Exam Data:

Exam Name: NRC August 2005 - 1 **ExamID:** NRC-082005-1
Exam Owner: Charles Bell **Exam Date:** 08/12/2005
Created by: Charles Bell **Created at:** Thu Jun 16 13:42:07 CDT 2005
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Approved By: Mickey Ellis **Approval Date:** Mon Jun 20 14:45:58 CDT 2005
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Access to this exam is restricted to:

- Charles Bell
- MikeRasch
- Mickey Ellis
- Tommy Harrelson

Exam History:

- Created by jbell at Thu Jun 16 13:42:07 CDT 2005
- Reviewed by mrasch at Mon Jun 20 14:45:03 CDT 2005
- Approved by mellis at Mon Jun 20 14:45:58 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 plant was operating at 96% power when the "B" Recirculation pump tripped.

Plant conditions include the following:

Reactor power	72% of rated
Core flow	68 Mlbm/hr
Feedwater temperature	415º F.
FCBB	1.34

PBDS is fully operable.

Which one of the following describes the actions to be taken for the present situation?

05-1-02-III-3 Reduction in Recriculation System Flowrate ONEP is provided.

A. Immediately place the reactor mode switch in the SHUTDOWN position.

B. Reduce the Recirculation loop flow to less than 44,600 and monitor for thermal hydraulic instability.

C. Initiate action to reduce core flow to exit the Power/ Flow graph Monitored region then verify FCBB is ≤ 1.0 within 15 minutes.

D. Immediately initiate action to reduce core flow to exit the Power/ Flow graph Restricted region then verify FCBB is ≤ 1.0 within 15 minutes.

Answer: B

Question Comments: The proper action is based on plotting within the Monitored region of the Power/Flow graph. (Figure 1 from 05-1-02-III-3) Answer A is incorrect because this action is for the Exclusion region. Answer B is CORRECT per step 3.5.1 of 05-1-02-III-3. Answer C is incorrect because this action is for the Monitored region with a concurrent loss of Feedwater heating. Answer D is incorrect because this action is for the Restricted region. This is a NEW question. Tier 1 Group 1 CFR 41.5/41.10/43.5

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00833

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-OPS-ONEP Objective: 2.0
2. CourseID: GLP-OPS-ONEP Objective: 25.0

KA References:

1. 295001 AA2.01 Power/flow map [3.5/3.8]
2. 295001 AK1.02 Power/flow distribution [3.3/3.5]
3. GENERIC 2.4.4 Ability to recognize abnormal indications for system operating parameters [4.0/4.3]

4. 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 Tue May 31 14:40:03 CDT 2005
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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
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Comments:

EB QUESTION: 2 (1.0 Points)

The GGNS Electrical line up is normal.

A large tree limb falling across a transmission line causes degraded 500 KV voltage.

The voltage to ALL ESF busses DROPS to 3120 volts for 5 seconds and then returns to normal.

Which one of the following statements describes the condition of the plant ESF busses 30 seconds after this voltage transient?

A.
15AA is being supplied from ESF 11

16AB is being supplied from ESF 21

17AC is being supplied from Div III D/G

B.
15AA is being supplied from Div I D/G

16AB is being supplied from Div II D/G

17AC is being supplied from Div III D/G

C.
15AA is being supplied from ESF 11

16AB is being supplied from ESF 21

17AC is being supplied from ESF 21

D.
15AA is being supplied from Div I D/G

16AB is being supplied from Div II D/G

17AC is being supplied from ESF 21

Answer: A

Question Comments: Answer A is correct because the voltage did not drop low enough for long enough for a Div I/II LSS BUV shed. [90%(3744) for 9 seconds or 70% (2912) for 0.5 seconds] However the Div III bus undervoltage relay trips the incoming feeder breaker at 73% (3045 volts) with no time delay and the Div III diesel generator will start and tie to the bus. Answer B is incorrect because LSS will not shed the DivI/II busses. Answer C is incorrect because the Div III bus undervoltage relay trips the incoming

feeder breaker at 73% (3045 volts) with no time delay and the Div III diesel generator will start and tie to the bus. Answer D is incorrect because the voltage did not drop low enough for long enough for a Div I/II LSS BUV shed. [90%(3744) for 9 seconds or 70%(2912) for 0.5 seconds], therefore the Div I/II diesel generators will not start and tie to their respective busses. However the Div III bus undervoltage relay trips the incoming feeder breaker at 73% (3045 volts) with no time delay and the Div III diesel generator will start and tie to the bus. TIER 1 GROUP 1 This is a MODIFIED question. NRC Exam 3/1998 ID# WRI 11 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

Objectives:

1. CourseID: GLP-OPS-R2100 Objective: 12.0
2. CourseID: GLP-OPS-R2100 Objective: 20.0

KA References:

1. 295003 AK3.01 Manual and auto bus transfer [3.3/3.5]
2. 295003 AK3.03 Load shedding [3.5/3.6]
3. 295003 AK1.03 Under voltage/degraded voltage effects on electricalloads [2.9/3.2]
4. 295003 AK1.04 Electrical bus divisional separation [3.1/3.2]

References:

1. Tech Spec TR3.3.8.1-1
2. 04-1-01-R21-1 section 5.1.1
3. 04-1-01-P81-1 section 3.22

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 Mon Jun 20 07:34:08 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
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13. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 3 **(1.0 Points)**

Which one of the following describes the interrelationship between the Control Power and the Standby Service Water System?

A. The 125 VDC System provides electrical power for SSW control and indication circuits.

The 120 VAC System provides electrical power for SSW initiation logic.

Loss of 125 VDC would NOT affect the initiation logic however, the SSW pumps and cooling tower fans CANNOT automatically start without DC control power.

B. The 125 VDC System provides electrical power for SSW initiation, control, and indication circuits.

Loss of 125 VDC would cause the initiation logic to initiate.

The SSW pumps and cooling tower fans CANNOT automatically start without DC

control power.

C. The 125 VDC System provides electrical power for SSW initiation, control, and indication circuits.

Loss of 125 VDC would cause the initiation logic to initiate.

The SSW pumps and cooling tower fans would automatically start and the system would align to supply cooling water to its plant loads.

D. The 120 VAC System provides electrical power for SSW control and indication circuits.

The 125 VDC System provides electrical power for SSW initiation logic.

Loss of 125 VDC would cause the initiation logic to initiate however, the SSW pumps and cooling tower fans CANNOT automatically start without AC control power.

Answer: B

Question Comments: Answer A is incorrect because the 120 VAC system does not supply electrical power to the SSW initiation logic. Answer B is correct because the 125 VDC System provides electrical power for SSW initiation, control, and indication circuits. Answer C is incorrect because the SSW pumps and cooling tower fans would not automatically start without DC control power. Answer D is incorrect because the 120 VAC System does not provide electrical power for SSW control and indication circuits. 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: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-OPS-P4100 Objective: 12.6
2. CourseID: GLP-OPS-I1100 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-1225

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
2. P41: Standby Service Water System

Categories:

1. Systems

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 07:29:14 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

Comments:

EB QUESTION: 4 (1.0 Points)

Based on EP-4 conditions, the "Operator at the Controls" has been directed to rapidly depressurize the reactor using the main turbine bypass valves .

The Condensate and Feedwater systems are in automatic with a 36 inch level setpoint and will respond to the event per design.

How should this be accomplished and how will these actions effect reactor water level?

A. The operator should immediately fully open the bypass valves and expect indicated narrow range reactor water level to drop and return to normal only after the ECCS initiates to augment the Condensate and Feedwater systems.

B. The operator should gradually open the bypass valves to full open and expect indicated narrow range reactor water level to rise, then drop, then return to normal only after the ECCS initiates to augment the Condensate and Feedwater systems.

C. The operator should immediately fully open the bypass valves and expect indicated narrow range reactor water level to rise, then drop, then return to normal as the Condensate and Feedwater systems respond.

D. The operator should gradually open the bypass valves to full open and expect indicated narrow range reactor water level to remain constant as the Condensate and Feedwater systems respond.

Answer: C

Question Comments: Answer A is incorrect because, reactor water level will initially rise, then drop. With no power being generated from the reactor, no ECCS initiation is expected. Answer B is incorrect because the operator should immediately fully open the bypass valves. With no power being generated from the reactor, no ECCS initiation is expected. Answer C is correct because the operator should immediately fully open the bypass valves and expect indicated reactor water to rise, then drop, then return to normal as the Condensate and Feedwater systems respond. Answer D is incorrect because the operator should immediately fully open the bypass valves and expect indicated reactor water level to rise, then drop, then return to normal as the Condensate and Feedwater systems respond. This is a NEW question. TIER 1 GROUP 1 CFR 41.3/41.5/41.7/41.10

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

TrainingPrograms:

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

Systems:

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

Categories:

1. Emergency Procedure Training
2. Fundamentals
3. Systems

Task References:

Question Last Revised By: MikeRasch at Tue May 10 13:58:26 CDT 2005
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1. Created by tharrelso at Thu Apr 21 12:17:26 CDT 2005
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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

Comments:

EB QUESTION: 5 (1.0 Points)

In which one of the following "Scram Reports" can the SRO determine that Shutdown Margin 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, bypass valves are available, and Feedwater is controlling reactor water level.

B. The Mode Switch is in Shutdown, APRMs indicate downscale, NO control rod position indication is available, bypass valves are available, and Feedwater is controlling reactor water level.

C. The Mode Switch is in Shutdown, reactor power is dropping, all control rods indicate full in except forty nine control rods indicate position 02, bypass valves are unavailable, and Feedwater is recoverable.

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, bypass valves are unavailable, and Feedwater is recoverable.

Answer: C

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 incorrect because even though the APRMs indicate downscale, no control rod position indication is available. Answer C is CORRECT because all control rods are inserted to at least position 02 and by analysis, this ensures adequate shutdown margin. 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
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 Tue May 24 10:14:09 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
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6. 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 one of the following describes the minimum actions necessary to scram the reactor from outside the Control Room?

A. Secure both CRD pumps using handswitches on the Remote Shutdown panels.

B. Open breakers CB2A and CB8A at panel 1C71-P001 ("A" RPS MG Set room).

C. Open breaker CB2B and CB8B at panel 1C71-P002 ("B" RPS MG Set room).

D. Open breakers CB2A or CB8A at panel 1C71-P001 ("A" RPS MG Set room) and CB2B or CB8B at panel 1C71-P002 ("B" RPS MG Set room).

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: 1: Fundamental Knowledge or Memory

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: MikeRasch at Tue May 24 10:30:20 CDT 2005
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6. 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 was operating at 100% power when the following alarms were observed.

Annunciators in alarm:

RECIRC PMP/MTR A/B TEMP HI

CTMT AREA FLR DRN SUMP LEAK HI

CCW SURGE TK LVL HI/LO

RWCU FLTR DMIN INL TEMP HI 130º F

CCW PMP A/C DISCH PRESS LO

Which one of the following is the appropriate response to this condition?

05-1-02-V-1, Loss of Component Cooling Water ONEP is provided.

A. Monitor Reactor Recirculation pump temperatures on 1B33-R601.

B. Verify the standby CCW pump starts. Isolate CCW to FPCCU. Isolate CCW to RWCU.

C. Reduce core flow to 67 Mlbm/hr. Refer to ONEP 05-1-02-III-3, Reduction in Reactor Recirculation Flow.

D. Manually scram the reactor. Manually trip the Reactor Recirculation pumps. Isolate CCW to the Containment.

Answer: D

Question Comments: Answer D is CORRECT per Step 3.3 of ONEP 05-1-02-V-1, Loss of Component Cooling Water. Answer A is incorrect because even though the Reactor Recirculation pumps are still running, this action is not enough because of the line break in Containment. Answer B is incorrect because even though this is a partial loss of CCW, this action is not enough because of the line break in Containment. Answer C is incorrect because even though the Reactor Recirculation pump temperatures are rising, this action is not enough because of the line break in Containment.

This is a NEW question. TIER 1 GROUP 1 CFR 41.4/41.5/41.6/41.7/41.10/43.5

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00838

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-OPS-ONEP Objective: 2.0
2. CourseID: GLP-OPS-B3300 Objective: 36.7
3. CourseID: GLP-OPS-B3300 Objective: 36.9
4. CourseID: GLP-OPS-B3300 Objective: 42.0
5. CourseID: GLP-OPS-B3300 Objective: 40.0
6. CourseID: GLP-OPS-G3336 Objective: 9.7
7. CourseID: GLP-OPS-G3336 Objective: 9.9
8. CourseID: GLP-OPS-G3336 Objective: 12.0
9. CourseID: GLP-OPS-G3336 Objective: 14.0
10. CourseID: GLP-OPS-P4200 Objective: 11.1
11. CourseID: GLP-OPS-P4200 Objective: 11.2
12. CourseID: GLP-OPS-P4200 Objective: 14.0
13. CourseID: GLP-OPS-P4200 Objective: 16.0

KA References:

1. 295018 AK1.01 Effects on component/system operations [3.5/3.6]
2. 295018 AK2.02 Plant operations [3.4/3.6]
3. 295018 AA2.03 Cause for partial or complete loss [3.2/3.5]

References:

1. 05-1-02-V-1 Step 3.3
2. 04-1-02-1H13-P870-5A-C2
3. 04-1-02-1H13-P870-5A-D2
4. 04-1-02-1H13-P601-22A-C2
5. 04-1-02-1H13-P680-3A-A8
6. 04-1-02-1H13-P680-11A-D6
7. 04-1-02-1H13-P870-9A-D2

TrainingPrograms:

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

Systems:

1. B33: Reactor Recirculation System
2. E31: Leak Detection System
3. G33: Reactor Water Cleanup
4. P42: Component Cooling Water System

Categories:

1. Off Normal Event Procedures
2. Systems

Task References:

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Comments:

EB QUESTION: 8 (1.0 Points)

The plant was operating at rated conditions when a leak in the Auxiliary Building Instrument Air header caused the header pressure to drop to 28 psig.

A manual scram was inserted.

NO LOCA isolations have occurred.

The Primary and Secondary Containment air operated isolation valves closed due to low pressure, and a bleed off valve in the Auxiliary Building automatically opened depressurizing the Auxiliary Building air header.

The Auxiliary Building Instrument Air header air leak was repaired.

Which one of the following describes the actions that will occur when Auxiliary Building Instrument Air header pressure is restored without operator action?

Upon restoration of header pressure, :

A. all of the Primary and Secondary Containment air operated isolation valves will

automatically re-open and the bleed off valve will have to be manually reclosed.

B. all of the Primary and Secondary Containment air operated isolation valves will require manual re-opening and the bleed off valve will have to be manually reclosed.

C. some of the Primary and Secondary Containment air operated isolation valves will require manual re-opening and the bleed off valve will automatically reclose.

D. some of the Primary and Secondary Containment air operated isolation valves will automatically re-open and the bleed off valve automatically re-close.

Answer: C

Question Comments: Upon Instrument air system pressure dropping < 60 psig Hiller actuators have a solenoid valve that isolate the valve. This will require operator actions to give the valve an open signal to return the valve to the open position. P53-F026A, F026B, and F001 are all hiller acutators. Once Instrument Air is restored to Secondary Containment the Instrument Air Bleed-off station P53-F531/F523 on 166' elevation will automatically reset itself by the Restricting Orifice (P53-RO-D026) creating a backpressure to close P53-F531. Answer A is incorrect because it has the isolation valves automatically re-opening and having to manually close the bleed-off station. Answer B is incorrect because it has the bleed-off station being manually reclosed. Answer C is CORRECT Primary and Secondary Containment isolation valves have to be manually realigned and the bleed-off station automatically reclosing. Answer D is incorrect because Primary and Secondary Containment valves do NOT automatically re-open. TIER 1 GROUP 1 This is a NEW question. CFR 41.4/41.5/41.7/41.9

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00548

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-OPS-P5300 Objective: 13.0
2. CourseID: GLP-OPS-P5300 Objective: 14.0

KA References:

1. 295019 AA2.01 Instrument air system pressure [3.5/3.6]
2. 295019 AA2.02 Status of safety-related instrument air system loads(see AK2 [3.6/3.7]
3. 295019 AK2.14 Plant air systems [3.2/3.2]
4. 295019 AK2.09 Containment [3.3/3.3]

References:

1. M-1067M
2. 05-1-02-V-9 Note after step 3.6
3. GG FSAR Table 9.3-1
4. GG FSAR Table 9.3-2
5. M-1067E

TrainingPrograms:

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

Systems:

1. P53: Instrument Air System

Categories:

1. Systems

Task References:

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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

Comments:

EB QUESTION: 9 (1.0 Points)

The plant was in Mode 4 during a forced outage when all forced circulation was 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: MikeRasch at Thu Jun 09 16:39:33 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

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. High in-plant dose rates

D. Inadequate decay heat removal

Answer: C

Question Comments: Answer C 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 D 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

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 Mon Jun 20 07:47:42 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

Comments:

EB QUESTION: 11 (1.0 Points)

The plant is at 100% power.

I&C is trouble shooting failure of Remote Shutdown reactor water level transmitter C61-N400B.

While isolating C61-N400B, a pressure spike on the reference leg of condensing pot B21-D004B affecting Wide Range Reactor Water Level transmitters B21-N091B and B21-N091F causes a spurious Low Reactor Water Level, Level 1, ECCS initiation.

NO other level or pressure transmitters common to that reference leg are affected.

The low water level initiation signal was only a spike and is now clear.

What is the sequence of operator actions necessary to prevent a reactor scram as a result of the ECCS initiation?

Selected sections of 17-S-06-5 and M-1077B are provided.

A. Restore Instrument Air and Plant Service Water to the Auxiliary Building and Containment; then, in order to shut down Drywell Purge Compressor A, depress LPCS/RHR A INIT RESET pushbutton on H13-P601, depress DIV 1 LSS PNL RESET on H13-P864, place CGCS DIV 1 MAN INIT RESET to RESET, then stop Drywell Purge Compressor A, E61-C001A, using its hand switch on H13-P870.

B. Restore Instrument Air and Plant Service Water to the Auxiliary Building and Containment; then, in order to shut down Drywell Purge Compressor B, depress RHR B/C INIT RESET pushbutton on H13-P601, depress DIV 2 LSS PNL RESET on H13-P864, place CGCS DIV 2 MAN INIT RESET to RESET, then stop Drywell Purge Compressor B, E61-C001B, using its hand switch on H13-P870.

C.

Depress LPCS/RHR A INIT RESET pushbutton on H13-P601, depress DIV 1 LSS PNL RESET on H13-P864, place CGCS DIV 1 MAN INIT RESET to RESET, then stop Drywell Purge Compressor A, E61-C001A, using its hand switch on H13-P870.

D. Depress RHR B/C INIT RESET pushbutton on H13-P601, depress DIV 2 LSS PNL RESET on H13-P864, place CGCS DIV 2 MAN INIT RESET to RESET, then stop Drywell Purge Compressor B, E61-C001B, using its hand switch on H13-P870.

Answer: D

Question Comments: Answer A is incorrect because the isolation signal for the listed valves originates from B21-N082A-D, not N091's, and B21-N091B&F initiate Division 2 ECCS, not Division 1 as listed. Answer B is incorrect because the isolation signal for the listed valves originates from B21-N082A-D, not N091's. Answer C is incorrect because B21-N091B&F initiate Division 2 ECCS, not Division 1 as listed. Answer D is correct since ECCS logic is designed such that a Division 2 ECCS initiation will occur if reactor water level signals from B21-N091B&F go below -150.3". 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

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 Mon Jun 20 13:33:34 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

Comments:

EB QUESTION: 12 (1.0 Points)

A Group 1 isolation resulted in a reactor scram.

RCIC was started manually and is injecting into the reactor with the controller in manual.

Reactor pressure is rising.

Which one of the following describes the RCIC system response to rising reactor pressure?

A. RCIC speed will drop with injection rate remaining stable.

B. RCIC speed will rise causing the injection rate to rise.

C. RCIC speed will rise slightly with injection rate remaining stable.

D. RCIC speed will rise slightly but injection rate will drop slightly.

Answer: D

Question Comments: As Reactor pressure rises speed will rise (about 50 rpm) because the governor valve is in a fixed position. This rise in speed is not sufficient to overcome approximately 100 psi change in Reactor pressure resulting in a slightly lower rate of injection. When an SRV opens to relieve pressure injection flow turns and rises. Answer A is incorrect because RCIC speed must increase, not drop due to a higher pressure supplied to the RCIC turbine. Answer B is incorrect because RCIC speed will rise as reactor pressure increases due to the governor valve being in a fixed position. This rise in speed will not be sufficient to overcome the rise in reactor pressure causing injection rate to drop. Answer C is incorrect because pump output will rise as speed of the turbine rises. Answer D is correct because a slight increase in RCIC speed will result from the higher reactor pressure however this rise in speed is not sufficient to overcome the change in reactor pressure causing injection rate to drop slightly. TIER 1 GROUP 1 This is a MODIFIED question. NRC ID WRI818 February 2004 CFR 41.4/41.5/41.14

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-E5100 Objective: 4.11
2. CourseID: GLP-OPS-E5100 Objective: 8.17

KA References:

1. 295025 EK3.05 RCIC operation: Plant-Specific [3.6/3.7]
2. 295025 EK2.07 RCIC: Plant-Specific [3.7/3.7]

References:

1. Vendor Manual 460000182
2. Simulator Response

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. Systems

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 12:49:23 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

Comments:

EB QUESTION: 13 (1.0 Points)

The plant is in an ATWS following a total loss of EHC due to large break in EHC piping.

Reactor power is 7%.

Reactor pressure is being controlled 800 psig to 1060 psig using SRVs and Main Steam Line Drains.

Reactor level is being controlled -167" to -192" using Reactor Feed Pump 'A'.

Reactor Feed Pump 'B' is available.

Suppression Pool Cooling 'B' is in service.

Suppression Pool Cooling 'A' is unavailable.

Suppression Pool level is 19.0 feet.

Suppression Pool temperature is 155°F and slowly rising.

Emergency Procedure Attachments to enable control rod insertion are NOT expected to be installed for at least 30 minutes.

Standby Liquid Control systems have failed.

Which one of the following is the basis for the requirement to lower Reactor Pressure using SRVs for this condition?

A. To raise Suppression Pool level which will raise its capacity as a heat sink by driving conditions away from HCTL.

B. To get more voiding which will lower power and slow rate of the Suppression Pool temperature rise.

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

D. Lower reactor pressure to within Condensate Booster Pump discharge pressure capability in case both Reactor Feed Pumps are lost.

Answer: C

Question Comments:

Image Reference: None

Open Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00840

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

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

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: Charles Bell at Fri Jun 17 14:31:55 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

Comments:

EB QUESTION: 14 (1.0 Points)

The plant is in an ATWS. An RWCU piping leak has occurred on the inlet to the non-regenerative heat exchangers. RWCU isolation valves have lost power and have failed to isolate.

Reactor power is 3%.

Reactor pressure is being controlled 450 psig to 600 psig using Main Bypass Valves.

Reactor level is being controlled -150 inches to -192 inches using Condensate Booster Pumps.

Primary Containment pressure is 2.0 psig.

Primary Containment temperature is 187°F.

Suppression Pool temperature is 125°F.

Suppression Pool level is 18.8 feet.

Both loops of Suppression Pool Cooling are maximized.

Standby Liquid Control systems have failed.

Which one of the following describes actions to control Primary Containment temperature and pressure?

A. Conduct an Emergency Depressurization of the RPV.

B. Vent Primary Containment using Emergency Procedure Attachment 14.

C. Operate all containment coolers using Emergency Procedure Attachment 7 to bypass all containment cooler isolation interlocks.

D. Initiate both loops of Containment Spray. Then, if Primary Containment temperature does NOT lower below 185°F, conduct an Emergency Depressurization of the RPV.

Answer: A

Question Comments: Answer A 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, and Containment Spray is not allowed due to being in the unsafe region of the CSIPL curve (EP-3 step 6). Answer B is incorrect since the vent path for EP attachment 14 is isolated and cannot be opened under the listed conditions. Answer C is incorrect since operation of containment coolers without P71 as a heat sink would be ineffective. Answer D is incorrect since Containment Spray is not allowed due to being in the unsafe region of the CSIPL curve (EP-3 step 6). 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-00844

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

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

TrainingPrograms:

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

Systems:

Categories:

1. Emergency Procedure Training

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 13:35:23 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

Comments:

EB QUESTION: 15 (1.0 Points)

A LOCA has occurred in the Drywell.

A reactor scram has occurred, and all control rods fully inserted.

All low pressure ECCS systems are available. HPCS is out of service.

RCIC, SLC, and CRD are injecting at rated flow.

Reactor pressure is 500 psig and slowly falling.

Wide Range reactor water level indication is -150 inches and slowly falling.

Post Accident Monitoring Fuel Zone level indication is -180 inches and slowly falling.

Compensated Fuel Zone level indication is -160 inches and slowly falling.

Average Drywell temperature is 190°F.

Drywell temperature at elevation 166 ft. is 200°F.

Average Containment temperature is 135°F.

Containment temperature at elevation 139 ft. is 150°F.

Which one of the following describes Reactor Water Level indication?

A. Emergency Depressurization should be conducted when Wide Range level trends offscale low.

B. Emergency Depressurization is required now by the Alternate Level Control leg

under these conditions.

C. Emergency Depressurization should be conducted when Compensated Fuel Zone level reaches -192 inches.

D. RPV Flooding should be entered now under these conditions.

Answer: B

Question Comments: Answer A is incorrect since EP Caution 1 prohibits use of wide range level indication when containment temperature at elev. 139' is >143°F. Answer B is correct since wide range level indication cannot be used and fuel zone level is < TAF (-167"). Compensated fuel zone is to be used for values of reactor level only during ATWS conditions per 02-S-01-27. Answer C is incorrect since compensated fuel zone is to be used for values of reactor level only during ATWS conditions per 02-S-01-27. Answer D is incorrect since reactor level can be determined by using fuel zone indication. 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

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 steps 53 - 71
2. 05-S-01-EP-2 Caution 1 section 2

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: MikeRasch at Thu Jun 16 12:51:16 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

Comments:

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

The plant is at 100% power.

High Pressure Core Spray system was operating on minimum flow with suction from the Suppression Pool for testing when a large break on the High Pressure Core Spray suction line upstream of HPCS Suction from Suppression Pool valve E22-F015 occurred.

HPCS pump was secured, but E22-F015 lost power when operators attempted to close it.

Annunciators HPCS RM SUMP LVL HI-HI (1H13-P680-8A1-B4) and HPCS ROOM FLOODED (1H13-P870-5A-H1) are in alarm.

The HPCS room water tight door has failed.

Suppression pool level is 15 feet and slowly falling.

Suppression Pool temperature is 80°F.

Emergency Procedures 3 and 4 have been entered.

Which one of the following describes further actions in response to the Suppression Pool leak?

A. Low Suppression Pool level would have caused HPCS suction to automatically align to the CST if E22-F015 had NOT lost power.

B. If Suppression Pool level continues to fall to 14.56 feet, Emergency Depressurization will be required due to inability to maintain the Heat Capacity Temperature Limit (HCTL) curve in the Safe Zone with the Reactor pressurized.

C. If one more Maximum Safe water level is reached in Secondary containment due to the Suppression Pool leak, Emergency Depressurization will be required by EP-4.

D. If Suppression Pool level continues to fall to 14.56 feet, Emergency Depressurization will be required to avert inadequate submergence of the horizontal vents in the Drywell wall with the Reactor pressurized.

Answer: D

Question Comments: Answer A is incorrect since HPCS suction has no automatic feature associated with low suppression pool level. Answer B is incorrect since suppression pool temperature would have to reach >140°F for the HCTL curve to require emergency depressurization with the reactor at rated pressure. Answer C is incorrect since the high area water level is not due to a leak from the reactor pressure boundary. Answer D is correct due to EP-3 step 42 requirements and basis. Tier 1 Group 1 This is a NEW question. 10 CFR 41.9/41.10/43.5

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00846

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

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

KA References:

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

References:

1. PSTG B 7-30
2. 05-S-01-EP-3

TrainingPrograms:

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

Systems:

Categories:

1. Emergency Procedure Training

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 13:37:44 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

Comments:

EB QUESTION: 17 **(1.0 Points)**

The plant was operating at 100% power when a small leak developed in the Drywell due to failure of both reactor head O-rings.

Drywell pressure is 1.5 psig and rising slowly.

All systems responded as designed.

The lowest reactor water level reached was -20 inches Wide Range.

Operators stopped High Pressure Core Spray (HPCS) pump using its Control Room hand switch to attempt to maintain reactor water level below Level 8.

Reactor water level is now 60" on Wide Range instruments due to swell.

Reactor pressure is stable at 960 psig being controlled automatically by bypass valves.

Which one of the following describes further operation of HPCS?

A. If HPCS INIT RESET push button is depressed on 1H13-P601, HPCS pump will immediately restart, but HPCS Injection Valve E22-F004 will remain closed.

B. HPCS Injection Valve E22-F004 can be opened by depressing and holding HPCS HI LVL RESET push button on 1H13-P601 and simultaneously holding the hand switch for E22-F004 to OPEN.

C. HPCS pump will automatically start and HPCS Injection Valve E22-F004 will automatically open with NO further operator action if reactor water level drops to Level 2.

D. HPCS pump will automatically start and HPCS Injection Valve E22-F004 will automatically open if reactor water level drops to Level 2 only if HPCS INIT RESET push button has been depressed on 1H13-P601.

Answer: D

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” 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”. Answer C is incorrect because once manually overridden with an initiation signal

sealed in, HPCS pump would have to be started manually with its hand switch or the initiation reset to break the manual override seal-in. Answer D is correct because depressing HPCS INIT RESET would reset the initiation signal and override high drywell pressure and break the manual override seal-in. 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

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 Mon Jun 20 13:38:53 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

Comments:

EB QUESTION: 18 (1.0 Points)

The plant had been operating at 100% power when a small tear occurred on the High Pressure Condenser boot seal.

A manual scram was directed due to degrading condenser vacuum, however, all control rods failed to fully insert.

All other equipment operated as designed.

EP-2A was entered.

Plant conditions are as follows:

Main Condenser vacuum is 11.5 in. Hg and falling very slowly.

Reactor power is 15 %.

Reactor water level is being maintained -70 inches to -130 inches using Condensate Booster Pumps.

Reactor pressure is being controlled 450 psig to 600 psig using SRVs.

MSIVs are open.

Main Bypass Stop Valves are closed.

Which one of the following describes required actions to be taken for RPV pressure control under these conditions?

A.

Attachment 8 should be installed.

B. Lowering Pressure Reference to 900 psig will cause reactor pressure to fall to less than the lowest Low-Low Set Valve reset setpoint.

C. The Main Condenser is NO longer "available" per EP-2A.

D. Main steam line drains should NOT be used since they discharge to the High Pressure Condenser shell.

Answer: A

Question Comments: Answer A is correct since power is >4%, the MSIVs are open (hence the main condenser is available), steam loads such as RFPTs are in service, and preserving the main condenser as a heat sink given the challenge to MSIVs from lowering reactor water level is desirable. This minimizes the heat input to containment and prolongs availability of feedwater. Therefore, EP-2A step 40 directs installation of Att. 8 to defeat the low water level closure of MSIVs. Answer B is incorrect because lowering P-ref will have no effect on pressure since turbine and bypass stop valves are all closed below 12”Hg vacuum. Answer C is incorrect because PSTGs conclude availability to reject heat to the condenser exists when MSIVs are open. Answer D is incorrect because discharging steam to the main condenser limits the challenge to containment. The MSIV isolation at 9”Hg vacuum would provide ample protection for the condenser to prevent over-pressurization and a possible radioactive release. Tier 1 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-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: MikeRasch at Mon Jun 20 13:40:07 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

Comments:

EB QUESTION: 19 (1.0 Points)

The plant was in Mode 4.

Standby Service Water (SSW) ‘A’ was in service supplying Fuel Pool Cooling and Clean-up (FPCC) heat exchanger ‘A’.

SSW ‘A’ basin cooling tower fans were temporarily secured due to low SSW ‘A’

temperature.

A small tube leak in the FPCC ‘A’ Heat Exchanger developed.

Drift from SSW ‘A’ basin was blowing westward and forming a puddle inside of the protected area fence.

Eventually, runoff from the puddle flowed under/through the protected area fence and to the storm drain in front of the Unit 2 Warehouse.

When would this be first considered to be a liquid effluent release regard to the Technical Requirements Manual?

TRM 6.11.1 and Definitions are provided.

A. When the effluent began to puddle inside the protected area fence.

B. When the effluent crossed the protected area fence.

C. When the effluent entered the storm drain in front of the warehouse.

D. When the effluent entered a navigable 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

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00851

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

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

KA References:

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

References:

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: MikeRasch at Mon Jun 20 13:42:54 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

Comments:

EB QUESTION: 20 (1.0 Points)

Which one of the following operator actions is NOT required in relation to a fire inside the protected area?

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

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

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

D. If the Main Control Room is evacuated due to a fire in the Main Control Room, always place Transfer Switch for Lockout Transfer Relay C61-HSS-M150 at 1H22-P152 to ON.

Answer: C

Question Comments: Answer A is incorrect because it is required by 04-1-01-P81-1 step 3.14. 04-1-01-P81-1 step 3.14 Answer B is incorrect because it is required by 05-1-02-II-1 step 3.4.1. Answer C is correct because the control room is not an area listed in 10-S-03-2 step 6.2.2f requiring Z17-B002 secured, since that does not directly serve the control room. Answer B is incorrect because it is required by 05-1-02-II-1 step 3.5.1 for a fire in the control room severe enough to result in control room evacuation. 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: MikeRasch at Fri Jun 10 08:33:06 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

Comments:

EB QUESTION: 21 (1.0 Points)

The plant is operating at 20% power with Main Generator output 250 MWe.

Circulating Pump ‘A’just tripped due to an electrical fault in the motor windings.

Circulating Water pump ’B’is in standby.

Main condenser vacuum is now 25 in. Hg vacuum.

Which one of the following sequence of events will occur as a result of degrading Main

Condenser vacuum if vacuum continually lowers at a rate of 1 in. Hg vacuum every 5 minutes?

A. Reactor scram, Main Turbine trip, Reactor Feed Pump trip, Main Bypass Stop Valve closure, MSIV closure

B. Main Turbine trip, Reactor Feed Pump trip, Reactor scram, Main Bypass Stop Valve closure, MSIV closure

C. Main Turbine trip, Reactor Feed Pump trip, Main Bypass Stop Valve closure, Reactor scram, MSIV closure

D. Main Turbine trip, Reactor Feed Pump trip, Main Bypass Stop Valve closure, MSIV closure, Reactor scram

Answer: B

Question Comments: Answer A is incorrect because there is no automatic scram directly from condenser vacuum. Answer B is correct because the turbine will trip at 21”Hg vacuum. The reactor scram from TSV/TCV closure is bypassed below 40% power, so no scram will occur from EHC fluid pressures. And bypass valves can accommodate steam flow for 35% power and are fast acting, so the effect on reactor pressure will be minimal, and pressure will stay below 1064.7 psig, the scram setpoint. Stable pressure will minimize shrink, so reactor level will remain relatively stable above the 11.4” scram setpoint. Also, neutron flux will not spike above the scram setpoint of 118% since reactor pressure will remain stable. RFPTs will trip at 16”Hg vacuum, causing inventory to steam off with only CRD for makeup. CRD will supply <1% rated makeup flow. With power 20%, steam flow is about 6600gpm, so the low level scram setpoint of 11.4” will be reached within less than 1 minute, calculating a level reduction rate of about 30 in/min, assuming 200 gal/in reactor level. This is much sooner than when bypass valves will close given the vacuum loss rate of 0.2”Hg vac/min. Then, Bypass Stop Valves will close at 12”Hg vacuum, and MSIVs will close at 9”Hg vacuum. Answer C is incorrect because the reactor will scram on low water level well before the bypass valve closure setpoint of 12”Hg vacuum is reached. 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 Mon Jun 20 13:43:36 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

Comments:

EB QUESTION: 22 (1.0 Points)

Under which one of the following conditions is installation of bottled gas to supply the Automatic Depressurization System (ADS) air receivers required?

A. Mode 3, following a scram due to loss of bus 15AA. SRVs are being used for reactor pressure control. ADS accumulators ‘A’ and ‘B’ pressures indicated on 1H13-P601 are 150 psig. The time estimated to restore bus 15AA is 8 hours.

B. An ATWS is in progress. The Main Condenser is available. Reactor power is 8%. The Auxiliary building is isolated due to low reactor water level and has not been restored.

C. An instrument air header rupture in the water treatment building has occurred. Repairs are expected to take 20 minutes. SRVs are being used for reactor pressure control. ADS accumulators ‘A’ and ‘B’ pressures are 165 psig and 163 psig, respectively.

D. Mode 4 during Operations Hydro (03-1-01-6) when one ADS Air Receiver is inoperable due to system maintenance.

Answer: A

Question Comments: 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. Answer B is incorrect because the instrument air isolation can be bypassed and restored under given conditions, so installing bottled gas is unnecessary. Answer C is incorrect because the stated expected out of service time for the normal air supply to SRV is less than the 6 hour limit in the referenced plant procedures. Answer D is incorrect because ADS is not required operable during Ops Hydro, only the relief function of 2 SRVs is required functional, which it would be with one ADS Air Receiver remaining operable. 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 Fri Jun 10 09:04:56 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

Comments:

EB QUESTION: 23 **(1.0 Points)**

The plant was at 99% power.

RHR 'A' pump is tagged out of service to replace the pump breaker protective relays.

A disturbance in the switchyard caused a loss of Service Transformers 11 and 21.

Three control rods remained at position 48 following the reactor scram.

Reactor power is 0% on APRMs.

MSIVs are closed.

Reactor level is being controlled +30 inches to -30 inches using RCIC.

Reactor pressure is being controlled 800 psig to 1060 psig using SRVs.

All ECCS has been initiated and overridden.

Average Suppression temperature is 95°F.

Instrument air and service air are unavailable.

Air header pressure in the Auxiliary Building is 45 psig.

Both ADS Air Receiver pressures are 160 psig.

RHR B TEST RETURN TO SUPP POOL VLV E12-F024B failed to open when placing Suppression Pool Cooling 'B' in service.

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

A. It is acceptable to allow Low-Low Set to cycle indefinitely under these conditions, as long as localized Suppression Pool temperatures remain below 185°F.

B. If there is only one leaking/weeping SRVs, 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 to allow for even heating of the Suppression Pool.

D. Only ADS SRVs should be used, and they should be rotated to allow for even heating of the Suppression Pool.

Answer: D

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 incorrect because ADS valves are specifically preferred when instrument air is unavailable since they will function longer. Answer D is correct because ADS valves are specifically preferred when instrument air is unavailable. Rotation to equalize pool heating and reduce the chance of exceeding containment design temperature limits is specified. 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: 1: Fundamental Knowledge or Memory

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 Mon Jun 20 13:45:28 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

Comments:

EB QUESTION: 24 (1.0 Points)

The plant was in power ascension during Mode 2.

RCIC was being warmed with the RCIC Low Reactor Pressure (60 psig) isolation defeated per IOI-1.

RCIC STM SPLY DRWL OTBD ISOL VLV E51-F064 and RCIC STM SPLY DRWL INBD ISOL VLV E51-F063 had been opened.

Reactor pressure had reached 62 psig.

A large leak developed from a crack in a weld on RCIC steam supply piping upstream of RCIC STM SPLY TO RCIC TURBINE valve E51-F045.

An operator has attempted to close E51-F063 and E51-F064, but their supply breakers have tripped before the valves reached full closed.

Reactor pressure has fallen to 58 psig.

Which one of the following actions is required for these conditions?

A. Immediately restore the RCIC Low Reactor Pressure isolation in accordance with IOI-1.

B. If RCIC Equipment Area Temperature reaches 185°F, enter EP-2 and place the Reactor Mode Switch in Shutdown.

C. If RCIC Equipment Area Temperature reaches 212°F, enter EP-2 and place the Reactor Mode Switch in Shutdown.

D. If RCIC Equipment Area Temperature reaches 212°F, enter EP-2 and conduct Emergency Depressurization.

Answer: C

Question Comments: Answer A is incorrect because the RCIC Low Reactor Pressure isolation is not required operable <150 psig reactor pressure, and restoring it would do nothing to mitigate the event. 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 E51F063&64 not fully closed, 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). Answer D is incorrect since only one area is affected, and temperature would have to exceed maximum safe levels in at least 2 areas to require emergency depressurization per EP-4 step 16. Tier 1 Group 2 This is a NEW question. 10 CFR 41.5/41.10/43.5

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00857

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis
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

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: MikeRasch at Mon Jun 20 13:46:09 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

Comments:

EB QUESTION: 25 (1.0 Points)

The plant is in Mode 5.

Fuel Handling Area Pool Sweep Exhaust Radiation Monitor 'C', D17K618C, has been removed by I&C for replacement.

Annunciator FP EXH DIV 2,3 RAD HI-HI/INOP (1H13-P601-19A-C10) is sealed in.

A fuel bundle was dropped inside the Reactor causing damage to fuel pins.

As a result of the event, the following radiation levels were reached, where they have now stabilized:

Containment/Drywell Vent Exhaust Radiation Monitors

D17K609A = 3.8 mr/hr

D17K609B = 4.0 mr/hr

D17K609C = 3.0 mr/hr

D17K609D = 4.0 mr/hr

Fuel Handling Area Vent Exhaust Radiation Monitors

D17K617A = 2.0 mr/hr

D17K617B = 4.0 mr/hr

D17K617C = 3.0 mr/hr

D17K617D = 4.0 mr/hr

Fuel Handling Area Pool Sweep Exhaust Radiation Monitors

D17K618A = 32 mr/hr

D17K618B = 20 mr/hr

D17K618C = removed for replacement/inoperable

D17K618D = 22 mr/hr

Which one of the following describes the response of Standby Gas Treatment System (SGTS) to the event?

A. 'A' and 'B' will remain in standby.

B.

'A' will automatically start.

C. 'B' will automatically start.

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: MikeRasch at Mon Jun 20 08:07:53 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

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.

Which one of the following is the implication of this alarm?

A. Greater personnel 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

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: MikeRasch at Fri Jun 10 09:40:21 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

Comments:

EB QUESTION: 27 (1.0 Points)

Plant is in Mode 3.

RHR 'A' has just been placed in Shutdown Cooling mode.

A large (2000 gpm) leak from the Standby Service Water 'A' supply piping to the RHR 'A' heat exchangers occurs in the RHR 'A' heat exchanger room.

Motor Control Center 11B12 is de-energized for connecting temporary power.

RHR Room 'A' Floor Drain Sump Pump 'A', P45C013A, is powered from breaker 52-111209.

RHR Room 'A' Floor Drain Sump Pump 'B', P45C013B, is powered from breaker 52-115104.

The current hand switch configuration for RHR 'A' Room Floor Drain Sump Pumps is:

P45C013A RHR Rm 'A' Flr Drn Smp Pmp 'A' HS-M020A AUTO

P45C013B RHR Rm 'A' Flr Drn Smp Pmp 'B' HS-M021A AUTO

P45C013A/B RHR Rm 'A' Flr Drn Smp Pmps 'A/B' Mode Switch HSS-M019A ALTERNATE

RHR Room A Floor Drain Sump Pump "A" was the last pump to run in this hand switch alignment.

Which one of the following describes operation as a result of these conditions?

A. RHR Rm 'A' Flr Drn Smp Pmp 'A', C013A, would start on High sump level. RHR Rm 'A' Flr Drn Smp Pmp 'B', C013B, would start on Hi-Hi sump level. Both pumps would continue to run. Alarms RHR RM A SMP LVL HI-HI (1H13-P680-8A1-A2) and RHR A RM FLOODED (1H13-P870-2A-E1) would be received in the control room.

B. RHR Rm 'A' Flr Drn Smp Pmp 'B', C013B, would start on Hi-Hi sump level and would continue to run. Only alarm RHR RM A SMP LVL HI-HI (1H13-P680-8A1-A2) would be received in the control room.

C. RHR Rm 'A' Flr Drn Smp Pmp 'B', C013B, would start on High sump level and would continue to run. Alarms RHR RM A SMP LVL HI-HI (1H13-P680-8A1-A2) and RHR A RM FLOODED (1H13-P870-2A-E1) would be received in the control room.

D. RHR Rm 'A' Flr Drn Smp Pmp 'A' and 'B' will remain off. Alarms RHR RM A SMP LVL HI-HI (1H13-P680-8A1-A2) and RHR A RM FLOODED (1H13-P870-2A-E1) would be received in the control room.

Answer: C

Question Comments: Answer A is incorrect because C013A has no power with 11B12 de-energized, and in ALTERNATE, C013B will start on high level since C013A was the last to run. Answer B is incorrect because pump B would have started on a High sump level as indicated in answer C you would also receive the Room Flooded alarm due to the capacity of the sump pump. Answer C is correct because C013B would not be able to pump

2000 gpm, so sumps would back up to the point where the room flooded alarm would come in (6” above the floor) also. Answer D is incorrect because C013B would start on high level, as previously described. 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 Mon Jun 20 08:13:30 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
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6. 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 small break LOCA occurred in the Drywell at 100% power.

Current plant conditions are:

Reactor power	0%
Reactor pressure	950 psig, stable
Reactor water level	-50 inches Wide Range, slowly falling
Drywell pressure	1.9 psig
Drywell Temperature	170°F
Containment pressure	0.7 psig
Containment Temperature	95°F

If the handswitch for RHR Pump ‘A’ on 1H13-P601 was momentarily placed in STOP, which one of the following describes RHR ‘A’ operation under these conditions?

A. If power supplying bus 15AA is lost and then restored, RHR ‘A’ pump will start and remain running.

B. If power supplying bus 15AA is lost and then restored, RHR ‘A’ pump will start but will immediately stop again.

C. If RHR ‘A’ Logic Power supply breaker 72-11A38 is opened, RHR ‘A’ pump will start and remain running.

D. If Wide Range reactor water level falls to -155 inches, RHR ‘A’ pump will start and remain running.

Answer: A

Question Comments: Answer A is correct because AC power from bus 15AA is monitored by relay e12-K3A. When power is lost, a contact on K3A opens which drops out the seal in for the initiation logic and the manual override. The pump handswitch is spring return to auto. If power from bus 15AA was lost, the seal in relay would de-energize, and when power was restored, the initiation logic would again trip due to drywell pressure being above 1.39 psig, and the pump would start. Answer B is incorrect because with the pump handswitch in auto after stop, the pump would start when power was restored, and no trip signal would exist to stop the pump. Answer C is incorrect because the DC logic to automatically start the pump is energize to trip. With no DC power, relays required to fire to start the pump would remain de-energized. Answer D is correct because with the initiation signal and the pump manual override sealed in, automatic start is prevented. The initiation logic is already sealed in, so subsequent initiation signals, though from a different parameter, have no additional effect. 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

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 Fri Jun 10 09:53:23 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

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’ mode, returning to the reactor via E12-F053B.

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

Which one of the following would cause RHR ‘B’pump to trip?

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

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: MikeRasch at Mon Jun 20 08:15:20 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

Comments:

EB QUESTION: 30 **(1.0 Points)**

A plant start up had been in progress at 32% power.

When Recirc pump 'A' was shifted to fast speed, a LOCA in the Drywell occurred.

Division 2 and 3 ECCS initiated properly due to high drywell pressure.

Division 1 ECCS failed to initiate due to loss of RHR 'A' Logic Power.

The following indications are present in the control room:

Reactor power 0%

Reactor pressure 800 psig

Drywell pressure 3 psig

Reactor water level 25 inches Wide Range

Annunciators in on 1H13-P601:

RHR A SYS OOSVC (20A-H6)

DRWL PRESS HI (21A-E7)

LPCS SYS ACTUATED (21A-B8)

LPCS SYS OOSVC (21A-H8)

Status light RHR A LOGIC PWR FAIL STATUS is on (1H13-P601-20B)

The green and amber lights on the Low Pressure Core Spray (LPCS) handswitch on P601 are on and the red light is off.

Which of the following describes availability of LPCS under these conditions?

A. LPCS pump CANNOT be started using any hand switch. LPCS Injection Valve E21F005 will NOT open automatically and CANNOT be opened by placing its control room hand switch on P601 to OPEN.

B. LPCS pump can be started by placing the control room hand switch to START. LPCS Injection Valve E21-F005 will open automatically or by placing its control room hand switch on P601 to OPEN when its pressure permissive is met.

C. LPCS pump can only be started by the local pistol grip hand switch on the front of LPCS Pump breaker 152-1506. LPCS Injection Valve E21-F005 can only be

opened using the local valve manual hand wheel.

D. LPCS pump and LPCS Injection Valve E21-F005 can be operated remotely only from the Division 1 Remote Shutdown Panel, 1H22-P150 (Area 25A, el. 111)

Answer: B

Question Comments: Answer A is incorrect because Logic power and control power to manually start LPCS pump and to automatically open E21F005 are unaffected. In the LPCS logic, all that is affected is a series contact from LSS in the auto start for LPCS pump does not close, since LSS is fired by an RHR ‘A’ relay. Answer B is correct because Logic power and control power to manually start LPCS pump and to automatically open E21F005 are unaffected. The P601 hand switch contact in the LPCS pump start circuit is in parallel with the LSS contact that is affected by the power loss. Answer C is incorrect because The P601 hand switch contact in the LPCS pump start circuit is in parallel with the LSS contact that is affected by the power loss. So the P601 handswitch will start LPCS pump. Answer D is incorrect because both LPCS pump and E21F005 can be operated from P601 as previously described. 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

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 Fri Jun 10 12:40:43 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

Comments:

EB QUESTION: 31 (1.0 Points)

Which of the following is NOT a method of altering Suppression Pool level using High Pressure Core Spray (HPCS) system?

Assume any associated prerequisites are met.

A. To raise Suppression Pool level with HPCS pump secured, gravity drain from the Condensate Storage Tank (CST) by aligning HPCS suction flow path from the CST and throttling open HPCS TEST RTN TO SUPP POOL, E22-F023.

B. To raise Suppression Pool level with HPCS pump running, align HPCS suction flow path from the CST, and close HPCS TEST RTN valves to CST, E22-F010 and E22-F011. Then ensure HPCS MIN FLOW valve E22-F012 is open.

C. To raise Suppression Pool level with HPCS pump running, check/align HPCS suction flow path from the CST, and throttle open HPCS TEST RTN TO SUPP POOL, E22-F023.

D. To lower Suppression Pool level, align HPCS suction flow path from the Suppression Pool, start HPCS pump, and ensure HPCS/RCIC Test Return to CST valves P11-F064 and P11-F065 are open. Then hold open hand switches for HPCS Test Return to CST valves E22-F010 and E22-F011 until the valves are full open, observe the valves automatically stroke closed, and repeat cycling them open, as necessary.

Answer: C

Question Comments: Answer A is incorrect because this is a method listed in 04-1-01-E22-1 step 6.4.2a. Answer B is incorrect because this is a method listed in 04-1-01-E22-1 step 6.4.2b. Answer C is correct because this is not an approved method of raising suppression pool level because it would result in too high of a flow rate. Answer D is incorrect because this is a method listed in 04-1-01-E22-1 step 6.3. 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-00864

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

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 Fri Jun 10 12:40:11 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

Comments:

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

Following a loss of Feedwater from 50% power, High Pressure Core Spray (HPCS) automatically initiated on low reactor water level and restored level.

HPCS Injection Valve, E22-F004, automatically closed as designed as level rose to Level 8.

Which one of the following describes the basis for closure of E22-F004 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 due to high reactor water level, Level 9.

Answer: C

Question Comments: Answer A is incorrect because this is not the reason listed in Tech Spec bases. In this case, SRVs would prevent over-pressurization. Answer B is incorrect because this is not the reason listed in Tech Spec bases. 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 would not prevent reaching level 9 in all cases due the time required for E22F004 to stroke closed. Tier 2 Group 1 This is a NEW question. 10CFR 41.8

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00865

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

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 Fri Jun 10 10:16:45 CDT 2005
Question History:

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2. Question Reviewed by mrasch at Mon Jun 20 14:44:19 CDT 2005
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4. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

EB QUESTION: 33 **(1.0 Points)**

Conditions exist that required Standby Liquid Control injection.

SLC ‘A’ has been initiated from 1H13-P601.

Reactor pressure is stable at 950 psig.

The following indications associated with SLC ‘A’ are present:

SLC A/B Discharge Pressure on P601 indicates 1700 psig.

SLC Storage Tank Level on P601 indicates 4800 gal.

SQUIB VLV READY C41-F004A white light is on.

SLC ‘A’ STOR TK OUTL VLV C41-F001A green light is off, red light is on.

SLC TEST TK OUTL VLV C41-F031 green light is on, red light is off.

SLC PUMP A green and amber lights are off, red light is on.

RWCU PMP SUCT CTMT OTBD ISOL valve is closed.

Squib Valve C41-F004A milliamp meter C41M600A in 1H13-P632 is pegged high.

Which one of the following conditions for SLC 'A' would cause these indications?

A. SLC 'A' is operating properly and injecting into the reactor at rated flow.

B. SLC 'A' discharge piping is ruptured.

C. SLC 'A' Squib Valve C41-F004A actuated and is open, but electrical leads to the valve have electrically shorted to one another.

D. SLC 'A' Squib Valve C41-F004A failed to actuate.

Answer: D

Question Comments: Answer A is incorrect because the squib valve continuity light is on, and discharge pressure is ~ 400 psig too high. Answer B is incorrect because discharge pressure is ~ 400 psig higher than normal. If the pipe was ruptured, discharge pressure would be lower than normal. Answer C is incorrect because usually if the leads short, the control power fuse will blow and the continuity light will then extinguish. Also, discharge pressure is ~ 400 psig too high. Answer D is correct because the squib valve continuity light is on, and discharge pressure is ~ 400 psig too high, indicating very low flow. 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-00866

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

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. 06-OP-1C41-R-0002 Note at Step 5.1.23

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. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 08:18:48 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
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5. 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% rated thermal power.

The ‘A’ Main Bypass Control Valve (BCV) has failed open, so startup has been suspended to troubleshoot the bypass valve.

The Baxter Wilson 500 KV line to the GGNS switchyard is lost causing breakers J5232 and J5228 to trip.

The Turbine Initial Pressure Control (IPC) system responds as designed, except BCV ‘A’ remains failed open.

How would the plant respond automatically for this event?

A. Bypass Control Valves ‘B’ and ‘C’ would open, Reactor Recirc Pumps would transfer to slow speed, and the reactor would scram on high water level.

B. Bypass Control Valves ‘B’ and ‘C’ would open, Reactor Recirc Pumps would transfer to slow speed and the reactor would scram due to TCV closure.

C. Bypass Control Valves ‘B’ and ‘C’ would open, Reactor Recirc Pumps would remain in fast speed, but the reactor would scram due to high neutron flux.

D. Bypass Control Valves ‘B’ and ‘C’ would open, Reactor Recirc Pumps would remain in fast speed, and the reactor would remain operating.

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 at 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

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: MikeRasch at Fri Jun 10 10:30:09 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

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

Comments:

EB QUESTION: 36 (1.0 Points)

Which one of the following sets of conditions will result in a control rod block during startup?

A.
SRM A is upscale and its joystick on 1H13-P680 is in BYPASS

SRM E is partially withdrawn and is reading 70 cps

IRM A is on Range 10 and its joystick on 1H13-P680 is in BYPASS

IRM E is on range 2

B.
SRM A is upscale and its joystick on 1H13-P680 is in BYPASS

SRM E is fully withdrawn and reads 200 cps

IRM A is on range 3

IRM E is on range 3

C.
SRM A is INOP and its joystick on 1H13-P680 is in BYPASS

SRM E is partially withdrawn and reads 200 cps

IRM A is on range 2

IRM E is on range 3

D.
SRM A is INOP and its joystick on 1H13-P680 is in BYPASS

SRM E is INOP

IRM A is on range 9

IRM E is on range 9

Answer: A

Question Comments: Answer A is correct because for SRM A a rod block will occur if it is upscale, even if bypassed, and SRM E is not full in and reads less than 100 counts. Answers B and C are incorrect because SRM E reads >100

counts and SRM A is bypassed. Answer D is incorrect because both IRMs associated with SRMs A and E are on range 9, thus bypassing the SRM rod block. 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

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

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: MikeRasch at Mon Jun 20 08:24:17 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

Comments:

EB QUESTION: 37 (1.0 Points)

The plant is at 100% power.

Which one of the following describes the affect to Average Power Range Monitors (APRMs) if the feeder to inverter panel 1Y86 were to trip open?

04-1-01-L62-1 Attachment V Table 1 is provided.

A. APRMs would be unaffected.

B. Rod block, only, due to APRMs D and H failing downscale due to loss of power. No half scram would occur since power is lost to Division 2 RPS sensors and logic, thus preventing them from tripping.

C. APRMs D and H would continue to indicate power accurately, but an inop signal would be generated due to loss of the LPRM count circuit, resulting in a Division 2 half scram and a rod block.

D. APRMs D and H would lose power. A Division 2 half scram and a rod block would be generated.

Answer: D

Question Comments: Answer A is wrong because 1Y86 feeds APRMs D and H. Answer B is wrong because RPS is de-energize to trip, so loss of sensor or logic power causes a half scram. Also, an APRM inop trip would occur, also producing a half scram. Answer C is incorrect because APRM indication

would fail downscale. Answer D is correct because 1Y86 supplies power to APRMs D and H, they would fail downscale, and Division 2 RPS trip units would lose power, causing some of them to be in their tripped state, resulting in a half scram as well. Tier 2 Group 1 This is a NEW question. 10 CFR 41.2/41.6/41.7

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00870

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

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
2. E-1173-14; 19
3. E-1172-05
4. 04-1-01-L62-1 Att V Table 1

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 Fri Jun 10 13:40:49 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

Comments:

EB QUESTION: 38 (1.0 Points)

The plant was at 100% when a LOCA occurred in the drywell.

RCIC has automatically initiated.

The following indications are present:

Reactor Power 0%

Reactor Pressure 950 psig, stable

Reactor Water Level -50 inches wide range, slowly falling

RCIC Flow Controller AUTO/ 15% output, stable

RCIC Turbine Steam Supply Pressure 950 psig, stable

RCIC Turbine Speed 2560 rpm, stable

RCIC Turbine Exhaust Pressure 3.9 psig, stable

RCIC Pump Suction Pressure -3.1 in Hg, stable

RCIC Pump Discharge Pressure 195 psig, stable

RCIC Pump Flow 810 gpm, stable

These conditions are indicative of which one of the following?

A. RCIC speed controller failure

B. Feed water line 'A' break in the Drywell

C. Feed water line 'B' break in the Drywell

D. RCIC is operating properly and injecting the design flow to the reactor

Answer: C

Question Comments: The key parameter here is a very low RCIC discharge pressure and flow controller output. Answer A is incorrect because the flow controller output is responding properly to the flow feedback signal and controlling ~800 gpm, its setpoint. Answer B is incorrect because RCIC injects into the 'B' feedwater line, which is segregated from the 'A' line by a check valve. Therefore a feedwater line 'A' break would not affect RCIC parameters. Answer C is correct because RCIC parameters indicate RCIC is pumping the correct flow rate against a head that is much lower than reactor pressure. If 'B' feedwater line was ruptured, RCIC would be pumping to the drywell, which is near atmospheric pressure. Answer D is incorrect because the discharge pressure would have to be above reactor pressure for water to go to the reactor 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 Mon Jun 20 08:26:05 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

Comments:

EB QUESTION: 39 (1.0 Points)

The plant was at 100% power when a Loss of Offsite Power coincident with a LOCA in the drywell occurred.

All systems initiated properly except DC breaker 72-11A23, which supplies Automatic Depressurization System (ADS) logic power and Division 1 SRV solenoid power, has tripped.

Drywell pressure is 7 psig.

Reactor water level is -100 inches wide range, falling slowly.

Reactor pressure is 800 psig, stable.

Which one of the following describes the effect these conditions have on ADS functionality?

Drawing E-1161-004 is provided.

A. ADS ‘A’ has initiated due to loss of logic power. ADS valves B21-F051A and B21-F051B are open since they have alternate power through the Division 1 Remote Shutdown Panel.

B. ADS ‘A’ has initiated due to loss of logic power. 8 ADS valves are open due to availability of Division 2 solenoid power.

C. ADS ‘A’ will NOT initiate automatically, but ADS valves will automatically open due to Division 2 power.

D. ADS ‘A’ will NOT initiate automatically, but 8 ADS valves can be opened manually using their 1H13-P601 hand switches.

Answer: C

Question Comments: Answers A and B are incorrect because ADS logic is “energize to initiate”. Answer C is correct because ADS logic is “energize to initiate”, so ADS ‘A’ cannot initiate. Division 2 ADS is unaffected, so Div 2 solenoids will open the ADS valves. Answer D is incorrect because P601 handswitches are for the Division 1 solenoids, which have lost power. Tier 2 Group 1 This is a NEW question. 10CFR 41.7

Image Reference: None

Open Reference Question

Handout 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]

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 Fri Jun 10 11:34:17 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

Comments:

EB QUESTION: 40 (1.0 Points)

The plant was operating at 100% power.

A large break LOCA caused Reactor Water Level to drop to -70 inches Wide Range.

RCIC is restoring RPV level and is -60 inches Wide Range.

An operator just reported that there is a large leak on the Drywell Chilled Water Return

piping inside containment.

You observe the Isolation Status Board and notice that NO isolation valves have closed.

What is the minimum required action regarding the isolation failure?

A. Manually isolate one valve in every penetration which should have isolated.

B. Isolate P53 to the Auxiliary Building to cause a complete isolation. All valves affected by the failure to isolate will fail closed.

C. Manually isolate one valve in each penetration which should have isolated, except Instrument Air (P53) and Plant Service Water (P44) do NOT need to be isolated if they will be immediately reopened.

D. Manually isolate one valve in each penetration which should have isolated, except Drywell Chilled Water (P72), Plant Service Water (P44), and Instrument Air (P53) do NOT need to be isolated if they will be immediately reopened.

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: 1: Fundamental Knowledge or Memory
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

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 Mon Jun 20 08:27:42 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

Comments:

EB QUESTION: 41 **(1.0 Points)**

Emergency Depressurization of the reactor had to be performed.

Eight ADS/SRVs (Automatic Depressurization System / Safety Relief Valves) were opened using their respective hand switches on H13-P601.

Reactor pressure lowered to zero psig.

NO other manipulations involving SRVs have been performed.

ADS logic is in standby.

NO failures have occurred in the ADS/SRV system.

Both ADS accumulator pressures are 160 psig.

What can the oncoming reactor operator now ascertain regarding the status of the ADS valves from H13-P601, H13-P631 (Main Control Room Back Panels), and H13-P628 (Upper Control Room)?

A. On section 19C (the apron section) of H13-P601, the red light above each ADS valve hand switch will be illuminated and green light will be off, indicating the respective ADS valve is open.

B. The ADS Logic A/E continuity lights on H13-P628 will be off, meaning power is applied to each ADS valve

C. The red ADS SRV Status light on section 19B (the vertical section) of H13-P601 will be illuminated, indicating power is applied to each ADS valve

D. All green lights will be illuminated and all red lights will be off for each ADS valve on H13-P601, H13-P631, and H13-P628. All continuity lights on H13-P631 and H13-P628 will be illuminated. ADS valve position CANNOT be positively determined from these indications.

Answer: D

Question Comments: Answer A is incorrect because the SRV tailpipe pressures are <30psig, the pressure switch setpoint feeding the P601 handswitch lights. Answer B is incorrect because ADS logic is in standby. Answer C is incorrect

because Div 2 solenoids have not been energized. Answer D is correct because all tailpipe pressure switches are reset due to depressurization, there is no indication for Div 1 solenoids which are energized, and logic systems are in standby, so all continuity lights are on. 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: 2: Comprehension or Analysis

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

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 08:29:28 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

Comments:

EB QUESTION: 42 (1.0 Points)

The plant is at 90% power.

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

Reactor level is 43 inches narrow range and slowly trending up.

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 operator response for this condition?

A. Immediately insert a manual scram.

B. 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.

C. 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.

D. Take manual control of the Master Level Controller and control level 32 inches to 42 inches.

Answer: B

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 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 correct because ONEP 05-1-02-V-6 specifically directs this action for the given failure. Answer C is incorrect because RFP ‘B’ controller is responding properly to the failure of ‘A’. Answer D is incorrect because the master level controller controller is responding properly to the failure of RFP ‘A’ speed controller. 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

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-6Step 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: MikeRasch at Mon Jun 20 08:31:07 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

Comments:

EB QUESTION: 43 (1.0 Points)

The plant is in Mode 1.

The equalizing valve for Main Steam Line Flow transmitter C34-N030A has begun to leak by the seat.

The steam flow signals inputting to the Digital Feedwater Control System (DFCS) are:

C34-N030A - 2.8 mlbm/hr

C34-N030B - 4.1 mlbm/hr

C34-N030C - 4.1 mlbm/hr

C34-N030D - 4.2 mlbm/hr

The Total Steam Flow estimated by DFCS for the current power level is 16.2 mlbm/hr.

Which one of the following describes the response of actual reactor water level and the Digital Feed Control System (DFCS)?

A. Reactor water level will be slightly lower than before and will remain stable. DFCS will remain in three element control.

B. Reactor water level will remain stable when the level dominance of the DFCS takes over and automatically substitutes calculated steam flow for the failed value.

C. Reactor water level will immediately rise then return to normal level when the DFCS system automatically de-selects and locks out three element control and

selects single element control.

D. Reactor water level will immediately fall then return to normal level when the DFCS system automatically de-selects and locks out three element control and selects single element control.

Answer: A

Question Comments: Answer A is correct because the difference between the highest and lowest stem flow signals does not exceed 1.6 mlbm/hr, so 3-element control would not be automatically disabled. Answer B is incorrect because there is no substitution for failed steam flow signals used at GGNS. Answer C is incorrect because level would initial fall as sensed steam flow lowered, and the difference between the highest and lowest stem flow signals does not exceed 1.6 mlbm/hr, so 3-element control would not be automatically disabled. Answer D is incorrect because the difference between the highest and lowest stem flow signals does not exceed 1.6 mlbm/hr, so 3-element control would not be automatically disabled. Tier 2 Group 1 This is a MODIFIED QUESTION. Last used August 2002 question id WRI705. 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: 10.6
2. CourseID: GLP-OPS-C3400 Objective: 10.7

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: MikeRasch at Mon Jun 13 08:03:40 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

Comments:

EB QUESTION: 44 (1.0 Points)

Standby Gas Treatment System (SGTS) 'A' was manually initiated 10 minutes ago for a planned move of radioactive materials on the refueling floor and is running properly.

Enclosure Building differential pressure is now -0.6 inches wc.

SGTS 'B' is to be manually initiated, too.

Which one of the following depicts how SGTS 'B' flow control damper T48-F005 (Steam Tunnel Outside Containment) and flow control vane T48-F500B should respond if operating properly?

A.

T48-F005 goes to intermediate position.

T48-F500B stays open until either -0.75 inches wc is reached or 120 seconds elapses, then it modulates.

B. T48-F005 goes to intermediate position.

T48-F500B stays open until either -0.75 inches wc is reached or 90 seconds elapses, then it modulates.

C. T48-F005 goes full open, and after 120 seconds throttles to intermediate position.

T48-F500B stays open until either -0.75 inches wc is reached or 90 seconds elapses, then it modulates.

D. T48-F005 goes full open, and after 90 seconds opens to intermediate position.

T48-F500B stays open until either -0.75 inches wc is reached or 120 seconds elapses, then it modulates.

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, and the proper pressure and timer values are stated for T48F500B. Answer B is incorrect because the wrong timer value for T48F500B is given. Answer C is incorrect because T48F005 would go to intermediate position right away instead of 90 sec later, and the wrong timer value for T48F500B is given. Answer D is incorrect because T48F005 would go to intermediate position right away instead of 90 sec later. 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: 1: Fundamental Knowledge or Memory
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: MikeRasch at Mon Jun 13 07:54:52 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

Comments:

EB QUESTION: 45 **(1.0 Points)**

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

High Pressure Core Spray (HPCS) pump is running on minimum flow with suction from the Suppression Pool.

Reactor level is normal.

Which one of the following describes operation of HPCS with bus 11DC de-energized?

A. HPCS Pump breaker 152-1702 can be tripped using its control room hand switch on 1H13-P601.

HPCS Injection Valve E22-F004 CANNOT be opened from the control room.

B. HPCS Pump breaker 152-1702 can be tripped by taking its local pistol grip hand switch on the front of breaker 152-1702 to OPEN, but it CANNOT be re-closed.

HPCS Injection Valve E22-F004 CANNOT be opened from the control room.

C. HPCS Pump breaker 152-1702 will automatically trip if windings in HPCS pump motor fault.

HPCS Injection Valve E22-F004 can be opened from the control room.

D. HPCS Pump breaker 152-1702 CANNOT be tripped with a hand switch and will NOT trip automatically.

HPCS Injection Valve E22-F004 can be opened 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 operates on and is controlled by 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 Mon Jun 13 09:08:22 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

Comments:

EB QUESTION: 46 (1.0 Points)

Static Inverter 1Y95 was on ALTERNATE supply for planned maintenance when a LOCA in the drywell occurred.

Division 1 and Division 2 ECCS initiated.

Fifteen minutes later, Bus 16AB locked out.

The following alarms are present on H13-P807:

STATIC INVERTER 1Y79 TROUBLE (3A-G4)

STATIC INVERTER 1Y80 TROUBLE (3A-H1)

STATIC INVERTER 1Y81 TROUBLE (3A-H2)

STATIC INVERTER 1Y82 TROUBLE (3A-H3)

STATIC INVERTER 1Y97 TROUBLE (3A-H4)

STATIC INVERTER 1Y98 TROUBLE (3A-G3)

What is the status of the Static Inverters?

04-1-01-L62-1 Attachment III is provided.

A. 1Y87, 1Y88, 1Y95, 1Y96 are on their normal supply.

1Y79, 1Y80, 1Y81, 1Y82, 1Y97, 1Y98, 1Y99 are on their alternate supply.

B. 1Y87, 1Y88, 1Y95, 1Y96 are on their alternate supply.

1Y79, 1Y80, 1Y81, 1Y82, 1Y97, 1Y98, 1Y99 are on their alternate supply.

C.

1Y87, 1Y88, 1Y96 are on their normal supply.

1Y79, 1Y80, 1Y81, 1Y82, 1Y97, 1Y98, 1Y99 are on their normal supply.

1Y95 is de-energized.

D. 1Y87, 1Y88, 1Y96 are on their normal supply.

1Y79, 1Y80, 1Y81, 1Y82, 1Y97, 1Y98, 1Y99 are on their alternate supply.

1Y95 is de-energized.

Answer: C

Question Comments: Answers A, B, and D are incorrect because no inverters would transfer from their normal DC supply to their alternate AC supply. Answer C is correct because no inverters would transfer from their normal DC supply to their alternate AC supply, and 1Y95 would have had its manual bypass switch in ALTERNATE (AC) and would not have been able to auto transfer to normal (DC). Bus 16AB supplies alternate power for 1Y95, so 1Y95 is de-energized. Tier 2 Group 1 This is a NEW question. 10 CFR 41.4

Image Reference: None

Open Reference Question

Handout 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 Mon Jun 20 08:34:35 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

Comments:

EB QUESTION: 47 (1.0 Points)

The plant is in Mode 1. The plant DC system is OPERABLE in its STANDBY configuration.

Battery chargers 1A4 and 1A5 normal/equalize switches are in NORMAL with load sharing ON.

Load Control Center 15BA6 trips, resulting in loss of power to battery charger 1A4.

How does this condition affect the Division 1 battery parameters?

A. The ‘A’ battery parameters will be unaffected, since battery charger 1A5 will pick

up load as necessary, without operator action.

B. The ‘A’ battery parameters will be unaffected only if battery charger 1A5 is manually placed into service immediately following loss of 15BA6.

C. ‘A’ battery bank voltage will slowly deteriorate. Average specific gravity will go down.

D. ‘A’ battery bank voltage will slowly deteriorate. Average specific gravity will go up.

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 Mon Jun 13 09:22:05 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

Comments:

EB QUESTION: 48 (1.0 Points)

The plant is at 50% power.

Due to grid instability, Division 2 Diesel Generator (DG12) is supplying bus 16AB which is separated from offsite power in accordance with the Loss of AC Power ONEP.

The reference leg for condensing pot B21-D004B ruptures in the drywell.

What will be the effect on DG12 and bus 16AB?

A. DG12 output breaker will open and Division 2 loads will be shed when sensed wide range reactor level goes below Level 1. After a time delay, DG12 output breaker will re-close and appropriate loads will sequence on.

B. DG12 output breaker will open and Division 2 loads will be shed when high drywell pressure is reached. After a time delay, DG12 output breaker will re-close and

appropriate loads will sequence on.

C. DG12 output breaker will remain closed. Division 2 loads will be shed when sensed wide range reactor level goes below Level 1. After a time delay, appropriate loads will sequence on.

D. DG12 output breaker will remain closed. Division 2 loads will be shed when high drywell pressure is reached. After a time delay, appropriate loads will sequence on.

Answer: D

Question Comments: Answers A and B are incorrect because DG12 output breaker will remain closed since it is carrying the bus alone. Answer C is incorrect because rupture of the reference leg would cause indicated level to go high, not low. Answer D is correct because DG12 output breaker will remain closed since it is carrying the bus alone, and a Div 2 LOCA signal would have to be due to drywell pressure since sensed level fails high. 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

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: MikeRasch at Mon Jun 20 09:04:22 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

Comments:

EB QUESTION: 49 (1.0 Points)

The plant was at 100% power when the feeder breaker to bus 16AB from ESF Transformer 21 tripped.

Division 2 Diesel Generator (DG12) automatically started and is running at 450 rpm, but DG12 output breaker, 152-1608, failed to close due to blowing control power fuses.

Bus 16AB is still de-energized 20 seconds following the loss of power to the bus.

Which one group of the following indications on 1H13-P864 associated with Division 2 Diesel Generator (DG12) would be expected for this condition?

Assume all indicators are operating as designed.

A. Annunciator DIV 2 LSS SYS FAIL (1H13-P864-2A-H1) illuminated

DG12 Frequency meter P75-R601B on 1H13-P864 downscale

DG-12 READY TO LOAD status light on 1H13-P864-2B extinguished

B. Annunciator DIV 2 LSS SYS FAIL (1H13-P864-2A-H1) illuminated

DG12 Frequency meter P75-R601B on 1H13-P864 downscale

DG-12 READY TO LOAD status light on 1H13-P864-2B illuminated

C. Annunciator DIV 2 LSS SYS FAIL (1H13-P864-2A-H1) illuminated

DG12 Frequency meter P75-R601B on 1H13-P864 indicating 60 Hz

DG-12 READY TO LOAD status light on 1H13-P864-2B extinguished

D. Annunciator DIV 2 LSS SYS FAIL (1H13-P864-2A-H1) extinguished

DG12 Frequency meter P75-R601B on 1H13-P864 indicating 60 Hz

DG-12 READY TO LOAD status light on 1H13-P864-2B illuminated

Answer: C

Question Comments: Answer A is incorrect because DG frequency would indicate ~60 hz since DG12 is running. Answer B is incorrect because the DG-12 READY TO LOAD status light is powered from bus 16AB, DG frequency would indicate ~60 hz since DG12 is running. Answer C correct because no power is being supplied to bus 16AB but the DG is running. Answer D is incorrect because the DG-12 READY TO LOAD status light is powered from bus 16AB. 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-00881

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis
Objectives:

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

KA References:

1. 264000
2. GENERIC 2.4.48 Ability to interpret control room indications to verify the status and operation of [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: MikeRasch at Mon Jun 20 09:05:22 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

Comments:

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

The plant is operating at 100% power.

Instrument Air Supply Header to Auxiliary Building valve P53-F026A fails closed due to a relay contact failure.

What effect will this have on the Drywell Coolers/Chillers?

A. The Division 1 Outlet dampers for the Drywell Coolers will fail open, and the Division 2 Outlet dampers will fail closed. Drywell Chiller Pressure Control valves ‘A’ will fail closed and Drywell Chiller Pressure Control valves ‘B’ will fail open.

B. The Division 1 Outlet dampers for the Drywell Coolers will fail closed, and the Division 2 Outlet dampers will fail open. Drywell Chiller Pressure Control valves ‘A’ will fail closed and Drywell Chiller Pressure Control valves ‘B’ will fail open.

C. The Division 1 Outlet dampers for the Drywell Coolers will fail closed, and the Division 2 Outlet dampers will fail open. Drywell Chiller Pressure Control valves ‘A’ will fail open and Drywell Chiller Pressure Control valves ‘B’ will fail closed.

D. The Division 1 Outlet dampers for the Drywell Coolers will fail open, and the Division 2 Outlet dampers will fail closed. Drywell Chiller Pressure Control valves ‘A’ will fail open and Drywell Chiller Pressure Control valves ‘B’ will fail closed.

Answer: B

Question Comments: Answer A is incorrect because Div 2 dampers fail open and Div 1 closed. Answer B is correct because dampers and valves fail as stated. Answer C is incorrect because Drywell chiller B valves fail open and A fails closed. Answer D is correct because Div 2 dampers and Drywell chiller B valves fail open and A dampers and valves fail closed. 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-00882

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-OPS-M5100 Objective: 9.3; 10.3

KA References:

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

References:

1. M-1101
2. M-1072B

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. M51: Drywell Cooling System
2. P53: Instrument Air System
3. P72: Drywell Chill Water System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 09:17:44 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

Comments:

EB QUESTION: 51 (1.0 Points)

The plant is operating at rated conditions.

Unit 2 Instrument Air Compressor is in service with Plant Air Dryer B P51-D001B in service.

Plant Air Dryer A P51-D001A is shutdown to a standby lineup.

Service Air Compressor A is in service.

Plant Air Dryer B After Filter P51-D003B has suddenly become clogged with a foreign object and is passing less than 1% of its normal volume of air.

Which one of the following describes the response of the Instrument, Plant and Service Air Systems to this problem?

Drawings M-1067A and G , M-1068D and M-1126 are provided.

A. Plant Air Dryer A will automatically align itself and supply the Instrument Air Header with air being supplied fro Unit 2 Instrument Air Compressor.

B. Unit 1 Instrument Air Compressor will auto start and align itself through the Unit 1 Instrument Air Dryer Skid.

C. Service Air Compressor A will automatically align itself to the Instrument Air Header via P52-F500 at 95 psig to maintain Instrument Air Header pressure.

D. Unit 2 Instrument Air Compressor will have to be manually aligned to the Instrument Air Header via Plant Air Dryer Bypass valve P51-F209B until Plant Air Dryer A can be started.

Answer: D

Question Comments: GGNS has been in the process of altering the Instrument Air, Service Air and Plant Air systems installing new air drying systems and air compressors. This is an attempt to upgrade the systems for reliability. Plant Air Compressor B has replace Service Air Compressor B. Unit 1 and Unit 2 Instrument Air Compressors are operational. Unit 1 Instrument Air Dryer skid is still installed but valved out of service per the System Operating Instructions. The new Plant Air Dryer skids are operational and supplying the Instrument Air drying needs. With one Plant Air Dryer Skid in service and the second skid shutdown, a clogged filter on the outlet of the in service Air Dryer will reduce/block air flow to the Instrument Air Header. No matter the source of the air, it must go through the in service

air dryer before going to the Instrument Air Header. Answer A is INCORRECT because the Plant Air Dryer that is out of service will not automatically align it self to take the load. Answer B is INCORRECT because the Unit 1 Instrument Air Dryer skid is valved out of service with manual valves and requires local operator action to place it in service. Answer C is INCORRECT because of the location of the Plant Air Dryer that is blocking flow will not allow the service air compressor to supply the system. Answer D is CORRECT because opening P51-F209B will bypass the air dryer and allow pressurization of the air header with raw compressed air. Tier 2 Group 1 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-00883

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

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:

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 Mon Jun 20 09:37:57 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

Comments:

EB QUESTION: 52 (1.0 Points)

The plant is in Mode 3.

RHR 'A' is 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 NOT be indicative of an RHR 'A' Heat Exchanger tube rupture if a tube leak occurred?

A. RHR 'A' pump discharge pressure trending down

B. Reactor water level trending up

C. SSW 'A' radiation monitor readings trending up

D. Reactor coolant temperature trending up

Answer: B

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 correct because RHR would be at a higher pressure than SSW, so RPV level would go down. Answer C is incorrect 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

Objectives:

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

KA References:

1. 400000 K1.01 Service water system [3.2/3.3]

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 Mon Jun 13 09:58:09 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
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4. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

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: 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

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: MikeRasch at Mon Jun 13 10:10:30 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

Comments:

EB QUESTION: 54 (1.0 Points)

A control rod sequence exchange is being performed at 80% power.

As part of the control rod movement plan, the operator selects control rod 28-17, which is at notch position 36, and begins to continuously withdraw it.

The control rod stops withdrawing and settles at notch position 40 due to an expected control rod block.

What is the purpose of this control rod block?

A. Mitigate a control rod drop event

B. Prevent exceeding fuel preconditioning limits

C. Prevent violation of the Minimum Critical Power Ratio (MCPR) Safety Limit

D. Normalize core exposure burn rates

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: MikeRasch at Mon Jun 13 10:15:34 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

Comments:

EB QUESTION: 55 **(1.0 Points)**

At 100% power, Reactor Recirculation loop flows must be within 5% of rated core flow of one another.

What is the basis of this requirement?

A. To normalize the core radial flux distribution.

B. To meet core flow coast down assumptions of the LOCA analysis.

C. To prevent excessive loading of Recirc pump motor windings and electrical penetration.

D. To minimize vibration stresses on Recirc piping caused by jet pump cavitation.

Answer: B

Question Comments: Tech Spec bases 3.4.1 specifically states that the limit for Recirc flow mismatch is an assumption in the LOCA analysis that assures sufficient flow and cooling during coastdown of the unbroken Recirc loop. That is

why answer B is correct. Answers A, C, and D are unrelated to this, and that is why they are incorrect. Tier 2 Group 2 This is a NEW question. 10CFR 41.3/41.10/43.2/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00887

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-OPS-B3300 Objective: 2; 44
2. CourseID: GLP-OPS-MCD16 Objective: 4

KA References:

1. 202001 K1.01 Core flow [3.6/3.7]

References:

1. Tech Spec Bases B3.4.1
2. Tech Spec Bases Surveillance 3.4.1.1

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 Mon Jun 13 10:28:15 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

Comments:

EB QUESTION: 56 (1.0 Points)

The plant was at 100%.

Reactor Recirc Flow Control Valves were at 68% valve position.

Then, Feed Water Line 'A' in the drywell, upstream of feed water check valve B21-F010A, suffered a guillotine break.

All systems responded as designed.

Maximum drywell pressure was 4.5 psig.

Maximum reactor pressure was 1025 psig.

Minimum reactor water level was -80 inches wide range.

Which one of the following describes the Reactor Recirc System status five minutes after the feed water line break?

A. Recirc Pump breakers CB-1A/B, CB-2A/B, CB-4A/B, and CB5A/B are OPEN. CB-3A/B are CLOSED.

Recirc Flow Control Valves 'A' and 'B' are approximately 20% open.

B. Recirc Pump breakers CB-1A/B, CB-2A/B, CB-3A/B, CB-4A/B, and CB5A/B are OPEN.

Recirc Flow Control Valves 'A' and 'B' are approximately 20% open.

C. Recirc Pump breakers CB-1A/B, CB-2A/B, CB-4A/B, and CB5A/B are OPEN. CB-3A/B are CLOSED.

Recirc Flow Control Valves 'A' and 'B' are approximately 68% open.

D. Recirc Pump breakers CB-1A/B, CB-2A/B, CB-3A/B, CB-4A/B, and CB5A/B are OPEN.

Recirc Flow Control Valves 'A' and 'B' are approximately 68% open.

Answer: C

Question Comments: For this event, all feedwater flow to the reactor would be lost. Water level would rapidly fall to -41.6 inches. Recirc pumps would trip to slow speed at 11.4 inches (CB-5s open, CB-1s and CB-2s close) and an ATWS RPT would occur at -41.6 inches (CB-1s, 2s, 4s, 5s open). The CB-3s are the only breakers that remain closed. They would only trip on EOC-RPT, which does not occur since power is <40%, where it is bypassed, by the time the turbine trips on reverse power. Recirc flow control valves do not run back to 20% because both still RFPs are running by the time Recirc pumps transfer out of fast speed. Also, both Recirc HPUs trip almost immediately due to high drywell pressure, 1.23 psig, from the FW line break. Answer A is incorrect because Recirc FCVs do not runback as stated above. Answer B is incorrect because CB-3s remain closed and Recirc FCVs do not runback as stated above. Answer C is correct because of reasons stated above. Answer D is incorrect because of reasons stated above. 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

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 Mon Jun 20 09:41:46 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

Comments:

EB QUESTION: 57 (1.0 Points)

Which one of the following is a requirement for use of Reactor Water Clean-up System (RWCU) as an alternate method of decay heat removal in Mode 5?

A. At least one recirculation pump must be placed in operation.

B.

The Component Cooling Water temperature control valve, P44-F501, must be set at 65°F.

C. RWCU REGEN HX BYP VLV G33-F107 must be fully opened and must remain fully opened for all conditions.

D. A temporary thermocouple must be installed to monitor reactor coolant temperature.

Answer: D

Question Comments: Answer A is incorrect because RWCU operation does not require recirculation pump operation. Answer B is incorrect because the minimum allowed CCW TCV setting is 70°F. Answer C is incorrect because G33F107 is required to be throttled open/closed as necessary to control temperature. Answer D is correct because there is insufficient RWCU flow to provide an accurate bulk coolant temperature. Tier 2 Group 2 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-00889

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-OPS-E1200 Objective: 14.1
2. CourseID: GLP-OPS-G3336 Objective: 10.1

KA References:

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

References:

1. 04-1-01-E12-1 3.8.16.c
2. 04-1-01-G33-1
3. 05-1-02-III-1

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

Task References:

Question Last Revised By: MikeRasch at Mon Jun 13 10:45:43 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 10:45:43 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

Comments:

EB QUESTION: 58 (1.0 Points)

The plant is in a Station Blackout with the following parameters:

Reactor power 0%

Reactor water level -110 inches wide range

Reactor pressure 550 psig

Suppression pool temperature 145°F

Suppression Pool level 10.3 feet

ADS accumulator pressure 145 psig

Which one of the following is the preferred method to be used to depressurize the RPV under these conditions?

A.

Automatic Depressurization System (ADS) valves

B. Reactor Core Isolation Cooling (RCIC)

C. Reactor Water Clean-up (RWCU)

D. Main Steam Drains and Offgas Preheater

Answer: B

Question Comments: Emergency depressurization is required due to suppression pool level <14.56'. Answer A is incorrect because SRV use is not allowed due to suppression pool level < 10.5'. Answer B is correct because it is a system listed by EP-2A step 55B that is inboard of MSIVs and is designed to function during a SBO. It can be used , even at low suppression pool level, during an emergency. Answer C is incorrect because RWCU is de-energized during a SBO and is not listed in EP-2A step 55B. Answer D is incorrect because MSIVs are closed in a SBO, so there is no steam flow path. 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. EP-3 step 43
2. EP-2A step 55B

3. 02-S-01-27 (No rod position indication during a SBO, therefore enter EP-2A)
4. 04-1-01-E51-1 Steps 3.3; 3.4

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
2. E51: Reactor Core Isolation Cooling System
3. G33: Reactor Water Cleanup
4. M71: Containment and Drywell Instrumentation System
5. R21: 4.16 KV AC Power System

Categories:

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

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 09:44:41 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

Comments:

EB QUESTION: 59 (1.0 Points)

The plant was operating at 100% power when a loss of all offsite power caused all Main Steam Isolation Valves to close and a Reactor Scram.

Hydrogen Water Chemistry was in service

Which one of the following describes the response of the Hydrogen Water Chemistry (HWC) System and its affect on the Offgas system?

A. Hydrogen Water Chemistry will commence a normal shutdown of the HWC system that will provide timed reduction in injection of Hydrogen and Oxygen into Offgas to allow for recombination of Hydrogen and Oxygen preventing the possibility of a fire in Offgas. The loss of heating steam to the Offgas Preheater will NOT significantly affect the Hydrogen concentrations in Offgas.

B. Hydrogen Water Chemistry will commence a Hydrogen Immediate Trip with Normal Oxygen shutdown of the HWC system that will immediately isolate Hydrogen to the Condensate System but allow a normal shutdown of the Oxygen injection to the Offgas System. Even though heating steam to the Offgas Preheater is lost, the chances of an Offgas Hydrogen fire are reduced.

C. The loss of power will result in an Emergency HWC shutdown causing elevated Hydrogen levels in the Offgas system since there is reduced Oxygen for the Hydrogen to recombine with. This combined with the loss of heating steam to the Offgas Preheater will raise the possibility of a fire in the Offgas system.

D. Hydrogen Water Chemistry will commence a normal shutdown of the HWC system that will provide timed reduction in injection of Hydrogen and Oxygen into Offgas, however due to the loss of heating steam to the Offgas Preheater the possibility of a fire in the Offgas System is raised.

Answer: B

Question Comments: Answers A and D are incorrect because hydrogen injection will immediately be isolated. Answer B is correct because the system is designed to immediately secure hydrogen with a timed reduction of oxygen to ensure excess oxygen available for recombination. This ensures very low hydrogen concentrations to reduce possibility of a fire. Answer C is incorrect because, as stated for answer B, the result is reduced levels of hydrogen and lower chances of fire. 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-P7300 Objective: 10; 11.4; 11.5; 11.6
2. CourseID: GLP-OPS-N6465 Objective: 14.3

KA References:

1. 239001 K3.04 Offgas system [2.8/2.8]
2. 239001 K1.07 Offgas system [2.9/3.1]
3. 271000 A1.13 Hydrogen gas concentration [3.2/3.7]
4. 271000 A1.14 Oxygen gas concentration [2.7/3.0]
5. 271000 K1.06 Main steam system [2.8/2.9]
6. 271000 K1.08 Oxygen injection system: Plant-Specific [2.3/2.3]
7. 271000 K4.04 The prevention of hydrogen explosions and/or fires 3 [3.3/3.6]

References:

1. 04-1-02-1H13-P845 1A-D7 1.3.6, 1.3.7, 3.1.2
2. 04-1-01-P73-1
3. E-7176-005, 006

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
3. P73: Hydrogen Water Chemistry

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 09:49:29 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

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 effluent of either MSIV LCS.

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) ‘A’ and ‘B’ are in operation.

D. Operation of the Outboard MSIV LCS in conjunction with Standby Gas Treatment System (SGTS) ‘B’ is the preferred alignment, since SGTS ‘B#146; provides the longest transport time and each is powered from Division 2.

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

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 Mon Jun 20 09:52:39 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

Comments:

EB QUESTION: 61 (1.0 Points)

The plant was operating at 99% power, when a winding fault caused Service Transformer (ST) 21 to lock out.

The running Electro-Hydraulic Control (EHC) fluid pumps tripped, and the standby EHC pump failed to automatically start.

The resulting pressure transient caused some amount of fuel damage.

Bus 14AE has been reenergized from ST11 and under-voltage lockout relays were reset.

EHC pumps have NOT been restarted but are available.

All other systems affected by the power loss have been recovered.

An ATWS currently exists with the following conditions:

Reactor power 15%

Reactor pressure 1050 psig controlled by Safety Relief Valves (SRVs) manually 800 - 1060 psig.

Reactor level -80 inches wide range, being controlled on startup level control with Reactor Feed Pump ‘A’

Offgas Pretreatment Radiation monitor 2000 mr/hr (above the high alarm, below the Hi-Hi alarm)

Average Main Steam Line Radiation Monitor reading 2300 mr/hr (below the high alarm)

Main Steam Isolation Valves (MSIVs) are open.

Average suppression pool temperature is 96°F, slowly rising.

Which one of the following represents the pressure control strategy to be followed under these conditions?

A. Close MSIVs. Manually control SRVs 800 psig to 1060 psig.

B. Place Suppression Pool Cooling A and B in service, and allow SRVs to cycle on Low-Low Set. Augment SRVs with Main Steam Line Drains.

C. Restart EHC pumps. Ensure Main Bypass Valves are operating, and lower Pressure Reference to 900 psig. Install Emergency Procedure Attachments 7 and 8.

D. Take manual control of SRVs and Main Steam Line Drains, and lower pressure to 450 psig to 600 psig.

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

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 Mon Jun 20 09:59:28 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

Comments:

EB QUESTION: 62 **(1.0 Points)**

During the Main Turbine roll to rated speed, procedures require at least a 55°F margin on the Turbine Stress Evaluator (TSE).

What is the basis for this requirement?

A.

Ensure adequate steam flow to cool latter blade stages.

B. Prevent blade failure due to resonance vibration due to sustained operation at critical speeds.

C. Ensure adequate steam flow for synchronization and minimum loading of the Main Generator.

D. Provide sufficient clearances between rotating and stationary turbine blades.

Answer: B

Question Comments: The caution at step7.1.5a of 03-1-01-1 states the 55°F margin on TSE is to ensure the turbine can be rolled to 1800 rpm with no hold points at critical speeds. Critical speed causes resonant vibrations that can damage turbine blades. This is why answer B is correct. Answer A is incorrect because this adverse effect is caused by sustained low steam flow at 1800 rpm. Answer C is incorrect because the associated requirement is to have 20% bypass valve opening, not TSE margin. Answer D is incorrect because this is a turbine relative expansion concern. Tier 2 Group 2 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-00894

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-OPS-N3000 Objective: 2.8; 8
2. CourseID: GLP-OPS-IOI01 Objective: 32.2

KA References:

1. 245000 A3.02 Turbine roll to rated speed [2.8/2.8]

References:

1. 03-1-01-1 section 2.12.3, caution section 7.1.5

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

Categories:

1. Integrated Plant Operations
2. Systems
3. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 10:02:26 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

Comments:

EB QUESTION: 63 (1.0 Points)

The plant is operating at 100% power.

Condensate Pump Recirc Isolation valve N19-F010 and Condensate Booster Pump Recirc Isolation valve N19-F057 are closed for I&C work on the Bailey INFI-90 system in H22-P171, scheduled to begin later in the shift.

Which one of the following describes what would happen if the flow signal from flow element N19-N065 on the outlet of the Low Pressure Feed Water Heaters were to intermittently fail to 2.9 mlbm/hr for 75 seconds, and then return to normal?

A. Condensate Pump A would trip. Condensate Booster Pumps would trip due to low suction pressure. The reactor would scram due to low water level.

B. Condensate Booster Pump A would trip. Reactor Feed Pumps would trip due to low suction pressure. The reactor would scram due to low water level.

C. Condensate Pump A would trip. Reactor Feed Pump speeds would rise to maintain normal normal water level.

D. Condensate Booster Pump A would trip. Reactor Feed Pump speeds would rise to maintain normal water level.

Answer: D

Question Comments: Minimum flow lines are isolated, so no additional flow will be established. Answer A is incorrect because the condensate low flow trip signal, < 1.2 mlbm/hr per pump, has to exist for 90 seconds to trip the first condensate pump, so no condensate pumps would trip. Also, reactor feed pumps would remain running, and the reactor would not scram. Answer B is incorrect because reactor feed pumps would remain running , and the reactor would not scram. Answer C is incorrect because the condensate low flow trip signal, < 1.2 mlbm/hr per pump, has to exist for 90 seconds to trip the first condensate pump, so no condensate pumps would trip. Answer D is correct because there would be no flow path to provide additional booster pump flow to meet the minimum requirement of 3 mlbm/hr total for 3 booster pumps running. One booster pump would trip in 60 seconds, and the minimum flow requirement would shift to 2 mlbm/hr, and would thus be satisfied. The condensate system is designed such that 2 booster pumps can provide enough flow for 100% power, so RFPs and the reactor would remain running. 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: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-OPS-N1900 Objective: 2; 13; 14; 21

KA References:

1. 259001 A2.03 Loss of condensate pump(s) [3.6/3.6]

References:

1. 04-1-01-N19-1 sections 3.4, 3.6, 3.20, 3.21
2. M-1053 A and B

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 Mon Jun 20 10:12:56 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

Comments:

EB QUESTION: 64 (1.0 Points)

Which one of the following is NOT an accepted method for monitoring Reactor Coolant System (RCS) leakage in the drywell to meet Tech Spec leakage limits?

Portions of the Daily Operations Log 06-OP-1000-D-0001 are provided.

A. Divide the volume of the Drywell Equipment Drain Sump by the time between automatic sump pump downs indicated on PDS to determine identified leakage.

B. Use the total rise in Drywell Equipment Drain Sump Level indicated on recorder E31-R185 (area 9, 139') along with the reading interval to calculate identified leakage.

C. Monitor for increases in noble gas activity on Drywell Atmospheric Monitoring System Recorder D23-R600 to identify gross changes in unidentified leakage.

D. Record unidentified leakage rate in gpm directly from a PDS computer point if the point is operable.

Answer: A

Question Comments: Answers B, C, and D are all procedural methods for monitoring drywell leakage described in the operator Tech Spec Rounds for determining leakage., and that makes them incorrect answers, since the question is asking what is NOT an approved method. Answer A is correct because the entire volume of the sump is NOT used when determining the leakage rate. 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-E3100 Objective: 5.4; 10
2. CourseID: GQC-CRO01 Section 2 Item 4 Objective: Item 4

KA References:

1. 268000 K1.04 Reactor building floor drains: Plant-Specific [2.7/2.9]

References:

1. 06-OP-1000-D-0001 Att I Data Sheet II Items 25, 13
2. Tech Spec Bases B SR3.4.5.1

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: MikeRasch at Mon Jun 20 14:01:47 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

Comments:

EB QUESTION: 65 **(1.0 Points)**

Excessive leakage from which one of the following Fuel Pool Cooling and Clean-up (FPCC) components would cause leak detection alarm FPCC FLTR DMIN SYS LEAK to alarm on 1H13-P680 due to high input into the FPCC leak detection standpipe?

Drawings M-1088D; 1090A; 1090B; 1098A; 1098B

A. FPCC Pump 'A'

B. Spent Fuel Pool liner

C. FPCC Heat Exchanger 'B'

D. FPCC Backwash Receiving Tank Transfer Pump

Answer: D

Question Comments: Answer A is incorrect because FPCC Pump "A" drains to Floor Drain Sump. Answer B is incorrect because Spent Fuel Pool liner drains to Floor Drain Sump. Answer C is incorrect because FPCC Heat Exchanger "B" drains to Equipment Drain Sump (G41 side) and Chem Waste Sump (P42 side). Answer D is correct because FPCC Backwash Receiving Tank Transfer Pump area drains go to the standpipe. Tier 2 Group 2 This is a NEW question. 10 CFR 41.4

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00897

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-OPS-E3100 Objective: 6.12

KA References:

1. 290001 K4.03 Fluid leakage collection [2.8/2.9]

References:

1. M1090B
2. M1098B
3. M1088D
4. M1098A

TrainingPrograms:

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

Systems:

1. E31: Leak Detection System
2. G41: Fuel Pool Cooling and Cleanup System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Mon Jun 13 12:01:57 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

Comments:

EB QUESTION: 66 (1.0 Points)

The plant was at 95% power when Reactor Feed Pump ‘B’ tripped, followed by a runback of both Recirc Flow Control Valves.

Reactor power is now swinging 72% to 74% on APRMs, with Total Core Flow now 56 mlbm/hr.

Given the following indications on H13-P680, which one would continued operation of the unit be allowed?

05-1-02-III-3 is provided.

A. APRM oscillations of 14% peak-to-peak

B. Annunciators PBDS ‘A’ INOP (5A-A6) and PBDS ‘B’ INOP (7A-A6) alarm 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

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: MikeRasch at Mon Jun 20 10:18:27 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

Comments:

EB QUESTION: 67 (1.0 Points)

The plant is in an ATWS with a steam leak in the Drywell.

Plant conditions are as follows:

Reactor power 7%

Reactor water level -172 inches and is being controlled in band.

Reactor pressure 530 psig and is being controlled in band.

Main Condenser is NOT available.

Drywell Pressure 9 psig, rising slowly

Drywell temperature 225°F, rising slowly

Containment pressure 7 psig, rising slowly

Containment temperature 125°F, rising slowly

Suppression Pool temperature 145°F, rising slowly

Suppression Pool level 24.6 feet, rising slowly

Standby Liquid Control systems have failed.

Which one of the following is the appropriate action for the given conditions?

A. Lower suppression pool level to <18.81 feet using RCIC or HPCS.

B. Lower reactor pressure, irrespective of resulting cool down rates, to maintain operation in the Safe Zone of the Heat Capacity Temperature Limit (HCTL) curve to avoid emergency depressurization.

C. Initiate RHR 'A' and 'B' in Containment Spray mode. If unable to maintain containment pressure in the Safe Zone or the 'N/A' regions of the Pressure Suppression Pressure curve, then perform emergency depressurization in accordance with EP-2A, including termination of injection.

D. Immediately perform emergency depressurization in accordance with EP-2A, including termination of injection.

Answer: D

Question Comments: Answer A is INCORRECT because HPCS and RCIC Test Return Lines to the CST are isolated due to an Auxiliary Building Isolation Signal. Answer B is INCORRECT because HCTL is not challenged and Emergency Depressurization is required per Containment/Suppression Pool Water level > 24.4 feet and no method available to lower level.

Answer C is INCORRECT because Emergency Depressurization is not an option it is required. Answer D is CORRECT because EP – 3 Step 52 cannot be answered YES so Emergency Depressurization is required and RPV Injection Termination is required per EP - 2A. Tier 3 This is a NEW question. 10CFR 41.7/41.9/41.10/43.5

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: GG-1-LP-RO-EP03 Objective: 3; 6
2. CourseID: GG-1-LP-RO-EP02A Objective: 7

KA References:

1. GENERIC 2.1.25 Ability to obtain and interpret station reference materials such as graphs / [2.8/3.1]

References:

1. EP-3 step 53
2. 04-1-01-E51-1 step 6.3.1c
3. 04-1-01-E22-1 Note at Step 6.3.2a
4. 05-1-02-III-5 for P11 isolations
5. 04-1-01-P11-2 section 5.8

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
2. E51: Reactor Core Isolation Cooling System
3. P11: Condensate and Refueling Water Transfer System

Categories:

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

4. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 10:23:16 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

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 Yellow plastic seals are used on Fire Protection Valves. Blue Tie-Wraps

Comments: 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

Comments:

EB QUESTION: 69 (1.0 Points)

Which one of the following situations presents an operability concern regarding Intermediate Range Monitors?

Data Sheets concerning IRMs from 06-OP-1000-D-0001; 03-1-01-1 and 03-1-01-3 are provided.

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 2×10^4 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

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 Jun 20 10:27:49 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

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.

For which one of the following activities would notification by control room personnel to Refueling Floor supervision be required?

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: MikeRasch at Mon Jun 20 10:29:40 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

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 Comments: 50% of the maximum radiation dose rate of a High Radiation Area of 999 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

Comments:

EB QUESTION: 72 (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.

A search and rescue team is being assembled to locate and extract the mechanic, who is believed to be gravely injured.

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?

10-S-01-17 is provided.

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 dose limits based on the situation. Further dose limit 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 to 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 3 This is a NEW question. 10CFR 41.10/41.12/43.4/43.5

Image Reference: None

Open Reference Question

Handout 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 6.1

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 Mon Jun 20 10:31:56 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

Comments:

EB QUESTION: 73 (1.0 Points)

Which one of the following describes the basis of Caution 1 of the Emergency Procedures?

A. The RPVST curve is based on de-gassing in the instrument runs.

B.

Caution 1 is only based on environmental temperature effects to the reference legs of level instruments, since variable legs would be unaffected.

C. The Condition 2 table of Indicated Level and Reference Leg Temperature protects against using an instrument for indication with actual level below the variable leg tap.

D. The question “Can RPV water level be determined?” on EP-2/2A must be answered‘NO’anytime operation is plotted in the Possible Boiling Region of the RPV Saturation Temperature –RPVST curve.

Answer: C

Question Comments: Caution 1 of the Emergency Procedures concerns the reliability of the RPV Level Instrumentation. It is divided into 2 sections. Section 1 concerns reference leg area temperatures vs RPV pressure and the possibility of reaching saturation conditions for the liquids inside the reference legs which possibly would cause erroneous readings. Section 2 concerns lower indicated levels with elevated reference leg temperatures that may cause actual level to be below the taps of the level instrument and still read on scale. Operations Philosophy allows instruments to be considered good if specifically Answer A is INCORRECT because the RPVST curve is based on saturation conditions for the Reference legs. Answer B is INCORRECT because Caution 1 is not just the reference legs it also includes variable legs. Answer C is a CORRECT statement concerning Section 2 of Caution 1. Answer D is INCORRECT because the conditions may be in the Unsafe region of Figure 2 but the Instrument not be showing the effects and still be reading true that is the reason is states POSSIBLE BOILING and Operations Philosophy 02-S-01-27 allows instruments to be used even with plant conditions in the Unsafe (Possible Boiling) region. 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: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GG-1-LP-RO-EP02 Objective: 11
2. CourseID: GLP-OPS-PROC Objective: 59.3

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 1
2. PSTG B-5-2 thru 10

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 Mon Jun 20 10:35:55 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

Comments:

EB QUESTION: 74 **(1.0 Points)**

The plant is at 100% power when a fire occurs at Main Transformer ‘C’ due to an oil leak.

Which one of the following describes the fire suppression for Main Transformer ‘C’?

A. A Dry Pipe system with fused-closed heads will automatically actuate fire water deluge when the fuse melts at a predetermined temperature.

B. Fire water deluge is only available by manually initiation and realignment of manual valves.

C. Fire water is the primary means of fire suppression, with carbon dioxide as the backup system.

D. Rate of rise heat detectors energize a solenoid valve on the deluge valve to automatically actuate fire water deluge.

Answer: D

Question Comments: The fire protection system for the Main Transformers is an Automatic Deluge System which operates off rate of rise heat detectors which energize a solenoid valve to open the Deluge Valve and admit fire protection water to open sprinkler heads surrounding the transformer. Answer A is INCORRECT because these systems are not classified as a Dry Pipe system. Answer B is INCORRECT because this describes a fire protection system similar to the Charcoal Filter trains. Transformer Automatic Deluge Systems require no operator action to actuate. Answer C is INCORRECT because even though the Transformer is electrical equipment there is NO CO2 fire suppression systems to back the water system. Answer D is CORRECT because this describes the operation of an Automatic Deluge Fire Suppression System. Tier 3 This is a NEW question. 10 CFR 41.4/41.8

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00905

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

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. 04-S-01-P64-1 Att VI
2. M-0035B and J

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. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 10:38:38 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

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

Comments:

END OF EXAM

Total Number of Questions: 75 Total Point Value: 75.0

Total Exam Question Difficulty: 120.0

Average Exam Question Difficulty: 1.6

Questions with Difficulty Level 1: 30

Questions with Difficulty Level 2: 45

Go to Action Menu. Go to Home Menu. Exit.

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: MikeRasch **Review Date:** Mon Jun 20 14:44:19 CDT 2005
Exam Review Status: Reviewed
Approved By: Mickey Ellis **Approval Date:** Mon Jun 20 14:46:49 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

Comments:

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

EB QUESTION: 1 (1.0 Points)

The plant was operating at 96% power when the "B" Recirculation pump tripped.

Plant conditions include the following:

Reactor power	72% of rated
Core flow	68 Mlbm/hr
Feedwater temperature	415° F.
FCBB	1.34

PBDS is fully operable.

Which one of the following describes the actions to be taken for the present situation?

05-1-02-III-3 Reduction in Recriculation System Flowrate ONEP is provided.

A. Immediately place the reactor mode switch in the SHUTDOWN position.

B. Reduce the Recirculation loop flow to less than 44,600 and monitor for thermal hydraulic instability.

C. Initiate action to reduce core flow to exit the Power/ Flow graph Monitored region then verify FCBB is ≤ 1.0 within 15 minutes.

D. Immediately initiate action to reduce core flow to exit the Power/ Flow graph Restricted region then verify FCBB is ≤ 1.0 within 15 minutes.

Answer: B

Question Comments: The proper action is based on plotting within the Monitored region of the Power/Flow graph. (Figure 1 from 05-1-02-III-3) Answer A is incorrect because this action is for the Exclusion region. Answer B is CORRECT per step 3.5.1 of 05-1-02-III-3. Answer C is incorrect because this action is for the Monitored region with a concurrent loss of Feedwater heating. Answer D is incorrect because this action is for the Restricted region. This is a NEW question. Tier 1 Group 1 CFR 41.5/41.10/43.5

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00833

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-OPS-ONEP Objective: 2.0
2. CourseID: GLP-OPS-ONEP Objective: 25.0

KA References:

1. 295001 AA2.01 Power/flow map [3.5/3.8]
2. 295001 AK1.02 Power/flow distribution [3.3/3.5]
3. GENERIC 2.4.4 Ability to recognize abnormal indications for system operating parameters [4.0/4.3]

4. 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 Tue May 31 14:40:03 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

Comments:

EB QUESTION: 2 (1.0 Points)

The GGNS Electrical line up is normal.

A large tree limb falling across a transmission line causes degraded 500 KV voltage.

The voltage to ALL ESF busses DROPS to 3120 volts for 5 seconds and then returns to normal.

Which one of the following statements describes the condition of the plant ESF busses 30 seconds after this voltage transient?

A.
15AA is being supplied from ESF 11

16AB is being supplied from ESF 21

17AC is being supplied from Div III D/G

B.
15AA is being supplied from Div I D/G

16AB is being supplied from Div II D/G

17AC is being supplied from Div III D/G

C.
15AA is being supplied from ESF 11

16AB is being supplied from ESF 21

17AC is being supplied from ESF 21

D.
15AA is being supplied from Div I D/G

16AB is being supplied from Div II D/G

17AC is being supplied from ESF 21

Answer: A

Question Comments: Answer A is correct because the voltage did not drop low enough for long enough for a Div I/II LSS BUV shed. [90%(3744) for 9 seconds or 70% (2912) for 0.5 seconds] However the Div III bus undervoltage relay trips the incoming feeder breaker at 73% (3045 volts) with no time delay and the Div III diesel generator will start and tie to the bus. Answer B is incorrect because LSS will not shed the DivI/II busses. Answer C is incorrect because the Div III bus undervoltage relay trips the incoming

feeder breaker at 73% (3045 volts) with no time delay and the Div III diesel generator will start and tie to the bus. Answer D is incorrect because the voltage did not drop low enough for long enough for a Div I/II LSS BUV shed. [90%(3744) for 9 seconds or 70%(2912) for 0.5 seconds], therefore the Div I/II diesel generators will not start and tie to their respective busses. However the Div III bus undervoltage relay trips the incoming feeder breaker at 73% (3045 volts) with no time delay and the Div III diesel generator will start and tie to the bus. TIER 1 GROUP 1 This is a MODIFIED question. NRC Exam 3/1998 ID# WRI 11 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

Objectives:

1. CourseID: GLP-OPS-R2100 Objective: 12.0
2. CourseID: GLP-OPS-R2100 Objective: 20.0

KA References:

1. 295003 AK3.01 Manual and auto bus transfer [3.3/3.5]
2. 295003 AK3.03 Load shedding [3.5/3.6]
3. 295003 AK1.03 Under voltage/degraded voltage effects on electricalloads [2.9/3.2]
4. 295003 AK1.04 Electrical bus divisional separation [3.1/3.2]

References:

1. Tech Spec TR3.3.8.1-1
2. 04-1-01-R21-1 section 5.1.1
3. 04-1-01-P81-1 section 3.22

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 Mon Jun 20 07:34:08 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

Comments:

EB QUESTION: 3 **(1.0 Points)**

Which one of the following describes the interrelationship between the Control Power and the Standby Service Water System?

A. The 125 VDC System provides electrical power for SSW control and indication circuits.

The 120 VAC System provides electrical power for SSW initiation logic.

Loss of 125 VDC would NOT affect the initiation logic however, the SSW pumps and cooling tower fans CANNOT automatically start without DC control power.

B. The 125 VDC System provides electrical power for SSW initiation, control, and indication circuits.

Loss of 125 VDC would cause the initiation logic to initiate.

The SSW pumps and cooling tower fans CANNOT automatically start without DC

control power.

C. The 125 VDC System provides electrical power for SSW initiation, control, and indication circuits.

Loss of 125 VDC would cause the initiation logic to initiate.

The SSW pumps and cooling tower fans would automatically start and the system would align to supply cooling water to its plant loads.

D. The 120 VAC System provides electrical power for SSW control and indication circuits.

The 125 VDC System provides electrical power for SSW initiation logic.

Loss of 125 VDC would cause the initiation logic to initiate however, the SSW pumps and cooling tower fans CANNOT automatically start without AC control power.

Answer: B

Question Comments: Answer A is incorrect because the 120 VAC system does not supply electrical power to the SSW initiation logic. Answer B is correct because the 125 VDC System provides electrical power for SSW initiation, control, and indication circuits. Answer C is incorrect because the SSW pumps and cooling tower fans would not automatically start without DC control power. Answer D is incorrect because the 120 VAC System does not provide electrical power for SSW control and indication circuits. 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: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-OPS-P4100 Objective: 12.6
2. CourseID: GLP-OPS-I1100 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-1225

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
2. P41: Standby Service Water System

Categories:

1. Systems

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 07:29:14 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

Comments:

EB QUESTION: 4 (1.0 Points)

Based on EP-4 conditions, the "Operator at the Controls" has been directed to rapidly depressurize the reactor using the main turbine bypass valves .

The Condensate and Feedwater systems are in automatic with a 36 inch level setpoint and will respond to the event per design.

How should this be accomplished and how will these actions effect reactor water level?

A. The operator should immediately fully open the bypass valves and expect indicated narrow range reactor water level to drop and return to normal only after the ECCS initiates to augment the Condensate and Feedwater systems.

B. The operator should gradually open the bypass valves to full open and expect indicated narrow range reactor water level to rise, then drop, then return to normal only after the ECCS initiates to augment the Condensate and Feedwater systems.

C. The operator should immediately fully open the bypass valves and expect indicated narrow range reactor water level to rise, then drop, then return to normal as the Condensate and Feedwater systems respond.

D. The operator should gradually open the bypass valves to full open and expect indicated narrow range reactor water level to remain constant as the Condensate and Feedwater systems respond.

Answer: C

Question Comments: Answer A is incorrect because, reactor water level will initially rise, then drop. With no power being generated from the reactor, no ECCS initiation is expected. Answer B is incorrect because the operator should immediately fully open the bypass valves. With no power being generated from the reactor, no ECCS initiation is expected. Answer C is correct because the operator should immediately fully open the bypass valves and expect indicated reactor water to rise, then drop, then return to normal as the Condensate and Feedwater systems respond. Answer D is incorrect because the operator should immediately fully open the bypass valves and expect indicated reactor water level to rise, then drop, then return to normal as the Condensate and Feedwater systems respond. This is a NEW question. TIER 1 GROUP 1 CFR 41.3/41.5/41.7/41.10

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

TrainingPrograms:

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

Systems:

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

Categories:

1. Emergency Procedure Training
2. Fundamentals
3. Systems

Task References:

Question Last Revised By: MikeRasch at Tue May 10 13:58:26 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

Comments:

EB QUESTION: 5 (1.0 Points)

In which one of the following "Scram Reports" can the SRO determine that Shutdown Margin 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, bypass valves are available, and Feedwater is controlling reactor water level.

B. The Mode Switch is in Shutdown, APRMs indicate downscale, NO control rod position indication is available, bypass valves are available, and Feedwater is controlling reactor water level.

C. The Mode Switch is in Shutdown, reactor power is dropping, all control rods indicate full in except forty nine control rods indicate position 02, bypass valves are unavailable, and Feedwater is recoverable.

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, bypass valves are unavailable, and Feedwater is recoverable.

Answer: C

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 incorrect because even though the APRMs indicate downscale, no control rod position indication is available. Answer C is CORRECT because all control rods are inserted to at least position 02 and by analysis, this ensures adequate shutdown margin. 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
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 Tue May 24 10:14:09 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

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 one of the following describes the minimum actions necessary to scram the reactor from outside the Control Room?

A. Secure both CRD pumps using handswitches on the Remote Shutdown panels.

B. Open breakers CB2A and CB8A at panel 1C71-P001 ("A" RPS MG Set room).

C. Open breaker CB2B and CB8B at panel 1C71-P002 ("B" RPS MG Set room).

D. Open breakers CB2A or CB8A at panel 1C71-P001 ("A" RPS MG Set room) and CB2B or CB8B at panel 1C71-P002 ("B" RPS MG Set room).

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: 1: Fundamental Knowledge or Memory

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: MikeRasch at Tue May 24 10:30:20 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

Comments:

EB QUESTION: 7 (1.0 Points)

The plant was operating at 100% power when the following alarms were observed.

Annunciators in alarm:

RECIRC PMP/MTR A/B TEMP HI

CTMT AREA FLR DRN SUMP LEAK HI

CCW SURGE TK LVL HI/LO

RWCU FLTR DMIN INL TEMP HI 130º F

CCW PMP A/C DISCH PRESS LO

Which one of the following is the appropriate response to this condition?

05-1-02-V-1, Loss of Component Cooling Water ONEP is provided.

A. Monitor Reactor Recirculation pump temperatures on 1B33-R601.

B. Verify the standby CCW pump starts. Isolate CCW to FPCCU. Isolate CCW to RWCU.

C. Reduce core flow to 67 Mlbm/hr. Refer to ONEP 05-1-02-III-3, Reduction in Reactor Recirculation Flow.

D. Manually scram the reactor. Manually trip the Reactor Recirculation pumps. Isolate CCW to the Containment.

Answer: D

Question Comments: Answer D is CORRECT per Step 3.3 of ONEP 05-1-02-V-1, Loss of Component Cooling Water. Answer A is incorrect because even though the Reactor Recirculation pumps are still running, this action is not enough because of the line break in Containment. Answer B is incorrect because even though this is a partial loss of CCW, this action is not enough because of the line break in Containment. Answer C is incorrect because even though the Reactor Recirculation pump temperatures are rising, this action is not enough because of the line break in Containment.

This is a NEW question. TIER 1 GROUP 1 CFR 41.4/41.5/41.6/41.7/41.10/43.5

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00838

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-OPS-ONEP Objective: 2.0
2. CourseID: GLP-OPS-B3300 Objective: 36.7
3. CourseID: GLP-OPS-B3300 Objective: 36.9
4. CourseID: GLP-OPS-B3300 Objective: 42.0
5. CourseID: GLP-OPS-B3300 Objective: 40.0
6. CourseID: GLP-OPS-G3336 Objective: 9.7
7. CourseID: GLP-OPS-G3336 Objective: 9.9
8. CourseID: GLP-OPS-G3336 Objective: 12.0
9. CourseID: GLP-OPS-G3336 Objective: 14.0
10. CourseID: GLP-OPS-P4200 Objective: 11.1
11. CourseID: GLP-OPS-P4200 Objective: 11.2
12. CourseID: GLP-OPS-P4200 Objective: 14.0
13. CourseID: GLP-OPS-P4200 Objective: 16.0

KA References:

1. 295018 AK1.01 Effects on component/system operations [3.5/3.6]
2. 295018 AK2.02 Plant operations [3.4/3.6]
3. 295018 AA2.03 Cause for partial or complete loss [3.2/3.5]

References:

1. 05-1-02-V-1 Step 3.3
2. 04-1-02-1H13-P870-5A-C2
3. 04-1-02-1H13-P870-5A-D2
4. 04-1-02-1H13-P601-22A-C2
5. 04-1-02-1H13-P680-3A-A8
6. 04-1-02-1H13-P680-11A-D6
7. 04-1-02-1H13-P870-9A-D2

TrainingPrograms:

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

Systems:

1. B33: Reactor Recirculation System
2. E31: Leak Detection 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 May 24 08:02:55 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

Comments:

EB QUESTION: 8 (1.0 Points)

The plant was operating at rated conditions when a leak in the Auxiliary Building Instrument Air header caused the header pressure to drop to 28 psig.

A manual scram was inserted.

NO LOCA isolations have occurred.

The Primary and Secondary Containment air operated isolation valves closed due to low pressure, and a bleed off valve in the Auxiliary Building automatically opened depressurizing the Auxiliary Building air header.

The Auxiliary Building Instrument Air header air leak was repaired.

Which one of the following describes the actions that will occur when Auxiliary Building Instrument Air header pressure is restored without operator action?

Upon restoration of header pressure, :

A. all of the Primary and Secondary Containment air operated isolation valves will

automatically re-open and the bleed off valve will have to be manually reclosed.

B. all of the Primary and Secondary Containment air operated isolation valves will require manual re-opening and the bleed off valve will have to be manually reclosed.

C. some of the Primary and Secondary Containment air operated isolation valves will require manual re-opening and the bleed off valve will automatically reclose.

D. some of the Primary and Secondary Containment air operated isolation valves will automatically re-open and the bleed off valve automatically re-close.

Answer: C

Question Comments: Upon Instrument air system pressure dropping < 60 psig Hiller actuators have a solenoid valve that isolate the valve. This will require operator actions to give the valve an open signal to return the valve to the open position. P53-F026A, F026B, and F001 are all hiller acutators. Once Instrument Air is restored to Secondary Containment the Instrument Air Bleed-off station P53-F531/F523 on 166' elevation will automatically reset itself by the Restricting Orifice (P53-RO-D026) creating a backpressure to close P53-F531. Answer A is incorrect because it has the isolation valves automatically re-opening and having to manually close the bleed-off station. Answer B is incorrect because it has the bleed-off station being manually reclosed. Answer C is CORRECT Primary and Secondary Containment isolation valves have to be manually realigned and the bleed-off station automatically reclosing. Answer D is incorrect because Primary and Secondary Containment valves do NOT automatically re-open. TIER 1 GROUP 1 This is a NEW question. CFR 41.4/41.5/41.7/41.9

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00548

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-OPS-P5300 Objective: 13.0
2. CourseID: GLP-OPS-P5300 Objective: 14.0

KA References:

1. 295019 AA2.01 Instrument air system pressure [3.5/3.6]
2. 295019 AA2.02 Status of safety-related instrument air system loads(see AK2 [3.6/3.7]
3. 295019 AK2.14 Plant air systems [3.2/3.2]
4. 295019 AK2.09 Containment [3.3/3.3]

References:

1. M-1067M
2. 05-1-02-V-9 Note after step 3.6
3. GG FSAR Table 9.3-1
4. GG FSAR Table 9.3-2
5. M-1067E

TrainingPrograms:

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

Systems:

1. P53: Instrument Air System

Categories:

1. Systems

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 07:45:31 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

Comments:

EB QUESTION: 9 (1.0 Points)

The plant was in Mode 4 during a forced outage when all forced circulation was 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: MikeRasch at Thu Jun 09 16:39:33 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

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. High in-plant dose rates

D. Inadequate decay heat removal

Answer: C

Question Comments: Answer C 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 D 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

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 Mon Jun 20 07:47:42 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

Comments:

EB QUESTION: 11 (1.0 Points)

The plant is at 100% power.

I&C is trouble shooting failure of Remote Shutdown reactor water level transmitter C61-N400B.

While isolating C61-N400B, a pressure spike on the reference leg of condensing pot B21-D004B affecting Wide Range Reactor Water Level transmitters B21-N091B and B21-N091F causes a spurious Low Reactor Water Level, Level 1, ECCS initiation.

NO other level or pressure transmitters common to that reference leg are affected.

The low water level initiation signal was only a spike and is now clear.

What is the sequence of operator actions necessary to prevent a reactor scram as a result of the ECCS initiation?

Selected sections of 17-S-06-5 and M-1077B are provided.

A. Restore Instrument Air and Plant Service Water to the Auxiliary Building and Containment; then, in order to shut down Drywell Purge Compressor A, depress LPCS/RHR A INIT RESET pushbutton on H13-P601, depress DIV 1 LSS PNL RESET on H13-P864, place CGCS DIV 1 MAN INIT RESET to RESET, then stop Drywell Purge Compressor A, E61-C001A, using its hand switch on H13-P870.

B. Restore Instrument Air and Plant Service Water to the Auxiliary Building and Containment; then, in order to shut down Drywell Purge Compressor B, depress RHR B/C INIT RESET pushbutton on H13-P601, depress DIV 2 LSS PNL RESET on H13-P864, place CGCS DIV 2 MAN INIT RESET to RESET, then stop Drywell Purge Compressor B, E61-C001B, using its hand switch on H13-P870.

C.

Depress LPCS/RHR A INIT RESET pushbutton on H13-P601, depress DIV 1 LSS PNL RESET on H13-P864, place CGCS DIV 1 MAN INIT RESET to RESET, then stop Drywell Purge Compressor A, E61-C001A, using its hand switch on H13-P870.

D. Depress RHR B/C INIT RESET pushbutton on H13-P601, depress DIV 2 LSS PNL RESET on H13-P864, place CGCS DIV 2 MAN INIT RESET to RESET, then stop Drywell Purge Compressor B, E61-C001B, using its hand switch on H13-P870.

Answer: D

Question Comments: Answer A is incorrect because the isolation signal for the listed valves originates from B21-N082A-D, not N091's, and B21-N091B&F initiate Division 2 ECCS, not Division 1 as listed. Answer B is incorrect because the isolation signal for the listed valves originates from B21-N082A-D, not N091's. Answer C is incorrect because B21-N091B&F initiate Division 2 ECCS, not Division 1 as listed. Answer D is correct since ECCS logic is designed such that a Division 2 ECCS initiation will occur if reactor water level signals from B21-N091B&F go below -150.3". 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

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 Mon Jun 20 13:33:34 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

Comments:

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

A Group 1 isolation resulted in a reactor scram.

RCIC was started manually and is injecting into the reactor with the controller in manual.

Reactor pressure is rising.

Which one of the following describes the RCIC system response to rising reactor pressure?

A. RCIC speed will drop with injection rate remaining stable.

B. RCIC speed will rise causing the injection rate to rise.

C. RCIC speed will rise slightly with injection rate remaining stable.

D. RCIC speed will rise slightly but injection rate will drop slightly.

Answer: D

Question Comments: As Reactor pressure rises speed will rise (about 50 rpm) because the governor valve is in a fixed position. This rise in speed is not sufficient to overcome approximately 100 psi change in Reactor pressure resulting in a slightly lower rate of injection. When an SRV opens to relieve pressure injection flow turns and rises. Answer A is incorrect because RCIC speed must increase, not drop due to a higher pressure supplied to the RCIC turbine. Answer B is incorrect because RCIC speed will rise as reactor pressure increases due to the governor valve being in a fixed position. This rise in speed will not be sufficient to overcome the rise in reactor pressure causing injection rate to drop. Answer C is incorrect because pump output will rise as speed of the turbine rises. Answer D is correct because a slight increase in RCIC speed will result from the higher reactor pressure however this rise in speed is not sufficient to overcome the change in reactor pressure causing injection rate to drop slightly. TIER 1 GROUP 1 This is a MODIFIED question. NRC ID WRI818 February 2004 CFR 41.4/41.5/41.14

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-E5100 Objective: 4.11
2. CourseID: GLP-OPS-E5100 Objective: 8.17

KA References:

1. 295025 EK3.05 RCIC operation: Plant-Specific [3.6/3.7]
2. 295025 EK2.07 RCIC: Plant-Specific [3.7/3.7]

References:

1. Vendor Manual 460000182
2. Simulator Response

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. Systems

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 12:49:23 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

Comments:

EB QUESTION: 13 (1.0 Points)

The plant is in an ATWS following a total loss of EHC due to large break in EHC piping.

Reactor power is 7%.

Reactor pressure is being controlled 800 psig to 1060 psig using SRVs and Main Steam Line Drains.

Reactor level is being controlled -167" to -192" using Reactor Feed Pump 'A'.

Reactor Feed Pump 'B' is available.

Suppression Pool Cooling 'B' is in service.

Suppression Pool Cooling 'A' is unavailable.

Suppression Pool level is 19.0 feet.

Suppression Pool temperature is 155°F and slowly rising.

Emergency Procedure Attachments to enable control rod insertion are NOT expected to be installed for at least 30 minutes.

Standby Liquid Control systems have failed.

Which one of the following is the basis for the requirement to lower Reactor Pressure using SRVs for this condition?

A. To raise Suppression Pool level which will raise its capacity as a heat sink by driving conditions away from HCTL.

B. To get more voiding which will lower power and slow rate of the Suppression Pool temperature rise.

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

D. Lower reactor pressure to within Condensate Booster Pump discharge pressure capability in case both Reactor Feed Pumps are lost.

Answer: C

Question Comments:

Image Reference: None

Open Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00840

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

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

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: Charles Bell at Fri Jun 17 14:31:55 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

Comments:

EB QUESTION: 14 (1.0 Points)

The plant is in an ATWS. An RWCU piping leak has occurred on the inlet to the non-regenerative heat exchangers. RWCU isolation valves have lost power and have failed to isolate.

Reactor power is 3%.

Reactor pressure is being controlled 450 psig to 600 psig using Main Bypass Valves.

Reactor level is being controlled -150 inches to -192 inches using Condensate Booster Pumps.

Primary Containment pressure is 2.0 psig.

Primary Containment temperature is 187°F.

Suppression Pool temperature is 125°F.

Suppression Pool level is 18.8 feet.

Both loops of Suppression Pool Cooling are maximized.

Standby Liquid Control systems have failed.

Which one of the following describes actions to control Primary Containment temperature and pressure?

A. Conduct an Emergency Depressurization of the RPV.

B. Vent Primary Containment using Emergency Procedure Attachment 14.

C. Operate all containment coolers using Emergency Procedure Attachment 7 to bypass all containment cooler isolation interlocks.

D. Initiate both loops of Containment Spray. Then, if Primary Containment temperature does NOT lower below 185°F, conduct an Emergency Depressurization of the RPV.

Answer: A

Question Comments: Answer A 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, and Containment Spray is not allowed due to being in the unsafe region of the CSIPL curve (EP-3 step 6). Answer B is incorrect since the vent path for EP attachment 14 is isolated and cannot be opened under the listed conditions. Answer C is incorrect since operation of containment coolers without P71 as a heat sink would be ineffective. Answer D is incorrect since Containment Spray is not allowed due to being in the unsafe region of the CSIPL curve (EP-3 step 6). 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-00844

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

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

TrainingPrograms:

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

Systems:

Categories:

1. Emergency Procedure Training

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 13:35:23 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

Comments:

EB QUESTION: 15 (1.0 Points)

A LOCA has occurred in the Drywell.

A reactor scram has occurred, and all control rods fully inserted.

All low pressure ECCS systems are available. HPCS is out of service.

RCIC, SLC, and CRD are injecting at rated flow.

Reactor pressure is 500 psig and slowly falling.

Wide Range reactor water level indication is -150 inches and slowly falling.

Post Accident Monitoring Fuel Zone level indication is -180 inches and slowly falling.

Compensated Fuel Zone level indication is -160 inches and slowly falling.

Average Drywell temperature is 190°F.

Drywell temperature at elevation 166 ft. is 200°F.

Average Containment temperature is 135°F.

Containment temperature at elevation 139 ft. is 150°F.

Which one of the following describes Reactor Water Level indication?

A. Emergency Depressurization should be conducted when Wide Range level trends offscale low.

B. Emergency Depressurization is required now by the Alternate Level Control leg

under these conditions.

C. Emergency Depressurization should be conducted when Compensated Fuel Zone level reaches -192 inches.

D. RPV Flooding should be entered now under these conditions.

Answer: B

Question Comments: Answer A is incorrect since EP Caution 1 prohibits use of wide range level indication when containment temperature at elev. 139' is >143°F. Answer B is correct since wide range level indication cannot be used and fuel zone level is < TAF (-167"). Compensated fuel zone is to be used for values of reactor level only during ATWS conditions per 02-S-01-27. Answer C is incorrect since compensated fuel zone is to be used for values of reactor level only during ATWS conditions per 02-S-01-27. Answer D is incorrect since reactor level can be determined by using fuel zone indication. 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

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 steps 53 - 71
2. 05-S-01-EP-2 Caution 1 section 2

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: MikeRasch at Thu Jun 16 12:51:16 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

Comments:

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

The plant is at 100% power.

High Pressure Core Spray system was operating on minimum flow with suction from the Suppression Pool for testing when a large break on the High Pressure Core Spray suction line upstream of HPCS Suction from Suppression Pool valve E22-F015 occurred.

HPCS pump was secured, but E22-F015 lost power when operators attempted to close it.

Annunciators HPCS RM SUMP LVL HI-HI (1H13-P680-8A1-B4) and HPCS ROOM FLOODED (1H13-P870-5A-H1) are in alarm.

The HPCS room water tight door has failed.

Suppression pool level is 15 feet and slowly falling.

Suppression Pool temperature is 80°F.

Emergency Procedures 3 and 4 have been entered.

Which one of the following describes further actions in response to the Suppression Pool leak?

A. Low Suppression Pool level would have caused HPCS suction to automatically align to the CST if E22-F015 had NOT lost power.

B. If Suppression Pool level continues to fall to 14.56 feet, Emergency Depressurization will be required due to inability to maintain the Heat Capacity Temperature Limit (HCTL) curve in the Safe Zone with the Reactor pressurized.

C. If one more Maximum Safe water level is reached in Secondary containment due to the Suppression Pool leak, Emergency Depressurization will be required by EP-4.

D. If Suppression Pool level continues to fall to 14.56 feet, Emergency Depressurization will be required to avert inadequate submergence of the horizontal vents in the Drywell wall with the Reactor pressurized.

Answer: D

Question Comments: Answer A is incorrect since HPCS suction has no automatic feature associated with low suppression pool level. Answer B is incorrect since suppression pool temperature would have to reach >140°F for the HCTL curve to require emergency depressurization with the reactor at rated pressure. Answer C is incorrect since the high area water level is not due to a leak from the reactor pressure boundary. Answer D is correct due to EP-3 step 42 requirements and basis. Tier 1 Group 1 This is a NEW question. 10 CFR 41.9/41.10/43.5

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00846

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

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

KA References:

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

References:

1. PSTG B 7-30
2. 05-S-01-EP-3

TrainingPrograms:

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

Systems:

Categories:

1. Emergency Procedure Training

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 13:37:44 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

Comments:

EB QUESTION: 17 (1.0 Points)

The plant was operating at 100% power when a small leak developed in the Drywell due to failure of both reactor head O-rings.

Drywell pressure is 1.5 psig and rising slowly.

All systems responded as designed.

The lowest reactor water level reached was -20 inches Wide Range.

Operators stopped High Pressure Core Spray (HPCS) pump using its Control Room hand switch to attempt to maintain reactor water level below Level 8.

Reactor water level is now 60" on Wide Range instruments due to swell.

Reactor pressure is stable at 960 psig being controlled automatically by bypass valves.

Which one of the following describes further operation of HPCS?

A. If HPCS INIT RESET push button is depressed on 1H13-P601, HPCS pump will immediately restart, but HPCS Injection Valve E22-F004 will remain closed.

B. HPCS Injection Valve E22-F004 can be opened by depressing and holding HPCS HI LVL RESET push button on 1H13-P601 and simultaneously holding the hand switch for E22-F004 to OPEN.

C. HPCS pump will automatically start and HPCS Injection Valve E22-F004 will automatically open with NO further operator action if reactor water level drops to Level 2.

D. HPCS pump will automatically start and HPCS Injection Valve E22-F004 will automatically open if reactor water level drops to Level 2 only if HPCS INIT RESET push button has been depressed on 1H13-P601.

Answer: D

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” 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”. Answer C is incorrect because once manually overridden with an initiation signal

sealed in, HPCS pump would have to be started manually with its hand switch or the initiation reset to break the manual override seal-in. Answer D is correct because depressing HPCS INIT RESET would reset the initiation signal and override high drywell pressure and break the manual override seal-in. 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

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 Mon Jun 20 13:38:53 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

Comments:

EB QUESTION: 18 (1.0 Points)

The plant had been operating at 100% power when a small tear occurred on the High Pressure Condenser boot seal.

A manual scram was directed due to degrading condenser vacuum, however, all control rods failed to fully insert.

All other equipment operated as designed.

EP-2A was entered.

Plant conditions are as follows:

Main Condenser vacuum is 11.5 in. Hg and falling very slowly.

Reactor power is 15 %.

Reactor water level is being maintained -70 inches to -130 inches using Condensate Booster Pumps.

Reactor pressure is being controlled 450 psig to 600 psig using SRVs.

MSIVs are open.

Main Bypass Stop Valves are closed.

Which one of the following describes required actions to be taken for RPV pressure control under these conditions?

A.

Attachment 8 should be installed.

B. Lowering Pressure Reference to 900 psig will cause reactor pressure to fall to less than the lowest Low-Low Set Valve reset setpoint.

C. The Main Condenser is NO longer "available" per EP-2A.

D. Main steam line drains should NOT be used since they discharge to the High Pressure Condenser shell.

Answer: A

Question Comments: Answer A is correct since power is >4%, the MSIVs are open (hence the main condenser is available), steam loads such as RFPTs are in service, and preserving the main condenser as a heat sink given the challenge to MSIVs from lowering reactor water level is desirable. This minimizes the heat input to containment and prolongs availability of feedwater. Therefore, EP-2A step 40 directs installation of Att. 8 to defeat the low water level closure of MSIVs. Answer B is incorrect because lowering P-ref will have no effect on pressure since turbine and bypass stop valves are all closed below 12"Hg vacuum. Answer C is incorrect because PSTGs conclude availability to reject heat to the condenser exists when MSIVs are open. Answer D is incorrect because discharging steam to the main condenser limits the challenge to containment. The MSIV isolation at 9"Hg vacuum would provide ample protection for the condenser to prevent over-pressurization and a possible radioactive release. Tier 1 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-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: MikeRasch at Mon Jun 20 13:40:07 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

Comments:

EB QUESTION: 19 (1.0 Points)

The plant was in Mode 4.

Standby Service Water (SSW) ‘A’ was in service supplying Fuel Pool Cooling and Clean-up (FPCC) heat exchanger ‘A’.

SSW ‘A’ basin cooling tower fans were temporarily secured due to low SSW ‘A’

temperature.

A small tube leak in the FPCC ‘A’ Heat Exchanger developed.

Drift from SSW ‘A’ basin was blowing westward and forming a puddle inside of the protected area fence.

Eventually, runoff from the puddle flowed under/through the protected area fence and to the storm drain in front of the Unit 2 Warehouse.

When would this be first considered to be a liquid effluent release regard to the Technical Requirements Manual?

TRM 6.11.1 and Definitions are provided.

A. When the effluent began to puddle inside the protected area fence.

B. When the effluent crossed the protected area fence.

C. When the effluent entered the storm drain in front of the warehouse.

D. When the effluent entered a navigable 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

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00851

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

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

KA References:

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

References:

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: MikeRasch at Mon Jun 20 13:42:54 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

Comments:

EB QUESTION: 20 (1.0 Points)

Which one of the following operator actions is NOT required in relation to a fire inside the protected area?

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

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

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

D. If the Main Control Room is evacuated due to a fire in the Main Control Room, always place Transfer Switch for Lockout Transfer Relay C61-HSS-M150 at 1H22-P152 to ON.

Answer: C

Question Comments: Answer A is incorrect because it is required by 04-1-01-P81-1 step 3.14. 04-1-01-P81-1 step 3.14 Answer B is incorrect because it is required by 05-1-02-II-1 step 3.4.1. Answer C is correct because the control room is not an area listed in 10-S-03-2 step 6.2.2f requiring Z17-B002 secured, since that does not directly serve the control room. Answer B is incorrect because it is required by 05-1-02-II-1 step 3.5.1 for a fire in the control room severe enough to result in control room evacuation. 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: MikeRasch at Fri Jun 10 08:33:06 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

Comments:

EB QUESTION: 21 (1.0 Points)

The plant is operating at 20% power with Main Generator output 250 MWe.

Circulating Pump ‘A’just tripped due to an electrical fault in the motor windings.

Circulating Water pump ’B’is in standby.

Main condenser vacuum is now 25 in. Hg vacuum.

Which one of the following sequence of events will occur as a result of degrading Main

Condenser vacuum if vacuum continually lowers at a rate of 1 in. Hg vacuum every 5 minutes?

A. Reactor scram, Main Turbine trip, Reactor Feed Pump trip, Main Bypass Stop Valve closure, MSIV closure

B. Main Turbine trip, Reactor Feed Pump trip, Reactor scram, Main Bypass Stop Valve closure, MSIV closure

C. Main Turbine trip, Reactor Feed Pump trip, Main Bypass Stop Valve closure, Reactor scram, MSIV closure

D. Main Turbine trip, Reactor Feed Pump trip, Main Bypass Stop Valve closure, MSIV closure, Reactor scram

Answer: B

Question Comments: Answer A is incorrect because there is no automatic scram directly from condenser vacuum. Answer B is correct because the turbine will trip at 21”Hg vacuum. The reactor scram from TSV/TCV closure is bypassed below 40% power, so no scram will occur from EHC fluid pressures. And bypass valves can accommodate steam flow for 35% power and are fast acting, so the effect on reactor pressure will be minimal, and pressure will stay below 1064.7 psig, the scram setpoint. Stable pressure will minimize shrink, so reactor level will remain relatively stable above the 11.4” scram setpoint. Also, neutron flux will not spike above the scram setpoint of 118% since reactor pressure will remain stable. RFPTs will trip at 16”Hg vacuum, causing inventory to steam off with only CRD for makeup. CRD will supply <1% rated makeup flow. With power 20%, steam flow is about 6600gpm, so the low level scram setpoint of 11.4” will be reached within less than 1 minute, calculating a level reduction rate of about 30 in/min, assuming 200 gal/in reactor level. This is much sooner than when bypass valves will close given the vacuum loss rate of 0.2”Hg vac/min. Then, Bypass Stop Valves will close at 12”Hg vacuum, and MSIVs will close at 9”Hg vacuum. Answer C is incorrect because the reactor will scram on low water level well before the bypass valve closure setpoint of 12”Hg vacuum is reached. 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 Mon Jun 20 13:43:36 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

Comments:

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

Under which one of the following conditions is installation of bottled gas to supply the Automatic Depressurization System (ADS) air receivers required?

A. Mode 3, following a scram due to loss of bus 15AA. SRVs are being used for reactor pressure control. ADS accumulators ‘A’ and ‘B’ pressures indicated on 1H13-P601 are 150 psig. The time estimated to restore bus 15AA is 8 hours.

B. An ATWS is in progress. The Main Condenser is available. Reactor power is 8%. The Auxiliary building is isolated due to low reactor water level and has not been restored.

C. An instrument air header rupture in the water treatment building has occurred. Repairs are expected to take 20 minutes. SRVs are being used for reactor pressure control. ADS accumulators ‘A’ and ‘B’ pressures are 165 psig and 163 psig, respectively.

D. Mode 4 during Operations Hydro (03-1-01-6) when one ADS Air Receiver is inoperable due to system maintenance.

Answer: A

Question Comments: 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. Answer B is incorrect because the instrument air isolation can be bypassed and restored under given conditions, so installing bottled gas is unnecessary. Answer C is incorrect because the stated expected out of service time for the normal air supply to SRV is less than the 6 hour limit in the referenced plant procedures. Answer D is incorrect because ADS is not required operable during Ops Hydro, only the relief function of 2 SRVs is required functional, which it would be with one ADS Air Receiver remaining operable. 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 Fri Jun 10 09:04:56 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

Comments:

EB QUESTION: 23 (1.0 Points)

The plant was at 99% power.

RHR 'A' pump is tagged out of service to replace the pump breaker protective relays.

A disturbance in the switchyard caused a loss of Service Transformers 11 and 21.

Three control rods remained at position 48 following the reactor scram.

Reactor power is 0% on APRMs.

MSIVs are closed.

Reactor level is being controlled +30 inches to -30 inches using RCIC.

Reactor pressure is being controlled 800 psig to 1060 psig using SRVs.

All ECCS has been initiated and overridden.

Average Suppression temperature is 95°F.

Instrument air and service air are unavailable.

Air header pressure in the Auxiliary Building is 45 psig.

Both ADS Air Receiver pressures are 160 psig.

RHR B TEST RETURN TO SUPP POOL VLV E12-F024B failed to open when placing Suppression Pool Cooling 'B' in service.

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

A. It is acceptable to allow Low-Low Set to cycle indefinitely under these conditions, as long as localized Suppression Pool temperatures remain below 185°F.

B. If there is only one leaking/weeping SRVs, 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 to allow for even heating of the Suppression Pool.

D. Only ADS SRVs should be used, and they should be rotated to allow for even heating of the Suppression Pool.

Answer: D

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 incorrect because ADS valves are specifically preferred when instrument air is unavailable since they will function longer. Answer D is correct because ADS valves are specifically preferred when instrument air is unavailable. Rotation to equalize pool heating and reduce the chance of exceeding containment design temperature limits is specified. 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: 1: Fundamental Knowledge or Memory

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 Mon Jun 20 13:45:28 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

Comments:

EB QUESTION: 24 (1.0 Points)

The plant was in power ascension during Mode 2.

RCIC was being warmed with the RCIC Low Reactor Pressure (60 psig) isolation defeated per IOI-1.

RCIC STM SPLY DRWL OTBD ISOL VLV E51-F064 and RCIC STM SPLY DRWL INBD ISOL VLV E51-F063 had been opened.

Reactor pressure had reached 62 psig.

A large leak developed from a crack in a weld on RCIC steam supply piping upstream of RCIC STM SPLY TO RCIC TURBINE valve E51-F045.

An operator has attempted to close E51-F063 and E51-F064, but their supply breakers have tripped before the valves reached full closed.

Reactor pressure has fallen to 58 psig.

Which one of the following actions is required for these conditions?

A. Immediately restore the RCIC Low Reactor Pressure isolation in accordance with IOI-1.

B. If RCIC Equipment Area Temperature reaches 185°F, enter EP-2 and place the Reactor Mode Switch in Shutdown.

C. If RCIC Equipment Area Temperature reaches 212°F, enter EP-2 and place the Reactor Mode Switch in Shutdown.

D. If RCIC Equipment Area Temperature reaches 212°F, enter EP-2 and conduct Emergency Depressurization.

Answer: C

Question Comments: Answer A is incorrect because the RCIC Low Reactor Pressure isolation is not required operable <150 psig reactor pressure, and restoring it would do nothing to mitigate the event. 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 E51F063&64 not fully closed, 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). Answer D is incorrect since only one area is affected, and temperature would have to exceed maximum safe levels in at least 2 areas to require emergency depressurization per EP-4 step 16. Tier 1 Group 2 This is a NEW question. 10 CFR 41.5/41.10/43.5

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00857

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis
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

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: MikeRasch at Mon Jun 20 13:46:09 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

Comments:

EB QUESTION: 25 (1.0 Points)

The plant is in Mode 5.

Fuel Handling Area Pool Sweep Exhaust Radiation Monitor 'C', D17K618C, has been removed by I&C for replacement.

Annunciator FP EXH DIV 2,3 RAD HI-HI/INOP (1H13-P601-19A-C10) is sealed in.

A fuel bundle was dropped inside the Reactor causing damage to fuel pins.

As a result of the event, the following radiation levels were reached, where they have now stabilized:

Containment/Drywell Vent Exhaust Radiation Monitors

D17K609A = 3.8 mr/hr

D17K609B = 4.0 mr/hr

D17K609C = 3.0 mr/hr

D17K609D = 4.0 mr/hr

Fuel Handling Area Vent Exhaust Radiation Monitors

D17K617A = 2.0 mr/hr

D17K617B = 4.0 mr/hr

D17K617C = 3.0 mr/hr

D17K617D = 4.0 mr/hr

Fuel Handling Area Pool Sweep Exhaust Radiation Monitors

D17K618A = 32 mr/hr

D17K618B = 20 mr/hr

D17K618C = removed for replacement/inoperable

D17K618D = 22 mr/hr

Which one of the following describes the response of Standby Gas Treatment System (SGTS) to the event?

A. 'A' and 'B' will remain in standby.

B.

'A' will automatically start.

C. 'B' will automatically start.

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: MikeRasch at Mon Jun 20 08:07:53 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

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.

Which one of the following is the implication of this alarm?

A. Greater personnel 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

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: MikeRasch at Fri Jun 10 09:40:21 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

Comments:

EB QUESTION: 27 (1.0 Points)

Plant is in Mode 3.

RHR ‘A’ has just been placed in Shutdown Cooling mode.

A large (2000 gpm) leak from the Standby Service Water ‘A’ supply piping to the RHR ‘A’ heat exchangers occurs in the RHR ’A’ heat exchanger room.

Motor Control Center 11B12 is de-energized for connecting temporary power.

RHR Room ‘A’ Floor Drain Sump Pump ‘A’, P45C013A, is powered from breaker 52-111209.

RHR Room ‘A’ Floor Drain Sump Pump ‘B’, P45C013B, is powered from breaker 52-115104.

The current hand switch configuration for RHR 'A' Room Floor Drain Sump Pumps is:

P45C013A RHR Rm 'A' Flr Drn Smp Pmp 'A' HS-M020A AUTO

P45C013B RHR Rm 'A' Flr Drn Smp Pmp 'B' HS-M021A AUTO

P45C013A/B RHR Rm 'A' Flr Drn Smp Pmps 'A/B' Mode Switch HSS-M019A ALTERNATE

RHR Room A Floor Drain Sump Pump "A" was the last pump to run in this hand switch alignment.

Which one of the following describes operation as a result of these conditions?

A. RHR Rm 'A' Flr Drn Smp Pmp 'A', C013A, would start on High sump level. RHR Rm 'A' Flr Drn Smp Pmp 'B', C013B, would start on Hi-Hi sump level. Both pumps would continue to run. Alarms RHR RM A SMP LVL HI-HI (1H13-P680-8A1-A2) and RHR A RM FLOODED (1H13-P870-2A-E1) would be received in the control room.

B. RHR Rm 'A' Flr Drn Smp Pmp 'B', C013B, would start on Hi-Hi sump level and would continue to run. Only alarm RHR RM A SMP LVL HI-HI (1H13-P680-8A1-A2) would be received in the control room.

C. RHR Rm 'A' Flr Drn Smp Pmp 'B', C013B, would start on High sump level and would continue to run. Alarms RHR RM A SMP LVL HI-HI (1H13-P680-8A1-A2) and RHR A RM FLOODED (1H13-P870-2A-E1) would be received in the control room.

D. RHR Rm 'A' Flr Drn Smp Pmp 'A' and 'B' will remain off. Alarms RHR RM A SMP LVL HI-HI (1H13-P680-8A1-A2) and RHR A RM FLOODED (1H13-P870-2A-E1) would be received in the control room.

Answer: C

Question Comments: Answer A is incorrect because C013A has no power with 11B12 de-energized, and in ALTERNATE, C013B will start on high level since C013A was the last to run. Answer B is incorrect because pump B would have started on a High sump level as indicated in answer C you would also receive the Room Flooded alarm due to the capacity of the sump pump. Answer C is correct because C013B would not be able to pump

2000 gpm, so sumps would back up to the point where the room flooded alarm would come in (6” above the floor) also. Answer D is incorrect because C013B would start on high level, as previously described. 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 Mon Jun 20 08:13:30 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

Comments:

EB QUESTION: 28 (1.0 Points)

A small break LOCA occurred in the Drywell at 100% power.

Current plant conditions are:

Reactor power	0%
Reactor pressure	950 psig, stable
Reactor water level	-50 inches Wide Range, slowly falling
Drywell pressure	1.9 psig
Drywell Temperature	170°F
Containment pressure	0.7 psig
Containment Temperature	95°F

If the handswitch for RHR Pump ‘A’ on 1H13-P601 was momentarily placed in STOP, which one of the following describes RHR ‘A’ operation under these conditions?

A. If power supplying bus 15AA is lost and then restored, RHR ‘A’ pump will start and remain running.

B. If power supplying bus 15AA is lost and then restored, RHR ‘A’ pump will start but will immediately stop again.

C. If RHR ‘A’ Logic Power supply breaker 72-11A38 is opened, RHR ‘A’ pump will start and remain running.

D. If Wide Range reactor water level falls to -155 inches, RHR ‘A’ pump will start and remain running.

Answer: A

Question Comments: Answer A is correct because AC power from bus 15AA is monitored by relay e12-K3A. When power is lost, a contact on K3A opens which drops out the seal in for the initiation logic and the manual override. The pump handswitch is spring return to auto. If power from bus 15AA was lost, the seal in relay would de-energize, and when power was restored, the initiation logic would again trip due to drywell pressure being above 1.39 psig, and the pump would start. Answer B is incorrect because with the pump handswitch in auto after stop, the pump would start when power was restored, and no trip signal would exist to stop the pump. Answer C is incorrect because the DC logic to automatically start the pump is energize to trip. With no DC power, relays required to fire to start the pump would remain de-energized. Answer D is correct because with the initiation signal and the pump manual override sealed in, automatic start is prevented. The initiation logic is already sealed in, so subsequent initiation signals, though from a different parameter, have no additional effect. 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

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 Fri Jun 10 09:53:23 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

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’ mode, returning to the reactor via E12-F053B.

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

Which one of the following would cause RHR ‘B’pump to trip?

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

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: MikeRasch at Mon Jun 20 08:15:20 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

Comments:

EB QUESTION: 30 (1.0 Points)

A plant start up had been in progress at 32% power.

When Recirc pump 'A' was shifted to fast speed, a LOCA in the Drywell occurred.

Division 2 and 3 ECCS initiated properly due to high drywell pressure.

Division 1 ECCS failed to initiate due to loss of RHR 'A' Logic Power.

The following indications are present in the control room:

Reactor power 0%

Reactor pressure 800 psig

Drywell pressure 3 psig

Reactor water level 25 inches Wide Range

Annunciators in on 1H13-P601:

RHR A SYS OOSVC (20A-H6)

DRWL PRESS HI (21A-E7)

LPCS SYS ACTUATED (21A-B8)

LPCS SYS OOSVC (21A-H8)

Status light RHR A LOGIC PWR FAIL STATUS is on (1H13-P601-20B)

The green and amber lights on the Low Pressure Core Spray (LPCS) handswitch on P601 are on and the red light is off.

Which of the following describes availability of LPCS under these conditions?

A. LPCS pump CANNOT be started using any hand switch. LPCS Injection Valve E21F005 will NOT open automatically and CANNOT be opened by placing its control room hand switch on P601 to OPEN.

B. LPCS pump can be started by placing the control room hand switch to START. LPCS Injection Valve E21-F005 will open automatically or by placing its control room hand switch on P601 to OPEN when its pressure permissive is met.

C. LPCS pump can only be started by the local pistol grip hand switch on the front of LPCS Pump breaker 152-1506. LPCS Injection Valve E21-F005 can only be

opened using the local valve manual hand wheel.

D. LPCS pump and LPCS Injection Valve E21-F005 can be operated remotely only from the Division 1 Remote Shutdown Panel, 1H22-P150 (Area 25A, el. 111)

Answer: B

Question Comments: Answer A is incorrect because Logic power and control power to manually start LPCS pump and to automatically open E21F005 are unaffected. In the LPCS logic, all that is affected is a series contact from LSS in the auto start for LPCS pump does not close, since LSS is fired by an RHR ‘A’ relay. Answer B is correct because Logic power and control power to manually start LPCS pump and to automatically open E21F005 are unaffected. The P601 hand switch contact in the LPCS pump start circuit is in parallel with the LSS contact that is affected by the power loss. Answer C is incorrect because The P601 hand switch contact in the LPCS pump start circuit is in parallel with the LSS contact that is affected by the power loss. So the P601 handswitch will start LPCS pump. Answer D is incorrect because both LPCS pump and E21F005 can be operated from P601 as previously described. 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

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 Fri Jun 10 12:40:43 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

Comments:

EB QUESTION: 31 (1.0 Points)

Which of the following is NOT a method of altering Suppression Pool level using High Pressure Core Spray (HPCS) system?

Assume any associated prerequisites are met.

A. To raise Suppression Pool level with HPCS pump secured, gravity drain from the Condensate Storage Tank (CST) by aligning HPCS suction flow path from the CST and throttling open HPCS TEST RTN TO SUPP POOL, E22-F023.

B. To raise Suppression Pool level with HPCS pump running, align HPCS suction flow path from the CST, and close HPCS TEST RTN valves to CST, E22-F010 and E22-F011. Then ensure HPCS MIN FLOW valve E22-F012 is open.

C. To raise Suppression Pool level with HPCS pump running, check/align HPCS suction flow path from the CST, and throttle open HPCS TEST RTN TO SUPP POOL, E22-F023.

D. To lower Suppression Pool level, align HPCS suction flow path from the Suppression Pool, start HPCS pump, and ensure HPCS/RCIC Test Return to CST valves P11-F064 and P11-F065 are open. Then hold open hand switches for HPCS Test Return to CST valves E22-F010 and E22-F011 until the valves are full open, observe the valves automatically stroke closed, and repeat cycling them open, as necessary.

Answer: C

Question Comments: Answer A is incorrect because this is a method listed in 04-1-01-E22-1 step 6.4.2a. Answer B is incorrect because this is a method listed in 04-1-01-E22-1 step 6.4.2b. Answer C is correct because this is not an approved method of raising suppression pool level because it would result in too high of a flow rate. Answer D is incorrect because this is a method listed in 04-1-01-E22-1 step 6.3. 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-00864

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

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 Fri Jun 10 12:40:11 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

Comments:

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

Following a loss of Feedwater from 50% power, High Pressure Core Spray (HPCS) automatically initiated on low reactor water level and restored level.

HPCS Injection Valve, E22-F004, automatically closed as designed as level rose to Level 8.

Which one of the following describes the basis for closure of E22-F004 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 due to high reactor water level, Level 9.

Answer: C

Question Comments: Answer A is incorrect because this is not the reason listed in Tech Spec bases. In this case, SRVs would prevent over-pressurization. Answer B is incorrect because this is not the reason listed in Tech Spec bases. 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 would not prevent reaching level 9 in all cases due the time required for E22F004 to stroke closed. Tier 2 Group 1 This is a NEW question. 10CFR 41.8

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00865

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

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 Fri Jun 10 10:16:45 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

Comments:

EB QUESTION: 33 (1.0 Points)

Conditions exist that required Standby Liquid Control injection.

SLC ‘A’ has been initiated from 1H13-P601.

Reactor pressure is stable at 950 psig.

The following indications associated with SLC ‘A’ are present:

SLC A/B Discharge Pressure on P601 indicates 1700 psig.

SLC Storage Tank Level on P601 indicates 4800 gal.

SQUIB VLV READY C41-F004A white light is on.

SLC ‘A’ STOR TK OUTL VLV C41-F001A green light is off, red light is on.

SLC TEST TK OUTL VLV C41-F031 green light is on, red light is off.

SLC PUMP A green and amber lights are off, red light is on.

RWCU PMP SUCT CTMT OTBD ISOL valve is closed.

Squib Valve C41-F004A milliamp meter C41M600A in 1H13-P632 is pegged high.

Which one of the following conditions for SLC ‘A’ would cause these indications?

A. SLC ‘A’ is operating properly and injecting into the reactor at rated flow.

B. SLC ‘A’ discharge piping is ruptured.

C. SLC ‘A’ Squib Valve C41-F004A actuated and is open, but electrical leads to the valve have electrically shorted to one another.

D. SLC ‘A’ Squib Valve C41-F004A failed to actuate.

Answer: D

Question Comments: Answer A is incorrect because the squib valve continuity light is on, and discharge pressure is ~ 400 psig too high. Answer B is incorrect because discharge pressure is ~ 400 psig higher than normal. If the pipe was ruptured, discharge pressure would be lower than normal. Answer C is incorrect because usually if the leads short, the control power fuse will blow and the continuity light will then extinguish. Also, discharge pressure is ~ 400 psig too high. Answer D is correct because the squib valve continuity light is on, and discharge pressure is ~ 400 psig too high, indicating very low flow. 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-00866

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

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. 06-OP-1C41-R-0002 Note at Step 5.1.23

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. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 08:18:48 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

Comments:

EB QUESTION: 34 **(1.0 Points)**

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

The ‘A’ Main Bypass Control Valve (BCV) has failed open, so startup has been suspended to troubleshoot the bypass valve.

The Baxter Wilson 500 KV line to the GGNS switchyard is lost causing breakers J5232 and J5228 to trip.

The Turbine Initial Pressure Control (IPC) system responds as designed, except BCV ‘A’ remains failed open.

How would the plant respond automatically for this event?

A. Bypass Control Valves ‘B’ and ‘C’ would open, Reactor Recirc Pumps would transfer to slow speed, and the reactor would scram on high water level.

B. Bypass Control Valves ‘B’ and ‘C’ would open, Reactor Recirc Pumps would transfer to slow speed and the reactor would scram due to TCV closure.

C. Bypass Control Valves ‘B’ and ‘C’ would open, Reactor Recirc Pumps would remain in fast speed, but the reactor would scram due to high neutron flux.

D. Bypass Control Valves ‘B’ and ‘C’ would open, Reactor Recirc Pumps would remain in fast speed, and the reactor would remain operating.

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 at 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

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: MikeRasch at Fri Jun 10 10:30:09 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

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

Comments:

EB QUESTION: 36 (1.0 Points)

Which one of the following sets of conditions will result in a control rod block during startup?

A.
SRM A is upscale and its joystick on 1H13-P680 is in BYPASS

SRM E is partially withdrawn and is reading 70 cps

IRM A is on Range 10 and its joystick on 1H13-P680 is in BYPASS

IRM E is on range 2

B.
SRM A is upscale and its joystick on 1H13-P680 is in BYPASS

SRM E is fully withdrawn and reads 200 cps

IRM A is on range 3

IRM E is on range 3

C.
SRM A is INOP and its joystick on 1H13-P680 is in BYPASS

SRM E is partially withdrawn and reads 200 cps

IRM A is on range 2

IRM E is on range 3

D.
SRM A is INOP and its joystick on 1H13-P680 is in BYPASS

SRM E is INOP

IRM A is on range 9

IRM E is on range 9

Answer: A

Question Comments: Answer A is correct because for SRM A a rod block will occur if it is upscale, even if bypassed, and SRM E is not full in and reads less than 100 counts. Answers B and C are incorrect because SRM E reads >100

counts and SRM A is bypassed. Answer D is incorrect because both IRMs associated with SRMs A and E are on range 9, thus bypassing the SRM rod block. 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

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

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: MikeRasch at Mon Jun 20 08:24:17 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

Comments:

EB QUESTION: 37 (1.0 Points)

The plant is at 100% power.

Which one of the following describes the affect to Average Power Range Monitors (APRMs) if the feeder to inverter panel 1Y86 were to trip open?

04-1-01-L62-1 Attachment V Table 1 is provided.

A. APRMs would be unaffected.

B. Rod block, only, due to APRMs D and H failing downscale due to loss of power. No half scram would occur since power is lost to Division 2 RPS sensors and logic, thus preventing them from tripping.

C. APRMs D and H would continue to indicate power accurately, but an inop signal would be generated due to loss of the LPRM count circuit, resulting in a Division 2 half scram and a rod block.

D. APRMs D and H would lose power. A Division 2 half scram and a rod block would be generated.

Answer: D

Question Comments: Answer A is wrong because 1Y86 feeds APRMs D and H. Answer B is wrong because RPS is de-energize to trip, so loss of sensor or logic power causes a half scram. Also, an APRM inop trip would occur, also producing a half scram. Answer C is incorrect because APRM indication

would fail downscale. Answer D is correct because 1Y86 supplies power to APRMs D and H, they would fail downscale, and Division 2 RPS trip units would lose power, causing some of them to be in their tripped state, resulting in a half scram as well. Tier 2 Group 1 This is a NEW question. 10 CFR 41.2/41.6/41.7

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00870

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

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
2. E-1173-14; 19
3. E-1172-05
4. 04-1-01-L62-1 Att V Table 1

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 Fri Jun 10 13:40:49 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

Comments:

EB QUESTION: 38 (1.0 Points)

The plant was at 100% when a LOCA occurred in the drywell.

RCIC has automatically initiated.

The following indications are present:

Reactor Power 0%

Reactor Pressure 950 psig, stable

Reactor Water Level -50 inches wide range, slowly falling

RCIC Flow Controller AUTO/ 15% output, stable

RCIC Turbine Steam Supply Pressure 950 psig, stable

RCIC Turbine Speed 2560 rpm, stable

RCIC Turbine Exhaust Pressure 3.9 psig, stable

RCIC Pump Suction Pressure -3.1 in Hg, stable

RCIC Pump Discharge Pressure 195 psig, stable

RCIC Pump Flow 810 gpm, stable

These conditions are indicative of which one of the following?

A. RCIC speed controller failure

B. Feed water line 'A' break in the Drywell

C. Feed water line 'B' break in the Drywell

D. RCIC is operating properly and injecting the design flow to the reactor

Answer: C

Question Comments: The key parameter here is a very low RCIC discharge pressure and flow controller output. Answer A is incorrect because the flow controller output is responding properly to the flow feedback signal and controlling ~800 gpm, its setpoint. Answer B is incorrect because RCIC injects into the 'B' feedwater line, which is segregated from the 'A' line by a check valve. Therefore a feedwater line 'A' break would not affect RCIC parameters. Answer C is correct because RCIC parameters indicate RCIC is pumping the correct flow rate against a head that is much lower than reactor pressure. If 'B' feedwater line was ruptured, RCIC would be pumping to the drywell, which is near atmospheric pressure. Answer D is incorrect because the discharge pressure would have to be above reactor pressure for water to go to the reactor 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 Mon Jun 20 08:26:05 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

Comments:

EB QUESTION: 39 (1.0 Points)

The plant was at 100% power when a Loss of Offsite Power coincident with a LOCA in the drywell occurred.

All systems initiated properly except DC breaker 72-11A23, which supplies Automatic Depressurization System (ADS) logic power and Division 1 SRV solenoid power, has tripped.

Drywell pressure is 7 psig.

Reactor water level is -100 inches wide range, falling slowly.

Reactor pressure is 800 psig, stable.

Which one of the following describes the effect these conditions have on ADS functionality?

Drawing E-1161-004 is provided.

A. ADS ‘A’ has initiated due to loss of logic power. ADS valves B21-F051A and B21-F051B are open since they have alternate power through the Division 1 Remote Shutdown Panel.

B. ADS ‘A’ has initiated due to loss of logic power. 8 ADS valves are open due to availability of Division 2 solenoid power.

C. ADS ‘A’ will NOT initiate automatically, but ADS valves will automatically open due to Division 2 power.

D. ADS ‘A’ will NOT initiate automatically, but 8 ADS valves can be opened manually using their 1H13-P601 hand switches.

Answer: C

Question Comments: Answers A and B are incorrect because ADS logic is “energize to initiate”. Answer C is correct because ADS logic is “energize to initiate”, so ADS ‘A’ cannot initiate. Division 2 ADS is unaffected, so Div 2 solenoids will open the ADS valves. Answer D is incorrect because P601 handswitches are for the Division 1 solenoids, which have lost power. Tier 2 Group 1 This is a NEW question. 10CFR 41.7

Image Reference: None

Open Reference Question

Handout 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]

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 Fri Jun 10 11:34:17 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

Comments:

EB QUESTION: 40 (1.0 Points)

The plant was operating at 100% power.

A large break LOCA caused Reactor Water Level to drop to -70 inches Wide Range.

RCIC is restoring RPV level and is -60 inches Wide Range.

An operator just reported that there is a large leak on the Drywell Chilled Water Return

piping inside containment.

You observe the Isolation Status Board and notice that NO isolation valves have closed.

What is the minimum required action regarding the isolation failure?

A. Manually isolate one valve in every penetration which should have isolated.

B. Isolate P53 to the Auxiliary Building to cause a complete isolation. All valves affected by the failure to isolate will fail closed.

C. Manually isolate one valve in each penetration which should have isolated, except Instrument Air (P53) and Plant Service Water (P44) do NOT need to be isolated if they will be immediately reopened.

D. Manually isolate one valve in each penetration which should have isolated, except Drywell Chilled Water (P72), Plant Service Water (P44), and Instrument Air (P53) do NOT need to be isolated if they will be immediately reopened.

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: 1: Fundamental Knowledge or Memory
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

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 Mon Jun 20 08:27:42 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

Comments:

EB QUESTION: 41 **(1.0 Points)**

Emergency Depressurization of the reactor had to be performed.

Eight ADS/SRVs (Automatic Depressurization System / Safety Relief Valves) were opened using their respective hand switches on H13-P601.

Reactor pressure lowered to zero psig.

NO other manipulations involving SRVs have been performed.

ADS logic is in standby.

NO failures have occurred in the ADS/SRV system.

Both ADS accumulator pressures are 160 psig.

What can the oncoming reactor operator now ascertain regarding the status of the ADS valves from H13-P601, H13-P631 (Main Control Room Back Panels), and H13-P628 (Upper Control Room)?

A. On section 19C (the apron section) of H13-P601, the red light above each ADS valve hand switch will be illuminated and green light will be off, indicating the respective ADS valve is open.

B. The ADS Logic A/E continuity lights on H13-P628 will be off, meaning power is applied to each ADS valve

C. The red ADS SRV Status light on section 19B (the vertical section) of H13-P601 will be illuminated, indicating power is applied to each ADS valve

D. All green lights will be illuminated and all red lights will be off for each ADS valve on H13-P601, H13-P631, and H13-P628. All continuity lights on H13-P631 and H13-P628 will be illuminated. ADS valve position CANNOT be positively determined from these indications.

Answer: D

Question Comments: Answer A is incorrect because the SRV tailpipe pressures are <30psig, the pressure switch setpoint feeding the P601 handswitch lights. Answer B is incorrect because ADS logic is in standby. Answer C is incorrect

because Div 2 solenoids have not been energized. Answer D is correct because all tailpipe pressure switches are reset due to depressurization, there is no indication for Div 1 solenoids which are energized, and logic systems are in standby, so all continuity lights are on. 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: 2: Comprehension or Analysis

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

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 08:29:28 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

Comments:

EB QUESTION: 42 **(1.0 Points)**

The plant is at 90% power.

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

Reactor level is 43 inches narrow range and slowly trending up.

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 operator response for this condition?

A. Immediately insert a manual scram.

B. 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.

C. 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.

D. Take manual control of the Master Level Controller and control level 32 inches to 42 inches.

Answer: B

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 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 correct because ONEP 05-1-02-V-6 specifically directs this action for the given failure. Answer C is incorrect because RFP ‘B’ controller is responding properly to the failure of ‘A’. Answer D is incorrect because the master level controller controller is responding properly to the failure of RFP ‘A’ speed controller. 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

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-6Step 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: MikeRasch at Mon Jun 20 08:31:07 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

Comments:

EB QUESTION: 43 (1.0 Points)

The plant is in Mode 1.

The equalizing valve for Main Steam Line Flow transmitter C34-N030A has begun to leak by the seat.

The steam flow signals inputting to the Digital Feedwater Control System (DFCS) are:

C34-N030A - 2.8 mlbm/hr

C34-N030B - 4.1 mlbm/hr

C34-N030C - 4.1 mlbm/hr

C34-N030D - 4.2 mlbm/hr

The Total Steam Flow estimated by DFCS for the current power level is 16.2 mlbm/hr.

Which one of the following describes the response of actual reactor water level and the Digital Feed Control System (DFCS)?

A. Reactor water level will be slightly lower than before and will remain stable. DFCS will remain in three element control.

B. Reactor water level will remain stable when the level dominance of the DFCS takes over and automatically substitutes calculated steam flow for the failed value.

C. Reactor water level will immediately rise then return to normal level when the DFCS system automatically de-selects and locks out three element control and

selects single element control.

D. Reactor water level will immediately fall then return to normal level when the DFCS system automatically de-selects and locks out three element control and selects single element control.

Answer: A

Question Comments: Answer A is correct because the difference between the highest and lowest stem flow signals does not exceed 1.6 mlbm/hr, so 3-element control would not be automatically disabled. Answer B is incorrect because there is no substitution for failed steam flow signals used at GGNS. Answer C is incorrect because level would initial fall as sensed steam flow lowered, and the difference between the highest and lowest stem flow signals does not exceed 1.6 mlbm/hr, so 3-element control would not be automatically disabled. Answer D is incorrect because the difference between the highest and lowest stem flow signals does not exceed 1.6 mlbm/hr, so 3-element control would not be automatically disabled. Tier 2 Group 1 This is a MODIFIED QUESTION. Last used August 2002 question id WRI705. 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: 10.6
2. CourseID: GLP-OPS-C3400 Objective: 10.7

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: MikeRasch at Mon Jun 13 08:03:40 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

Comments:

EB QUESTION: 44 (1.0 Points)

Standby Gas Treatment System (SGTS) 'A' was manually initiated 10 minutes ago for a planned move of radioactive materials on the refueling floor and is running properly.

Enclosure Building differential pressure is now -0.6 inches wc.

SGTS 'B' is to be manually initiated, too.

Which one of the following depicts how SGTS 'B' flow control damper T48-F005 (Steam Tunnel Outside Containment) and flow control vane T48-F500B should respond if operating properly?

A.

T48-F005 goes to intermediate position.

T48-F500B stays open until either -0.75 inches wc is reached or 120 seconds elapses, then it modulates.

B. T48-F005 goes to intermediate position.

T48-F500B stays open until either -0.75 inches wc is reached or 90 seconds elapses, then it modulates.

C. T48-F005 goes full open, and after 120 seconds throttles to intermediate position.

T48-F500B stays open until either -0.75 inches wc is reached or 90 seconds elapses, then it modulates.

D. T48-F005 goes full open, and after 90 seconds opens to intermediate position.

T48-F500B stays open until either -0.75 inches wc is reached or 120 seconds elapses, then it modulates.

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, and the proper pressure and timer values are stated for T48F500B. Answer B is incorrect because the wrong timer value for T48F500B is given. Answer C is incorrect because T48F005 would go to intermediate position right away instead of 90 sec later, and the wrong timer value for T48F500B is given. Answer D is incorrect because T48F005 would go to intermediate position right away instead of 90 sec later. 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: 1: Fundamental Knowledge or Memory
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: MikeRasch at Mon Jun 13 07:54:52 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

Comments:

EB QUESTION: 45 **(1.0 Points)**

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

High Pressure Core Spray (HPCS) pump is running on minimum flow with suction from the Suppression Pool.

Reactor level is normal.

Which one of the following describes operation of HPCS with bus 11DC de-energized?

A. HPCS Pump breaker 152-1702 can be tripped using its control room hand switch on 1H13-P601.

HPCS Injection Valve E22-F004 CANNOT be opened from the control room.

B. HPCS Pump breaker 152-1702 can be tripped by taking its local pistol grip hand switch on the front of breaker 152-1702 to OPEN, but it CANNOT be re-closed.

HPCS Injection Valve E22-F004 CANNOT be opened from the control room.

C. HPCS Pump breaker 152-1702 will automatically trip if windings in HPCS pump motor fault.

HPCS Injection Valve E22-F004 can be opened from the control room.

D. HPCS Pump breaker 152-1702 CANNOT be tripped with a hand switch and will NOT trip automatically.

HPCS Injection Valve E22-F004 can be opened 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 operates on and is controlled by 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 Mon Jun 13 09:08:22 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

Comments:

EB QUESTION: 46 (1.0 Points)

Static Inverter 1Y95 was on ALTERNATE supply for planned maintenance when a LOCA in the drywell occurred.

Division 1 and Division 2 ECCS initiated.

Fifteen minutes later, Bus 16AB locked out.

The following alarms are present on H13-P807:

STATIC INVERTER 1Y79 TROUBLE (3A-G4)

STATIC INVERTER 1Y80 TROUBLE (3A-H1)

STATIC INVERTER 1Y81 TROUBLE (3A-H2)

STATIC INVERTER 1Y82 TROUBLE (3A-H3)

STATIC INVERTER 1Y97 TROUBLE (3A-H4)

STATIC INVERTER 1Y98 TROUBLE (3A-G3)

What is the status of the Static Inverters?

04-1-01-L62-1 Attachment III is provided.

A. 1Y87, 1Y88, 1Y95, 1Y96 are on their normal supply.

1Y79, 1Y80, 1Y81, 1Y82, 1Y97, 1Y98, 1Y99 are on their alternate supply.

B. 1Y87, 1Y88, 1Y95, 1Y96 are on their alternate supply.

1Y79, 1Y80, 1Y81, 1Y82, 1Y97, 1Y98, 1Y99 are on their alternate supply.

C.

1Y87, 1Y88, 1Y96 are on their normal supply.

1Y79, 1Y80, 1Y81, 1Y82, 1Y97, 1Y98, 1Y99 are on their normal supply.

1Y95 is de-energized.

D. 1Y87, 1Y88, 1Y96 are on their normal supply.

1Y79, 1Y80, 1Y81, 1Y82, 1Y97, 1Y98, 1Y99 are on their alternate supply.

1Y95 is de-energized.

Answer: C

Question Comments: Answers A, B, and D are incorrect because no inverters would transfer from their normal DC supply to their alternate AC supply. Answer C is correct because no inverters would transfer from their normal DC supply to their alternate AC supply, and 1Y95 would have had its manual bypass switch in ALTERNATE (AC) and would not have been able to auto transfer to normal (DC). Bus 16AB supplies alternate power for 1Y95, so 1Y95 is de-energized. Tier 2 Group 1 This is a NEW question. 10 CFR 41.4

Image Reference: None

Open Reference Question

Handout 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 Mon Jun 20 08:34:35 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

Comments:

EB QUESTION: 47 (1.0 Points)

The plant is in Mode 1. The plant DC system is OPERABLE in its STANDBY configuration.

Battery chargers 1A4 and 1A5 normal/equalize switches are in NORMAL with load sharing ON.

Load Control Center 15BA6 trips, resulting in loss of power to battery charger 1A4.

How does this condition affect the Division 1 battery parameters?

A. The ‘A’ battery parameters will be unaffected, since battery charger 1A5 will pick

up load as necessary, without operator action.

B. The ‘A’ battery parameters will be unaffected only if battery charger 1A5 is manually placed into service immediately following loss of 15BA6.

C. ‘A’ battery bank voltage will slowly deteriorate. Average specific gravity will go down.

D. ‘A’ battery bank voltage will slowly deteriorate. Average specific gravity will go up.

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 Mon Jun 13 09:22:05 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

Comments:

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

The plant is at 50% power.

Due to grid instability, Division 2 Diesel Generator (DG12) is supplying bus 16AB which is separated from offsite power in accordance with the Loss of AC Power ONEP.

The reference leg for condensing pot B21-D004B ruptures in the drywell.

What will be the effect on DG12 and bus 16AB?

A. DG12 output breaker will open and Division 2 loads will be shed when sensed wide range reactor level goes below Level 1. After a time delay, DG12 output breaker will re-close and appropriate loads will sequence on.

B. DG12 output breaker will open and Division 2 loads will be shed when high drywell pressure is reached. After a time delay, DG12 output breaker will re-close and

appropriate loads will sequence on.

C. DG12 output breaker will remain closed. Division 2 loads will be shed when sensed wide range reactor level goes below Level 1. After a time delay, appropriate loads will sequence on.

D. DG12 output breaker will remain closed. Division 2 loads will be shed when high drywell pressure is reached. After a time delay, appropriate loads will sequence on.

Answer: D

Question Comments: Answers A and B are incorrect because DG12 output breaker will remain closed since it is carrying the bus alone. Answer C is incorrect because rupture of the reference leg would cause indicated level to go high, not low. Answer D is correct because DG12 output breaker will remain closed since it is carrying the bus alone, and a Div 2 LOCA signal would have to be due to drywell pressure since sensed level fails high. 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

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: MikeRasch at Mon Jun 20 09:04:22 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

Comments:

EB QUESTION: 49 (1.0 Points)

The plant was at 100% power when the feeder breaker to bus 16AB from ESF Transformer 21 tripped.

Division 2 Diesel Generator (DG12) automatically started and is running at 450 rpm, but DG12 output breaker, 152-1608, failed to close due to blowing control power fuses.

Bus 16AB is still de-energized 20 seconds following the loss of power to the bus.

Which one group of the following indications on 1H13-P864 associated with Division 2 Diesel Generator (DG12) would be expected for this condition?

Assume all indicators are operating as designed.

A. Annunciator DIV 2 LSS SYS FAIL (1H13-P864-2A-H1) illuminated

DG12 Frequency meter P75-R601B on 1H13-P864 downscale

DG-12 READY TO LOAD status light on 1H13-P864-2B extinguished

B. Annunciator DIV 2 LSS SYS FAIL (1H13-P864-2A-H1) illuminated

DG12 Frequency meter P75-R601B on 1H13-P864 downscale

DG-12 READY TO LOAD status light on 1H13-P864-2B illuminated

C. Annunciator DIV 2 LSS SYS FAIL (1H13-P864-2A-H1) illuminated

DG12 Frequency meter P75-R601B on 1H13-P864 indicating 60 Hz

DG-12 READY TO LOAD status light on 1H13-P864-2B extinguished

D. Annunciator DIV 2 LSS SYS FAIL (1H13-P864-2A-H1) extinguished

DG12 Frequency meter P75-R601B on 1H13-P864 indicating 60 Hz

DG-12 READY TO LOAD status light on 1H13-P864-2B illuminated

Answer: C

Question Comments: Answer A is incorrect because DG frequency would indicate ~60 hz since DG12 is running. Answer B is incorrect because the DG-12 READY TO LOAD status light is powered from bus 16AB, DG frequency would indicate ~60 hz since DG12 is running. Answer C correct because no power is being supplied to bus 16AB but the DG is running. Answer D is incorrect because the DG-12 READY TO LOAD status light is powered from bus 16AB. 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-00881

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis
Objectives:

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

KA References:

1. 264000
2. GENERIC 2.4.48 Ability to interpret control room indications to verify the status and operation of [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: MikeRasch at Mon Jun 20 09:05:22 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

Comments:

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

The plant is operating at 100% power.

Instrument Air Supply Header to Auxiliary Building valve P53-F026A fails closed due to a relay contact failure.

What effect will this have on the Drywell Coolers/Chillers?

A. The Division 1 Outlet dampers for the Drywell Coolers will fail open, and the Division 2 Outlet dampers will fail closed. Drywell Chiller Pressure Control valves ‘A’ will fail closed and Drywell Chiller Pressure Control valves ‘B’ will fail open.

B. The Division 1 Outlet dampers for the Drywell Coolers will fail closed, and the Division 2 Outlet dampers will fail open. Drywell Chiller Pressure Control valves ‘A’ will fail closed and Drywell Chiller Pressure Control valves ‘B’ will fail open.

C. The Division 1 Outlet dampers for the Drywell Coolers will fail closed, and the Division 2 Outlet dampers will fail open. Drywell Chiller Pressure Control valves ‘A’ will fail open and Drywell Chiller Pressure Control valves ‘B’ will fail closed.

D. The Division 1 Outlet dampers for the Drywell Coolers will fail open, and the Division 2 Outlet dampers will fail closed. Drywell Chiller Pressure Control valves ‘A’ will fail open and Drywell Chiller Pressure Control valves ‘B’ will fail closed.

Answer: B

Question Comments: Answer A is incorrect because Div 2 dampers fail open and Div 1 closed. Answer B is correct because dampers and valves fail as stated. Answer C is incorrect because Drywell chiller B valves fail open and A fails closed. Answer D is correct because Div 2 dampers and Drywell chiller B valves fail open and A dampers and valves fail closed. 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-00882

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-OPS-M5100 Objective: 9.3; 10.3

KA References:

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

References:

1. M-1101
2. M-1072B

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. M51: Drywell Cooling System
2. P53: Instrument Air System
3. P72: Drywell Chill Water System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 09:17:44 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

Comments:

EB QUESTION: 51 **(1.0 Points)**

The plant is operating at rated conditions.

Unit 2 Instrument Air Compressor is in service with Plant Air Dryer B P51-D001B in service.

Plant Air Dryer A P51-D001A is shutdown to a standby lineup.

Service Air Compressor A is in service.

Plant Air Dryer B After Filter P51-D003B has suddenly become clogged with a foreign object and is passing less than 1% of its normal volume of air.

Which one of the following describes the response of the Instrument, Plant and Service Air Systems to this problem?

Drawings M-1067A and G , M-1068D and M-1126 are provided.

A. Plant Air Dryer A will automatically align itself and supply the Instrument Air Header with air being supplied fro Unit 2 Instrument Air Compressor.

B. Unit 1 Instrument Air Compressor will auto start and align itself through the Unit 1 Instrument Air Dryer Skid.

C. Service Air Compressor A will automatically align itself to the Instrument Air Header via P52-F500 at 95 psig to maintain Instrument Air Header pressure.

D. Unit 2 Instrument Air Compressor will have to be manually aligned to the Instrument Air Header via Plant Air Dryer Bypass valve P51-F209B until Plant Air Dryer A can be started.

Answer: D

Question Comments: GGNS has been in the process of altering the Instrument Air, Service Air and Plant Air systems installing new air drying systems and air compressors. This is an attempt to upgrade the systems for reliability. Plant Air Compressor B has replace Service Air Compressor B. Unit 1 and Unit 2 Instrument Air Compressors are operational. Unit 1 Instrument Air Dryer skid is still installed but valved out of service per the System Operating Instructions. The new Plant Air Dryer skids are operational and supplying the Instrument Air drying needs. With one Plant Air Dryer Skid in service and the second skid shutdown, a clogged filter on the outlet of the in service Air Dryer will reduce/block air flow to the Instrument Air Header. No matter the source of the air, it must go through the in service

air dryer before going to the Instrument Air Header. Answer A is INCORRECT because the Plant Air Dryer that is out of service will not automatically align it self to take the load. Answer B is INCORRECT because the Unit 1 Instrument Air Dryer skid is valved out of service with manual valves and requires local operator action to place it in service. Answer C is INCORRECT because of the location of the Plant Air Dryer that is blocking flow will not allow the service air compressor to supply the system. Answer D is CORRECT because opening P51-F209B will bypass the air dryer and allow pressurization of the air header with raw compressed air. Tier 2 Group 1 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-00883

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

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:

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 Mon Jun 20 09:37:57 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

Comments:

EB QUESTION: 52 (1.0 Points)

The plant is in Mode 3.

RHR 'A' is 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 NOT be indicative of an RHR 'A' Heat Exchanger tube rupture if a tube leak occurred?

A. RHR 'A' pump discharge pressure trending down

B. Reactor water level trending up

C. SSW 'A' radiation monitor readings trending up

D. Reactor coolant temperature trending up

Answer: B

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 correct because RHR would be at a higher pressure than SSW, so RPV level would go down. Answer C is incorrect 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

Objectives:

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

KA References:

1. 400000 K1.01 Service water system [3.2/3.3]

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 Mon Jun 13 09:58:09 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

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: 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

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: MikeRasch at Mon Jun 13 10:10:30 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

Comments:

EB QUESTION: 54 (1.0 Points)

A control rod sequence exchange is being performed at 80% power.

As part of the control rod movement plan, the operator selects control rod 28-17, which is at notch position 36, and begins to continuously withdraw it.

The control rod stops withdrawing and settles at notch position 40 due to an expected control rod block.

What is the purpose of this control rod block?

A. Mitigate a control rod drop event

B. Prevent exceeding fuel preconditioning limits

C. Prevent violation of the Minimum Critical Power Ratio (MCPR) Safety Limit

D. Normalize core exposure burn rates

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: MikeRasch at Mon Jun 13 10:15:34 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

Comments:

EB QUESTION: 55 (1.0 Points)

At 100% power, Reactor Recirculation loop flows must be within 5% of rated core flow of one another.

What is the basis of this requirement?

A. To normalize the core radial flux distribution.

B. To meet core flow coast down assumptions of the LOCA analysis.

C. To prevent excessive loading of Recirc pump motor windings and electrical penetration.

D. To minimize vibration stresses on Recirc piping caused by jet pump cavitation.

Answer: B

Question Comments: Tech Spec bases 3.4.1 specifically states that the limit for Recirc flow mismatch is an assumption in the LOCA analysis that assures sufficient flow and cooling during coastdown of the unbroken Recirc loop. That is

why answer B is correct. Answers A, C, and D are unrelated to this, and that is why they are incorrect. Tier 2 Group 2 This is a NEW question. 10CFR 41.3/41.10/43.2/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00887

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-OPS-B3300 Objective: 2; 44
2. CourseID: GLP-OPS-MCD16 Objective: 4

KA References:

1. 202001 K1.01 Core flow [3.6/3.7]

References:

1. Tech Spec Bases B3.4.1
2. Tech Spec Bases Surveillance 3.4.1.1

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 Mon Jun 13 10:28:15 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

Comments:

EB QUESTION: 56 (1.0 Points)

The plant was at 100%.

Reactor Recirc Flow Control Valves were at 68% valve position.

Then, Feed Water Line 'A' in the drywell, upstream of feed water check valve B21-F010A, suffered a guillotine break.

All systems responded as designed.

Maximum drywell pressure was 4.5 psig.

Maximum reactor pressure was 1025 psig.

Minimum reactor water level was -80 inches wide range.

Which one of the following describes the Reactor Recirc System status five minutes after the feed water line break?

A. Recirc Pump breakers CB-1A/B, CB-2A/B, CB-4A/B, and CB5A/B are OPEN. CB-3A/B are CLOSED.

Recirc Flow Control Valves 'A' and 'B' are approximately 20% open.

B. Recirc Pump breakers CB-1A/B, CB-2A/B, CB-3A/B, CB-4A/B, and CB5A/B are OPEN.

Recirc Flow Control Valves 'A' and 'B' are approximately 20% open.

C. Recirc Pump breakers CB-1A/B, CB-2A/B, CB-4A/B, and CB5A/B are OPEN. CB-3A/B are CLOSED.

Recirc Flow Control Valves 'A' and 'B' are approximately 68% open.

D. Recirc Pump breakers CB-1A/B, CB-2A/B, CB-3A/B, CB-4A/B, and CB5A/B are OPEN.

Recirc Flow Control Valves 'A' and 'B' are approximately 68% open.

Answer: C

Question Comments: For this event, all feedwater flow to the reactor would be lost. Water level would rapidly fall to -41.6 inches. Recirc pumps would trip to slow speed at 11.4 inches (CB-5s open, CB-1s and CB-2s close) and an ATWS RPT would occur at -41.6 inches (CB-1s, 2s, 4s, 5s open). The CB-3s are the only breakers that remain closed. They would only trip on EOC-RPT, which does not occur since power is <40%, where it is bypassed, by the time the turbine trips on reverse power. Recirc flow control valves do not run back to 20% because both still RFPs are running by the time Recirc pumps transfer out of fast speed. Also, both Recirc HPUs trip almost immediately due to high drywell pressure, 1.23 psig, from the FW line break. Answer A is incorrect because Recirc FCVs do not runback as stated above. Answer B is incorrect because CB-3s remain closed and Recirc FCVs do not runback as stated above. Answer C is correct because of reasons stated above. Answer D is incorrect because of reasons stated above. 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

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 Mon Jun 20 09:41:46 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

Comments:

EB QUESTION: 57 (1.0 Points)

Which one of the following is a requirement for use of Reactor Water Clean-up System (RWCU) as an alternate method of decay heat removal in Mode 5?

A. At least one recirculation pump must be placed in operation.

B.

The Component Cooling Water temperature control valve, P44-F501, must be set at 65°F.

C. RWCU REGEN HX BYP VLV G33-F107 must be fully opened and must remain fully opened for all conditions.

D. A temporary thermocouple must be installed to monitor reactor coolant temperature.

Answer: D

Question Comments: Answer A is incorrect because RWCU operation does not require recirculation pump operation. Answer B is incorrect because the minimum allowed CCW TCV setting is 70°F. Answer C is incorrect because G33F107 is required to be throttled open/closed as necessary to control temperature. Answer D is correct because there is insufficient RWCU flow to provide an accurate bulk coolant temperature. Tier 2 Group 2 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-00889

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-OPS-E1200 Objective: 14.1
2. CourseID: GLP-OPS-G3336 Objective: 10.1

KA References:

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

References:

1. 04-1-01-E12-1 3.8.16.c
2. 04-1-01-G33-1
3. 05-1-02-III-1

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

Task References:

Question Last Revised By: MikeRasch at Mon Jun 13 10:45:43 CDT 2005
Question History:

1. Created by mrasch at Mon Jun 13 10:45:43 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

Comments:

EB QUESTION: 58 (1.0 Points)

The plant is in a Station Blackout with the following parameters:

Reactor power 0%

Reactor water level -110 inches wide range

Reactor pressure 550 psig

Suppression pool temperature 145°F

Suppression Pool level 10.3 feet

ADS accumulator pressure 145 psig

Which one of the following is the preferred method to be used to depressurize the RPV under these conditions?

A.

Automatic Depressurization System (ADS) valves

B. Reactor Core Isolation Cooling (RCIC)

C. Reactor Water Clean-up (RWCU)

D. Main Steam Drains and Offgas Preheater

Answer: B

Question Comments: Emergency depressurization is required due to suppression pool level <14.56’. Answer A is incorrect because SRV use is not allowed due to suppression pool level < 10.5’. Answer B is correct because it is a system listed by EP-2A step 55B that is inboard of MSIVs and is designed to function during a SBO. It can be used , even at low suppression pool level, during an emergency. Answer C is incorrect because RWCU is de-energized during a SBO and is not listed in EP-2A step 55B. Answer D is incorrect because MSIVs are closed in a SBO, so there is no steam flow path. 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. EP-3 step 43
2. EP-2A step 55B

3. 02-S-01-27 (No rod position indication during a SBO, therefore enter EP-2A)
4. 04-1-01-E51-1 Steps 3.3; 3.4

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
2. E51: Reactor Core Isolation Cooling System
3. G33: Reactor Water Cleanup
4. M71: Containment and Drywell Instrumentation System
5. R21: 4.16 KV AC Power System

Categories:

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

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 09:44:41 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

Comments:

EB QUESTION: 59 (1.0 Points)

The plant was operating at 100% power when a loss of all offsite power caused all Main Steam Isolation Valves to close and a Reactor Scram.

Hydrogen Water Chemistry was in service

Which one of the following describes the response of the Hydrogen Water Chemistry (HWC) System and its affect on the Offgas system?

A. Hydrogen Water Chemistry will commence a normal shutdown of the HWC system that will provide timed reduction in injection of Hydrogen and Oxygen into Offgas to allow for recombination of Hydrogen and Oxygen preventing the possibility of a fire in Offgas. The loss of heating steam to the Offgas Preheater will NOT significantly affect the Hydrogen concentrations in Offgas.

B. Hydrogen Water Chemistry will commence a Hydrogen Immediate Trip with Normal Oxygen shutdown of the HWC system that will immediately isolate Hydrogen to the Condensate System but allow a normal shutdown of the Oxygen injection to the Offgas System. Even though heating steam to the Offgas Preheater is lost, the chances of an Offgas Hydrogen fire are reduced.

C. The loss of power will result in an Emergency HWC shutdown causing elevated Hydrogen levels in the Offgas system since there is reduced Oxygen for the Hydrogen to recombine with. This combined with the loss of heating steam to the Offgas Preheater will raise the possibility of a fire in the Offgas system.

D. Hydrogen Water Chemistry will commence a normal shutdown of the HWC system that will provide timed reduction in injection of Hydrogen and Oxygen into Offgas, however due to the loss of heating steam to the Offgas Preheater the possibility of a fire in the Offgas System is raised.

Answer: B

Question Comments: Answers A and D are incorrect because hydrogen injection will immediately be isolated. Answer B is correct because the system is designed to immediately secure hydrogen with a timed reduction of oxygen to ensure excess oxygen available for recombination. This ensures very low hydrogen concentrations to reduce possibility of a fire. Answer C is incorrect because, as stated for answer B, the result is reduced levels of hydrogen and lower chances of fire. 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-P7300 Objective: 10; 11.4; 11.5; 11.6
2. CourseID: GLP-OPS-N6465 Objective: 14.3

KA References:

1. 239001 K3.04 Offgas system [2.8/2.8]
2. 239001 K1.07 Offgas system [2.9/3.1]
3. 271000 A1.13 Hydrogen gas concentration [3.2/3.7]
4. 271000 A1.14 Oxygen gas concentration [2.7/3.0]
5. 271000 K1.06 Main steam system [2.8/2.9]
6. 271000 K1.08 Oxygen injection system: Plant-Specific [2.3/2.3]
7. 271000 K4.04 The prevention of hydrogen explosions and/or fires 3 [3.3/3.6]

References:

1. 04-1-02-1H13-P845 1A-D7 1.3.6, 1.3.7, 3.1.2
2. 04-1-01-P73-1
3. E-7176-005, 006

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
3. P73: Hydrogen Water Chemistry

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 09:49:29 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

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 effluent of either MSIV LCS.

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) ‘A’ and ‘B’ are in operation.

D. Operation of the Outboard MSIV LCS in conjunction with Standby Gas Treatment System (SGTS) ‘B’ is the preferred alignment, since SGTS ‘B#146; provides the longest transport time and each is powered from Division 2.

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

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 Mon Jun 20 09:52:39 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

Comments:

EB QUESTION: 61 **(1.0 Points)**

The plant was operating at 99% power, when a winding fault caused Service Transformer (ST) 21 to lock out.

The running Electro-Hydraulic Control (EHC) fluid pumps tripped, and the standby EHC pump failed to automatically start.

The resulting pressure transient caused some amount of fuel damage.

Bus 14AE has been reenergized from ST11 and under-voltage lockout relays were reset.

EHC pumps have NOT been restarted but are available.

All other systems affected by the power loss have been recovered.

An ATWS currently exists with the following conditions:

Reactor power 15%

Reactor pressure 1050 psig controlled by Safety Relief Valves (SRVs) manually 800 - 1060 psig.

Reactor level -80 inches wide range, being controlled on startup level control with Reactor Feed Pump ‘A’

Offgas Pretreatment Radiation monitor 2000 mr/hr (above the high alarm, below the Hi-Hi alarm)

Average Main Steam Line Radiation Monitor reading 2300 mr/hr (below the high alarm)

Main Steam Isolation Valves (MSIVs) are open.

Average suppression pool temperature is 96°F, slowly rising.

Which one of the following represents the pressure control strategy to be followed under these conditions?

A. Close MSIVs. Manually control SRVs 800 psig to 1060 psig.

B. Place Suppression Pool Cooling A and B in service, and allow SRVs to cycle on Low-Low Set. Augment SRVs with Main Steam Line Drains.

C. Restart EHC pumps. Ensure Main Bypass Valves are operating, and lower Pressure Reference to 900 psig. Install Emergency Procedure Attachments 7 and 8.

D. Take manual control of SRVs and Main Steam Line Drains, and lower pressure to 450 psig to 600 psig.

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

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 Mon Jun 20 09:59:28 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

Comments:

EB QUESTION: 62 (1.0 Points)

During the Main Turbine roll to rated speed, procedures require at least a 55°F margin on the Turbine Stress Evaluator (TSE).

What is the basis for this requirement?

A.

Ensure adequate steam flow to cool latter blade stages.

B. Prevent blade failure due to resonance vibration due to sustained operation at critical speeds.

C. Ensure adequate steam flow for synchronization and minimum loading of the Main Generator.

D. Provide sufficient clearances between rotating and stationary turbine blades.

Answer: B

Question Comments: The caution at step7.1.5a of 03-1-01-1 states the 55°F margin on TSE is to ensure the turbine can be rolled to 1800 rpm with no hold points at critical speeds. Critical speed causes resonant vibrations that can damage turbine blades. This is why answer B is correct. Answer A is incorrect because this adverse effect is caused by sustained low steam flow at 1800 rpm. Answer C is incorrect because the associated requirement is to have 20% bypass valve opening, not TSE margin. Answer D is incorrect because this is a turbine relative expansion concern. Tier 2 Group 2 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-00894

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-OPS-N3000 Objective: 2.8; 8
2. CourseID: GLP-OPS-IOI01 Objective: 32.2

KA References:

1. 245000 A3.02 Turbine roll to rated speed [2.8/2.8]

References:

1. 03-1-01-1 section 2.12.3, caution section 7.1.5

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

Categories:

1. Integrated Plant Operations
2. Systems
3. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 10:02:26 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

Comments:

EB QUESTION: 63 **(1.0 Points)**

The plant is operating at 100% power.

Condensate Pump Recirc Isolation valve N19-F010 and Condensate Booster Pump Recirc Isolation valve N19-F057 are closed for I&C work on the Bailey INFI-90 system in H22-P171, scheduled to begin later in the shift.

Which one of the following describes what would happen if the flow signal from flow element N19-N065 on the outlet of the Low Pressure Feed Water Heaters were to intermittently fail to 2.9 mlbm/hr for 75 seconds, and then return to normal?

A. Condensate Pump A would trip. Condensate Booster Pumps would trip due to low suction pressure. The reactor would scram due to low water level.

B. Condensate Booster Pump A would trip. Reactor Feed Pumps would trip due to low suction pressure. The reactor would scram due to low water level.

C. Condensate Pump A would trip. Reactor Feed Pump speeds would rise to maintain normal normal water level.

D. Condensate Booster Pump A would trip. Reactor Feed Pump speeds would rise to maintain normal water level.

Answer: D

Question Comments: Minimum flow lines are isolated, so no additional flow will be established. Answer A is incorrect because the condensate low flow trip signal, < 1.2 mlbm/hr per pump, has to exist for 90 seconds to trip the first condensate pump, so no condensate pumps would trip. Also, reactor feed pumps would remain running, and the reactor would not scram. Answer B is incorrect because reactor feed pumps would remain running , and the reactor would not scram. Answer C is incorrect because the condensate low flow trip signal, < 1.2 mlbm/hr per pump, has to exist for 90 seconds to trip the first condensate pump, so no condensate pumps would trip. Answer D is correct because there would be no flow path to provide additional booster pump flow to meet the minimum requirement of 3 mlbm/hr total for 3 booster pumps running. One booster pump would trip in 60 seconds, and the minimum flow requirement would shift to 2 mlbm/hr, and would thus be satisfied. The condensate system is designed such that 2 booster pumps can provide enough flow for 100% power, so RFPs and the reactor would remain running. 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: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-OPS-N1900 Objective: 2; 13; 14; 21

KA References:

1. 259001 A2.03 Loss of condensate pump(s) [3.6/3.6]

References:

1. 04-1-01-N19-1 sections 3.4, 3.6, 3.20, 3.21
2. M-1053 A and B

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 Mon Jun 20 10:12:56 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

Comments:

EB QUESTION: 64 (1.0 Points)

Which one of the following is NOT an accepted method for monitoring Reactor Coolant System (RCS) leakage in the drywell to meet Tech Spec leakage limits?

Portions of the Daily Operations Log 06-OP-1000-D-0001 are provided.

A. Divide the volume of the Drywell Equipment Drain Sump by the time between automatic sump pump downs indicated on PDS to determine identified leakage.

B. Use the total rise in Drywell Equipment Drain Sump Level indicated on recorder E31-R185 (area 9, 139') along with the reading interval to calculate identified leakage.

C. Monitor for increases in noble gas activity on Drywell Atmospheric Monitoring System Recorder D23-R600 to identify gross changes in unidentified leakage.

D. Record unidentified leakage rate in gpm directly from a PDS computer point if the point is operable.

Answer: A

Question Comments: Answers B, C, and D are all procedural methods for monitoring drywell leakage described in the operator Tech Spec Rounds for determining leakage., and that makes them incorrect answers, since the question is asking what is NOT an approved method. Answer A is correct because the entire volume of the sump is NOT used when determining the leakage rate. 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-E3100 Objective: 5.4; 10
2. CourseID: GQC-CRO01 Section 2 Item 4 Objective: Item 4

KA References:

1. 268000 K1.04 Reactor building floor drains: Plant-Specific [2.7/2.9]

References:

1. 06-OP-1000-D-0001 Att I Data Sheet II Items 25, 13
2. Tech Spec Bases B SR3.4.5.1

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: MikeRasch at Mon Jun 20 14:01:47 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

Comments:

EB QUESTION: 65 (1.0 Points)

Excessive leakage from which one of the following Fuel Pool Cooling and Clean-up (FPCC) components would cause leak detection alarm FPCC FLTR DMIN SYS LEAK to alarm on 1H13-P680 due to high input into the FPCC leak detection standpipe?

Drawings M-1088D; 1090A; 1090B; 1098A; 1098B

A. FPCC Pump 'A'

B. Spent Fuel Pool liner

C. FPCC Heat Exchanger 'B'

D. FPCC Backwash Receiving Tank Transfer Pump

Answer: D

Question Comments: Answer A is incorrect because FPCC Pump "A" drains to Floor Drain Sump. Answer B is incorrect because Spent Fuel Pool liner drains to Floor Drain Sump. Answer C is incorrect because FPCC Heat Exchanger "B" drains to Equipment Drain Sump (G41 side) and Chem Waste Sump (P42 side). Answer D is correct because FPCC Backwash Receiving Tank Transfer Pump area drains go to the standpipe. Tier 2 Group 2 This is a NEW question. 10 CFR 41.4

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00897

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-OPS-E3100 Objective: 6.12

KA References:

1. 290001 K4.03 Fluid leakage collection [2.8/2.9]

References:

1. M1090B
2. M1098B
3. M1088D
4. M1098A

TrainingPrograms:

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

Systems:

1. E31: Leak Detection System
2. G41: Fuel Pool Cooling and Cleanup System

Categories:

1. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Mon Jun 13 12:01:57 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

Comments:

EB QUESTION: 66 (1.0 Points)

The plant was at 95% power when Reactor Feed Pump ‘B’ tripped, followed by a runback of both Recirc Flow Control Valves.

Reactor power is now swinging 72% to 74% on APRMs, with Total Core Flow now 56 mlbm/hr.

Given the following indications on H13-P680, which one would continued operation of the unit be allowed?

05-1-02-III-3 is provided.

A. APRM oscillations of 14% peak-to-peak

B. Annunciators PBDS ‘A’ INOP (5A-A6) and PBDS ‘B’ INOP (7A-A6) alarm 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

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: MikeRasch at Mon Jun 20 10:18:27 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

Comments:

EB QUESTION: 67 **(1.0 Points)**

The plant is in an ATWS with a steam leak in the Drywell.

Plant conditions are as follows:

Reactor power 7%

Reactor water level -172 inches and is being controlled in band.

Reactor pressure 530 psig and is being controlled in band.

Main Condenser is NOT available.

Drywell Pressure 9 psig, rising slowly

Drywell temperature 225°F, rising slowly

Containment pressure 7 psig, rising slowly

Containment temperature 125°F, rising slowly

Suppression Pool temperature 145°F, rising slowly

Suppression Pool level 24.6 feet, rising slowly

Standby Liquid Control systems have failed.

Which one of the following is the appropriate action for the given conditions?

A. Lower suppression pool level to <18.81 feet using RCIC or HPCS.

B. Lower reactor pressure, irrespective of resulting cool down rates, to maintain operation in the Safe Zone of the Heat Capacity Temperature Limit (HCTL) curve to avoid emergency depressurization.

C. Initiate RHR 'A' and 'B' in Containment Spray mode. If unable to maintain containment pressure in the Safe Zone or the 'N/A' regions of the Pressure Suppression Pressure curve, then perform emergency depressurization in accordance with EP-2A, including termination of injection.

D. Immediately perform emergency depressurization in accordance with EP-2A, including termination of injection.

Answer: D

Question Comments: Answer A is INCORRECT because HPCS and RCIC Test Return Lines to the CST are isolated due to an Auxiliary Building Isolation Signal. Answer B is INCORRECT because HCTL is not challenged and Emergency Depressurization is required per Containment/Suppression Pool Water level > 24.4 feet and no method available to lower level.

Answer C is INCORRECT because Emergency Depressurization is not an option it is required. Answer D is CORRECT because EP – 3 Step 52 cannot be answered YES so Emergency Depressurization is required and RPV Injection Termination is required per EP - 2A. Tier 3 This is a NEW question. 10CFR 41.7/41.9/41.10/43.5

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: GG-1-LP-RO-EP03 Objective: 3; 6
2. CourseID: GG-1-LP-RO-EP02A Objective: 7

KA References:

1. GENERIC 2.1.25 Ability to obtain and interpret station reference materials such as graphs / [2.8/3.1]

References:

1. EP-3 step 53
2. 04-1-01-E51-1 step 6.3.1c
3. 04-1-01-E22-1 Note at Step 6.3.2a
4. 05-1-02-III-5 for P11 isolations
5. 04-1-01-P11-2 section 5.8

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
2. E51: Reactor Core Isolation Cooling System
3. P11: Condensate and Refueling Water Transfer System

Categories:

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

4. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 10:23:16 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

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 Yellow plastic seals are used on Fire Protection Valves. Blue Tie-Wraps

Comments: 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

Comments:

EB QUESTION: 69 (1.0 Points)

Which one of the following situations presents an operability concern regarding Intermediate Range Monitors?

Data Sheets concerning IRMs from 06-OP-1000-D-0001; 03-1-01-1 and 03-1-01-3 are provided.

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 2×10^4 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

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 Jun 20 10:27:49 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

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.

For which one of the following activities would notification by control room personnel to Refueling Floor supervision be required?

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: MikeRasch at Mon Jun 20 10:29:40 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

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 Comments: 50% of the maximum radiation dose rate of a High Radiation Area of 999 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

Comments:

EB QUESTION: 72 (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.

A search and rescue team is being assembled to locate and extract the mechanic, who is believed to be gravely injured.

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?

10-S-01-17 is provided.

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 dose limits based on the situation. Further dose limit 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 to 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 3 This is a NEW question. 10CFR 41.10/41.12/43.4/43.5

Image Reference: None

Open Reference Question

Handout 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 6.1

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 Mon Jun 20 10:31:56 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

Comments:

EB QUESTION: 73 **(1.0 Points)**

Which one of the following describes the basis of Caution 1 of the Emergency Procedures?

A. The RPVST curve is based on de-gassing in the instrument runs.

B.

Caution 1 is only based on environmental temperature effects to the reference legs of level instruments, since variable legs would be unaffected.

C. The Condition 2 table of Indicated Level and Reference Leg Temperature protects against using an instrument for indication with actual level below the variable leg tap.

D. The question “Can RPV water level be determined?” on EP-2/2A must be answered‘NO’anytime operation is plotted in the Possible Boiling Region of the RPV Saturation Temperature –RPVST curve.

Answer: C

Question Comments: Caution 1 of the Emergency Procedures concerns the reliability of the RPV Level Instrumentation. It is divided into 2 sections. Section 1 concerns reference leg area temperatures vs RPV pressure and the possibility of reaching saturation conditions for the liquids inside the reference legs which possibly would cause erroneous readings. Section 2 concerns lower indicated levels with elevated reference leg temperatures that may cause actual level to be below the taps of the level instrument and still read on scale. Operations Philosophy allows instruments to be considered good if specifically Answer A is INCORRECT because the RPVST curve is based on saturation conditions for the Reference legs. Answer B is INCORRECT because Caution 1 is not just the reference legs it also includes variable legs. Answer C is a CORRECT statement concerning Section 2 of Caution 1. Answer D is INCORRECT because the conditions may be in the Unsafe region of Figure 2 but the Instrument not be showing the effects and still be reading true that is the reason is states POSSIBLE BOILING and Operations Philosophy 02-S-01-27 allows instruments to be used even with plant conditions in the Unsafe (Possible Boiling) region. 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: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GG-1-LP-RO-EP02 Objective: 11
2. CourseID: GLP-OPS-PROC Objective: 59.3

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 1
2. PSTG B-5-2 thru 10

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 Mon Jun 20 10:35:55 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

Comments:

EB QUESTION: 74 (1.0 Points)

The plant is at 100% power when a fire occurs at Main Transformer ‘C’ due to an oil leak.

Which one of the following describes the fire suppression for Main Transformer ‘C’?

A. A Dry Pipe system with fused-closed heads will automatically actuate fire water deluge when the fuse melts at a predetermined temperature.

B. Fire water deluge is only available by manually initiation and realignment of manual valves.

C. Fire water is the primary means of fire suppression, with carbon dioxide as the backup system.

D. Rate of rise heat detectors energize a solenoid valve on the deluge valve to automatically actuate fire water deluge.

Answer: D

Question Comments: The fire protection system for the Main Transformers is an Automatic Deluge System which operates off rate of rise heat detectors which energize a solenoid valve to open the Deluge Valve and admit fire protection water to open sprinkler heads surrounding the transformer. Answer A is INCORRECT because these systems are not classified as a Dry Pipe system. Answer B is INCORRECT because this describes a fire protection system similar to the Charcoal Filter trains. Transformer Automatic Deluge Systems require no operator action to actuate. Answer C is INCORRECT because even though the Transformer is electrical equipment there is NO CO2 fire suppression systems to back the water system. Answer D is CORRECT because this describes the operation of an Automatic Deluge Fire Suppression System. Tier 3 This is a NEW question. 10 CFR 41.4/41.8

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00905

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

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. 04-S-01-P64-1 Att VI
2. M-0035B and J

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. Systems
2. Continuing Training

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 10:38:38 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

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

Comments:

EB QUESTION: 76 (1.0 Points)

It is a Division 1 work day with the plant at 100% power.

During an I&C surveillance in 1H13-P629, an incidental electrical short causes Topaz inverters E21-K701 (E21A-PS1) and E21-K702 (E21A-PS2) fuses to blow.

The DC feeder breaker to the two Topaz inverters, 72-11A18, also trips.

The following illuminate on 1H13-P601:

Annunciator ECCS DIV 1 125VDC ISOL PWRLOSS (21A-G7)
Annunciator LPCS SYS OOSVC (21A-H8)
Annunciator RCIC SYS OOSVC (21A-H5)
Annunciator RHR A SYS OOSVC (20A-H6)

Status light LPCS LOGIC PWR FAIL (21B)

What affect will this have on the Low Pressure Coolant Injection (LPCI) mode RHR 'A'?

A. RHR 'A' will NOT automatically start and align for injection to the reactor, but it can be manually started and aligned from the Division 1 Remote Shutdown Panel, 1H22-P150. Once open, RHR Injection Valve E12-F042A CANNOT be closed from 1H13-P601.

B. RHR 'A' will NOT automatically start and align for injection to the reactor, but it can be manually started and lined up for injection from the Division 1 Remote Shutdown Panel, 1H22-P150. Also, RHR Injection Valve E12-F042A can be closed from 1H13-P601, but it will NOT open from 1H13-P601, and RHR 'A' pump can be started and stopped from 1H13-P601.

C. RHR 'A' will NOT automatically start and align for injection to the reactor. RHR 'A' pump and RHR Injection Valve E12-F042A CANNOT be controlled from the Main Control Room or the Division 1 Remote Shutdown Panel. RHR 'A' pump must be manually started by placing the local pistol grip hand switch on the front of RHR 'A' Pump breaker 152-1509 to CLOSE. E12-F042A must be opened manually using its local hand wheel.

D. RHR 'A' pump will NOT automatically start but can be manually started using its 1H13-P601 hand switch. E12-F042A will then automatically open when its pressure permissive is met.

Answer: B

Question Comments: The loss of 72-11A18 circuit breaker removes logic power to LPCS relays for ECCS initiation. This same logic supplies the signals to RHR ‘A’ LPCI logic. This logic is energize to operate. The RHR ‘A’ Pump start for manual operation from either H13-P601 or H22-P150 does not rely on this logic. Automatic operation for LPCI mode does. E12-F042A opening from H13-P601 either automatic from LPCI or handswitch requires pressure permissive. The controls for E12-F042A at the Remote Shutdown Panel bypass pressure permissives when manually operated. The automatic functions still use the Logic system. Closing E12-F042A from H13-P601 does not require any logic systems. Answer A is INCORRECT because E12-F042A can be closed from H13-P601 as long as the valve has power. Answer B is CORRECT. Answer C is INCORRECT because RHR Pump ‘A’ as long as it has power and control power it can be started and stopped from H13-P601 and E12-F042A can be closed. Answer D is INCORRECT because E12-F042A uses the same logic controls from LPCS for automatic opening on a LOCA signal therefore it will not automatically open. 10CFR 41.7, 41.10; 43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00908

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-OPS-E1200 Objective: 9.2; 13.2; 13.6; 2
2. CourseID: GLP-OPS-E2100 Objective: 9.2; 10.2; 17

KA References:

1. 295004 AA2.02: 3.5/3.9; AK2.03: 3.3/3.3
2. 203000 A2.05: 3.0/3.2; A2.14: 3.8/3.9

References:

1. ARI 04-1-02-1H13-P601-20A-H6; 21A-H8; 21A-G7; 20A-H6; 21A-H5
2. E1182-023; 024; 026; 028
3. E-1181-037; 043; 067; 081

TrainingPrograms:

Systems:

Categories:

Task References:

Question Last Revised By: Charles Bell at Thu Jun 16 14:06:58 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

Comments:

EB QUESTION: 77 (1.0 Points)

The plant is in an ATWS condition.

The Main Turbine has tripped, but one Combined Main Stop and Control Valve remained partially open.

Personnel must enter the Turbine bio-shield to assess the failure.

Postings at the entrance to this area read "HIGH RADIATION AREA" and "TECH SPEC MONITORING REQUIRED".

Which one of the following meets the minimum radiological monitoring requirements for this entry?

Technical Specification 5.7.1 is provided.

A. Thermo-luminescent Dosimeter (TLD) and a radiation monitoring device that continuously integrates the radiation dose rate and alarms when a preset integrated dose is received, if the dose rate levels in the area are unknown and personnel are unaware of them.

B. Thermo-luminescent Dosimeter (TLD), a radiation monitoring device that continuously integrates the radiation dose rate and alarms when a preset integrated dose is received, if the dose rate levels in the area are known and personnel are aware of them, and an individual qualified in radiation protection procedures with a radiation dose rate measuring device to perform periodic area surveys.

C. Thermo-luminescent Dosimeter (TLD), a radiation monitoring device that continuously indicates the dose rate in the area, and an individual qualified in radiation protection procedures with a radiation dose rate measuring device to perform periodic area surveys.

D. Thermo-luminescent Dosimeter (TLD) and a radiation monitoring device that continuously indicates the dose rate in the area.

Answer: D

Question Comments: Comments: Tech Spec 5.7.1a in addition to the normal TLD for entry into High Radiation Areas / Tech Spec Monitoring Required requires a minimum of one other device which 5.7.1a indicates a radiation monitoring device that continuously indicates radiation dose rate in the area. Answer A is INCORRECT because it requires additional personnel to enter the area and perform radiation assessment to allow the use of this type of device. Answer B is INCORRECT because it requires extra personnel and radiation monitoring dose rate monitoring devices. Answer C is INCORRECT because it requires both monitoring devices and additional personnel be in the Radiation Area. Answer D is CORRECT because it is the minimum equipment/personnel to meet the Tech Spec Monitoring requirements. 10 CFR 41.12; 43.4

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00909

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory
Objectives:

1. CourseID: Rad Worker Training, ELP-GET-RWT Objective: RWT44

KA References:

1. 295005 Generic 2.3.5: 2.3/2.5

References:

1. Tech Spec 5.7.1

TrainingPrograms:

Systems:

Categories:

Task References:

Question Last Revised By: Charles Bell at Thu Jun 16 14:09:50 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

Comments:

EB QUESTION: 78 (1.0 Points)

The plant is in an ATWS following a loss of all Electro-Hydraulic Control (EHC) pumps.

The following conditions exist:

Reactor power 10%

Reactor water level -175 inches; compensated fuel zone
(being controlled in band -167 inches to -197 inches)

Reactor pressure 510 psig (being controlled in band 450 - 600 psig)

Drywell Pressure 1.2 psig, rising slowly

Drywell temperature 125°F, rising slowly

Containment pressure 0.02 psig, rising slowly

Containment temperature 90°F, rising slowly

Suppression Pool temperature 145°F, rising slowly

Suppression Pool level 18.4 feet, rising slowly

Standby Liquid Control systems have failed.

Which one of the following actions would expand the margin to the Heat Capacity Temperature Limit curve the most?

A. Lowering the pressure band by 50 psig

B. Placing both loops of Suppression Pool Cooling in service

C. Initiating Suppression Pool Makeup

D. Opening Main Steam Line Drains to the condenser

Answer: C

Question Comments: Comments: This question applies the given Suppression Pool Level and Temperature and applies given Reactor Pressure to determine acceptability and a course of action. Answer A is INCORRECT because lowering the lower limit of the Reactor Pressure will only gain about 5 degrees F margin. This would also add heat to the Suppression Pool to make the change. Answer B is INCORRECT because Suppression Pool Cooling has insufficient capability to reduce Suppression Pool temperature avoiding the limits of HCTL at the amount of heat being added. Answer C is CORRECT because adding relatively cool water to the Suppression Pool lowers average temperature and raises the level of the water (about 5 ft.) in the Suppression Pool raising the amount of heat the pool is capable of accepting. (At least a 10 degree F rise in margin.) Answer D is INCORRECT because MSL Drains are only capable of passing about 5% power and at the present values this would not expand the margin to HCTL that greatly it just redirects where the heat is being deposited. 10 CFR 41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00910

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

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:

Systems:

Categories:

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 10:44:37 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

Comments:

EB QUESTION: 79 (1.0 Points)

The basis for the Technical Specification LCO limit for Primary Containment average air temperature is:

A. Prevent exceeding the Containment design temperature limit which would degrade the Containment structure under accident loads during a Design Basis Accident

(DBA) LOCA.

B. Prevent exceeding the Containment design pressure limit which would degrade the equipment inside the Containment during a Design Basis Accident (DBA) LOCA.

C. Prevent exceeding the Containment design temperature limit which would degrade the equipment inside the Containment during an ATWS.

D. Prevent exceeding the Containment design pressure limit which would degrade the Containment structure under accident loads during an ATWS.

Answer: A

Question Comments: Answer A is CORRECT per the EP Bases. Answer B is INCORRECT because Containment Average Temperature for accident conditions is not established for preventing exceeding Containment Design Pressure of 15psig would entail a saturation temperature of 250 degrees F which is above the design temperature in addition the equipment is designed to operate and is capable of operating under accident conditions. Answer C is INCORRECT because the design basis accident considered for the Containment Temperature is DBA LOCA not ATWS. Preventing exceeding the Design Temperature limit is correct. Answer D is INCORRECT because the design basis accident considered for the Containment Temperature is DBA LOCA not ATWS. Preventing exceeding the Design Pressure limit is incorrect. The limiting component is correct in the Containment Structure. 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

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:

Systems:

Categories:

Task References:

Question Last Revised By: Charles Bell at Thu Jun 16 14:15:44 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

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 14.5 feet.

Division 2 Suppression Pool level is 14.0 feet.

Average Suppression Pool temperature is 135°F.

Which one of the following lists all of the indications that would read erroneously used under these conditions?

A. Division 1 Drywell pressure and Division 1 Suppression Pool level

B. Division 1 Suppression Pool level and Average Suppression Pool temperature

C. Division 1 Suppression Pool level only

D. All of the listed indications are within channel check criteria and are valid

Answer: B

Question Comments: Given there is a 6 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

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: Charles Bell at Thu Jun 16 14:18:53 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

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 is NOT a required action for this event?

A. Close any open Secondary Containment doors and penetrations

B. Initiate one train of Standby Gas Treatment System

C. Perform 06-OP-1T10-M-0001, Secondary Containment Integrity Check

D. Close the 208' containment airlock

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 due to not in modes 1, 2, or 3. 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 required per Conduct of Operations and the High Radiation during Fuel Handling ONEP. Answer D is CORRECT because Technical Specifications identify Primary Containment as NOT required in Modes 4 or 5 and Primary Containment Air Locks are NOT required in modes 4 & 5. 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

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: Charles Bell at Thu Jun 16 14:22:52 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

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

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: Charles Bell at Thu Jun 16 14:26:04 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

Comments:

EB QUESTION: 83 (1.0 Points)

The plant is in Mode 1.

The coil for air supply solenoid valve for PCW SPLY TO SMPL WTR CLRS/CTMT CLRS valve P71-F150 has faulted causing P71-F150 to fail closed.

Containment temperatures are rising. Current containment temperature readings are:

CTMT 119' AZ 45 79°F	CTMT 208' AZ 45 92°F
CTMT 119' AZ 135 80°F	CTMT 208' AZ 135 90°F
CTMT 119' AZ 225 80°F	CTMT 208' AZ 225 90°F
CTMT 119' AZ 315 82°F	CTMT 208' AZ 315 88°F
CTMT 139' AZ 45 87°F	CTMT 240' AZ 45 121°F
CTMT 139' AZ 135 89°F	CTMT 240' AZ 135 118°F
CTMT 139' AZ 225 88°F	CTMT 240' AZ 225 137°F
CTMT 139' AZ 315 84°F	CTMT 240' AZ 315 130°F

Considering requirements for containment temperatures ONLY, which one of the following describes the action(s), if any, required by Technical Specifications/Technical Requirements Manual (TRM) for this condition?

Technical Specification 3.6.1.5 and TRM 6.7.3 are provided.

A. Containment parameters are within allowances, NO action is required.

B. Restore containment temperatures to within limits within 8 hours, AND be in Mode 3 in 12 hours AND Mode 4 in 36 hours.

C. Restore containment temperatures to within limits within 8 hours, AND immediately initiate action to provide a record of the amount by which and the cumulative time the temperature in the affected area exceeded its limit and an analysis to demonstrate continued operability of affected equipment.

D. Restore containment temperatures to within limits within 8 hours, AND immediately initiate action to provide a record of the amount by which and the cumulative time the temperature in the affected area exceeded its limit and an analysis to demonstrate continued operability of affected equipment, AND declare affected equipment inoperable within 4 hours.

Answer: D

Question Comments: The Average temperature of Containment calculated is 95.9 degrees F taking an arithmetical average per TR3.6.1.5. This exceeds the LCO requiring entry into Tech Spec 3.6.1.5 Condition A. The four readings on elevation 240 Ft. are in excess of the TRM Table 6.7.3-1 value of 105 degrees F outside the Drywell. Azimuth 225 is in excess of 30 degrees F of the 105 degrees F limit which means Conditions A and C are in effect. Answer A is INCORRECT because actions are required. Answer B is INCORRECT because placing the plant into Modes 3 and 4 is NOT

required in the times specified. Answer C is INCORRECT because it is incomplete for the Azimuth 225 being > 30 degrees from the limit. Answer D is CORRECT because it includes the correct actions for the conditions given and the appropriate Tech Specs and TRM Conditions. 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 Condition A
2. TRM 6.7.3 Condition A and C
3. (1535/16 = average temperature 95.9°F for these readings)

TrainingPrograms:

Systems:

Categories:

Task References:

Question Last Revised By: Charles Bell at Thu Jun 16 14:29:08 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

Comments:

EB QUESTION: 84 **(1.0 Points)**

The plant was at 100% power when a control rod drift alarm was received.

The operator determined a single rod was drifting out of the core and had drifted from position 12 to position 30.

The operator selected the control rod and has attempted to insert the control rod to position 00.

The control rod could not be inserted due to RCIS malfunction, so Core Flow was reduced to 67 Mlbm/hr.

Cyclops shows NO challenges to Thermal Limits.

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. Operations Management and Reactor Engineering are expected to be notified and this is a level 3 reactivity event.

B. NRC Region IV Headquarters and Reactor Engineering are expected to be notified and this is a level 4 reactivity precursor.

C. Operations Management and NRC Region IV Headquarters are expected to be notified and this is a level 3 reactivity event.

D. Operations Management and Reactor Engineering are expected to be notified and this is a level 4 reactivity precursor.

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 a mispositioned control rod more than one notch beyond its intended position. 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

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:

Systems:

Categories:

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 12:38:21 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

Comments:

EB QUESTION: 85 (1.0 Points)

The plant is at 100% power during a Division 2 work week.

Unknown to anyone, the Division 1 solenoid valve for Inboard Main Steam Isolation Valve (MSIV) B21-F022D had been stuck in the de-energized position since an I&C surveillance the previous week.

Reactor Protection System Motor Generator Set ‘B’ trips due to overload.

What would be the plant response to this event?

A. half scram only

B. full scram from MSIV Limit Switches

C. full scram from High Reactor Pressure

D. full scram from Low Reactor Water Level

Answer: C

Question Comments: A half scram is received when the RPS MG trips on Division 2. MSIV B21-F022D will go closed due to both solenoids being de-energized. This causes a rapid rise in Reactor Pressure causing 1064.7 psig RPS Scram Signal to be achieved. This initiates a full scram. MSIVs will remain open since the High Steam Line flow is not achieved. Answer A is INCORRECT because a full scram occurs. Answer B is INCORRECT because the only one Main Steam Line is affected. Answer C is CORRECT because of the above. Answer D is INCORRECT because the Reactor Water level changes are not as rapid as the pressure transient causing the level change. 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: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-OPS-C7100 Objective: 9 and 26

KA References:

1. 295020 AA2.03: 3.7/3.7
2. Generic 2.4.48: 3.5/3.8

References:

1. FSAR 15.2.4.1.2.2,
2. Tech Spec Bases B 3.3.1.1 function 6

TrainingPrograms:

Systems:

Categories:

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 14:21:57 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

Comments:

EB QUESTION: 86 (1.0 Points)

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

The only available ECCS systems are RHR A and B.

Current parameters are as follows:

Reactor power 0%

Reactor pressure 950 psig being controlled by Main Steam Bypass Valves

Reactor level is - 159 inches wide range, slowly falling at a rate of 1 inch per minute.

Drywell pressure 9.0 psig

Drywell temperature 180 degrees F.

Containment pressure 7.0 psig

Containment temperature 100 degrees F.

Suppression Pool temperature 125 degrees F.

Suppression Pool level 17.0 feet

Both loops of Suppression Pool Cooling are maximized.

Recator Core Isolation Cooling (RCIC), Standby Liquid Control (SLC) and Control Rod Drive (CRD) are injecting at their maximum rates.

Which one of the following will be the alignment of RHR Systems to limit the offsite release rate under these conditions?

A. Initiate both loops of Containment Spray.

B. Initiate one loop of Containment Spray, and leave one loop of Suppression Pool Cooling in service.

C. Leave both loops of Suppression Pool Cooling in service.

D. Align 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 -159 inches and falling all ECCS will be aligned to the RPV

until no longer needed. Suppression Pool temperature while above the point to need two loops in Suppression Pool cooling, RHR is still required for the adequate core cooling issue. Once this has been corrected then 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 keeping the core covered. This is a NEW question. 10CFR41.8/41.10/43.2/43.5

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-3C
4. PSP curve,
5. CSIPL curve, PSTG pgs B-7-7

TrainingPrograms:

Systems:

Categories:

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 13:03:33 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

Comments:

EB QUESTION: 87 **(1.0 Points)**

The plant is at 100% power in the middle of the current operating cycle.

Division 1 work week has just begun.

Work inside the Low Pressure Core Spray (LPCS) Pump room is scheduled to begin tomorrow.

The work is expected to last four days and will include scaffold erection and periodically blocking open the LPCS room door.

The Assistant Operations Manager – Shift (AOM-shift) desires the Shift Managers and Control Room Supervisors to make frequent tours of the area to ensure housekeeping and other standards are being upheld during the course of the work.

What method should be used to communicate this to the Operations personnel being assigned to perform the tours?

A. E-mail

B. Plant mail

C. One Night Order

D. Standing Order

Answer: D

Question Comments: Answers A and B are INCORRECT these two methods are not approved methods of management to convey information to the Shift personnel. Section 6.1.1 and 6.1.2 are the methods described in the Station Operating Orders procedure. These are Night Orders good for One Day and must be reissued if needed. Since Answer C states One Night Order. This means Answer C is INCORRECT. Answer D is CORRECT because Standing Orders are applicable until rescinded. 10CFR 41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00919

Review Status: Reviewed

Difficulty: 1: Fundamental Knowledge or Memory

Objectives:

1. CourseID: GLP-OPS-PROC Objective: 54.2 and 54.6

KA References:

1. 209001 Generic 2.1.15: 2.3/3.0

References:

1. 02-S-01-12 step 6.1.1; 6.1.2 and 6.2.4f

TrainingPrograms:

Systems:

Categories:

Task References:

Question Last Revised By: Charles Bell at Thu Jun 16 14:45:59 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

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 Recirc Pump double downshift.

The Control Room Supervisor (CRS) has received the scram report and entered EP-2 due to reactor water level.

The CRS has just reached step 20, the reactor water level control step, when the Shift Technical Advisor advocates performance of the Reactor Scram Off-Normal Event Procedure (ONEP) subsequent actions to ensure power is going down on the Intermediate Range Monitors (IRMs).

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 other subsequent actions of the Reactor Scram ONEP have been done.

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. Then, the CRS may give directions to insert IRMs.

B. The CRS should complete the primary control loop in EP-2. Then, if NO emergency exists, he may give directions to insert IRMs.

C. The CRS should complete the primary control loop in EP-2. Then, if NO emergency exists, he may direct performance of subsequent actions. Then, all subsequent actions of the Reactor Scram ONEP must be done in order.

D. The CRS should complete the primary control loop in EP-2. The Reactor Operator may insert IRMs, without CRS direction or permission, as soon as he completes all immediate actions of the Reactor Scram ONEP.

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 an SRO to perform subsequent actions of an ONEP out of sequence. 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

Objectives:

1. CourseID: GLP-OPS-PROC Objective: 49.5
2. CourseID: GG-1-LP-RO-EP02 Objective: 4, 9

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:

Systems:

Categories:

Task References:

Question Last Revised By: Charles Bell at Thu Jun 16 14:49:34 CDT 2005

Question History:

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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

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 TR3.3.2.1.3

B. SR 3.3.1.1.5 and SR TR3.3.2.1.3 and SR TR 3.3.2.1.9

C. SR 3.3.1.2.4 and SR TR3.3.2.1.3 and SR 3.3.1.1.5

D. SR 3.3.1.2.1 and SR 3.3.1.2.4 and SR TR3.3.2.1.3 and SR TR3.3.2.1.9

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

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: Charles Bell at Thu Jun 16 14:53:41 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

Comments:

EB QUESTION: 90 **(1.0 Points)**

Severe Accident Procedure 1 (SAP-1) has been entered.

Containment flooding is in progress.

All Suppression Pool level instruments are INOP.

The following conditions exist:

Containment pressure is 1.5 psig

Average Suppression Pool temperature is 104°F.

RCIC pump suction pressure is 16.6 psig

Emergency Procedure attachment 2 has been installed

Reactor Core Isolation Cooling (RCIC) suction has been aligned to suppression pool (E51-F031 open)

RCIC suction from Condensate Storage Tank (CST) E51-F010 is closed

RCIC pump is shut down

What is containment water level for these conditions and how far is containment water level from the minimum level desired for control?

05-S-01-EP-2 Attachment 29 is provided.

CONTAINMENT WATER LEVEL	DELTA TO MIN CTMT WATER LEVEL FOR CONTROL

A. 40.8 FT 7.9 FT below

B. 40.8 FT 21.2 FT below

C. 47.8 FT 0.9 FT below

D. 47.8 FT 14.2 FT below

Answer: B

Question Comments: Target Containment water level for conditions is 62 feet. The calculated actual containment water level is 40.8 ft based on a differential pressure of 15.1 psi. The distractor water level of 47.8 ft is derived from adding 16.6 and 1.5 psi to get 18.1 psid. Answer A is INCORRECT because it used the improper Containment Water Level for the Target water level per SAP-2. Answer B is CORRECT because it has the proper Containment Water Level calculation and is based on the correct Target Containment Water level per SAP-2. Answer C is INCORRECT because it used the improper Containment Water Level for the Target water level per SAP-2 and incorrectly calculated actual Containment Water Level. Answer D is INCORRECT because it incorrectly calculated actual

Containment Water Level. It calcualted based on the correct Target Containment Water level per SAP-2. 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

Objectives:

1. CourseID: GQCOPS-LOQC1 Objective: Attachment 1 page 44 of 133

KA References:

1. 217000 Generic 2.1.25: 2.8/3.1

References:

1. SAP-1 steps 35 - 38
2. 05-S-01-EP-2 attachment 29

TrainingPrograms:

Systems:

Categories:

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 13:12:37 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

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

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: Charles Bell at Thu Jun 16 14:58:47 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

Comments:

EB QUESTION: 92 (1.0 Points)

Containment Spray is required due to Pressure Suppression Pressure concerns.

RHR pump 'A' and 'B' are overridden OFF per EP-2A.

The DC breaker for RHR B/C logic power 72-11B14 immediately trips when the Containment Spray 'B' push button is armed and depressed.

The logic fails to actuate.

H13-P601 alarms RHR B SYS OOSVC (17A-H2) and RHR C SYS OOSVC (17A-H3) are ON.

H13-P601 status light RHR B/C LOGIC PWR FAIL is ON.

Which one of the following describes capability to spray the containment with Containment Spray 'B' under these conditions?

A. Spraying the Containment with RHR 'B' Containment Spray is unavailable from either the Main Control Room or Remote Shutdown Panel under these conditions.

B. RHR pump ‘B’ can be manually started from H13-P601, and CTMT SPR B SPARGER INL VLV E12-F028B can be manually opened by placing its hand switch to OPEN on H13-P601.

C. RHR pump ‘B’ can be manually started from H13-P601, and CTMT SPR B SPARGER INL VLV E12-F028B can be manually opened and will remain open by simultaneously placing its hand switch to OPEN and RHR B SYS SHUTOFF VLV E12-F027B hand switch to CLOSE on H13-P601.

D. RHR pump ‘B’ can be manually started from H13-P601, and CTMT SPR B SPARGER INL VLV E12-F028B can be manually opened by placing its hand switch to OPEN on Remote Shutdown Panel H22-P151.

Answer: A

Question Comments: Answer A is CORRECT because without the logic power Containment Spray Valve cannot be realigned defeating their interlocks E123-F027B and E12-F028B. Answer B is INCORRECT because E12-F028B cannot be opened from H13-P601 due to an interlock between it and E12-F027B to prevent inadvertently spraying Containment during testing. Answer C is INCORRECT because the interlock between E12-F027B and E12-F028B is a local limit switch interlock in the open circuits of the valves. Answer D is INCORRECT because there is NO handswitch on H22-P151 for E12-F028B. 10CFR 41.10/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00924

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-OPS-E1200 Objective: 8.11, 10.1, 10.2, 13.2

KA References:

1. 226001 2.8/2.9

References:

1. ARI 04-1-02-1H13-P601-17A-H2
2. E-1181-40; 42; 44; 68; 69

TrainingPrograms:

Systems:

Categories:

Task References:

Question Last Revised By: Charles Bell at Thu Jun 16 15:01:13 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

Comments:

EB QUESTION: 93 (1.0 Points)

The plant is in a refueling outage.

Core fuel shuffle is in progress when the At the Controls Operator (ACRO) notices a control rod block is NOT received when the refueling SRO states over the head set that a fuel assembly is being lifted out of the core.

Which one of the following describes the allowances for movement of fuel with this condition?

Technical Specification 3.9.1 is provided.

A. All fuel movement must be suspended except to place fuel in a safe condition until the interlock is repaired and retested satisfactorily without exception.

B. Fuel movement is allowed in the Upper Containment Spent Fuel Pool, however, in-vessel fuel movement must be suspended until the interlock is repaired and retested satisfactorily without exception.

C. In-vessel fuel movement must be suspended until a control rod block is inserted and after that, all control rods are verified to be fully inserted for fueled cells, then all fuel movements may resume.

D.

In-vessel fuel movement must be suspended until surveillance requirement SR 3.9.3.1 is verified current and after that, the Control Rod Withdraw pushbutton on H13-P680 is red tagged in the NOT depressed state.

Answer: C

Question Comments: Answer A is INCORRECT because Tech Spec 3.9.1 Condition A has allowances for alternative actions to allow fuel movement until the rod block is repaired. Answer B is INCORRECT because of the same reason is Answer A. Answer C is CORRECT because this is an action allowed by Tech Spec 3.9.1 Condition A Actions A.2. Answer D is INCORRECT because Tech Spec 3.9.1 does not allow for administrative controls for resuming fuel movement. 10CFR 43.2/43.7

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00925

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

Objectives:

1. CourseID: GLP-RF-F1105 Objective: 12

KA References:

1. 234000 A2.01: 3.3/3.7

References:

1. TS 3.9.1 action A
2. SR 3.9.1.1
3. TS Bases B 3.9.1 for Action A.1, A.2.1, A.2.2

TrainingPrograms:

Systems:

Categories:

Task References:

Question Last Revised By: Charles Bell at Thu Jun 16 15:04:03 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

Comments:

EB QUESTION: 94 **(1.0 Points)**

Which one of the following should be evaluated for indications of possible tampering and/or sabotage?

02-S-01-13 is provided.

A. Scaffolding is erected in the Division 2 Switchgear Room (area 25A, elev. 111') and there are workmen going in and out of the room, but you see NO Red Tags in the general vicinity inside the room.

B. You are performing a plant tour and find two remnants of blue and black tie-wraps on the floor inside containment and one stuffed inside a tube of support steel in the RHR 'A' room.

C. Inside the Control Room Standby Fresh Air 'B' (SBFA) room, where workers have been painting walls, components, and ductwork, you find masking tape blocking the instrument air exhaust port of the solenoid valve for SBFA 'B' Outlet Damper Z51-F013, and the tape has been partially painted over. On other dampers with fresh paint, the tape has been removed.

D. You find six empty tubes of Super Glue inside the Division 2 Remote Shutdown Panel (H22-P151) room stuffed behind the pager, and there is a white residue around the controls on H22-P151 and H22-P150.

Answer: D

Question Comments: Answer A is INCORRECT because there are instances where scaffolding is erected to support tagging and future work and the work may be kaowool project that does not use red tags. Answer B is INCORRECT because remnants of different tie-wraps are found in various locations where personnel have dropped them. Blue tie-wraps are used as locking devices but have been found used for other things. Answer C is INCORRECT because painters have continuous projects of preservation of the plant and tape is used to cover things that are desired not to be painted and the final inspection by painting supervisors may not have occurred. Answer D is CORRECT because Super Glue requires chemical

permits and there in no mention of chemical controls permits and the use of glue on controls is not permitted. 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

Objectives:

1. CourseID: GLP-OPS-PROC Objective: 55.4

KA References:

1. K/A Generic 2.1.2: 3.0/4.0

References:

1. 02-S-01-13 sections 5.1; 5.2; 6.2.2

TrainingPrograms:

Systems:

Categories:

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 13:14:29 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

Comments:

EB QUESTION: 95 **(1.0 Points)**

An equipment clearance is being prepared for installation of a design change modification by the Engineering Review (ER) process.

The ER will install a bypass line around the Control Rod Drive suction filters.

Which of the following describes requirements for determining the boundaries of the required equipment clearance?

A. A Drawing Revision Notice (DRN) denoting the design change must be stamped “as-built”, issued, and placed on control room stick files for Operations Sensitive drawings before the clearance can be reviewed.

B. The Responsible Engineer should provide drawings electronically attached to “Folder Documents” to the implementing ER in the Engineering Review Database (ERD) marked as “pending” or provide hard copies to Tagging Group.

C. Configuration control requirements do NOT apply to the drawings required for review, since the design change is NOT installed. Sketches may be used to determine boundary isolation points.

D. All required drawings needed for the clearance review must be approved “as-built” under the controlled drawings section on IDEAS. They may be marked as temporary attached drawings to the base drawing.

Answer: B

Question Comments: Protective tagging requires the use of Controlled Drawings and documents for determination of boundaries. The exception is when developing tagouts to support plant modifications where the controlled drawings have not been issued. This requires verification using ER Support documentation. In addition the preparers should perform walkdowns of the equipment to ensure hazards have been identified and eliminated. Answer A is INCORRECT because plant modifications have allowances to use ER documentation. Answer B is CORRECT for the reasons listed above. Answer C is INCORRECT because drawings used in protective tagging must have some method of controls in place. The ER Drawings are supported by the ER Process. Answer D is INCORRECT because all ER drawings for ERs in process may not be in the IDEAS program. 10 CFR41.10/43.3/43.5

Image Reference: None

Closed Reference Question

Handout Not Required with Exam

QuestionID: GGNS-NRC-00927

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis
Objectives:

1. CourseID: GLP-OPS-TAGPR Objective: 2.6

KA References:

1. Generic 2.1.24: 2.8/3.1

References:

1. 02-S-01-38 step 6.1.7
2. 01-S-06-51 steps 6.12.1, 6.12.2
3. EN-OP-102 section 5.4(1)(b)

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 Mon Jun 20 05:25:01 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

Comments:

EB QUESTION: 96 (1.0 Points)

Which one of the following could be worked without the generation of a work request?

EN-WM-100 and 01-S-07-1 are provided.

A. Replace the Acid Offload pump with a one for one replacement of a new pump. There is sufficient acid in the tanks to support plant operations during the replacement.

B. A Coaxial cable is to be run from the Refuel Floor to the Control Room to support a Plasma screen in the Control Room for monitoring Refuel Floor Operations.

C. A Load Control Center has tripped that jeopardizes continued plant operation and a spare circuit breaker is needed to replace the damaged feeder breaker.

D. An additional telephone line is required to be added to the Technical Support Center from a junction box inside the Control Room envelope.

Answer: D

Question Comments: Answer A is INCORRECT because this work would require generation of a tag out. To obtain a tag out a work request must be generated to describe the scope of the work to ensure adequate boundaries are established. Answer B is INCORRECT because this will involve the opening of Primary and Secondary Containment penetrations and the Control Room Envelope. These penetrations are controlled and require documentation. This means a work request would need to be generated to track the penetrations. Answer C is INCORRECT because this is Priority 1 Expedited Work Order. Work may begin without the work order but a work request is generated. Answer D is CORRECT because this is considered TOOL POUCH MAINTENANCE per the definitions in 01-S-07-1 and EN-WM-100. Tool Pouch Maintenance does not require a work request. 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

Objectives:

1. CourseID: GLP-OPS-PROC Objective: 23.1; 24.1; 24.2

KA References:

1. 2.2.19: 2.1/3.1

References:

1. EN-WM-100 section 5.2.1 and page 11 of 19 definition of Tool Pouch Maintenance.
2. 01-S-07-1 section 6.4.1; 6.4.2

TrainingPrograms:

Systems:

Categories:

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 13:16:38 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

Comments:

EB QUESTION: 97 (1.0 Points)

A Temporary Alteration will be used to temporarily install specialized test equipment in place of a permanent plant recorder for testing activities.

The testing will last one week.

Which one of the following is NOT responsible for approval of the installation of the Temporary Alteration?

A. Control Room Supervisor

B. Shift Manager

C. Manager, System Engineering

D. Assistant Manager, Operations -Shift

Answer: C

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 C 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: 2: Comprehension or Analysis

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

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 13:18:19 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

Comments:

EB QUESTION: 98 (1.0 Points)

Which one of the following batch releases would NOT require a discharge permit approved by Operations personnel?

Procedures 01-S-08-11; 01-S-08-12; TRM 6.11.1 are provided.

A. Discharge of residual water in the condenser water boxes with samples taken were analyzed at < 1 x 10^{-7} microcuries/ml from principle gamma emitters.

B. Discharge of water from the CST dike with samples taken were analyzed at 1 x 10^{-6} microcuries/ml from principle gamma emitters.

C. Draining of SSW 'A' basin with samples taken were analyzed at < 1 x 10^{-7} microcuries/ml from principle gamma emitters.

D. Rain water collected in the Main Transformer berm area that has had a chemistry check declaring the absence of oil or chemicals.

Answer: D

Question Comments: Answer A is INCORRECT because draining of a system or component (Condenser Water Boxes) requires documentation on a discharge permit per 01-S-08-12 section 6.6.1h & 6.6.7a. Answer B is INCORRECT because CST water (be it rain water or other) must meet the limits of TRM Table 6.11.1-1 which is 5x10-7 µci/ml. 1x10-6 and #181;ci/ml is greater than the limit. Therefore permit required. Answer C is INCORRECT because SSW basin water requires documentation on a discharge permit per 01-S-08-12 section 6.6.1d & 6.6.3b. Answer D is CORRECT because rainwater in transformer berms that has been confirmed as having no oil or chemicals may be drained per 01-S-08-12

section 6.2.1. 10CFR 41.10/41.13/43.4/43.5

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00930

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

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 sections 6.1, 6.4.3
2. 01-S-08-12 sections 6.2, 6.4.7, 6.6, 6.6.3, 6.6.7
3. TRM 6.11.1

TrainingPrograms:

1. Senior Reactor Operator Training Program

Systems:

Categories:

1. Administrative Requirements

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 13:23:28 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

Comments:

EB QUESTION: 99 **(1.0 Points)**

Plant startup is in progress.

All Reactor Recirc pump parameters had been normal while in slow speed.

Reactor Recirc pump 'A' has just been shifted to fast speed.

Now, the following indications are present on H13-P680:

Alarm RECIRC PMP A SEAL STG FLO HI/LO (P680-3A-B5) is ON

Seal Staging flow at C11-FI-R020A is higher than normal.

Recirc Pump 'A' Seal Cavity #1 is 1050 psig

Recirc Pump 'A' Seal Cavity #2 is 1000 psig

Seal Temperatures as indicated on B33-TJR-R601 are lower than normal.

Which one of the following describes the status of the Recirc Pump seals and the appropriate action for this condition?

04-1-01-B33-1 Section 4.2 is provided.

A. This is indication of Seal #1 failure. Transfer Recirc pump'A' back to slow speed, verify the Recirc seal restages or notify System Engineering to implement seal monitoring, and then shift Recirc pump 'A' back to fast speed.

B. This is an indication of Seal Staging Orifice plugging. Transfer Recirc pump 'A' back to slow speed, verify the Recirc seal restages or notify System Engineering to implement seal monitoring, and then shift Recirc pump 'A' back to fast speed.

C. This is indication of Seal #1 failure. Notify System Engineering to implement seal monitoring and continue with startup.

D. This is an indication of Seal Staging Orifice plugging. Notify System Engineering to implement seal monitoring and continue with startup.

Answer: A

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 A is CORRECT because this is indicated in the answer. Answers B & D are INCORRECT because this is not indications of Seal Staging Orifice plugging. Answer C is INCORRECT because it does not indicate the need to shift pump speeds to attempt to restage the seals. This is a NEW question. 10CFR 41.10/43.5

Image Reference: None

Open Reference Question

Handout Required with Exam

QuestionID: GGNS-NRC-00931

Review Status: Reviewed

Difficulty: 2: Comprehension or Analysis

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:

Systems:

1. B33: Reactor Recirculation System

Categories:

Task References:

Question Last Revised By: MikeRasch at Mon Jun 20 13:25:42 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

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

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 Mon Jun 20 13:29:02 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
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4. Question used on ExamName: NRC August 2005 - 3 ExamID: NRC-082005-3 Exam Date: 08/12/2005

Comments:

END OF EXAM

Total Number of Questions: 100 Total Point Value: 100.0

Total Exam Question Difficulty: 161.0

Average Exam Question Difficulty: 1.61

Questions with Difficulty Level 1: 39

Questions with Difficulty Level 2: 61

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