1, Reactor Scram and what is the basis for the action?

A. Close EC steam isolation valves to prevent flooding EC steam lines.

B. Close HPCI level control valves to prevent flooding main steam lines.

C. Trip both motor driven feed pumps to prevent flooding EC steam lines.

D. Reset HPCI logic with pump switches to prevent flooding main steam lines.

Answer: C

Answer Explanation: C. is correct, per N1-SOP-1. Override step for level control directs verifying feed pumps are off, if level is 85 inches and rising. The discussion section 5.6 provides the reason for tripping the pumps. This is to maintain the heat removal capabilities of the Emergency Condensers, which have their steam lines at about 95 inches RPV level.
A. is incorrect because SOP-1 does not require the ECs to be isolated on a high level, even though flooding the EC steam line is the reason for taking action to control level below 85 inches.
B. and D are incorrect because the EC steam lines are lower than the MSLs. Action taken is to maintain heat removal capability from ECs. D. includes action in the SOP to restart tripped feed pumps by resetting HPCI with pump control switches.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

EXAMINATION ANSWER KEY

Reactor Operator Written Examination Nine Mile Point Unit 1 2007

Question 69 Details

Question Type:	Multiple Choice
Topic:	NRC RO 69
System ID:	21294
User ID:	NRC 2006 RO 69
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 30, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Technical References: N1-SOP-1 Reactor Scram **Reference Provided: NONE** Enabling Objective: O1-OPS-001-259-1-02 EO-1.7

Question 69 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

PROC

- N1-SOP-1 Rev. NA

NUREG 1123 KA Catalog Rev. 2

- 295008 AK3.04 3.3/3.5 Knowledge of thE reasons for the following responses as they apply to HIGH REACTOR WATER LEVEL: Reactor feed pump trip: Plant-Specific

Level of Difficulty

- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level

- 2 Higher Cognitive Level comprehensive

10CFR55

- 41(b)(5)

Question Source

- New

EXAMINATION ANSWER KEY

Reactor Operator Written Examination Nine Mile Point Unit 1 2007

70 **USERID: NRC 2006 RO 70** **SYSID: 21295** **Points: 1.00**

The plant is operating at 100% power with the following:

- All Drywell Coolers are in service
- Drywell pressure is 1.7 psig and steady
- Drywell average temperature is 130°F and steady
- THEN Drywell Cooler 12 trips and cannot be restarted

Which one of the following is the affect on Drywell parameters?

A. Pressure rises and reaches an EOP entry condition.

B. Temperature rises and reaches an EOP entry condition.

C. Temperature rises and stabilizes below EOP entry condition.

D. Temperature and pressure remain at their initial pre-trip values.

Answer: C

Answer Explanation: C is correct. Six drywell coolers can maintain drywell temperature below 135°F. Five drywell coolers can maintain drywell temperature below 150°F. On the loss of one cooler, expected temperature rise is about 5°F. Since initial temperature is 130°F, temperature will rise, but remain below the EOP entry condition of 150°F.

A. is incorrect because pressure will not rise to the EOP entry condition of 3.5 psig. Pressure may rise slightly from the initial value of 1.7 psig

B. is incorrect temperature will rise but stabilize below the EOP entry condition of 150°F.

D. is incorrect because temperature and pressure will rise and not stay at the pre-trip value.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 70 Details

Question Type:	Multiple Choice
Topic:	NRC RO 70
System ID:	21295
User ID:	NRC 2006 RO 70
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 30, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Technical References: N1-OP-8, Section B **Reference Provided: NONE** Enabling Objective: O1-OPS-001-223-1-02 EO-1.3

Question 70 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

PROC

- N1-OP-8, Rev. NA

NUREG 1123 KA Catalog Rev. 2

- 295012 AK2.02 3.6/3.7 Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: Drywell cooling

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 2 Higher Cognitive Level comprehensive

10CFR55

- 41(b)(7)

Question Source

- New

71 **USERID: NRC 2006** **SYSID: 21296** **Points: 1.00**
RO 71

The plant is being started up following an outage with the following:

- Torus water temperature is 80°F
- CRS orders Torus Cooling placed in service to lower temperature
- Torus Cooling is being placed in service per N1-OP-14, Containment Spray System

Which one of the following local actions must be performed before starting a Containment Spray Pump and why?

A. Associated Containment Spray IV stem stop nut gap must be verified to be sufficient to ensure the valve is closed to prevent inadvertent containment spray.

B. Associated Containment Spray Bypass stem stop nut gap must be verified to be sufficient to ensure the valve is closed to prevent inadvertent containment spray.

C. Associated Containment Spray Raw Water Pump discharge valve must be closed then throttled open six turns to prevent pump runout, when the raw water pump is started.

D. Associated Containment Spray Raw Water system inter-tie valves must be locally verified to be in the correct position to prevent inadvertent injection of raw water to the torus.

Answer: A

Answer Explanation: A is correct. N1-OP-14 requires the stop nut gap verified, so that the Containment Spray IV can be assured to be closed. If stem nut gap is insufficient, the valve may not be fully closed and when the pump is started, inadvertent spray can result (N1-OP-14 Precaution and Limitation D.12.0).
B. is incorrect because it identifies the incorrect valve.
C. is incorrect because throttling the discharge is not performed when implementing this procedure, but it is a local action performed when lining up (cross-tie) raw water to the Containment Spray system.
D. is incorrect because local verification of inter-tie valve position is not required when starting torus cooling.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 71 Details

Question Type:	Multiple Choice
Topic:	NRC RO 71
System ID:	21296
User ID:	NRC 2006 RO 71
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 16, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Technical Reference(s): N1-226001 N1-OP-14 N1-226001 RBO 9 **Reference Provided: NONE**

Question 71 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

PROC

- N1-OP-14 Rev. NA

NUREG 1123 KA Catalog Rev. 2

- G2.1.30 3.9/3.4 Ability to locate and operate components, including local controls
- 295013 High Suppression Pool Temperature

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 1 Memory/Fundamental Knowledge

10CFR55

- 41(b)(7)

Question Source

- New

72 **USERID: NRC 2006 RO 72** **SYSID: 21297** **Points: 1.00**

The plant is in a startup after a two day forced outage, with the following initial conditions:

- The reactor has just been made critical on rod 30-31 at 06
- Period is constant at 200 seconds with no rod motion
- All SRMs are reading about 8 X 10^4 cps and rising
- All IRMs are reading about 25 on Range 2
- NO rod blocks exist

THEN, the following occurs:

- Annunciator F3-2-6, CONTROL ROD DRIFT, alarms
- Control rod 30-31 is observed to be at 08, 10, then 12 and continues to move
- SRM PERIOD is observed at 20 seconds
- IRMs remain on Range 2

Which one of the following identifies how the Neutron Monitoring System (NMS) responds to this reactivity change, after one minute?

A. No NMS scram occurred but the SRMs initiated a rod withdrawal block.

B. No NMS scram occurred but the IRMs initiated a rod withdrawal block.

C. Automatic scram occurred when counts reached SRM high-high setpoint.

D. Automatic scram occurred when readings reached IRM high-high setpoint.

Answer: D

Answer Explanation: D. is correct. Period = 1.44*Doubling Time, so if period is 20 seconds the DT is about 14 seconds. IRMs are initially reading 25, so just over two doubling (30 seconds) will result in IRM HI-HI scram trip at 120/125 (96% of scale).

A. is incorrect. The SRMs will generate a rod block at 1 X 10^5 cps, but after a minute a scram has also occurred due to reaching the Hi Hi IRM scram setpoint.

B. is incorrect. An IRM rod block will be generated but after one minute an IRM scram will also occurred due to reaching the Hi Hi IRM scram setpoint.

C. is incorrect. The SRM scram is bypassed in COINCIDENT mode. An SRM scram (>5 X 10^5 cps) would have occurred after 3 doubling times (1minute), if SRMs were in NON-COINCIDENT. The NON COINCIDENT mode of operation is only used during a core reload and verification.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 72 Details

Question Type:	Multiple Choice
Topic:	NRC RO 72
System ID:	21297
User ID:	NRC 2006 RO 72
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Jan 17, 2007
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Technical References: N1-215000 **Reference Provided: NONE** Enabling Objective: N1-215000 RBO 8

Question 72 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2

- 295014 AA1.05 3.9/3.9 Ability to operate and / or monitor the following as they apply to INADVERTENT REACTIVITY ADDITION: Neutron monitoring system

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 2 Higher Cognitive Level comprehensive

10CFR55

- 41(b)(7)

Question Source

- New

73 **USERID: NRC 2006 RO 73** **SYSID: 21298** **Points: 1.00**

The plant is experiencing an ATWS with the following:

- Reactor power is 15%
- RPV Water Level is 70 inches
- Torus water temperature is 80°F and rising
- ERVs are cycling
- CRS orders injection to be terminated and prevented

Which one of the following is the reason for lowering water level per the EOP Basis Document?

A. Suppress power oscillations by reducing inlet subcooling.

B. Suppress power oscillations by reducing natural circulation flow.

C. Minimize heat rejection to the torus by reducing inlet subcooling.

D. Minimize heat rejection to the torus by reducing natural circulation flow.

Answer: A

Answer Explanation: A. is correct, per EOP Basis Document NER-1M-095 page 117. If power is above 6% and level is above -41 inches, level is lowered to at least -41 inches to uncover the feedwater spargers. This raises feedwater temperature entering the core (reduced subcooling) to prevent thermal hydraulic instabilities.
B. is incorrect because lowering level to -41 inches is not performed to reduce natural circulation core flow. Reducing core flow is the concept behind lowering level to -109 and -84, to reduce power, to minimize heat rejection to the torus.
C. and D. are incorrect because this is performed with torus temperature above 110°F to lower power to protect the containment.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 73 Details

Question Type:	Multiple Choice
Topic:	NRC RO 73
System ID:	21298
User ID:	NRC 2006 RO 73
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 30, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Technical References: NER-1M-095 NMP 1 EOP Basis Document **Reference Provided: NONE** Enabling Objective: O1-OPS-006-344-1-03 EO-1.3

Question 73 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2

- 295015 AK1.03 3.8/3.9 Knowledge of the operational implications of the following concepts as they apply to INCOMPLETE SCRAM: Reactivity effects

Level of Difficulty

- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level

- 1 Memory/Fundamental Knowledge

10CFR55

- 41(b)(8)

Question Source

- New

74 **USERID: NRC 2006** **SYSID: 21299** **Points: 1.00**
RO 74

If Torus water level can NOT be restored and maintained below 13.5 feet, N1-EOP-4 directs a reactor blowdown.

Which one of the following is prevented by performing this reactor blowdown?

A. Loss of Torus air space venting capability.

B. Submerging Torus to Drywell vacuum breakers.

C. Containment damage due to high ERV clearing load.

D. Loss of pressure suppression capability of the containment.

Answer: D

Answer Explanation: D is correct because the basis document NER-1M-095 states that 13.5 feet is the highest primary containment water level at which the pressure suppression capability of the containment can be maintained.

A. is incorrect because the ability to vent the Torus is lost at 27 feet.

B. is incorrect because the Torus to Drywell vacuum breakers will not get submerged until ~27 feet.

C. is incorrect because there is no ERV clearing load limit because it is not a limiting condition for the Mark I containment.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 74 Details

Question Type:	Multiple Choice
Topic:	NRC RO 74
System ID:	21299
User ID:	NRC 2006 RO 74
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 30, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: NONE** Enabling Objective: O1-OPS-006-344-1-04 EO-1.3

Question 74 Cross References (table item links)

<u>Question Setting</u>
- C1 (License class closed reference)

<u>OTHER_REFS</u>
- (BOOK) NMP1 EOP/SAP BASIS DOCUMENT NER-1M-095, Rev, NA

<u>NUREG 1123 KA Catalog Rev. 2</u>
- 295029 EK3.01 3.5/3.9* Knowledge of the reasons for the following responses as they apply to HIGH SUPPRESSION POOL WATER LEVEL: Emergency depressurization

<u>Level of Difficulty</u>
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

<u>Cognitive Level</u>
- 1 Memory/Fundamental Knowledge

<u>10CFR55</u>
- 41(b)(5)

<u>Question Source</u>
- New

75 **USERID: NRC 2006 RO 75** **SYSID: 21300** **Points: 1.00**

The plant is operating at 100% power with the following:

- RWCU Heat Exchanger room is 200°F and steady
- RWCU leak is unisolable
- 8.2×10^3 mR/hr is reported on RB 261 East
- 9.2×10^3 mR/hr is reported on RB 261 West

Which one of the following describes the proper implementation of EOP-5 with regards to area radiation levels?

A. Only one area is affected by elevated radiation levels. Continue to try to isolate the leak. A normal plant shutdown is NOT required.

B. Two areas are affected by elevated radiation levels. A normal plant shutdown IS required. RPV Blowdown is NOT required.

C. Only one area is affected by elevated radiation levels. A manual scram IS required but an RPV Blowdown is NOT required.

D. Two areas are affected by elevated radiation levels. A manual scram IS required AND an RPV Blowdown IS required.

Answer: D

Answer Explanation: D. is correct. The west and east side of RB 261 are separate areas. If two areas are reading above Max Safe Values and the discharge cannot be isolated, then a reactor scram and Blowdown is required by EOP-5.

A is incorrect. Two areas are affected, not one.

B is incorrect. These actions are directed from SC-8 step. IF a primary system were not discharging into the area, this would be correct. Based on temperatures AND radiation levels, a primary system IS discharging into the Reactor Building, so SC-12 applies.

C is incorrect. A scram is required and so is an RPV Blowdown with 2 areas affected. If reactor building 261 East and reactor building 261 West were treated as the same area because they are on the same floor of the Reactor Building, this would be correct.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

EXAMINATION ANSWER KEY

Reactor Operator Written Examination Nine Mile Point Unit 1 2007

Question 75 Details

Question Type:	Multiple Choice
Topic:	NRC RO 75
System ID:	21300
User ID:	NRC 2006 RO 75
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Dec 19, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: EOP-5 flowchart.** Enabling Objective: O1-OPS-006-344-1-05 EO-1.2

Question 75 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2
- 295033 EA2.01 3.8/3.9 Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS: Area radiation levels

Level of Difficulty
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 41(b)(10)

Question Source
- New

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TO USE SUBJECTIVE SCORE FEATURE:
- Mark total possible subjective points
- Only one mark per line on key
- 163 points maximum

EXAMPLE OF STUDENT SCORE:

NAME	
SUBJECT	TEST NO.
DATE	HOUR

TEST RECORD

PART 1	
PART 2	
TOTAL	

PART 1

(T) (F) KEY

Question	Marked answer
1	A
2	B
3	A
4	C
5	D
6	B
7	A
8	C
9	D
10	B
11	C
12	A
13	A
14	D
15	B
16	C
17	B
18	A
19	D
20	C
21	D
22	B
23	D
24	B
25	C

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ES-401 **Site-Specific SRO Written Examination Cover Sheet** Form ES-401-8

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

Applicant Information

Name:	
Date: March 16, 2007	Facility/Unit: Nine Mile Point / Unit 1
Region: I	Reactor Type: GE
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination you must achieve a final grade of at least 80.00 percent overall, with 70.00 percent or better on the SRO-only items if given in conjunction with the RO exam; SRO-only exams given alone require a final grade of 80.00 percent to pass. You have 8 hours to complete the combined examination, and 3 hours if you are only taking the SRO portion.

Applicant Certification

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

____
Applicant's Signature

Results

RO/SRO-Only/Total Examination Values	75 / 25 / 100	Points
Applicant's Score	___ / ___ / ___	Points
Applicant's Grade	___ / ___ / ___	Percent

SUBJECTIVE SCORE INSTRUCTOR USE ONLY

100	90	80	70	60
50	40	30	20	10
9	8	7	6	5
4	3	2	1	0
(T)	(F)			KEY
%	2	3		5

IMPORTANT

- USE NO. 2 PENCIL ONLY
- MAKE DARK MARKS
- ERASE COMPLETELY TO CHANGE
- EXAMPLE: A B D E

TO USE SUBJECTIVE SCORE FEATURE:
- Mark total possible subjective points
- Only one mark per line on key
- 153 points maximum

EXAMPLE OF STUDENT SCORE:

PART 2

NAME			
SUBJECT		TEST NO.	
DATE		HOUR	

TEST RECORD	
PART 1	
PART 2	
TOTAL	

51	A	B	C	D	E
52	A	B	C	D	E
53	A	B	C	D	E
54	A	B	C	D	E
55	A	B	C	D	E
56	A	B	C	D	E
57	A	B	C	D	E
58	A	B	C	D	E
59	A	B	C	D	E
60	A	B	C	D	E
61	A	B	C	D	E
62	A	B	C	D	E
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64	A	B	C	D	E
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96	A	B	C	D	E
97	A	B	C	D	E
98	A	B	C	D	E
99	A	B	C	D	E
100	A	B	C	D	E

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EXAMINATION ANSWER KEY
(ANSWERS ONLY)

NRC Senior Reactor Operator Written Examination Nine Mile Point Unit 1 2007

1		**ID: NRC 2006 SRO 1**	**Points: 1.00**
	Answer: A		
2		**ID: NRC 2006 SRO 2**	**Points: 1.00**
	Answer: B		
3		**ID: NRC 2006 SRO 3**	**Points: 1.00**
	Answer: A		
4		**ID: NRC 2006 SRO 4**	**Points: 1.00**
	Answer: C		
5		**ID: NRC 2006 SRO 5**	**Points: 1.00**
	Answer: D		
6		**ID: NRC 2006 SRO 6**	**Points: 1.00**
	Answer: B		
7		**ID: NRC 2006 SRO 7**	**Points: 1.00**
	Answer: A		
8		**ID: NRC 2006 SRO 8**	**Points: 1.00**
	Answer: C		
9		**ID: NRC 2006 SRO 9**	**Points: 1.00**
	Answer: D		
10		**ID: NRC 2006 SRO 10**	**Points: 1.00**
	Answer: B		

EXAMINATION ANSWER KEY
(ANSWERS ONLY)

NRC Senior Reactor Operator Written Examination Nine Mile Point Unit 1 2007

11		**ID: NRC 2006 SRO 11**	**Points: 1.00**
	Answer: C		
12		**ID: NRC 2006 SRO 12**	**Points: 1.00**
	Answer: A		
13		**ID: NRC 2006 SRO 13**	**Points: 1.00**
	Answer: A		
14		**ID: NRC 2006 SRO 14**	**Points: 1.00**
	Answer: D		
15		**ID: NRC 2006 SRO 15**	**Points: 1.00**
	Answer: B		
16		**ID: NRC 2006 SRO 16**	**Points: 1.00**
	Answer: C		
17		**ID: NRC 2006 SRO 17**	**Points: 1.00**
	Answer: B		
18		**ID: NRC 2006 SRO 18**	**Points: 1.00**
	Answer: A		
19		**ID: NRC 2006 SRO 19**	**Points: 1.00**
	Answer: D		
20		**ID: NRC 2006 SRO 20**	**Points: 1.00**
	Answer: C		

EXAMINATION ANSWER KEY
(ANSWERS ONLY)

NRC Senior Reactor Operator Written Examination Nine Mile Point Unit 1 2007

21 **ID: NRC 2006 SRO 21** **Points: 1.00**

Answer: D

22 **ID: NRC 2006 SRO 22** **Points: 1.00**

Answer: B

23 **ID: NRC 2006 SRO 23** **Points: 1.00**

Answer: D

24 **ID: NRC 2006 SRO 24** **Points: 1.00**

Answer: B

25 **ID: NRC 2006 SRO 25** **Points: 1.00**

Answer: C

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

Applicant Information

Name:	
Date: March 16, 2007	Facility/Unit: Nine Mile Point / Unit 1
Region: I	Reactor Type: GE
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination you must achieve a final grade of at least 80.00 percent overall, with 70.00 percent or better on the SRO-only items if given in conjunction with the RO exam; SRO-only exams given alone require a final grade of 80.00 percent to pass. You have 8 hours to complete the combined examination, and 3 hours if you are only taking the SRO portion.

Applicant Certification

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

____
Applicant's Signature

Results

RO/SRO-Only/Total Examination Values	75 / 25 / 100	Points
Applicant's Score	___ / ___ / ___	Points
Applicant's Grade	___ / ___ / ___	Percent

EXAMINATION ANSWER KEY

Senior Reactor Operator Written Examination Nine Mile Point Unit 1 2006

1 **USERID: NRC 2006 SRO 1** **SYSID: 21301** **Points: 1.00**

The plant is operating at 100% power with the following:

Time	Event
t = 0900	A8-2-2, TRANS. 101N SWITCH 168 OPEN, is in alarm
t = 0900	Diesel Generator 102 starts and powers PB 102
t = 0905	A fire is reported in the cable spreading room
t = 0910	H2-1-6, SEISMIC DETECTION EQUIPMENT EVENT is in alarm
t = 0911	Earthquake felt by all Control Room personnel
t = 0918	Cable spreading room fire is reported extinguished
t = 0920	It is reported that the fire damage is limited to the Reactor Building Ventilation System

Which one of the following describes when the SM has to assume the role of the Emergency Director (ED)?

A. 0911 because UNUSUAL EVENT condition now exists.

B. 0911 because ALERT condition now exists.

C. 0916 because UNUSUAL EVENT condition now exists.

D. 0920 because ALERT condition now exists.

Answer: A

Answer Explanation: A. is correct. EAL 8.4.1 is an UNUSUAL EVENT for an earthquake felt in plant based upon a consensus of Control Room Operators on duty and either NMP-1 seismic instrumentation ACTUATED or CONFIRMATION of earthquake received on NMP-2 or JAFNPP seismic instrumentation.

B. is incorrect. In order for a seismic event to be classified as an ALERT it must have that seismic instrumentation indicated >0.11 g (EAL. 8.4.4) or damage to equipment required to ESTABLISH or MAINTAIN SAFE PLANT SHUTDOWN (8.4.6). This information is not given making it incorrect to assume an ALERT condition.

C. is incorrect. EAL. 6.1.1 is a loss of all offsite AC power, which requires both T101N and T101S to have a loss of power for >15 min. In this case only T101N has had a loss of power for >15 min.

D. is incorrect. EAL. 8.2.2 is fire or explosion in any plant area, which results in damage to plant equipment or structures required to ESTABLISH or MAINTAIN SAFE PLANT SHUTDOWN, Table 5 or Table 6. The Reactor Building Ventilation System is not required to ESTABLISH or MAINTAIN SAFE PLANT SHUTDOWN so declaring an ALERT condition would be incorrect.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 1 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 1
System ID:	21301
User ID:	NRC 2006 SRO 1
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Jan 08, 2007
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Reference Provided: EAL MATRIX, EPIP-EPP-01 Enabling Objective: O3-OPS-006-350-3-21 EO-1.3

Question 1 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

OTHER_REFS

- EAL Matrix, Rev. NA

NUREG 1123 KA Catalog Rev. 2

- G2.1.6 2.1/4.3 Ability to supervise and assume a management role during plant transients and upset conditions

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 2 Higher Cognitive Level comprehensive

10CFR55

- 43(b)(5)

Question Source

- New

2 **USERID: NRC 2006 SRO 2** **SYSID: 21302** **Points: 1.00**

The plant is experiencing a transient with the following:

- N1-EOP-4 has been entered
- Torus Pressure is 20 psig
- Torus level is 11 feet
- The CRS has directed Containment Spray per N1-EOP-1 Attachment 17
- Once Containment Spray is in service Torus pressure immediately drops to 17 psig

Which one of the following describes when Containment Spray is considered to be in service and what is the next required action following placing containment spray in service?

A. When two Containment Spray Pumps are started and up to rated flow. An evaluation of Torus pressure in relation to the Pressure Suppression Pressure curve must be performed.

B. When the first Containment Spray Pump is started and up to rated flow. An evaluation of Torus pressure in relation to the Pressure Suppression Pressure curve must be performed.

C. When two Containment Spray Pumps are started and up to rated flow. An RPV BLOWDOWN must be performed.

D. When the first Containment Spray pump is started and up to rated flow. An RPV BLOWDOWN must be performed.

Answer: B

Answer Explanation: B. is correct. When the Suppression Chamber pressure is greater that the Pressure Suppression Pressure (PSP) upon entry into the Primary Containment Control EOP, all steps up to evaluating proximity to PSP are to be implemented prior to evaluating Torus Pressure against the PSP curve. Containment sprays are considered to be "in service" when one train of Containment Spray is initiated (The other additional loop of Containment Spray that is started is for Appendix J Water Seal requirements). Evaluation of the Torus pressure in relation to the Pressure Suppression Pressure (PSP) curve is expected to occur once Containment Sprays are "in service."

A. is incorrect. Containment Spray should be considered to be in service after the first pump has been started and is up to rated flow.

C. is incorrect. Containment Spray should be considered to be in service after the first pump has been started and is up to rated flow. An RPV Blowdown will only be directed after evaluating Torus pressure in relation to the Pressure Suppression Pressure.

D. is incorrect. An RPV Blowdown will only be directed after evaluating Torus pressure in relation to the Pressure Suppression Pressure.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 2 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 2
System ID:	21302
User ID:	NRC 2006 SRO 2
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 27, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Reference Provided: N1-EOP-4 Enabling Objective: O1-OPS-006-344-1-04 EO-1.2

Question 2 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2

- G2.1.7 3.7/4.4 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 2 Higher Cognitive Level comprehensive

10CFR55

- 41(b)(10)
- 43(b)(5)

Question Source

- New

3 **USERID: NRC 2006 SRO 3** **SYSID: 21303** **Points: 1.00**

The plant is experiencing a transient with the following:

t = 0 sec	RPV pressure decreases to 750 psig
t = 1 sec	Reactor power decreases to 50%
t = 2 sec	MSIVs begin closing
t = 5 sec	MSIVs are fully closed
t = 8 sec	RPV pressure rises to 1360 psig
t = 14 sec	RPV pressure drops to 1100 psig due to ERV and SRV operation

Which one of the following Safety Limits has been exceeded and what is the basis behind the limit?

A. MCPR. Prevents fuel damage as a result of an abnormal operational transient.

B. MCPR. Prevents heat generation rates following a LOCA from exceeding design limits.

C. RPV pressure. Ensures that a pressure transient does not damage the fuel cladding.

D. RPV pressure. Ensures that the integrity of the Reactor pressure vessel is maintained.

Answer: A

Answer Explanation: A. is correct. When reactor pressure is <800 psig or Core Flow is <10% reactor power must remain below 25%. T.S. States that the fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient.

B. is incorrect. This describes the basis behind APLHGR.

C & D are incorrect. The safety limit for reactor pressure has not been exceeded (1375 psig)

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 3 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 3
System ID:	21303
User ID:	NRC 2006 SRO 3
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Jan 08, 2007
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Reference Provided: NONE Enabling Objective: O1-OPS-008-362-1-03 EO-1.4

Question 3 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2

- G2.2.25 2.5/3.7 Knowledge of bases in technical specifications for limiting conditions for operations and safety limits

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 2 Higher Cognitive Level comprehensive

10CFR55

- 43(b)(2)

Question Source

- New

4 **USERID: NRC 2006 SRO 4** **SYSID: 21304** **Points: 1.00**

The plant is in a refueling outage with the following:

- A spiral reload is planned at the start of the next shift
- No fuel has been loaded yet
- SRMs 12 & 13 are reading 2 cps

Which one of the following requirements (actions) must be performed to permit the loading of fuel?

A. Special movable dunking type detectors will be used and connected into the normal SRM circuits 12 and 13.

B. Fuel is not allowed to be moved back into the core until the 3 cps requirement is met on both SRM channels 12 & 13.

C. Two fuel assemblies will be loaded in different cells which contain control blades around SRM channel 12 & 13 to obtain 3 cps.

D. SRM operability will be verified by using a portable external source every 24 hours until SRM channel 12 & 13 are reading 3 cps.

Answer: C

Answer Explanation: C. is correct. P&L 4.1.12 of N1-FHP-27B states prior to and during spiral reload, SRM operability will be verified by loading two fuel assemblies in different cells containing control blades around each SRM to obtain the required 3 cps. Until these two assemblies have been loaded, the 3 cps requirement is not necessary (T/S 3.5.3e). As an alternate, a neutron source may be used to verity SRM operability per T.S. 3.5.3.e.

A is incorrect. Dunking detectors do not meet T.S. 3.5.3e requirements.

B. is incorrect. T.S. 3.5.3.e states that fuel is allowed to be inserted with SRM <3 cps to load two fuel assemblies around each SRM.

D. is incorrect. T.S. 3.5.3.e states that if a Portable Source is used to verify operability that it is required to be done every 12 hours.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

EXAMINATION ANSWER KEY

Senior Reactor Operator Written Examination Nine Mile Point Unit 1 2006

Question 4 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 4
System ID:	21304
User ID:	NRC 2006 SRO 4
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 27, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	995
Num Field 2:	
Text Field:	
Comments:	Reference Provided: NONE Enabling Objective: O1-OPS-001-234-1-01 EO-1.11

Question 4 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

PROC
- N1-FHP-27B Rev. na

TECHSPEC
- 3.5.3.E Rev. na

NUREG 1123 KA Catalog Rev. 2
- G2.2.27 2.6/3.5 Knowledge of the refueling process

Level of Difficulty
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 43(b)(6)

Question Source
- Bank

5 **USERID: NRC 2006 SRO 5** **SYSID: 21305** **Points: 1.00**

The plant is operating at 95% with the following:

- A seismic event occurs
- A loss of coolant accident and a loss of off-site power occur
- All rods are at 00
- Drywell pressure is 14.5 psig
- 3 ERVs are open
- RPV pressure is 150 psig and lowering
- RPV water level is -90 inches
- Torus level is 12.5 ft
- Chemistry has sampled the drywell and expects release rate to stay within Tech Spec limits
- Drywell hydrogen concentration is 1.8%
- Drywell oxygen concentration is 2.5%

Which one of the following actions is required?

A. Line up to vent the Drywell through RB Emergency Ventilation.

B. Lineup to vent the Torus through RB Emergency Ventilation.

C. Perform an RPV Blowdown when level drops to -108 inches.

D. Line up power to the Drywell and Torus Vent and Purge Fan.

Answer: D

Answer Explanation: D. is correct. A drywell pressure of 3.5 is an entry requirement for N1-EOP-4. When the hydrogen leg is executed it states that if there is detectable hydrogen in the torus or the drywell then to enter N1-EOP-4.2, HYDROGEN CONTROL. With a hydrogen concentration of 1.8% and an oxygen concentration of 2.5% we are directed to section 31 of N1-EOP-4.2. Section 31 directs the operator to vent the containment per detail Z1 and purge per detail Z2. Detail Z1 directs the operator to vent the torus through the vent and purge fan. With a loss of PB 11 & 12 the drywell vent and purge fan has to be lined up to EDG 102 or EDG 103 per N1-EOP-1 ATT 20.

A. is incorrect. With Torus level at 12.5 feet detail Z1 directs venting through the torus.

B. is incorrect. The drywell pressure is at 14.5 psig. During post transient conditions torus pressure is within 3 psig of drywell pressure. So the lowest torus pressure could be is 11.5 psig. With torus pressure greater than 3 psig detail Z1 directs venting through the vent and purge fan.

C. is incorrect. Given that 3 ERVs are open and the RPV pressure is 150 psig and lowering is enough information to determine that a blowdown is already in progress.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 5 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 5
System ID:	21305
User ID:	NRC 2006 SRO 5
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Feb 20, 2007
Cross Reference Number:	LC1 05-01
Num Field 1:	2138
Num Field 2:	
Text Field:	
Comments:	Reference Provided: EOP-4 and EOP-4.2 flowcharts. Enabling Objective: O1-OPS-006-344-1-04 EO-1.2

Question 5 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

PROC
- N1-EOP-4 Rev. NA
- N1-EOP-2 Rev. NA
- N1-EOP-4.1 Rev. NA
- N1-EOP-4.2 Rev. NA

NUREG 1123 KA Catalog Rev. 2
- G2.3.9 2.5/3.4 Knowledge of the process for performing a containment purge

Level of Difficulty
- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 43(b)(4)

Question Source
- Bank

EXAMINATION ANSWER KEY

Senior Reactor Operator Written Examination Nine Mile Point Unit 1 2006

6 **USERID: NRC 2006** **SYSID: 21306** **Points: 1.00**
SRO 6

The plant is experiencing a transient, with the following:

Time	Event
t = 0	A primary system leak has occurred in the secondary containment
t = 10 min	Stack radiation monitor (RN10B) is reading 5.2×10^{6} cps
t = 15 min	The leak has been isolated
t = 24 min	Field Surveys at the site boundary indicates TEDE dose of 50 mRem
t = 25 min	Secondary containment general area temperatures are stabilized at 125°F

Which one of the following actions is required based on these radiological conditions?

A. Declare an ALERT. Entry into N1-EOP-8 is required.

B. Declare an ALERT. Entry into N1-EOP-8 is NOT required.

C. Declare a SITE AREA EMERGENCY. Entry into N1-EOP-8 is required.

D. Declare a SITE AREA EMERGENCY. Entry into N1-EOP-8 is NOT required.

Answer: B

Answer Explanation: B. is correct. EAL 5.1.3 states a valid reading from an unplanned release on any monitors for Table 3 "SAE" column for > 15 min unless dose assessment can confirm releases are below Table 4 column "SAE" within this time period. A field survey determined that TEDE was 50 mRem 14 minutes after the high stack radiation level. This dose projection is classified as an ALERT on Table 4. This corresponds to EAL 5.2.3 where it states that dose projections of field surveys resulting from actual or imminent release which indicate doses/dose rates ≥ Table 4 column "ALERT" at the site boundary or beyond.

A. and C. are incorrect. Entry into N1-EOP-8 is not required because all general area temperatures are less than 135 °F.

D. is incorrect. Field surveys were able to determine that dose was less than Table 4 SAE levels before 15 minutes which makes declaring EAL 5.1.3 incorrect.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 6 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 6
System ID:	21306
User ID:	NRC 2006 SRO 6
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 27, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Reference Provided: EAL MATRIX, EPIP-EPP-01; EOP-5 flowchart. Enabling Objectives: O3-OPS-006-350-3-21 EO-1.3 O1-OPS-006-344-1-08 EO-1.2

Question 6 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2

- G2.4.4 4/4.3 Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures

Level of Difficulty

- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level

- 2 Higher Cognitive Level comprehensive

10CFR55

- 43(b)(2)

Question Source

- New

7 **USERID: NRC 2006** **SYSID: 21250** **Points: 1.00**
SRO 7

The plant is operating at 100% power with the following:

t = 0 sec	A loss of Feedwater injection to the RPV occurs
t = 10 sec	Reactor is manually scrammed
t = 30 sec	Reactor power is 10%
t = 30 sec	Reactor Operator (RO) restarts a Feedwater Pump
t = 1:00 min	CRS orders EOP jumpers installed per EOP-1 Att 2
t = 1:30 min	CRS orders injection terminated and prevented
t = 2:00 min	RPV water level is 0 inches and lowering
t = 2:30 min	RO installs first EOP jumper inside N Panel
t = 3:00 min	RPV water level is -45 inches
t = 3:30 min	Liquid Poison injection to RPV begins
t = 4:00 min	RO completes installation of EOP-1 Att 2 jumpers and reports to CRS

Which one of the following is the impact of these events on the MSIVs and the actions required as a result?

A. MSIVs are closed but can be opened now to restore condenser as a heat sink.

B. MSIVs are closed and must remain closed until the Liquid Poison Tank is empty.

C. MSIVs are still open and RPV depressurization will continue using turbine bypass valves.

D. MSIVs are still open and RPV water level is to be maintained between -84 inches and -41 inches.

Answer: A

Answer Explanation: A. is correct. The MSIVs have automatically isolated when level reached 5 inches because the EOP jumpers are not yet installed. Once the jumpers are installed, EOP action direct the MSIVs be reopened, as long as the condenser is available and no indication of a steam break exists.

B. is incorrect because MSIVs are to be reopened.

C. and D. are incorrect because the MSIV are not still open. The MSIVs have automatically isolated when level reached 5 inches because the EOP jumpers are not yet installed.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 7 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 7 REV 1
System ID:	21250
User ID:	NRC 2006 SRO 7
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Jan 24, 2007
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	**Reference Provided: N1-EOP-3 Flowchart** Enabling Objective: O1-OPS-006-344-1-03 EO-1.2 Initially submitted as RO 25. Changed to resolve NRC comments.

Question 7 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

PROC
- N1-EOP-3 Rev. NA

NUREG 1123 KA Catalog Rev. 2
- G2.4.16 3/4 Knowledge of EOP implementation hierarchy and coordination with other support procedures

Level of Difficulty
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 41(b)(7)

Question Source
- New

8 **USERID: NRC 2006 SRO 8** **SYSID: 21308** **Points: 1.00**

The plant is experiencing a transient, with the following::

- RPV is depressurized
- Core Spray Pump 111 amps are 80 amps
- Core Spray Pump 112 amps are 0 amps
- Core Spray System 11 is injecting 210 x 10^4 lbm/hr
- RPV level is -84 inches and rising 10 inches/minute
- Torus pressure is 2.3 psig
- Torus temperature is 163 °F
- Torus water level is 10.1 feet

Which one of the following is the correct DIRECTION regarding Core Spray System 11 flow that prevents damaging the pumps while level is restored to 53 inches?

A. Maintain flow rate even though the limit is currently exceeded.

B. Maintain flow rate because adequate margin to the limit exists.

C. Lower flow to at least 190 x 10^4 lbm/hr to get below the limit.

D. Raise flow but stay below 240 x 10^4 lbm/hr to stay below the limit.

Answer: C

Answer Explanation: C. is correct. Torus Overpressure = Torus Pressure +0.433(Torus Level-4.5) = 2.3 psig +0.433(10.1-4.5) = 2.3+2.42 = 4.72 psig. So the 0 psig curve is used on detail N1 in N1-EOP-2. With one pump running and Torus temperature = 163 °F the maximum flow is 190x10^4 lbm/hr. In order to prevent damaging the Core Spray pump, flow must be reduced to at least 190x10^4 lbm/hr.

A. is incorrect. Maintaining the current flow rate may damage the pump, but under different conditions, such as inability to raise level above TAF or if this is the only pump running, EOP basis would allow operation under these conditions.

B. is incorrect. Currently the flow rate exceeds the limit. This answer would be correct if the 5 psig Torus Overpressure curve was chosen.

D. is incorrect. Raising the flow rate may damage the pump. This answer would be correct if the 5 psig Torus Overpressure curve was chosen.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 8 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 8
System ID:	21308
User ID:	NRC 2006 SRO 8
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 27, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Reference Provided: EOP-2, Enlarged EOP-2 Detail N1 for legibility Enabling Objective: O1-OPS-006-344-1-02 EO-1.2

Question 8 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

OTHER_REFS

- (BOOK) NMP1 EOP/SAP BASIS DOCUMENT NER-1M-095, Rev, NA

PROC

- N1-EOP-2 Rev. NA

NUREG 1123 KA Catalog Rev. 2

- 209001 A2.10 3.1/3.4 Ability to (a) predict the impacts of the following on the LOW PRESSURE CORE SPRAY SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High suppression pool temperature

Level of Difficulty

- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level

- 2 Higher Cognitive Level comprehensive

10CFR55

- 43(b)(5)

Question Source

- New

The plant is at rated power with the following:

- RPS Bus 11 is deenergized due to a ground
- Scram air header pressure lowered to 58 psig and is rising
- F3-3-2, CRD CONTROL AIR PRESSURE HI-LO, is in alarm

Which one of the following describes the procedural entry requirements?

A. N1-SOP-1.1, Emergency Power Reduction, to meet Feedwater minimum temperature requirements for entering the reactor.

B. N1-SOP-1.1, Emergency Power Reduction, in anticipation of a scoop tube lock up on the RRMG sets.

C. N1-EOP-2, RPV Control, due to low reactor water level following a reactor scram. N1-EOP-3, Failure to Scram, is not required.

D. N1-EOP-2, RPV Control, due to low reactor water level following a reactor scram. Then transition to N1-EOP-3, Failure to Scram.

Answer: D

Answer Explanation: D is correct. With the scram air header pressure < 60 psig a scram is required by N1-SOP-21.2. RPIS is powered from RPS 11. With RPS Bus 11 de-energized, RPIS data will be unavailable to confirm rod position on a scram. Therefore, requiring entry into N1-EOP-3.

A and B are incorrect. With the scram air header <60 psig a reactor scram is required by SOP-21.2. An Emergency Power Reduction is not the correct action to take. A is plausible because loss of RPS 11 does result in a loss of Feedwater Heater strings. B is plausible because a loss of instrument air will cause a scoop tube lockup.

C. is incorrect. Without being able to verify rod position N1-EOP-3 is required to be entered.

Question 1 Details

Associated objective(s):

1. LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question Type:	Multiple Choice
Topic:	NRC SRO 9 REV 1
System ID:	26981
User ID:	NRC 2006 SRO 9
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	4.00
Time to Complete:	3
Point Value:	1.00
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Bank: 27015 O1-OPS-001-201-1-02 Reactor Manual Control & Rod Position Indication Systems EO-1.7, As it relates to the RPIS, from memory, describe the basic operational actions, and using appropriate NMP procedures, identify specific actions in relation to: e. Correcting alarm conditions EO-1.8, Describe the impact of component malfunctions on the RPIS system. **REFERENCES PROVIDED: NONE**

Question 1 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2
- 212000 A2.02 3.7/3.9 RPS bus power supply failure

Level of Difficulty
- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 43(b)(5)

Question Source
- Bank

10 **USERID: NRC 2006 SRO 10** **SYSID: 21310** **Points: 1.00**

The plant is operating at rated power with the following:

- Turbine Generator output drops from 640 to 590 MWe
- Torus temperature starts to rise
- No acoustic monitors are in alarm
- ERV tail pipe temperature for ERV 121 is 300 °F

Which one of the following identifies the required procedure to enter and the limit placed on plant operation?

A. Enter N1-SOP-1.4, STUCK OPEN ERV. Reduce pressure below 110 psig in 10 hours and restore ERV 121 instrumentation to an operable status within 30 days.

B. Enter N1-SOP-1.4, STUCK OPEN ERV. Reduce pressure below 110 psig in 10 hours and restore ERV 121 instrumentation to an operable status during the next cold shutdown.

C. Execute N1-OP-1 Cycling Leaking ERV(s) During Power Operation. Restore ERV 121 instrumentation to an operable status within 30 days or be in hot shutdown within the next 12 hours.

D. Execute N1-OP-1 Cycling Leaking ERV(s) During Power Operation. Restore ERV 121 instrumentation to an operable status during the next cold shutdown.

Answer: B

Answer Explanation: B. is correct. A drop in reactor power, torus temperature rising and a high ERV tailpipe temperature are all indications of a stuck open ERV. Since the acoustic monitor should have alarmed it is required to be declared inoperable. When an ERV lifts at power it is considered to be unreliable due to unnecessarily opening and is declared inoperable. T.S. 3.1.5 says that with less than 6 ERV's operable it is required to drop pressure below 110 psig within 10 hours. T.S. 3.6.11 says that if 1 relief valve position indicator is inoperable then it is required to be fixed during the next cold shutdown.

A. and C. are incorrect. It would be correct to have the relief valve position indication fixed within 30 days if 2 indicators were inoperable. The conditions in the stem only lead to a determination that one ERV position indication is inoperable.

D. is incorrect. With a rise in torus temperature and an ERV tail pipe temperature of 300 F the ERV is open. A determination that the ERV was just leaking by would be incorrect.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 10 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 10
System ID:	21310
User ID:	NRC 2006 SRO 10
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Feb 20, 2007
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Reference Provided: T.S. 3.1.5, 3.2.9, 3.6.11 and the steam tables Enabling Objectives: O1-OPS-001-239-1-01 EO-1.8 O3-OPS-006-350-3-21 EO-1.3

Question 10 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2
- G2.4.11 3.4/3.6 Knowledge of abnormal condition procedures

Level of Difficulty
- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 43(b)(5)

Question Source
- New

11 **USERID: NRC 2006 SRO 11** **SYSID: 21311** **Points: 1.00**

The plant is in a refueling outage, with the following:

- Irradiated fuel is in the vessel
- RPV water level is 70 inches
- Maintenance requires reactor water level to be lowered to -35 inches

Which of the following is an additional requirement that must be met per Tech Specs before level is lowered below -10 inches?

A. Fuel zone water level instruments must be verified operable.

B. Valves which may lower water level must be caution tagged.

C. Redundant reactor water level instrumentation must be installed.

D. Water level must be monitored in the reactor building by an operator.

Answer: C

Answer Explanation: C. is correct. Per T.S. 2.1.1.d with irradiated fuel in the vessel the water level shall not be lower than -10 inches except as specified in 2.1.1.e which states that in order to lower reactor water level less than -10 inches redundant instrumentation will be provided to monitor the reactor water level.

A. is incorrect. Fuel zone water level instruments are existing instruments and do not get applied to the requirement to have additional level instrumentation.

B. is incorrect. Valves that have the potential to lower reactor water level are required to be danger tagged.

D. is incorrect. Water level is required to monitored continuously from the control room by a control room operator.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

EXAMINATION ANSWER KEY

Senior Reactor Operator Written Examination Nine Mile Point Unit 1 2006

Question 11 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 11
System ID:	21311
User ID:	NRC 2006 SRO 11
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Jan 09, 2007
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Reference Provided: None Enabling Objective: O1-OPS-001-216-1-01 EO-1.11

Question 11 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

TECHSPEC

- 2.1.1 Rev.

NUREG 1123 KA Catalog Rev. 2

- 259002 Reactor Water Level Control 2.2.24 Ability to analyze the affect of maintenance activities on LCO status 3.8

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 1 Memory/Fundamental Knowledge

10CFR55

- 43(b)(2)

Question Source

- New

12 **USERID: NRC 2006 SRO 12** **SYSID: 21312** **Points: 1.00**

A reactor startup is in progress with the following conditions:

- IRM 14 bypassed
- Annunciator F2-3-6, IRM 11-14 is received
- IRM 11 is spiking between downscale and upscale alarm set points
- All other IRMs are 25 on range 4

Which one of the following actions is required?

A. Trip RPS system 11 within 12 hours. Rod withdrawal is prohibited.

B. Trip RPS system 11 within 12 hours. Rod withdrawal is permitted.

C. Verify sufficient channels operable to maintain trip capability within 1 hour and trip RPS system 11 within 6 hours. Rod withdrawal is prohibited.

D. Verify sufficient channels operable to maintain trip capability within 1 hour and trip RPS system 11 within 6 hours. Rod withdrawal is permitted.

Answer: A

Answer Explanation: A. is correct. With less than 3 IRMs operable T.S. Table 3.6.2a note "o" requires a trip to be inserted on RPS Bus 11 with in 12 hours. T.S. 3.6.2g for instruments that cause control rod withdrawal blocks also applies. Since this LCO is not met and there is no compensatory action provided within the specification, T.S. 3.6.2(a)(7) must be applied, which prohibits control rods withdrawal.

B. is incorrect. Per T.S. 3.6.2(a)(7) with 2 IRMs inoperable withdrawing control rods in prohibited.

C. and D. are incorrect. These would be correct if 2 channels required by Table 3.6.2(a) were inoperable. Table 3.6.2(a) only required 3 channels to be operable and 2 IRM channels remain.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 12 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 12
System ID:	21312
User ID:	NRC 2006 SRO 12
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 27, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Reference Provided: Tech Spec 3.6.2 Enabling Objective: O1-OPS-001-215-1-02 EO-1.11

Question 12 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

TECHSPEC
- 3.6.2 Rev. NA

NUREG 1123 KA Catalog Rev. 2
- 215003 A2.05 3.3/3.5 Ability to (a) predict the impacts of the following on the IRM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Faulty or erratic operation of detectors/system

Level of Difficulty
- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 43(b)(1)

Question Source
- New

13 **USERID: NRC 2006 SRO 13** **SYSID: 21313** **Points: 1.00**

The plant is operating at 100% power with the following:

- Level transmitter 36-05A is isolated for surveillance testing

THEN 2 hours later the following occurs:

- A seismic event
- Test signal inputted to the analog trip unit corresponds to a level of 30 inches
- Level transmitter 36-05C fails upscale
- Pressure transmitter 36-07C fails downscale

Which one of the following is the impact on plant operations and applicable LCO actions and completion times?

A. ADS will not initiate on a valid signal. Shutdown and cool down to less than 110 °F within 10 hours. Ensure the non-ADS instrument channels are tripped within 12 hours.

B. ADS will not initiate on a valid signal. Shutdown and cool down to less than 110 °F within 10 hours. Ensure one non-ADS instrument channel is tripped within 12 hours and the other is tripped within 24 hours.

C. ADS will initiate on a valid signal. Place the inoperable ADS channel in the tripped condition within 24 hours. Ensure the non-ADS instrument channels are tripped within 12 hours.

D. ADS will initiate on a valid signal. Place the inoperable ADS channel in the tripped condition within 24 hours. Ensure one non-ADS instrument channel is tripped within 12 hours and the other is tripped within 24 hours.

Answer: A

Answer Explanation: A. is correct. In order for ADS to initiate it needs a valid initiation signal from one channel 11 level instrument, one channel 11 pressure instrument, one channel 12 level instrument and one channel 12 pressure instrument. With the channel 11 level instruments, 36-05A and 36-05C, reading above the ADS set point of -10 inches, ADS will not initiate on a valid ADS initiation signal. Per T.S. 3.6.2(f) if two channels of a trip system are inoperable then T.S. 3.6.2(a) applies. For an instrument that initiates ADS T.S. 3.6.2(a) states that T.S. 3.1.5 applies. T.S. 3.1.5 requires that the plant be shut down and cooled down to less than 110 °F within 10 hours. PT 36-07C inputs to both the primary coolant isolation logic and the scram logic. Per T.S. 3.6.2(b), primary coolant isolation, with one channel inoperable place the inoperable channel in the tripped condition within 12 hours for parameters common to scram instrumentation. Per T.S. 3.6.2(a) with one channel inoperable place the inoperable channel and/or that trip system in the tripped condition within 12 hours. Therefore, both non-ADS instrument channels are required to be tripped within 12 hours.

B. is incorrect. Because PT 36-07C provides both vessel isolation and scram functions T.S. 3.6.2(b) requires the channel to be tripped within 12 hours. If PT 36-07C was not common to a scram instrument it would have to be tripped within 24 hours.

C. and D. are incorrect. ADS will not initiate on a valid initiation signal because both channel 11 instruments will not respond to changes in RPV level and they are currently above the ADS set point of -10 inches.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 13 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 13
System ID:	21313
User ID:	NRC 2006 SRO 13
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 27, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Reference Provided: Tech Specs 3.1.5, 3.6.2 Enabling Objective: O1-OPS-001-216-1-01 EO-8 & EO-1.11

Question 13 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2
- 216000 A2.12 2.8/2.9 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: Instrument isolation valve closures

Level of Difficulty
- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 43(b)(5)

Question Source
- New

14 **USERID: NRC 2006 SRO 14** **SYSID: 21314** **Points: 1.00**

The plant is experiencing a seismic event with the following:

- An ATWS has occurred
- Drywell pressure is 22 psig
- Torus level is 11 ft
- Torus pressure is 20 psig and rising
- Torus temperature is 120 °F
- Both PB 102 and 103 have tripped on faults
- Reactor power is downscale
- RPV pressure is 675 psig and stable
- RPV water level is -84 inches and lowering

Which one of the following actions must the CRS direct?

	RPV pressure	RPV water level
A.	Maintain 600-800 psig	Maintain -84 to -109 inches
B.	RPV blowdown	Maintain -84 to -109 inches
C.	Maintain 600-800 psig	Terminate and prevent
D.	RPV blowdown	Terminate and prevent

Answer: D

Answer Explanation: D. is correct. With torus pressure at 20 psig and torus level at 11 the Pressure Suppression Pressure (PSP) curve has been violated. Without PB 102 or PB 103 containment spray can not be initiated, which means a RPV BLOWDOWN (N1-EOP-8) is required per N1-EOP-4 PCP 6. When N1-EOP-8 is entered the level leg of N1-EOP-3 is exited. N1-EOP-8 directs RPV injection to be terminated and prevented to prevent an uncontrolled cold water injection which could cause a power excursion and core damage.

A. and B. are incorrect. This would be the direction given by the CRS if the level leg of N1-EOP-3 was not exited when an RPV BLOWDOWN is directed.

C. is incorrect. PSP has been violated which requires a reactor blowdown.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 14 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 14
System ID:	21314
User ID:	NRC 2006 SRO 14
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Jan 09, 2007
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Reference Provided: N1-EOP-3, 4, 8 flowcharts. Enabling Objectives: O1-OPS-006-344-1-03 EO-1.2 O1-OPS-006-344-1-04 EO-1.2 O1-OPS-006-344-1-08 EO-1.2

Question 14 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

PROC
- N1-EOP-3 Rev. NA
- N1-EOP-5 Rev. NA
- N1-EOP-8 Rev. NA

NUREG 1123 KA Catalog Rev. 2
- G2.4.6 3.1/4 Knowledge symptom based EOP mitigation strategies

Level of Difficulty
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 43(b)(5)

Question Source
- New

15 **USERID: NRC 2006** **SYSID: 26979** **Points: 1.00**
SRO 15

The plant is operating at rated power with the following:

- N1-ST-Q6A, CONTAINMENT SPRAY SYSTEM LOOP 111 QUARTERLY OPERABILITY TEST, has just been completed
- Containment Spray Raw Water Pump 111 has a flow rate of 2900 gpm
- Containment Spray Pump 122 suction valve was closed due to excessive pump seal leakage

Which one of the following describes the status of the Containment Spray System and the applicable LCO?

A. A redundant component in one system is inoperable. Return the component within 15 days.

B. A redundant component in each system is inoperable. Return the components within 7 days.

C. One system is inoperable. Return the system to operation within 7 days.

D. Both systems are inoperable. A shutdown shall begin within one hour.

Answer: B

Answer Explanation: B. is correct. Containment Spray Loop 111 is inoperable because Containment Spray Raw Water Pump 111 has a flow rate of < 3000 gpm. In the basis for T.S. 3.3.7 it states the containment spray raw water cooling system is considered operable when the flow rate is not less than 3000 gpm. Containment Spray Loop 122 is inoperable because the suction valve for Containment Spray Pump 122 is shut. One of these pumps is in system 11 and one is in system 12. Therefore, a redundant component in each system is inoperable and a 7 day LCO applies. N1-OP-14 Precaution and Limitation D.1.0 describes the application of redundant components and application of the LCO.

A. is incorrect. This answer would be true if Containment Spray Raw Water Pump 122 was operable.

C. is incorrect. This answer would be correct if Containment Spray Pump 111 and 122 were in the same system.

D. is incorrect. This answer would be correct if the loss of both Containment Spray loop 111 and 122 made both system 11 and system 12 inoperable.

Question 15 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 15 REV 1
System ID:	26979
User ID:	NRC 2006 SRO 15
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	3.00
Time to Complete:	3
Point Value:	1.00
Date Changed:	Feb 22, 2007
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	226000 RBO 14

REFERENCES PROVIDED: NONE

Question 15 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2
- G2.2.25 2.5/3.7 Knowledge of bases in technical specifications for limiting conditions for operations and safety limits
- 226001 RHR/LPCI: Containment Spray System Mode

Level of Difficulty
- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 43(b)(2)

Question Source
- New

16 **USERID: NRC 2006** **SRO 16** **SYSID: 21316** **Points: 1.00**

The plant is operating at 90% power with the following:

- 5 Recirc Loops are operating
- Recirc Pump 14 trips
- CRS has directed the RO to close Recirc Pump 14 discharge valve

Which one of the following describes the impact on the Flow-Biased Scram set point & Technical Specifications after the RRP 14 discharge valve is full closed?

	Flow-Biased Scram set point	TECH SPEC IMPACT
A.	Lower	Reactor power limit is 90.5%
B.	Higher	Reactor power limit is 90.5%
C.	Lower	APRMs will be operable
D.	Higher	APRMs will be operable

Answer: C

Answer Explanation: C. is correct. When the RRP 14 discharge valve is shut the indicated flow will lower to match actual core flow, which makes the APRMs operable.

A. and B. are incorrect. Reactor power is only limited to 90.5% if a RRP loop is isolated.

D. is incorrect. The Flow-Biased Scram set point will lower because indicated flow will lower.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 16 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 16
System ID:	21316
User ID:	NRC 2006 SRO 16
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 27, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Reference Provided: NONE Enabling Objective: O1-OPS-001-202-1-01 EO-1.7

Question 16 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

PROC

- N1-SOP-1.3 Rev. NA

NUREG 1123 KA Catalog Rev. 2

- 295001 AA2.03 3.3/3.3 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Actual core flow

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 2 Higher Cognitive Level comprehensive

10CFR55

- 43(b)(6)

Question Source

- New

17 **USERID: NRC 2006** **SYSID: 21317** **Points: 1.00**
SRO 17

The plant is operating at 100% power with the following:

12/10/06 0800 Core Spray Pump 111 is declared inoperable
12/10/06 0800 Core Spray surveillance with inoperable components is performed
12/10/06 1200 R10 Breaker trips and cannot be re-closed
12/10/06 1600 EDG103 is declared inoperable

Which one of the following is the LATEST time that a plant shutdown can be initiated while still complying with Tech Specs?

A. 12/10/06 at 1300

B. 12/10/06 at 1700

C. 12/11/06 at 1700

D. 12/17/06 at 0900

Answer: B

Answer Explanation: B. is correct. TS 3.1.4 is entered when Core Spray Pump is inoperable and when EDG 103 is declared inoperable at 12/10 1600, 3.6.3 spec g is entered. This states that ***when operating with only one diesel generator, all emergency equipment aligned to the operable diesel generator shall have no inoperable components.*** Since Core Spray Pump 111 is inoperable and aligned to EDG 102, specification g is not met. Normal orderly shutdown must be initiated within one hour, or 12/10 at 1700. TS 3.0.1 also applies which requires TS 3.1.4.d to be implemented.

TS 3.6.3 spec b is entered when R10 trips at 12/10 1200, with 7 day completion time.

A. is incorrect because tech specs do not require a plant shutdown initiated with a Core Spray pump inoperable concurrent with an offsite power source being inoperable. Time is based on initiating a shutdown within one hour (12/10 at 1300) of the offsite line being inoperable at 12/10 at 1200.

C. is incorrect because the earlier condition of the inoperable Core Spray pump with only one diesel generator operable requires the shutdown be initiates first. If the Core Spray pump were operable, then the limiting spec would be 3.6.3 spec c, which requires the diesel be restored within 24 hours, if inoperable coincident with a 115 KV line being de-energized or inoperable. If not met, this requires the shutdown to be initiated after 24 hours later at 12/11 at 1700, after applying the additional 1 hour to initiate a shutdown.

D. is incorrect but applies the expiration of 7 day completion time from when the first component became inoperable on 12/10 at 0800. 12/17 at 0900 is the expiration of the 7 day LCO time after applying the additional 1 hour to initiate a shutdown.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 17 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 17
System ID:	21317
User ID:	NRC 2006 SRO 17
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 27, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Reference Provided: TS 3.1.4 and 3.6.3 and surrounding sections Enabling Objective: O1-OPS-001-209-1-01 EO-1.11

Question 17 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

TECHSPEC

- 3.1.4 Rev. NA
- 3.6.3 Rev. NA

NUREG 1123 KA Catalog Rev. 2

- 295003 AA2.04 3.5/3.7 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: System lineups

Level of Difficulty

- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level

- 2 Higher Cognitive Level comprehensive

10CFR55

- 43(b)(5)

Question Source

- New

18 **USERID: NRC 2006** **SYSID: 21318** **Points: 1.00**
SRO 18

The plant is operating at 100% power with the following:

- Torus water temperature instrument TI 201.2-522B is reading downscale

Which one of the following actions is required, if any?

A. Operation can continue beyond 30 days because the parameter can still be monitored on the redundant remote shutdown panel.

B. Operation can continue beyond 30 days because the parameter can still be monitored using primary instruments in the main control room.

C. Instrument must be returned to operable status in 30 days or the plant must be in HOT SHUTDOWN within the next 12 hours.

D. Instrument must be returned to operable status in 7 days or the plant must be in HOT SHUTDOWN within the next 12 hours.

Answer: A

Answer Explanation: A is correct. N1-SOP-29.1 Attachment 1 Table page 6 identifies the instrument as remote shutdown panel RSP 12 torus temperature indicator. TS Table 3.6.13-1 identifies the minimum operable channels per function as one. The TS bases state that "one channel for each function provides the necessary capabilities...." With one inoperable channel, the function is still maintained by the RSP 11 instrument. The bases also states that "Therefore only one channel on either remote shutdown monitoring instrument is required to be operable." Establish alternate monitoring within 30 days and must restore within 90 days.

B. is incorrect because the control room instruments and the remote shutdown instruments have different functions and tech specs.

C. is incorrect because the monitoring functional capability is still met. Therefore, specification c is not entered and the 30 day completion time does not apply.

D. is incorrect because a 7 day LCO is not entered. The 7 day LCO is from TS 3.6.11 parameter 8, control room monitoring instruments. Action 2(a) applies if an accident monitoring instrument channel is inoperable.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

EXAMINATION ANSWER KEY

Senior Reactor Operator Written Examination Nine Mile Point Unit 1 2006

Question 18 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 18
System ID:	21318
User ID:	NRC 2006 SRO 18
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Jan 09, 2007
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Reference Provided: N1-SOP-29.1, TS 3.6.13 and 3.6.11 and surrounding sections. Enabling Objective: O1-OPS-001-212-1-01 EO-1.11

Question 18 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

PROC

- N1-SOP-29.1, Rev. NA

TECHSPEC

- 3.6.11 Rev. NA
- 3.6.13 Rev.

NUREG 1123 KA Catalog Rev. 2

- 295016 AA2.04 3.9/4.1 Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT: Suppression pool temperature

Level of Difficulty

- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level

- 2 Higher Cognitive Level comprehensive

10CFR55

- 43(b)(5)

Question Source

- New

19 **USERID: NRC 2006 SRO 19** **SYSID: 21319** **Points: 1.00**

The phase 1 core shuffle is in progress. Which one of the following conditions meets the Fuel Handling Procedure (FHP) criteria for stopping fuel movement per N1-FHP-25, general description of Fuel moves?

A. Last performance of the Refueling Platform Interlocks Test was completed twenty-four (24) hours ago.

B. Air bubbles are observed from the vicinity of the grapple head when a fuel assembly is latched or disengaged.

C. The fuel assembly nose piece is lowered to two (2) feet above the core top guide before establishing the correct orientation.

D. A fuel assembly is moved to the spent fuel pool and the rod block interlock light clears when the bridge is clear of the reactor core.

Answer: B

Answer Explanation: Objective: O1-OPS-001-234-1-02, EO-1.8.b

N1-FHP-25, Attachment 4

B. is correct. Air leakage on the fuel grapple requires stopping fuel movement. Evidence of air bubbles indicates air leakage.

A. is incorrect. This interval for this surveillance is 7 days. The surveillance is current. A similar surveillance N1-PM-SO, Refuel Platform and Grapple Inspection, is required to be completed every 12 hours.

C. is incorrect. It is acceptable to lower the fuel assembly before establishing the correct orientation, however, the correct orientation is to be established before lowering the fuel assembly into the assigned core location.

D. is incorrect. This is correct operation of the refueling interlocks. The rod block interlock light is lit when the refueling bridge is over the reactor core and the main hoist is loaded. When clear of the reactor core (proximity switch) the rod block clears.

References Provided: NONE

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 19 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 19
System ID:	21319
User ID:	NRC 2006 SRO 19
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 27, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	21097
Num Field 2:	
Text Field:	
Comments:	Reference Provided: NONE Enabling Objective: O1-OPS-001-234-1-02 EO-1.8

Question 19 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

PROC

- N1-FHP-25 Rev. NA

NUREG 1123 KA Catalog Rev. 2

- G2.2.28 2.6/3.5 Knowledge of new and spent fuel movement procedures
- 295023 Refueling Acc

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 1 Memory/Fundamental Knowledge

10CFR55

- 43(b)(7)

Question Source

- Bank

20 **USERID: NRC 2006 SRO 20** **SYSID: 21320** **Points: 1.00**

A major air leak in the scram air header results in an ATWS with the following conditions:

- Bypass valves are failed shut
- Reactor power is 20 %
- Reactor water level is at -50 inches and lowering
- Both emergency condensers are in service
- Reactor pressure is 700 psig
- Torus water temperature is 136 °F
- Torus water level is 11.1 ft
- Initial tank level was 1450 gal
- Current liquid poison tank level is 1000 gallons

Which one of the following indicates the **NEXT** action that is required to be directed by the Control Room Supervisor?

A. Perform RPV Blowdown due to HCTL violation.

B. Raise reactor water level to 53 to 95 inches to mix boron.

C. Reduce RPV pressure to stay below applicable HCTL limit.

D. Install RPS jumpers to manually drive control rods with RMCS.

Answer: C

Answer Explanation: C. is correct. For a torus water level of 11.1 ft curve A on the HCTL graph is used. When RPV pressure is 700 psig then the torus temperature limit on curve A is 140 °F. Reactor is at 20% power. Emergency condensers take away 6 % reactor power. An excess of 14% reactor power is being sent to the torus. The CRS should direct reactor pressure to be lowered to stay below HCTL per N1-EOP-2 P4.

A. is incorrect. A blow down is not required unless HCTL has been violated. This answer would be the correct answer if curve B is used on the HCTL graph. Curve B is only used if Torus level is 9-10.5 ft or 11.25-13.0 ft.

B. is incorrect. Level is only raised in N1-EOP-3 when hot shutdown boron is injected. Hot shutdown boron is 600 gallons and only 450 gallons have been injected.

D. is incorrect. An air leak in the scram air header caused the initial reactor scram. Even if the scram was reset the header would not repressurize, which means that installing RPS jumpers to insert Control Rods will not work.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 20 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 20
System ID:	21320
User ID:	NRC 2006 SRO 20
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Jan 09, 2007
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Reference Provided: N1-EOP-2,3,4 & HCTL GRAPH Enabling Objectives: O1-OPS-006-344-1-02 EO-1.2 O1-OPS-006-344-1-04 EO-1.2

Question 20 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

OTHER_REFS
- HCTL GRAPH, Rev. NA

PROC
- N1-EOP-2 Rev. NA
- N1-EOP-3 Rev. NA
- N1-EOP-4 Rev. NA

NUREG 1123 KA Catalog Rev. 2
- 295026 EA2.02 3.8/3.9 Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Suppression pool level

Level of Difficulty
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 43(b)(5)

Question Source
- New

21 **USERID: NRC 2006** **SYSID: 21321** **Points: 1.00**
SRO 21

The plant is experiencing a transient from rated power with the following:

• Core Spray Loop 11 status	Injecting to RPV
• Reactor pressure	50 psig
• Drywell temperature Elevation 319	266°F
• Fuel Zone level indication	NOT flashing
• Wide Range level indication	1.5 feet
• GEMAC Channel 11	0 inches
• GEMAC Channel 12	10 inches
• Lo-Lo-Lo Channel 11	50 inches
• Lo-Lo-Lo Channel 12	14 inches

Which one of the following RPV level instruments are allowed to be used to monitor Reactor Water level?

A. Lo-Lo-Lo Channel 11 and Fuel Zone.

B. Lo-Lo-Lo Channel 12 and Wide Range.

C. Lo-Lo-Lo Channel 11 and GEMAC Channel 11.

D. Lo-Lo-Lo Channel 12 and GEMAC Channel 12.

Answer: D

Answer Explanation: D. is correct. Use EOP-2 Detail A to determine useable level instruments. Lo-Lo-Lo channel 12 minimum usable level is -20" and its actual level is reading 14". Also Core Spray 12 is not injecting. Core Spray 12 injects through the variable leg tap of Lo-Lo-Lo channel 12. Therefore, Lo-Lo-Lo channel 12 instrument is useable. GEMAC Channel 12 minimum useable level is 0 and its actual level is reading 10", which means GEMAC Channel 12 can also be used.

A. is incorrect. Core Spray 11 is injecting into the reactor. Core Spray 11 injects through the variable leg of the Lo-Lo-Lo Channel 11 instrument, which makes it inoperable.

B. is incorrect. Wide Range minimum usable level is 1.5 feet and it is at 1.5 feet, which means it cannot be used because level is at the variable leg tap and is no longer accurately indicating reactor water level.

C. is incorrect. Core Spray 11 is injecting into the reactor. Core Spray 11 injects through the variable leg of the Lo-Lo-Lo Channel 11 instrument, which makes it inoperable. GEMAC Channel 11's minimum usable level is 0 and it is at 10", therefore, GEMAC Channel 11 is useable.

The plant is experiencing a transient with the following:

t = 0001	N1-EOP-9, Steam Cooling is being used to maintain core cooling
t = 0003	Torus temperature is 135 °F and rising
t = 0006	Torus level is 10.6 feet and stable
t = 0007	RPV pressure is 950 psig and stable
t = 0009	Reactor water level is -111 inches and lowering

Which one of the following describes the required operator action and the basis behind those actions?

A. Immediately open 3 ERVs. The resulting swell will provide temporary adequate core cooling.

B. Immediately open 3 ERVs. Delaying a blowdown could cause a failure of the Primary Containment.

C. Wait until -121 inches then open 3 ERVs. The resulting swell will provide temporary adequate core cooling.

D. Wait until -121 inches then open 3 ERVs. Waiting until -121 inches ensures the lowest PCT while performing a blowdown below -109 inches.

Answer: B

Answer Explanation: B is correct. Per the EOP Basis, N1-EOP-9 Steam Cooling requires a blowdown at -121 inches. However, since HCTL is violated, EOP-4 requires a blowdown now. An RPV blowdown is required by EOP-4 so N1-EOP-8 is entered, overriding pressure control instructions in N1-EOP-9. The consequences of not performing an RPV blowdown when required could include failure of the primary or secondary containment, ultimately resulting in uncontrolled release of radioactivity to the environment.

A is incorrect. Initially a blowdown is performed at -121 inches due to the swell providing temporary core cooling. However, we are not yet at -121 inches and this blowdown is performed for primary containment concerns.

C and D are incorrect. Per the EOP Basis document when executing N1-EOP-9 a blowdown will not be delayed when it is required by another EOP.

Question 1 Details

Associated objective(s):

1. LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question Type:	Multiple Choice
Topic:	NRC SRO 22 REV 1
System ID:	26980
User ID:	NRC 2006 SRO 22
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	3.00
Time to Complete:	3
Point Value:	1.00
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	O1-OPS-006-344-1-09 EO-1.2 Given the EOPs, if appropriate, and a set of operating parameters and/or conditions, determine the appropriate actions to be taken. **REFERENCES PROVIDED: N1-EOP-4 AND 9**

Question 1 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2

- G2.4.6 3.1/4 Knowledge symptom based EOP mitigation strategies
- 295031 Reactor Low Water Level

Level of Difficulty

- Level 3: Moderately discriminating. Expected miss rate of 20% - 30%

Cognitive Level

- 2 Higher Cognitive Level comprehensive

10CFR55

- 43(b)(5)

Question Source

- New

23 **USERID: NRC 2006 SRO 23** **SYSID: 21323** **Points: 1.00**

The plant is operating at 100% power with the following:

- Diesel Fire Pump is out of service
- Core Spray Pump 111 is out of service
- I&C requests permission to perform regularly scheduled half scram surveillance testing on RPS Channel 12
- Maintenance requests CRD Pump 11 control switch placed in P-T-L while performing a visual inspection on its power supply breaker cubicle

Which one of the following describes how the risk associated with these maintenance activities is managed and the basis?

A. Allow the CRD pump breaker inspection because the activity will be completed before its allowable out of service time expires.

B. Allow the CRD pump breaker inspection only after verifying redundant CRD Pump will remain operable during the inspection.

C. Don't allow the CRD pump breaker inspection because vessel injection capability is degraded with Core Spray inoperable.

D. Don't allow the CRD pump breaker inspection because vessel injection capability is degraded with Fire Pump inoperable.

Answer: D

Answer Explanation: D. is correct. GAP-OPS-117 Attachment 20 does not allow a CRD pump and a Diesel Fire Pump to be out of service at the same time during a planned maintenance period, unless supported by PRA.

A. is incorrect because the restriction in GAP-OPS-117 does not allow this. Removing a CRD pump does have an allowable 7 day out of service time per TS 3.1.6.

B. is incorrect because the restriction in GAP-OPS-117 does not allow this. If a CRD pump is out of service, SR 4.1.6 specification c requires the redundant component to be verified operable immediately and daily thereafter.

C. is incorrect because although injection is degraded, it is because of the fire pump, not because Core Spray is out of service. GAP-OPS-117 does not restrict a CRD pump and Core Spray pump from being unavailable together.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 23 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 23
System ID:	21323
User ID:	NRC 2006 SRO 23
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Oct 27, 2006
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	10CFR 55.43(b)(5) NOTE: Selection of the appropriate procedure is implied rather than stated. To select the correct answer, the appropriate procedure or knowledge of that procedure was required. Reference Provided: NONE Enabling Objective: O3-OPS-006-343-3-60 EO-1.6

Question 23 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

PROC

- GAP-OPS-117 Rev. NA

NUREG 1123 KA Catalog Rev. 2

- G2.2.17 2.3/3.5 Knowledge of the process for managing maintenance activities during power operations
- 295022 Loss of CRD Pumps

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 2 Higher Cognitive Level comprehensive

10CFR55

- 43(b)(5)

Question Source

- New

24 **USERID: NRC 2006 SRO 24** **SYSID: 21324** **Points: 1.00**

The plant is operating at rated power with the following:

- 33-41, AO BLOCKING valve inadvertently closes
- Attempts to reopen 33-41 have failed
- No other operator actions have been taken

Which one of the following describes the impact on the Reactor Water Cleanup (RWCU) System and what operator actions are required?

A. Isolates on low flow. Remove RWCU from the thermal power calculation.

B. Isolates on low flow. Chemistry establishes an alternate sample path.

C. Heat exchanger damage. Remove RWCU from the thermal power calculation.

D. Heat exchanger damage. Chemistry establishes an alternate sample path.

Answer: B

Answer Explanation: B. is correct. When 33-41 shuts it stops flow in the RWCU system. The RWCU system will then isolate on low flow. Chemistry shifts the sample line up to RRP 11 to meet the required Technical Specification conductivity sample frequency.

A is incorrect. RWCU does not need to be removed from the thermal power calculation. RWCU can introduce errors into the thermal power calculation when rejecting to the condenser or to rad waste.

C. is incorrect. Closing 33-41 prevents damaging the heat exchanger. A RWCU isolation where 33-41 did not isolate would cause Heat exchanger damage. RWCU does not need to be removed from the thermal power calculation. RWCU can introduce errors into the thermal power calculation when rejecting to the condenser or to rad waste.

D. is incorrect. Closing 33-41 prevents damaging the heat exchanger. A RWCU isolation where 33-41 did not isolate would cause Heat exchanger damage.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 24 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 24
System ID:	21324
User ID:	NRC 2006 SRO 24
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Jan 11, 2007
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Reference Provided: T.S. 3.2.3 Enabling Objective: O1-OPS-001-204-1-01 EO-1.8 & 1.11

Question 24 Cross References (table item links)

Question Setting

- C1 (License class closed reference)

PROC

- N1-OP-3 Rev.

TECHSPEC

- 3.2.3 Rev. NA

NUREG 1123 KA Catalog Rev. 2

- 295020 AA2.06 3.4/3.8 Ability to determine and/or interpret the following as they apply to INADVERTENT CONTAINMENT ISOLATION: Cause of isolation

Level of Difficulty

- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level

- 1 Memory/Fundamental Knowledge

10CFR55

- 43(b)(5)

Question Source

- New

25 **USERID: NRC 2006 SRO 25** **SYSID: 27154** **Points: 1.00**

The plant is operating at 100% power with the following:

- A seismic event occurs
- Torus water level is slowly lowering
- H2-2-1, R BLDG FL DR SUMPS 11-16 AREA WTR LEVEL HIGH, alarms
- Computer Pt. F189 NW RB CORNER RM WTR LVL HIGH is in alarm
- The leak is between the Torus and Containment Spray Pump 111 Suction Isolation Valve
- The leak is determined to be 500 gpm

Which one of the following states the operability of Containment Spray Pump 121 and the required operator action per the Emergency Operating Procedures?

	Containment Spray Pump 121	Operator Action
A.	Operable	Continue Operation
B.	Operable	Scram
C.	Inoperable	Continue Operation
D.	Inoperable	Scram

Answer: C

Answer Explanation: C. is correct. Containment Spray Pumps 111 and 121 are in the NW corner room and are the components affected by the water level in the room. The alarm for F189 is actuated at a water level of 5 feet (Table S N1-EOP-5) in the NW corner room, which is the maximum safe value. The max safe value is defined to be the highest value at which equipment necessary for safe shutdown of the plant will operate. Therefore, Containment Spray Pumps 111 and 121 are inoperable. There is no isolation valve to isolate a leak between the Torus and the suction isolation valve for a containment spray pump. Therefore, the Torus leak is not isolable. However, the Torus is not a primary system because it is not directly linked to the reactor. In N1-EOP-4 a scram is only required if a primary system is leaking into the RB and cannot be isolated, which means a scram is not required and operation will continue until two values of the same parameter are above a max safe value. A containment spray raw water pump can maintain Torus water level with a 500 gpm Torus leak. A scram will not be required due to low Torus water level.

A. and B are incorrect. Containment Spray Pumps 111 and 121 are inoperable at 5 feet, which means they are inoperable at the current level.

D. is incorrect. There is no isolation valve to isolate a leak between the Torus and the suction isolation valve for a containment spray pump. Therefore, the Torus leak is not isolable. However, the Torus is not a primary system because it is not directly linked to the reactor. In N1-EOP-4 a scram is only required if a primary system is leaking into the RB and cannot be isolated, which means a scram is not required and operation will continue until two values of the same parameter are above a max safe value. A containment spray raw water pump can maintain Torus water level with a 500 gpm Torus leak. A scram will not be required due to low Torus water level.

Associated objective(s):

LC1 05-01 NRC EXAM DEVELOPMENT AREA

Question 25 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 25 REV 1
System ID:	27154
User ID:	NRC 2006 SRO 25
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Date Changed:	Jan 31, 2007
Cross Reference Number:	LC1 05-01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Reference Provided: N1-EOP-5 Enabling Objective: O1-OPS-006-344-1-05 EO-1.2

Question 25 Cross References (table item links)

Question Setting
- C1 (License class closed reference)

NUREG 1123 KA Catalog Rev. 2
- G2.1.25 2.8/3.1 Ability to obtain and interpret station reference materials such as graphs, monographs, and tables which contain performance data
- 295036 Secondary Containment High Sump/Area Water Level

Level of Difficulty
- Level 2: Slightly discriminating. Expected miss rate of 10% - 20%

Cognitive Level
- 2 Higher Cognitive Level comprehensive

10CFR55
- 43(b)(5)

Question Source
- New