Exam Data:

Exam Name: NRC August 2005 - 3
ExamID: NRC-082005-3
Exam Owner: Charles Bell
Exam Date: 08/12/2005
Created by: Charles Bell
Created at: Fri Jun 17 08:59:22 CDT 2005
Last Revised by: Charles Bell
Last Revised at: Fri Jun 17 08:59:22 CDT 2005
Reviewed By: Mickey Ellis
Review Date: Thu Aug 04 07:58:11 CDT 2005
Exam Review Status: Reviewed
Approved By: MikeRasch
Approval Date: Thu Aug 04 07:59:21 CDT 2005
Exam Approval Status: Approved

Access to this exam is restricted to:

- Charles Bell
- MikeRasch
- Mickey Ellis
- Tommy Harrelson

Exam History:

- Created by jbell at Fri Jun 17 08:59:22 CDT 2005
- Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
- Approved by mellis at Mon Jun 20 14:46:49 CDT 2005
- Questions added by mrasch at Mon Jul 18 15:28:20 CDT 2005
- Question 57 removed by mrasch at Mon Jul 18 15:29:00 CDT 2005
- Question 100 renumbered to 57 by mrasch at Mon Jul 18 15:29:18 CDT 2005
- Question 100 renumbered to 58 by mrasch at Mon Jul 18 15:31:57 CDT 2005
- Question 100 renumbered to 59 by mrasch at Mon Jul 18 15:33:34 CDT 2005
- Question 100 renumbered to 60 by mrasch at Mon Jul 18 15:35:19 CDT 2005
- Question 61 renumbered to 62 by mrasch at Mon Jul 18 15:36:28 CDT 2005
- Question 100 renumbered to 61 by mrasch at Mon Jul 18 15:37:00 CDT 2005
- Reviewed by mellis at Thu Aug 04 07:58:11 CDT 2005
- Approved by mrasch at Thu Aug 04 07:59:21 CDT 2005

Comments:

Edit Exam | Mail Message | Compare This Exam | Print Exam Data | Print Exam and Key | Generate E

EB QUESTION: 1 (1.0 Points)

The reactor is at 95% power and I&C is performing a surveillance on the Reactor Recirc Flow Control System. Suddenly, the A Reactor Recirc Flow Control Valve (FCV) unexpectedly ramps back to its minimum position.

Based on this:

A.

the reactor will scram on Flow Control Trip Reference (FCTR) and ONEP 05-1-02-I-1, Reactor Scram, should be entered.

B. reactor power will drop and ONEP 05-1-02-III-3, Reduction in Recirculation System Flow Rate should be entered.

C. the reactor will scram on high reactor water level and ONEP 05-1-02-I-1, Reactor Scram, should be entered.

D. reactor power will drop and Reactor Recirc FCV B should immediately be closed to its minimum position.

Answer: B

Question Comments: The reactor will not scram on a runback to minimum position. There are no immediate actions for this event, however it is an entry condition to ONEP Reduction in Recirculation System Flow Rate. Therefore B is th only correct answer. This is a NEW question. Tier 1 Group 1 CFR 41.5/41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00833

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-OPS-ONEP Objective: 2.0

KA References:

1. 295001 AK1.02 Power/flow distribution [3.3/3.5]
2. GENERIC 2.4.4 Ability to recognize abnormal indications for system operating parameters [4.0/4.3]
3. GENERIC 2.4.11 Knowledge of abnormal condition procedures [3.4/3.6]

References:

1. 05-1-02-III-3sect 3.1; 3.4; 3.5; Figure 1

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. B33: Reactor Recirculation System
2. C51-6: Period Based Detection System

Categories:

1. Off Normal Event Procedures
2. Systems

Task References:

Question Last Revised By: MikeRasch at Fri Jul 29 14:58:38 CDT 2005
Question History:

1. Created by tharrelso at Wed Apr 20 14:49:03 CDT 2005
2. Modified by tharrelso at Wed Apr 20 14:58:13 CDT 2005
3. Modified by tharrelso at Thu Apr 28 08:31:36 CDT 2005
4. Modified by mrasch at Tue May 24 07:52:01 CDT 2005
5. Modified by mrasch at Tue May 24 08:22:58 CDT 2005
6. Modified by mrasch at Tue May 31 14:40:03 CDT 2005
7. Question Reviewed by mellis at Tue May 31 14:57:01 CDT 2005
8. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
9. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
10. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
11. Modified by mrasch at Thu Jul 28 06:34:57 CDT 2005
12. Modified by tharrelso at Fri Jul 29 06:27:20 CDT 2005
13. Modified by tharrelso at Fri Jul 29 10:26:57 CDT 2005
14. Modified by mrasch at Fri Jul 29 14:58:38 CDT 2005
15. Question Reviewed by mellis at Thu Aug 04 07:23:44 CDT 2005
16. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 2 **(1.0 Points)**

Which one of the following describes the reasons for transferring Divisions 1 and 2 buses from offsite power sources to the diesel generators during degraded grid voltage situations?

A. Safety related loads are only allowed to be supplied power from Emerency Diesel Generators during accident conditions.

B. To preclude damage to safety related electrical equipment as a result of extended operation at reduced voltage levels.

C. To prevent core damage by ensuring safety systems are connected to a reliable power source and minimize response time during an accident.

D. To ensure the safety related equipment is supplied with a power source of sufficient capacity to ensure the plant can be maintained in a safe shutdown condition during an accident.

Answer: B

Question Comments: Answer A is INCORRECT because Offsite power sources are the preferred power sources for the ESF buses. Answer B is CORRECT because voltage degradation in conjunction with operating equipment can cause damage to safety related equipment normally operated on these buses. Answer C is incorrect because the transient times associated with transferring power to alternate feeds (Diesel Generators) is consistent with the accident analyses. Answer D is incorrect because the offsite power sources have sufficient capacity to handle all loads required for safe shutdown during accident conditions and do not have the lower limitations of the diesel generators. TIER 1 GROUP 1 This is a NEW question. CFR 41.7/41.8

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00464

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-R2100 Objective: 2
2. CourseID: GLP-OPS-R2100 Objective: 3.1

3. CourseID: GLP-OPS-R2100 Objective: 1

KA References:

1. 295003 AK3.01 Manual and auto bus transfer [3.3/3.5]

References:

1. Tech Spec Bases 3.3.8.1-1
2. UFSAR 8.3.1.1.2.2

TrainingPrograms:

1. Nonlicensed Operator Training Program
2. Reactor Operator Training Program
3. Senior Reactor Operator Training Program
4. Licensed Operator Requalification Training Program

Systems:

1. R21: 4.16 KV AC Power System

Categories:

1. Systems

Task References:

Question Last Revised By: MikeRasch at Tue Aug 09 10:45:04 CDT 2005
Question History:

1. Used on December 2000 NRC Exam
2. Used on August 2002 Audit RO Exam
3. Converted from MSWord on Tue May 25 14:16:47 CDT 2004
4. Imported at Tue May 25 14:24:02 CDT 2004
5. Modified by tharrelso at Wed Apr 20 15:54:30 CDT 2005
6. Modified by mrasch at Tue May 10 13:42:42 CDT 2005
7. Modified by mrasch at Tue May 24 07:55:38 CDT 2005
8. Question Reviewed by mellis at Tue May 31 14:56:59 CDT 2005
9. Modified by jbell at Thu Jun 16 16:50:13 CDT 2005
10. Modified by mrasch at Mon Jun 20 07:34:08 CDT 2005
11. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
12. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
13. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
14. Modified by mrasch at Fri Jul 01 11:04:00 CDT 2005
15. Question Reviewed by mrasch at Fri Jul 15 16:31:39 CDT 2005

16. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
17. Modified by mrasch at Tue Jul 26 12:14:06 CDT 2005
18. Modified by mrasch at Wed Jul 27 07:48:01 CDT 2005
19. Modified by mrasch at Wed Jul 27 12:27:53 CDT 2005
20. Question Reviewed by mellis at Thu Aug 04 07:23:42 CDT 2005
21. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
22. Modified by mrasch at Tue Aug 09 10:45:04 CDT 2005
23. Question Reviewed by mellis at Tue Aug 09 11:59:48 CDT 2005

Comments:

EB QUESTION: 3 (1.0 Points)

With RHR A in standby, how does a complete loss of Division I 125 VDC affect placing RHR A in Shutdown Cooling (SDC) mode of operation?

A. All necessary component manipulations can be performed at the Remote Shutdown Panel.

B. The RHR A pump can be started from the Control Room. The SDC Return Valve E12-F053A will require local operation.

C. All necessary component manipulations can be performed in the Main Control Room

D. The RHR A pump can be started from the breaker cubicle. The SDC Return Valve E12-F053A can be manipulated from the Control Room.

Answer: D

Question Comments: RHR A Pump to be operated anywhere other than the circuit breaker requires DC Control power. The E12-F053A control power is 120 VAC and is available for control room operation. Based on these conditions Answer D is the only correct answer. TIER 1 GROUP 1 This is a NEW question. CFR 41.7/41.8

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00834

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-P4100 Objective: 12.6
2. CourseID: GLP-OPS-L1100 Objective: 8a

KA References:

1. 295004 AA1.02 Systems necessary to assure safe plant shutdown [3.8/4.1]
2. 295004 AA1.03 AC electrical distribution [3.4/3.6]

References:

1. E-1181

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. E12: Residual Heat Removal System
2. L11: Plant DC Electical System

Categories:

1. Systems

Task References:

Question Last Revised By: MikeRasch at Fri Jul 29 15:00:26 CDT 2005
Question History:

1. Created by tharrelso at Thu Apr 21 09:43:45 CDT 2005
2. Modified by mrasch at Tue May 10 13:54:14 CDT 2005
3. Modified by mrasch at Tue May 24 10:10:51 CDT 2005
4. Question Reviewed by mellis at Tue May 31 14:57:01 CDT 2005
5. Modified by mrasch at Mon Jun 20 07:29:14 CDT 2005
6. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
7. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
8. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

9. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
10. Modified by mrasch at Wed Jul 27 16:59:35 CDT 2005
11. Modified by mrasch at Thu Jul 28 06:54:35 CDT 2005
12. Modified by tharrelso at Fri Jul 29 06:28:31 CDT 2005
13. Modified by mrasch at Fri Jul 29 15:00:26 CDT 2005
14. Question Reviewed by mellis at Thu Aug 04 07:23:44 CDT 2005
15. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 4 (1.0 Points)

The plant is operating at rated conditions when a turbine trip occurs.

Immediately following the transient, the reactor water level shrink associated with the reactor pressure change will:

A. cause reactor water level to initially drop, then recover to a value lower than the initial level.

B. not be seen because of the sharp drop in steam flow.

C. not be seen because of setpoint setdown.

D. cause reactor water level to rapidly drop and then recover to it's initial level.

Answer: A

Question Comments: Answer A is CORRECT because reactor water level will initially drop as the rise in reactor pressure collapses voids. When the bypass valves open to combat the reactor pressure rise, voids reform and reactor water level recovers. However, because of the setpoint setdown feature, reactor water level stabilizes at a new value lower than is normal for an operating reactor. Answer B is INCORRECT because the sharp drop in steam flow following a reactor scram reduces the void content in the reactor, resulting in a drop in reactor water level. Answer C is INCORRECT because the setpoint setdown feature, by design, will change reactor water level following a scram. Answer D is INCORRECT because reactor water level will initially drop, but setpoint setdown is designed to recover reactor water level to a value lower than the level normal for an operating reactor. This is a NEW question. TIER 1 GROUP 1 CFR 41.3/41.5/41.7

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00835

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GG-1-LP-RO-EP02 Objective: 7a

KA References:

1. 295005 AK1.03 Pressure effects on reactor level [3.5/3.7]

References:

1. GGNS PSTG, Appendix B Pages B-6-27 and 28
2. 02-S-01-27 Step 6.6.8.k
3. Simulator and plant operating data

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Shift Technical Advisor Training Program

Systems:

1. B21: Nuclear Boiler System
2. N19: Condensate System
3. N21: Feedwater System
4. N32: EHC Control System

Categories:

1. Fundamentals
2. Off Normal Event Procedures
3. Operating Experience
4. Systems
5. STA Training

Task References:

Question Last Revised By: Tommy Harrelson at Fri Jul 29 10:28:23 CDT 2005
Question History:

1. Created by tharrelso at Thu Apr 21 12:17:26 CDT 2005
2. Modified by tharrelso at Thu Apr 21 13:52:50 CDT 2005
3. Modified by mrasch at Tue May 10 13:58:26 CDT 2005
4. Question Reviewed by mellis at Tue May 31 14:57:02 CDT 2005
5. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
7. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
8. Modified by tharrelso at Thu Jul 28 12:59:40 CDT 2005
9. Modified by tharrelso at Fri Jul 29 06:29:46 CDT 2005
10. Modified by tharrelso at Fri Jul 29 10:28:23 CDT 2005
11. Question Reviewed by mrasch at Thu Aug 04 07:21:41 CDT 2005
12. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 5 (1.0 Points)

In which one of the following Scram Reports can it be determined that Shutdown Margin is assured without Reactor Engineering assistance?

A. The Mode Switch is in Shutdown, reactor power is dropping, all control rods indicate full in except four control rods indicate position 08.

B. The Mode Switch is in Shutdown, reactor power is dropping, all control rods indicate full in except forty-nine control rods indicate position 02.

C. The Mode Switch is in Shutdown, APRMs indicate downscale, NO control rod position indication is available.

D.

The Mode Switch is in Shutdown, reactor power is dropping, all control rods indicate full in except two peripheral control rods indicates position 04.

Answer: B

Question Comments: Answer A is incorrect because all control rods must be inserted to at least position 02 to ensure adequate shutdown margin. Answer B is CORRECT because all control rods are inserted to at least position 02 and by analysis, this ensures adequate shutdown margin. Answer C is incorrect because even though the APRMs indicate downscale, no control rod position indication is available. Answer D is incorrect because all control rods must be inserted to at least position 02 to ensure adequate shutdown margin. No distinction is made for lower worth peripheral control rods. This is a NEW question. TIER 1 GROUP 1 CFR 41.1/41.2/41.5/41.6/41.10/43.1/43.2/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00836

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 2

Objectives:

1. CourseID: GG-1-LP-RO-EP02 Objective: 11.0

KA References:

1. 295006 AK1.02 Shutdown margin [3.4/3.7]

References:

1. GE SIL No. 529 dated February 19, 1991
2. GE SIL No. 529 Supplement 1dated March 14, 1997
3. Technical Specification Bases 3.1.1

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. B13: Reactor Pressure Vessel
2. C11-2: Rod Control and Information System
3. C51-5: Average Power Range Nuclear Instrumentation System

Categories:

1. Emergency Procedure Training
2. FSAR
3. Mitigation of Core Damage
4. Systems
5. Technical Specifications

Task References:

Question Last Revised By: MikeRasch at Fri Jul 29 15:04:04 CDT 2005
Question History:

1. Created by tharrelso at Thu Apr 21 16:40:40 CDT 2005
2. Modified by mrasch at Tue May 10 14:00:12 CDT 2005
3. Modified by mrasch at Tue May 24 10:14:09 CDT 2005
4. Question Reviewed by mellis at Tue May 31 14:57:02 CDT 2005
5. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
7. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
8. Modified by tharrelso at Fri Jul 29 06:31:03 CDT 2005
9. Modified by tharrelso at Fri Jul 29 10:34:48 CDT 2005
10. Modified by mrasch at Fri Jul 29 15:04:04 CDT 2005
11. Question Reviewed by mellis at Thu Aug 04 07:23:45 CDT 2005
12. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 6 (1.0 Points)

A small fire with heavy smoke has forced the Operations crew to abandon the Control Room before the reactor could be shutdown.

Which of the following will scram the reactor from outside the Control Room?

A. Open breakers CB3A and CB7B.

B. Open breakers CB2A and CB8A.

C. Open breakers CB2B and CB8B.

D. Open breakers CB2A and CB2B.

Answer: D

Question Comments: Answer A is incorrect because this will have no effect on the RPS logic. Answer B is incorrect because opening only one division of breakers will result in only a half scram. Answer C is incorrect because opening only one division of breakers will result in only a half scram. Answer D is CORRECT because at least 1 RPS breaker in division (1 and 2) is required for full scram. This is a NEW question. TIER 1 GROUP 1 CFR 41.6/41.7/41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00837

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-OPS-C7100 Objective: 5.4
2. CourseID: GLP-OPS-C7100 Objective: 6.4
3. CourseID: GLP-OPS-C7100 Objective: 5.3

KA References:

1. 295016 AA1.01 RPS [3.8/3.9]

References:

1. 05-1-02-II-1 Step 3.3
2. E-1173-014

TrainingPrograms:

1. Nonlicensed Operator Training Program
2. Reactor Operator Training Program
3. Senior Reactor Operator Training Program
4. Licensed Operator Requalification Training Program

Systems:

1. C61: Remote Shutdown System
2. C71: Reactor Protection System

Categories:

1. Off Normal Event Procedures
2. Systems

Task References:

Question Last Revised By: Tommy Harrelson at Fri Jul 29 10:39:06 CDT 2005
Question History:

1. Created by tharrelso at Fri Apr 22 09:41:15 CDT 2005
2. Modified by mrasch at Tue May 10 14:01:49 CDT 2005
3. Modified by mrasch at Tue May 24 10:30:20 CDT 2005
4. Question Reviewed by mellis at Tue May 31 14:57:03 CDT 2005
5. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
7. Modified by mrasch at Fri Jul 01 12:10:03 CDT 2005
8. Question Reviewed by mrasch at Fri Jul 15 16:31:39 CDT 2005
9. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
10. Modified by tharrelso at Fri Jul 29 06:38:02 CDT 2005
11. Modified by tharrelso at Fri Jul 29 10:39:06 CDT 2005
12. Question Reviewed by mrasch at Thu Aug 04 07:21:06 CDT 2005
13. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 7 (1.0 Points)

The plant is operating at rated conditions when a rupture on the discharge header of the Component Cooling Water (CCW) System results in a complete loss of CCW.

Which of the following would be affected by the loss of CCW?

A. Control Rod Drive Hydraulic pump oil temperatures.

B.

Reactor Water Clean Up containment penetration temperatures.

C. Ambient Drywell temperatures.

D. RHR A pump seal temperatures.

Answer: A

Question Comments: The only component cooled by CCW listed in the answers is CRD pump lube oil. Drywell cooling is from Drywell Chilled Water. The RWCU penetration are cooled by Plant Chilled Water. RHR A pump seal is cooled by SSW A.Therefore A is the only correct answer. This is a NEW question. TIER 1 GROUP 1 CFR 41.4/41.5/41.7

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00838

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-OPS-ONEP Objective: 2.0
2. CourseID: GLP-OPS-P4200 Objective: 10

KA References:

1. 295018 AK1.01 Effects on component/system operations [3.5/3.6]

References:

1. 05-1-02-V-1 Section 4.0

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program

Systems:

1. B33: Reactor Recirculation System
2. C11-1A: CRD Hydraulic System

3. G33: Reactor Water Cleanup
4. P42: Component Cooling Water System

Categories:

1. Off Normal Event Procedures
2. Systems

Task References:

Question Last Revised By: MikeRasch at Tue Aug 09 10:47:33 CDT 2005
Question History:

1. Created by tharrelso at Fri Apr 22 11:04:55 CDT 2005
2. Modified by tharrelso at Fri Apr 22 13:11:21 CDT 2005
3. Modified by mrasch at Tue May 24 08:02:55 CDT 2005
4. Question Reviewed by mellis at Tue May 31 14:57:03 CDT 2005
5. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
7. Modified by mrasch at Fri Jul 01 15:38:53 CDT 2005
8. Question Reviewed by mrasch at Fri Jul 15 16:31:39 CDT 2005
9. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
10. Modified by mrasch at Thu Jul 28 07:29:23 CDT 2005
11. Modified by tharrelso at Fri Jul 29 09:52:03 CDT 2005
12. Modified by tharrelso at Fri Jul 29 09:53:32 CDT 2005
13. Modified by tharrelso at Fri Jul 29 09:59:16 CDT 2005
14. Modified by tharrelso at Fri Jul 29 10:01:13 CDT 2005
15. Question Reviewed by mrasch at Thu Aug 04 07:21:47 CDT 2005
16. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
17. Modified by mrasch at Tue Aug 09 10:47:33 CDT 2005
18. Question Reviewed by mellis at Tue Aug 09 11:59:49 CDT 2005

Comments:

EB QUESTION: 8 **(1.0 Points)**

Unit I Instrument Air Compressor is tagged out for maintenance and the Unit II Instrtument Air Compressor just failed.

With NO operator action the minimum Instrument Air pressure the operator would expect to see is approximately:

A. 80-85 psig

B. 90-95 psig

C. 100-105 psig

D. 110-115 psig

Answer: B

Question Comments: As instrument air pressure lowers, it will eventually reach the setpoint for auto opening of the service air to instrument air cross tie valve. Once this valve opens service air wil maintain instrument air pressure at approximately 90 psig per the loss of Instrument Air ONEP. Tier 1 Group 1 This is a NEW question. CFR 41.4/41.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00548

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-OPS-P5300 Objective: 4

KA References:

1. 295019 AA2.01 Instrument air system pressure [3.5/3.6]
2. 295019 AA1.01 Backup air supply [3.5/3.3]

References:

1. M-1068D
2. 05-1-02-V-9 section 3.2.3

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. P52: Service Air System
2. P53: Instrument Air System

Categories:

1. Systems

Task References:

Question Last Revised By: Tommy Harrelson at Fri Jul 29 10:41:54 CDT 2005
Question History:

1. No exam history found for this question during conversion. Converted from MSWord on Tue May 25 14:16:50 CDT 2004
2. Imported at Tue May 25 14:24:22 CDT 2004
3. Modified by tharrelso at Thu Apr 28 08:20:24 CDT 2005
4. Modified by mrasch at Tue May 24 11:11:40 CDT 2005
5. Question Reviewed by mellis at Tue May 31 14:57:00 CDT 2005
6. Modified by mrasch at Thu Jun 09 16:38:33 CDT 2005
7. Modified by jbell at Thu Jun 16 16:51:16 CDT 2005
8. Modified by mrasch at Mon Jun 20 07:45:31 CDT 2005
9. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
10. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
11. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
12. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
13. Modified by mrasch at Thu Jul 28 07:51:00 CDT 2005
14. Modified by mrasch at Thu Jul 28 08:28:06 CDT 2005
15. Modified by tharrelso at Fri Jul 29 10:41:54 CDT 2005
16. Question Reviewed by mrasch at Thu Aug 04 07:21:22 CDT 2005
17. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 9 (1.0 Points)

The plant is in Mode 4 and all forced circulation has been lost.

Which of the following is the reason for raising reactor water level to +82 inches per 05-1-02-III-1, Inadequate Decay Heat Removal?

A. This is the indicated level required to establish flow through open Safety Relief Valves to the Suppression Pool.

B. This is the indicated level required to establish alternate cooling using the Reactor Water Cleanup system.

C. This is the indicated level used in the FSAR accident analysis for the "Loss of Forced Circulation" Time to Boil Curves.

D. This is the indicated level required to allow natural circulation through the core and feedwater annulus.

Answer: D

Question Comments: Answer A is INCORRECT because the indicated level required to establish flow through open safety relief valves to the suppression pool is between +101 to 129 inches. Answer B is INCORRECT because the Reactor Water Cleanup System can be used as an alternate cooling method at normal water level. Answer C is INCORRECT because each Time to Boil Curve specifies the initial conditions of time after shutdown and RPV level for thier valid analysis. Answer D is CORRECT because this is the level at which sufficient driving head exists to establish natural circulation through the core. Tier 1 Group 1 MODIFIED idWRI 515 NRC Exam June 2001 Question 33 10CFR 41.5/41.10/41.14/42.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00078a

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-OPS-ONEP Objective: 2.0
2. CourseID: GLP-OPS-ONEP Objective: 20.0
3. CourseID: GLP-OPS-B1300 Objective: 5.12
4. CourseID: GLP-OPS-B3300 Objective: 42.0

KA References:

1. 295021 AK3.01 Raising reactor water level [3.3/3.4]

References:

1. 05-1-02-III-1, Inadequate Decay Heat Removal Step 3.3.2

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program

Systems:

1. B13: Reactor Pressure Vessel
2. B21: Nuclear Boiler System
3. E12: Residual Heat Removal System

Categories:

1. Off Normal Event Procedures
2. Systems

Task References:

Question Last Revised By: Tommy Harrelson at Fri Jul 29 10:43:27 CDT 2005
Question History:

1. Created by tharrelso at Mon Apr 25 10:18:40 CDT 2005
2. Created by tharrelso at Mon Apr 25 10:18:40 CDT 2005 from parent QuestionID GGNS-NRC-00078
3. Modified by mrasch at Tue May 24 08:31:58 CDT 2005
4. Question Reviewed by mellis at Tue May 31 14:56:57 CDT 2005
5. Modified by mrasch at Thu Jun 09 16:16:34 CDT 2005
6. Modified by mrasch at Thu Jun 09 16:37:24 CDT 2005
7. Modified by mrasch at Thu Jun 09 16:39:33 CDT 2005
8. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
9. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
10. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
11. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
12. Modified by tharrelso at Fri Jul 29 10:43:27 CDT 2005
13. Question Reviewed by mrasch at Thu Aug 04 07:20:36 CDT 2005
14. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 10 (1.0 Points)

In Mode 5 during a Shutdown Margin Demonstration (SMD), which one of the following would be the immediate concern in the event of inadvertent criticality?

A. Unplanned mode change

B. Fuel damage

C. Inadequate decay heat removal

D. High in-plant dose

Answer: D

Question Comments: Answer D is CORRECT because of the near proximity of workers in the drywell and on 208' of containment relative to the reactor core during a refueling outage. Localized criticality would raise dose rates in the drywell and possibly within line of sight of the core on elev. 208' of containment. Procedural guidance clearly prioritizes dose rate monitoring for an inadvertent criticality event. Answer A is INCORRECT since in Mode 5, a mode change is not made based on the effects of criticality, rising flux or coolant temperature, but only by Reactor Mode Switch position. Answer B is INCORRECT because any postulated criticality would be localized, not global, since control rod density would be ~99% for the SMD. Also, IRMs would generate a rod block and/or scram to limit the power excursion. Manual control rod insertion or system interlocks would mitigate the local power rise. Answer C is INCORRECT due to the lengthy amount of time it would take for the large volume of reactor coolant during Mode 5 to reach a temperature of significant concern. Other methods of decay heat removal could be placed in service during that time. Moreover, the control rod(s) withdrawn for the SDM would be quickly inserted to terminate the event. Tier 1 Group 1 This is a NEW Question. 10CFR 41.1/41.12/43.4/43.6

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00848

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-PROC Objective: 8.30

KA References:

1. 295023 AK1.03 Inadvertent criticality [3.7/4.0]

References:

1. 01-S-06-2 step 6.7.12
2. FSAR 15.4.1.1.4

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. J11: Reactor Fuel

Categories:

1. Off Normal Event Procedures
2. Refueling Training
3. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Tue Aug 09 10:50:27 CDT 2005
Question History:

1. Created by mrasch at Thu Jun 09 16:52:06 CDT 2005
2. Modified by mrasch at Mon Jun 20 07:47:42 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
7. Modified by tharrelso at Fri Jul 29 10:44:56 CDT 2005
8. Modified by mrasch at Fri Jul 29 15:07:01 CDT 2005

9. Question Reviewed by mellis at Thu Aug 04 07:23:47 CDT 2005
10. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
11. Modified by mrasch at Tue Aug 09 10:50:27 CDT 2005
12. Question Reviewed by mellis at Tue Aug 09 11:59:51 CDT 2005

Comments:

EB QUESTION: 11 (1.0 Points)

The plant is operating at 100% power when a spurious Division 1 ECCS initiation occurs.

Drywell pressure is rising due to Drywell Purge Compressor operation.

What must be reset first to perform a normal shutdown of Drywell Purge Compressor A?

A. Division 1 Load Shedding and Sequencing panel

B. Division 1 Combustible Gas Control System logic

C. Division 1 Containment/Drywell isolation logic

D. Division 1 ECCS logic

Answer: D

Question Comments: Answer A is incorrect because the LSS LOCA initiation signal originates from ECCS Logic. Answer B is incorrect because CGCS initiation logic LOCA signal originates from the ECCS logic. Answer C is incorrect because the CGCS and ECCS logics are separate from Containment isolation logic. Answer D is CORRECT because it is the top tier logic. Tier 1 Group 1 This is a NEW question. 10CFR 41.5/41.7

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00849

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 2
Objectives:

1. CourseID: GLP-OPS-E6100 Objective: 6.8, 19.0

KA References:

1. 295024 Generic 2.1.23: 3.9/4.0

References:

1. 17-S-06-5 Att. I pgs 3, 4; Att. II pgs 16,19,21,27,37,38
2. 04-1-01-E12-1 Att. IX
3. M-1077B

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. E12: Residual Heat Removal System
2. E61: Combustible Gas Control System
3. M71: Containment and Drywell Instrumentation System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Fri Jul 29 15:08:57 CDT 2005
Question History:

1. Created by mrasch at Thu Jun 09 17:02:20 CDT 2005
2. Modified by mrasch at Mon Jun 20 13:33:34 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
7. Modified by mrasch at Wed Jul 27 15:19:10 CDT 2005
8. Modified by tharrelso at Fri Jul 29 10:45:38 CDT 2005

9. Modified by tharrelso at Fri Jul 29 10:46:10 CDT 2005
10. Modified by tharrelso at Fri Jul 29 10:52:43 CDT 2005
11. Modified by mrasch at Fri Jul 29 15:08:57 CDT 2005
12. Question Reviewed by mellis at Thu Aug 04 07:23:47 CDT 2005
13. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 12 **(1.0 Points)**

Which one of the following describes the reason for automatic actuation of Alternate Rod Insertion (ARI) in response to rising reactor pressure?

A. ARI acts independently of RPS to insert control rods during conditions that should have resulted in a reactor scram.

B. ARI utilizes the RPS trip units to provide an alternate vent path for the scram air header as a backup to RPS.

C. ARI anticipates a reactor scram and assists RPS by establishing alternate vent paths for the scram air header.

D. ARI acts in conjunction with the Recirculation pumps to prevent exceeding the RPV pressure safety limit.

Answer: A

Question Comments: Answer A is CORRECT because ARI is an independent system to RPS which utilizes separate B21 trip units to open three independent vent paths in the event reactor pressure exceeds the 1064.7 psig scram setpoint. The ARI setpoint is at 1126 psig which is sufficiently above the scram setpoint to minimize inadvertent trips. Answer B is incorrect because ARI uses separate B21 trip units from those used by RPS. Answer C is incorrect because ARI actuation setpoint is higher than the RPS actuation. Answer D is incorrect because the Safety Relief Valves will prevent challenging the RPV Pressure Safety Limit. TIER 1 GROUP 1 This is a NEW question. CFR 41.5/41.6

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00690

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-OPS-C111A Objective: 3.13

KA References:

1. 295025 EK3.06 Alternate rod insertion: Plant-Specific [4.2/4.4]

References:

1. UFSAR 4.6.1.1.1.1.1(d)
2. UFSAR 4.6.1.1.2.4.5

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. C11-1A: CRD Hydraulic System

Categories:

1. Systems

Task References:

Question Last Revised By: Tommy Harrelson at Fri Jul 29 10:55:25 CDT 2005

Question History:

1. Used on NRC 2004 Exam
2. Converted from MSWord on Wed May 26 18:04:19 CDT 2004
3. Imported at Wed May 26 18:04:47 CDT 2004
4. Modified by tharrelso at Wed Apr 27 13:41:14 CDT 2005
5. Modified by mrasch at Tue May 10 13:47:54 CDT 2005
6. Question Reviewed by mellis at Tue May 31 14:57:00 CDT 2005
7. Modified by mrasch at Mon Jun 20 06:38:30 CDT 2005
8. Modified by mrasch at Mon Jun 20 10:58:22 CDT 2005
9. Modified by mrasch at Mon Jun 20 12:49:23 CDT 2005
10. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
11. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam

Date: 08/12/2005
12. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
13. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
14. Modified by mrasch at Wed Jul 27 11:30:12 CDT 2005
15. Modified by mrasch at Wed Jul 27 14:15:49 CDT 2005
16. Modified by tharrelso at Fri Jul 29 10:55:25 CDT 2005
17. Question Reviewed by mrasch at Thu Aug 04 07:21:28 CDT 2005
18. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 13 **(1.0 Points)**

Which one of the following is the basis to emergency depressurize the RPV prior to exceeding the Heat Capacity Temperature Limit (HCTL) of the Suppression Pool?

A. The emergency depressurization adds negative reactivity to the core and thereby reduces the total energy addition into the containment.

B. Prevent introduction of energy into the suppression pool at a rate that exceeds the energy removal rate of plant systems.

C. Emergency depressurizing the RPV prior to exceeding HCTL ensures the ECCS pumps will maintain NPSH if required to operate.

D. Prevent exceeding containment pressure and temperature limits in the event emergency depressurization is required.

Answer: D

Question Comments: HCTL requires emergency depressurization of the RPV in the event conditions are in the UNSAFE region to protect equipment in the pressure suppression chamber (Containment) from the maximum temperature and pressure capability per the PSTGs. Therefore Answer D is the only correct answer. Answer B is incorrect because it says when the reactor is still pressurized. Tier 1 Group 1 This is a NEW question. 10CFR55. 41.9

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00840

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 3

Objectives:

1. CourseID: GG-1-LP-RO-EP02A Objective: 2
2. CourseID: GG-1-LP-RO-EP02A Objective: 5

KA References:

1. 295026 EK3.01 Emergency/normal depressurization [3.8/4.1]

References:

1. GGNS PSTG App. B 17.5

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. M41-1: Containment

Categories:

1. Emergency Procedure Training
2. Continuing Training

Task References:

Question Last Revised By: Mickey Ellis at Fri Jul 29 12:17:42 CDT 2005
Question History:

1. Created by tharrelso at Mon May 02 07:18:33 CDT 2005
2. Modified by mrasch at Tue May 10 14:19:27 CDT 2005
3. Question Reviewed by mellis at Tue May 31 14:57:04 CDT 2005
4. Modified by jbell at Fri Jun 17 14:31:55 CDT 2005
5. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
7. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam

Date: 08/12/2005

8. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
9. Modified by mrasch at Thu Jul 28 09:50:15 CDT 2005
10. Modified by tharrelso at Fri Jul 29 06:53:01 CDT 2005
11. Modified by tharrelso at Fri Jul 29 07:46:38 CDT 2005
12. Modified by mellis at Fri Jul 29 12:13:17 CDT 2005
13. Modified by mellis at Fri Jul 29 12:16:15 CDT 2005
14. Modified by mellis at Fri Jul 29 12:17:42 CDT 2005
15. Question Reviewed by mrasch at Thu Aug 04 07:21:53 CDT 2005
16. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 14 **(1.0 Points)**

ATWS conditions exist.

Average Containment temperature is 187°F and slowly rising.

Suppression Pool temperature is 147°F and rising.

Suppression Pool level is 20 feet and rising.

Which one of the following describes the action that should be taken to control Containment parameters?

A. Vent the Containment

B. Emergency depressurize the reactor

C. Operate all containment coolers utilizing EP Attachment 7

D. Initiate Containment Spray

Answer: B

Question Comments: Answer A is incorrect since the vent path for EP attachment 14 is isolated and cannot be opened under the listed conditions.Answer B is correct since EP-3 step 28 requires emergency depressurization if containment temperature cannot be restored below 185°F. Containment temperature cannot be restored below 185°F since P71 is isolated to the containment under these conditions, so normal containment cooling is unavailable. Answer C is incorrect since operation of containment coolers without P71 as a heat sink would be ineffective. Answer D is incorrect since EP-3 Step 29 requires emergency depressurization and any other action would be inappropriate. Tier 1 Group 1 This is a NEW question. 10 CFR 41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00844

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 2

Objectives:

1. CourseID: GG-1-LP-RO-EP03 Objective: 3

KA References:

1. 295027 EA1.03 Emergency depressurization: Mark-III [3.5/3.8]

References:

1. PSTG B 7-19
2. 05-S-01-EP-3 Step 29

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. E12: Residual Heat Removal System
2. M41-1: Containment

Categories:

1. Emergency Procedure Training

Task References:

Question Last Revised By: Mickey Ellis at Fri Jul 29 12:18:13 CDT 2005
Question History:

1. Created by mrasch at Mon May 16 15:27:06 CDT 2005
2. Question Reviewed by mellis at Tue May 31 14:57:04 CDT 2005
3. Modified by mrasch at Fri Jun 10 07:59:51 CDT 2005
4. Modified by mrasch at Mon Jun 20 13:35:23 CDT 2005
5. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
7. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
8. Modified by mrasch at Mon Jul 11 08:11:54 CDT 2005
9. Question Reviewed by mrasch at Fri Jul 15 16:31:39 CDT 2005
10. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
11. Modified by tharrelso at Wed Jul 27 13:39:49 CDT 2005
12. Modified by tharrelso at Wed Jul 27 13:44:08 CDT 2005
13. Modified by tharrelso at Fri Jul 29 07:08:25 CDT 2005
14. Modified by mellis at Fri Jul 29 12:18:13 CDT 2005
15. Question Reviewed by mrasch at Thu Aug 04 07:21:57 CDT 2005
16. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 15 (1.0 Points)

A LOCA occurred, the following conditions exist:

Drywell temperature at the 166 foot elevation is 200 degrees F.

Reactor Recirc Pumps are off.

Reactor pressure is 600 psig.

Containment temperature at the 139 foot elevation is 95 degrees F.

If the above conditions remain stable, the Wide Range Reactor Water Level instrument:

CAUTION 1 of EP-2 is provided.

A. is limited to use below -130 inches indicated level but will indicate higher than actual water level.

B. can be used as long as it is on scale but will indicate higher than actual water level.

C. is limited to use above -130 inches indicated level but will indicate higher than actual water level.

D. can be used as long as it is on scale but will indicate lower than actual water level.

Answer: B

Question Comments: Answer A is incorrect because since the RPV Saturation Curve is still safe, the instrument is allowed full use as long as the instrument tracks per the Operations Philosphy. Answer B is CORRECT because the elevated temperature of the reference leg will cause the instrument to read higher than actual and Operations Philosophy allows its use. Answer C is incorrect because Operations Philosophy allows full use of the instrument as long as it tracks even with the elevated reference leg temperatures. Answer D is incorrect because indicated level will be higher than acutal level. Tier 1 Group 1 This is a NEW question. 10CFR 41.7/41.10/43.5

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00845

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 4

Objectives:

1. CourseID: GG-1-LP-RO-EP02 Objective: 3
2. CourseID: GLP-OPS-PROC Objective: 59.3

KA References:

1. 295028 EK1.02 Equipment environmental qualification [2.9/3.1]

References:

1. 05-S-01-EP-2 Caution 1
2. 02-S-01-27 section 6.1.8

TrainingPrograms:

1. Reactor Operator Training Program

2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. B21: Nuclear Boiler System

Categories:

1. Emergency Procedure Training

Task References:

Question Last Revised By: Mickey Ellis at Fri Jul 29 12:18:58 CDT 2005
Question History:

1. Created by mrasch at Mon May 16 15:36:22 CDT 2005
2. Modified by mrasch at Mon May 16 15:54:53 CDT 2005
3. Question Reviewed by mellis at Tue May 31 14:57:05 CDT 2005
4. Modified by mrasch at Fri Jun 10 08:04:08 CDT 2005
5. Modified by mrasch at Mon Jun 13 13:12:39 CDT 2005
6. Modified by mrasch at Thu Jun 16 12:51:16 CDT 2005
7. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
8. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
9. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
10. Modified by mrasch at Mon Jul 11 08:46:21 CDT 2005
11. Question Reviewed by mrasch at Fri Jul 15 16:31:39 CDT 2005
12. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
13. Modified by mrasch at Wed Jul 27 13:21:06 CDT 2005
14. Modified by mrasch at Wed Jul 27 13:30:51 CDT 2005
15. Modified by mrasch at Wed Jul 27 14:16:09 CDT 2005
16. Modified by mellis at Fri Jul 29 12:18:58 CDT 2005
17. Question Reviewed by mrasch at Thu Aug 04 07:22:03 CDT 2005
18. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 16 **(1.0 Points)**

The reactor is at rated conditions.

An RO is performing the daily Technical Specification surveillance rounds.

A Suppression Pool Limiting Condition for Operation (LCO) would require entry per the technical specifications if:

A. One narrow range Suppression Pool level recorder pen on H13-P870 is stuck.

B. Suppression Pool Average Temperature is 93 degrees F.

C. Suppression Pool Water Level is 18 feet 2 inches.

D. Suppression Pool Water Level is 18 feet 8 inches.

Answer: C

Question Comments: TS 3.6.2.1 Suppression Pool Average Temperature limit is 95 degrees F. TS 3.6.2.2 Suppression Pool Level must be between 18 feet 4 1/2 inches and 18 feet 9 3/4 inches. TR3.6.2.2 for Suppression Pool Level Alarms and TS 3.3.3.1 Post Accident monitoring are Instrumentation Specifications not affecting the operability of the Suppression Pool itself. Therefore Answer C is the only correct answer. Tier 1 Group 1 This is a NEW question. 10 CFR 41.9/41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00846

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 3

Objectives:

1. CourseID: GLP-OPS-M4101 Objective: 11

KA References:

1. 295030
2. GENERIC 2.2.12 Knowledge of surveillance procedures [3.0/3.4]

References:

1. TS3.3.3.1
2. TS3.6.2.1

3. TS3.6.2.2
4. TR3.6.2.2

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. M41-1: Containment

Categories:

1. Administrative Requirements
2. Systems
3. Technical Specifications

Task References:

Question Last Revised By: MikeRasch at Tue Aug 09 10:51:21 CDT 2005
Question History:

1. Created by mrasch at Mon May 16 15:53:01 CDT 2005
2. Question Reviewed by mellis at Tue May 31 14:57:05 CDT 2005
3. Modified by mrasch at Fri Jun 10 08:09:11 CDT 2005
4. Modified by mrasch at Mon Jun 20 13:37:44 CDT 2005
5. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
7. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
8. Modified by mrasch at Tue Jul 12 15:41:48 CDT 2005
9. Question Reviewed by mrasch at Fri Jul 15 16:31:39 CDT 2005
10. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
11. Modified by mrasch at Thu Jul 28 08:26:37 CDT 2005
12. Modified by tharrelso at Fri Jul 29 07:14:44 CDT 2005
13. Modified by tharrelso at Fri Jul 29 07:15:16 CDT 2005
14. Modified by mellis at Fri Jul 29 12:19:43 CDT 2005
15. Question Reviewed by mrasch at Thu Aug 04 07:22:07 CDT 2005
16. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
17. Modified by mrasch at Tue Aug 09 10:51:21 CDT 2005
18. Question Reviewed by mellis at Tue Aug 09 11:59:49 CDT 2005

Comments:

EB QUESTION: 17 (1.0 Points)

Following a reactor scram on high Drywell pressure, RPV water level initially dropped to - 20 inches on wide range.

HPCS automatically initiated.

As level was being restored, the HS for the HPCS Pump was stopped and the HPCS Injection Valve E22-F004 was taken to close.

RPV level is now +60 inches Narrow Range and Drywell pressure is 1.5 psig.

In this situation, if the HPCS INIT RESET pushbutton is depressed, the:

A. HPCS pump will auto start immediately but the HPCS Injection Valve E22-F004 will have to be opened using the handswitch.

B. HPCS pump will auto start immediately and the HPCS Injection Valve E22-F004 will automatically open.

C. HPCS pump will auto start after reactor water level drops below Level 2.

D. HPCS pump will auto start after reactor water level drops below Level 8.

Answer: C

Question Comments: Answer A is incorrect because depressing HPCS INIT RESET would reset the initiation signal and override high drywell pressure. HPCS pump would not auto start until -41.6 in. level was reached. Answer B is incorrect because the HPCS HI LVL RESET push button does not bypass high level but only resets the seal in when level falls below 53.5 in.. Answer C is CORRECT because the High Drywell pressure signal is bypassed such that HPCS is reset to a standby condition and will auto initiate on level 2. Answer D is incorrect because depressing HPCS INIT RESET would reset the initiation signal and override high drywell pressure the system will remain in standby until level 2 is reached. Tier 1 Group 1 This is a NEW question. 10CFR 41.7/41.8

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00847

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-E2200 Objective: 9.3
2. CourseID: GLP-OPS-E2200 Objective: 9.4
3. CourseID: GLP-OPS-E2200 Objective: 20
4. CourseID: GLP-OPS-E2200 Objective: 21

KA References:

1. 295031 EA1.04 High pressure core spray: Plant-Specific [4.3/4.2]

References:

1. E-1183-03
2. E-1183-23
3. E-1188-19

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. E22-1: High Pressure Core Spray System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Tue Aug 09 16:10:27 CDT 2005
Question History:

1. Created by mrasch at Mon May 16 16:08:02 CDT 2005
2. Question Reviewed by mellis at Tue May 31 14:57:06 CDT 2005
3. Modified by mrasch at Thu Jun 09 17:22:58 CDT 2005
4. Modified by jbell at Thu Jun 16 16:52:57 CDT 2005
5. Modified by mrasch at Mon Jun 20 13:38:53 CDT 2005
6. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
7. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam

Date: 08/12/2005
8. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
9. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
10. Modified by mrasch at Thu Jul 28 08:44:45 CDT 2005
11. Modified by tharrelso at Fri Jul 29 07:21:21 CDT 2005
12. Modified by mellis at Fri Jul 29 12:21:05 CDT 2005
13. Question Reviewed by mrasch at Thu Aug 04 07:22:11 CDT 2005
14. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
15. Modified by mrasch at Tue Aug 09 10:52:52 CDT 2005
16. Question Reviewed by mellis at Tue Aug 09 11:59:50 CDT 2005
17. Modified by mrasch at Tue Aug 09 16:10:27 CDT 2005
18. Question Reviewed by mellis at Tue Aug 09 16:14:20 CDT 2005

Comments:

EB QUESTION: 18 (1.0 Points)

The plant has experienced a transient and currently:

Reactor power stabilized at 8% following the reactor scram.

MSIVs are open.

RPV water level is - 20 inches and lowering.

Offsite power is available.

In this situation, the EOPs direct the operators to defeat the low level auto closure of the MSIVs.

One of the bases for this direction is to:

A. facilitate a reactor plant cooldown to 200 degrees F.

B. prevent the requirement of injection of Standby Liquid Control.

C. keep the main condenser available to reduce the probability of exceeding the HCTL.

D. keep the Turbine Bypass Valves available in the event a rapid depressurization is required.

Answer: C

Question Comments: The MSIVs are kept open in order to keep the Main Condenser available as a heat sink and maintain Condensate and Feedwater available. This reduces the heat input into the containment and minimizes the probability of exceeding the HCTL. This makes Answer C correct. Tier 1 Group 1 This is a NEW question. 10CFR 41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00850

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GG-1-LP-RO-EP02A Objective: 5

KA References:

1. 295037 EK3.06 Maintaining heat sinks external to the containment [3.8/4.1]

References:

1. EP-2A STEP 40
2. PSTG pg B-14-13

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

Categories:

1. Emergency Procedure Training
2. Continuing Training

Task References:

Question Last Revised By: Tommy Harrelson at Fri Jul 29 07:22:09 CDT 2005
Question History:

1. Created by mrasch at Fri Jun 10 08:18:07 CDT 2005
2. Modified by jbell at Thu Jun 16 16:54:03 CDT 2005
3. Modified by mrasch at Mon Jun 20 07:58:06 CDT 2005
4. Modified by mrasch at Mon Jun 20 13:40:07 CDT 2005
5. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
7. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
8. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
9. Modified by mrasch at Thu Jul 28 09:08:20 CDT 2005
10. Modified by tharrelso at Fri Jul 29 07:22:09 CDT 2005
11. Question Reviewed by mrasch at Thu Aug 04 07:22:21 CDT 2005
12. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 19 (1.0 Points)

A radioactive liquid leak has formed a puddle inside the protected area fence.

Eventually, the puddle flowed under the protected area fence, into a storm drain, and then into a waterway.

This would be considered a liquid effluent release when the radioactive water:

A. initially started to leak.

B. crossed the protected area fence.

C. entered the storm drain.

D. entered the waterway.

Answer: B

Question Comments: Answer B is correct because because the protected area fence is considered to be the boundary between “onsite” and “offsite” with respect to releases. Answer A is incorrect because it is not transgressed outside the protected area fence. is Answers C, and D are incorrect because the areas listed in those answers are well beyond the protected area fence, which would be crossed by the liquid first, before reaching the other listed locations. Tier 1 Group 1 This is a NEW question. 10CFR 41.10/41.13/43.4/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00851

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-OPS-TS001 Objective: 34

KA References:

1. 295038 EA2.01 Off-site [3.3/4.3]

References:

1. TS6.11.1

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

Categories:

1. Technical Specifications
2. Continuing Training

Task References:

Question Last Revised By: Tommy Harrelson at Fri Jul 29 10:15:30 CDT 2005
Question History:

1. Created by mrasch at Fri Jun 10 08:24:53 CDT 2005
2. Modified by mrasch at Mon Jun 20 13:42:54 CDT 2005

3. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
7. Modified by mrasch at Thu Jul 28 09:14:56 CDT 2005
8. Modified by tharrelso at Fri Jul 29 10:15:30 CDT 2005
9. Question Reviewed by mrasch at Thu Aug 04 07:22:25 CDT 2005
10. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 20 (1.0 Points)

Which one of the following operator actions is required in relation to a fire inside the protected area?

A. If a fire is reported in the Division 3 Diesel Generator room, start the Division 3 Diesel Generator Room Outside Air Fan from 1H13-P870.

B. If a fire occurs in the Upper Cable Spreading Room, always place Transfer Switch for Lockout Transfer Relay C61-HSS-M150 at 1H22-P152 to ON.

C. Before manning the Remote Shutdown Panel due to a fire in the Main Control Room, defeat the Division 3 Switchgear Room CO_2 system.

D. If a fire occurs in the Main Control Room, secure the Control Building Fan Coil Unit, Z17-B002.

Answer: C

Question Comments: Answer A is incorrect because the fan in the room with the fire is required to be stopped by 04-1-01-P81-1 step 3.14. 04-1-01-P81-1 step 3.14 Answer B is incorrect because it is not required by 05-1-02-II-1 step 3.5.1 for a fire outside the main control room. Answer C is correct because it is required by 05-1-02-II-1 step 3.4.1. Answer D is incorrect because the control room is not an area listed in 10-S-03-2 step 6.2.2f requiring Z17-B002 to be secured, since that does not directly serve the control room. Tier 1 Group 1 This is a NEW question. 10 CFR 41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00852

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-OPS-PROC Objective: 67.3

KA References:

1. 600000 AK3.04 Actions contained in the abnormal procedure for plantfire on site [2.8/3.4]

References:

1. 04-1-01-P81-1 step 3.14
2. 05-1-02-II-1 Steps 3.4.1; 3.5.1
3. 10-S-03-2 Step 6.2.2.f

TrainingPrograms:

1. Nonlicensed Operator Training Program
2. Reactor Operator Training Program
3. Senior Reactor Operator Training Program
4. Licensed Operator Requalification Training Program

Systems:

1. P64: Fire Water Protection System

Categories:

1. Administrative Requirements
2. Emergency Plan Training
3. Off Normal Event Procedures
4. Continuing Training

Task References:

Question Last Revised By: Tommy Harrelson at Fri Jul 29 10:09:05 CDT 2005
Question History:

1. Created by mrasch at Fri Jun 10 08:33:06 CDT 2005
2. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
3. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
4. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
6. Modified by tharrelso at Thu Jul 28 08:02:53 CDT 2005
7. Modified by tharrelso at Fri Jul 29 10:09:05 CDT 2005
8. Question Reviewed by mrasch at Thu Aug 04 07:22:28 CDT 2005
9. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 21 **(1.0 Points)**

The plant is operating at 20% power when a loss of condenser vacuum occurred due to air inleakage.

What will be the first cause of a direct reactor scram as condenser vacuum slowly lowers?

A. Main Turbine Trip

B. Low Reactor water level

C. High Reactor Water Level

D. Group 1 isolation

Answer: B

Question Comments: Answer A is incorrect because the Main Turbine Trip Scram signals are bypassed. Answer B is CORRECT because the reactor feed pump turbine trip will cause reactor level to drop causing a Level 3 Scram signal. Answer C is incorrect because the reactor will drop when the RFPTs trip. Answer D is incorrect because the reactor will scram on low water level well before the MSIV closure setpoint of 9”Hg vacuum is reached. Tier 1 Group 2 This is a NEW question. 10 CFR

41.4/41.5/41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00854

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-OPS-ONEP Objective: 39.0
2. CourseID: GLP-OPS-C7100 Objective: 9; 10

KA References:

1. 295002 AK2.01 RPS [3.5/3.5]

References:

1. 05-1-02-V-8 Section 5.0
2. Tech Spec Bases B3.3.1.1 Functions 9; 10

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. C71: Reactor Protection System
2. N62: Condenser Air Removal System

Categories:

1. Off Normal Event Procedures
2. Systems
3. Technical Specifications
4. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Wed Jul 27 15:25:51 CDT 2005
Question History:

1. Created by mrasch at Fri Jun 10 08:55:35 CDT 2005
2. Modified by mrasch at Mon Jun 20 13:43:36 CDT 2005

3. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
7. Modified by mrasch at Wed Jul 27 15:25:51 CDT 2005
8. Question Reviewed by mellis at Thu Aug 04 07:23:48 CDT 2005
9. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 22 **(1.0 Points)**

The plant was at the end of the current operating cycle following a 400 day run when the plant was scrammed due to bus 15AA lockout.

SRVs are being used for reactor pressure control.

The time estimated to restore bus 15AA is 16 hours.

Which one of the following should be performed to control reactor pressure?

A. Install nitrogen bottles on the Instrument Air header supply to the ADS air receivers in the Auxiliary Building.

B. Bypass the MSIVs and bleed reactor pressure to the Main Condenser through the Main Steam drains from the control room.

C. Allow Low-Low Set function of the SRVs to maintain pressure by cycling only two SRVs.

D. Install air jumpers in the Auxiliary Building to open Instrument Air Isolation valves restoring Instrument Air to the SRV receivers .

Answer: A

Question Comments: This task is performed by Non-Licensed Operators and Mechanics in the Auxiliary Building to supply air (nitrogen) to the ADS valves for reactor pressure control. Answer A is CORRECT since the estimated time to restore bus 15AA and restore instrument air to supply SRVs exceeds the 6 hour limit in the referenced plant procedures. This installs Nitrogen bottles to ADS receivers allowing them to be used for pressure control. Answer B is incorrect because the loss of bus 15AA prevents opening the Secondary Containment Isolation Air Operated Valves. Answer C is incorrect because the air supply for the Non-ADS SRVs would be exhausted after a short period of time. Answer D is incorrect because would require work instructions to control installation of air jumpers and defeat secondary containment. GGNS Scram April 2003 Tier 1 Group 2 This is a NEW question. 10 CFR 41.4/41.7/41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00855

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-OPS-ONEP Objective: 2.0
2. CourseID: GG-1-LP-RO-EP02A Objective: 5; 6

KA References:

1. GENERIC 2.4.35 Knowledge of local auxiliary operator tasks during emergency operations [3.3/3.5]
2. 295007

References:

1. 05-S-01-EP-2 Att 7 Step 2.4
2. EP-2A Steps 40; 54
3. 05-1-02-V-9 Step 3.12
4. 05-1-02-I-4 Step 3.2.4

TrainingPrograms:

1. Nonlicensed Operator Training Program
2. Reactor Operator Training Program
3. Senior Reactor Operator Training Program
4. Licensed Operator Requalification Training Program

Systems:

1. E22-2: Automatic Depressurization System
2. P53: Instrument Air System

Categories:

1. Emergency Procedure Training
2. Off Normal Event Procedures
3. Systems
4. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Wed Jul 27 16:32:16 CDT 2005
Question History:

1. Created by mrasch at Fri Jun 10 09:04:56 CDT 2005
2. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
3. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
4. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
5. Modified by mrasch at Mon Jul 11 09:39:57 CDT 2005
6. Modified by mrasch at Wed Jul 13 09:05:11 CDT 2005
7. Modified by mrasch at Wed Jul 13 10:19:52 CDT 2005
8. Question Reviewed by mrasch at Fri Jul 15 16:31:39 CDT 2005
9. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
10. Modified by mrasch at Tue Jul 26 13:33:20 CDT 2005
11. Modified by mrasch at Tue Jul 26 15:41:35 CDT 2005
12. Modified by mrasch at Wed Jul 27 16:32:16 CDT 2005
13. Question Reviewed by mellis at Thu Aug 04 07:23:49 CDT 2005
14. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 23 (1.0 Points)

A three (3) rod out ATWS condition exists with MSIVs closed.

Instrument Air is in normal lineup to the Containment and Drywell.

Which one of the following describes the use of SRVs to control reactor pressure?

A. Allow Low-Low Set to cycle under these conditions.

B. If there is only one leaking/weeping SRV, as designated by a colored key, only that SRV should be used.

C. Only non-ADS SRVs should be used to preserve ADS Air Receiver pressure, and they should be rotated.

D. Only ADS SRVs should be used, and they should be rotated.

Answer: C

Question Comments: Answer A is incorrect because low-low set is not allowed during an ATWS, as defined by being in EP-2A. Answer B is incorrect because use of only 1 SRV is limited to situations when suppression pool cooling is in service for pool circulation. Answer C is correct because non-ADS valves are specifically preferred when instrument air is available and rotation is required since Suppression Pool Cooling is not in service. Answer D is incorrect because ADS valves are specifically preferred only when instrument air is unavailable. Tier 1 Group 2 This is a NEW question. 10 CFR 41.3/41.7/41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00856

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-PROC Objective: 59.3
2. CourseID: GLP-OPS-B1300 Objective: 14.1; 14.2

KA References:

1. 295013 AA2.02 Localized heating/stratification [3.2/3.5]

References:

1. 04-1-01-B21-1 Step 4.2.2c
2. 02-S-01-27 Step 6.2.4

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. E22-2: Automatic Depressurization System

Categories:

1. Emergency Procedure Training
2. Off Normal Event Procedures
3. Systems
4. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Tue Aug 09 10:55:30 CDT 2005
Question History:

1. Created by mrasch at Fri Jun 10 09:16:04 CDT 2005
2. Modified by jbell at Thu Jun 16 16:57:45 CDT 2005
3. Modified by mrasch at Mon Jun 20 13:45:28 CDT 2005
4. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
5. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
7. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
8. Modified by mrasch at Thu Jul 28 13:58:05 CDT 2005
9. Modified by mellis at Fri Jul 29 12:36:41 CDT 2005
10. Question Reviewed by mrasch at Thu Aug 04 07:22:42 CDT 2005
11. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
12. Modified by mrasch at Tue Aug 09 10:55:30 CDT 2005
13. Question Reviewed by mellis at Tue Aug 09 11:59:51 CDT 2005

Comments:

EB QUESTION: 24 (1.0 Points)

The plant is in startup with reactor pressure at 80 psig.

RCIC has developed a steam leak within the RCIC room that cannot be isolated.

The CRS has ordered a reactor scram due to elevated RCIC room temperature.

Which one of the following is the reason for the reactor scram for these conditions?

These actions prevent:

A. An excessive release of radioactive material.

B. Exceeding any applicable TRM operability limits.

C. Failure of safe shutdown equipment and allows personnel access to the RCIC room.

D. Contaminating the Auxiliary Building and manually starting the Standby Gas Treatment system.

Answer: C

Question Comments: Answer A is incorrect because there is no evidence of fuel failure and therefor no source for gross amounts of radioactive material. Answer B is incorrect because EP-4 does not require entering EP-2 to effect a scram until the maximum safe temperature, 212°F, is reached. 185°F is the operating limit, only. Answer C is correct because 212°F is the maximum safe temperature for the RCIC room, and with an unisolable leak, a system that cannot be isolated from the RPV is discharging outside primary containment. Per EP-4 step 14, EP-2 should be entered, and it will require manual scram (EP-2 step 3). This will limit the heat added to the area preventing a challenge to equipment in the area and allowing personnel access. Answer D is incorrect since only one area is affected, and temperature would have to exceed maximum safe levels in at least 2 areas before contaminating the Auxiliary Building would becom an issue. Tier 1 Group 2 This is a NEW question. 10 CFR 41.5/41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00857

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory **Difficulty Rating:** 2

Objectives:

1. CourseID: GG-1-LP-RO-EP04 Objective: 7

KA References:

1. 295032 EK3.02 Reactor SCRAM [3.6/3.8]

References:

1. EP-4 Steps 9, 10, 13, 14
2. 03-1-01-1 Step 6.2.6f
3. PSTG Appendix B 16.14

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. E51: Reactor Core Isolation Cooling System

Categories:

1. Emergency Procedure Training
2. Continuing Training

Task References:

Question Last Revised By: Mickey Ellis at Fri Jul 29 12:38:04 CDT 2005
Question History:

1. Created by mrasch at Fri Jun 10 09:25:18 CDT 2005
2. Modified by mrasch at Mon Jun 20 13:46:09 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Modified by mrasch at Wed Jul 13 10:16:02 CDT 2005
7. Question Reviewed by mrasch at Fri Jul 15 16:31:39 CDT 2005
8. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
9. Modified by tharrelso at Wed Jul 27 14:07:56 CDT 2005
10. Modified by tharrelso at Fri Jul 29 07:25:06 CDT 2005
11. Modified by mellis at Fri Jul 29 12:38:04 CDT 2005
12. Question Reviewed by mrasch at Thu Aug 04 07:22:45 CDT 2005
13. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 25 (1.0 Points)

How will the Standby Gas Treatment System respond to an event resulting in the following radiation levels detected in ventilation systems:

	A	B	C	D
Containment Vent Exhaust Radiation Monitors	3.8 mr/hr	4.0 mr/hr	3.0 mr/hr	4.0 mr/hr
Fuel Handling Area Exhaust Radiation Monitors	2.0 mr/hr	4.0 mr/hr	3.0 mr/hr	4.0 mr/hr
Fuel Pool Sweep Exhaust Radiation Monitors	32 mr/hr	20 mr/hr	INOP	22 mr/hr

A. A and B will remain in standby.

B. A will automatically start and B will remain in standby.

C. B will automatically start and A will remain in standby.

D. A and B will automatically start.

Answer: A

Question Comments: Initiation setpoint for SGTS from Fuel Handling Area Vent Exh Rad Monitors is 3.6 mr/hr. Initiation setpoint for SGTS from Fuel Handling Area Pool Sweep Exh Rad Monitors is 30 mr/hr. SGTS does not receive an auto start from Containment/Drywell Vent Exhaust Radiation Monitors. The initiation logic requires Channels 'A' and 'D' to initiate SGTS 'A', or 'B' and 'C' to initiate SGTS 'B'. With D17K618C removed, a channel 'B' trip of FPS Rad Monitor (K618B) would start SGTS 'B'. Answer A is correct (and answers B,C, and D are incorrect) because D17K618B, D17K618D, D17K618A, and D17K617C do not reach their trip setpoints, so neither division completes a full logic initiation. Tier 1 Group 2 This is a NEW question. 10CFR 41.4/41.7

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00858

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-OPS-T4801 Objective: 8.6

KA References:

1. 295034 EK2.03 SBGT/FRVS: Plant-Specific [4.3/4.5]

References:

1. 17-S-06-5 Att II pages 35 and 36

TrainingPrograms:

1. Nonlicensed Operator Training Program
2. Reactor Operator Training Program
3. Senior Reactor Operator Training Program
4. Licensed Operator Requalification Training Program

Systems:

1. D17: Process Radiation Monitoring System
2. T48: Standby Gas Treatment System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: Tommy Harrelson at Fri Jul 29 07:42:21 CDT 2005
Question History:

1. Created by mrasch at Fri Jun 10 09:32:37 CDT 2005
2. Modified by mrasch at Mon Jun 20 08:07:53 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
7. Modified by mrasch at Wed Jul 27 13:33:00 CDT 2005
8. Modified by mrasch at Wed Jul 27 13:47:18 CDT 2005
9. Modified by mrasch at Wed Jul 27 13:52:29 CDT 2005
10. Modified by tharrelso at Fri Jul 29 07:26:52 CDT 2005
11. Modified by tharrelso at Fri Jul 29 07:40:57 CDT 2005
12. Modified by tharrelso at Fri Jul 29 07:42:21 CDT 2005
13. Question Reviewed by mrasch at Thu Aug 04 07:22:48 CDT 2005
14. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 26 (1.0 Points)

The plant is in a refueling outage when a fuel handling accident occurs.

Standby Gas Treatment System (SGTS) A is manually initiated.

on 1H13-P870, amber alarm ENCL BLDG NEG PRESS LO (2A-E3) is subsequently received and does not clear.

What is the operational implication of this alarm?

A. Greater personnel industrial safety hazard when entering or exiting the auxiliary building.

B. Possible damage to the enclosure building due to high pressure.

C. Possible excessive filtered leakage from secondary containment.

D. Possible excessive unmonitored leakage from secondary containment.

Answer: D

Question Comments: Answer A is incorrect because the alarm is indicative of a higher pressure in secondary containment (i.e. lower dp relative to outside secondary containment). A safety concern only exists when there is a higher dp, resulting in high forces on doors which could cause them to open quickly when unlatched. Answer B is incorrect because the alarm is indicative of a low dp with respect to outside, not a high dp which could cause excessive forces on enclosure building coverings. Answer C is incorrect because no amount of filtered leakage would be excessive. SGTS flow rates would be governed predominantly by the flow control circuit, so approximately the same flow rate through the SGTS filter train would exist after the 120 sec timer had expired. Answer D is correct because prevention of exfiltration could not be assured if pressure was higher than -0.25"wc. The given alarm occurs at -0.2"wc or higher pressure. That leakage would bypass the SGTS filter train and exhaust radiation monitoring system. Tier 1 Group 2 This is a NEW question. 10CFR 41.8/41.10/41.13/43.4/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00859

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-T4801 Objective: 2.0; 11.0
2. CourseID: GG-1-LP-RO-EP04 Objective: 6

KA References:

1. 295035 EK1.02 Radiation release [3.7/4.2]

References:

1. ARI 04-1-02-1H13-P870 2A-E3
2. EP-4 Bases pg B-8-2, 6

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. T48: Standby Gas Treatment System

Categories:

1. Emergency Procedure Training
2. Continuing Training

Task References:

Question Last Revised By: Mickey Ellis at Fri Jul 29 12:42:47 CDT 2005
Question History:

1. Created by mrasch at Fri Jun 10 09:40:21 CDT 2005
2. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
3. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
4. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
6. Modified by mellis at Fri Jul 29 12:39:22 CDT 2005
7. Modified by mellis at Fri Jul 29 12:42:47 CDT 2005
8. Question Reviewed by mrasch at Thu Aug 04 07:22:51 CDT 2005
9. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 27 **(1.0 Points)**

RHR A room sump pumps are each in AUTO with their selector switch in ALTERNATE.

RHR A is operating in Shutdown Cooling when a 10 gpm leak develops inside RHR A room.

RHR Room A Floor Drain Sump Pump A, P45-C013A, started and pumped down almost to the low sump level but its breaker tripped due to a fault.

How will this affect subsequent operation of the other sump pump in RHR A room, P45-C013B?

A. P45-C013B will alternate starting on high-high and high sump levels, each time pumping until low sump level is reached.

B. P45-C013B will come on at high-high sump level and off at high sump level and continue to cycle that way.

C. P45-C013B will come on at high-high sump level and off at low sump level and continue to cycle that way.

D. P45-C013B will come on at high sump level and off at low sump level one time. It will not automatically start after that.

Answer: D

Question Comments: If the High alarm has cleared on the initial start from pump A then the B pump will start on a subsequent high level alarm but only once since the logic thinks pump A should start next. Therefore Answer D is correct. Tier 1 Group 2 This is a NEW question. 10 CFR 41.4/41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00860

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-OPS-P4500 Objective: 7.1; 11.1; 11.2

KA References:

1. 295036 EA1.01 Secondary containment equipment and floor drainsystems [3.2/3.3]

References:

1. 04-1-01-P45-2 Steps 3.5; Note 4.2.2a
2. M-1094A
3. M-1098B

TrainingPrograms:

1. Nonlicensed Operator Training Program
2. Reactor Operator Training Program
3. Senior Reactor Operator Training Program
4. Licensed Operator Requalification Training Program

Systems:

1. P45: Floor and Equipment Drain System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Fri Jul 29 07:46:20 CDT 2005
Question History:

1. Created by mrasch at Fri Jun 10 09:46:57 CDT 2005
2. Modified by mrasch at Fri Jun 10 13:23:02 CDT 2005
3. Modified by mrasch at Mon Jun 20 08:13:30 CDT 2005
4. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
5. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
7. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
8. Modified by mrasch at Fri Jul 29 07:46:20 CDT 2005
9. Question Reviewed by mellis at Thu Aug 04 07:23:50 CDT 2005
10. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 28 (1.0 Points)

A plant shutdown is in progress and conditions are as follows:

Reactor pressure 30 psig

Reactor water level 36 inches Narrow Range

Both reactor recirculation pumps are shutdown.

Shutdown cooling was lost due to a failure of the Shutdown Cooling common suction isolation valve E12-F008.

Which of the following would provide an alternate method to ensure core cooling is maintained?

A. Allowing natural circulation to initiate.

B. Initiating RCIC.

C. Placing one loop of RHR in the LPCI mode and opening SRVs.

D. Starting one of the Reactor Recirculation pumps.

Answer: C

Question Comments: Answer A is incorrect because level is not high enough to allow natural circulation. Answer B is incorrect because Reactor pressure is below RCIC isolation setpoint making RCIC unavailable. Answer C is CORRECT since these actions are consistent with guidance in ONEP Inadequate Decay Heat Removal 05-1-02-III-1. Answer D is incorrect because the restoration of a Recirc pump does not result in heat removal. Tier 2 Group 1 This is a NEW question. 10CFR41.7

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00861

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-E1200 Objective: 9.7

KA References:

1. 203000 K5.02 Core cooling methods [3.5/3.7]

References:

1. E-1181-67

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. E12: Residual Heat Removal System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Tue Aug 09 11:12:23 CDT 2005
Question History:

1. Created by mrasch at Fri Jun 10 09:53:23 CDT 2005
2. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
3. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
4. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
6. Modified by tharrelso at Fri Jul 29 06:26:05 CDT 2005
7. Modified by tharrelso at Fri Jul 29 08:34:42 CDT 2005
8. Modified by mellis at Fri Jul 29 12:44:03 CDT 2005
9. Question Reviewed by mrasch at Thu Aug 04 07:23:02 CDT 2005
10. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
11. Modified by mrasch at Tue Aug 09 11:12:23 CDT 2005
12. Question Reviewed by mellis at Tue Aug 09 11:59:52 CDT 2005

Comments:

EB QUESTION: 29 **(1.0 Points)**

The plant is in Mode 5.

RHR B has been placed in Shutdown Cooling with suction from Recirc Loop B, returning to the reactor via RHR B SHUTDN CLG RTN TO FW valve E12-F053B.

RHR B ADHRS MODE TRIP ENABLE hand switch on 1H13-P618 is in NORMAL.

Which one of the following will cause RHR B pump to trip?

A. Opening the power supply to RHR SHUTDN CLG INBD SUCT VLV E12-F009.

B. RHR SHUTDN CLG OTBD SUCT VLV E12-F008 were to close to 50%.

C. RHR B FPC ASSIST SUCTION VLV E12-F066B were to open to 50%.

D. RHR B ADHRS MODE TRIP ENABLE hand switch on 1H13-P618 were to be placed in ADHRS position.

Answer: B

Question Comments: The purpose of the RHR pump suction path interlocks is to trip the RHR pump when no fully open suction path exists. Answer A is incorrect because the trip circuit for RHR pump 'B' looks directly at the limit switch contact for E12F009. It does not rely on control power for E12F009, but uses RHR B/C DC logic power. Answer B is correct because the trip circuit for RHR pump 'B' looks directly at the limit switch for E12F008. RHR pump B trip coil is energized when the valve closes to "not full open" position, ~95% open. Answer C is incorrect because either a suction path from the reactor, one from the suppression pool, or one from the fuel pool has to exist for RHR pump to remain running. In this case, a suction path exists from the reactor, so isolation of the suction path from the fuel pool does not energize the pump trip coil. Answer D is incorrect because placing RHR B ADHRS MODE TRIP ENABLE hand switch in ADHRS position only removes E12F066B as a permissive to run RHR pump B. As long as the suction path from the reactor is aligned fully open, the pump will continue to run. Tier 2 Group 1 This is a NEW question. 10CFR 41.7

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00862

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-OPS-E1200 Objective: 8.1

KA References:

1. 205000 K4.04 Adequate pump NPSH [2.6/2.6]

References:

1. E-1160-09; 10
2. E-1181-05; 44; 68

TrainingPrograms:

1. Nonlicensed Operator Training Program
2. Reactor Operator Training Program
3. Senior Reactor Operator Training Program
4. Licensed Operator Requalification Training Program

Systems:

1. E12: Residual Heat Removal System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: Tommy Harrelson at Fri Jul 29 10:18:44 CDT 2005
Question History:

1. Created by mrasch at Fri Jun 10 09:58:56 CDT 2005
2. Modified by mrasch at Mon Jun 20 08:15:20 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
7. Modified by tharrelso at Fri Jul 29 08:36:03 CDT 2005
8. Modified by tharrelso at Fri Jul 29 10:14:03 CDT 2005
9. Modified by tharrelso at Fri Jul 29 10:17:45 CDT 2005
10. Modified by tharrelso at Fri Jul 29 10:18:44 CDT 2005
11. Question Reviewed by mrasch at Thu Aug 04 07:23:06 CDT 2005
12. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 30 (1.0 Points)

Due to a blown fuse, logic power for RHR A has been lost.

In this situation, if a LOCA were to occur, the LPCS pump:

A. would auto start but RHR Pump A would have to be manually started.

B. would have to be manually started, and RHR Pump A is unavailable.

C. and RHR Pump A are both available but each would have to be manually started.

D. and RHR Pump A are both unavailable.

Answer: C

Question Comments: The loss of logic power for RHR A will preclude an auto start of either pump. However,, control power is still available so both pumps could be manually started if desired. This makes Answer C the only correct answer. Tier 2 Group 1 This is a NEW question. 10CFR 41.7

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00863

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-E2100 Objective: 9.2; 10.2; 16.0

KA References:

1. 209001 A4.01 Core spray pump [3.8/3.6]

References:

1. ARI 04-1-02-1H13-P601 20A-H6

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. E21: Low Pressure Core Spray System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Tue Aug 09 11:12:54 CDT 2005
Question History:

1. Created by mrasch at Fri Jun 10 10:04:09 CDT 2005
2. Modified by mrasch at Fri Jun 10 12:40:43 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
7. Modified by mrasch at Thu Jul 28 09:36:27 CDT 2005
8. Question Reviewed by mellis at Thu Aug 04 07:23:50 CDT 2005
9. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
10. Modified by mrasch at Tue Aug 09 11:12:54 CDT 2005
11. Question Reviewed by mellis at Tue Aug 09 11:59:53 CDT 2005

Comments:

EB QUESTION: 31 (1.0 Points)

The plant is in Mode 1.

RHR C Jockey Pump has been shutdown for 5 days.

How will this affect High Pressure Core Spray (HPCS) pump suction automatic re-alignment to Suppression Pool if actual Suppression Pool Level rises?

A. HPCS suction will automatically re-align when Suppression Pool level reaches the setpoint.

B. HPCS suction will automatically re-align at a Suppression Pool level that is actually higher than the setpoint.

C. HPCS suction will automatically re-align at a Suppression Pool level that is actually lower than the setpoint.

D. HPCS suction will automatically re-align on Low Condensate Storage Tank level only.

Answer: A

Question Comments: Answer A is correct. RHR C jockey Pump does not affect HPCS Suppression Pool Level Instrumentation. Tier 2 Group 1 This is a NEW question. 10 CFR 41.7/41.8

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00864

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-E2201 Objective: 22.0

KA References:

1. 209002 A4.09 Suppression pool level: BWR-5,6 [3.4/3.5]

References:

1. 04-1-01-E22-1 Sections 6.3; 6.4

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. E22-1: High Pressure Core Spray System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Tue Aug 09 11:15:05 CDT 2005
Question History:

1. Created by mrasch at Fri Jun 10 10:11:05 CDT 2005
2. Modified by mrasch at Fri Jun 10 12:40:11 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
7. Modified by mrasch at Wed Jul 27 16:01:51 CDT 2005
8. Modified by mellis at Fri Jul 29 12:47:02 CDT 2005
9. Question Reviewed by mrasch at Thu Aug 04 07:28:59 CDT 2005
10. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
11. Modified by mrasch at Tue Aug 09 11:15:05 CDT 2005
12. Question Reviewed by mellis at Tue Aug 09 11:59:53 CDT 2005

Comments:

EB QUESTION: 32 **(1.0 Points)**

Why does E22-F004, HPCS Injection Valve, automatically close on high reactor water level?

A. Prevent over pressurizing the Reactor Pressure Vessel (RPV).

B. Prevent excessive cool down rates for RPV internals.

C.

Prevent overflow into the main steam lines.

D. Prevent Main Turbine and Reactor Feed Pump trips.

Answer: C

Question Comments: The FSAR design function and Tech Spec bases for the High Reactor Water Level isolation of E22-F004 HPCS Injection Valve is to prevent water introduction into the Main Steam Lines. Answer A is incorrect because this is not the reason listed in Tech Spec bases and FSAR. In this case, SRVs would prevent over-pressurization. Answer B is incorrect because this is not the reason listed in Tech Spec bases and FSAR. The design function of HPCS is to provide adequate core cooling, not to control cool down rates. Answer C is CORRECT because this is the reason listed in Tech Spec bases and is assumed in the accident analysis. Answer D is incorrect because this is not the reason listed in Tech Spec bases and FSAR, and would not prevent reaching level 9 in all cases due the time required for E22 F004 to stroke closed. Tier 2 Group 1 This is a NEW question. 10CFR 41.8/43.2

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00865

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-E2201 Objective: 5.5; 9.4; 17.0

KA References:

1. GENERIC 2.1.28 Knowledge of the purpose and function of major system components and controls [3.2/3.3]
2. 209002

References:

1. Tech Sec Bases B3.3.5.1 function 3c
2. FSAR 7.3.1.1.1.3.6

TrainingPrograms:

1. Nonlicensed Operator Training Program
2. Reactor Operator Training Program
3. Senior Reactor Operator Training Program
4. Licensed Operator Requalification Training Program

Systems:

1. E22-1: High Pressure Core Spray System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Tue Aug 09 11:15:45 CDT 2005
Question History:

1. Created by mrasch at Fri Jun 10 10:16:45 CDT 2005
2. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
3. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
4. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
5. Modified by mrasch at Mon Jul 11 09:58:23 CDT 2005
6. Modified by mrasch at Wed Jul 13 10:59:56 CDT 2005
7. Question Reviewed by mrasch at Fri Jul 15 16:31:39 CDT 2005
8. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
9. Modified by mrasch at Tue Jul 26 13:37:59 CDT 2005
10. Modified by mrasch at Tue Jul 26 13:39:15 CDT 2005
11. Modified by tharrelso at Fri Jul 29 08:37:33 CDT 2005
12. Modified by mellis at Fri Jul 29 12:48:12 CDT 2005
13. Question Reviewed by mrasch at Thu Aug 04 07:29:00 CDT 2005
14. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
15. Modified by mrasch at Tue Aug 09 11:15:45 CDT 2005
16. Question Reviewed by mellis at Tue Aug 09 11:59:54 CDT 2005

Comments:

EB QUESTION: 33 (1.0 Points)

Both squib valves have failed to fire on a Standby Liquid Control (SLC) initiation.

What affect will this have on the ability of SLC to inject and what acceptable alternate means of SLC injection should be utilized?

A. SLC will inject at a much lower injection rate. Inject Sodium Pentaborate directly into piping downstream of the Condensate demineralizers such as the Reactor Feedwater pump suction.

B. SLC will NOT inject. Dump required amounts of boric acid and borax into the Refueling Water Storage Tank (RWST) and inject with the Condensate Transfer Pumps.

C. SLC will inject at a much lower injection rate. Dump required amounts of boric acid and borax into the Condensate Storage Tank (CST) and inject with the Condensate Transfer pumps.

D. SLC will NOT inject. Dump required amounts of boric acid and borax into the Condensate Storage Tank (CST) and mix the contents then inject with RCIC.

Answer: D

Question Comments: If neither squib valve fires then SLC will not inject requiring the use of EP-2A Attachment 28 Alternate SLC. This involves adding 5000 lbs. each of Boric Acid and Borax to the CST and mixing and injecting with either the RCIC pump or HPCS. Therefore Answer D is Correct. Tier 2 Group 1 This is a NEW question. 10 CFR 41.7/41.8/41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00866

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-C4100 Objective: 6; 10.1; 10.4; 12

KA References:

1. 211000 A2.02 Failure of explosive valve to fire [3.6/3.9]

References:

1. 04-1-01-C41-1 ATT. VI
2. 05-S-01-EP-2 Attachment 28

TrainingPrograms:

1. Nonlicensed Operator Training Program
2. Reactor Operator Training Program
3. Senior Reactor Operator Training Program
4. Licensed Operator Requalification Training Program

Systems:

1. C41: Standby Liquid Control System

Categories:

1. Emergency Procedure Training
2. Systems
3. Continuing Training

Task References:

Question Last Revised By: Mickey Ellis at Fri Jul 29 12:54:23 CDT 2005
Question History:

1. Created by mrasch at Fri Jun 10 10:22:40 CDT 2005
2. Modified by mrasch at Mon Jun 20 08:18:48 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Modified by mrasch at Mon Jul 11 11:17:18 CDT 2005
7. Modified by mrasch at Thu Jul 14 08:50:36 CDT 2005
8. Question Reviewed by mrasch at Fri Jul 15 16:31:39 CDT 2005
9. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
10. Modified by mrasch at Thu Jul 28 14:25:00 CDT 2005
11. Modified by mellis at Fri Jul 29 12:54:23 CDT 2005
12. Question Reviewed by mrasch at Thu Aug 04 07:29:01 CDT 2005
13. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 34 (1.0 Points)

A plant startup is in progress at 36% of rated thermal power.

The A Main Bypass Control Valve (BCV) has failed open and startup has been suspended.

The Baxter Wilson 500 KV line into GGNS trips causing breakers J5228 and J5232 to open.

The Turbine Initial Pressure Control (IPC) system responds as designed, except the A Main Bypass Control Valve remains open.

How would the plant automatically respond to this event and what operator actions should be taken?

A. The Reactor Recirculation Pumps would transfer to slow speed and the reactor would scram on high water level. Enter the Reactor Scram ONEP.

B. The Reactor Recirculation Pumps would transfer to slow speed and the reactor would scram due to the Turbine Control Valve closure. Enter the Reactor Scram ONEP.

C. The Reactor Recirculation Pumps would remain in fast speed and the reactor would scram on high neutron flux. Enter the Reactor Scram ONEP.

D. The Reactor Recirculation Pumps would remain in fast speed and the reactor would continue to operate. Suspend plant startup until the BCV A failure is corrected.

Answer: D

Question Comments: Though at 36% power, turbine 1st stage inlet pressure was equivalent to ~26% power with one bypass valve full open (~13%) worth). TSV/TCV scrams bypassed until 40% power based on Turbine 1st stage inlet pressure. Therefore, EOC/RPT and scram from TCV/TSV closure is automatically bypassed. B and C bypass valves can accommodate steam flow for the remaining 26% power, so reactor pressure is relatively unaffected. Answer A and B are incorrect because EOC/RPT is bypassed under given conditions. Answer C is incorrect because bypass valves are fast acting and can accommodate all steam flow for the specified power level, so the pressure/flux transient is minimal, well below what would cause a high flux scram. Answer D is CORRECT for the reasons previously stated. Tier 2 Group 1 This is a NEW question. 10CFR 41.5/41.6/41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00867

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-OPS-N3202 Objective: 16
2. CourseID: GLP-OPS-B3300 Objective: 27.5
3. CourseID: GLP-OPS-C7100 Objective: 10

KA References:

1. 212000 A2.12 Main turbine stop control valve closure [4.0/4.1]

References:

1. FSAR 7.2.1.1.4.4.2

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. B33: Reactor Recirculation System
2. C71: Reactor Protection System
3. N32: EHC Control System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: Tommy Harrelson at Wed Jul 27 14:40:48 CDT 2005
Question History:

1. Created by mrasch at Fri Jun 10 10:30:09 CDT 2005
2. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005

3. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
4. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
5. Modified by mrasch at Thu Jul 14 09:04:20 CDT 2005
6. Question Reviewed by mrasch at Fri Jul 15 16:31:39 CDT 2005
7. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
8. Modified by tharrelso at Wed Jul 27 14:40:48 CDT 2005
9. Question Reviewed by mellis at Thu Aug 04 07:24:12 CDT 2005
10. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 35 **(1.0 Points)**

Which of the following combinations of plant activities would be allowed by plant procedures?

A. Driving in IRM ‘A’ while I&C performs a surveillance for Scram Discharge Volume Water Level High channel B

B. Driving in IRM ‘A’ with the under-vessel service platform out of its standby position and the under-detector grid sections installed

C. Driving out IRM ‘A’ while driving out IRM ‘C’ in Mode 2

D. Driving out IRM ‘A’and IRM ‘B’ simultaneously in Mode 1

Answer: C

Question Comments: Answer A is incorrect because driving IRM might cause a Div 2 half scram during the Div 1 half scram surveillance, resulting in a full scram. Answer B is incorrect because this might cause damage to the IRM drive cables. Answer C is correct because both IRMs are Div 1, so the worst consequence for driving one IRM, a Div 1 half scram, is no worse than that for driving the two IRMs simulataneously. Answer D is incorrect because driving IRM ‘A’ might cause a Div 1 half scram, driving IRM ‘B’ might cause a Div 2 half scram, resulting in a full scram. Tier 2 Group 1 This is a NEW question. 10CFR 41.7/41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00868

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-OPS-C5102 Objective: 12.2

KA References:

1. 215003 K5.03 Changing detector position [3.0/3.1]

References:

1. 04-1-01-C51-1 sections 3.5, 3.7, cautions section 4.2.2

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. C51-2: Intermediate Range Nuclear Instrumentation System

Categories:

1. Administrative Requirements
2. Systems
3. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 08:22:45 CDT 2005

Question History:

1. Created by mrasch at Fri Jun 10 10:40:26 CDT 2005
2. Modified by mrasch at Mon Jun 20 08:22:45 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
7. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 36 **(1.0 Points)**

With the plant in Mode 2, which of the following will result in a control rod block?

A. SRM A is upscale and bypassed.

SRM E is NOT full in and reads 70 cps.

All IRMs are on Range 2.

B. SRMs A and E are INOP.

IRMs A and E are on Range 9.

C. SRM A is upscale and bypassed.

SRM B is NOT full in and reads 275 cps.

All IRMs are on range 2.

D. SRMs A and B are INOP.

IRMs A and B are on Range 9.

Answer: A

Question Comments: With SRM E not full in and less than 70 CPS a rod block will occur because SRM s are single coincidence logic and SRM E will generate a rod block. None of the other conditions would generate a control rod block, therefore Answer A is the only correct answer. Tier 2 Group 1 This is a NEW question. 10CFR 41.2/41.6

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00869

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-C5101 Objective: 8.2; 11.1

KA References:

1. GENERIC 2.2.33 Knowledge of control rod programming [2.5/2.9]
2. 215004

References:

1. 04-1-01-C51-1 Step 3.8
2. E-1171-20
3. FSAR 7.6.2.5.1.1

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. C11-2: Rod Control and Information System
2. C51-1: Source Range Nuclear Instrumentation System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: Mickey Ellis at Fri Jul 29 12:58:07 CDT 2005

Question History:

1. Created by mrasch at Fri Jun 10 10:53:52 CDT 2005
2. Modified by mrasch at Mon Jun 20 08:24:17 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

6. Modified by mrasch at Mon Jul 11 13:03:15 CDT 2005
7. Question Reviewed by mrasch at Fri Jul 15 16:31:39 CDT 2005
8. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
9. Modified by mrasch at Thu Jul 28 10:08:00 CDT 2005
10. Modified by mellis at Fri Jul 29 12:58:07 CDT 2005
11. Question Reviewed by mrasch at Thu Aug 04 07:29:02 CDT 2005
12. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 37 **(1.0 Points)**

The power supply for APRM D is:

A. 1Y87

B. 1Y88

C. 1Y95

D. 1Y96

Answer: C

Question Comments: per the referenced L62 SOI and C51 SOI 1Y95 Inverter is the power supply to APRM cabinet D. Tier 2 Group 1 This is a NEW question. 10 CFR 41.6

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00870

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-OPS-C5104 Objective: 11.3

KA References:

1. 215005 K2.02: 2.6/2.8 Knowledge of power supplies to APRM Channels.

References:

1. 04-1-01-L62-1 Att VI Table 3

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. C51-5: Average Power Range Nuclear Instrumentation System
2. L62: Uninterruptible Power Supply System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Thu Jul 28 10:14:55 CDT 2005
Question History:

1. Created by mrasch at Fri Jun 10 11:02:59 CDT 2005
2. Modified by mrasch at Fri Jun 10 13:40:49 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
7. Modified by mrasch at Thu Jul 28 10:14:55 CDT 2005
8. Question Reviewed by mellis at Thu Aug 04 07:24:20 CDT 2005
9. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 38 (1.0 Points)

The reactor is at 1000 psig and slowly lowering.

RCIC is in operation in automatic control.

In response to the RPV pressure lowering, RCIC discharge pressure will:

A. slowly rise due to the governor valve slowly opening.

B. slowly lower due to RCIC turbine speed slowly lowering.

C. remain constant due to RCIC speed slowly rising.

D. remain constant due to the RCIC throttle valve slowly opening.

Answer: B

Question Comments: As RPV pressure lowers, the flow controller will close down on the governor valve thereby reducing RCIC speed and discharge pressure making B the correct answer. This makes the other three distractors incorrect. Tier 2 Group 1 This is a NEW question. 10 CFR 41.4/41.7/41.14

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00871

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-OPS-E5100 Objective: 3.1; 8.17; 19.0

KA References:

1. 217000 A1.02 RCIC pressure [3.3/3.3]

References:

1. M-1083A
2. M-1085A

3. M-1077D

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. E12: Residual Heat Removal System
2. E51: Reactor Core Isolation Cooling System
3. N21: Feedwater System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Thu Jul 28 10:24:22 CDT 2005
Question History:

1. Created by mrasch at Fri Jun 10 11:17:17 CDT 2005
2. Modified by mrasch at Mon Jun 20 08:26:05 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
7. Modified by mrasch at Thu Jul 28 10:24:22 CDT 2005
8. Question Reviewed by mellis at Thu Aug 04 07:24:23 CDT 2005
9. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 39 **(1.0 Points)**

The power supply for Division 1 ADS Logic is Division 1:

A. RPS.

B. UPS.

C. 120 VAC.

D. 125 VDC.

Answer: D

Question Comments: The power supply to ADS Logic is from the Divisional DC power supply per the facility electrical drawings. Therefore Answer D is the only correct answer. Tier 2 Group 1 This is a NEW question. 10CFR 41.7

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00872

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-OPS-E2202 Objective: 9.1; 9.2; 10.2; 15.0; 19.3; 25.0; 27.0

KA References:

1. 218000 K2.01 ADS logic [3.1/3.3]
2. 218000 K6.06 D [3.4/3.6]

References:

1. E-1161-04

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. E22-2: Automatic Depressurization System

2. L11: Plant DC Electical System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Thu Jul 28 10:40:13 CDT 2005
Question History:

1. Created by mrasch at Fri Jun 10 11:34:17 CDT 2005
2. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
3. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
4. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
5. Modified by mrasch at Mon Jul 11 13:08:46 CDT 2005
6. Modified by mrasch at Thu Jul 14 09:18:07 CDT 2005
7. Question Reviewed by mrasch at Fri Jul 15 16:31:39 CDT 2005
8. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
9. Modified by mrasch at Thu Jul 28 10:40:13 CDT 2005
10. Question Reviewed by mellis at Thu Aug 04 07:24:26 CDT 2005
11. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 40 **(1.0 Points)**

A condenstate system transient occurred while at 100% power which caused reactor water level to drop to -50 inches wide range.

Level 2 Trip units for Groups 6, 7, 8, 10 and Auxiliary Building (B21-N682A and B21-N682B) failed to trip.

A large leak on Drywell Chilled Water piping inside containment has been reported.

Regarding building isolations only, what is the appropriate response for these conditions?

A. All Division 1 valves have failed to isolate. All Division 2 valves have isolated. Manually close all Division 1 valves that failed to isolate.

B. Only Division 1 and 2 primary containment valves have failed to isolate. All secondary containment valves have closed. Isolate all primary containment valves.

C. No primary or secondary containment valves have isolated. Close at least one valve in each penetration, except for Instrument Air and Plant Service Water.

D. No primary or secondary containment valves have isolated. Close at least one valve in each penetration, except for Drywell Chilled Water, Instrument Air and Plant Service Water.

Answer: C

Question Comments: Many isolations should have occurred due to low reactor water level, level 2. Operations Philosophy, 02-S-01-27, states in a failure to isolate situation, only one valve in each penetration that should be isolated needs to be shut, and to not isolate P53, P44, or P72 if those systems are intact and are going to be unisolated per the EPs. Answer A is incorrect because only one valve in each penetration is required to be closed. Answer B is incorrect because some isolation valves that failed are MOVs, and loss of air has no effect on them. Answer C is correct because it mirrors Operations Philosophy as stated above. P72 should be isolated because of a system breach. Answer D is correct because P72 should be isolated because of a system breach. Tier 2 Group 1 This is a NEW question. 10CFR 41.9/41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00873

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 3

Objectives:

1. CourseID: GLP-OPS-PROC Objective: 59.3

KA References:

1. 223002 A2.03 System logic failures [3.0/3.3]

References:

1. 02-S-01-27 Step 6.1.3
2. 17-S-06-5 Att I page 3; Att II 28, 30, 31, 32, 34, 35, 36

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. P44: Plant Service Water System
2. P53: Instrument Air System
3. P72: Drywell Chill Water System

Categories:

1. Administrative Requirements
2. Systems
3. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Fri Jul 29 12:28:02 CDT 2005
Question History:

1. Created by mrasch at Fri Jun 10 12:20:51 CDT 2005
2. Modified by mrasch at Mon Jun 20 08:27:42 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
7. Modified by mrasch at Thu Jul 28 17:46:39 CDT 2005
8. Modified by tharrelso at Fri Jul 29 08:52:04 CDT 2005
9. Modified by mrasch at Fri Jul 29 12:23:26 CDT 2005
10. Modified by mrasch at Fri Jul 29 12:28:02 CDT 2005
11. Question Reviewed by mellis at Thu Aug 04 07:24:29 CDT 2005
12. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 41 (1.0 Points)

In the event of a loss of Instrument Air to Containment, what is the minimum number of manual actuations the Automatic Depressurization System air system is designed to provide Safety Relief Valve B21-F051D over a 6 hour period?

A. 100

B. 50

C. 25

D. 10

Answer: A

Question Comments: Tier 2 Group 1 This is a NEW question. 10CFR 41.3/41.5/41.7

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00874

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-OPS-E2202 Objective: 18

KA References:

1. 239002 K4.09 Manual opening of the SRV [3.7/3.6]

References:

1. E-1161-13; 16

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. E22-2: Automatic Depressurization System
2. P53: Instrument Air System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: Tommy Harrelson at Fri Jul 29 10:23:47 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 07:37:50 CDT 2005
2. Modified by mrasch at Mon Jun 13 07:39:03 CDT 2005
3. Modified by mrasch at Mon Jun 20 08:29:28 CDT 2005
4. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
5. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
7. Modified by mrasch at Mon Jul 11 13:19:16 CDT 2005
8. Modified by mrasch at Thu Jul 14 10:04:14 CDT 2005
9. Question Reviewed by mrasch at Fri Jul 15 16:31:39 CDT 2005
10. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
11. Modified by mrasch at Wed Jul 27 16:06:08 CDT 2005
12. Modified by tharrelso at Fri Jul 29 10:23:47 CDT 2005
13. Question Reviewed by mellis at Thu Aug 04 07:24:32 CDT 2005
14. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 42 (1.0 Points)

The plant is at 90% power.

Annunciator RX LVL HI/LO is received on 1H13-P680.

You notice the output of the Master Level Controller is slowly trending down, the output of RFP A speed controller is slowly trending up, and the output of RFP B speed controller is slowly trending down.

Which of the following describes the cause of these indications and the operator response?

A. RFP A runout due to the Master Level Controller failing in automatic operation. Immediately insert a manual scram.

B. RFP B failing to the low speed stop due to the Master Level Controller failing in automatic operation. Take manual control of the Master Level Controller and control level 32 inches to 42 inches.

C. RFP A runout due to the RFP A speed control failing upscale. Attempt to take manual control of RFP A, and control level 32 inches to 42 inches. If it CANNOT be controlled manually, trip RFP A and verify a Recirc Flow Control Runback occurs.

D. RFP B failing to the low speed stop due to the RFP B speed control failing downscale. Attempt to take manual control of RFP B, and control level 32 inches to 42 inches. If it CANNOT be controlled manually, trip RFP B and verify a Recirc Flow Control Runback occurs.

Answer: C

Question Comments: The condition is indicative of failure upward of the RFP A speed controller. The master level controller and RFP B speed controller are responding to the increased inventory added by RFP A. Answer A is incorrect because The Master Controller output is responding to RPV level. The transient is stated to be slow, and scramming is not conservative if it can be avoided. Procedural guidance exists to avoid a scram. Answer B is incorrect because the master level controller controller is responding properly to the failure of RFP A speed controller. Answer C is CORRECT because ONEP 05-1-02-V-6 specifically directs this action for the given failure. Answer D is incorrect because RFP B controller is responding properly to the failure of A. Tier 2 Group 1 This is a NEW question. 10 CFR 41.4/41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00875

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-N2100 Objective: 17; 31.2; 37
2. CourseID: GLP-OPS-ONEP Objective: 37

KA References:

1. 259002 A2.04 RFP runout condition: Plant-Specific [3.0/3.1]

References:

1. 05-1-02-V-6 Step 2.2

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. C34: Feedwater Level Control System
2. N21: Feedwater System

Categories:

1. Off Normal Event Procedures
2. Systems
3. Continuing Training

Task References:

Question Last Revised By: Mickey Ellis at Fri Jul 29 12:30:52 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 07:45:26 CDT 2005
2. Modified by mrasch at Mon Jun 20 08:31:07 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Modified by mrasch at Thu Jul 14 09:35:26 CDT 2005
7. Question Reviewed by mrasch at Fri Jul 15 16:31:39 CDT 2005
8. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
9. Modified by mrasch at Tue Jul 26 14:16:14 CDT 2005
10. Modified by mrasch at Wed Jul 27 09:47:44 CDT 2005

11. Modified by mrasch at Wed Jul 27 12:40:35 CDT 2005
12. Modified by mrasch at Wed Jul 27 12:46:39 CDT 2005
13. Modified by mellis at Fri Jul 29 12:30:52 CDT 2005
14. Question Reviewed by mrasch at Thu Aug 04 07:29:03 CDT 2005
15. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 43 **(1.0 Points)**

The plant is at 100% power.

Which one of the following will cause the Feedwater THREE ELEMENT pushbutton DISABLED back light to illuminate on 1H13-P680-2C?

A. B Feedwater flow signal drifts up to 11 mlbm/hr.

B. The difference between highest and lowest Main Steam Line Flow signals diverges to 1 mlbm/hr.

C. The difference between Feedwater Flow A and B diverge to 0.6 mlbm/hr greater than normal.

D. Plant power reduction to 45% rated thermal power.

Answer: A

Question Comments: Answer A is correct because when a feedwater flow signal goes above 10.6 mlbm/hr it is a hard failure which disables three element control. Answer B is incorrect because Steam flow divergence is 1.6 mlbm/hr to disable three element control. Answer C is incorrect because feedwater flow divergence is 0.8 mlbm/hr to disable three element control. Answer D is incorrect because on a power decrease three element control is disabled at 30% power (4.95 mlbm/hr). Tier 2 Group 1 10CFR41.7/41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00233a

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-OPS-C3400 Objective: 6.3
2. CourseID: GLP-OPS-C3400 Objective: 5.3

KA References:

1. 295009 AA4.06: 3.1/3.2
2. 259002 A4.06 DP/Single/three element control selector switch:Plant-Specific [3.1/3.2]

References:

1. 04-1-02-1H13-P680 2A-C9

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. C34: Feedwater Level Control System
2. B21: Nuclear Boiler System

Categories:

1. Systems

Task References:

Question Last Revised By: Tommy Harrelson at Fri Jul 29 08:53:29 CDT 2005
Question History:

1. Created by tharrelso at Wed May 04 16:49:27 CDT 2005
2. Created by tharrelso at Wed May 04 16:49:27 CDT 2005 from parent QuestionID GGNS-NRC-00233
3. Modified by mrasch at Tue May 24 08:46:46 CDT 2005
4. Modified by mrasch at Tue May 24 10:05:45 CDT 2005
5. Question Reviewed by mellis at Tue May 31 14:56:58 CDT 2005
6. Modified by tharrelso at Tue Jun 07 15:36:27 CDT 2005
7. Modified by mrasch at Mon Jun 13 07:56:49 CDT 2005
8. Modified by mrasch at Mon Jun 13 08:03:40 CDT 2005
9. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005

10. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
11. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
12. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
13. Modified by mrasch at Thu Jul 28 15:15:56 CDT 2005
14. Modified by tharrelso at Fri Jul 29 08:53:29 CDT 2005
15. Question Reviewed by mrasch at Thu Aug 04 07:20:40 CDT 2005
16. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 44 (1.0 Points)

A planned move of radioactive material is planned on the refueling floor.

As a precaution, Standby Gas Treatment System (SGTS) A has been started and applicable parameters have stabilized.

Enclosure Building differential pressure is -0.6 inches wc and stable.

An operator is about to manually start Standby Gas Treatment system B.

How should the Steam Tunnel Outside Containment damper, T48-F005 respond?

A. T48-F005 will immediately throttle to its intermediate position.

B. T48-F005 will travel full open and throttle to its intermediate position after 90 seconds.

C. T48-F005 will travel full open and throttle to its intermediate position after 120 seconds.

D. T48-F005 will travel full open and throttle to its intermediate position after 120 seconds or Enclosure Building pressure reaches -.75 inches wc.

Answer: A

Question Comments: Answer A is correct because enclosure building pressure is less than -0.2 inches wc when SGTS B is initiated, so T48F005 goes to intermediate position right away. Answer B is incorrect because the T48F005 damper will not travel to it full open position before throttling. Answer C is incorrect because T48F005 would go to intermediate position right away instead of 120 sec later. Answer D is incorrect because T48F005 would go to intermediate position right away instead of 120 sec or -.75 inches wc in the enclosure building. These times are actually for flow control vane T48F500B. Tier 2 Group 1 This is a NEW question. 10CFR 41.4/41.13

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00876

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-OPS-T4801 Objective: 8.4; 8.5; 8.7

KA References:

1. 261000 A3.03 Valve operation [3.0/2.9]

References:

1. 04-1-01-T48-1 Step 5.2.1b NOTE

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. T48: Standby Gas Treatment System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: Mickey Ellis at Fri Jul 29 13:02:31 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 07:54:52 CDT 2005
2. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
3. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
4. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
6. Modified by tharrelso at Wed Jul 27 15:02:15 CDT 2005
7. Modified by mellis at Fri Jul 29 13:01:50 CDT 2005
8. Modified by mellis at Fri Jul 29 13:02:31 CDT 2005
9. Question Reviewed by mrasch at Thu Aug 04 07:29:04 CDT 2005
10. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 45 (1.0 Points)

The plant is operating in mode 1 with HPCS running for a surveillance with suction from the suppression pool.

A fire in the Division 3 Diesel Generator room de-energized bus 11DC.

How does this affect the operation of the HPCS Pump breaker 152-1702 and the HPCS Injection Valve E22-F004?

A. Pump breaker can be tripped using its control room hand switch but the injection valve must be operated locally.

B. Pump breaker must be tripped locally at the breaker and the injection valve must be operated locally.

C. Pump breaker can be tripped from the control room and the injection valve can be operated from the control room.

D. Pump breaker must be tripped locally at the breaker and the injection valve can be operated from the control room.

Answer: D

Question Comments: Answer A is incorrect because 11DC supplies control power for 152-1702, and E22F004 could be operated from the control room since it is AC. Answer B is incorrect because 11DC supplies control power for 152-1702, and E22F004 could be operated from the control room since it is AC. Answer C is incorrect because 11DC supplies control power for 152-1702. Answer D is correct because there is no control power to energize the trip coil for 152-1702, and E22F004 control power is AC power. Tier 2 Group 1 This is a NEW question. 10CFR 41.7

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00877

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-OPS-E2201 Objective: 13.2; 13.3

KA References:

1. 262001 K6.01 D [3.1/3.4]

References:

1. E-1183-03
2. E-1188-19

TrainingPrograms:

1. Nonlicensed Operator Training Program
2. Reactor Operator Training Program
3. Senior Reactor Operator Training Program
4. Licensed Operator Requalification Training Program

Systems:

1. E22-1: High Pressure Core Spray System
2. L11: Plant DC Electical System
3. R21: 4.16 KV AC Power System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Wed Jul 27 16:37:13 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 09:08:22 CDT 2005
2. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
3. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
4. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
6. Modified by mrasch at Wed Jul 27 14:10:33 CDT 2005
7. Modified by mrasch at Wed Jul 27 16:37:13 CDT 2005
8. Question Reviewed by mellis at Thu Aug 04 07:24:42 CDT 2005
9. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 46 (1.0 Points)

A LOCA occurred shortly after placing static inverter 1Y95 on its alternate power supply.

Fifteen minutes later Bus 16AB locked out.

Based on this information, the reactor operator should conclude:

A. 1Y80 is receiving power from its alternate supply.

B. 1Y95 is receiving power from its normal supply.

C. 1Y88 is receiving power from its normal supply.

D.

1Y87 is receiving power from its alternate supply.

Answer: C

Question Comments: 1Y95 was placed on its alternate power supply and does not have an auto re-transfer feature therefore when 16AB bus is locked out the loads for 1Y95 are lost. 1Y88 is also a division 2 powered inverter however it will remain on its DC Normal power supply when its alternate supply is lost. 1Y87 is division 1 powered and would have had a momentary loss of its alternate power source during the LOCA signal but remain on its normal power supply. Tier 2 Group 1 This is a NEW question. 10 CFR 41.4

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00878

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-OPS-L6200 Objective: 4.1; 4.2; 8; 9.1; 9.2; 10.1; 15

KA References:

1. 262002 K4.01 Transfer from preferred power to alternate powersupplies [3.1/3.4]

References:

1. 04-1-01-L62-1 Steps 3.4; 3.5 Attachment III

TrainingPrograms:

1. Nonlicensed Operator Training Program
2. Reactor Operator Training Program
3. Senior Reactor Operator Training Program
4. Licensed Operator Requalification Training Program

Systems:

1. L62: Uninterruptible Power Supply System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Thu Jul 28 10:52:29 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 09:17:16 CDT 2005
2. Modified by mrasch at Mon Jun 20 08:34:35 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
7. Modified by mrasch at Thu Jul 28 10:52:29 CDT 2005
8. Question Reviewed by mellis at Thu Aug 04 07:24:47 CDT 2005
9. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 47 (1.0 Points)

The plant DC system is in its normal lineup when a loss of Load Control Center 15BA6 occurs

What effect will this have on Division 1 battery voltage and specific gravity?

A. Both will remain stable.

B. Both will lower.

C. Voltage will remain constant and specific gravity will go down.

D. Voltage will lower and specific gravity will remain constant.

Answer: A

Question Comments: Answer A is correct because load is shared between chargers 1A4 and 1A5, and either charger is rated to maintain battery parameters by itself. Answer B is incorrect because with load sharing, 1A5 will pick up load automatically. Answers C and D are incorrect because charger 1A5 is 100% duty rated and will maintain battery parameters. Tier 2 Group 1 This is a NEW question. 10CFR 41.4

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00879

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-OPS-L1100 Objective: 2; 7; 16

KA References:

1. 263000 K4.01 Manual/ automatic transfers of control: Plant-Specific [3.1/3.4]

References:

1. 04-1-01-L11-1 Attachment IIIA
2. Tech Spec Bases B3.8.4

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. L11: Plant DC Electical System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Thu Jul 28 11:00:25 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 09:22:05 CDT 2005
2. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
3. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005

4. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
6. Modified by mrasch at Thu Jul 28 11:00:25 CDT 2005
7. Question Reviewed by mellis at Thu Aug 04 07:24:54 CDT 2005
8. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 48 **(1.0 Points)**

The plant is at 50% power due to grid disturbances.

16AB bus is separated from the grid with DG12 carrying the bus.

With respect to DG 12, if a valid LOCA were to occur in this electrical lineup, the DG 12 output breaker would:

A. open, load shedding would occur, and the output breaker would re-close automatically. No operator action is required.

B. remain closed and load shedding would occur, and loads are sequenced on. No operator action is required.

C. remain closed and load shedding would NOT occur. The operator would manually secure the unnecessary loads.

D. open, load shedding would occur, and the operator would have to re-close the output breaker.

Answer: B

Question Comments: With the Diesel carrying the bus by itself the diesel output breaker will remain closed but the Load Shedding and Sequencing system will shed the loads on the bus resequencing the loads per the LOCA sequencing process. Therefore Answer B is the only correct answer. Tier 2 Group 1 This is a NEW question. 10CFR 41.7

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00880

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-R2100 Objective: 11; 14; 34
2. CourseID: GLP-OPS-P7500 Objective: 26

KA References:

1. 264000 A2.10 LOCA [3.9/4.2]

References:

1. 04-1-01-P75-1 ATT V page 5
2. M-1077B
3. E-1109-24
4. E-1120-04

TrainingPrograms:

1. Nonlicensed Operator Training Program
2. Reactor Operator Training Program
3. Senior Reactor Operator Training Program
4. Licensed Operator Requalification Training Program

Systems:

1. B21: Nuclear Boiler System
2. P75: Div 1 and 2 Diesel Generator System
3. R21: 4.16 KV AC Power System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: Mickey Ellis at Fri Jul 29 13:05:01 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 09:28:08 CDT 2005
2. Modified by mrasch at Mon Jun 20 09:04:22 CDT 2005

3. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
7. Modified by mrasch at Thu Jul 28 11:10:46 CDT 2005
8. Modified by tharrelso at Fri Jul 29 08:56:44 CDT 2005
9. Modified by mellis at Fri Jul 29 13:05:01 CDT 2005
10. Question Reviewed by mrasch at Thu Aug 04 07:29:05 CDT 2005
11. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 49 (1.0 Points)

The plant is operating normally at 100% power when the normal supply breaker to bus 16AB trips open. As expected, DG 12 starts and comes up to 450 RPM. However, the Ready to Load light is not lit and the generator frequency and voltage meters are both reading downscale. The output breaker did not close and the LSS failure alarm has annunciated.

Based on these indications the operator would suspect:

A. generator frequency is low.

B. generator voltage is low.

C. the Parallel Reset Switch needs to be reset.

D. the lockout relay for bus 16AB has tripped.

Answer: B

Question The ready to load light is powered off the AC bus and will not light until

Comments: the DG output breaker closes. Because the engine is at 450 RPM the generator is turning at the electrical equivalent to 60 Hz. However, the frequency meter will not indicate a value if the generator has no voltage output. Therefore the immediate suspicion would be the generator field flash failed and the generator output voltage is zero. Tier 2 Group 1 This is a NEW question. 10CFR55.41 (7)

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00881

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 4

Objectives:

1. CourseID: GLP-OPS-P7500 Objective: 27

KA References:

1. 264000
2. GENERIC 2.4.48 5 EDG/Diagnose EDG status from control room indications. [3.5/3.8]

References:

1. ARI 04-1-02-1H13-P864 2A-H1

TrainingPrograms:

1. Nonlicensed Operator Training Program
2. Reactor Operator Training Program
3. Senior Reactor Operator Training Program
4. Licensed Operator Requalification Training Program

Systems:

1. P75: Div 1 and 2 Diesel Generator System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: Mickey Ellis at Fri Jul 29 13:09:20 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 09:37:34 CDT 2005
2. Modified by mrasch at Mon Jun 20 09:05:22 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
7. Modified by tharrelso at Fri Jul 29 08:04:47 CDT 2005
8. Modified by mellis at Fri Jul 29 13:09:20 CDT 2005
9. Question Reviewed by mrasch at Thu Aug 04 07:29:09 CDT 2005
10. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 50 **(1.0 Points)**

The plant is operating at 100% power when instrument air is lost to the Auxiliary Building.

Pressure is now approximately 60 psig, slowly lowering and the restoration efforts thus far have been unsuccessful.

If pressure continues to lower, the Auxiliary Building air bleed off valve (PV-F531) is set to automatically open at approximately:

A. 20 psig.

B. 30 psig.

C. 40 psig.

D. 50 psig.

Answer: B

Question Comments: The referenced lesson plan states the bleed off valve will open at approximately 30 psig, making Answer B the correct answer. Tier 2 Group 1 This is a NEW question. 10 CFR 41.7

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00882

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-P5300 Objective: 5.6; 14

KA References:

1. 300000 K3.01 Containment air system [2.7/2.9]

References:

TrainingPrograms:

1. Nonlicensed Operator Training Program
2. Reactor Operator Training Program
3. Senior Reactor Operator Training Program
4. Licensed Operator Requalification Training Program

Systems:

1. P53: Instrument Air System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Tue Aug 09 11:17:02 CDT 2005

Question History:

1. Created by mrasch at Mon Jun 13 09:42:58 CDT 2005
2. Modified by mrasch at Mon Jun 20 09:17:44 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005

7. Modified by mrasch at Thu Jul 28 17:24:47 CDT 2005
8. Question Reviewed by mellis at Thu Aug 04 07:25:05 CDT 2005
9. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
10. Modified by mrasch at Tue Aug 09 11:17:02 CDT 2005
11. Question Reviewed by mellis at Tue Aug 09 11:59:54 CDT 2005

Comments:

EB QUESTION: 51 (1.0 Points)

A break in the instrument air header in the Turbine Building has occurred.

Unit 2 Instrument Air Compressor and Plant Air Dryer B are in service.

Service Air Compressor A is in service.

The resulting high air flow through the in service drying tower causes the desiccant to clog the tower.

Dryer outlet pressure reaches 45 psig.

What will occur as a result of this condition?

A. Plant Air Dryer A will automatically align itself to Unit 2 Instrument Air Compressor.

B. Unit 1 Instrument Air Compressor will automatically start and align through Plant Air Dryer A.

C. Service Air Compressor A will automatically align through Plant Air Dryer A.

D. Plant Air Dryer B will undergo an Executed Stop and its other tower will go into drying mode.

Answer: D

Question Comments: There is no automatic shifting or starts for the Instrument Air Compressors or Air Dryers. The in service air dryer will go through an executed stop. Therefore Answer D is correct. Tier 2 Group 1 This is a NEW question. 10CFR 41.4/41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00883

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 4

Objectives:

1. CourseID: GLP-OPS-P5300 Objective: 3; 4; 23; 27; 31; 33
2. CourseID: GLP-OPS-ONEP Objective: 40

KA References:

1. 300000 K6.13 Filters [2.8/2.3]
2. 300000 A2.01 Air dryer and filter malfunctions [2.9/2.8]

References:

1. M-1067G
2. M-1126
3. M-1068D
4. M-1067A
5. ONEP 05-1-02-V-9 section 3.6; 3.8.1; 3.20
6. ARI 04-1-02-1H13-P870-7A-E3

TrainingPrograms:

1. Nonlicensed Operator Training Program
2. Reactor Operator Training Program
3. Senior Reactor Operator Training Program
4. Licensed Operator Requalification Training Program

Systems:

1. P51: Plant Air System
2. P52: Service Air System
3. P53: Instrument Air System

Categories:

1. Off Normal Event Procedures
2. Systems
3. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Tue Aug 09 11:17:49 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 09:51:52 CDT 2005
2. Modified by mrasch at Mon Jun 20 05:29:03 CDT 2005
3. Modified by mrasch at Mon Jun 20 09:37:57 CDT 2005
4. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
5. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
7. Modified by mrasch at Thu Jul 14 10:46:22 CDT 2005
8. Question Reviewed by mrasch at Fri Jul 15 16:31:39 CDT 2005
9. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
10. Modified by mrasch at Thu Jul 28 15:32:03 CDT 2005
11. Modified by mellis at Fri Jul 29 13:10:28 CDT 2005
12. Question Reviewed by mrasch at Thu Aug 04 07:29:11 CDT 2005
13. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
14. Modified by mrasch at Tue Aug 09 11:17:49 CDT 2005
15. Question Reviewed by mellis at Tue Aug 09 11:59:55 CDT 2005

Comments:

EB QUESTION: 52 (1.0 Points)

The plant is in Mode 3 with RHR A in Shutdown Cooling mode.

Reactor coolant temperature is 338°F.

Reactor Water level and temperature were stable before the event.

Which one of the following would be indicative of an RHR A Heat Exchanger tube rupture if a tube leak occurred?

A. RHR A pump discharge pressure trending up

B. Reactor water level trending up

C. SSW A radiation monitor readings trending up

D. Reactor coolant temperature trending down

Answer: C

Question Comments: Coolant temperature 338°F equates to ~ 100 psig. Tube leakage would be from RHR to SSW. Answer A is incorrect because a leak into SSW would be less flow restriction, RHR flow would go up and discharge pressure would go down. Answer B is incorrect because RHR would be at a higher pressure than SSW, so RPV level would go down. Answer C is CORRECT because RHR would leak into SSW. Reactor water is of higher activity than SSW, so rad monitor readings would go up. Answer D is incorrect because less RHR flow to the reactor, which would be at a higher temperature, would result. So, higher temperature IS indicative of a tube leak. Tier 2 Group 1 This is a NEW question. 10CFR 41.7/41.13/43.4

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00884

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-E1200 Objective: 4.2; 21
2. CourseID: GLP-OPS-P4100 Objective: 21

KA References:

1. 400000 K1.04 Reactor coolant system, in order to determine source (s)of RCS leakage into CCWS [2.9/3.1]

References:

1. ARI 04-1-02-1H13-P601 18A-F6
2. SFD-1085-001
3. SFD-1085-002
4. SFD-1061C

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. D17: Process Radiation Monitoring System
2. E12: Residual Heat Removal System
3. P41: Standby Service Water System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Tue Aug 09 11:18:51 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 09:58:09 CDT 2005
2. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
3. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
4. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
5. Modified by mrasch at Thu Jul 14 11:00:04 CDT 2005
6. Question Reviewed by mrasch at Fri Jul 15 16:31:39 CDT 2005
7. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
8. Modified by mrasch at Tue Jul 26 15:39:19 CDT 2005
9. Modified by mrasch at Wed Jul 27 07:50:34 CDT 2005
10. Question Reviewed by mellis at Thu Aug 04 07:25:11 CDT 2005
11. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
12. Modified by mrasch at Tue Aug 09 11:18:51 CDT 2005
13. Question Reviewed by mellis at Tue Aug 09 11:59:56 CDT 2005

Comments:

EB QUESTION: 53 (1.0 Points)

The plant is at 100% power.

The Turbine Building Cooling Water (TBCW) temperature control valve, P44-F513, fails closed.

Which one of the following will necessitate plant shutdown first, assuming Loss of TBCW ONEP actions are performed, but P44-F513 remains closed?

A. Main Turbine Lube Oil temperature

B. Loss of Instrument Air Compressors

C. Reactor Feed Pump Oil temperature

D. Generator Seal Oil temperature

Answer: D

Question Comments: This question is based on actual plant events and the changes made to the ONEP for Loss of TBCW. At 100% power, seal oil temperature is normally ~ 115°F. This is only 10°F margin to the limit of 125°F specified in plant procedures where plant shutdown is required. The ONEP lists temperatures in their expected order of priority. That sequence is expected based on plant data for a universal degradation of TBCW heat removal capacity. Seal oil is expected to reach its limit first based on plant and simulator data, given its normal operating temperature. That is why answer D is correct and answers A, B, and C are incorrect. Tier 2 Group 1 This is a NEW question. 10CFR 41.4/41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00885

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-ONEP Objective: 1; 2; 44

KA References:

1. 400000 A1.02 CCW temperature [2.8/2.8]

References:

1. 05-1-02-V-2

TrainingPrograms:

1. Nonlicensed Operator Training Program
2. Reactor Operator Training Program
3. Senior Reactor Operator Training Program
4. Licensed Operator Requalification Training Program

Systems:

1. N42: Seal Oil Syste
2. P43: Turbine Building Cooling Water System
3. P44: Plant Service Water System

Categories:

1. Off Normal Event Procedures
2. Systems
3. Continuing Training

Task References:

Question Last Revised By: Mickey Ellis at Fri Jul 29 13:23:59 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 10:10:30 CDT 2005
2. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
3. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
4. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
5. Modified by mrasch at Fri Jul 15 11:19:02 CDT 2005
6. Question Reviewed by mrasch at Fri Jul 15 16:31:39 CDT 2005
7. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
8. Modified by mellis at Fri Jul 29 13:23:59 CDT 2005
9. Question Reviewed by mrasch at Thu Aug 04 07:29:12 CDT 2005
10. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 54 **(1.0 Points)**

With reactor power above the high power setpoint, what is the purpose for the control rod withdrawal limiter?

A. Mitigates the effects from a control rod drop event

B. Prevents inadvertently exceeding fuel preconditioning limits

C. Prevents violating the Minimum Critical Power Ratio (MCPR) Safety Limit

D. Ensures core exposure burn rates are evenly distributed

Answer: C

Question Comments: The basis for RWL as specifically stated in Tech Spec bases is to prevent exceeding the MCPR safety limit. That is why answer C is correct. Answers A, B, and D are not related to that limit, so they are incorrect. Tier 2 Group 2 This is a NEW question. 10CFR 41.2/41.6/43.2

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00886

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-OPS-C1102 Objective: 2; 6

KA References:

1. 201005 K5.10 Rod withdrawal limiter: BWR-6 [3.2/3.3]

References:

1. Tech Spec Bases F3.3.2.1 Function 1a

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. C11-2: Rod Control and Information System

Categories:

1. Systems
2. Technical Specifications
3. Continuing Training

Task References:

Question Last Revised By: Tommy Harrelson at Fri Jul 29 09:04:26 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 10:15:34 CDT 2005
2. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
3. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
4. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
6. Modified by tharrelso at Wed Jul 27 15:20:46 CDT 2005
7. Modified by tharrelso at Fri Jul 29 09:04:26 CDT 2005
8. Question Reviewed by mellis at Thu Aug 04 07:25:25 CDT 2005
9. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 55 (1.0 Points)

The plant has been performing a special core physics test that involves varying reactor recirc pump flow. A human performance error by the lead test engineer leads to a reactor transient.

Currently:

Reactor power is 28% and stable.

Core Flow is 8% and stable.

RPV pressure is 990 psig and stable.

RPV water level is +36 inches and stable.

The Turbine is in operation and the bypass valves are closed.

One Reactor Recirc Pump is operating in slow speed.

Based on this information, the crew should:

A. Raise core flow by starting the second Reactor Recirc pump.

B. Reduce reactor power to 25%.

C. Raise core flow by opening the operating pump flow control valve.

D. Insert all control rods.

Answer: D

Question Comments: Power greater than 25% with core flow less than 10% is a safety limit violation. This requires all control rods be inserted within two hours. Tier 2 Group 2 This is a NEW question. 10CFR55.41 (5)

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00887

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-TS001 Objective: 29

KA References:

1. GENERIC 2.4.11 Knowledge of abnormal condition procedures [3.4/3.6]
2. 202001

References:

1. Tech Spec 2.1.1

TrainingPrograms:

1. Reactor Operator Training Program

2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. B33: Reactor Recirculation System

Categories:

1. Systems
2. Technical Specifications
3. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Tue Aug 09 11:21:32 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 10:28:15 CDT 2005
2. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
3. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
4. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
5. Modified by mrasch at Thu Jul 14 13:00:42 CDT 2005
6. Question Reviewed by mrasch at Fri Jul 15 16:31:39 CDT 2005
7. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
8. Modified by mrasch at Thu Jul 28 16:48:58 CDT 2005
9. Modified by tharrelso at Fri Jul 29 08:26:14 CDT 2005
10. Modified by tharrelso at Fri Jul 29 09:05:15 CDT 2005
11. Modified by mellis at Fri Jul 29 13:23:09 CDT 2005
12. Question Reviewed by mrasch at Thu Aug 04 07:29:13 CDT 2005
13. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
14. Modified by mrasch at Tue Aug 09 11:21:32 CDT 2005
15. Question Reviewed by mellis at Tue Aug 09 11:59:56 CDT 2005

Comments:

EB QUESTION: 56 (1.0 Points)

The plant is operating at 100% rated thermal power with the Reactor Recirc Flow Control Valves at 68% valve position when a LOCA in the Drywell occurs.

As a result, drywell pressure reached 4.5 psig.

RPV water level reached - 80 inches.

Five minutes later the Reactor Operator should observe:

A. Recirc Pump breakers CB-3A/B are closed and Recirc Flow Control Valves are 20% open.

B. Recirc Pump breakers CB-3A/B are closed and Recirc Flow Control Valves are 68% open.

C. All the Recirc Pump breakers are open and Recirc Flow Control Valves are 20% open.

D. All the Recirc Pump breakers are open and Recirc Flow Control Valves are 68% open

Answer: B

Question Comments: Drywell pressure will cause the Recirc FCV HPUs to trip preventing valve motion on the FCVs (remain at 68%). The Drywell pressure will cause a reactor scram preventing the EOC actuation. RPV level dropping will cause ATWS RPT to occur opening CB-1, 2, 4, 5 for both Recirc pumps. Therefore Answer B is the correct answer. Tier 2 Group 2 This is a NEW question. 10CFR 41.3/41.5/41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00888

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-B3300 Objective: 27.5; 28.2; 28.3; 47

KA References:

1. 202002 K1.01 Recirculation system [3.5/3.6]

References:

1. ARI 04-1-02-1H13-P680 3A-E3
2. Tech Spec Bases B3.3.4.1; B3.3.4.2
3. 17-S-06-5 Att II pages 12; 13

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. B33: Reactor Recirculation System

Categories:

1. Off Normal Event Procedures
2. Systems
3. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Tue Aug 09 11:22:45 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 10:39:55 CDT 2005
2. Modified by mrasch at Wed Jun 15 12:21:58 CDT 2005
3. Modified by mrasch at Wed Jun 15 12:26:43 CDT 2005
4. Modified by mrasch at Wed Jun 15 12:32:08 CDT 2005
5. Modified by mrasch at Mon Jun 20 09:41:46 CDT 2005
6. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
7. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
8. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
9. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
10. Modified by mrasch at Thu Jul 28 14:39:03 CDT 2005
11. Modified by tharrelso at Fri Jul 29 09:07:58 CDT 2005
12. Modified by mellis at Fri Jul 29 13:24:51 CDT 2005
13. Question Reviewed by mrasch at Thu Aug 04 07:29:14 CDT 2005
14. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
15. Modified by mrasch at Tue Aug 09 11:22:45 CDT 2005
16. Question Reviewed by mellis at Tue Aug 09 11:59:57 CDT 2005

Comments:

EB QUESTION: 57 (1.0 Points)

Which one of the following conditions associated with the RWCU system will cause a reduction in plant efficiency when operating at 100% power?

Assume none of the listed actions cause a RWCU filter/deminerilizer isolation.

A. A rise of 5°F in Component Cooling Water temperature at the inlet of the RWCU Non-Regenerative Heat Exchangers.

B. The RWCU Filter/Demineralizer Bypass valve is leaking by a value of 100 gpm.

C. An RWCU flow controller malfunction resulting in a reduction in total RWCU flow of 100 gpm.

D. The RWCU Regenerative Heat Exchanger Bypass valve is throttle open 25%.

Answer: D

Question Comments: Answer A is incorrect because a rise in CCW temperature results in a rise in RWCU temperature that causes a less heat having to be made up by the core thus a rise in efficiency. Answer B is incorrect because bypassing the Filter Demineralizers does not affect the heat loss of RWCU such that there is no effect on plant efficiency. Answer C is incorrect because a reduction in RWCU flow means less cool water to be reheated by the core thus a rise in plant efficiency. Answer D is CORRECT because less cooling flow through the Regenerative heat exchanger results in less preheating of the return RWCU water to the reactor causing a loss in plant efficiency. Tier 2 Group 2 This is a NEW question. 10CFR 41.5/41.14

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00889a

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-OPS-G3336 Objective: 4.2; 9.9; 18; 19

KA References:

1. 204000 A4.06 System flow [3.0/2.9]

References:

1. SFD-1079

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. G33: Reactor Water Cleanup

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Fri Jul 15 16:04:42 CDT 2005
Question History:

1. Created by mrasch at Fri Jul 15 11:57:02 CDT 2005
2. Created by mrasch at Fri Jul 15 11:57:02 CDT 2005 from parent QuestionID GGNS-NRC-00889
3. Modified by mrasch at Fri Jul 15 11:58:32 CDT 2005
4. Modified by mrasch at Fri Jul 15 16:04:42 CDT 2005
5. Question Reviewed by mrasch at Fri Jul 15 16:31:39 CDT 2005
6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
7. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 58 (1.0 Points)

Prior to emergency depressurizing using the SRVs, EP-2, RPV Control, requires the operating crew to verify Suppression Pool level greater than 10.5 feet.

The basis for this verification is to ensure:

A. there is sufficient heat capacity in the Suppression Pool to absorb the energy from a rapid depressurization.

B. there is sufficient water to preclude the rise in Suppression Pool temperature affecting the operation of the ECCS injection pumps.

C. there is sufficient backpressure to limit steam flowrates and potential damage to the SRVs.

D. the SRV discharge quenchers in the Suppression Pool are submerged.

Answer: D

Question Comments: 10.5 feet in the suppression pool is based on coverage of the SRV quenchers being covered to prevent the introduction of steam into Containment which would present a challenge to the Containment Sturcture. Therefore Answer D is the correct answer. Tier 2 Group 2 This is a NEW question. 10CFR 41.3/41.4/41.7/41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00890

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-OPS-EP02A Objective: 7
2. CourseID: GLP-OPS-EP03 Objective: 3

KA References:

1. 223001 K1.08 Relief/safety valves [3.6/3.8]

References:

1. PSTG

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. E22-2: Automatic Depressurization System
2. M41-1: Containment

Categories:

1. Emergency Procedure Training
2. Systems
3. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Thu Jul 28 12:16:42 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 10:58:48 CDT 2005
2. Modified by mrasch at Mon Jun 20 09:44:41 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
7. Modified by mrasch at Thu Jul 28 12:16:42 CDT 2005
8. Question Reviewed by mellis at Thu Aug 04 07:25:43 CDT 2005
9. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 59 (1.0 Points)

The plant is operating at 100% power when the steam supply to the Offgas Preheaters is lost.

What effect will this have on the Offgas System?

A. Less hydrogen will be removed from the process stream by the catalytic recombiners.

B. Fewer radioactive particulates will be removed from the process stream by the catalytic recombiners.

C. Offgas Pre-treatment radiation levels will rise.

D. Offgas Post-treatment radiation levels will rise.

Answer: A

Question Comments: The loss of steam to the Preheaters will result in a less efficiency in the recombiner component causing offgas hydrogen content to rise. The Offgas recombiners do not affect radioactive materials in offgas. Therefore Answer A is correct. Tier 2 Group 2 This is a NEW question. 10 CFR 41.4/41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00891

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-OPS-N6465 Objective: 5.1; 14.3

KA References:

1. 239001 K3.04 Offgas system [2.8/2.8]

References:

1. 04-1-02-1H13-P845 1A-A2

TrainingPrograms:

1. Nonlicensed Operator Training Program
2. Reactor Operator Training Program
3. Senior Reactor Operator Training Program

4. Licensed Operator Requalification Training Program

Systems:

1. N11: Main Steam System
2. N64: Offgas System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Thu Jul 28 12:36:49 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 11:10:19 CDT 2005
2. Modified by mrasch at Mon Jun 20 09:49:29 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Modified by mrasch at Thu Jul 14 13:39:50 CDT 2005
7. Question Reviewed by mrasch at Fri Jul 15 16:31:39 CDT 2005
8. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
9. Modified by mrasch at Thu Jul 28 12:36:49 CDT 2005
10. Question Reviewed by mellis at Thu Aug 04 07:25:47 CDT 2005
11. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 60 (1.0 Points)

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

Severe Accident Procedures have been entered.

The Control Room Supervisor has directed initiation of the Outboard Main Steam Isolation Valve Leakage Control System (MSIV LCS).

Which one of the following describes the effect of Standby Gas Treatment System (SGTS) on the effluent of the MSIV LCS?

A.

Either Standby Gas Treatment System (SGTS) A or B will process most of the effluent from either MSIV LCS subsystem.

B. Effluent of the Outboard MSIV LCS is piped directly to Standby Gas Treatment System (SGTS) A ducting, therefore SGTS A is the preferred system to operate.

C. Simultaneous operation of both Inboard MSIV LCS and Outboard MSIV LCS is prohibited unless both Standby Gas Treatment Systems (SGTS) are in operation.

D. Operation of the Outboard MSIV LCS in conjunction with Standby Gas Treatment System (SGTS) B is preferred.

Answer: A

Question Comments: Answer A is correct because MSIV LCS exhausts to auxiliary building corridors on 119' elev. SGTS takes suction on these areas and maintains negative pressure in the auxiliary building, therefore essentially all MSIV LCS exhaust will be eventually processed by SGTS. Answer B is incorrect because MSIV LCS is not piped directly to SGTS, but only in the vicinity of an intake to SGTS ductwork. Answer C is incorrect because simultaneous operation of both inboard and outboard MSIV LCS is always prohibited. Answer D is incorrect because SGTS A is preferred with the outboard MSIV LCS due to the shorter associated transport time. Tier 2 Group 2 This is a NEW question. 10CFR 41.4/41.14/43.4

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00892

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-E3200 Objective: 12.3; 13.1; 13.2

KA References:

1. 239003 K1.02 Standby gas treatment system: BWR-4,5,6(P-Spec) [2.9/3.0]

References:

1. 04-1-01-E32-1 steps 3.1; 3.2; 3.7; 3.8; 5.2.1c

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. E32: MSIV Leakage Control System
2. T48: Standby Gas Treatment System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Tue Aug 09 16:11:37 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 11:23:14 CDT 2005
2. Modified by mrasch at Mon Jun 20 09:52:39 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
7. Modified by tharrelso at Fri Jul 29 09:12:42 CDT 2005
8. Modified by tharrelso at Fri Jul 29 09:19:47 CDT 2005
9. Question Reviewed by mellis at Thu Aug 04 07:25:53 CDT 2005
10. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
11. Modified by mrasch at Tue Aug 09 11:23:52 CDT 2005
12. Question Reviewed by mellis at Tue Aug 09 11:59:58 CDT 2005
13. Modified by mrasch at Tue Aug 09 16:11:37 CDT 2005
14. Question Reviewed by mellis at Tue Aug 09 16:14:22 CDT 2005

Comments:

EB QUESTION: 61 (1.0 Points)

Following a scram from rated conditions, pressure is being controlled with the Main

Bypass Valves.

There is a steam leak in the RHR A room that CANNOT be isolated.

EP-2 and EP-4 are being performed.

The CRS has determined Emergency Depressurization is anticipated due to EP-4 concerns.

What action should be taken for this condition?

A. Open 6 Safety Relief Valves to lower the pressure band to 450 psig to 650 psig.

B. Throttle open Main Bypass Valves to reduce the driving head of the leak, staying within Technical Specification cooldown rate limitations.

C. Fully open the Main Bypass Valves using the Manual Bypass Jack to fully depressurize the reactor.

D. Lower Pressure Reference to 450 psig to reduce the driving head of the leak.

Answer: C

Question Comments: The PSTGs state it is preferred to discharge steam to the main condenser to limit the challenge to containment. No conditions exist that require closing MSIVs. EP-2A gives guidance for using bypass valves and maintaining the MSIVs open. Operations Philosophy disallows use of low-low set operation of SRVs during ATWS conditions. EHC pumps can be restarted from the control room since bus 14AE undervoltage lockouts are reset. Answer A, B, and D are incorrect because EP-2A bases prefers using bypass valves and maintaining the MSIVs open. Answer C is correct because it includes guidance for using bypass valves and maintaining the MSIVs open. Tier 2 Group 2 This is a NEW question. 10CFR 41.5/41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00893

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 2
Objectives:

1. CourseID: GLP-OPS-EP02A Objective: 2; 5
2. CourseID: GLP-OPS-PROC Objective: 59.3

KA References:

1. GENERIC 2.4.6 Knowledge symptom based EOP mitigation strategies [3.1/4.0]
2. 241000

References:

1. PSTG B-6-25, 38, 41, 42, 43
2. PSTG B-14-9, 11
3. PSTG B-16-5
4. 02-S-01-27 steps 6.1.6; 6.2.4; 6.6.8d

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. B21: Nuclear Boiler System

Categories:

1. Administrative Requirements
2. Emergency Procedure Training
3. Systems
4. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Tue Aug 09 11:24:55 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 11:35:20 CDT 2005
2. Modified by mrasch at Mon Jun 20 09:59:28 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Modified by mrasch at Thu Jul 14 14:03:56 CDT 2005

7. Modified by mrasch at Thu Jul 14 14:21:37 CDT 2005
8. Modified by mrasch at Thu Jul 14 14:22:52 CDT 2005
9. Question Reviewed by mrasch at Fri Jul 15 16:31:40 CDT 2005
10. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
11. Modified by mrasch at Thu Jul 28 15:38:39 CDT 2005
12. Modified by tharrelso at Fri Jul 29 09:21:49 CDT 2005
13. Question Reviewed by mellis at Thu Aug 04 07:25:57 CDT 2005
14. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
15. Modified by mrasch at Tue Aug 09 11:24:55 CDT 2005
16. Question Reviewed by mellis at Tue Aug 09 11:59:59 CDT 2005

Comments:

EB QUESTION: 62 **(1.0 Points)**

Which one of the following would automatically occur during Main Turbine roll up from 400 rpm to 1800 rpm?

A. Primary Water Circulating Pump will start.

B. Turbine Turning Gear valves will close.

C. Shaft Lift Oil Pump will stop.

D. Generator Field Breaker will close.

Answer: C

Question Comments: Answer A is incorrect because the Primary Water Ciruclating Pump stops on turbine roll up. Answer B is incorrect because the Turning gear valves close around 240 RPM and must be closed before 400 RPM. Answer D is incorrect because the Generator Field Breaker is only enabled at 1710 RPM but must be manually closed from the control room. Answer C is CORRECT because the Shaft Lift Oil Pump will automatically stop at about 540 RPM. Tier 2 Group 2 This is a NEW question. 10CFR 41.4/41.10

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00894

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-N4151 Objective: 8.2
2. CourseID: GLP-OPS-N4300 Objective: 10
3. CourseID: GLP-OPS-N3402 Objective: 6.3: 6.4
4. CourseID: GLP-OPS-IOI01 Objective: 33

KA References:

1. 245000 A3.02 Turbine roll to rated speed [2.8/2.8]

References:

1. 03-1-01-2 sections 7.1.4; 7.1.5; 7.2
2. 04-1-01-N40-1 section 3.12

TrainingPrograms:

1. Nonlicensed Operator Training Program
2. Reactor Operator Training Program
3. Senior Reactor Operator Training Program
4. Licensed Operator Requalification Training Program

Systems:

1. N30: Main Turbine
2. N34: Main Turbine Lube Oil System
3. N40: Main Generator
4. N43: Primary Water System

Categories:

1. Integrated Plant Operations
2. Systems
3. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Tue Aug 09 11:25:33 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 11:43:21 CDT 2005
2. Modified by mrasch at Mon Jun 20 10:02:26 CDT 2005

3. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
7. Modified by mrasch at Tue Jul 26 17:41:15 CDT 2005
8. Modified by mrasch at Wed Jul 27 09:39:56 CDT 2005
9. Modified by tharrelso at Fri Jul 29 09:23:50 CDT 2005
10. Question Reviewed by mellis at Thu Aug 04 07:26:01 CDT 2005
11. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
12. Modified by mrasch at Tue Aug 09 11:25:33 CDT 2005
13. Question Reviewed by mellis at Tue Aug 09 12:00:00 CDT 2005

Comments:

EB QUESTION: 63 **(1.0 Points)**

The plant is operating at 100% power when Condensate Pump A spuriously trips.

What is the preferred action for this condition?

A. Scram the reactor due to inevitable Reactor Feed Pump trip on low suction flow.

B. Initiate Reactor Core Isolation Cooling due to the inevitable Reactor Feed Pump trip on low suction pressure.

C. Re-start Condensate Pump A in accordance with the SOI to prevent inevitable Condensate Booster Pump trips on low suction pressure.

D. Lower reactor power to raise margin to Reactor Feed Pump trip setpoints, staying within power to flow limitations.

Answer: D

Question Comments: A loss of a single Condensate pump will cause a lower suction pressure for the Condensate Booster Pumps and Reactor Feed pumps but will not cause a reactor scram. GGNS is designed to operate full power with a single condensate and condesate booster pump out of service. Tier 2

Group 2 This is a NEW question. 10CFR 41.4/41.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00895

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-OPS-N1900 Objective: 2; 25

KA References:

1. 259001 A2.03 Loss of condensate pump(s) [3.6/3.6]

References:

1. 05-1-02-V-7
2. 04-1-02-1H13-P680-1A-A1

TrainingPrograms:

1. Nonlicensed Operator Training Program
2. Reactor Operator Training Program
3. Senior Reactor Operator Training Program
4. Licensed Operator Requalification Training Program

Systems:

1. N19: Condensate System
2. N21: Feedwater System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Thu Jul 28 15:55:48 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 11:48:54 CDT 2005
2. Modified by mrasch at Mon Jun 20 10:07:44 CDT 2005
3. Modified by mrasch at Mon Jun 20 10:12:56 CDT 2005
4. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005

5. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
7. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
8. Modified by mrasch at Thu Jul 28 15:55:48 CDT 2005
9. Question Reviewed by mellis at Thu Aug 04 07:26:07 CDT 2005
10. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 64 (1.0 Points)

A LOCA has occurred causing Drywell pressure to rise to 4.2 psig.

Which one of the following describes a possible flowpath to transfer the contents of the Containment Floor Drain sump under these conditions?

A. The sump cannot be transferred.

B. The sump will automatically transfer directly to the Floor Drain Collector tank.

C. The sump will automatically transfer to the Auxiliary Building Floor Drain Transfer tank then to the Floor Drain Collector tank.

D. The sump can be transferred to the Suppression Pool via AUX BLDG EQ/FL DR PMPBK TO SUPP POOL valves, P45-F273 and P45-F274.

Answer: A

Question Comments: With Drywell pressure at 4.2 psig, an NSSSS isolation signal exists. This closes the Primary and Secondary Containment isolation valves for P45 and cannot be overridden except P45-F273 and F274 which may be overridden 30 seconds after the isolation signal. Answer A is CORRECT because the flowpath from the containment cannot be opened with the isolation signal present. Answer B is INCORRECT because the Containment Floor Drain Sump must pass through the Auxiliary Building Transfer tank before exiting the Auxiliary Building to Radwaste. Answer C is INCORRECT because the flowpath is correct, however the primary and secondary containment isolation valves are closed and cannot be reopened. Answer D is INCORRECT because the Primary Containment isolation valves are closed and cannot be reopened. Tier 2 Group 2 This is a NEW question. 10CFR 41.4/41.10/43.2/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00896

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-OPS-P4500 Objective: 9, 10.9

KA References:

1. 268000 K1.04 Reactor building floor drains: Plant-Specific [2.7/2.9]

References:

1. M1094

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. P45: Floor and Equipment Drain System

Categories:

1. Administrative Requirements
2. Systems
3. Technical Specifications
4. Continuing Training

Task References:

Question Last Revised By: Tommy Harrelson at Fri Jul 29 09:24:20 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 11:56:48 CDT 2005
2. Modified by mrasch at Mon Jun 20 14:01:47 CDT 2005

3. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Modified by mrasch at Mon Jul 11 14:19:19 CDT 2005
7. Modified by mrasch at Thu Jul 14 14:56:24 CDT 2005
8. Modified by mrasch at Thu Jul 14 16:37:19 CDT 2005
9. Question Reviewed by mrasch at Fri Jul 15 16:31:40 CDT 2005
10. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
11. Modified by tharrelso at Wed Jul 27 15:36:45 CDT 2005
12. Modified by tharrelso at Fri Jul 29 09:24:20 CDT 2005
13. Question Reviewed by mellis at Thu Aug 04 07:26:14 CDT 2005
14. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 65 **(1.0 Points)**

Which one of the following identifies where the Auxiliary Building Steam Tunnel Floor Drains are directly routed?

A. The RHR A room Floor Drain sump

B. The Auxiliary Building Floor Drain Transfer tank

C. The Containment Steam Tunnel Floor Drain sump

D. The RCIC room Floor Drain sump

Answer: D

Question Comments: Answer A, B, and C are incorrect because the Auxiliary Building Steam Tunnel Floor Drains are hard piped directly to the RCIC room Floor Drain sump, which makes answer D correct. Tier 2 Group 2 This is a

NEW question. 10 CFR 41.9

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00897

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-OPS-P4500 Objective: 3.13

KA References:

1. 290001 K4.03 Fluid leakage collection [2.8/2.9]

References:

1. M1098A

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. P45: Floor and Equipment Drain System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: Tommy Harrelson at Fri Jul 29 09:25:47 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 12:01:57 CDT 2005
2. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
3. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
4. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam

Date: 08/12/2005

6. Modified by tharrelso at Wed Jul 27 16:51:23 CDT 2005
7. Modified by tharrelso at Wed Jul 27 16:53:58 CDT 2005
8. Modified by tharrelso at Thu Jul 28 06:16:34 CDT 2005
9. Modified by tharrelso at Fri Jul 29 09:25:47 CDT 2005
10. Question Reviewed by mellis at Thu Aug 04 07:26:21 CDT 2005
11. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 66 **(1.0 Points)**

The plant was at 95% power when Reactor Feed Pump B tripped.

Operation is in the Restricted Region of the Power-Flow Map.

Which of the following indications on 1H13-P680 would allow continued operation of the unit?

A. APRM oscillations of 14% peak-to-peak.

B. Annunciators PBDS A INOP (5A-A6) and PBDS B INOP (7A-A6) in due to failed cards.

C. PBDS A has generated a HI-HI DECAY RATIO (5A-A10) only and PBDS B has generated a HI DECAY RATIO (7A-C4) only.

D. Alarm PBDS B HI-HI DECAY RATIO (7A-C6) and PDS computer point PBDS Channel B Highest Counts reads 12 counts.

Answer: C

Question Comments: PBDS is normally a computer indication of APRM oscillations. Answer A is INCORRECT because APRM oscillations of > 10% is an indication of the onset of instability requiring plant shutdown. Answer B is INCORRECT because with operation in the Restricted Region of the Power to Flow Map and both PBDS channels INOP action with FCTR in NORMAL, calls for immediately placing the reactor mode switch to shutdown. Answer C is CORRECT because Channel A has a HI-HI without a HI which indicates a bad channel and Channel B only has a HI alarm. These conditions allow continued operation but require actions to be taken to immediately exit the region. Answer D is INCORRECT because both the HI-HI alarm on Channel B and indication of 12 counts on the computer point are indications of the Onset of Instability requiring a reactor scram. Tier 3 This is a NEW question. 10CFR 41.1/41.5/41.6/41.10/43.5/43.6

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00898

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-C5106 Objective: 1; 14.1; 14.2; 23.2

KA References:

1. GENERIC 2.1.19 Ability to use plant computer to obtain and evaluate parametric information on [3.0/3.0]
2. GENERIC 2.4.7 Knowledge of event based EOP mitigation strategies [3.1/3.8]

References:

1. Power to Flow Map (Restricted Region)
2. 05-1-02-III-3 Steps 2.1; 4.9
3. ARI 04-1-02-1H13-P680 5A-A6, A10; 7A-A6, C4, C6

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. B33: Reactor Recirculation System
2. C51-3: Local Power Range Nuclear Instrumentation System
3. C51-5: Average Power Range Nuclear Instrumentation System
4. C51-6: Period Based Detection System

Categories:

1. Off Normal Event Procedures
2. Systems
3. Continuing Training

Task References:

Question Last Revised By: Mickey Ellis at Fri Jul 29 13:38:58 CDT 2005

Question History:

1. Created by mrasch at Mon Jun 13 12:14:50 CDT 2005
2. Modified by mrasch at Mon Jun 20 10:18:27 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
7. Modified by tharrelso at Fri Jul 29 09:27:45 CDT 2005
8. Modified by mellis at Fri Jul 29 13:38:58 CDT 2005
9. Question Reviewed by mrasch at Thu Aug 04 07:29:15 CDT 2005
10. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 67 (1.0 Points)

The plant is operating at 100% power when the System Load Dispatcher calls and requests Grand Gulf change their generator power factor from1.0 to 0.975 capacitive to help with grid stability.

Generator hydrogen pressure is 60 psig.

Given NO generator operating limits will be exceeded, the operating crew should:

Generator V Curves are provided.

A. depress the voltage regulator LOWER pushbutton until approximately -300 MVARs are being carried by the generator.

B. depress the voltage regulator RAISE pushbutton until approximately 300 MVARs are being carried by the generator.

C. depress the voltage regulator RAISE pushbutton until approximately 250 MVARs are being carried by the generator.

D.

depress the voltage regulator LOWER pushbutton until approximately -250 MVARs are being carried by the generator.

Answer: D

Question Comments: From a unity power factor to 0.975 capacitive requires the generator to be under excited so the voltage regulator must be lowered. The limit on the GGNS Generator reactive load is 250 MVARs. Therefore Answer D is the Correct answer. Tier 3 This is a NEW question. 10CFR 41.4/41.10/43.5

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00899

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-OPS-N4151 Objective: 7; 13

KA References:

1. GENERIC 2.1.25 Ability to obtain and interpret station reference materials such as graphs / [2.8/3.1]

References:

1. 04-1-01-N40-1 step 3.8
2. 03-1-01-2 Figure 2

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. N40: Main Generator
2. N41: Generator Auxiliaries

Categories:

1. Integrated Plant Operations
2. Systems

Task References:

Question Last Revised By: Tommy Harrelson at Fri Jul 29 09:28:32 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 12:21:49 CDT 2005
2. Modified by mrasch at Mon Jun 20 10:23:16 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
7. Modified by mrasch at Thu Jul 28 13:03:59 CDT 2005
8. Modified by tharrelso at Fri Jul 29 09:28:32 CDT 2005
9. Question Reviewed by mellis at Thu Aug 04 07:26:33 CDT 2005
10. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 68 **(1.0 Points)**

An operator is restoring a red tag clearance on a manual valve.

The required position for the valve is THROTTLED, 1 1/2 TURNS OPEN.

There are NO red tie wraps available at GGNS.

Which one of the following is an acceptable method to lock the valve in the required position?

A. Cable with a padlock

B. Yellow valve seal

C. Black tie wrap

D. Blue tie wrap

Answer: A

Question Comments: Yellow plastic seals are used on Fire Protection Valves. Blue Tie-Wraps are used as a locking devices for valves other than throttled valves. Lockwire is used by I&C for sealing instrument valves in position. Chains and/or cables with padlocks are acceptable alternatives to Red and Blue Tie Wraps in the event the appropriate tie-wraps are unavailable. This is per Attachment III of 02-S-01-2 Component Position Verification. Based on this the ONLY CORRECT answer is answer A. This question is MODIFIED. Stem changed to Locked Throttled Valve from Locked Closed Valve. Answer Lockwire was replaced with Black Tie Wrap which is normally used when attaching red tags to components. Original question used RO Audit Examination December 2000 Question # 82 ID WRIA082. Similar question used RO NRC Examination April 2000 Question # 89 ID WRI289. Tier 3 10CFR 41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00237a

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-OPS-PROC Objective: 49.14

KA References:

1. Generic 2.1.29: 3.4/3.3

References:

1. 02-S-01-2 Att III

TrainingPrograms:

1. Nonlicensed Operator Training Program
2. Reactor Operator Training Program
3. Senior Reactor Operator Training Program
4. Licensed Operator Requalification Training Program

Systems:

Categories:

1. Administrative Requirements

Task References:

Question Last Revised By: MikeRasch at Thu Jun 09 09:25:39 CDT 2005
Question History:

1. Created by tharrelso at Tue May 10 13:33:43 CDT 2005
2. Created by tharrelso at Tue May 10 13:33:43 CDT 2005 from parent QuestionID GGNS-NRC-00237
3. Modified by mrasch at Thu May 12 15:52:49 CDT 2005
4. Question Reviewed by mellis at Tue May 31 14:56:58 CDT 2005
5. Modified by mrasch at Thu Jun 09 08:53:48 CDT 2005
6. Modified by mrasch at Thu Jun 09 09:25:39 CDT 2005
7. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
8. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
9. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
10. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
11. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 69 **(1.0 Points)**

Which one of the following situations presents an operability concern regarding Intermediate Range Monitors?

A. In Mode 2 preparing to transfer to RUN, all IRMs are fully inserted and indicate 15 to 20 on range 10 while all APRMs indicate 6% to 8% power.

B. During startup in Mode 2 with IRMs fully inserted, all IRMs indicate approximately 10 on range 1 when all SRMs, fully inserted, indicate approximately 2 x 10^4 cps.

C. During a "soft" shutdown with IRMs fully inserted, preparing to go from Mode 1 to Mode 2, APRMS indicate 4% power while the highest reading IRM indicates 110 on range 10 and the lowest IRM indicates 30 on range 10.

D. In Mode 2 with IRMs fully inserted, the highest reading IRM indicates 70 on range 8 and the lowest IRM indicates 20 on range 7.

Answer: C

Question Comments: Answer A is INCORRECT because overlap of IRMs to APRMs is of concern because system design will initiate rod blocks if adequate overlap is not maintained during power increases. Answer B is INCORRECT because proper overlap is being observed between SRMs and IRMs on plant startup. Answer C is CORRECT because APRMS are reading 4% with at least one IRM >108/125, this does not meet the overlap requirements of 03-1-01-3 section 5.4.7 of Attachment I. The IRM channel check is within a factor of 4 (limit is 10). Answer D is INCORRECT because the IRM channel check is within a factor of 4 (limit is 10). 20 on range 7 is 20 on range 8. Even ranges are just an expansion of the previous odd range. Tier 3 This is a NEW question. 10CFR 41.2/41.6/43.2

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00900

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-C5102 Objective: 9; 10; 15
2. CourseID: GLP-OPS-IOI01 Objective: 20

KA References:

1. GENERIC 2.2.1 Ability to perform pre-startup procedures for the facility / including operating those [3.7/3.6]

References:

1. 06-OP-1000-D-0001
2. 03-1-01-1 sections 5.28, 6.2.16
3. 03-1-01-3 sections 5.4.7, 5.4.8, 5.4.9 caution step 5.7
4. TS Bases 3.3.1.1 SR3.3.1.1.5 and 3.3.1.1.6

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program

3. Licensed Operator Requalification Training Program

Systems:

1. C51-2: Intermediate Range Nuclear Instrumentation System

Categories:

1. Integrated Plant Operations
2. Systems
3. Technical Specifications
4. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Mon Aug 01 16:03:50 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 12:29:04 CDT 2005
2. Modified by mrasch at Mon Jun 20 10:27:49 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
7. Modified by mrasch at Mon Aug 01 16:03:50 CDT 2005
8. Question Reviewed by mellis at Thu Aug 04 07:26:38 CDT 2005
9. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 70 (1.0 Points)

The plant is in Mode 5.

Core Alterations are in progress.

High Pressure Core Spray (HPCS) is tagged out of service to perform an internal inspection of HPCS TESTABLE CHK VLV E22-F005.

Fuel Pool Cooling and Clean Up (FPCC) pumps A and B are in service.

NO FPCC Filter Demin is in service.

RHR B is in Shutdown Cooling, returning to the RPV via RHR B SHUTDN CLG RTN TO FW valve E12-F053B.

Reactor Water Clean Up (RWCU) pump A is tagged out of service for breaker preventive maintenance, only.

Which one of the following activities would require notification by control room personnel to Refuel Floor supervision?

A. Clearing red tags and restoring HPCS to standby.

B. Returning RWCU pump A to standby.

C. Placing Standby Service Water B in service to FPCC heat exchangers.

D. Making necessary adjustments to RHR HX B OUTLT VLV E12-F003B to maintain constant reactor coolant temperature.

Answer: A

Question Comments: Answer A is CORRECT because there is a potential of an air bubble rising into the reactor could cause problems on the Refuel floor. Answer B is INCORRECT because this realignment of RWCU would not change flows to the Reactor. Answer C is INCORRECT because this will only alter the cooling medium for FPCCU not the actual flow of the system. Answer D is INCORRECT because this is only changing the amount of flow from Shutdown Cooling not swapping the system providing shutdown cooling. Tier 3 This is a NEW question. 10CFR 41.10/43.5/43.6/43.7

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00901

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-OPS-IOI05 Objective: 2.3
2. CourseID: GLP-OPS-PROC Objective: 8.30

KA References:

1. GENERIC 2.2.30 Knowledge of RO duties in the control room during fuel handling such as alarms [3.5/3.3]

References:

1. 01-S-06-2 step 6.7.29
2. 03-1-01-5 steps 2.24; 2.34; 2.35

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. E12: Residual Heat Removal System
2. E22-1: High Pressure Core Spray System
3. G33: Reactor Water Cleanup
4. G41: Fuel Pool Cooling and Cleanup System
5. P41: Standby Service Water System

Categories:

1. Administrative Requirements
2. Integrated Plant Operations
3. Continuing Training

Task References:

Question Last Revised By: Tommy Harrelson at Fri Jul 29 09:40:46 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 12:36:19 CDT 2005
2. Modified by mrasch at Mon Jun 20 10:29:40 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005

5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
7. Modified by tharrelso at Fri Jul 29 09:32:42 CDT 2005
8. Modified by tharrelso at Fri Jul 29 09:40:46 CDT 2005
9. Question Reviewed by mellis at Thu Aug 04 07:26:46 CDT 2005
10. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 71 **(1.0 Points)**

The Auxiliary Building Operator must enter a High Radiation Area to hang red tags.

The general area dose rate is 50% of the maximum dose rate that could be experienced for a High Radiation Area.

The operator is male, 30 years old, and has accumulated 200 mrem year-to-date Total Effective Dose Equivalent (TEDE).

His lifetime TEDE is 1200 mrem.

NO extension of the operator’s dose limit will be granted.

Which one of the following times is the longest the operator could stay in the general area dose rate for the High Radiation Area without exceeding the administrative dose limit for TEDE?

A. 1.8 hours

B. 3.6 hours

C. 4.8 hours

D. 9.6 hours

Answer: B

Question 50% of the maximum radiation dose rate of a High Radiation Area of 999

Comments: mrem/hr is 499 mrem/hr. Worker's dose margin to the annual administrative limit is 1800 mrem based on 200mrem already received and a 2000 mrem per year administrative TEDE limit. Entry into a 499mrem/hr field with an 1800 mrem maximum dose gives a stay time of 3.6 hours. This is Answer B. Answer A is based on 1000 mrem/hr. Answers C and D are INCORRECT because they are based on the NRC TEDE Limit of 5 Rem/Yr. Tier 3 This is a NEW question. 10CFR 41.10/41.12/43.4/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00902

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: ELP-GET-RWT Objective: RWT30; 32; 43; 44

KA References:

1. GENERIC 2.3.1 Knowledge of 10 CFR: 20 and related facility radiation control requirements [2.6/3.0]

References:

1. 01-S-08-2 section 6.5.1d
2. NMM ENS-RP-201 section 5.2.3.1

TrainingPrograms:

1. Nonlicensed Operator Training Program
2. Reactor Operator Training Program
3. Senior Reactor Operator Training Program
4. Licensed Operator Requalification Training Program

Systems:

Categories:

1. Administrative Requirements
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 10:30:45 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 12:41:38 CDT 2005
2. Modified by mrasch at Mon Jun 20 10:30:45 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
7. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 72 **(1.0 Points)**

A Site Area Emergency has been declared.

You must leave the control room to perform operations at an Alternate Shutdown Panel.

For this situation, your administrative annual dose limit is automatically extended to:

A. 2.5 Rem

B. 4.5 Rem

C. 5 Rem

D. 10 Rem

Answer: C

Question Comments: Emergency Support personnel are administratively extended to the federal limits of 10 CFR20 at the declaration of an Alert or above. The federal limit is 5 Rem TEDE per year Based on the above discussion Answer C is the only CORRECT answer. Tier 3 This is a NEW question. 10CFR 41.10/41.12/43.4/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00903

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: ELP-GET-RWT Objective: RWT36

KA References:

1. GENERIC 2.3.4 Knowledge of radiation exposure limits and contamination control / including [2.5/3.1]

References:

1. 10-S-01-17 section 6.8.1

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

Categories:

1. Administrative Requirements
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Thu Jul 28 17:09:15 CDT 2005

Question History:

1. Created by mrasch at Mon Jun 13 12:50:27 CDT 2005
2. Modified by mrasch at Mon Jun 20 10:31:56 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Modified by mrasch at Thu Jul 14 15:05:11 CDT 2005
7. Question Reviewed by mrasch at Fri Jul 15 16:31:40 CDT 2005
8. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam

Date: 08/12/2005

9. Modified by mrasch at Thu Jul 28 17:09:15 CDT 2005
10. Question Reviewed by mellis at Thu Aug 04 07:26:52 CDT 2005
11. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 73 (1.0 Points)

An ATWS is in progress.

Emergency Depressurization has just commenced.

The Control Room Supervisor has directed you to inject with 2 mlbm/hr using Condensate through the Start-Up Level Control Valve when reactor pressure lowers to the Minimum Steam Cooling Pressure (MSCP).

What is the basis for initializing injection at the rate of 2 mlbm/hr?

A. To prevent damage to reactor internals due to thermal shock.

B. To prevent exceeding DP restrictions for the Start-Up Level Control Valve.

C. To prevent substantial fuel damage due to large power excursions.

D. To prevent exceeding the upper end of the allowed level band.

Answer: C

Question Comments: EP-2A Caution 5. Answer C is the only correct answer. Tier 3 This is a NEW question. 10 CFR 41.7/41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00904

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 2

Objectives:

1. CourseID: GG-1-LP-RO-EP02A Objective: 2; 9

KA References:

1. GENERIC 2.4.20 Knowledge of operational implications of EOP warnings / cautions / and notes [3.3/4.0]

References:

1. 05-S-01-EP-2 Caution 5

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

1. B21: Nuclear Boiler System

Categories:

1. Administrative Requirements
2. Emergency Procedure Training
3. Systems
4. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Tue Aug 09 16:12:35 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 12:55:42 CDT 2005
2. Modified by mrasch at Mon Jun 20 10:35:55 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
7. Modified by mrasch at Thu Jul 28 16:11:24 CDT 2005
8. Modified by mrasch at Thu Jul 28 17:33:52 CDT 2005
9. Question Reviewed by mellis at Thu Aug 04 07:26:57 CDT 2005
10. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
11. Modified by mrasch at Tue Aug 09 11:27:46 CDT 2005

12. Question Reviewed by mellis at Tue Aug 09 12:00:01 CDT 2005
13. Modified by mrasch at Tue Aug 09 16:12:35 CDT 2005
14. Question Reviewed by mellis at Tue Aug 09 16:14:23 CDT 2005

Comments:

EB QUESTION: 74 **(1.0 Points)**

Both Standby Gas Treatment systems have automatically started because of a small break LOCA in the drywell.

The Roving operator was verifying that the systems started properly when the following annunciators alarmed.

SGTS FLTR TR A CHAR TEMP HI

SGTS FLTR TR A CHAR TEMP HI-HI

What immediate action should the operator take in response to these alarms after placing handswitch SGTS FLTR TR A HTR to the OFF position?

A. Dispatch a building operator to open the exhaust filter train fan A breaker.

B. Secure SGTS train A per the T48 SOI.

C. Dispatch a building operator to start the motor driven fire pump.

D. Secure both SGTS trains per the T48 SOI.

Answer: A

Question Comments: The SGTS automatically started due to exceeding a NSSSS LOCA signal setpoint and cannot be secured until the NSSSS signal is cleared and reset. Therefore answers B and D are incorrect. Answer A is correct because, per the ARI, the exhaust filter train fan for the affected system is opened to minimize the amount of air available to feed the fire. Answer C is incorrect because the motor driven fire should automatically start once the filter train deluge system is activated. Tier 3 This is a NEW question. 10 CFR 41.7

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00905

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-OPS-P6400 Objective: 3.8

KA References:

1. GENERIC 2.4.25 Knowledge of fire protection procedures [2.9/3.4]

References:

1. 10-S-03-2
2. 04-1-02-1H13-P870-2A-C2
3. 04-1-02-1H13-P870-2A-B2

TrainingPrograms:

1. Nonlicensed Operator Training Program
2. Reactor Operator Training Program
3. Senior Reactor Operator Training Program
4. Licensed Operator Requalification Training Program

Systems:

1. P64: Fire Water Protection System
2. T48: Standby Gas Treatment System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: Tommy Harrelson at Thu Jul 28 06:43:33 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 13:00:50 CDT 2005
2. Modified by mrasch at Mon Jun 20 10:38:38 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
4. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam

Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
7. Modified by tharrelso at Thu Jul 28 06:43:33 CDT 2005
8. Question Reviewed by mellis at Thu Aug 04 07:27:05 CDT 2005
9. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 75 (1.0 Points)

Which of the following is the preferred back-up method for notifications to state and local agencies when the Operational Hot Line (OHL) is inoperative during implementation of the Emergency Plan?

A. UHF radio

B. Satellite telephone

C. Commercial telephone

D. Entergy fiber optic lines

Answer: C

Question Comments: Per 10-S-01-6 all of the above are backup communications but per 6.3.1 lists in order of use the backup communications listing Commercial Telephone as the first method. Therefore answer C is CORRECT. Tier 3 This is a NEW question. 10CFR 41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00906

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-EP-EPT6 Objective: 3

KA References:

1. GENERIC 2.4.43 Knowledge of emergency communications systems and techniques [2.8/3.5]

References:

1. 10-S-01-6 section 6.3

TrainingPrograms:

1. Nonlicensed Operator Training Program
2. Reactor Operator Training Program
3. Senior Reactor Operator Training Program
4. Licensed Operator Requalification Training Program

Systems:

Categories:

1. Emergency Plan Training
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Mon Jun 13 13:04:50 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 13:04:50 CDT 2005
2. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
3. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
4. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
5. Question used on ExamName: NRC August 2005 - 1 ExamID: NRC-082005-1 Exam Date: 08/12/2005
6. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 76 **(1.0 Points)**

The plant was operating at 100% power in a normal configuration with NO inoperable equipment.

Then, at 0800 this morning, the Division 1 125 VDC battery was declared inoperable and removed from service to allow maintenance to work on an emergency battery issue.

To perform the battery work, both Division 1 battery chargers were also tagged out.

Later in the day, at 1500, a problem was discovered on the HPCS pump breaker and HPCS was declared inoperable.

In this situation, the plant must enter Mode 3, NO later than:

Tech Specs are provided.

A. 2000

B. 2100

C. 2200

D. 0400 (the following day)

Answer: C

Question Comments: When Division 1 125 VDC was declared inoperable and forced an entry into TS 3.8.4C giving 2 hours to restore the DC system. When the 2 hours expire, this forces an entry into 3.8.4.E and this gives the crew 12 hours to get into Mode 3. Therefore based on this they have 14 hours from 0800 or 2200. When HPCS is declared inoperable this forces an entry into 3.0.3 per TS 3.5.1. This requires the plant to be in mode 3 within 13 hours of discovery (0400). distractor A is based on entering 3.8.4.E immediately and not taking into consideration the 2 hours allowed to meet the LCO. While this is permitted, it does not change the actual TS due date. Therefore C is the only correct answer because it is the most limiting LCO. Tier 1 Group 1 This is a NEW question. 10CFR 43.2

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00908

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 2
Objectives:

1. CourseID: GLP-OPS-TS001 Objective: 34

KA References:

1. 295004 AA2.02: 3.5/3.9; AK2.03: 3.3/3.3

References:

1. Tech Specs 3.8.4; 3.5.3; 3.5.1

TrainingPrograms:

1. Senior Reactor Operator Training Program
2. Licensed Operator Requalification Training Program

Systems:

Categories:

1. Technical Specifications

Task References:

Question Last Revised By: MikeRasch at Thu Aug 04 08:11:56 CDT 2005
Question History:

1. Created by jbell at Thu Jun 16 14:05:36 CDT 2005
2. Modified by jbell at Thu Jun 16 14:06:58 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
4. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
5. Modified by mrasch at Fri Jul 29 06:34:25 CDT 2005
6. Question Reviewed by mellis at Thu Aug 04 07:27:17 CDT 2005
7. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
8. Modified by mrasch at Thu Aug 04 08:11:56 CDT 2005
9. Question Reviewed by mellis at Thu Aug 04 08:17:51 CDT 2005
10. Question used on ExamName: NRC August 2005 - 2 ExamID: NRC-082005-2 Exam Date: 08/12/2005

Comments:

EB QUESTION: 77 (1.0 Points)

The plant was at 100% power when a steam line break occurred in the Turbine Building.

A site evacuation was conducted, however, one mechanic who was logged into the Turbine Building at the time of the event could NOT be accounted for.

NO one on the search and rescue team would voluntarily accept the assignment.

What is the highest administrative dose limit extension that may be approved for NON-Volunteers by the Emergency Director if he believes it to be for a life saving activity?

A. 5 Rem

B. 10 Rem

C. 25 Rem

D. 50 rem

Answer: C

Question Comments: During an emergency the Emergency Director /Offsite Emergency Coordinator may authorize extensions of dosse limits based on the situation. Further dose limmit extensions are applicable only to volunteers. Up to 25 Rem may be authorized for NON-volunteers of search and rescue teams and repair teams. The highest dose limit extension for a Non-Volunteer is 25 Rem for protection of populations or saving a life. Protection of property is only authorized up to 10 Rem. Greater than 25 Rem is strictly voluntary for protection of populations and saving a life. Based on the above discussion Answer C is the only CORRECT answer. Tier 1 Group 1 This is a NEW question. 10CFR41.10/41.12/43.4/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00909

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory **Difficulty Rating:** 2

Objectives:

1. CourseID: Rad Worker Training, ELP-GET-RWT Objective: RWT36

KA References:

1. GENERIC 2.3.4 Knowledge of radiation exposure limits and contamination control / including [2.5/3.1]
2. 295005

References:

1. 10-S-01-17 Section 6.1

TrainingPrograms:

1. Senior Reactor Operator Training Program
2. Licensed Operator Requalification Training Program

Systems:

Categories:

1. Administrative Requirements
2. Emergency Plan Training

Task References:

Question Last Revised By: MikeRasch at Tue Aug 09 11:30:49 CDT 2005
Question History:

1. Created by jbell at Thu Jun 16 14:09:50 CDT 2005
2. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
3. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
4. Modified by mrasch at Mon Jul 18 16:30:24 CDT 2005
5. Modified by mrasch at Wed Jul 27 15:03:23 CDT 2005
6. Question Reviewed by mellis at Thu Aug 04 07:27:22 CDT 2005
7. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
8. Modified by mrasch at Thu Aug 04 08:12:40 CDT 2005
9. Question Reviewed by mellis at Thu Aug 04 08:17:53 CDT 2005
10. Modified by mrasch at Tue Aug 09 11:30:49 CDT 2005
11. Question Reviewed by mellis at Tue Aug 09 12:00:02 CDT 2005
12. Question used on ExamName: NRC August 2005 - 2 ExamID: NRC-082005-2 Exam Date: 08/12/2005

Comments:

EB QUESTION: 78 **(1.0 Points)**

The plant is in an ATWS.

Main Steam Isolation Valves are closed.

Suppression Pool level is 19.2 feet.

Average Suppression Pool temperature is 160 degrees F.

Reactor pressure is 1000 psig.

What is the highest priority action for this condition?

Figure 1 HCTL is provided.

A. Enter EP-3 and place both loops of pool cooling in service

B. Enter EP-2A emergency depressurization leg

C. Enter EP-3 and inhibit Suppression Pool Make-Up

D. Enter EP-2A and lower the pressure control band to 450-600 psig

Answer: B

Question Comments: With Suppression Pool temperature at 160 degrees F and Suppression Pool Level at 19.2 feet and the Reactor at 1000 psig conditions are UNSAFE on the HCTL curve requiring emergency depressurization per EP-2A. Tier 1 Group 1 This is a NEW question. 10 CFR 41.10/43.5

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00910

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 2

Objectives:

1. CourseID: GG-1-LP-RO-EP03 Objective: 2, 3, 6

KA References:

1. 295026 EA2.03: 3.9/4.0
2. Generic 2.4.6: 3.1/4.0

References:

1. HCTL curve

TrainingPrograms:

1. Senior Reactor Operator Training Program
2. Licensed Operator Requalification Training Program

Systems:

Categories:

1. Emergency Procedure Training

Task References:

Question Last Revised By: MikeRasch at Thu Aug 04 08:13:17 CDT 2005
Question History:

1. Created by jbell at Thu Jun 16 14:12:42 CDT 2005
2. Modified by jbell at Thu Jun 16 15:38:32 CDT 2005
3. Modified by mrasch at Mon Jun 20 10:44:37 CDT 2005
4. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Modified by mrasch at Fri Jul 29 06:45:02 CDT 2005
7. Question Reviewed by mellis at Thu Aug 04 07:27:27 CDT 2005
8. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
9. Modified by mrasch at Thu Aug 04 08:13:17 CDT 2005
10. Question Reviewed by mellis at Thu Aug 04 08:17:54 CDT 2005
11. Question used on ExamName: NRC August 2005 - 2 ExamID: NRC-082005-2 Exam Date: 08/12/2005

Comments:

EB QUESTION: 79 (1.0 Points)

The basis for the Technical Specification limit for Average Primary Containment temperature is:

A. To maintain containment air temperature below 185 degrees F during a LOCA.

B. To prevent degradation of equipment important to safety during plant operation.

C. To maintain containment pressure less than 22.4 psig during a LOCA.

D. To maintain errors in reactor water level instrumentation within the initial assumptions of the accident analyses.

Answer: A

Question Comments: This the bases of Tech Spec 3.6.1.5 and answer A is the only correct answer. Tier 1 Group 1 This is a NEW question. 10CFR 43.2

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00911

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-M4101 Objective: 4, 11, 12
2. CourseID: GLP-OPS-M4100 Objective: 17

KA References:

1. 295027 Generic 2.2.22: 3.4/4.1
2. EK1.03: 3.8/3.8

References:

1. Tech Spec Bases B 3.6.1.5

TrainingPrograms:

1. Senior Reactor Operator Training Program
2. Licensed Operator Requalification Training Program

Systems:

Categories:

1. Technical Specifications

Task References:

Question Last Revised By: MikeRasch at Tue Aug 09 16:13:24 CDT 2005
Question History:

1. Created by jbell at Thu Jun 16 14:15:44 CDT 2005
2. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
3. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
4. Modified by mrasch at Thu Jul 14 15:38:29 CDT 2005
5. Modified by mrasch at Fri Jul 29 06:55:37 CDT 2005
6. Question Reviewed by mellis at Thu Aug 04 07:27:49 CDT 2005
7. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
8. Modified by mrasch at Tue Aug 09 16:13:24 CDT 2005
9. Question Reviewed by mellis at Tue Aug 09 16:14:24 CDT 2005
10. Question used on ExamName: NRC August 2005 - 2 ExamID: NRC-082005-2 Exam Date: 08/12/2005

Comments:

EB QUESTION: 80 (1.0 Points)

A strong earthquake has occurred, causing large LOCA in the drywell and a Suppression Pool leak into the Low Pressure Core Spray (LPCS) pump room.

The LPCS room watertight door has failed.

Also, electrical bus 15AA has tripped due to a fault.

The following indications of containment parameters exist:

Division 1 Drywell pressure is 8.5 psig

Division 2 Drywell pressure is 8.6 psig

Average Drywell temperature is 170°F.

Average Containment temperature is 135°F.

Division 1 Suppression Pool level is 14.6 feet.

Division 2 Suppression Pool level is 14.0 feet.

Average Suppression Pool temperature is 135°F.

What action should be directed based on evaluation of these conditions?

A. Enter EP-4 and direct operation of all available sump pumps.

B. Execute emergency depressurization per EP-2.

C. Execute EP-2 Attachment 29, Primary Containment Water Level Determination.

D. Rapidly depressurize using the Bypass Jack per EP-2.

Answer: B

Question Comments: Given there is a 7 inch difference between Division 1 and 2 Suppression Pool level instruments and a loss of power to the Division 1 Suppression Pool level instrument reference leg Division 1 Suppression Pool Level instrument is incorrect. Suppression Pool level on the correct reading level instrument is below 14.25 ft per Caution 2 of the Emergency Procedures Suppression Pool Temperature Instruments are suspect and since they are reading the same temperature as Containment Air temperature the Suppression Pool Temperature readings are inaccurate. Since Answer B is the only answer listing these two items it is the only correct answer. 10CFR 41.7; 41.10; 43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00912

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory **Difficulty Rating:** 2

Objectives:

1. CourseID: GG-1-LP-RO-EP03 Objective: 3, 6

KA References:

1. 295030 EA2.02: 3.9/3.9
2. Generic 2.4.3: 3.5/3.8

References:

1. EP caution 2

2. PSTG Appendix B Caution 2
3. 04-1-01-E21-1 step 3.8
4. ARI 04-1-02-1H13-P601-21A-C7 step 4.1.3

TrainingPrograms:

Systems:

Categories:

Task References:

Question Last Revised By: MikeRasch at Tue Aug 09 11:33:10 CDT 2005
Question History:

1. Created by jbell at Thu Jun 16 14:18:53 CDT 2005
2. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
3. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
4. Modified by mrasch at Fri Jul 29 12:29:54 CDT 2005
5. Modified by mellis at Fri Jul 29 14:30:45 CDT 2005
6. Modified by mrasch at Fri Jul 29 14:50:32 CDT 2005
7. Question Reviewed by mellis at Thu Aug 04 07:27:53 CDT 2005
8. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
9. Modified by mrasch at Tue Aug 09 11:33:10 CDT 2005
10. Question Reviewed by mellis at Tue Aug 09 12:00:03 CDT 2005
11. Question used on ExamName: NRC August 2005 - 2 ExamID: NRC-082005-2 Exam Date: 08/12/2005

Comments:

EB QUESTION: 81 **(1.0 Points)**

The plant is in day seven of a refueling outage.

Secondary Containment is NOT required operable.

During Core Alterations, a malfunction of the Refueling Bridge has caused a fuel damaging event in the containment fuel racks.

Which one of the following actions is required to limit the offsite release rate?

A. Close the 208' containment airlock

B.

Close the containment equipment hatch

C. Start Fuel Pool Sweep Ventillation system

D. Close any open Secondary Containment doors

Answer: D

Question Comments: Conduct of Operations states during a fuel handling accident to start Standby Gas Treatment, establish secondary containment and perform the secondary containment verification surveillance to ensure all penetrations are closed since during an outage work can be in progress with these penetrations open. Primary Containment is NOT required to be Operable per Technical Specifications in modes 4 or 5. Secondary Containment is NOT required to be Operable per Technical Specifications due to being greater than 24 hours after shutdown. Answers A, B, and C are not required per Conduct of Operations, 01-S-06-2, and the High Radiation during Fuel Handling ONEP, 05-1-02-II-8. Answer D is CORRECT because Conduct of Operations, 01-S-06-2, and High Radiation During Fuel Handling, 05-1-02-II-8, require closing openings in Secondary Containment. 10CFR 41.10/43.2/43.4/43.5/43.7

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00913

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-PROC Objective: 8.29
2. CourseID: GLP-OPS-ONEP Objective: 2
3. CourseID: GLP-RF-F1105 Objective: 3.7

KA References:

1. 295038 Generic 2.2.28: 2.6/3.5

References:

1. 05-1-02-II-8 steps 3.2, 3.3
2. 01-S-06-2 steps6.7.26c, 6.7.23
3. Tech Specs 3.6.1.1; 3.6.1.2; 3.6.4.1 and bases

4. FSAR 15.7.4

TrainingPrograms:

Systems:

Categories:

Task References:

Question Last Revised By: MikeRasch at Fri Jul 29 12:31:13 CDT 2005
Question History:

1. Created by jbell at Thu Jun 16 14:22:52 CDT 2005
2. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
3. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
4. Modified by mellis at Tue Jul 26 11:03:03 CDT 2005
5. Modified by mrasch at Fri Jul 29 12:31:13 CDT 2005
6. Question Reviewed by mellis at Thu Aug 04 07:28:00 CDT 2005
7. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
8. Question used on ExamName: NRC August 2005 - 2 ExamID: NRC-082005-2 Exam Date: 08/12/2005

Comments:

EB QUESTION: 82 (1.0 Points)

There is a fire in the Control Room.

Which one of the following describes the basis for separating the Main Control Room from the Remote Shutdown Panels?

A. Essential equipment for both divisions in the Main Control Room is isolated to prevent a fire from causing equipment malfunctions preventing placing the plant in cold shutdown.

B. All Division I equipment associated with Remote Shutdown and systems required to place the plant in cold shutdown can be isolated from the Main Control Room controls.

C. All Division II equipment associated with Remote Shutdown and systems required to place the plant in cold shutdown can be isolated from the Main Control Room controls.

D. All Division I equipment and selected Division II equipment associated with Remote Shutdown and systems required to place the plant in cold shutdown can be isolated from the Main Control Room controls.

Answer: B

Question Comments: GGNS design utilized Division I Remote Shutdown Panel as the Safe Shutdown Systems for placing the unit in Cold Shutdown following a Fire in the Main Control Room that could potentially compromise control of systems required to maintain the reactor in a safe cold shutdown condition. 10CFR50 Appendix R sections III.G and III.L require only one train (division). Answers A, C, and D include Division II equipment making them INCORRECT. Answer B is CORRECT because it identifies the correct division and reason for separating Remote Shutdown Systems from the Main Control Room. 10CFR41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00914

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-C6100 Objective: 3; 9; 10; 11

KA References:

1. 600000 AA2.16: 3.0/3.5

References:

1. UFSAR 7.4.1.5.1.1
2. 05-1-02-II-1 Attachment IV

TrainingPrograms:

Systems:

Categories:

Task References:

Question Last Revised By: MikeRasch at Fri Jul 29 12:43:10 CDT 2005
Question History:

1. Created by jbell at Thu Jun 16 14:26:04 CDT 2005
2. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
3. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
4. Modified by mrasch at Fri Jul 29 12:43:10 CDT 2005
5. Question Reviewed by mellis at Thu Aug 04 07:28:05 CDT 2005
6. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
7. Question used on ExamName: NRC August 2005 - 2 ExamID: NRC-082005-2 Exam Date: 08/12/2005

Comments:

EB QUESTION: 83 (1.0 Points)

The plant is in Mode 2 during startup, preparing to transfer to Mode 1.

PCW SPLY TO SMPL WTR CLRS/CTMT CLRS valve P71-F150 has failed closed due to a blown control power fuse.

Average Containment Temperature is 97 degrees F.

Transferring to Mode 1 is:

Tech Specs are provided.

A. Not allowed only due to containment temperature.

B. Not allowed only due to inoperability of P71-F150.

C. Allowed only within 8 hours of containment temperature exceeding the limit.

D. Allowed provided containment temperature is returned within limits within 8 hours of entering Mode 1.

Answer: A

Question Comments: Tech Spec 3.6.1.5 is out of spec and is not 3.0.4 exempt for plant mode change. Tech Spec 3.6.1.3 is met for P71-F150 because the valve is closed. Therefore Answer A is the only correct answer. Tier 1 Group 2 This is a NEW question. 10CFR 43.2

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00915

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-OPS-TS001 Objective: 34

KA References:

1. 295011 AA2.01: 3.6/3.9
2. Generic 2.1.12: 2.9/4.0

References:

1. Tech Spec 3.6.1.5
2. Tech Spec 3.6.1.3
3. Tech Spec 3.0.4

TrainingPrograms:

1. Senior Reactor Operator Training Program
2. Licensed Operator Requalification Training Program

Systems:

Categories:

1. Technical Specifications

Task References:

Question Last Revised By: MikeRasch at Fri Jul 29 07:16:12 CDT 2005
Question History:

1. Created by jbell at Thu Jun 16 14:29:08 CDT 2005
2. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
3. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
4. Modified by mrasch at Fri Jul 29 07:16:12 CDT 2005

5. Question Reviewed by mellis at Thu Aug 04 07:28:09 CDT 2005
6. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
7. Question used on ExamName: NRC August 2005 - 2 ExamID: NRC-082005-2 Exam Date: 08/12/2005

Comments:

EB QUESTION: 84 (1.0 Points)

The plant was operating at 100% power when a control rod drift annunciator was received.

The ACRO determined that a single control rod was drifting out of the core from position 12 to position 30.

He selected the control rod and attempted to insert it.

An RC&IS malfunction prevented the control rod from fully inserting so core flow was reduced to 67 Mlbm/hr.

Cyclops shows NO challenges to Thermal Limits, but reactor power did rise by 0.6% before lowering Recirc Flow.

What level of Reactivity Management Event/Precursor is this classified as and who at a minimum is expected to be notified of the occurrence?

NMM OP-103 is provided.

A. The event is classified as a level 3 reactivity event. Operations Management and Reactor Engineering are notified.

B. The event is classified as a level 4 precursor. Reactor Engineering and NRC Region IV Headquarters are notified.

C. The event is classified as a level 3 reactivity event. Operations Management and NRC Region IV Headquarters are notified.

D. The event is classified as a level 4 precursor. Operations Management and Reactor Engineering are notified.

Answer: A

Question Comments: Subsequent actions of the ONEP for Control Rod/Drive Malfunctions for a control rod drift requires notification of Reactor Engineering for analysis of the core conditions with an out of position control rod. Operations Philosophy Operations Section Procedure requires analysis by Reactor Engineering. Entergy Policy PL-163 section 2.2 standards requires the notification of Operations Management and Reactor Engineering if abnormal reactivity changes occur in the core. Entergy Nuclear Management Manual Procedure OP-103 Reactivity Management sets this as a Level 3 Reactivity Management Event since it is an unintended power rise by greater than 0.5 percent. It is NOT a Level 4 Reactivity Management Precursor because this is the lesser severity of reactivity management event. This event does not reach the level to make notification to the NRC Regional Headquarters. It would include a courtesy call to the NRC Resident Inspector. Answer A is CORRECT because it is the only answer identifying the correct notifications and level of reactivity management event. 10CFR41.10/43.5/43.6

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00916

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-PROC Objective: 59.3; 74.3; 76.2
2. CourseID: GLP-OPS-ONEP Objective: 31

KA References:

1. 295014 Generic 2.1.14: 2.5/3.3
2. Generic 2.1.6: 2.1/4.3

References:

1. 02-S-01-27 section 6.5.3
2. ONEP 05-1-02-IV-1 section 3.2.3
3. NMM Policy ENS-PL-163 Attachment 9.1 section 2.2 item 19
4. NMM OP-103 section 5.3.3 and Attachment 9.1 Level 3 bullet 3.

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Shift Manager Training Program
4. Licensed Operator Requalification Training Program

Systems:

Categories:

1. Administrative Requirements
2. Off Normal Event Procedures
3. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Tue Aug 09 11:36:07 CDT 2005
Question History:

1. Created by jbell at Thu Jun 16 14:33:17 CDT 2005
2. Modified by mrasch at Mon Jun 20 12:38:21 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
4. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
5. Modified by tharrelso at Thu Jul 28 07:02:17 CDT 2005
6. Modified by mrasch at Fri Jul 29 12:36:33 CDT 2005
7. Question Reviewed by mellis at Thu Aug 04 07:28:14 CDT 2005
8. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
9. Modified by mrasch at Tue Aug 09 11:36:07 CDT 2005
10. Question Reviewed by mellis at Tue Aug 09 12:00:05 CDT 2005
11. Question used on ExamName: NRC August 2005 - 2 ExamID: NRC-082005-2 Exam Date: 08/12/2005

Comments:

EB QUESTION: 85 **(1.0 Points)**

The plant is at 100% power when secondary containment instrument air isolation valve P53-F026B fails closed due to loss of control power and CANNOT be re-opened.

The first action that should be directed by the Control Room Supervisor for this condition is:

A. start Reactor Core Isolation Cooling (RCIC) in anticipation of entry into the Loss of Feedwater Flow ONEP due to Main Steam Isolation Valve closure.

B. attempt to restore power to P53-F026B in accordance with the Loss of AC Power ONEP.

C. scram the reactor in anticipation of entry into the Control Rod Drive Malfunctions ONEP due to multiple control rod drifts/scrams.

D. enter EP-2 and lower reactor pressure to 450-600 psig in anticipation of losing the main condenser as a heat sink.

Answer: C

Question Comments: When air is isolated to the containment it is also lost to the Scram Air header this allows slow depressurization of the header and control rods to begin to drift/scram due to the scram inlet and outlet valves opening. Therefore Answer C is the correct answer for the given conditions. Tier 1 Group 2 This is a NEW question. 10CFR 41.10/43.2/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00917

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-ONEP Objective: 1

KA References:

1. 295020 AA2.03: 3.7/3.7
2. Generic 2.4.48: 3.5/3.8

References:

1. 05-1-02-V-9
2. 05-1-02-IV-1

TrainingPrograms:

1. Senior Reactor Operator Training Program
2. Licensed Operator Requalification Training Program

Systems:

Categories:

1. Off Normal Event Procedures

Task References:

Question Last Revised By: MikeRasch at Fri Jul 29 12:38:16 CDT 2005
Question History:

1. Created by jbell at Thu Jun 16 14:38:07 CDT 2005
2. Modified by mrasch at Mon Jun 20 14:21:57 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
4. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
5. Modified by mrasch at Fri Jul 29 07:28:33 CDT 2005
6. Modified by mrasch at Fri Jul 29 12:38:16 CDT 2005
7. Question Reviewed by mellis at Thu Aug 04 07:28:18 CDT 2005
8. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
9. Question used on ExamName: NRC August 2005 - 2 ExamID: NRC-082005-2 Exam Date: 08/12/2005

Comments:

EB QUESTION: 86 (1.0 Points)

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

EP2 and EP3 are being executed.

The only available ECCS systems are RHR A and B.

Current parameters are as follows:

Reactor level is - 194 inches compensated fuel zone and falling.

Pressure Suppression Pressure (PSP) is in the unsafe region.

Both loops of Suppression Pool Cooling are in service, but not maximized.

Reactor Core Isolation Cooling (RCIC), Standby Liquid Control (SLC) and Control Rod Drive (CRD) are injecting at their maximum rates.

How should the CRS direct use of RHR Systems to limit the offsite release rate under these conditions?

A.

Execute EP-3 and direct initiation of both loops of Containment Spray.

B. Execute EP-3 and direct initiation of one loop of Containment Spray, and leave one loop of Suppression Pool Cooling in service.

C. Execute EP-3 and direct maximizing both loops of Suppression Pool Cooling in service.

D. Execute EP-2 and direct alignment of RHR A and B for injection into the Reactor.

Answer: D

Question Comments: Background for ECCS This question involves the difference between use of RHR for control of Containment parameters to prevent a challenge to the Containment structure and the control of RPV parameters to prevent the degradation of the first fission product barrier. EP-3 step 8 says to initiate those loops of RHR not required for adequate core cooling. With RPV level at -194 inches and falling all ECCS will be aligned to the RPV until no longer needed. Suppression Pool temperature, while above the point to need Suppression Pool cooling, is not critical. RHR is still required for the adequate core cooling issue. Once adequate core cooling is assured, other priorities can be established. Answer D is the only answer to align ECCS Systems (RHR 'A' and 'B' for LPCI Injection mode). Answers A, B, and C align RHR 'A' and 'B' for Containment Spray and Suppression Pool Cooling which are lower priorities than recovering adequate core cooling. This is a NEW question. 10CFR41.8/41.10/43.2/43.4/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00918

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GG-1-LP-RO-EP02 Objective: 4, 11
2. CourseID: GG-1-LP-RO-EP03 Objective: 2, 3, 6

KA References:

1. 203000 Generic 2.3.11: 2.7/3.2
2. Generic 2.4.22: 3.0/4.0

References:

1. TS Bases 3.5.1 Background
2. EP-2
3. EP-3
4. PSP curve,
5. CSIPL curve, PSTG pgs B-7-7

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Shift Manager Training Program
4. Licensed Operator Requalification Training Program
5. Shift Technical Advisor Training Program

Systems:

1. E12: Residual Heat Removal System

Categories:

1. Emergency Procedure Training
2. FSAR
3. Integrated Plant Operations
4. Mitigation of Core Damage
5. Technical Specifications
6. Continuing Training
7. STA Training

Task References:

Question Last Revised By: Mickey Ellis at Fri Jul 29 13:58:09 CDT 2005
Question History:

1. Created by jbell at Thu Jun 16 14:42:50 CDT 2005
2. Modified by mrasch at Mon Jun 20 12:52:50 CDT 2005
3. Modified by mrasch at Mon Jun 20 13:03:33 CDT 2005
4. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Modified by tharrelso at Thu Jul 28 11:00:12 CDT 2005
7. Modified by mrasch at Fri Jul 29 12:41:38 CDT 2005
8. Modified by mrasch at Fri Jul 29 12:44:41 CDT 2005
9. Modified by mellis at Fri Jul 29 13:58:09 CDT 2005

10. Question Reviewed by mrasch at Thu Aug 04 07:29:16 CDT 2005
11. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
12. Question used on ExamName: NRC August 2005 - 2 ExamID: NRC-082005-2 Exam Date: 08/12/2005

Comments:

EB QUESTION: 87 (1.0 Points)

The plant is in Mode 2 at the point of adding heat.

I&C is removing a chart recorder in 1H13-P629, when incidental electrical contact inside the panel causes Low Pressure Core Spray to initiate and inject into the reactor.

The resulting power excursion causes an IRM scram.

Notification of this event to the NRC Operations Center is required within:

(01-S-06-5 Attachment III is provided)

A. 1 hour.

B. 4 hours.

C. 8 hours.

D. 24 hours.

Answer: B

Question Comments: Answer B is correct since valid RPS actuations with the reactor critical are 4 hour reports per 10 CFR 50.72(b)(2)(iv)(B). Answer C is incorrect because a 8 hour report is required only for RPS initiations when the Reactor is NOT critical. Answers A and D are incorrect since this does not fall into one of the events listed for 1 hr or 24 hr reports. SRO only 10CFR 41.10/43.5

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00919

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-PROC Objective: 11.13

KA References:

1. 209001 Generic 2.1.15: 2.3/3.0

References:

1. 01-S-06-5 Att. III
2. 10 CFR 50.72(b)(2)(iv)(B)

TrainingPrograms:

Systems:

Categories:

Task References:

Question Last Revised By: MikeRasch at Tue Aug 09 11:41:20 CDT 2005

Question History:

1. Created by jbell at Thu Jun 16 14:45:59 CDT 2005
2. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
3. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
4. Modified by mrasch at Fri Jul 29 12:45:16 CDT 2005
5. Modified by mellis at Fri Jul 29 15:11:44 CDT 2005
6. Modified by mellis at Fri Jul 29 15:16:11 CDT 2005
7. Question Reviewed by mrasch at Thu Aug 04 07:29:16 CDT 2005
8. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
9. Modified by mrasch at Tue Aug 09 11:41:20 CDT 2005
10. Question Reviewed by mellis at Tue Aug 09 12:00:06 CDT 2005
11. Question used on ExamName: NRC August 2005 - 2 ExamID: NRC-082005-2 Exam Date: 08/12/2005

Comments:

EB QUESTION: 88 (1.0 Points)

The plant had been operating at 100% power when a scram occurred due to failure of an optical isolator causing a Recirculation Pump double downshift.

The Control Room Supervisor (CRS) has received the scram report and entered EP-2 due to reactor water level.

Reactor water level is + 18 inches narrow range and stable.

The CRS has made NO assessments other than what was provided in the scram report.

NO Reactor Scram ONEP subsequent actions have been performed.

Which one of the following is proper with respect to procedure execution and hierarchy in this situation?

A. Since reactor water level is back above +11.4 inches, EP-2 should be exited immediately. Then, the CRS should enter the Reactor Scram ONEP and direct subsequent actions.

B. The CRS should complete the primary control loop in EP-2. The CRS should direct complimentary subsequent actions from the Reactor Scram ONEP concurrently during EP-2 execution.

C. The CRS should complete the primary control loop in EP-2. Then, if NO emergency exists, he should direct performance of subsequent actions from the Reactor Scram ONEP only in the order listed in the ONEP.

D. The CRS should complete the primary control loop in EP-2. The ACRO may insert IRMs without CRS direction or permission as soon as he completes all Reactor Scram ONEP immediate actions.

Answer: B

Question Comments: Answer A is INCORRECT because to exit the Emergency Procedures a determination must be made per 05-S-01-EP-2 section 2.3 that an emergency NO longer exists and that is not stated in the answer. Answer B is CORRECT because it ensures control is established per EP-2 then allows other actions that are not in contradiction to the EPs to be performed. In addition, 02-S-01-2 allows ONEP subsequent actions to be performed out of sequence at the discretion of an SRO . Answer C is

INCORRECT because subsequent actions of an ONEP are not required to be performed in order if an ON-Shift SRO authorizes out of order performance. Answer D is INCORRECT because the CRS gives the directions for action during the execution of the EPs and all other procedures are subordinate to the EPs. 10CFR 41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00920

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-PROC Objective: 49.5
2. CourseID: GG-1-LP-RO-EP02 Objective: 4, 9
3. CourseID: GLP-OPS-ONEP Objective: 3.0

KA References:

1. 215003 2.4.16: 3.0/4.0

References:

1. 02-S-01-2 step 6.9.6d
2. 05-S-01-EP-2 steps 2.3, 2.4

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Shift Manager Training Program
4. Shift Technical Advisor Training Program

Systems:

Categories:

1. Administrative Requirements
2. Emergency Procedure Training
3. Integrated Plant Operations
4. Off Normal Event Procedures
5. STA Training

Task References:

Question Last Revised By: Mickey Ellis at Fri Jul 29 14:00:10 CDT 2005

Question History:

1. Created by jbell at Thu Jun 16 14:49:34 CDT 2005
2. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
3. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
4. Modified by tharrelso at Thu Jul 28 09:53:14 CDT 2005
5. Modified by mrasch at Fri Jul 29 12:46:28 CDT 2005
6. Modified by mellis at Fri Jul 29 13:46:57 CDT 2005
7. Modified by mellis at Fri Jul 29 14:00:10 CDT 2005
8. Question Reviewed by mrasch at Thu Aug 04 07:29:17 CDT 2005
9. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
10. Question used on ExamName: NRC August 2005 - 2 ExamID: NRC-082005-2 Exam Date: 08/12/2005

Comments:

EB QUESTION: 89 (1.0 Points)

Source Range Monitor (SRM) A failed to respond greater than 300 cps during plant startup.

I&C has reworked the connector where the detector signal cable meets the SRM A drawer.

Only the flux signal to the SRM was affected (e.g. detector position function was unaffected).

The plant startup is still in progress, and Intermediate Range Monitors are on range 7.

Which one of the following would satisfy the minimum retest requirements for SRM A to return it to operable status for ALL plant modes?

(Do NOT consider whether plant conditions would support a particular re-test.)

Technical Specifications 3.3.1.1; 3.3.1.2 and TR 3.3.2.1 are provided.

A. SR 3.3.1.1.5 and SR 3.3.1.2.1 and SR TR 3.3.2.1.3 and SR TR 3.3.2.1.9

B. SR 3.3.1.1.5 and SR 3.3.1.2.4 and SR TR 3.3.2.1.3 and SR TR 3.3.2.1.9

C. SR 3.3.1.1.5 and SR 3.3.1.2.1 and SR 3.3.1.2.4 and SR TR 3.3.2.1.3

D. SR 3.3.1.2.1 and SR 3.3.1.2.4 and SR TR 3.3.2.1.3 and SR TR 3.3.2.1.9

Answer: D

Question Comments: To declare an SRM operable it must pass the surveillance requirements for Channel Checks, Channel Calibration, Channel Functional Test and a minimum count rate. SRM/IRM overlap surveillance per 3.3.1.1 is not required for SRM operability it is for IRM operability. Answer A is INCORRECT because it does not include a Minimum Count Rate and a Channel Calibration. SRM /IRM overlap is not required for SRM Operability. Answer B is INCORRECT because it does not include a Channel Check and Minimum Count Rate. SRM /IRM overlap is not required for SRM Operability. Answer C is INCORRECT because it does not include a Minimum Count Rate and a Channel Calibration. SRM /IRM overlap is not required for SRM Operability. Answer D is CORRECT because it includes the necessary Channel Checks, Calibration, Functional Test and Minimum Count Rate checks. SRM /IRM overlap is not required for SRM Operability. 10CFR 41.10/43.2/43.5

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00921

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 3

Objectives:

1. CourseID: GLP-OPS-TS001 Objective: 3.3; 4.2; 4.3; 4.4; 4.11; 13;

KA References:

1. 215004 Generic 2.2.21: 2.3/3.5

References:

1. Tech Spec SR 3.3.1.1.5, SR 3.3.1.2.1 through 3.3.1.2.6, TRM table TR3.3.2.1-2,
2. Tech Spec 1.0 Definitions – Channel Check; Channel Calibration; Channel Functional Test, Operable-Operability
3. LCO3.0.5
4. SR3.0.1

TrainingPrograms:

Systems:

Categories:

Task References:

Question Last Revised By: MikeRasch at Tue Aug 09 11:55:32 CDT 2005
Question History:

1. Created by jbell at Thu Jun 16 14:53:41 CDT 2005
2. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
3. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
4. Modified by mrasch at Fri Jul 29 12:47:49 CDT 2005
5. Question Reviewed by mellis at Thu Aug 04 07:30:28 CDT 2005
6. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
7. Modified by mrasch at Tue Aug 09 11:51:23 CDT 2005
8. Modified by mrasch at Tue Aug 09 11:55:32 CDT 2005
9. Question Reviewed by mellis at Tue Aug 09 12:00:07 CDT 2005
10. Question used on ExamName: NRC August 2005 - 2 ExamID: NRC-082005-2 Exam Date: 08/12/2005

Comments:

EB QUESTION: 90 (1.0 Points)

The plant is in Mode 2 with reactor pressure 125 psig.

An I&C technician just reported he discovered a bad relay in the RCIC actuation logic.

Subsequently, RCIC was declared inoperable and HPCS was verified to be operable.

In this situation, the Technical Specifications:

A. require the reactor to be placed in Mode 3 within 12 hours.

B. allow the startup to continue, but exceeding 150 psig is not permitted.

C. allow the startup to continue to normal operating pressure, but entering Mode 1 is not permiteed.

D.

allow the startup to continue into Mode 1, however RCIC must be restored within 14 days or LCO condition 3.5.5.B must be entered.

Answer: B

Question Comments: Answer B is the only correct answer because RCIC may be out of service in this condition as long as RPV pressure is less than 150 psig. Since RCIC is required to be operable when RPV pressure is above 150 psig the change is not allowed. Tier 2 Group 1 This is a NEW question. 10CFR 41.10/43.5

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00922

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-TS01 Objective: 34

KA References:

1. 217000
2. GENERIC 2.1.10 Knowledge of conditions and limitations in the facility license [2.7/3.9]

References:

1. Tech Spec 3.5.3
2. Tech Spec 3.0.4

TrainingPrograms:

1. Senior Reactor Operator Training Program
2. Licensed Operator Requalification Training Program

Systems:

1. E51: Reactor Core Isolation Cooling System

Categories:

1. Integrated Plant Operations
2. Technical Specifications

Task References:

Question Last Revised By: MikeRasch at Fri Jul 29 14:07:38 CDT 2005
Question History:

1. Created by jbell at Thu Jun 16 14:56:39 CDT 2005
2. Modified by mrasch at Mon Jun 20 13:12:37 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
4. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
5. Modified by mrasch at Fri Jul 29 12:49:38 CDT 2005
6. Modified by mrasch at Fri Jul 29 14:07:38 CDT 2005
7. Question Reviewed by mellis at Thu Aug 04 07:30:28 CDT 2005
8. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
9. Question used on ExamName: NRC August 2005 - 2 ExamID: NRC-082005-2 Exam Date: 08/12/2005

Comments:

EB QUESTION: 91 **(1.0 Points)**

A large break LOCA with failure of Divisions 1 and 2 ECCS has occurred.

Emergency Depressurization has been conducted.

High Pressure Core Spray is injecting at rated flow, but it is determined that reactor water level is unable to be restored above -192 inches.

Reactor water level is currently -210 inches and slowly lowering.

Severe Accident Procedure (SAP) 3 is entered.

Which one of the following is the reason why SAP-3 is entered?

A. Core damage is occurring requiring the Containment to be flooded and these actions are described in the SAPs.

B. Emergency Procedures have insufficient systems allocated for restoration of RPV level.

C. Conditions in the reactor require venting of the vessel to facilitate RPV flooding.

D. Reactor conditions fail to assure adequate core cooling.

Answer: D

Question Comments: Emergency Procedures are based upon maintaining adequate core cooling. Severe Accident Procedures are based upon conditions no longer assure adequate core cooling and the focus is to shift operations to ensure Primary Containment is not challenged. Answer A is INCORRECT because core damage is not indicated per definitions in the given information at this time. Answer B. is INCORRECT because the same systems are listed for use in RPV level control in both the EPs and SAPs. Answer C is INCORRECT because RPV Flooding is contained on EP-2. Answer D is CORRECT because the entry conditions of 05-S-01-SAP-1 state that adequate core cooling cannot be assured and with the given conditions adequate core cooling is not assured. 10CFR41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00923

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-EP-EPT19 Objective: 7

KA References:

1. 290002 Generic 2.4.14: 3.0/3.9

References:

1. 05-S-01-SAP-1 step 2.1

TrainingPrograms:

Systems:

Categories:

Task References:

Question Last Revised By: MikeRasch at Thu Aug 04 08:15:37 CDT 2005
Question History:

1. Created by jbell at Thu Jun 16 14:58:47 CDT 2005
2. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
3. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
4. Modified by mrasch at Fri Jul 29 12:51:27 CDT 2005
5. Question Reviewed by mellis at Thu Aug 04 07:30:29 CDT 2005
6. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
7. Modified by mrasch at Thu Aug 04 08:15:37 CDT 2005
8. Question Reviewed by mellis at Thu Aug 04 08:17:57 CDT 2005
9. Question used on ExamName: NRC August 2005 - 2 ExamID: NRC-082005-2 Exam Date: 08/12/2005

Comments:

EB QUESTION: 92 (1.0 Points)

The plant is in Mode 1.

Containment Spray A time delay relay E12-K93A has been declared inoperable due to it failing to actuate during an electrical surveillance and the relay has been removed for troubleshooting.

An alarm and status light are received on 1H13-P601 indicating CTMT SPRB SPARGER INL VLV E12-F028B control power fuses have blown.

What is the most limiting impact of these conditions regarding RHR Containment Spray?

Tech Specs and 02-S-01-17 Att II are provided.

A. Containment Spray A will not automatically initiate; the inoperable channel must be placed into trip within 24 hours and E12-F028B restored within 7 days.

B. Containment Spray A can be considered OPERABLE via manual initiation capability; E12-F028B must be restored within 7 days.

C. Containment Spray A will not automatically initiate; Mode 3 must be entered within 20 hours.

D. Containment Spray A will not automatically initiate; Mode 3 must be entered within 21 hours.

Answer: D

Question Comments: This involves a safety function determination for instrumentation. 02-S-01-17 says if a safety function determination involves instrumentation, take action per the instrumentation Technical Specification. Here, automatic initiation capability for containment spray has been lost due to a non-functional relay on Div 1 and a non-functional spray valve on Div 2. The appropriate action would be TS 3.3.6.3 action B.1, which requires, after 1 hr, declaring both sprays inop. TS 3.6.1.7 requires, after 8 hours, entering a 12 hr shutdown statement for 2 inop sprays. Thus 1+8+12=21 which is action d. It would be inappropriate to declare both sprays inop immediately, as represented by answer c. All other answers are wrong since they state times other that 21 hrs. Tier 2 Group 2 This is a NEW question 10CFR 43.2

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00924

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-E1200 Objective: 8.11, 10.1, 10.2, 13.2

KA References:

1. 226001 A2.13 Valve logic failure [2.8/2.9]

References:

1. E-1181-40; 42; 44; 68; 69
2. Tech Specs 3.3.6.3
3. Tech Specs 3.6.1.7

TrainingPrograms:

1. Senior Reactor Operator Training Program
2. Licensed Operator Requalification Training Program

Systems:

Categories:

1. Technical Specifications

Task References:

Question Last Revised By: MikeRasch at Tue Aug 09 11:57:41 CDT 2005
Question History:

1. Created by jbell at Thu Jun 16 15:01:13 CDT 2005
2. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
3. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
4. Modified by mrasch at Fri Jul 29 08:18:52 CDT 2005
5. Modified by mrasch at Fri Jul 29 12:52:42 CDT 2005
6. Question Reviewed by mellis at Thu Aug 04 07:30:30 CDT 2005
7. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
8. Modified by mrasch at Tue Aug 09 11:57:41 CDT 2005
9. Question Reviewed by mellis at Tue Aug 09 12:00:08 CDT 2005
10. Question used on ExamName: NRC August 2005 - 2 ExamID: NRC-082005-2 Exam Date: 08/12/2005

Comments:

EB QUESTION: 93 **(1.0 Points)**

The Refueling Floor SRO must grant permission to bypass a Refueling Platform interlock manifested by which one of the following Refueling Platform Interlock Display indicating lights?

A. Safety Travel interlock

B. Backup Hoist interlock

C. Rod Block Interlock No.1

D. Bridge Reverse Stop No.1

Answer: A

Question Comments: Fuel handling precaution and limitation specifies the travel interlock as requiring Refuel SRO permission to bypass. Answer A is the correct answer. Tier 3 10CFR 43.2/43.7

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00925

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory **Difficulty Rating:** 2
Objectives:

1. CourseID: GLP-RF-F1101 Objective: 15.3

KA References:

1. 234000 A2.01 Interlock failure [3.3/3.7]

References:

1. 04-1-01-F11-1 step 3.19

TrainingPrograms:

1. Senior Reactor Operator Training Program
2. Licensed Operator Requalification Training Program

Systems:

1. F11: Reactor Refueling Equipment

Categories:

1. Administrative Requirements
2. Systems

Task References:

Question Last Revised By: MikeRasch at Fri Jul 29 12:54:23 CDT 2005
Question History:

1. Created by jbell at Thu Jun 16 15:04:03 CDT 2005
2. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
3. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
4. Modified by mrasch at Thu Jul 28 18:13:07 CDT 2005
5. Modified by mrasch at Fri Jul 29 12:54:23 CDT 2005
6. Question Reviewed by mellis at Thu Aug 04 07:30:30 CDT 2005
7. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
8. Question used on ExamName: NRC August 2005 - 2 ExamID: NRC-082005-2 Exam Date: 08/12/2005

Comments:

EB QUESTION: 94 (1.0 Points)

The plant is in Mode 5 with core alterations in progress.

A major evolution involving use of the Polar crane is scheduled.

Who must the Refueling Floor Supervisor notify before the evolution may begin?

A. The Shift Supervisor and the Outage Director

B. The Shift Manager and the Outage Director

C. The Shift Manager and the Refueling Floor SRO

D. The Shift Supervisor and the Refueling Floor SRO

Answer: C

Question Comments: 01-S-06-2, Section 6.7.5 requires the Refueling Floor Supervisor to notify the Shift Manager of any refueling floor major activity. When core alterations are in progress, he must also notify the Refueling Floor SRO therefore, Answer C is CORRECT. Answer A is INCORRECT because neither of these positions is required to be notified. Answer B is INCORRECT because only one of thses positions must be notified. Answer D is INCORRECT because only one of thses positions must be notified. 10CFR 41.10/43.5

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00926

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 3

Objectives:

1. CourseID: GLP-OPS-PROC Objective: 55.4

KA References:

1. K/A Generic 2.1.2: 3.0/4.0

References:

1. 01-S-06-2 sections 6.7.5

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Shift Manager Training Program
4. Licensed Operator Requalification Training Program

Systems:

Categories:

1. Administrative Requirements
2. Industrial Safety
3. Operating Experience
4. Refueling Training
5. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Fri Jul 29 12:55:20 CDT 2005
Question History:

1. Created by jbell at Thu Jun 16 15:06:04 CDT 2005
2. Modified by mrasch at Mon Jun 20 13:14:29 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
4. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
5. Modified by tharrelso at Thu Jul 28 07:24:58 CDT 2005
6. Modified by mrasch at Fri Jul 29 12:55:20 CDT 2005
7. Question Reviewed by mellis at Thu Aug 04 07:30:31 CDT 2005
8. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
9. Question used on ExamName: NRC August 2005 - 2 ExamID: NRC-082005-2 Exam Date: 08/12/2005

Comments:

EB QUESTION: 95 **(1.0 Points)**

The plant is in Mode 1.

Standby Service Water Pump A is tagged out of service for planned maintenance when RHR B Jockey Pump trips.

What is the latest time until the plant must be in Mode 3 for this condition?

Tech Specs are provided.

A. 12 hours

B. 13 hours

C. 20 hours

D. 84 hours

Answer: B

Question Comments: This is a NEW question. Tier 3 10 CFR41.10/43.3/43.5

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00927

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 3

Objectives:

1. CourseID: GLP-OPS-TS001 Objective: 34; 35

KA References:

1. GENERIC 2.1.12 Ability to apply technical specifications for a system [2.9/4.0]

References:

1. Tech Specs 3.5.1; 3.6.1.7; 3.6.1.8; 3.6.2.3; 3.7.1
2. 02-S-01-17
3. Tech Specs 3.0.6; 3.0.3

TrainingPrograms:

1. Senior Reactor Operator Training Program
2. Licensed Operator Requalification Training Program

Systems:

Categories:

1. Administrative Requirements

2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Fri Jul 29 12:57:14 CDT 2005
Question History:

1. Created by jbell at Thu Jun 16 15:08:46 CDT 2005
2. Modified by mrasch at Mon Jun 20 05:25:01 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
4. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
5. Modified by mrasch at Thu Jul 28 17:55:30 CDT 2005
6. Modified by mrasch at Fri Jul 29 12:57:14 CDT 2005
7. Question Reviewed by mellis at Thu Aug 04 07:30:32 CDT 2005
8. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
9. Question used on ExamName: NRC August 2005 - 2 ExamID: NRC-082005-2 Exam Date: 08/12/2005

Comments:

EB QUESTION: 96 (1.0 Points)

A surveillance test on a Control Room Air Conditioning Unit indicates the system has significantly degraded but is still operable.

The work to repair the unit will require the unit to be out of service for 32 days.

Which one of the following describes the proper work prioritization?

EN-WM-100 and Tech Spec 3.7.4 are provided.

A. Priority 1 Work on-line.

B. Priority 2 work on-line.

C. Priority 2 work in outage.

D. Priority 3 work in outage

Answer: C

Question The work per the prioritization matrix of EN-WM-100 for safety related

Comments: equipment covered by Tech Specs that is operable and available is Yellow Priority 2 per attachment 9.2 of EN-WM-100 work that would take a component out of service for a period of time that would exceed the TS shutdown requirements must be deferred to an outage. This is a NEW question. 10CFR 41.10/43.5

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00928

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-PROC Objective: 24.4

KA References:

1. 2.2.19: 2.1/3.1

References:

1. EN-WM-100 Attachments 9.1 and 9.2
2. Tech Spec 3.7.4

TrainingPrograms:

1. Senior Reactor Operator Training Program
2. Licensed Operator Requalification Training Program

Systems:

Categories:

1. Administrative Requirements

Task References:

Question Last Revised By: MikeRasch at Fri Jul 29 13:01:47 CDT 2005

Question History:

1. Created by jbell at Thu Jun 16 15:12:44 CDT 2005
2. Modified by mrasch at Mon Jun 20 13:16:38 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
4. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
5. Modified by mrasch at Thu Jul 28 13:14:00 CDT 2005

6. Modified by mrasch at Thu Jul 28 13:16:26 CDT 2005
7. Modified by mrasch at Fri Jul 29 13:01:47 CDT 2005
8. Question Reviewed by mellis at Thu Aug 04 07:30:32 CDT 2005
9. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
10. Question used on ExamName: NRC August 2005 - 2 ExamID: NRC-082005-2 Exam Date: 08/12/2005

Comments:

EB QUESTION: 97 **(1.0 Points)**

A Temporary Alteration will be used to install specialized test equipment in place of a permanent plant recorder for testing activities.

The testing will last one week.

One of the individuals responsible for approving the installation of this Temporary Alteration is the:

A. Control Room Supervisor.

B. Manager, Planning and Scheduling.

C. Manager, System Engineering.

D. Radiation Protection Manager.

Answer: A

Question Comments: The following personnel are required to approve a temporary alteration prior to its installation per 01-S-06-3: Manager, Operations (or designee Assistant Operations Manager - Shift), Shift Manager, Control Room Supervisor. The Manager, System Engineering is a process owner and requests Temporary Alterations but doe not have to approve them. Therefore Answer B is CORRECT. This is a NEW question. 10CFR 41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00929

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-PROC Objective: 9.9

KA References:

1. Generic 2.2.16: 1.9/2.6

References:

1. 01-S-06-3 sections 2.4.1; 2.5; 6.1.6; 6.1.8; 6.1.13; 6.1.14
2. FSAR 13.1.2.3.7; 18.1.13 response a.

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

Categories:

1. Administrative Requirements
2. Industrial Safety
3. Plant Modifications

Task References:

Question Last Revised By: MikeRasch at Tue Aug 09 11:59:08 CDT 2005
Question History:

1. Created by jbell at Thu Jun 16 15:15:42 CDT 2005
2. Modified by mrasch at Mon Jun 20 05:19:07 CDT 2005
3. Modified by mrasch at Mon Jun 20 13:18:19 CDT 2005
4. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
5. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
6. Modified by tharrelso at Thu Jul 28 08:24:29 CDT 2005
7. Modified by mrasch at Fri Jul 29 13:03:19 CDT 2005
8. Modified by mrasch at Fri Jul 29 14:37:39 CDT 2005
9. Modified by mrasch at Fri Jul 29 14:39:52 CDT 2005
10. Modified by mrasch at Fri Jul 29 14:47:59 CDT 2005
11. Question Reviewed by mellis at Thu Aug 04 07:30:33 CDT 2005

12. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
13. Modified by mrasch at Tue Aug 09 11:59:08 CDT 2005
14. Question Reviewed by mellis at Tue Aug 09 12:00:08 CDT 2005
15. Question used on ExamName: NRC August 2005 - 2 ExamID: NRC-082005-2 Exam Date: 08/12/2005

Comments:

EB QUESTION: 98 **(1.0 Points)**

The individual responsible for authorizing the commencement of a release on a Batch Liquid Radwaste Discharge Permit is the:

A. Radwaste Specialist.

B. Chemistry Supervisor.

C. Control Room Supervisor.

D. Shift Manager.

Answer: D

Question Comments: The Shift Manager is the approval signature for Radwaste Discharges per 01-S-08-11. Tier 3 This is a NEW question. 10CFR 41.10/41.13/43.4/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00930

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-OPS-PROC Objective: 36 and 37

KA References:

1. Generic 2.3.6: 2.1/3.1

References:

1. 01-S-08-11

TrainingPrograms:

1. Senior Reactor Operator Training Program

Systems:

Categories:

1. Administrative Requirements

Task References:

Question Last Revised By: MikeRasch at Fri Jul 29 14:13:03 CDT 2005
Question History:

1. Created by jbell at Thu Jun 16 15:18:23 CDT 2005
2. Modified by mrasch at Mon Jun 20 13:23:28 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
4. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
5. Modified by mrasch at Fri Jul 29 13:04:44 CDT 2005
6. Modified by mrasch at Fri Jul 29 14:13:03 CDT 2005
7. Question Reviewed by mellis at Thu Aug 04 07:30:33 CDT 2005
8. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
9. Question used on ExamName: NRC August 2005 - 2 ExamID: NRC-082005-2 Exam Date: 08/12/2005

Comments:

EB QUESTION: 99 **(1.0 Points)**

A plant startup is in progress and all reactor recirc system indications are normal.

When Reactor Recirc Pumps are shifted to fast speed, the reactor operator reports:

The REACTOR PMP A SEAL STG FLO HI/LO alarm has annunciated.

Seal staging flow indicates higher than normal.

Reactor Recirc pump A seal cavity # 1 is 1050 psig.

Reactor Recirc pump B seal cavity # 2 is 1000 psig.

Indicated Reactor Recirc seal temperatures are lower than normal.

Based on this information, the SRO should enter procedure:

A. 05-1-02-III-3, Reduction in Recirculation System Flow Rate because of improper seal cavity differential pressures.

B. 04-1-01-B33-1, Reactor Recirculation System, because of improper seal cavity differential pressures.

C. 04-1-01-C11-1, Control Rod Hydraulic System, because of improper seal staging flow.

D. 04-1-01-B33-1, Reactor Recirculation System, because of lower than normal seal temperatures.

Answer: B

Question Comments: This is an indication of a failure of Seal #1 due to the following RECIRC PMP A SEAL STG FLO HI/LO annunciator being in, seal staging flow indicating high, Seal # 2 pressure approaching Seal # 1 pressure and seal temperatures dropping. If there are signs of seal de-staging the Recirc pump is to be shifted back to slow speed to attempt to restage the seals then return its operation to fast speed. Even if the seals do not restage the pump may continue operation in fast speed allowing the plant to raise power with monitoring of the seals. Answer B is the correct answer. The procedure to be entered to downshift the recirc pumps to attempt to fix the seal staging. This is not a change in recirc flow requiring entry into the ONEP. This is a NEW question. 10CFR 41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00931

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 3

Objectives:

1. CourseID: GLP-OPS-B3300 Objective: 29.1; 29.4

KA References:

1. Generic 2.4.47: 3.4/3.7

References:

1. 04-1-01-B33-1 sect 3.13.1; 4.2.2a(12), (13), (14), (15)
2. ARI 04-1-02-1H13-P680-3A-B5
3. M-1081B
4. M-1078A

TrainingPrograms:

1. Senior Reactor Operator Training Program
2. Licensed Operator Requalification Training Program

Systems:

1. B33: Reactor Recirculation System

Categories:

1. Systems

Task References:

Question Last Revised By: MikeRasch at Thu Aug 04 08:17:26 CDT 2005
Question History:

1. Created by jbell at Thu Jun 16 15:21:23 CDT 2005
2. Modified by mrasch at Mon Jun 20 13:25:42 CDT 2005
3. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
4. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
5. Modified by mrasch at Fri Jul 29 13:07:05 CDT 2005
6. Modified by mrasch at Fri Jul 29 14:27:49 CDT 2005
7. Question Reviewed by mellis at Thu Aug 04 07:30:34 CDT 2005
8. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
9. Modified by mrasch at Thu Aug 04 08:17:26 CDT 2005
10. Question Reviewed by mellis at Thu Aug 04 08:18:01 CDT 2005
11. Question used on ExamName: NRC August 2005 - 2 ExamID: NRC-082005-2 Exam Date: 08/12/2005

Comments:

EB QUESTION: 100 (1.0 Points)

A Reactor Recirc LOCA and Feed Water line break in the drywell has occurred.

Reactor Core Isolation Cooling (RCIC) has tripped due to high RCIC room ambient temperature that occurred five minutes ago and failed to isolate.

RCIC room blowout panel alarms are in on the Fire Computer in the control room.

NO systems listed on Table 1 of the Alternate Level Control leg of EP-2 are available for injection. RHR B Injection Valve E12-F042B has lost power in the closed position.

Control Rod Drive flow and Standby Liquid Control flow are maximized.

The following conditions exist:

Reactor Power 0%
Drywell Pressure 7.6 psig slowly falling
Reactor Pressure 750psig slowly falling
Drywell Temperature 192°F slowly falling
Reactor level -170 inches slowly falling
Drywell Radiation 6R/hr slowly rising
Containment pressure 2.0 psig slowly falling
Containment Temperature 108°F slowly falling
Suppression Pool Temperature 125°F stable
Suppression Pool Level 20.8 ft. stable

Dose commitment at the site boundary is 20 mr/hr TEDE and 130 mr/hr CEDE Thyroid.

What are the Emergency Classification and Protective Action Recommendation (PAR), if any, for this event?

10-S-01-1 is provided.

A. Alert, NO PAR

B. Site Area Emergency, NO PAR

C. General Emergency, Evacuate 2 miles all sectors and 5 miles downwind and shelter the remainder of the 10 mile Emergency Planning Zone

D. General Emergency, Evacuate 2 miles all sectors and 10 miles downwind and shelter the remainder of the 10 mile Emergency Planning Zone

Answer: C

Question Comments: With RPV level below TAF and falling (Potential Loss of Fuel Cladding), high Drywell Pressure/ RCIC Steam line Break outside Containment

(Loss of Reactor Pressure boundary), RCIC has failed to isolate (Loss of Primary Containment), Loss of 2 of 3 fission product barriers with a potential of the 3rd is a General Emergency per EAL 3.4. Due to the offsite radiation levels the Standard PAR is issued for a General Emergency. Answers A and B are INCORRECT because the wrong emergency classification. Answer D is INCORRECT because the wrong PAR is issued for the given radiation conditions. Answer C is CORRECT because the correct emergency classification and PAR are identified. This is a NEW question. 10CFR41.10/43.5

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00932

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis **Difficulty Rating:** 2

Objectives:

1. CourseID: GLP-EP-EPTS6 Objective: 1; 2

KA References:

1. Generic 2.4.44: 2.1/4.0

References:

1. 10-S-01-1 EAL 3.4, 10-S-01-1 step 6.1.4k(1)

TrainingPrograms:

1. Reactor Operator Training Program
2. Senior Reactor Operator Training Program
3. Licensed Operator Requalification Training Program

Systems:

Categories:

1. Emergency Plan Training

Task References:

Question Last Revised By: MikeRasch at Fri Jul 29 13:08:04 CDT 2005
Question History:

1. Created by jbell at Thu Jun 16 15:23:55 CDT 2005
2. Modified by mrasch at Mon Jun 20 13:29:02 CDT 2005

3. Question Reviewed by mrasch at Mon Jun 20 14:44:20 CDT 2005
4. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
5. Modified by mrasch at Fri Jul 29 13:08:04 CDT 2005
6. Question Reviewed by mellis at Thu Aug 04 07:30:35 CDT 2005
7. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005
8. Question used on ExamName: NRC August 2005 - 2 ExamID: NRC-082005-2 Exam Date: 08/12/2005

Comments:

END OF EXAM

Total Number of Questions: 100 Total Point Value: 100.0

Total Exam Question Difficulty: 163.0

Average Exam Question Difficulty: 1.63

Questions with Difficulty Level 1: 37

Questions with Difficulty Level 2: 63

Go to Action Menu. Go to Home Menu. Exit.