UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

ENCLOSURE 1

Examination Report No. 50-400/OL-89-02

Facility Licensee: Carolina Power and Light Company
P. O. Box 1551
Raleigh, NC 27602

Facility Name: Shearon Harris Nuclear Power Plant

Facility Docket No: 50-400

Written examinations and operating tests were conducted at the Shearon Harris Nuclear Power Plant site near New Hill, North Carolina.

Chief Examiner: for Jesse A. Arildsen — 1/8/90 Date Signed

Approved By: Charles A. Casto, Chief — 1/8/90 Date Signed
Operator Licensing Section 2
Division of Reactor Safety

Summary:

Requalification examinations were conducted during the weeks of October 3, 1989, and October 9, 1989.

Written examinations and operating tests were administered to eight Reactor Operators (ROs) and sixteen Senior Reactor Operators (SROs). Of the eight ROs tested, six passed the examination. Of the 16 SROs tested, 14 passed the examination. One of the six crews tested failed the simulator part of the operating examination.

REPORT DETAILS

1. Facility Employees Attending Exit Meeting:

   R. Richey, Site Manager
   C. Hinnant, Plant Manager
   L. Martin, Nuclear Training Manager
   J. Collins, Operations Manager
   A. Powell, HTU Manager
   J. Hudson, Licensee Training Manager
   A. Orton, HTU Staff
   J. Pierce, HTU Staff
   C. Fleming, HTU Staff
   R. Smith, HTU Staff
   P. Mazzola, HTU Staff
   H. Stroup, HTU Staff
   J. Bryan, HTU Staff
   A. Gorrou, Emergency Preparedness

2. Examiners

   *J. A. Arildsen, NRC, Region II
   T. P. Guilfoil, Sonalysts
   B. C. Haagensen, Sonalysts
   I. G. Kingsley, Sonalysts
   *Chief Examiner

3. Exit Meeting

   At the conclusion of the site visit, the examiners met with representatives of the plant staff to discuss the results of the examinations. The following items were addressed:

   <u>Examination Development</u>

   After review of the licensee proposed questions for the written examinations, the examination team made modifications to enhance the examination and format the questions to be more objective.

   The walk-through examination's Job Performance Measures (JPMs) required the rephrasing of several of their associated oral questions in order to enhance clarity. Additionally, a few of the JPMs required corrections due to technical inaccuracies.

   The proposed simulator scenarios required only minor changes prior to administration.

Examination Administration

During the administration of the simulator examinations the evaluators appeared to conduct the scenarios in a well planned, and well ordered manner. As a result, only one minor simulator scenario problem required an on-the-spot amendment to the scenario.

During the administration of the walk-through examinations (JPMs), one minor inconsistency was exhibited in the use of impromptu follow-up questions.

The facility was unable to provide a different Shift Technical Advisor (STA) for each crew. This necessitated that the STAs used during the administration of the simulator examinations were required to be limited in support such as not to display any initiative during the scenarios. They were allowed only to perform as specifically directed since they had prior knowledge of scenario events. This restriction adversely impacts the ability of the NRC examiners to assess the crews' performance as they would normally operate in the control room.

WLOK 6

Enclosure 3

* NRC MASTER COPY *

10/6/89

SHNPP RO
ANNUAL REQUALIFICATION EXAM
PART A WRITTEN EXAM
ANSWER KEY

16.0 pts

1. A09-019

QUESTION: PT. VALUE: ~~1.5~~ 1.0

When the failed PRZ pressure channel is removed from service, which bistable status lights that are not currently lit, should become lit?

ANSWER:

(0.25 pts. each)

- PB456A
- PB456B
- Loop B 0/TEMP DELTA-T, TB422C1
- Loop B 0/TEMP DELTA-T, TB422C2

REFERENCES:

OWP-RP (Rev.2/AC-3)

Scenario Applicability: 13

2. A09-038

QUESTION: PT. VALUE: 1.0

Following removal of PRZ pressure channel 456 from service IAW the OWP, what effect would the Loop "A" protection T_c RTD failing low have on current plant and protection system operation?

ANSWER:

This condition will result in a reactor trip due to meeting the 2/3 coincidence on OT-Delta-T. (1.0 pt.)

REFERENCES:

SD-103 (Rev.4/AC-2)
Technical Specification Table 2.2-1

Scenario Applicability: 13

3. A09-018

QUESTION: PT. VALUE: 1.0

Explain why the difference exists between the three OT-DELTA-T setpoint channels.

ANSWER:

PRZ press channel 456 has failed which causes the "B" loop OT-DELTA-T setpoint to be decreased. (1.0 pt.)

REFERENCES:

SD-103 (Rev.4/AC-2)
Technical Specification Table 2.2-1

Scenario Applicability: 13

4. A09-039

QUESTION: PT. VALUE: 1.5

Currently, PB-456B on TSLB-3 is de-energized, PB-455B and 457B for high RCS pressure are energized yet RCS actual pressure is at or somewhat less than normal operating pressure.

a. If PT-456 had NOT failed, should PB-455B, 456B and 457B be energized? Assume all other plant conditions are normal for existing conditions. (0.5 pts)
b. What do PB-455B, 456B and 457B represent and what is their actuating setpoint? (1.0 pt.)

ANSWER:

a. Yes (0.5 pts.)
b. P-11 interlock (.5) setpoint 2000 psig (.5)

REFERENCES:

1. OWP-RP (Rev.2/AC-3)
2. SD-103 (Rev.4/AC-2)
3. Logics

Scenario Applicability: 13

5. A05-018

QUESTION: PT. VALUE: 1.0

Summarize the effects on plant operation of a continued decrease in main condenser vacuum to atmospheric pressure. Include any resulting trip(s) and interlock(s) effected, the direct cause(s) of the trip(s) and interlock(s), and the final heat removal method in your answer.

ANSWER:

Main turbine will trip on low condenser vacuum. (0.25 pts.)
The turbine trip will cause a reactor trip. (0.25 pts.)
The steam dumps will be unavailable due to the low vacuum. (0.25 pts.)
So decay heat will have to be removed via SG PORVs. (0.25 pts.)

REFERENCES:

SD-131.01 (Rev.2)
SD-103 (Rev.4)
SD-126.01 (Rev.1/AC-3)
AOP-012 (Rev.3/AC-1)
APP-ALB-20 (Rev.2)
Logics

Scenario Applicability: 13

6. A02-028

QUESTION: PT. VALUE: 1.0

T_{avg} and PRZ level have been showing a decreasing trend. Identify the event and briefly explain how the event is causing the transient.

ANSWER:

The decrease in condenser vacuum (0.25 pts.) is reducing secondary efficiency. (0.25 pts.) The increase in steam flow to maintain constant turbine load (0.25 pts.) has reduced T_{ave} and PRZ level is following T_{ave}. (0.25 pts.)

REFERENCES:

SD-100.03 (Rev.1)

Scenario Applicability: 13

7. A04-010

QUESTION: PT. VALUE: 1.0

Assume that "C" Steam Generator Steam Flow Channel FT-494 immediately failed offscale low under the current plant conditions. Briefly describe the initial response of "C" main feedwater regulating valve, including a description of how or why the control circuitry generates the initial response.

ANSWER:

No effect. (0.5 pts.) FT-495 is selected for level control. (0.5 pts.)

REFERENCES:

SD-126.02 (Rev.1)

Scenario Applicability: 3, 11, 13, 18

8. A04-014

QUESTION: PT. VALUE: 1.0

Describe the initial response of the "C" SG feed reg valve if steam pressure transmitter PT-496 were to fail low at this time, including a description of how or why the control circuitry generates the initial response.

ANSWER:

The valve would initially shut. (0.2 pts.) Loss of density compensation to the steam flow calculation will cause a decrease in the steam flow input (0.4 pts.) SGWLC will see feed flow greater than steam flow and shut the feed reg valve. (0.4 pts.)

REFERENCES:

SD-126.02 (Rev.1)

Scenario Applicability: 11, 13, 18

9. A02-022

QUESTION: PT. VALUE: 1.0

While natural circulation is developing it is necessary to establish a driving head which is a result of differential temperature. What would you expect the Delta-T value to reach immediately following a trip from full power?

ANSWER:

Approximately normal full power Delta-T, (60°F ± 5°F). (1.0 pt.)

REFERENCES:

Fluid Flow Manual Chapter 5

Scenario Applicability: 10

10. A07-013

QUESTION: PT. VALUE: 1.5

Has natural circulation been established? Justify your answer by listing the parameters used to verify natural circulation per procedure, and their current value and/or trend as necessary to support your answer.

ANSWER:

- Yes (0.25 pts.)
- Subcooling is > 10/20°F (60±5°F) (0.25 pts.)
- Steam pressure is stable/decreasing (~1100 psig) (0.25 pts.)
- RCS hot leg temperature is stable/decreasing (~595°F) (0.25 pts.)
- Core exit TCs are stable/decreasing (~598°F) (0.25 pts.)
- RCS cold leg temperature is trending to saturation temperature for steam pressure (~555°F) (0.25 pts.)

REFERENCES:

EOP-EPP-004 (Rev.3)

Scenario Applicability: 10

11. A03-014

QUESTION: PT. VALUE: 1.0

What means of PRZ pressure control are currently available for pressure reduction?

ANSWER:

PRZ PORVs, (0.5 pts.) and Aux. Spray. (0.5 pts.)

REFERENCES:

SD-100.03 (Rev.1)
EOP-EPP-004 (Rev. 3)

Scenario Applicability: 10

12. A02-023

QUESTION: PT. VALUE: 1.0

To what temperature could the RCS currently be cooled down, while maintaining system pressure constant without exceeding any of the cooldown pressure/temperature limits?

ANSWER:

500°F (+10°F, -0°F) (1.0 pt.) (OR 515°F IF WITH ASSUMED PRESSURE OF 2300#)

REFERENCES:

Technical Specification Figure 3.4-2
GP-007 (Rev. 3/AC-3)

Scenario Applicability: 10

13. A01-019

QUESTION: PT. VALUE: 1.0

Emergency boration is in progress. Identify the symptom that required emergency boration.

ANSWER:

Two or more rods not fully inserted following a Rx trip (1.0 pt.) (Since no indication on DRPI)

REFERENCES:

AOP-002 (Rev.5)

Scenario Applicability: 10

14. A05-013

QUESTION: PT. VALUE: 0.5

The steam driven AFW pump started automatically due to:

a. Lo-Lo level in 2/3 SGs
b. "A" and "B" Main Feed pump breakers open
c. 6.9 KV bus 1A-SA or 1B-SB de-energized
d. The steam driven AFW pump should not have automatically started

ANSWER:

c. 6.9 KV buss 1A-SA or 1B-SB de-energized (0.5 pts.)

REFERENCES:

SD-137 (Rev.1)

Scenario Applicability: 10

15. A05-014

QUESTION: PT. VALUE: 1.0

a. What caused the "B" motor driven AFW pump recirc valve to shut?

b. What is the reason for this automatic feature?

ANSWER:

a. Due to the opposite train emergency bus being de-energized. (0.5 pts.)

b. To ensure adequate AFW flow to the SGs (0.5 pts.)

REFERENCES:

SD-137 (Rev.1)
AOP-025 (Rev.4)

Scenario Applicability: 10

16. A07-002

QUESTION: PT. VALUE: 0.5

Which of the following EDG trips is NOT bypassed during the current EDG Mode of operation ?

a. High jacket cooling temperature trip
b. High vibration trip
c. Overspeed
d. Low lube oil pressure trip

ANSWER:

c. Overspeed (0.5 pts.)

REFERENCES:

SD-155.01 (Rev.0/AC-1)

Scenario Applicability: 1, 4, 10, 16, 17, 20, 21

WK 1 PART A
RO

~~WK 2~~

To: Mr. Jesse Arildsen, NRC Region II

From: Jim Pierce, CP&L

Subject: Weeks one and two LOR witten exam comments

WEEK I

★ A02-023 Answer "unless assume a stated pressure" Justification is that during the exam, ERFIS was not responding accurately. Our answer was based on an ERFIS input. Therefore, an answer would be accepted provided that it agrees with an assumed stated pressure.

2300# ⇒ 515 °F

★ A09-019 1.0pt not 1.5pts

B02-029 "Q" It is stated that: "Manual action may be required to restart safeguards equipment following a loss of offsite power after SI reset." Specifically, which equipment may need to be restarted or realigned?

"A" (4 required .25 pts ea.)
RHR Pumps
CNMT Spray Pumps
RAB Emergency Exhaust
CNMT Fan Coolers to Low Speed
S-2 Fans (Primary Shield Fans)
S-4 Fans (Rx Support Cooling)

B03-053 Delete "...during the recirculation phase." This was stated in the question.

WEEK II

B02-029 Part B answer, The SR detectors must be manually re-energized(.75)

Jesse, if you have any comments or questions with regards to these requests, please call me as (919)-362-2638. Thank you!

* MASTER *
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10/6/89

SHNPP RO
ANNUAL REQUALIFICATION EXAM
PART B WRITTEN EXAM
ANSWER KEY

13 pts

1. B01-030

QUESTION: **PT. VALUE: 1.0**

A normal plant heatup is in progress. Mode 4 has just been entered and actions to draw a bubble in the PRZ have begun. The operator will be able to tell that a bubble is forming in the PRZ when:

A. Stable pressure with stable letdown flow
B. Increase pressure with increased charging flow
C. Stable pressure with increased letdown flow.
D. Pressure with decreased charging flow decrease

ANSWER:

C

REFERENCES:

GP-002, Normal Plant Heatup (Rev. 3/AC-3)

1.0/1.0

2. B02-029

QUESTION: PT. VALUE: 1.0

It is stated that:
"Manual action may be required to restart safeguards equipment following a loss of offsite power after SI reset."
Specifically, which equipment is this referring to? *
(Assume, initial SI was due to Hi Containment Press which is ~~reached~~ 17 psig)
JdWL 10/6/92

ANSWER:

(0.25 pts each) (4 REQUIRED)
-RHR pumps
-Containment fan coolers to low speed (started on high speed)
-RAB emergency exhaust fans (E-6)
-Cont. spray pumps
- S-2 FANS
- S-4 FANS

REFERENCES:

SD-155.02, Emergency Sequencing System (Rev.1, AC/001)
EOP-EPP-008, SI Termination (Rev.3)
EOP-User's Guide

* (Equipment may need to be restarted or realigned.)

3. B01-065

QUESTION: PT. VALUE: 0.5

The plant is operating at 100% power with the rod control system in automatic. The COMPUTER ALARM ROD DEV/SEQ NIS PWR RANGE TILTS annunciator alarms. Investigation reveals that rod B6 in Control Bank A is indicating 13 steps lower than its group step counter. There are no other alarms, and all other parameters are normal. Which of the following is the proper response in this situation?

A. Prepare a PRI-1 work request and contact I & C.
B. Immediately commence inserting the control rods to shut down the reactor and be in hot standby within 6 hours.
C. Trip the reactor.
D. Transfer rod control to manual to ensure no further rod motion until direction by Reactor Engineering.

ANSWER:

D

REFERENCES:

AOP-001, Rev.4

4. B04-007

QUESTION: PT. VALUE: 1.0

A plant cooldown is in progress. All but one RCP has been stopped to assist in the cooldown. RHR has been placed in service per procedure. Component Cooling Water heat exchanger outlet temperature has increased to 115°F since placing RHR in service. Seal injection flow is lost to the running RCP. Explain why the RCP should or should not be tripped immediately due to the loss of seal injection under the above stated plant conditions.

ANSWER:

RCPs should not be tripped because immediate RCP trip is only required if CCW HX outlet temp. is greater than 120°F (0.5 pts) when RCS temperature is less than 400°F (0.5 pts).

REFERENCES:

AOP-018 (Rev.3)
OP-111 (Rev.4)

5. B01-067

QUESTION: PT. VALUE: 0.5

A reactor trip has occurred without SI being required, and the operators are in the process of ensuring that the primary system stabilizes at no-load conditions. A check of the rod bottom lights and rod position indicators shows that one control rod has not fully inserted. The operators continue on with the Reactor Trip Response procedure without taking action concerning the stuck control rod. Which of the following statements explains why no action is taken at this point concerning the stuck rod?

A. Xenon is building into the core following the trip and inserting sufficient (negative) reactivity to compensate for the stuck rod.
B. The core is designed for adequate shutdown margin with one rod stuck out.
C. Samarium is building into the core following the trip and inserting sufficient (negative) reactivity to compensate for the stuck rod.
D. Relative shutdown reactivity is a major concern during a reactor trip only in conjunction with a rapid RCS cooldown.

ANSWER:

B. The core is designed for adequate shutdown margin with one rod stuck out.

REFERENCES:

1. EOP-EPP-004, Rev.3
2. ERG Background, Reactor Trip Response

6. B02-017

QUESTION: **PT. VALUE: 1.5**

How much subcooling is required during a natural circulation cooldown in each of the following cases? Also, briefly explain why subcooling is required in each case.

a. All CRDM fans running and computer unavailable. (.75 pts)

b. No CRDM fans running and computer available. (.75 pts)

ANSWER:

a. $70^{o}F$ (.5)

b. $110^{o}F$ (.5)

The subcooling in each case precludes void formation in the vessel head. (.5)

REFERENCES:

EOP-EPP-005 (Rev.4)
WOG ERG Background Document (ES-0.2, HP Rev.1)

7. B03-014

QUESTION: **PT. VALUE: 0.5**

A loss of coolant accident is in progress. Control room operators are performing FRP-C.2, "Response to Degraded Core Cooling" in response to a MAGENTA path on the CORE COOLING CSF Status Tree. Which one of the following statements is correct concerning transitions out of this procedure.

The operators would immediately transition to:

a. FRP-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, if the HEAT SINK CSFST indicates a RED path condition.
b. FRP-P.1, RESPONSE TO IMMINENT PRESSURIZER THERMAL SHOCK if RCS INTEGRITY CSFST indicates a MAGENTA path condition.
c. FRP-S.2, RESPONSE TO LOSS OF CORE SHUTDOWN, if the SUBCRITICALITY CSFST indicates a YELLOW path condition.
d. FRP-J.1, RESPONSE TO HIGH CONTAINMENT PRESSURE, if the CONTAINMENT CSFST indicates a MAGENTA path condition.

ANSWER:

a. (0.5 pts.)

REFERENCES:

Emergency Operating Procedure User's Guide (Rev. 1/AC-1)

8. B03-080

QUESTION: **PT. VALUE: 0.5**

PRZ PORV 1RC-116 is stuck slightly open. High pressure safety injection is maintaining RCS pressure at 1910 psig, and the PRZ vapor space temperature is 630°F. The PRT level and pressure have been stabilized at 80% and 35 psig by draining and spraying. A Control Room Operator notes that TI-463, the PORV line temperature indicator, is reading approximately 390°F. He states that he believes there is a problem with the indication, since it is not reading as he expects for the current conditions. select the response that most closely corresponds to your assessment of the Control Room Operator's statement.

a. Agree. The indication should be offscale high.
b. Agree. The indication should be offscale low.
c. Agree. The indication should be reading approximately 280°F.
d. Agree. The indication should be reading approximately 260°F
e. Disagree. The indication is reading as expected

ANSWER:

c. Agree. The indication should be reading approximately 280°F. (0.5 pts.)

REFERENCES:

ASME Steam Tables, Mollier Diagram

9. B04-012

QUESTION: PT. VALUE: 1.0

The primary is being drained to mid-loop level in preparation for SG tube inspections. Level is currently between 36 in. and 67 in. below the reactor vessel flange and is being monitored every 30 minutes by observation of the standpipe. "B" RHR train is operating in the shutdown cooling mode with a total flow through the train of 3000 gpm.

What item(s) in this situation would you need to correct in order to be in compliance with station procedures?

ANSWER:

-Standpipe watch should be continuous and level logged every 15 minutes. (0.5)
-RHR loop flow should be reduced to ≤1500 gpm. (0.5)

REFERENCES:

OP-111, Residual Heat Removal System, (Rev.4)

10. B04-028

QUESTION: PT. VALUE: 0.5

The Control Room Operators are responding to a LOCA. The STA monitoring the CSFST's observes that all Core Exit TCs are reading greater than 1200°F. The SCO enters the correct FRP. Which one of the following methods will be the most effective in restoring the critical safety function associated with these symptoms ?

a. Rapidly depressurize the secondary to cool and depressurize the RCS.
b. Reduce RCS pressure by opening all available RCS vent paths to containment.
c. Start all RCP's
d. Establish high-pressure SI flow.

ANSWER:

d. Establish high-pressure SI flow. (0.5)

REFERENCES:

EOP-FRP-C.1 (Rev.3)
WOG ERG Background Documents

11. B-NRC-02

QUESTION: **PT. VALUE: 1.0**

1. If the plant process computer is not available, the operators should determine subcooling margin using:

   a. The average of all core exit thermocouple readings on the Inadequate Core Cooling Monitor (ICCM).

   b. The single highest reading core exit thermocouple reading on the Inadequate Core Cooling Monitor (ICCM).

   c. The active loop(s) wide range hot leg temperature indicators (TI-413, 423, 433).

   d. The inactive loop(s) wide range hot leg temperature indicators (TI-413, 423, 433).

2. What Tech Spec(s) (if any) is (are) applicable if the process computer is inoperable?

ANSWER:

1. b.

2. Reference OWP-ERFIS-01 for applicable T.S.

(.007 pt. each)

1. T.S. 3.3.3.4 (MET TOWER)
2. T.S. 3.3.3.6 ITEM 11 (S.G.)
3. T.S. 3.3.3.6 ITEM 14 (FIRE SAFETY)
4. T.S. 3.4.6.1.b (SUMP LEAKAGE)
5. T.S. 4.1.3.1.1 (RPDM)
6. T.S. 4.1.3.2 (RPI)
7. T.S. 4.2.1.1 (AFD MONITOR)

REFERENCES:

1. EOP User's Guide (6.2) (Rev. 1/AC-1)
2. Tech Specs

12. B05-002

QUESTION: PT. VALUE: 0.5

With regard to operation of the Auxiliary Control Panel (ACP) after a transfer to the ACP, which of the following conditions is true?

a. The automatic start signals for the Turbine Driven Auxiliary Feedwater Pump (TDAFW) and the Motor Driven Auxiliary Feedwater Pumps (MDAFW) are blocked.
b. Only the auto start for the TDAFW pump is blocked.
c. Only the auto start for the MDAFW pump is blocked.
d. Both the TDAFW and MDAFW pumps retain their auto start signals.

ANSWER:

b. (1.0 pt.)

REFERENCES:

Abnormal Operating Procedure (AOP-004)
"Safe Shutdown in case of a fire or control room inaccessibility." (Rev. 5/AC-2)

13. B07-012

QUESTION: **PT. VALUE: 1.0**

The "B" DG is operating and synchronized to 1B-SB. The DG trips and the "DIESEL GENERATOR B TRIP" annunciator is received on the MCB. The operator in the DG building reports that the "LOSS OF BTH GEN POT CKS TRIP" AND "GEN POT CKT LOSS OF FUSE" annunciators have alarmed.

a. After fuses have been replaced to correct the problem, how is the DG trip signal reset? (0.5 pts.)
b. At the time the DG output breaker (126) tripped, the normal supply breaker (125) also tripped. As a result of this, what response, if any, will the DG have when the trip signal is reset? (0.5 pts.)

ANSWER:

a. By manually depressing the Emergency Stop Reset (0.5)
b. The DG will auto start (0.5 pts.)

REFERENCES:

SD-155.01, Emergency Diesel Generator System, (Rev. 0/AC-1)

14. B07-022

QUESTION: **PT. VALUE: 0.5**

At 10:00 a.m. the plant is operating at 100% capacity when onsite and offsite electrical power is lost and an SI signal is simultaneously generated. The Control Operators verify a reactor trip, turbine trip, and both AC emergency buses energized by the emergency diesels. Two minutes into the event, the Turbine-Driven AFW Pump is found to be operating, but the Motor-Driven AFW Pumps are not.

Should the Motor-Driven AFW Pumps have started automatically by this time? Why?

a. No. Load Program "A" will not start the Motor-Driven AFW Pumps for another 10 seconds.
b. No. Load Program "B" does not start the Motor-Driven AFW Pumps for a loss of offsite power with SI.
c. Yes. Load Program "B" should have started the Motor-Driven AFW Pumps by now.
d. Yes. Load Program "A" should have started the Motor-Driven AFW Pumps by now.

ANSWER:

c. Yes. Load Program "B" should have started the Motor-Driven AFW Pumps by now.

REFERENCES:

SD-155.02, Rev.1/AC-1

15. B09-008

QUESTION: **PT. VALUE: 1.0**

The plant is operating at 5% power. Maintenance is to be performed on the instrument string for Source Range detector N-32. What action can the Reactor Operator take that will prevent a reactor trip in the event that the Source Range inadvertently became energized during the maintenance work?

ANSWER:

Taking the "LEVEL TRIP" switch to the "BYPASS" position. (1.0 pt.)

REFERENCES:

SD-105, Excore Nuclear Instrumentation (Rev.4/AC-1)

16. B-NRC-04

QUESTION: PT. VALUE: 0.5

Which controlled keys are located in the Control Room and are to be used for emergency situations only?

a. Keys affording access to plant Vital Areas.

b. Keys to the containment personnel hatch.

c. Keys to locked High Radiation Areas.

d. Keys necessary to transfer the Diesel Generator from Remote to Local, to implement AOP-004.

ANSWER: a

REFERENCES: OMM-001 (REV 4)

Deleted per J. Anderson 10/6/89

17. B-NRC-05

QUESTION: **PT. VALUE: 1.0**

You are on site and off-shift when personnel accountability is implemented following declaration of a Site Emergency. Which one of the following is NOT your responsibility?

a. Quickly notifying the Control Room of your location.

b. Expeditiously going to your emergency facility or Shelter/Assembly area.

c. Reporting to your Foreman/Supervisor.

d. Keeping your Foremen/Supervisor informed of your location after the initial assembly.

ANSWER: a.

REFERENCES: PEP-382 (REV 2)

WK 1 RO
PART B

To: Mr. Jesse Arildsen, NRC Region II

From: Jim Pierce, CP&L

Subject: Weeks one and two LOR witten exam comments

WEEK I

A02-023 Answer "unless assume a stated pressure" Justification is that during the exam, ERFIS was not responding accurately. Our answer was based on an ERFIS input. Therefore, an answer would be accepted provided that it agrees with an assumed stated pressure. 2300# ⇒ 515°F

A09-019 1.0pt not 1.5pts

★ B02-029 "Q" It is stated that: "Manual action may be required to restart safeguards equipment following a loss of offsite power after SI reset." Specifically, which equipment may need to be restarted or realigned?

"A" (4 required .25 pts ea.)
RHR Pumps
CNMT Spray Pumps
RAB Emergency Exhaust
CNMT Fan Coolers to Low Speed
S-2 Fans (Primary Shield Fans)
S-4 Fans (Rx Support Cooling)

B03-053 Delete "...during the recirculation phase." This was stated in the question.

WEEK II

B02-029 Part B answer, The SR detectors must be manually re-energized(.75)

Jesse, if you have any comments or questions with regards to these requests, please call me as (919)-362-2638. Thank you!

WK 1 RO/SRO
B

10/24/89

To: Mr: Jesse Arildsen

From: Jim Pierce, CP&L

Subject: Week One Part-B-RO/SRO Exam Answer Key

B-NRC-02, change answer for part 2 "Reference OWP-ERFIS-01 for applicable T.S." to

1. T.S. 3.3.3.4 (Met Tower)
2. T.S. 3.3.3.6 item 11 (Subcooling Margin)
3. T.S. 3.3.3.6 item 14 (Prz Safeties)
4. T.S. 3.4.6.1.b (Sump Leakage)
5. T.S. 4.1.3.1.1 (Rod Position Deviation Monitor)
6. T.S. 4.1.3.2 (Rod Position Indication)
7. T.S. 4.2.1.1 (AFD Monitor)

Each part is worth .07pts and either the T.S. number or noun name description is acceptable.

Jim Pierce

MASTER
WK 2

Enclosure 3

SHNPP RQ
ANNUAL REQUALIFICATION EXAM
PART A WRITTEN EXAM
ANSWER KEY

1. A01-028

QUESTION: **PT. VALUE: 0.5**

When the No. 1 Seal Leakoff Isolation Valve for the "C" RCP is shut, what is the expected response of seal injection flows?

a. They would return to approximately the same (normal) values as before the seal failure.
b. "C" RCP will go to zero while A and B assume equal amounts.
c. Values would stay as presently indicated.
d. Only manual action of HC-186.1 will affect seal injection flows.

ANSWER:

a. They would return to approximately the same values as before the seal failure. (0.5 pt.)

REFERENCES:

SD-100.01 (Rev. 0)

Scenario Applicability: 15

2. A04-008

QUESTION: **PT. VALUE: 1.0**

What immediate action(s) is/are required to respond to the primary plant conditions as they are shown?

ANSWER:

Shut the No. 1 seal leakoff isolation valve 1CS-437 (1.0 pt.).

REFERENCES:

AOP-018 (Rev. 3)

Scenario Applicability: 15

3. A04-011

QUESTION: PT. VALUE: 1.0

a. With the current problem on "C" RCP, within what time frame must "C" RCP be tripped?
b. Below what power level must the plant be to avoid a Rx trip when the RCP is tripped?

ANSWER:

a. Within 30 minutes of detecting the seal failure (0.5 pts.)
b. Below P-8 Setpoint (49% power) (0.5 pts.)

REFERENCES:

AOP-018 (Rev. 3)

Scenario Applicability: 15

4. A01-027

QUESTION: **PT. VALUE: 1.0**

Explain the reason for the initial VCT level increase.

ANSWER:

The excessive seal leakoff due to the failed No. 1 seal is being directed to the VCT (via the seal water Hx) (1.0 pt.)

REFERENCES:

SD-107 (Rev. 2)

Scenario Applicability: 15

5. A09-017

QUESTION: **PT. VALUE: 1.0**

List the present MCB indications that identify the failure of PT-447. (Four required for full credit).

ANSWER: (Any 4, 0.25 pts. each)

PI-447 meter = 0 psig

PB-447E (P-13 bistable status light) is deenergized

C-7A permissive status lights are energized

C-7B permissive status lights are energized

ALB-13-2-2 (SR High Flux at shutdown blocked) annunciators are energized

ALB-13-2-3 (SR Loss of Detector Volts) annunciator is energized

REFERENCES:

APP-ALB-013 (Rev. 2)
SD-126.01 (Rev. 1/AC-3)

Scenario Applicability: 15

6. A09-016

QUESTION: PT. VALUE: 1.0

Why have C-7A and C-7B permissive status lights energized?

ANSWER:

PT-447 failed low generated the bistables due to rate (> 10%/120 sec and > 40%/120 sec) (1.0 pt.)

REFERENCES:

SD-103 (Rev. 4/AC-2)

Scenario Applicability: 15

7. A09-015

QUESTION: **PT. VALUE: 1.0**

Why are the "SOURCE RANGE HIGH FLUX LVL AT SHUTDOWN ALARM BLOCKED" and "SOURCE RANGE LOSS OF DETECTOR VOLTAGE" annunciators illuminated, which is contrary to dark board concept?

ANSWER:

The loss of Impulse pressure channel 447 deenergized its associated P-13 bistable which unblocked these annunciator lights. (1.0 pt.)

REFERENCES:

PCR-1866
APP-ALB-013 windows 2-2 & 2-3 (Rev.2/AC-1)

Scenario Applicability: 15

8. A05-016

QUESTION: **PT. VALUE: 0.5**

If a reactor trip occurred with present plant conditions, which statement below best describes the response of the steam dump system?

a. The turbine trip controller would position the steam dump valves to control T_{avg} at 557°F.
b. The turbine trip controller would position the steam dump valves to control T_{avg} to within 5°F of T_{ref}.
c. The steam dump valves would be controlled by PK-464.1 (STM HDR DMP PRESS CONT) according to its present setpoint.
d. The steam dump system would not actuate due to the present status of the system.

ANSWER:

c. (0.5 pts.)

REFERENCES:

SD-126.01 (Rev. 1/AC-3)

Scenario Applicability: 15

9. A04-009

QUESTION: **PT. VALUE: 0.5**

With present plant conditions, if "B" steam generator feedwater flow transmitter FT-486 failed to 3.4×10^6 lbm/hr, the <u>initial</u> response of "B" steam generator main feedwater regulating valve would be to:

a. Move in the shut direction due to feedwater > steam flow
b. Move in the shut direction due to steam flow > feedwater
c. Move in the open direction due to steam flow > feedwater
d. Would not be affected. Would remain in present position.

ANSWER:

a. (0.5 pts.)

REFERENCES:

SD-126.02 (Rev. 1)

Scenario Applicability: 15

10. A09-045

QUESTION: **PT. VALUE: 1.0**

When I&C was dispatched to place the bistables in a tripped condition, the I&C technician inadvertently tripped the bistables associated with PT-456. The bistables were tripped in the following order:

PB 456A, High-Pressure Reactor Trip
PB 456C, Low-Pressure Reactor Trip
PB 456D, Pressirizer Low-Pressure SI
TB 422C1, OT-Delta-T Trip

"REACTOR TRIP PRESSURIZER SI" is the first out annunciator. Why didn't tripping of the Low-Pressure Reactor Trip bistable cause the reactor trip?

ANSWER:

Power level was below the P-7 interlock which blocks the low pressure reactor trip. (1.0 pt.)

REFERENCES:

SD-103 (Rev.4)

Scenario Applicability: 20

11. A09-013

QUESTION: PT. VALUE: 1.0

Why has the "GROSS FAILED FUEL DET TROUBLE" annunciator alarmed? Failed fuel is not suspected.

ANSWER:

Low neutron activity due to sample flow being isolated on SI. (1.0 pt.)

REFERENCES:

APP-ALB-026-2-1 (Rev. 2)
SD-117 (Rev.3/AC-1)

Scenario Applicability: 1, 3, 4, 14, 17, 20, 21

12. A03-040

QUESTION: PT. VALUE: 1.0

Explain how and why the the loop 1 OT-Delta-T setpoint changed.

ANSWER:

The PT-455 failure resulted in a reduction of the Loop 1 OT-Delta-T setpoint (0.5 pts.) as a result of a penalty being generated due to the pressure input decrease. (0.5 pts.)

REFERENCES:

SD-103 (Rev. 4/AC-2)
Technical Specifications Table 2.2-1
SD-100.03 (Rev. 1)
PLS

Scenario Applicability: 20

13. A04-001

QUESTION: PT. VALUE: 1.0

FR-154B "RCP Seal Leakoff" has been cycling. Explain why this is occurring.

ANSWER:

Seal return valves 1CS-470/472 shut on the SI/Phase A, (0.5 pts.) and the seal return relief valve is cycling. (0.5 pts.)

REFERENCES:

SD-107 (Rev.2)
SFD, S-1303 (Rev.7)

Scenario Applicability: 1, 2, 4, 9, 14, 16, 17, 20

14. A02-041

QUESTION: PT. VALUE: 0.5

Which one of the following correctly describes what is necessary to reset SI and prevent a re-actuation of automatic SI?

a. SI timer timed out
b. All reactor trip and bypass breakers open
c. SI times timed out and all reactor trip and bypass breakers open
d. No, automatic SI actuation setpoint exceeded and SI timer timed out

ANSWER:

c. SI timer timed out and all reactor trip and bypass breakers open. (0.5 pts.)

REFERENCES:

SD-103 (Rev. 4/AC-2)
Logics

Scenario Applicability: 1, 2, 4, 9, 17, 20, 21

15. A05-038

QUESTION: PT. VALUE: 1.0

What condition (signal) generated the motor-driven AFW pump start?

ANSWER:

SI (1.0 pt.)

REFERENCES:

SD-137 (Rev. 1)
Logics

Scenario Applicability: 20

16. A03-004

QUESTION: **PT. VALUE: 1.0**

What operator action(s) would be required, (if any), to determine PRZ PORV Isolation Valve position from the main control room ?

ANSWER:

Energize 480V electrical busses A1 & B1 (1.0 pt)

REFERENCES:

SD-100.3 (Rev. 1)
EOP-Path-1 (Rev. 5)

Scenario Applicability: 1, 2, 4, 14, 16, 17, 20, 21

17. A01-049

QUESTION: PT. VALUE: 1.0

Alternate charging valve 1CS-480 indicates open and yet its control switch is positioned to shut. State whether the actual valve position is open or shut and explain your decision.

ANSWER:

Open. (0.25 pts.) 1CS-480 failed open on a loss of air to CNMT. (0.75 pts.) (Which occurred as a result of the phase A CNMT isolation.)

REFERENCES:

SD-107 (Rev.2)
AOP-017, Att.1 (Rev. 3/AC-1)

Scenario Applicability: 1, 3, 4, 17, 20, & 21

18. A01-005

QUESTION: **PT. VALUE: 1.0**

The controller for PCV-145, the letdown line pressure control valve, is in automatic and its controller output indicates that the PCV is receiving a shut signal. Justify the existence of a shut signal to PCV-145.

ANSWER:

Letdown isolated (on Phase A) so the pressure in the letdown line is less than the controller's setpoint. (0.5 pts.) So the controller is sending a shut signal to try to raise the pressure in the line. (0.5 pts.)

REFERENCES:

SD-107 (Rev. 2)

Scenario Applicability: 1, 2, 3, 4, 14, 16, 17, 20, & 21

19. A02-046

QUESTION: PT. VALUE: 1.0

Should automatic steamline isolation have occurred? Justify your answer.

ANSWER:

No. (0.2 pts.) CNMT pressure has remained below 3 psig, (0.4 pts.) and steamline pressure has remained above 601 psig. (0.4 pts.)

REFERENCES:

SD-103 (Rev.4/AC-2)
SD-126.01 (Rev.1/AC-3)
Westinghouse Functional Diagram 108D831, Sh8

Scenario Applicability: 2, 3, 4, 20

NJNSTEG
WK 2

Master

SHNPP RO
ANNUAL REQUALIFICATION EXAM
PART B WRITTEN EXAM
ANSWER KEY

1. B00-011

QUESTION: PT. VALUE: 0.5

While performing a manual dose calculation, which of the following should be used for wind speed and direction ? (Assume these are the only values available)

a. National weather service at RDU reports 20 mph from 180°
b. Corporate weather center reports 15 mph from 175°
c. Wind sock at chlorine storage building indicates 9 mph from 180°
d. ERFIS

ANSWER:

d. (0.5 pts.)

REFERENCES:

PEP-341, Manual Dose Calculation (Rev.5)

2. B-NRC-07

QUESTION: **PT. VALUE: 1.0**

During a plant emergency, a reactor operator volunteers to enter an area where radiation levels are estimated to be over 150 Rem/hr to shut a valve to save a piece of vital station equipment. What is the maximum exposure he could be authorized to receive by the Station Emergency Coordinator (10CFR20 limits apply) during the entry into the accident area.

Exposure history:
Age: 48
Lifetime dose: 21 Rem
Current year: 4 Rem
Current quarter: 2 Rem
Current admin limit extended to 2500 mrem

ANSWER: 25 Rem

REFERENCES:

PEP-371
10CFR20

3. B01-019

QUESTION: PT. VALUE: 0.5

Select the correct answer from the choices provided in the following paragraph concerning the effects of moderator temperature changes on MTC.

As moderator temperature decreases, the change in _____ per degree change in the moderator decreases. This causes a _____ reactivity effect per degree temperature change, which causes MTC to get _____ negative.

A. enthalpy, smaller, less
B. density, smaller, more
C. enthalpy, larger, more
D. density, smaller, less

ANSWER:

D.

REFERENCES:

Cycle 2 Curve book, Figures:
- C-2-1 (Rev.0)
- C-2-2 (Rev.0)
- C-2-3 (Rev.0)
ASME Steam Tables

4. B03-091

QUESTION: **PT. VALUE: 0.5**

While responding to a small-break LOCA, it is determined that a RED path exists on the RCS Integrity CSFST. The SRO enters the appropriate Function Restoration Procedure and check for possible sources of excessive RCS cooldown and then checks if SI can be terminated. Current RCS subcooling does not support SI termination, but it does support the starting of an RCP. None are currently running.

RCP operation under these conditions will decrease the likelihood of pressurized thermal shock by: (1.0)

A. Providing mixing of the incoming SI water and reactor coolant.
B. Providing additional RCS subcooling.
C. Providing more decay heat removal to the S/G's.
D. Collapsing upper vessel head void formation.

ANSWER: A (.5pt.)

REFERENCES:

1. EOP-FRP-P.1, Rev. 3
2. ERG Background Document (FR-P.1, Rev. 1A)

5. B01-053

QUESTION: PT. VALUE: 1.0

The reactor operator reports that there is a demand for automatic rod withdrawal, but the Rod Control System is not responding. The RO is directed to place rod control in manual and attempt to withdraw rods. The rods move outward in manual. What two possible rod withdrawal interlocks may have existed to cause this scenario? (Name and setpoints required, if applicable.)

ANSWER:

C-5, (Turbine power <15%) (0.5 pts)
C-11, (Bank D rod withdrawal stop) (0.5 pts)

REFERENCES:

SD-104 (Rev.5/AC-1)

6. B01-072

QUESTION: PT. VALUE: 0.5

The Control Room Operators are responding to a reactor trip with SI when a red condition is indicated on the Subcriticality CSFST. The operators re-verify that the reactor and turbine have tripped, ensure that the AFW pumps are running, initiate emergency boration, verify all ESFAS equipment has actuated properly, and begin checking for possible causes of the loss of subcriticality. Their investigation reveals the following:

- RCS temperature and pressure are decreasing rapidly.
- The pressure in SG "A" is decreasing uncontrollably.
- The MSIV's and MSIV bypass valves for all SG's are closed.
- Reactor makeup pumps are off.
- Reactor makeup water valve is shut.
- BTRS bypass valve is in the bypass position (to VCT)
- Boric acid batch tank outlet valve is shut.

Based on these indications, what is the probable cause of this loss of subcriticality?

a. Excessive cooldown caused by an unisolated ruptured SG.
b. Excessive cooldown caused by an unisolable faulted SG.
c. Inadvertent dilution flow path alignment.
d. Loss of secondary heat sink.

ANSWER:

b. Excessive cooldown caused by an unisolable faulted SG.

REFERENCES:

1. EOP-FRP-S.1, Rev.3/AC-1
2. WOG ERG Background Document (FR-S.1, Rev.1)

7. B01-076

QUESTION: PT. VALUE: 0.5

The reactor has failed to automatically trip when required and cannot be manually tripped. The turbine is tripped, the AFW pumps are running, and emergency boration is in progress. Pressurizer pressure is 2385 psig. Both pressurizer PORV's are open, but their associated block valves are closed. A Reactor Operator succeeds in opening one block valve and reduces pressurizer pressure to below 2135 psig. Under these conditions, which of the following is the main reason for reducing pressure?

a. Prevent the rapid overpressurization transient expected with most ATWS events.
b. Minimize primary-to-secondary leakage in case of the most limitimg ATWS event, a SGTR, until other recovery actions can be taken.
c. Allow enough borated water to flow into the RCS to ensure the addition of negative reactivity to the core.
d. Begin a slow, controlled cooldown and depressurization, thereby minimizing positive reactivity feedback via a negative MTC.

ANSWER:

c. Allow enough borated water to flow into the RCS to ensure the addition of negative reactivity to the core.

REFERENCES:

1. EOP-FRP-S.1, Rev.3/AC-1
2. WOG ERG Background Document (FR-S.1, Rev.1A)

8. B01-079

QUESTION: PT. VALUE: 0.5

The Control Room Operators are verifying that the appropriate automatic actions have occurred following a reactor trip with SI. Containment pressure has exceeded the containment spray automatic actuation setpoint, so the operators verify that containment spray has initiated and that the containment isolation phase "B" valves have closed. At this point, the applicable procedure instructs the operators to trip the RCP's.

What would be the probable consequences of NOT tripping the RCP's at this point?

a. Forced circulation will make the containment high pressure transient more severe by sustaining an excessive energy release rate to containment.
b. The RCP motor bearings will overheat because Component Cooling Water flow to them has been isolated by the Phase "B" actuation.
c. Radioactive materials will be released from containment because a potential release path remains unisolated.
d. The RCP's will trip on low pressure because it is assumed that if containment spray actuates, an RCS depressurization will be in progress.

ANSWER:

b. The RCP motor bearings will overheat because Component Cooling Water flow to them has been isolated by the Phase "B" actuation.

REFERENCES:

1. EOP-PATH-1,Rev.5
2. ERG Background Document (E-0, Rev.1)

9. B-NRC-06

QUESTION: PT. VALUE: 0.5

The 1A-SA and 1B-SB busses are deenergized due to loss of AC power following a station blackout. EPP-001 has been entered and RCP seals have been locally isolated. When power is restored to one AC emergency bus, the first required action is to:

a. Check ESW system operation.

b. Verify equipment loaded on the AC emergency bus.

c. Stabilize steam generator pressure.

d. Check containment status.

ANSWER: c

REFERENCES: EOP-EPP-001 (REV. 4)

10. B01-063

QUESTION: **PT. VALUE: 1.0**

A review of plant operations and shutdown margin determinations shows the following sequence of events:

0800 July 21 - Plant in Mode 3
1200 July 21 - Shutdown margin determination
1600 July 22 - Shutdown margin determination
0900 July 23 - Shift foreman gave permission for approach to criticality
1400 July 23 - Reactor critical operating in Mode 2
1500 July 23 - Shutdown margin determination
2300 July 23 - Shutdown margin determination
0400 July 24 - Plant operating in Mode 1
1400 July 24 - Shutdown margin determination

Based on the sequence of events described above, determine whether or not any Technical Specification violations have occurred. Justify your answer.

ANSWER:

A violation has occurred. (.25) On July 23 a shutdown margin determination should have been completed within 4 hours prior to achieving criticality. (.75)

REFERENCES:

1. Tech Specs. 3.1.1.1 & 3.1.1.2, Amend.7
2. GP-004, Rev.3/AC-1

11. B04-005

QUESTION: PT. VALUE: 1.0

The Reactor Coolant System (RCS) is being maintained @ 325°F by steam dumps, and RCS pressure is being maintained at 350 psig. The 'A' RHR pump has been running for greater than five minutes and chemistry has reported that the RCS boron concentration is 2010 ppm and that the RHR A-train boron concentration is 1990 ppm. Explain why the 'A' RHR train should not be placed in service under these conditions.

ANSWER:

The boron concentration of the RHR system should be equal to or greater than that of the RCS before allowing flow from the RHR system to enter the RCS. (1.0 pt.)

REFERENCES:

OP-111 (Rev.4)

12. B-NRC-11

QUESTION: POINTS 0.5

Which one of the following have to be completed or reviewed to ensure that regulatory and administrative requirements for plant heatup have been met?

a. Shift Orders.

b. NRC Event Notification Worksheet.

c. Equipment Inoperable Record Book.

d. Caution Tag Log.

ANSWER: c

REFERENCES: GP-002 (REV. 3)

13. B02-011

QUESTION: **PT. VALUE: 1.0**

The interlocks between RCS pressure and the RHR Pumps from RCS loops suction valves, allow the valves to be open when RCS pressure is less than _____ psig and automatically close the valves when RCS pressure exceeds _____ psig.

ANSWER:

363 (± 5), 700 (0.5 pts. each)

REFERENCES:

1. System Description (SD-111) Residual Heat Removal System (Rev.2)

14. B02-026

QUESTION: PT. VALUE: 0.5

Which one of the following HVAC systems is NOT affected by a CRAV isolation signal.

a. RAB Switchgear Room HVAC
b. Electrical Equipment Protection Room HVAC
c. Fuel Handling Building HVAC
d. Computer and Communications Rooms HVAC

ANSWER:

c. Fuel Handling Building HVAC (0.5 pts)

REFERENCES:

SD-173, Control Room HVAC System (Rev.2)

15. B02-044

QUESTION: PT. VALUE: 1.0

The plant is operating at 100% power with all control systems in automatic. VCT HIGH-LOW LEVEL alarm is received. The CO observes that VCT indicated level has failed high at 100% and that letdown is being diverted to the RHT. The CO is informed that local VCT level is reading less than 5% and is decreasing. What automatic actions failed to occur due to LT-115 failing high?

ANSWER:

When actual level decreased to less than 20%, automatic makeup should have occurred.(0.25) At less than 5%, LCV-115B and LCV-115D should have automatically opened to supply CSIP from the RWST. (0.75)

REFERENCES:

1. AOP-003, Rev.3
2. APP-ALB-007-4-3, Rev.2

16. B02-028

QUESTION: PT. VALUE: 0.5

The AMSAC modification will send a trip and/or actuation signal to:

a. Trip the turbine and actuate main steam isolation and auxiliary feedwater isolation circuitry.
b. Trip the reactor, trip the turbine and actuate AFW.
c. Trip the turbine and actuate main feedwater isolation.
d. Trip the turbine and actuate AFW

ANSWER:
d (.5 pts.)

REFERENCES:

APP-ALB-017-1-1 (Rev.2/AC-1)

17. B05-033

QUESTION: PT. VALUE: 1.0

The plant is operating at full power when the "SG B NR LVL/SP HI/LO DEV" annunciator alarms. The same alarms for the "A" and "C" SG's follow momentarily. The operators observe SG levels decreasing. They also observe the main feed reg. valves opening and CNDBSTR speed increasing. No CNDST, CNDBSTR, HD or FW pumps have tripped.

a. What is the cause of the decreasing SG levels? (0.4 pts) (Note: Assume there are no leaks or breaks in any of the associated piping)
b. What two actions should be taken to prevent a trip on low-low SG level? (0.6 pts)

ANSWER:

a. FW (or CNDST) recirc valve opened (0.4 pts)
b. Reduce turbine load (0.3 pts) or isolate the recirc line (0.3 pts)

REFERENCES:

AOP-010 (Rev.4/AC-1)

18. B10-001

QUESTION: **PT. VALUE: 1.0**

During power operations the operator observes that VCT level is trending up on ERFIS and that letdown is diverting to the HUT's. The "CCW SURGE TANK HIGH-LOW" alarm annunciates and subsequently the operating CCW pump trips. Based on this information, what is the most likely location of the CCW leak?

ANSWER:

Seal water heat exchanger. (1.0 pts.)

REFERENCES:

AOP-014, Loss of Component Cooling Water (Rev. 4)

19. B-NRC-09

QUESTION: **PT. VALUE: 1.0**

1. What is the root cause of the periodic OT delta T spikes on loop 3 (432)?

2. What two actions have been prescribed to avert a runback?

ANSWER:

1. Flow stratification in the reactor core discharge flow path that periodically affects Loop 3 delta T. (0.5)

2. a. Bypass loop 3 delta T. (0.25)

   b. Ensure T_{avg} is reduced by 1.0 to 1.3° below T_{ref} whenever maintenance is performed that affects one of the redundant loops (412 or 422). (0.25)

REFERENCES:

Standing Orders; "2/21/89 Loop 3 OT Delta T Spikes"

20. B03-060

QUESTION: PT. VALUE: 0.5

During a plant heatup the PRZ pressure is 900 psia when a PRZ PORV spuriously opens. Steam is relieved to PRT at 50 psia. Which description of the relieved steam is correct ?

a. Saturated steam at 320°F
b. Superheated steam at 320°F
c. Saturated steam at 540°F
d. Superheated steam at 540°F

ANSWER:

b. (1.0 pt.)

REFERENCES:

ASME Steam Tables

WEEK 1

Enclosure 3

* NRC MASTER COPY *

10/4/89

SHNPP SRO
ANNUAL REQUALIFICATION EXAM
PART A WRITTEN EXAM
ANSWER KEY

1. A09-019

QUESTION: PT. VALUE: ~~1.5~~ 1.0

When the failed PRZ pressure channel is removed from service, which bistable status lights that are not currently lit, should become lit?

ANSWER:

(0.25 pts. each)

- PB456A
- PB456B
- Loop B O/TEMP DELTA-T, TB422C1
- Loop B O/TEMP DELTA-T, TB422C2

REFERENCES:

OWP-RP (Rev.2/AC-3)

Scenario Applicability: 13

2. A09-038

QUESTION: **PT. VALUE: 1.0**

Following removal of PRZ pressure channel 456 from service IAW the OWP, what effect would the Loop "A" protection T_c RTD failing low have on current plant and protection system operation?

ANSWER:

This condition will result in a reactor trip due to meeting the 2/3 coincidence on OT-Delta-T. (1.0 pt.)

REFERENCES:

SD-103 (Rev.4/AC-2)
Technical Specification Table 2.2-1

Scenario Applicability: 13

3. A09-018

QUESTION: PT. VALUE: 1.0

Explain why the difference exists between the three OT-DELTA-T setpoint channels.

ANSWER:

PRZ press channel 456 has failed which causes the "B" loop OT-DELTA-T setpoint to be decreased. (1.0 pt.)

REFERENCES:

SD-103 (Rev.4/AC-2)
Technical Specification Table 2.2-1

Scenario Applicability: 13

4. A09-039

QUESTION: PT. VALUE: 1.5

Currently, PB-456B on TSLB-3 is de-energized, PB-455B and 457B for high RCS pressure are energized yet RCS actual pressure is at or somewhat less than normal operating pressure.

a. If PT-456 had NOT failed, should PB-455B, 456B and 457B be energized? Assume all other plant conditions are normal for existing conditions. (0.5 pts)
b. What do PB-455B, 456B and 457B represent and what is their actuating setpoint? (1.0 pt.)

ANSWER:

a. Yes (0.5 pts.)
b. P-11 interlock (.5) setpoint 2000 psig (.5)

REFERENCES:

1. OWP-RP (Rev.2/AC-3)
2. SD-103 (Rev.4/AC-2)
3. Logics

Scenario Applicability: 13

5. A02-028

QUESTION: PT. VALUE: 1.0

T_{avg} and PRZ level have been showing a decreasing trend. Identify the event and briefly explain how the event is causing the transient.

ANSWER:

The decrease in condenser vacuum (0.25 pts.) is reducing secondary efficiency. (0.25 pts.) The increase in steam flow to maintain constant turbine load (0.25 pts.) has reduced T_{ave} and PRZ level is following T_{ave}. (0.25 pts.)

REFERENCES:

SD-100.03 (Rev.1)

Scenario Applicability: 13

6. A02-029

QUESTION: **PT. VALUE: 1.0**

Assume the plant subsequently trips from this transient. Following the trip, if all plant conditions and equipment are restored to normal/operational status with the exception of the "B" CSIP still being out of service, would a reactor startup be allowed? Justify your answer.

ANSWER:

No. (0.25 pts.) 2 CSIP's are required to be operable and tech spec 3.0.4 does not allow entering a mode unless the LCO is met without reliance on the action statement. (0.75 pts.)

REFERENCES:

Technical Specifications:
- 3.1.2.4
- 3.0.4

Scenario Applicability: 13

8. A02-054

QUESTION: PT. VALUE: 1.5

OST-1026 is performed and identifies a total RCS leakage rate of 6.0 gpm. 5.8 gpm of this is determined to be leakage into the "A" Cold Leg Accumulator.

a. Is this a violation of Tech. Specs.? Justify your answer. (0.75 pts.)
b. The STA recommends that you declare "A" CLA inoperable due to inleakage and that you shut the breaker for 1SI-246 and close the discharge isolation valve (1SI-246). Under current plant conditions, once power is restored to the valve, can you accomplish what the STA recommends from the MCB? Justify your answer. (0.75 pts.)

ANSWER:

a. Yes, (.25) leakage > 5.0 gpm past the accumulator discharge check valves. (.5)
b. No, (.25) primary pressure must be less than 2000 psig (P-11). (.5)

REFERENCES:

1. Tech. Spec. 3.4.6.2.f
2. SD-103 (Rev. 4/AC-2)
3. Logics

Scenario Applicability: 13

9. A07-010

QUESTION: PT. VALUE: 1.5

Based on the present electrical system status in what mode must the plant be placed to comply with tech. specs.? Include the time frame(s) associated with the required actions.

ANSWER:

Within 1 hour action must be taken to place the unit in: (0.5 pts.)
Hot Shutdown within the next 6 hours, (0.5 pts.) and
Cold Shutdown within the subsequent 24 hours. (0.5 pts.)

REFERENCES:

Technical Specifications:
- 3.8.1.1
- 3.0.3

Scenario Applicability: 10

10. A07-013

QUESTION: PT. VALUE: 1.5

Has natural circulation been established? Justify your answer by listing the parameters used to verify natural circulation per procedure, and their current value and/or trend as necessary to support your answer.

ANSWER:

- Yes (0.25 pts.)
- Subcooling is > 10/20°F (60±5°F) (0.25 pts.)
- Steam pressure is stable/decreasing (~1100 psig) (0.25 pts.)
- RCS hot leg temperature is stable/decreasing (~595°F) (0.25 pts.)
- Core exit TCs are stable/decreasing (~598°F) (0.25 pts.)
- RCS cold leg temperature is trending to saturation temperature for steam pressure (~555°F) (0.25 pts.)

REFERENCES:

EOP-EPP-004 (Rev.3)

Scenario Applicability: 10

11. A01-019

QUESTION: PT. VALUE: 1.0

Emergency boration is in progress. Identify the symptom that required emergency boration.

ANSWER:

Two or more rods not fully inserted following a Rx trip (1.0 pt.) (Since no indication on DRPI)

REFERENCES:

AOP-002 (Rev.5)

Scenario Applicability: 10

12. A05-013

QUESTION: PT. VALUE: 0.5

The steam driven AFW pump started automatically due to:

a. Lo-Lo level in 2/3 SGs
b. "A" and "B" Main Feed pump breakers open
c. 6.9 KV bus 1A-SA or 1B-SB de-energized
d. The steam driven AFW pump should not have automatically started

ANSWER:

c. 6.9 KV buss 1A-SA or 1B-SB de-energized (0.5 pts.)

REFERENCES:

SD-137 (Rev.1)

Scenario Applicability: 10

13. A05-014

QUESTION: PT. VALUE: 1.0

a. What caused the "B" motor driven AFW pump recirc valve to shut?

b. What is the reason for this automatic feature?

ANSWER:

a. Due to the opposite train emergency bus being de-energized. (0.5 pts.)

b. To ensure adequate AFW flow to the SGs (0.5 pts.)

REFERENCES:

SD-137 (Rev.1)
AOP-025 (Rev.4)

Scenario Applicability: 10

14. A07-002

QUESTION: PT. VALUE: 0.5

Which of the following EDG trips is NOT bypassed during the current EDG Mode of operation?

a. High jacket cooling temperature trip
b. High vibration trip
c. Overspeed
d. Low lube oil pressure trip

ANSWER:

c. Overspeed (0.5 pts.)

REFERENCES:

SD-155.01 (Rev.0/AC-1)

Scenario Applicability: 1, 4, 10, 16, 17, 20, 21

15. A00-007

QUESTION: PT. VALUE: 1.0

Classify the event in progress IAW the Emergency Plan, based on current plant conditions. Briefly explain the basis for the classification level.

ANSWER:

Unusual Event. (0.5 pts.) Due to loss of offsite power. (0.5 pts.)

REFERENCES:

PEP-101 Flowpath (Rev.3/AC-1)

Scenario Applicability: 10

WK 1 SRO
PART A

To: Mr. Jesse Arildsen, NRC Region II

From: Jim Pierce, CP&L

Subject: Weeks one and two LOR witten exam comments

WEEK I

A02-023 Answer "unless assume a stated pressure" Justification is that during the exam, ERFIS was not responding accurately. Our answer was based on an ERFIS input. Therefore, an answer would be accepted provided that it agrees with an assumed stated pressure.

2300# ⇒ 515 °F

★ A09-019 1.0pt not 1.5pts

B02-029 "Q" It is stated that: "Manual action may be required to restart safeguards equipment following a loss of offsite power after SI reset." Specifically, which equipment may need to be restarted or realigned?

"A" (4 required .25 pts ea.)
RHR Pumps
CNMT Spray Pumps
RAB Emergency Exhaust
CNMT Fan Coolers to Low Speed
S-2 Fans (Primary Shield Fans)
S-4 Fans (Rx Support Cooling)

B03-053 Delete "...during the recirculation phase." This was stated in the question.

WEEK II

B02-029 Part B answer, The SR detectors must be manually re-energized(.75)

Jesse, if you have any comments or questions with regards to these requests, please call me as (919)-362-2638. Thank you!

* MASTER *
NRC
COPY

SHNPP SRO
ANNUAL REQUALIFICATION EXAM
PART B WRITTEN EXAM
ANSWER KEY

10/6/89

1. B01-065

QUESTION: PT. VALUE: 0.5

The plant is operating at 100% power with the rod control system in automatic. The COMPUTER ALARM ROD DEV/SEQ NIS PWR RANGE TILTS annunciator alarms. Investigation reveals that rod B6 in Control Bank A is indicating 13 steps lower than its group step counter. There are no other alarms, and all other parameters are normal. Which of the following is the proper response in this situation?

A. Prepare a PRI-1 work request and contact I & C.
B. Immediately commence inserting the control rods to shut down the reactor and be in hot standby within 6 hours.
C. Trip the reactor.
D. Transfer rod control to manual to ensure no further rod motion until direction by Reactor Engineering.

ANSWER:

D

REFERENCES:

AOP-001, Rev.4

2. B04-007

QUESTION: **PT. VALUE: 1.0**

A plant cooldown is in progress. All but one RCP has been stopped to assist in the cooldown. RHR has been placed in service per procedure. Component Cooling Water heat exchanger outlet temperature has increased to 115°F since placing RHR in service. Seal injection flow is lost to the running RCP. Explain why the RCP should or should not be tripped immediately due to the loss of seal injection under the above stated plant conditions.

ANSWER:

RCPs should not be tripped because immediate RCP trip is only required if CCW HX outlet temp. is greater than 120°F (0.5 pts) when RCS temperature is less than 400°F (0.5 pts).

REFERENCES:

AOP-018 (Rev.3)
OP-111 (Rev.4)

3. B01-067

QUESTION: PT. VALUE: 0.5

A reactor trip has occurred without SI being required, and the operators are in the process of ensuring that the primary system stabilizes at no-load conditions. A check of the rod bottom lights and rod position indicators shows that one control rod has not fully inserted. The operators continue on with the Reactor Trip Response procedure without taking action concerning the stuck control rod. Which of the following statements explains why no action is taken at this point concerning the stuck rod?

A. Xenon is building into the core following the trip and inserting sufficient (negative) reactivity to compensate for the stuck rod.
B. The core is designed for adequate shutdown margin with one rod stuck out.
C. Samarium is building into the core following the trip and inserting sufficient (negative) reactivity to compensate for the stuck rod.
D. Relative shutdown reactivity is a major concern during a reactor trip only in conjunction with a rapid RCS cooldown.

ANSWER:

B. The core is designed for adequate shutdown margin with one rod stuck out.

REFERENCES:

1. EOP-EPP-004, Rev.3
2. ERG Background, Reactor Trip Response

4. B01-070

QUESTION: PT. VALUE: 0.5

The reactor fails to trip when required. The Control Room Operators take actions as per appropriate procedure(s) and obtain the required plant/system/component responses, except that the reactor is still not tripped, and emergency boration cannot be initiated because of blockage in the boration flow paths. All Power Range channels indicate 3%, and the startup rate is zero on both Intermediate Range channels. Which of the following describes the correct operator actions under these conditions AND the primary reason for taking these actions?

a. Return to the procedure and step in effect. Power is less than 5% and the IR startup rate is zero.
b. Allow the RCS to heat up while continuing efforts to establish emergency boration. The heatup will insert negative reactivity.
c. Go to FRP-S.2. This is required by the Subcriticality CSFST based on current reactor conditions.
d. Maintain RCS temperature stable while continuing efforts to establish emergency boration. Stable temperatures preclude positive reactivity insertion by cooldown.

ANSWER:

b. Allow the RCS to heat up while continuing efferts to establish emergency boration. The heatup will insert negative reactivity.

REFERENCES:

1. EOP-FRP-S.1, Rev.3/AC-1
2. WOG ERG Background Document (FR-S.1, Rev.1)

5. B03-014

QUESTION: **PT. VALUE: 0.5**

A loss of coolant accident is in progress. Control room operators are performing FRP-C.2, "Response to Degraded Core Cooling" in response to a MAGENTA path on the CORE COOLING CSF Status Tree. Which one of the following statements is correct concerning transitions out of this procedure.

The operators would immediately transition to:

a. FRP-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, if the HEAT SINK CSFST indicates a RED path condition.

b. FRP-P.1, RESPONSE TO IMMINENT PRESSURIZER THERMAL SHOCK if RCS INTEGRITY CSFST indicates a MAGENTA path condition.

c. FRP-S.2, RESPONSE TO LOSS OF CORE SHUTDOWN, if the SUBCRITICALITY CSFST indicates a YELLOW path condition.

d. FRP-J.1, RESPONSE TO HIGH CONTAINMENT PRESSURE, if the CONTAINMENT CSFST indicates a MAGENTA path condition.

ANSWER:

a. (0.5 pts.)

REFERENCES:

Emergency Operating Procedure User's Guide (Rev. 1/AC-1)

6. B-NRC-02

QUESTION: **PT. VALUE: 1.0**

1. If the plant process computer is not available, the operators should determine subcooling margin using:

   a. The average of all core exit thermocouple readings on the Inadequate Core Cooling Monitor (ICCM).

   b. The single highest reading core exit thermocouple reading on the Inadequate Core Cooling Monitor (ICCM).

   c. The active loop(s) wide range hot leg temperature indicators (TI-413, 423, 433).

   d. The inactive loop(s) wide range hot leg temperature indicators (TI-413, 423, 433).

2. What Tech Spec(s) (if any) is (are) applicable if the process computer is inoperable?

ANSWER:

1. b.

2. ~~Reference OWP-ERFIS-01 for applicable T.S.~~

REFERENCES:

1. EOP User's Guide (6.2) (Rev. 1/AC-1)
2. Tech Specs

   T.S. ~~4.3.3.4~~ 3.3.3.4

   T.S. 3.3.3.6 (Item 11)

   T.S. 3.3.3.6 (Item 14)

   T.S. 3.4.6.1.b

   T.S. 4.1.3.1.1

   T.S. 4.1.3.2

   T.S. 4.2.1.1

10/6/85

7. B02-005

QUESTION: **PT. VALUE: 1.5**

With Unit 1 at 100% power the A RHR Pump is disassembled for maintenance. The Reactor Operator notes that one of the RWST suction motor operated valves on the B RHR pump is closed and cannot be reopened. Assuming it will take 10 hours to return the A RHR Pump to service and the suction motor operated valve on the B RHR Pump cannot be opened:

1. What ~~is~~ are all the actions required IAW Tech. Specs.? ~~(1.0)~~ (0.20)
2. What are the reportability requirements? ~~(0.25)~~ (0.30)
3. Classify the event. ~~(0.25)~~ (0.30)

ANSWER:

1. a. Within one hour initiate action to place unit in a mode in which the specification does not apply. ~~(0.25)~~ (0.30)
   b. Be in at least Hot Standby within the next 6 hours. ~~(0.25)~~ (0.30)
   c. Be in at least Hot Shutdown within the following 6 hours. ~~(0.25)~~ (0.30)
   ~~d. Be in at least Cold Shutdown within the subsequent 24 hours. (0.25)~~ Deleted 10/6/89

2. One hour notification ~~(0.25)~~ (0.30)

3. ~~Alert~~ UNUSUAL EVENT ~~(0.25)~~ (0.30)

Only one train of RHR is required in Hot or Cold S/D (Mode 4+5) One HSD is ~~actual~~ achieved, CSD is not required.

REFERENCES:

1. Technical Specifications (3.0.3) Applicability (Amendment-7)
2. Administrative Procedure (AP-615) NRC Reporting Requirements (Rev.4/AC-7)
3. Plant Emergency Procedure (PEP-101) (Rev. 3/AC-1)

(Equipment may need to be restarted or realigned.) 10/6/89

8. B02-029

QUESTION: PT. VALUE: 1.0

It is stated that:
"Manual action may be required to restart safeguards equipment following a loss of offsite power after SI reset."
Specifically, which equipment is this referring to?
(Assume, initial SI was due to Hi Containment Press which is ~~reached~~ 17 psig)

10/6/89

ANSWER:
(4 of 6 required)
(0.25 pts each)
-RHR pumps
-Containment fan coolers to low speed (started on high speed)
-RAB emergency exhaust fans (E-6)
-Cont. spray pumps
- S-2 FANS
- S-4 FANS

REFERENCES:

SD-155.02, Emergency Sequencing System (Rev.1, AC/001)
EOP-EPP-008, SI Termination (Rev.3)
EOP-User's Guide

9. B05-048

QUESTION: PT. VALUE: 0.5

The plant has experienced a main steamline break. Because of difficulties in closing the MSIV's, two of the SG's have blown dry. RCS temperature is less than 230°F and decreasing. EOP-EPP-014 "Faulted Steam Generator Isolation" is being implemented. SI flow is still being supplied to the RCS and AFW is being supplied to the SG's. All RCP's have been stopped. Which of the following concerns should be given the highest priority?

a. A crack could develop in the reactor vessel wall due to a pressurized thermal shock event.
b. Injection of ECCS accumulator nitrogen into the RCS is imminent.
c. A significant reduction in the heat sink capacity has occurred due to two SG's being blown dry.
d. The loss of thermal driving head in the dry SG's will reduce the amount of natural circulation flow, due to stagnant coolant loops.

ANSWER:

a. A crack could develop in the reactor vessel wall due to a pressurized thermal shock event.

REFERENCES:

1. ERG Background Document (FRP-P.1, Rev.3)
2. ERG Exective Volume, Generic Issue, Stagnant Reactor Coolant Loops (Rev.1)

10. B04-036

QUESTION: PT. VALUE: 0.5

The plant is in an emergency condition, and an operator is manually monitoring the Core Cooling Critical Safety Function Status Tree. The use of hot leg temperatures is NOT recommended for this purpose for which of the following reasons?

a. The hot leg temperatures may react to core uncovery more slowly than the Core Exit Thermocouples.
b. The hot leg temperatures indicate the local temperature; while Core Exit Thermocouples indicate the average core temperature.
c. The hot leg temperatures are unreliable due to the susceptibility to failure of their associated instrumentation during inadequate core cooling conditions.
d. The hot leg temperatures are unreliable due to the loss of forced circulation that causes inadequate core cooling conditions in the first place.

ANSWER:

a. The hot leg temperatures may react to core uncovery more slowly than the Core Exit Thermocouples.

REFERENCES:

ERG Background Document —

11. B09-035

QUESTION: PT. VALUE: 0.5

The plant is operating at 100% power, and all systems and controls are in normal alignment. A LOCA occurs. Which of the following correctly describes what will happen as RCS pressure decreases?

a. When pressure decreases to less than 1960 psig on 2 of 3 PRZ pressure instruments, a trip signal will be generated in both protection trains, and both reactor trip breakers will open.
b. When pressure decreases to less than 1960 psig on 1 of 3 PRZ pressure instruments, a trip signal will be generated in one protection train, and one reactor trip breaker will open.
c. When pressure decreases to less than 1860 psig on 2 of 3 PRZ pressure instruments, a trip signal will be generated in one protection train, and both reactor trip breakers will open.
d. When pressure decreases to less than 1860 psig on 1 of 3 PRZ pressure instruments, a trip signal will be generated in both protection trains, and both reactor trip breakers will open.

ANSWER:

a. When pressure decreases to less than 1960 psig on 2 of 3 PRZ pressure instruments, a trip signal will be generated in both protection trains, and both reactor trip breakers will open. (0.5)

REFERENCES:

1. Tech. Spec. 2.2.1
2. Tech. Spec. 3.3.1, Amend. 7
3. SD-103, Rev. 4/AC-2

12. B-NRC-01

QUESTION: **PT. VALUE: 0.5**

An operator is directed to perform a position check of the Hydrogen Side Seal Oil Pump RECIRC. Valve (ISO-242). The valve position is supposed to be 4 turns open. The operator should:

a. Remove the locking device, position the valve to the shut position counting the number of turns until it reaches it's shut seat, note the number of turns, open the valve to the original position, relock the locking device and report the valve position.

b. Remove the locking device, position the valve to the shut position counting the number of turns until it reaches it's shut seat, note the number of turns, open the valve to the position specified on the valve line up form, relock the locking device and report any deviation between original valve position and required valve position.

c. Visually inspect the valve to verify that the locking device is in place and that the valve appears to be in the correct position.

d. Visually inspect the valve to verify that the locking device is in place.

ANSWER: c.

REFERENCES:

1. OMM-011 (5.4) (Rev. 2/AC-2)

13. B05-002

QUESTION: PT. VALUE: 0.5

With regard to operation of the Auxiliary Control Panel (ACP) after a transfer to the ACP, which of the following conditions is true?

a. The automatic start signals for the Turbine Driven Auxiliary Feedwater Pump (TDAFW) and the Motor Driven Auxiliary Feedwater Pumps (MDAFW) are blocked.
b. Only the auto start for the TDAFW pump is blocked.
c. Only the auto start for the MDAFW pump is blocked.
d. Both the TDAFW and MDAFW pumps retain their auto start signals.

ANSWER:

b. (1.0 pt.)

REFERENCES:

Abnormal Operating Procedure (AOP-004)
"Safe Shutdown in case of a fire or control room inaccessibility." (Rev. 5/AC-2)

14. B07-022

QUESTION: **PT. VALUE: 0.5**

At 10:00 a.m. the plant is operating at 100% capacity when onsite and offsite electrical power is lost and an SI signal is simultaneously generated. The Control Operators verify a reactor trip, turbine trip, and both AC emergency buses energized by the emergency diesels. Two minutes into the event, the Turbine-Driven AFW Pump is found to be operating, but the Motor-Driven AFW Pumps are not.

Should the Motor-Driven AFW Pumps have started automatically by this time? Why?

a. No. Load Program "A" will not start the Motor-Driven AFW Pumps for another 10 seconds.
b. No. Load Program "B" does not start the Motor-Driven AFW Pumps for a loss of offsite power with SI.
c. Yes. Load Program "B" should have started the Motor-Driven AFW Pumps by now.
d. Yes. Load Program "A" should have started the Motor-Driven AFW Pumps by now.

ANSWER:

c. Yes. Load Program "B" should have started the Motor-Driven AFW Pumps by now.

REFERENCES:

SD-155.02, Rev.1/AC-1

15. B11-007

QUESTION: **PT. VALUE: 1.0**

You are on shift in the Control Room during refueling operations. You receive a report from containment that a new fuel assembly has been dropped into the core and the SRO-Fuel Handling has evacuated the people in the immediate area. You receive "RAD MONITOR SYSTEM TROUBLE" alarm (ALB-10, 4-5). You observe that the gas channel on the Containment Leak Detection System RM-23 (3502 A-SA) is in alarm and several containment ARM's are indicating ALERT on RM-11.

a. What ESFAS signal(s) will be generated based on the known data? (0.5 pts.)
b. What is/are your immediate action(s)? (0.5 pts.)

ANSWER:

a. Containment Ventilation Isolation (0.5 pts.)
b. Sound the local evacuation alarm (0.25 pts.) and evacuate the affected area (0.25 pts.)

REFERENCES:

AOP-013, Fuel Handling Accident (Rev. 4)

16. B-NRC-03

QUESTION: PT. VALUE: 0.5

The plant is currently at end-of-life in its fuel cycle and will be refueled next month. How will the value of the refueled core's moderator temperature coefficient (MTC) at BOL compare to the present core MTC at EOL?

A. The refueled core's MTC will be less negative.
B. The refueled core's MTC will be more negative.
C. The MTC will change very little.
D. A comparison cannot be made with the available information.

ANSWER: A

REFERENCES:

Westinghouse, Reactor Core Control for Large Pressureized Water Reactors, Chapter 3, page 21.

17. B00-022

QUESTION: PT. VALUE: 1.5

Classify each of the following events. Consider each event separately. Limit your answer to None, Unusual Event, Alert, Site Emergency or General Emergency. Assume that any condition not stated is normal and satisfies any applicable LCO requirements.

-a. A fire results in evacuation of the control room for at least an hour. It is discovered upon evacuation that the ACP is involved in the fire and cannot be accessed.
-b. The 1B CSIP has been out of service over 7 days. Assume mode 1. 1C CSIP is not available.
-c. A steam generator code safety is blowing by and cannot be shut.
-d. A steam generator tube fails, resulting in a 70 gpm leak in the 1B SG. Leak rate prior to the failure was 3 gph.
-e. An aircraft crashes in the switchyard. The resulting fire requires complete deenergizing of the switchyard.
-f. RCS specific activity has been 50 Microcuries/gram Dose Equivalent I-131 for 12 hours. Assume mode 1.

ANSWER: (0.25 each)

a. Site Emergency
b. Unusual Event
c. Alert
d. Alert
e. Unusual Event
f. None

REFERENCES:

1. PEP-101, EAL Flowpath (Rev.3)
2. PEP-100, EAL Interpretation #4 (rev.1)

18. B03-053

QUESTION: PT. VALUE: 1.0

When transfering to cold leg recirc, either 1SI-340 or 1SI-341 must be shut. What is the reason for this ?

ANSWER:

Prevent RHR pump runnout in the event of a loss of 1 RHR pump ~~(during the recirc. phase.)~~ (1.0 pt.)

REFERENCES:

EOP-EPP-010, Transfer to Cold Leg Recirculation (Rev.3)
FSAR Amendment #39

[redacted] SRO
WK1 PART B

To: Mr. Jesse Arildsen, NRC Region II

From: Jim Pierce, CP&L

Subject: Weeks one and two LOR witten exam comments

WEEK I

A02-023 Answer "unless assume a stated pressure" Justification is that during the exam, ERFIS was not responding accurately. Our answer was based on an ERFIS input. Therefore, an answer would be accepted provided that it agrees with an assumed stated pressure. 2300# ⇒ 515°F

A09-019 1.0pt not 1.5pts

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CNMT Fan Coolers to Low Speed
S-2 Fans (Primary Shield Fans)
S-4 Fans (Rx Support Cooling)

* B03-053 Delete "...during the recirculation phase." This was stated in the question.

WEEK II

B02-029 Part B answer, The SR detectors must be manually re-energized(.75)

Jesse, if you have any comments or questions with regards to these requests, please call me as (919)-362-2638. Thank you!

WK 1 SRO B

10/24/89

To: Mr. Jesse Al en, Nrc Region II

From: Jim Pierce, CP&L

Subject: Week One Part-B-SRO Exam Answer Key

B02-C05, part 3 "Classify the event" change answer to Unusual Event vice Alert.

Jesse, please accept my apology for the confusion.

Jim Pierce

WK 1 RO/SRO
B

10/24/89

To: Mr: Jesse Arildsen

From: Jim Pierce, CP&L

Subject: Week One Part-B-RO/SRO Exam Amswer Key

B-NRC-02, change answer for part 2 "Reference OWP-ERFIS-01 for applicable T.S." to

1. T.S. 3.3.3.4 (Met Tower)
2. T.S. 3.3.3.6 item 11 (Subcooling Margin)
3. T.S. 3.3.3.6 item 14 (Prz Safeties)
4. T.S. 3.4.6.1.b (Sump Leakage)
5. T.S. 4.1.3.1.1 (Rod Position Deviation Monitor)
6. T.S. 4.1.3.2 (Rod Position Indication)
7. T.S. 4.2.1.1 (AFD Monitor)

Each part is worth .07pts and either the T.S. number or [illegible] name description is acceptable.

Jim Pierce

MASTER
WK 2

Enclosure 3

SHNPP SRO
ANNUAL REQUALIFICATION EXAM
PART A WRITTEN EXAM
ANSWER KEY

2. A04-008

QUESTION: PT. VALUE: 1.0

What immediate action(s) is/are required to respond to the primary plant conditions as they are shown?

ANSWER:

Shut the No. 1 seal leakoff isolation valve 1CS-437 (1.0 pt.).

REFERENCES:

AOP-018 (Rev. 3)

Scenario Applicability: 15

3. A04-011

QUESTION: PT. VALUE: 1.0

a. With the current problem on "C" RCP, within what time frame must "C" RCP be tripped?
b. Below what power level must the plant be to avoid a Rx trip when the RCP is tripped?

ANSWER:

a. Within 30 minutes of detecting the seal failure (0.5 pts.)
b. Below P-8 Setpoint (49% power) (0.5 pts.)

REFERENCES:

AOP-018 (Rev. 3)

Scenario Applicability: 15

4. A01-027

QUESTION: PT. VALUE: 1.0

Explain the reason for the initial VCT level increase.

ANSWER:

The excessive seal leakoff due to the failed No. 1 seal is being directed to the VCT (via the seal water Hx) (1.0 pt.)

REFERENCES:

SD-107 (Rev. 2)

Scenario Applicability: 15

5. A09-017

QUESTION: **PT. VALUE: 1.0**

List the present MCB indications that identify the failure of PT-447. (Four required for full credit).

ANSWER: (Any 4, 0.25 pts. each)

PI-447 meter = 0 psig

PB-447E (P-13 bistable status light) is deenergized

C-7A permissive status lights are energized

C-7B permissive status lights are energized

ALB-13-2-2 (SR High Flux at shutdown blocked) annunciators are energized

ALB-13-2-3 (SR Loss of Detector Volts) annunciator is energized

REFERENCES:

APP-ALB-013 (Rev. 2)
SD-126.01 (Rev. 1/AC-3)

Scenario Applicability: 15

6. A09-015

QUESTION: **PT. VALUE: 1.0**

Why are the "SOURCE RANGE HIGH FLUX LVL AT SHUTDOWN ALARM BLOCKED" and "SOURCE RANGE LOSS OF DETECTOR VOLTAGE" annunciators illuminated, which is contrary to dark board concept?

ANSWER:

The loss of Impulse pressure channel 447 deenergized its associated P-13 bistable which unblocked these annunciator lights. (1.0 pt.)

REFERENCES:

PCR-1866
APP-ALB-013 windows 2-2 & 2-3 (Rev.2/AC-1)

Scenario Applicability: 15

7. A05-016

QUESTION: PT. VALUE: 0.5

If a reactor trip occurred with present plant conditions, which statement below best describes the response of the steam dump system?

a. The turbine trip controller would position the steam dump valves to control T_{avg} at 557°F.
b. The turbine trip controller would position the steam dump valves to control T_{avg} to within 5°F of T_{ref}.
c. The steam dump valves would be controlled by PK-464.1 (STM HDR DMP PRESS CONT) according to its present setpoint.
d. The steam dump system would not actuate due to the present status of the system.

ANSWER:

c. (0.5 pts.)

REFERENCES:

SD-126.01 (Rev. 1/AC-3)

Scenario Applicability: 15

8. A07-037

QUESTION: PT. VALUE: 1.0

If "A" EDG is declared inoperable, all required systems that depend on the remaining EDG as an emergency power source must be verified operable. What does the word verified imply? (i.e. what has to be done to perform the verification)

ANSWER:

Means to administratively check, by examining logs or other information, to determine if certain components are out of service. (1.0 pt.)

REFERENCES:

Tech. Spec. Basis 3/4.8.1

Scenario Applicability: 15

9. A07-039

QUESTION: **PT. VALUE: 1.0**

When performing the OST to verify the remaining EDG operable, why wouldn't you want to bar the EDG? (1.0)

ANSWER:

During the time of barring, the only remaining EDG would also be inoperable. (1.0 pt.)

REFERENCES:

OST-1073, Rev.1/AC-5
Technical Specification 3.8.1.1
OP-155 (Rev.5)

Scenario Applicability: 15

10. A07-035

QUESTION: PT. VALUE: 1.0

If the "B" ESW pump was found to be inoperable under current conditions, what would be the most restrictive Tech. Spec. action requirement(s) and why? (1.0 pt.)

ANSWER:

Action statement f. of Tech. Spec. 3.8.1.1, (.25) because with "B" ESW pump inoperable the "B" EDG must be placed in MAINT mode which now makes both diesels inoperable. (.75)

REFERENCES:

Technical Specification 3.8.1.1

Scenario Applicability: 15

11. A09-029

QUESTION: PT. VALUE: 0.5

When I&C was dispatched to place the bistables in a tripped condition, the I&C technician inadvertently tripped the bistables associated with PT-456. The bistables were tripped in the following order:

PB 456A, High-Pressure Reactor Trip
PB 456C, Low-Pressure Reactor Trip
PB 456D, Pressurizer Low-Pressure SI
TB 422C1, OT-Delta-T Trip

Which bistable generated the first reactor trip signal when it was placed in the tripped condition?

ANSWER:

PB 456D, Pressurizer Low-Pressure SI (1.0 pt.)

REFERENCES:

SD-103 (Rev. 4)

Scenario Applicability: 20

12. A09-032

QUESTION: **PT. VALUE: 1.0**

Assume that the inoperable reactor trip breaker had been discovered while the reactor was at power and was due to a mechanical failure. Explain any limitations or conditions that would affect operating under this condition.

ANSWER:

The plant would have to be placed in Hot Standby within 6 hours (IAW Tech Specs) (1.0 pt.)

REFERENCES:

Technical Specification 3.3.1

Scenario Applicability: 20

13. A01-047

QUESTION: PT. VALUE: 0.5

While performing Path-1 it is observed that reactor trip breaker B is closed. A manual reactor trip is performed, and the breaker remains closed. The operator should perform which of the following?

a. Continue with the actions of Path-1
b. Transition to FRP-S.1
c. Dispatch personnel to manually open the breaker, and continue with Path-1 once the breaker is opened.
d. Commence emergency boration

ANSWER:

a. Continue with the actions of Path-1 (0.5 pts.)

REFERENCES:

EOP Users Guide (Rev. 1/AC-1)
WOG ERG Background Documents

Scenario Applicability: 20

14. A00-014

QUESTION: PT. VALUE: 1.0

When will implementation of Function Restoration Procedures be allowed during this event?

ANSWER:

Upon transition to EPP-008 (1.0 pt.)

REFERENCES:

EOP Users Guide (Rev. 1/AC-1)
EOP-EPP-Path-1 (Rev. 5)

Scenario Applicability: 20

15. A04-001

QUESTION: **PT. VALUE: 1.0**

FR-154B "RCP Seal Leakoff" has been cycling. Explain why this is occurring.

ANSWER:

Seal return valves 1CS-470/472 shut on the SI/Phase A, (0.5 pts.) and the seal return relief valve is cycling. (0.5 pts.)

REFERENCES:

SD-107 (Rev.2)
SFD, S-1303 (Rev.7)

Scenario Applicability: 1, 2, 4, 9, 14, 16, 17, 20

16. A02-041

QUESTION: PT. VALUE: 0.5

Which one of the following correctly describes what is necessary to reset SI and prevent a re-actuation of automatic SI?

a. SI timer timed out
b. All reactor trip and bypass breakers open
c. SI times timed out and all reactor trip and bypass breakers open
d. No, automatic SI actuation setpoint exceeded and SI timer timed out

ANSWER:

c. SI timer timed out and all reactor trip and bypass breakers open. (0.5 pts.)

REFERENCES:

SD-103 (Rev. 4/AC-2)
Logics

Scenario Applicability: 1, 2, 4, 9, 17, 20, 21

17. A01-049

QUESTION: PT. VALUE: 1.0

Alternate charging valve 1CS-480 indicates open and yet its control switch is positioned to shut. State whether the actual valve position is open or shut and explain your decision.

ANSWER:

Open. (0.25 pts.) 1CS-480 failed open on a loss of air to CNMT. (0.75 pts.) (Which occurred as a result of the phase A CNMT isolation.)

REFERENCES:

SD-107 (Rev.2)
AOP-017, Att.1 (Rev. 3/AC-1)

Scenario Applicability: 1, 3, 4, 17, 20, & 21

18. A01-005

QUESTION: PT. VALUE: 1.0

The controller for PCV-145, the letdown line pressure control valve, is in automatic and its controller output indicates that the PCV is receiving a shut signal. Justify the existence of a shut signal to PCV-145.

ANSWER:

Letdown isolated (on Phase A) so the pressure in the letdown line is less than the controller's setpoint. (0.5 pts.) So the controller is sending a shut signal to try to raise the pressure in the line. (0.5 pts.)

REFERENCES:

SD-107 (Rev. 2)

Scenario Applicability: 1, 2, 3, 4, 14, 16, 17, 20, & 21

19. A00-013

QUESTION: **PT. VALUE: 1.0**

Classify the event in progress IAW the Emergency Plan, based on current plant conditions. Briefly explain the basis for the classification level.

ANSWER:

Unusual Event (0.5 pts.). Unplanned ECCS discharge to vessel (0.5 pts.)

REFERENCES:

PEP-101 Flowpath (Rev. 3/AC-1)

Scenario Applicability: 20

MASTER
WK 2

Master

SHNPP SRO
ANNUAL REQUALIFICATION EXAM
PART B WRITTEN EXAM
ANSWER KEY

1. B00-014

QUESTION: POINT VALUE: 2.0

A LOCA has occurred, and the Containment Ventilation Isolation radiation monitor is in alarm and reading 5 x 10^6 mR/hr. The Shift Foreman is unavailable to fill the SEC position and implement the Emergency Plan. The following activities concerning Emergency Plan implementation need to be performed.

1. The event needs to be classified.
2. The classification needs to be announced.
3. An operations leader needs to be assigned.
4. An emergency communicator needs to be assigned.
5. The TSC, OSC and EOF need to be activated.
6. Determination of notifications needed to be made.
7. ~~Shelter~~ recommendations for people downwind of the plant
Determine ~~needs to be made~~
8. Callouts for additional manpower need to be made.

a. Who (by job position) whould serve as the primary alternate SEC-CR? (0.5 pts.)
b. Which of the activities listed above must the SEC-CR perform himself? (As opposed to delegating to someone else) (0.75 pts.)
c. As a minimum, what would this event be classified? (0.75 pts.)

ANSWER:

a. Roving SCO (0.5 pts.)
b. 1. Classify event (0.25 pts.)
   6. Determine notifications (0.25 pts.)
   7. Make shelter recommendation (0.25 pts)
c. Site Emergency (0.75 pts.)

REFERENCES:

1. PEP-102, Site Emergency Coordinator - Control Room (Rev.6)
2. PEP-101, Emergency Action Level Flowpath (Rev.3)

2. B00-016

QUESTION: PT. VALUE: 1.0

A plant cooldown in preparation for a mini-outage is being performed. Once below 350°F, RHR is placed in service and the cooldown is continued. A clearance is to be placed on the 1A Motor Driven AFW pump for preventative maintenance. Explain why (or why not) independent verification is (is not) required for the alignment changes associated with this clearance?

ANSWER:
~~Independent~~ verification is not required (0.5 pts)

Independent verification is only required when alignment changes to components that provide a safety function have been made ~~(0.5 pts.)~~ in a mode where the system is required. (0.5 pts.)

REFERENCES:

1. AP-002(Rev./AC-2)
2. PLP-702 (Rev.1/AC-1)

3. B03-034

QUESTION: PT. VALUE: 1.0

A SGTR has occurred. The SCO has implemented EOP-PATH-2. In preparation for RCS cooldown, the operator correctly determined the required core exit temperature to be 420 degrees. During the rapid RCS cooldown the RO observes that if the cooldown continues the Tech Spec cooldown limits will be exceeded (127 degrees in 27 minutes). The SCO decides to continue the cooldown to 420 degrees. Is this action appropriate? Justify your response.

ANSWER:

Yes (0.25 pts.). The intent of PATH-2 is to stop primary-to-secondary leakage and to establish and maintain sufficient indications of adequate cooldown inventory as quickly as possible, hence cooldown should not be limited to 100°F/Hr IAW Tech. Specs. (0.75 pts.).

REFERENCES:

1. Westinghouse Owner's Group Emergency Response Guideline E-3 Steam Generator Tube Rupture Background Document.

4. B00-024

QUESTION: PT. VALUE: 0.5

The Control Room Operators are performing FRP-C.2,"Response to Degraded Core Cooling," in response to a magenta path condition shown on the Core Cooling Critical Safety Function Status Tree(CSFST). Which ONE of the following statements is CORRECT with regard to transitions out of this procedure?

-a. The operators must leave this procedure before completion and go to FRP-S.2,"Response to Loss of Core Shutdown,"if the Subcriticality status tree indicates a yellow path condition.

-b. The operators may leave this procedure at any step as soon as the Core Cooling CSFST is satisfied (Green).

-c. The operators must leave this procedure before completion and go to FRP-H.1,"Response of Loss of Secondary Heat Sink", if the Heat Sink CSFST indicates a red path condition.

-d. The operators must leave this procedure before completion and go to FRP-J.1," Response to High Containment Pressure", if the Containment CSFST indicates an magenta path condition.

ANSWER:

c. The operators must leave this procedure before completion and go to FRP-H.1,"Response to Loss of Secondary Heat Sink", if the Heat Sink CSFST indicates a red path condition.

REFERENCES:

1. EOP Users Guide (Rev.1)

5. B01-078

QUESTION: PT. VALUE: 0.5

The reactor trips from 100% power with SI actuation. The cause is a steamline break downstream of the MSIV's. The Control Room Operators take actions per the appropriate procedure(s) and obtain the required plant, system, and component responses, except for the following: the MSIV associated with SG "A" will not shut, and the maximum total AFW flow rate that can be achieved is 200 KPPH. SG levels are as follows:

- SG "A": offscale low, narrow range
- SG "B": offscale low, narrow range
- SG "C": 15% narrow range

Which of the following correctly describes the impact of this configuration on secondary heat removal capability?

a. Secondary heat removal capability has been lost because AFW flow is inadequate.
b. Secondary heat removal capability has been lost because SG levels are inadequate.
c. The loss of secondary heat removal capability is imminent, unless SG "A" is isolated.
d. Secondary heat removal capability will be adequate as long as current SG levels are maintained

ANSWER:

d. Secondary heat removal capability will be adequate as long as current SG levels are maintained.

REFERENCES:

1. EOP-PATH-1, Rev.5
2. ERG Background Document (E-0, Rev.1A)

6. B01-079

QUESTION: PT. VALUE: 0.5

The Control Room Operators are verifying that the appropriate automatic actions have occurred following a reactor trip with SI. Containment pressure has exceeded the containment spray automatic actuation setpoint, so the operators verify that containment spray has initiated and that the containment isolation phase "B" valves have closed. At this point, the applicable procedure instructs the operators to trip the RCP's.

What would be the probable consequences of NOT tripping the RCP's at this point?

a. Forced circulation will make the containment high pressure transient more severe by sustaining an excessive energy release rate to containment.
b. The RCP motor bearings will overheat because Component Cooling Water flow to them has been isolated by the Phase "B" actuation.
c. Radioactive materials will be released from containment because a potential release path remains unisolated.
d. The RCP's will trip on low pressure because it is assumed that if containment spray actuates, an RCS depressurization will be in progress.

ANSWER:

b. The RCP motor bearings will overheat because Component Cooling Water flow to them has been isolated by the Phase "B" actuation.

REFERENCES:

1. EOP-PATH-1,Rev.5
2. ERG Background Document (E-0, Rev.1)

7. B-NRC-08

QUESTION: PT. VALUE: 0.5

Following a complete loss of seal cooling, a RCP should NOT be started unless required:

a. to obtain an accurate RVLIS indication.

b. to ensure RTD bypass temperatures and associated interlocks will be accurate.

c. to provide normal pressurizer spray flow.

d. by an FRP that addresses a red or magenta Critical Safety Function.

ANSWER: d

REFERENCES: EOP-EPP-009 (REV 3)

8. B-NRC-06

QUESTION: **PT. VALUE: 0.5**

The 1A-SA and 1B-SB busses are deenergized due to loss of AC power following a station blackout. EPP-001 has been entered and RCP seals have been locally isolated. When power is restored to one AC emergency bus, the first required action is to:

a. Check ESW system operation.

b. Verify equipment loaded on the AC emergency bus.

c. Stabilize steam generator pressure.

d. Check containment status.

ANSWER: c

REFERENCES: EOP-EPP-001 (REV. 4)

9. B01-017

QUESTION: PT. VALUE: 1.0

Due to a leak on the letdown line, letdown was taken out of service and charging was isolated. The operator observes that pressurizer level is slowly increasing even with charging isolated and attributes the increase to seal injection. What two general actions must be taken by the operator in order to prevent a Reactor Trip on high pressurizer level?

ANSWER:

Seal Injection must be reduced to a minimum in order to minimize makeup to the RCS (0.5 pts.) AND Excess Letdown should be placed in service in order to remove the inventory being added by seal injection. (.5 pts)

REFERENCES:

1. Operating Procedure (OP-107) Chemical and Volume Control System (Rev 3/AC-2)
2. System Description (SD-107) Chemical and Volume Control System (Rev.1)

10. B09-023

QUESTION: PT. VALUE: 0.5

The reactor is critical at 10^{-8} amps. The channel II instrument inverter output breaker trips open. This causes a loss of the IDP-S2 Instrument Bus. These conditions result in which of the following ?

a. The loss of power supply for Source, Intermediate, and Power Range Nuclear Instruments, but no change in reactor power.
b. A reactor trip due to the de-energization of IR channel N-35.
c. A reactor trip due to the de-energization of IR channel N-36.
d. A SR High Flux trip due to the de-energization of permissive P-6.

ANSWER:

c. A reactor trip due to the de-energization of IR channel N-36 (0.5 pts)

REFERENCES:

SD-105 (Rev.4)
SD-156 (Rev.2)
AOP-024 (Rev.4)

11. B09-029

QUESTION: PT. VALUE: 1.5

The plant is operating at 100% power. IR channel N-35 is out of service for a power supply replacement. IR channel N-36 subsequently fails. low.

a. Explain why continued operation at 100% power is allowed under these conditions.

b. If the plant is shutdown or trips, how will the SR detectors be re-energized ?

ANSWER:

a. IR operability requirements are only applicable in Mode 1 below the P-10 setpoint (0.75 pts)

b. ~~When power is reduced below P-10 (0.25 pts) the auto-block of the SR high voltage will be removed (0.25 pts) The SR detectors will be re-energized since P-6 will already be cleared. (0.25 pts).~~

The SR Detectors must be MANUALLY Re-energized (.75) (Due to IR Channel N-35 being taken out-of-service per OWP. P-6 would be energized. And Prevent Re-energizing SR. Automatically.)

REFERENCES:

Technical Specification 3.3.1
SD-105 (Rev.4/AC-1)

12. B01-059

QUESTION: PT. VALUE: 1.5

The plant is operating at 100% power with all control systems in automatic. PRESSURIZER AUX. SPRAY VLV FULL OPEN alarm ALB-6 has been received in the Control Room. The operator observes that all pressurizer heaters are energized with pressure at 2150 psig and decreasing. The operator observes that 1CS-487 (Auxiliary Spray Valve) is open. Attempts to close 1CS-487 are unsuccessful.

A. What action(s) should the operator take to terminate the pressure transient?
B. What Tech Spec LCO's have been entered, if any?

ANSWER:

A. Secure letdown, secure charging, ~~(.5)~~ (1.0) and place excess letdown in service ~~(specific order not required). (.5)~~
B. You have entered the LCO for the DNB parameters Tech Spec. 3.2.5 with Pressurizer pressure <2205 psig. (0.5)

REFERENCES:

1. APP-ALB-006-4-5, Rev.2
2. Tech. Spec. 3.2.5
3. AOP-019, Rev.3

13. B10-001

QUESTION: PT. VALUE: 1.0

During power operations the operator observes that VCT level is trending up on ERFIS and that letdown is diverting to the HUT's. The "CCW SURGE TANK HIGH-LOW" alarm annunciates and subsequently the operating CCW pump trips. Based on this information, what is the most likely location of the CCW leak?

ANSWER:

Seal water heat exchanger. (1.0 pts.)

REFERENCES:

AOP-014, Loss of Component Cooling Water (Rev. 4)

14. B02-023

QUESTION: PT. VALUE: 1.0

Due to the failure of a SI relay, the "A" RHR pump auto starts with the unit at 100% power. Determine the reportability requirements for this event.

ANSWER:

This is a 4 hour report as an ESF actuation. (1.0 pt)

REFERENCES:

1. Administrative Procedure (AP-615) NRC Reporting Requirements (Rev.4/AC-6)

15. B03-060

QUESTION: **PT. VALUE: 0.5**

During a plant heatup the PRZ pressure is 900 psia when a PRZ PORV spuriously opens. Steam is relieved to PRT at 50 psia. Which description of the relieved steam is correct ?

a. Saturated steam at 320°F
b. Superheated steam at 320°F
c. Saturated steam at 540°F
d. Superheated steam at 540°F

ANSWER:

b. (1.0 pt.)

REFERENCES:

ASME Steam Tables

unidentified leakage

16. B03-105

QUESTION: PT. VALUE: 0.5

During normal full power operation, radiation monitors for SG blowdown and condenser vacuum pump effluent were observed to be increasing. OST-1026 "RCS Leakage Evaluation" was performed and results were 0.6 gpm. SG chemistry samples identify a primary to secondary leak on "C" SG of 0.2 gpm.

Based on this information, which of the following actions is correct? (0.5)

a. Power operation can continue, but continue to monitor RCS leakage.
b. Place the reactor in a Hot Shutdown condition within 12 hours.
c. Place the reactor in a Cold Shutdown condition within 24 hours.
d. Place the reactor in a Cold Shutdown condition within 30 hours.

ANSWER:

a. Power operation can continue, but continue to monitor RCS leakage. (0.5)

REFERENCES:

1. AOP-016, Rev. 5
2. Tech. Spec. 3.4.6.2

17. B04-015

QUESTION: PT. VALUE: 0.5

A plant cooldown is being performed with both RHR pumps and heat exchangers in service. Pressure and temperature data are being recorded every 15 minutes in accordance with GP-007. The data below has been obtained over the last 75 minutes.

Time	RCS Temp (Wide Range)	RCS Pressure
00:00	321°F	363 psig
00:15	310°F	356 psig
00:30	298°F	365 psig
00:45	287°F	370 psig
01:00	265°F	352 psig
01:15	225°F	326 psig

Based upon an analysis of this data, the operators should take which of the following actions?

a. Restore RCS temperature to 237°F within 30 minutes.
b. Perform soaking requirements IAW FRP-P.1.
c. Continue cooldown within limits to establish RCS temperature of less than 200°F within 30 hours.
d. Restore cooldown rate to within Tech. Spec. limits and notify the NRC of the violation of the Tech. Spec. cooldown rate.

ANSWER:

d

REFERENCES:

1. Tech. Spec. 3.4.9.2 (Interpretation 89-002, May 1989)
2. GP-007, Rev.3/AC-3

18. B05-046

QUESTION: PT. VALUE: 0.5

The Shift Foreman has just taken over the shift with a RCS heatup in progress. T_{avg} is at 330°F. The shift turnover is to continue the heatup IAW GP-002 to 547°F. You note that the "A" motor-driven AFW Pump is under clearance and out of service. Which ONE of the following is the correct course of action to be taken for this situation?

a. Continue the plant heatup to 547°F.
b. Write a temporary change to GP-002 concerning the AFW pump and then continue the heatup to 547°F.
c. Do not continue the heatup above 350°F until "B" motor-driven AFW Pump is proven operable.
d. Do not continue the heatup above 350°F until "A" motor-driven AFW Pump is returned to service.

ANSWER:

d. Do not continue the heatup above 350°F until "A" motor-driven AFW Pump is returned to service.

REFERENCES:

Technical Specifications
- 3.0.4
- 3.7.1.2

19. B-NRC-11

QUESTION: POINTS: 0.5

Which one of the following have to be completed or reviewed to ensure that regulatory and administrative requirements for plant heatup have been met?

a. Shift Orders.

b. NRC Event Notification Worksheet.

c. Equipment Inoperable Record Book.

d. Caution Tag Log.

ANSWER: c

REFERENCES: GP-002 (REV. 3)

20. B-NRC-10

QUESTION: PT. VALUE: 1.0

The plant is at 98% power and is in the process of increasing power to 100%. All systems are operating in automatic. The A Heater Drain Pump trips due to a loss of NPSH.

1. What automatic actions will occur?

   a. The B Heater Drain Pump will trip on loss of suction, and the turbine will runback to 62% turbine load.

   b. The turbine will runback to 0% turbine load and the reactor will trip.

   c. The main turbine will runback to 86% turbine load.

   d. No automatic actions will occur.

2. Flow to the feed pumps will be reduced by:

   a. 30%

   b. 15%

   c. 5%

   d. 0%

ANSWER:

1. d. (accept c for ~~[illegible]~~ if reference given) KJN

2. b. ~~(accept~~ KJN

REFERENCES:

Licensee Event of October 30, 1988
SD-131.05 (Page 14) (Rev. 3)
SD-136 (Page 11) (Rev. 2)
PCR 4519 (8/27/89)

SRO
PART B

To: Mr. Jesse Arildsen, NRC Region II

From: Jim Pierce, CP&L

Subject: Weeks one and two LOR witten exam comments

WEEK I

A02-023 Answer "unless assume a stated pressure" Justification is that during the exam, ERFIS was not responding accurately. Our answer was based on an ERFIS input. Therefore, an answer would be accepted provided that it agrees with an assumed stated pressure.

2300# ⇒ 515°F

A09-019 1.0pt not 1.5pts

B02-029 "Q" . It is stated that: "Manual action may be required to restart safeguards equipment following a loss of offsite power after SI reset." Specifically, which equipment may need to be restarted or realigned?

"A" (4 required .25 pts ea.)
RHR Pumps
CNMT Spray Pumps
RAB Emergency Exhaust
CNMT Fan Coolers to Low Speed
S-2 Fans (Primary Shield Fans)
S-4 Fans (Rx Support Cooling)

B03-053 Delete "...during the recirculation phase." This was stated in the question.

WEEK II

* ~~B02-029~~ B09-029 Part B answer, The SR detectors must be manually re-energized(.75)

Jesse, if you have any comments or questions with regards to these requests, please call me as (919)-362-2638. Thank you!

ENCLOSURE 4

REQUALIFICATION PROGRAM EVALUATION

The reference material provided by the licensee was reviewed to determine if it was adequate to support the examination. System descriptions and procedures to support the test items were satisfactory. A sampling plan which included topics covered in the requalification program was reviewed and several improvements were identified.

The number of open reference questions provided was consistent with NRC requirements for this stage of program implementation. A majority of these questions were highly subjective, and many of these questions were of a look up nature or required only the ability to match information in the question stem to the reference material in order to answer correctly. As a result, the exam team had to modify or develop many of the written exam questions used for the exam. Questions from exams administered previously at SHNPP were also used in supplement of requal bank areas of deficiencies. The exams that were developed required minor post-exam changes. Of the 15 changes made, 10 were considered substantive. The number of changes made is of concern to the NRC since it is reflective of the facility's ability to properly prepare an exam. Since this program is in the developing stages, the changes were accepted and were not grounds for rating the SHNPP requalification program unsatisfactory. It is expected that future exams will require fewer, if any, changes.

The Job Performance Measures (JPMs) written by the facility covered important tasks; all were related to the Task Analysis and NUREG-1122. The JPMs originally submitted required changes to include evaluator cues and to better define standards for steps of a task. These changes were to ensure that they would be consistently administered by all evaluators. The changes were made in a prompt manner. The follow-up questions to the JPMs were released with the JPMs. A few technical inaccuracies, errors, and additions/deletions to the performance requirements and questions in the JPMs were determined during the administration of the JPMs. It is expected that future exams will require fewer on-the-spot changes.

The simulator exam scenarios submitted required minor changes to emphasize emergency operations and to include passive malfunctions which would allow positive evaluation of completion of immediate operator actions for Emergency Operating Procedures. The changes were promptly made. The passive malfunctions that were added enabled the examiners to objectively measure the operator's performance of critical tasks. Scenarios for future exams should be written such that these items would cause plant conditions to degrade or impede recovery.

The facility evaluators were attentive and did not miss items important to the evaluation. Some minor instances of prompting were exhibited which did not significantly impact the exam. In a few instances the facility evaluators added additional, impromptu questions to a JPM. These questions were not follow-up questions given for clarification nor required by a noted weak area. The addition of such questions can give rise to potential exam inconsistency, grading concerns, question validity concerns due to no prior research for K/A value and technical accuracy, and confusing phraseology.

Based on the results of the examination, the SHNPP Requalification Program is evaluated as satisfactory.

ENCLOSURE 5

SIMULATION FACILITY FIDELITY REPORT

Facility Licensee: Carolina Power and Light Company

Facility Docket No.: 50-400

Operating Tests Administered On: October 3 and 9, 1989

This form is used only to report observations. These observations do not constitute audit or inspection findings and are not, without further verification and review, indicative of noncompliance with 10 CFR 55.45(b). These observations do not affect NRC certification or approval of the simulation facility other than to provide information which may be used in future evaluations. No licensee action is required in response to these observations.

During the conduct of the simulator portions of the operating tests, the following items were observed:

1. The S/G PORVs appear to lift early during S/G tube rupture event.

2. There is no reverse power trip modeled on the EDG output breaker.

3. The EDG manual (emergency) trip switch is labeled backwards.

4. DEH controls for the main turbine will not accept certain load values.

5. FI-940 is mislabeled reading "SI HOT LEG HDR FLOW," and it should read similar to "ALTERNATE HIGH HEAD INJECTION."