<u>ENCLOSURE 1</u>

EXAMINATION REPORT - 50-400/OL-88-01

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 administered at Shearon Harris Nuclear Power Plant near New Hill, North Carolina.

Chief Examiner: William M. Dean 6/10/88
William M. Dean Date Signed

Approved by: for William M. Dean 6/10/88
Kenneth E. Brockman, Section Chief Date Signed
Operator Licensing Section 2

Summary:

Examinations on April 25-28, 1988.

Written and operating tests were administered to 11 reactor operator (RO) candidates, 11 of whom passed. One senior reactor operator (SRO) candidate was administered a written re-examination, whom also passed.

Based on the results described above, 11 of 11 ROs passed and one of one SRO passed.

12 of 25 (48%) changes made to the written examinations were due to inadequate or incorrect facility training material supplied to the examiners for exam item development. Reference material submitted to the NRC should accurately reflect the current plant configuration so that post-examination modifications are minimized. Other problems were noted with the training material and these are discussed in the following report details.

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REPORT DETAILS

1. Facility Employees Contacted:

*Joseph L. Harness, SHNPP General Manager
A. W. Powell, Director of Training
*Chuck Olexik, Operations Supervisor
*Lou Martin, Manager, Nuclear Licensing
*J. T. Bryan, Project Specialist Simulator
*Michael Wallace, Regulatory Compliance, Senior Specialist
*John Hudson, Project Specialist, License Training

*Attended Exit Meeting

2. Examiners:

*William Dean
Siegfried Guenther (OLB:NRR)
D. Charles Payne
Michael Ernstes
Frank Victor (Sonalysts)
Gary Weale (Sonalysts)

*Chief Examiner

3. Examination Review Meeting

At the conclusion of the written examinations, the examiners provided your training staff with a copy of the written examination and answer key for review. The NRC Resolutions to facility comments are listed below.

a. SRO Exam (Applicable RO exam questions in parentheses)

(1) Question 5.03 (1.26)

Comment accepted. The answer key has been modified to accept either answers "b" or "d" for full credit.

(2) Question 5.04 (1.25)

Comment accepted. Due to lack of specificity, this question has been deleted.

(3) Question 5.09

Comment partially accepted. It is recognized that some confusion may have existed as a result of the wording of the question. As a result, parts (a) and (b) have been deleted from the exam. It should be noted that the training material in lesson plan RT-LP-3.7 is confusing and should be reviewed by the facility. In particular, transparencies RT-TP-135.0, 136.0 and 136.1 are difficult to interpret and conflict with information within the body of the lesson plan. Also, changing the fuel loading pattern with each example further confuses the reader as to the geometric relationship between the detector, source and core. Parts (a) and (b) have been deleted.

(4) Question 5.16 (b)

Comment partially accepted. Since both effects are present with no distinction between their level of significance, both are required for a complete answer. Moderator temperature defect reduction has been added to the answer key as part of the answer required for full credit.

(5) Question 5.18 (b)

Comment accepted. Increasing ESW flow has been added to the answer key with two out of three answers required for full credit.

(6) Question 6.01 (3.20)

Comment accepted. Upon review of the facility supplied controlled prints it is acknowledged that the facility training material (in several lesson plans) is in error and should be corrected to reflect current system status.

(7) Question 6.02

Comment accepted. Facility training material (SD-103) incorrectly states that shunt trip coils are on the main breakers "only". This should be corrected to coincide with the current system status. Question has been deleted.

(8) Question 6.04

Comment accepted. The situation postulated in the question was observed using the SHNPP simulator. The behavior of the S/G level did not sufficiently deviate from program level, though it did oscillate. Answer "d" will also be accepted as correct.

(9) Question 6.07 (b)

Comment accepted. Facility training material (SD-134) reflects the answer originally required. This SD should be corrected to reflect current system status. Question has been delted.

(10) Question 6.08

Comment accepted. Recommended answer will be added to the answer key as an additional correct response.

(11) Question 6.13

Comment accepted. Recommended answer will be added to the answer key as an additional correct response.

(12) Question 6.20 (a)

Comment noted. Based on the facility's contention that the candidate has to make an assumption regarding the cause of the EDG start, the facility's recommended answer will be accepted, with additional modifications to make it a complete answer (i.e. address interlocks that must be met).

(13) Question 6.20 (b)

Comment accepted. Facility reference material (SACP-LP-3.0 and SD-155-01) both stated that the control device in question was a "push button" vice a switch. The facility reference material should be corrected to reflect current system status. Answer key has been changed to reflect the answer recommended by the facility.

(14) Question 8.03

Comment accepted. The referenced TS is not consistent in its treatment of RHR suction paths between the LCO and the action statements. Since answers "b" and "d" were very similar, both will be accepted for full credit.

(15) Question 8.05

Comment accepted. The intent of distractor "b" of the question was to fail the input to the OP Delta T circuitry. However, it is acknowledged that, in fact, only the indication was stated to have failed. As a result, either answer "b" or "d" will be accepted for full credit.

(16) Question 8.09

Comment accepted. Facility recommended answer will be added to the answer key as an additional correct response.

(17) Question 8.12

Comment accepted. It was assumed that 1C CSIP would be put in service, but since this was not expressly stated, the facility's recommended response will be included in the answer key in case the candidate made the assumption 1C CSIP would not be put in service.

b. RO Exam

(1) Question 1.04 (a)

Comment accepted. The answer key has been changed to accept the additional answers provided by the facility as alternative answers to those already listed.

(2) Question 1.04 (b)

Comment not accepted. The question provides only two choices, increase or decrease. While the increase may be slight it is still the only correct choice.

(3) Question 1.09

Comment noted. Credit will be allowed for answer "a" if the candidate states that he assumed that a rod was stuck out. Note that OST-1036, "SDM Calculation" (for MODES 3-5) does not include the worth of the most reactive rod unless a rod is actually stuck/immovable.

(4) Question 1.17

Comment accepted. The answer key has been changed as recommended by the facility.

(5) Question 1.18

Comment accepted. The answer key has been changed as recommended by the facility.

(6) Question 1.20

Comment accepted. Answer "a" will also be accepted as correct.

(7) Question 1.24

Comment not accepted. While the question does not address the status of the downcomer, the question specifically solicits the effect on Excore Source Range indication resulting only from phase separation near the top of the core.

(8) Question 2.01

Comment accepted. The answer key has been changed to add the recommended answer as an additional correct response. SD-155.01 should be updated to reflect this information.

(9) Question 2.07 (b)

Comment accepted. Part (b) has been deleted since the information concerning the location of the designated relief valves is incorrect.

(10) Question 2.08

Comment not accepted. The fire detection panels are obviously under the cognizance of the COs, and they should have an understanding of what types of malfunctions would cause an abnormal system status alarm. This is supported by the learning objective referenced by the facility (though it is too general in nature to be a good objective) as well as NUREG 1122 K/A 086000A402 which has a 3.5 importance value. No change to answer key.

(11) Question 2.09 (a)

Comment accepted. The answer key has been changed to accept the answer provided by the facility and to redistribute the point value. The incorrect information contained in the System Description should be updated. Since the question asked for THREE COMPONENTS, the candidate will not be penalized for providing a third answer.

(12) Question 2.09 (b)

Comment accepted. Facility reference material should be updated to eliminate conflicting information within the body of the same system description (SD-145).

(13) Question 2.15

Comment not accepted. The referenced learning objective, though general in nature supports this knowledge as being pertinant, as does the referenced NUREG 1122 K/A. No change to answer key.

(14) Question 2.19 (b)

Comment accepted. Part (b) has been deleted and the point value reduced accordingly. The incorrect information contained in the ESWSS Lesson Plan should be updated.

(15) Question 3.07 (b)

Comment accepted. Since the temperature of the system was not specified, the answer key has been changed as recommended by the facility.

(16) Question 3.16 (b)

Comment noted. Since the answers are equivalent, no change to the answer key is required.

(17) Question 3.16 (b)

Comment noted. Since the change in the dynamic head is very slight, either no change or a change from 100% to slightly above 100% will be accepted.

(18) Question 3.18

Comment accepted. The answer key has been changed to accept the additional response recommended by the facility. The facility training material should be corrected to ensure consistent information is provided within the same training material.

(19) Question 4.01

Comment partially accepted. The additional answer provided by the facility will be added to the list of acceptable answers. There is no change to the number of responses required for full credit, as no justification exists for making this modification.

(20) Question 4.02 (a)

Comment noted. Full credit will be given for the two answers which refer to flow which does not pass through the blender, provided the candidate states the assumption that flow through the blender is not available.

(21) Question 4.04

Comment accepted. If the candidate states "greater than" rather than "greater than or equal to" they will not be penalized. The facility training material (EOP-LP-3.16) should be modified to reflect the guidance contained within the EOP User's Guide.

(22) Question 4.09

Comment accepted. The question has been deleted.

(23) Question 4.11

Comment partially accepted. The additional symptom provided by the facility has been added to the answer key. There is no change to the number of responses required for full credit, as no justification exists for making this modification.

(24) Question 4.12

Comment noted. Information provided by the facility specifically states that the SG PORV controller setpoint is set greater than no load pressure in order to minimize atmospheric releases from the ruptured steam generator. This is the answer required for full credit. The additional information concerning challenges to the code safety value will not result in a deduction if it is included in the answer.

(25) Question 4.16 (c)

Comment partially accepted. The answer key has been modified to the facility's recommended answer, however the point value has remained the same.

4. Exit Meeting

At the conclusion of the site visit the examiners met with representatives of the plant staff to discuss the results of the examination.

There were no generic weaknesses noted during the oral examinations. However, it was noted that most of the Reactor Operators do not complete their immediate actions following a reactor trip/safety injection, rather they wait for the Senior Reactor Operator to read through these steps on a flow chart. This results in a momentary delay in completing verification of automatic actions and is not consistent with the purpose and intent of having designated immediate actions. This appears to be a training vice a knowledge deficiency, as the operators are very familiar with the required response to an SI signal as demonstrated during disscussions with the examiners.

Difficulties in utilizing and obtaining reference material for examination development were also discussed. There were no indices provided for the lesson plans, operating procedures and system descriptions nor were the lesson plans clearly tabbed. This made use of the material as a reference source unwieldy and time consuming. Learning objectives were not clearly related to a job task analysis and tended to be general in nature. They did not specify the conditions (e.g. - "using a system drawing" or "while referring to Technical Specifications") under which the learning objective was to be demonstrated and did not specify the standards of performance with respect to completeness/comprehension other than phrases such as "list, state or explain".

Several requests had to be made by the NRC to obtain complete sets of material. (e.g. reactor theory manual, various lesson plans, administrative, abnormal and normal operating procedures).

Problems with equipment on the simulator being inoperable and continuing simulator lockups were discussed. These are addressed in more detail in Enclosure 4 to this report.

Problems were noted with AOP-14, "Loss of CCW, Increasing Surge Tank" in that step 4 does not provide clear direction on sequential isolation of components in determining the source of leakage.

Procedure OWP-RP-9, "FW Flow Transmitter Failure", was noted to have the incorrect bistable tripped for a failure of protection channel 4 (BS 488B vice 488A).

The cooperation given to the examiners and the effort to ensure an atmosphere in the control room conducive to oral examinations was also noted and appreciated.

The licensee did not identify as proprietary any of the material provided to or reviewed by the examiners.

U. S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR LICENSE EXAMINATION

FACILITY: SHEARON HARRIS 1&2

REACTOR TYPE: PWR-WEC3

DATE ADMINISTERED: 88/04/25

EXAMINER: VICTOR, F.

CANDIDATE: MASTER

INSTRUCTIONS TO CANDIDATE:

Use separate paper for the answers. Write answers on one side only. Staple question sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires at least 70% in each category and a final grade of at least 80%. Examination papers will be picked up six (6) hours after the examination starts.

CATEGORY VALUE	% OF TOTAL	CANDIDATE'S SCORE	% OF CATEGORY VALUE		CATEGORY
29.00 ~~30.00~~	~~25.44~~ [illegible] ~~25.00~~			1.	PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW
[illegible] ~~26.50~~ ~~30.00~~	[illegible] ~~23.25~~ ~~25.00~~			2.	PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS
30.00	[illegible] ~~26.32~~ ~~25.00~~			3.	INSTRUMENTS AND CONTROLS
28.50 ~~30.00~~	[illegible] ~~25.00~~			4.	PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
115.5 ~~114.00~~ ~~120.00~~		Final Grade	%		Totals

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

Candidate's Signature

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

1. Cheating on the examination means an automatic denial of your application and could result in more severe penalties.
2. Restroom trips are to be limited and only one candidate at a time may leave. You must avoid all contacts with anyone outside the examination room to avoid even the appearance or possibility of cheating.
3. Use black ink or dark pencil <u>only</u> to facilitate legible reproductions.
4. Print your name in the blank provided on the cover sheet of the examination.
5. Fill in the date on the cover sheet of the examination (if necessary).
6. Use only the paper provided for answers.
7. Print your name in the upper right-hand corner of the first page of <u>each</u> section of the answer sheet.
8. Consecutively number each answer sheet, write "End of Category __" as appropriate, start each category on a <u>new</u> page, write <u>only on one side</u> of the paper, and write "Last Page" on the last answer sheet.
9. Number each answer as to category and number, for example, 1.4, 6.3.
10. Skip at least <u>three</u> lines between each answer.
11. Separate answer sheets from pad and place finished answer sheets face down on your desk or table.
12. Use abbreviations only if they are commonly used in facility <u>literature</u>.
13. The point value for each question is indicated in parentheses after the question and can be used as a guide for the depth of answer required.
14. Show all calculations, methods, or assumptions used to obtain an answer to mathematical problems whether indicated in the question or not.
15. Partial credit may be given. Therefore, ANSWER ALL PARTS OF THE QUESTION AND DO NOT LEAVE ANY ANSWER BLANK.
16. If parts of the examination are not clear as to intent, ask questions of the <u>examiner</u> only.
17. You must sign the statement on the cover sheet that indicates that the work is your own and you have not received or been given assistance in completing the examination. This must be done after the examination has been completed.

18. When you complete your examination, you shall:

a. Assemble your examination as follows:

(1) Exam questions on top.

(2) Exam aids - figures, tables, etc.

(3) Answer pages including figures which are part of the answer.

b. Turn in your copy of the examination and all pages used to answer the examination questions.

c. Turn in all scrap paper and the balance of the paper that you did not use for answering the questions.

d. Leave the examination area, as defined by the examiner. If after leaving, you are found in this area while the examination is still in progress, your license may be denied or revoked.

QUESTION 1.01 (1.00)

During a reactor startup, control rods are withdrawn such that 1,000 pcm (1% delta-K/K) of reactivity is added. Before the withdrawal Keff was 0.97 and count rate was 500 cps. Select the final steady state count rate following rod withdrawal from the following choices:

a. 750 cps.

b. 1000 cps.

c. 2000 cps.

d. 2250 cps.

QUESTION 1.02 (1.00)

The moderator temperature coefficient (MTC) becomes LEAST NEGATIVE (the absolute value becomes smallest) under which ONE of the following conditions?

a. Moderator temperature is decreased while boron concentration is decreased.

b. Moderator temperature is increased while boron concentration is increased.

c. Moderator temperature is decreased while boron concentration is increased.

d. Moderator temperature is increased while boron concentration is decreased.

QUESTION 1.03 (1.00)

The fuel temperature coefficient (FTC) becomes most negative (the absolute value becomes largest) under which ONE of the following conditions?

a. Low core power, late in core life.

b. Low core power, early in core life.

c. High core power, late in core life.

d. High core power, early in core life.

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.04 (1.50)

There are two primary effects that cause differential boron worth (DBW) to change as the core ages.

a. List the TWO effects and their relative impact on DBW (increase or decrease).

b. State what the total resultant effect is on DBW over core life (increase or decrease).

QUESTION 1.05 (1.00)

The average effective delayed neutron fraction (B-eff) decreases over core life. This is primarily due to the buildup of:

a. Xenon isotopes.

b. Samarium isotopes.

c. Plutonium isotopes.

d. Uranium isotopes.

QUESTION 1.06 (2.00)

Using steam tables, compute the amount of RCS subcooling for Shearon Harris Unit 1 if the plant were operating at 80% power with all parameters at design operating setpoints (for example, pressurizer pressure at 2235 psig). SHOW WORK.

QUESTION 1.07 (1.00)

Indicate for each of the following conditions whether a greater tensile stress is generated on the INNER or OUTER wall of the pressure vessel:

a. Heating up the Reactor Coolant System at 75 degrees F per hour.

b. Increasing system pressure from 2000 psig to 2250 psig.

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.08 (1.00)

The plant is operating at 30 percent load during a load ramp to full power. If steam flow density compensation fails at its 30 percent load value, which ONE of the following describe the affected steam flow indication when at full power:

a. Indicated steam flow is greater than 100%.

b. Indicated steam flow is greater than 30% but less than 100%.

c. Indicated steam flow is equal to 100%.

d. Indicated steam flow is equal to 30%.

QUESTION 1.09 (1.00)

A reactor has been shut down from 100 percent power and cooled down to 140 degrees F over 5 days. During the cooldown, boron concentration was increased by 100 ppm. Given the following absolute values of reactivity which ONE of the answers below would be the value of the shutdown margin?

Rods =	6918 pcm
Temperature =	500 pcm
Boron =	1040 pcm (100 ppm increase)
Power Defect =	1575 pcm

a. minus 3803 pcm
b. minus 4803 pcm
c. minus 5883 pcm
d. minus 6883 pcm

QUESTION 1.10 (2.00)

Answer the following concerning the coefficients that contribute to power defect:

a. State which coefficient contributes most to the CHANGE of power defect over core life and explain why. (1.0)

b. State which coefficient acts first to affect reactivity on a sudden power change due to rod movement and explain why. (1.0)

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.11 (1.00)

During a reactor startup, which ONE of the following statements best describes the change in count rate resulting from a short rod withdrawal when Keff is 0.99 as compared to an identical rod withdrawal when Keff is 0.95?

a. Less time will be required to reach steady-state following the rod withdrawal and the count rate will be greater with Keff at 0.99.

b. More time will be required to reach steady-state following the rod withdrawal and the change in count rate will be less with Keff at 0.99.

c. Less time will be required to reach steady-state following the rod withdrawal and the count rate will be less with Keff at 0.99.

d. More time will be required to reach steady-state following the rod withdrawal and the change in count rate will be greater with Keff at 0.99.

QUESTION 1.12 (1.00)

Assume that a reactor is operating at 100% power when a reactor trip occurs. If the rate of the reactor power decay stabilizes at 4 minutes after the trip with reactor power at 0.01 percent, which ONE of the following represent the number of additional minutes required for power to decrease to 0.0001 percent?

a. 3
b. 4
c. 5
d. 6

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.13 (1.00)

How will RCS loop hot leg temperature indication be affected if the associated resistance temperature detector (RTD) becomes open-circuited?

a. Hot leg temperature indication will be equal to actual hot leg temperature.

b. Hot leg temperature indication will fail as is.

c. Hot leg temperature indication will be higher than actual hot leg temperature.

d. Hot leg temperature indication will be lower than actual hot leg temperature.

QUESTION 1.14 (1.00)

How will affected pressurizer level indication compare to actual pressurizer level during a high-energy line break that raises containment temperature from 100 degrees F to 180 degrees F? Consider only reference leg heating effects.

a. Indicated pressurizer level will be lower than actual pressurizer level.

b. Indicated pressurizer level will be equal to actual pressurizer level.

c. Indicated pressurizer level will fail as is.

d. Indicated pressurizer level will be higher than actual pressurizer level.

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.15 (1.00)

An operator drops his self-reading pocket dosimeter (SRPD) resulting in no physical damage but only a partial discharge of the SRPD. The SRPD will then:

a. Read lower than actual exposure.

b. Read higher than actual exposure.

c. Fail as is.

d. Read equal to actual exposure.

QUESTION 1.16 (1.00)

Which ONE of the following conditions will result in criticality occurring at a higher than estimated control rod position?

a. Inadvertent dilution of RCS boron concentration during rod withdrawal.

b. Misadjusting the steam dump controller such that steam pressure is maintained 50 psig higher than the required no-load setting.

c. Delaying the time of startup from 16 hours to 20 hours following a trip from 100% power equilibrium conditions.

d. A malfunction resulting in control rod speed being faster than normal speed.

QUESTION 1.17 (1.00)

What is the primary reason for arranging symmetrical control rods in groups?

QUESTION 1.18 (1.00)

What are the indications of a cavitating RCP?

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.19 (1.50)

Answer the following TRUE or FALSE concerning Xenon behavior following a reactor trip.

a. Xenon peaks later if the reactor trip occurs at high power as compared to low power.

b. The Xenon concentration decreases following the peak because the half-life of Xenon is shorter than the half-life of Iodine.

c. The Xenon concentration increases as the iodine decay rate becomes smaller than the xenon decay rate.

QUESTION 1.20 (1.00)

If the control rods are NOT maintained above the rod insertion limits during routine reactor operations at power, which ONE of the following is most likely already outside specification limits?

a. Local Power Density (KW/ft)

b. Departure from Nucleate Boiling Ratio (DNBR)

c. Axial Flux Difference (AFD).

d. Quadrant Power Tilt Ratio (QPTR)

QUESTION 1.21 (1.00)

A control rod has its greatest reactivity worth if it is inserted in which ONE of the following locations?

a. Near the edge of the core.

b. Near the center of the core.

c. In a region with high poison concentration.

d. In a region with low fuel concentration.

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.22 (1.00)

Over core life, the reactor vessel Nil- Ductility Temperature (NDT):

a. Increases continuously.

b. Decreases continuously.

c. Increases, then decreases.

d. Decreases, then increases.

QUESTION 1.23 (1.00)

Which ONE of the following core conditions will decrease the departure from nucleate boiling ratio (DNBR)? Consider each separately.

a. Coolant temperature decreases.

b. Coolant flow decreases.

c. Coolant pressure increases.

d. Reactor power decreases.

QUESTION 1.24 (1.00)

The plant has experienced a loss-of-coolant accident (LOCA) with degraded safety injection flow. The reactor coolant pumps are manually tripped and the resulting phase separation causes the upper portion of the core to uncover. (Core is only slightly uncovered.) Which ONE of the following describes Excore Source Range (BF3) neutron level indication relative to indication just prior to partial core uncovery?

a. Significantly less than actual neutron level.

b. Significantly greater than actual neutron level.

c. Essentially unchanged.

d. Impossible to estimate with the given core conditions.

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

~~QUESTION 1.25 (1.00)~~ DELETED

Which ONE of the following situations will the insertion of control rods cause Delta I to become MORE positive?

a. Burnout of Xenon in the top of the core with rods intially fully withdrawn.

b. Positive MTC during a reactor startup.

c. Bank D control rods inserted toward the core midplane.

d. Excessively negative MTC at EOL.

QUESTION 1.26 (1.00)

Select the ONE statement below that is correct if the Power Range instruments have been adjusted to 100% based on a calculated calorimetric.

a. If the feedwater temperature used in the calorimetric calculation was HIGHER than actual feedwater temperature, actual power will be LESS than indicated power.

b. If the reactor coolant pump heat input used in the calorimetric calculation is OMITTED, actual power will be LESS than indicated power.

c. If the steam flow used in the calorimetric calculation was LOWER than actual steam flow, actual power will be LESS than indicated power.

d. If the steam pressure used in the calorimetric calculation is LOWER then actual steam pressure, actual power will be LESS than indicated power.

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.27 (1.00)

Initially, one centrifugal pump is in operation when a second centrifugal pump (in parallel with the first pump) is also put into operation. Select the ONE statement below which correctly describes the effect on system volumetric flow rate and system head loss.

a. Same flow rate, same head loss

b. Higher flow rate, same head loss

c. Same flow rate, higher head loss

d. Higher flow rate, higher head loss

(***** END OF CATEGORY 01 *****)

QUESTION 2.01 (1.00)

List the FOUR conditions that will trip the Emergency Diesel Generator during EMERGENCY operations, in addition to the Emergency Stop Pushbuttons.

QUESTION 2.02 (2.00)

Answer each of the following concerning Reactor Makeup TRUE or FALSE:

a. When the Reactor Makeup Control System is operating in the MANUAL mode, the boric acid and makeup water pumps automatically stop when preset quantities of boric acid and makeup water have been added.

b. In the AUTOMATIC makeup mode placing the RMW Control Switch to START will initiate flow only if the VCT level is less than 20%.

c. On a Safety Injection actuation, the VCT Outlet Valves (LCV-115C and LCV-115E) automatically CLOSE when either of the RWST charging pump suction valves (LCV-115B OR LCV-115D) OPEN.

d. The Alternate Mini-flow Line Isolation valves (ICS-752 and ICS-746) open on a Safety Injection actuation to provide a recirculation path to the RWST if pump discharge pressure exceeds 2300 psig.

QUESTION 2.03 (1.50)

List both the preferred and the emergency source of makeup water to the Component Cooling Water (CCW) System surge tank, if makeup from the CCW Holdup Tank or the CCW Drain Tank is not available.

QUESTION 2.04 (1.00)

List TWO CVCS relief valves that discharge into the Volume Control Tank.

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)

QUESTION 2.05 (1.00)

Which ONE of the following correctly describes how power is NORMALLY supplied to a typical protection channel bus?

a. 480 VAC from EMERG bus, transformed to 120 VAC, rectified to 125 VDC, inverted to 118 VAC.

b. 480 VAC from EMERG bus, transformed to 118 VAC.

c. 250 VDC from battery bus, inverted to 250 VAC, transformed to 118 VAC.

d. 480 VAC from EMERG bus, transformed to 250 VAC, rectified to 250 VDC, inverted to 250 VAC, transformed 118 VAC.

QUESTION 2.06 (1.50)

State the Auxiliary Feedwater (AFW) System design feature and how it operates to protect the motor driven AFW pumps from a run out condition.

QUESTION 2.07 (1.00) ~~(2.00)~~

a. With the Residual Heat Removal System (RHRS) in a normal lineup and the reactor plant operating at 100% power describe how the following are isolated:

1. The RCS hot leg supply to the RHR pumps. (0.5)

2. The RHR pump discharge to the RCS cold legs. (0.5)

(List type of valves and state how they are arranged: series/parallel)

~~b. What is the design basis for the size (flow rate) of the relief valves (1RH 7 & 45) located in the lines leading from the RCS loops to the suction of the RHR pumps?~~ ~~(1.0)~~

~~QUESTION 2.08 (1.50)~~ ~~DELETED~~ left in exam WMD 5/25/88

~~List FIVE conditions that activate amber lights at both the Local Fire Detection Control Panel (LFDCP) and the Main Fire Detection Information Center (MFDIC), as well as actuate an audible alarm distinct from the fire alarms (fire horn).~~

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)

QUESTION 2.09 (2.00) ~~(2.50)~~

a. List TWO ~~THREE~~ components that have their Component Cooling Water supply isolated on a phase A signal. (1.0) ~~(1.5)~~

b. List the TWO loads supplied by each Component Cooling Water essential loop. (1.0)

QUESTION 2.10 (1.50)

Describe the operation of the Containment Fan Coolers when a Loss of Coolant accident causes the ESF system to actuate. Include the position of the dampers and STATE what design feature ensures the proper damper lineup.

QUESTION 2.11 (2.50)

An undervoltage condition on an ESF 6.9 kV bus occurs 20 seconds after the receipt of a Safety Injection signal.

a. What is the sequence of events with respect to the stripping and subsequent loading of LOCA related loads?

b. What conditions must exist before loads can be started manually?

QUESTION 2.12 (2.00)

State the design purpose of each of the following Reactor Coolant Pump (RCP) components. For each purpose, include the circumstance or event that the component is intended to mitigate.

a. Flywheel

b. Thermal Barrier Heat Exchanger

QUESTION 2.13 (1.50)

List THREE sources of hydrogen in the containment structure following a Design Basis Accident?

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)

QUESTION 2.14 (1.50)

a. State the design purpose for the Auxiliary Feedwater (AFW) Isolation Signal. (0.5)

b. What condition generates the AFW isolation signal? (1.0)

QUESTION 2.15 (1.00)

Which ONE of the following would result in the modulation of the Instrument and Service Air Crosstie valve (1IA-648):

a. Instrument air pressure is 80 psig and Service air pressure is 92 psig.

b. Instrument air pressure is 88 psig and Service air pressure is 92 psig.

c. Instrument air pressure is 98 psig and Service air pressure is 78 psig.

d. Instrument air pressure is 93 psig and Service air pressure is 88 psig.

QUESTION 2.16 (1.00)

What TWO design features of the spent fuel racks ensure criticality does not occur in the Spent Fuel Pool?

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)

QUESTION 2.17 (1.50)

a. The Gross Failed Fuel Detector System (GFFDS) monitors which ONE of the following:

1. Gross radioactivity

2. Delayed neutrons

3. Radioactive Iodine

4. Cesium-137 (1.0)

b. TRUE or FALSE

The GFFDS system provides an alarm and also provides a quantitative measure of the extent of a fuel failure. (0.5)

QUESTION 2.18 (1.00)

For each of the following loads, LIST the bus that provides the normal source of electrical power.

a. Main Feedwater Pump 1B

b. Pressurizer Heater Group C

c. EMERGENCY Service Water Pump 1A

d. High Head Safety Injection Pump 1C

QUESTION 2.19 (1.00) ~~(1.50)~~

Answer EACH of the following with regard to the Emergency Service Water System:

a. LIST two (2) design features of the ESW system that prevent the escape of radioactivity from containment via the ESW header during a loss of coolant accident.

~~b. A valve interlock prevents opening the ESW pump backup suction supply valves while the preferred supply valves are still open. STATE the purpose of this interlock.~~

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)

QUESTION 2.20 (1.00)

Four (4) plant atmospheric release points are equipped with wide range gas monitors to satisfy post-TMI requirements. LIST these four locations.

(***** END OF CATEGORY 02 *****)

QUESTION 3.01 (2.00)

The reactor plant is operating in a normal 100% power lineup with all control systems in automatic when Pressurizer pressure detector 1PT-445 fails high. Describe the effect this failure will have on the pressurizer Power-Operated Relief Valves (PORVs) including the functioning of interlocks and/or permissives. Continue the description until either the reactor trips or the plant stablizes. Assume no operator action.

QUESTION 3.02 (1.00)

List FOUR conditions that will cause a Power Cabinet Urgent Failure alarm.

QUESTION 3.03 (1.00)

What TWO conditions must be satisfied to allow the fast acting automatic bus transfer feature to shift power from the Unit Auxiliary Transformers to the Start-up Transformers in response to a reactor trip.

QUESTION 3.04 (1.50)

The reactor is critical at 5% rated thermal power during a normal reactor startup. List SIX (6) reactor trips which are DISABLED in this condition.

QUESTION 3.05 (1.00)

Which ONE of the following parameters is utilized by the Rod Insertion Monitoring System to establish power level when calculating rod insertion limits?

a. RCS Tavg

b. Turbine First Stage Pressure

c. RCS DELTA T

d. Power Range NI signal

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.06 (2.00)

The plant is operating at 50% power with all control systems in automatic. Bank D rods are at 150 steps.

For each of the following, state the direction in which the rods will move and list the parameters and signals which cause the rods to move when each of the following events occur. (assume no operator action (unless stated) and the reactor does NOT trip.) Consider each case separately.

a. Loop A narrow-range Thot instrument fails high. (1.0)

b. Turbine load is reduced to 20% at 5% per minute. (1.0)

QUESTION 3.07 (1.50)

a. List the THREE conditions that will satisfy the RHR System interlocks and allow the RHRS hot leg suction valves (RH-1,RH-2,RH-39,RH-40) to be opened. (1.0)

b. What condition will automatically OPEN and what condition will automatically CLOSE the RHRS miniflow valves (RH-31 and RH-69)? (0.5)

QUESTION 3.08 (1.00)

At 100% power, with the steam dump control system in the Tavg mode, a 10% step loss of load occurs. Assuming no reactor trip occurs, the condenser is available, and rod control is in manual operation, which ONE of the following would occur if Bank 1 (PCV-408F,G,H) steam dump valves failed to open?

a. Bank 2 (PCV-408A,B,C) would open.

b. Atmospheric dumps would open.

c. S/G safeties would open.

d. No steam dump valves would open.

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.09 (1.00)

Match the interlock descriptions in column A with the appropriate coincidence logic required to cause rod withdrawal to be blocked in column B. (Column B items may be used more than once.)

COLUMN A	COLUMN B
	1. 1/1
a. Power Range High Flux @ 103% power	2. 1/2
	3. 2/2
b. Overtemperature Delta T rod stop	4. 1/3
	5. 2/3
c. Intermediate Range High Flux	6. 1/4
	7. 2/4
d. Automatic Rod Control Interlock (C-5)	8. 3/4

QUESTION 3.10 (2.00)

Answer the following concerning the Steam Generator Water Level Control System (SGWLC):

a. Describe how the SGWLC generates a level error signal. (0.75)

b. List the THREE SGWLC inputs and describe how they are used to generate a flow error signal. (1.25)

QUESTION 3.11 (1.00)

Answer the following concerning the Auxiliary Feedwater System TRUE or FALSE.

a. The Trip and Throttle Valve, the Main Steam Supply Valve, and the TDP Flow Control valves are all powered from DC sources.

b. When EITHER Main Steam Supply Valve reaches the full open position, ALL three Flow Control Valves for the Turbine Driven Pump are energized.

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.12 (1.00)

Which ONE of the following statements describes the signal path from the Source Range detector to the Source Range level meter on the Main Control Board?

a. Detector, Pre Amp, Discriminator, Log Integrator, Meter

b. Detector, Log Integrator, Pulse Shaper, Pulse Counter, Meter

c. Detector, Pre Amp, Log Integrator, Discriminator, Meter

d. Detector, Pre Amp, Log Integrator, Log Amp, Meter

QUESTION 3.13 (1.00)

The Detector Current Comparator receives input from all FOUR upper and lower power range detectors.

a. Describe how these inputs are compared? (0.5)

b. What is the alarm setpoint? (0.25)

c. When is this circuitry in operation? (0.25)

QUESTION 3.14 (2.00)

a. List the FOUR control circuits that utilize the output of the T-average auctioneer circuit. (1.0)

b. State the FOUR protection signals generated by the T-average signal from each loop. (1.0)

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.15 (1.50)

Fill in the blanks in the following statements regarding the CVCS System.

a. The CVCS mixed beds and cation demineralizers are protected from high temperature by diverting flow from the demineralizers to the (1) ________ when letdown temperature reaches or exceeds (2)________ degrees F.

b. CVCS emergency makeup from the RWST is initiated when (1)________ VCT level indicator(s) sense(s) (2)________ tank level.

c. The orifice isolation valves are interlocked so they can be manually opened only if pressurizer level is (1)________ and both letdown isolation valves are (2)________.

QUESTION 3.16 (1.50)

The following pertain to indications on the Reactor Vessel Level Indicating System.

a. What will the upper range indication show when a RCP is running in the associated loop?

b. How does dynamic head indication change as reactor power is increased from 0 - 100%?

c. Is OPERABILITY of the Reactor Vessel Level Indicating System required by Technical Specification in Mode 1?

QUESTION 3.17 (2.00)

During a successful starting sequence for the Emergency Diesel Engine, the engine speed has increased to 200 rpm. List FOUR diesel system control functions associated with reaching this speed.

QUESTION 3.18 (2.00)

a. List the TWO types of power (voltage,phase,frequency) supplied to the DC Hold Cabinet AND state the source for each type. (1.2)

b. List the functions of the 125 VDC and 70 VDC power outputs from the DC Hold Cabinet. (0.8)

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.19 (1.00)

What is the reason for NOT using RTDs which are located inside thermowells for temperature inputs to the Reactor Protection System.

QUESTION 3.20 (1.00)

Which ONE of the following statements correctly describes the operation of the Main Steam Line isolation logic?

a. Any ESFAS signal which isolates the MSIVs will also isolate the steam supplies to the turbine driven auxiliary feedwater pump.

b. A low steam line pressure signal in one channel of 2/3 main steam lines will initiate an isolation signal.

c. A trip signal to an MSIV causes redundant solenoid valves to energize and bleed air from pilot valves.

d. A retentive memory in the isolation logic prevents the MSIVs from being reset with the actuation signal still present.

QUESTION 3.21 (2.00)

Answer EACH of the following with regard to the Engineered Safety Features Actuation System (ESFAS):

a. State THREE conditions/interlocks which must be satisfied or actions taken to restore the operators' ability to rearrange ESF equipment lineups after a safety injection has occurred. (1.5)

b. State any additional action(s) which must be taken to re-enable (unblock) subsequent automatic ESF initiation signals. (0.5)

(***** END OF CATEGORY 03 *****)

QUESTION 4.01 (1.50)

List SIX indications, other than annunciators or radiation monitors, that are symptoms of excessive RCS leakage, as listed in AOP-16, Excessive Primary Plant Leakage.

QUESTION 4.02 (1.50)

a. Abnormal Procedure AOP-002, Emergency Boration, lists five available paths to deliver boric acid to the suction of the charging pumps. If the normal path (through the blender) and the preferred Emergency Boration path (through 1CS-278) are not available, list the THREE remaining paths. (0.75)

b. There are four paths available to the Charging Pumps for delivery of boric acid to the RCS. Two of the paths are the normal charging line and the alternate charging line to the RCS loops. List, in order of preference, the remaining TWO paths as listed in AOP-002. (0.75)

QUESTION 4.03 (1.00)

General Procedure GP-007, Normal Plant Cooldown, states that with RCS temperature less than 200 degrees F (Mode 5) only one Reactor Makeup Pump will be operable. What is the basis for requiring this action?

QUESTION 4.04 (1.50)

State the THREE criteria that determine when adverse containment parameters should be monitored, including setpoints where applicable.

QUESTION 4.05 (1.00)

Fill in the blanks in the following statement regarding Nuclear Instrumentation requirements for critical operations:

> The minimum number of operable Power Range instruments to proceed to critical operations is (1)_____. It is required that (2)_____ Intermediate Range channel (s) be operable unless (3)_____ of (4)_____ Power Range channels indicate (s) greater than 10% full power. (0.25 for each answer)

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.06 (1.00)

Answer the following TRUE or FALSE concerning the use of Emergency Operating Procedures (EOP) as discussed in the EOP-Users' Guide, Volume 3 Part 4:

a. Foldout pages contain action requirements that MUST be taken as soon as the symptoms associated with the action are recognized.

b. There are certain contingency End Path Procedures (EPP) which could take precedence over a Red or Magenta condition associated with a Function Restoration Procedure (FRP).

QUESTION 4.07 (1.00)

Answer EACH of the following with regard to Hot Leg Recirculation:

a. Per EOP-Guide-1, STATE when, after a loss of coolant accident initiation, the Safety Injection System is realigned for Hot Leg Recirculation.

b. EXPLAIN why this realignment is necessary.

QUESTION 4.08 (1.00)

Which ONE of the following would cause the greatest biological damage to an individual?

a. 0.1 Rad of Fast Neutron radiation.

b. 0.1 Rem of Gamma radiation.

c. 2.0 Rem of Beta radiation.

d. 0.5 Rad of Alpha radiation.

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

~~QUESTION 4.09 (1.50)~~ DELETED

~~Answer the following questions concerning procedure PLP-702, Independent Verfication.~~

a. Attachment 7.1 to PLP-702 lists systems, subsystems and components which require independent verification. Under what conditions, as specified in PLP-702, would a system, subsystem or component NOT listed in attachment 7.1 require independent verification?

b. When may the Shift Foreman waive the requirements for independent verification?

c. How does a qualified person outside the Shearon Harris organization receive approval to perform independent verification on plant systems or equipment?

QUESTION 4.10 (1.50)

EOP-ECA-0.0, "Loss of All AC Power" requires the operator to check the RCS isolated as one of the immediate actions. List THREE unique systems or components that the operator must check to accomplish this step.

QUESTION 4.11 (2.00)

a. List FOUR indications, other than annunciators,of a Partial Loss of Condenser Vacuum, as listed in AOP-012, Partial Loss of Condenser Vacuum. (Do not include circulating water flow and pressure nor condenser vacuum.) (1.0)

b. If one of the three running Circulating Water Pumps were to trip resulting in the standby vacuum pump automatically starting, what TWO immediate operator actions are required per AOP-012, Partial Loss of Condenser Vacuum? (1.0)

QUESTION 4.12 (1.50)

Emergency Procedure, Path-2 (Path-2 Guide) directs the operator to adjust the ruptured SG PORV controller setpoint to 8.8 (1145 psig) and shut the MSIV and MSIV Bypass valves. What are TWO reason for requiring this action?

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.13 (1.00)

Answer EACH of the following TRUE or FALSE:

a. When directed to check a trended parameter by the EOPs, the operator should use the computer CRTs as the primary indication and the recorder panels in the back of the Control Room as a backup.

b. Under adverse containment conditions, the Reactor Coolant Pumps do NOT have to be tripped if ONLY one pressure instrument indicates less than the value contained in brackets for RCP TRIP CRITERIA.

QUESTION 4.14 (1.50)

After Natural Circulation has been established, what THREE indications are monitored to determine RCS COOLDOWN, according to EOP-EPP-005, Natural Circulation Cooldown?

QUESTION 4.15 (1.00)

Loss of a fission product barrier resulting in High RCS Activity would most likely be detected and reported by Chemistry Technicians. List TWO other indications available to the Control Room Operator which would be symptoms of High RCS Activity, as listed in AOP-032.

QUESTION 4.16 (2.00)

Answer the following questions concerning EOP usage.

a. What indication is used in an EOP procedure to inform the operator that a task must be completed before proceeding to a subsequent step? (0.5)

b. What is the operator required to do if a response not achieved contingency action is required, but CANNOT be successfully completed, and no additional contingency actions are listed? (0.5)

c. What operator actions are required if, during performance of steps in PATH-1, a MAGENTA terminus on a CSF Status is encountered? (1.0)

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.17 (1.00)

The precautions and limitations for GP-007, Normal Plant Cooldown, state that Shutdown Banks C and D must be fully withdrawn when the reactor is subcritical and positive reactivity is being inserted. Which ONE of the following is NOT an approved exception to this rule:

a. Physics testing requires that all shutdown bank rods be fully inserted.

b. The plant is maintained at Hot Standby with a hot Xenon-free boron concentration.

c. The plant is being cooled down with a Cold Shutdown boron concentration.

d. Shutdown Bank A or B can be substituted for Shutdown Banks C and D under special testing circumstances.

QUESTION 4.18 (1.00)

What is the reason for the note in EOP-EPP-008 stating that high temperature in the CCW lines of the thermal barrier heat exchanger may flash to steam when 1CC-252 (RCP thermal barrier flow control valve) is initially opened.

QUESTION 4.19 (2.50)

While proceeding through PATH-1, the operator is directed to "Manually Trip Reactor". If a NO response is obtained in the next step, "Reactor Tripped", list the FIVE immediate actions that the operator is required to initiate.

QUESTION 4.20 (1.00)

OP-107, "Chemical and Volume Control System," cautions that, when isolating a charging pump for maintenance, the discharge isolation valve must be closed prior to closing the suction isolation valve. STATE the basis for this precaution.

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.21 (1.00)

OP-100, "Reactor Coolant System," cautions that seal leakoff valves must be closed when the seal injection water is not supplied and the RCS pressure is less than 100 psig. STATE the basis for this operating precaution.

QUESTION 4.22 (1.00)

In accordance with GP-002, "Normal Plant Heatup From Cold Solid to Hot Subcritical (Mode 5 to Mode 3)", when RCS temperature is >350 deg F and before increasing pressure above 400 psig, the PORV Isolation Valves (1RC-113, 1RC-117) are cycled closed. Annunciators ALB-09-1-5 and ALB-09-3-5 (LOW AUC RCS TEMP & PZR RELF ISOL VLV A (C) SHUT) are verified NOT to alarm, then the isolation valves are cycled back open again. STATE the basis for performing these actions.

QUESTION 4.23 (1.00)

The plant is operating at full power and all systems are functioning within their normal operating bands. WHICH one of the following conditions/malfunctions would require an IMMEDIATE trip of the affected Reactor Coolant Pump, per AOP-018, "Reactor Coolant Pump Abnormal Conditions".

a. Thrust bearing temperature increases to 180 degrees F.

b. Seal inlet temperature increases to 210 degrees F.

c. Motor winding temperature increases to 310 degrees F.

d. Seal injection flow is lost.

(***** END OF CATEGORY 04 *****)
(************* END OF EXAMINATION ***************)

EQUATION SHEET

$f = ma$

$w = mg$

$E = mC^2$

$KE = \frac{1}{2}mv^2$

$PE = mgh$

$W = v\Delta P$

$\Delta E = 931\Delta m$

$\dot{Q} = \dot{m}C_p\Delta T$

$\dot{Q} = UA\Delta T$

$Pwr = W_f\,\dot{m}$

$P = P_o\ 10^{SUR(t)}$

$P = P_o\ e^{t/T}$

$SUR = 26.06/T$

$T = 1.44\ DT$

$SUR = 26\left(\frac{\lambda_{eff}\rho}{\bar{\beta} - \rho}\right)$

$T = (\ell^*/\rho) + [(\bar{\beta} - \rho)/\lambda_{eff}\rho]$

$T = \ell^*/(\rho - \bar{\beta})$

$T = (\bar{\beta} - \rho)/\lambda_{eff}\rho$

$\rho = (K_{eff} - 1)/K_{eff} = \Delta K_{eff}/K_{eff}$

$\rho = [\ell^*/TK_{eff}] + [\bar{\beta}/(1 + \lambda_{eff}T)]$

$P = \Sigma\phi V/(3 \times 10^{10})$

$\Sigma = N\sigma$

$v = s/t$

$s = v_o t + \frac{1}{2}at^2$

$a = (v_f - v_o)/t$

$v_f = v_o + at$

$\omega = \theta/t$

WATER PARAMETERS

1 gal. = 8.345 lbm

1 gal. = 3.78 liters

1 ft^3 = 7.48 gal.

Density = 62.4 lbm/ft^3

Density = 1 gm/cm^3

Heat of vaporization = 970 Btu/lbm

Heat of fusion = 144 Btu/lbm

1 Atm = 14.7 psi = 29.9 in. Hg.

1 ft. H_2O = 0.4335 lbf/in^2

$\text{Cycle efficiency} = \frac{\text{Net Work (out)}}{\text{Energy (in)}}$

$A = \lambda N \qquad A = A_o e^{-\lambda t}$

$\lambda = \ln 2/t_{1/2} = 0.693/t_{1/2}$

$t_{1/2}(eff) = \frac{(t_{1/2})(t_b)}{(t_{1/2} + t_b)}$

$I = I_o e^{-\Sigma x}$

$I = I_o e^{-\mu x}$

$I = I_o\ 10^{-x/TVL}$

$TVL = 1.3/\mu$

$HVL = 0.693/\mu$

$SCR = S/(1 - K_{eff})$

$CR_x = S/(1 - K_{effx})$

$CR_1(1 - K_{eff})_1 = CR_2(1 - K_{eff})_2$

$M = 1/(1 - K_{eff}) = CR_1/CR_0$

$M = (1 - K_{eff})_0/(1 - K_{eff})_1$

$SDM = (1 - K_{eff})/K_{eff}$

$\ell^* = 1 \times 10^{-5}$ seconds

$\lambda_{eff} = 0.1\ \text{seconds}^{-1}$

$I_1 d_1 = I_2 d_2$

$I_1 d_1^2 = I_2 d_2^2$

$R/hr = (0.5\ CE)/d^2(\text{meters})$

$R/hr = 6\ CE/d^2(\text{feet})$

MISCELLANEOUS CONVERSIONS

1 Curie = 3.7×10^{10} dps

1 kg = 2.21 lbm

1 hp = 2.54×10^3 BTU/hr

1 Mw = 3.41×10^6 Btu/hr

1 Btu = 778 ft-lbf

1 inch = 2.54 cm

$^oF = 9/5\,^oC + 32$

[illegible]

ANSWERS -- SHEARON HARRIS 1&2 -88/04/25-VICTOR, F.

ANSWER 1.01 (1.00)

a.

REFERENCE
SHEARON HARRIS RT-LP-3.7 p.30 to 33
Westinghouse, Reactor Core Control for Large Pressurized Water Reactors, 1983, p. 9-10.
192008K104 3.8 ...(KA'S)

ANSWER 1.02 (1.00)

c.

REFERENCE
SHEARON HARRIS RT-LP-3.9 p.9,10,11
Westinghouse, Reactor Core Control for Large PWRs, 1983, p. 3-20 to 3-28.

192004K106 3.1 ...(KA'S)

ANSWER 1.03 (1.00)

a.

REFERENCE
SHEARON HARRIS RT-LP-3.8 p.17,18
Westinghouse, Reactor Core Control for Large PWRs, 1983, p.2-42, 2-46.

192004K107 2.9 ...(KA'S)

ANSWER 1.04 (1.50)

a. 1. Boron concentration decreases over core life which INCREASES DBW (or decreasing boron concentration decreases the amount of spectrum hardening or competition which INCREASES DBW). (0.5)

2. Fission products build up or depletion of BPRA decreases DBW. ~~(0.5)~~ or fuel burn up increases DBW (0.5)

b. INCREASES over core life.(0.5)

ANSWERS -- SHEARON HARRIS 1&2 -88/04/25-VICTOR, F.

REFERENCE
SHEARON HARRIS RT-LP-3.11 p.10,11
CNTO, "Reactor Core Control" p.5-15/16
192004K109 2.8 ...(KA'S)

ANSWER 1.05 (1.00)

c.

REFERENCE
SHEARON HARRIS RT-LP-3.8 p.12
Westinghouse, Fundamentals of Nuclear Reactor Physics, 1983, p.7-37.
192003K104 2.4 ...(KA'S)

ANSWER 1.06 (2.00)

For 2235 psig(2250 psia) sat temp= 653 (+/- 1 degrees) (0.4)
Nominal Full Power T avg= 588.8 - 557(Tc)=DELTA T/2= 32 degrees F (+/-1 degrees)
100% DELTA T = 64 degrees F (+/- 2 degrees) (0.4)
100% DELTA T X 80% = 51 degrees (+/-2 degrees) (0.4)
CORE EXIT TEMP is 557 plus 51= 608 degrees F (+/-2 degrees) (0.4)
SUBCOOLING is 652 degrees F MINUS 608 degrees F=44 degrees F (+/-3 degrees) (0.4)

REFERENCE
STEAM TABLES
SHEARON HARRIS T.S. Table 2.2-1 p.2-8, SD-100.03 p.7
193008K115 3.6 ...(KA'S)

ANSWER 1.07 (1.00)

a. OUTER
b. INNER

REFERENCE
SHEARON HARRIS MTSC-LP-3.4 File No. 12.13 p.4
002000K518 3.3 ...(KA'S)

ANSWERS -- SHEARON HARRIS 1&2 -88/04/25-VICTOR, F.

ANSWER 1.08 (1.00)

a.

REFERENCE
SHEARON HARRIS T&AA-LP-2.9 p.59
Westinghouse, Mitigating Core Damage, 1984, p.6.26
Westinghouse, Thermal-Hydraulic Principles and Applications to the PWR, Vol.2, 1982, p.11-19, 11-20
191002K102 2.7 ...(KA'S)

ANSWER 1.09 (1.00)

c. or (a) if they state that they assume a stuck rod exists

REFERENCE
SHEARON HARRIS RT-LP-3.13 p.7
Westinghouse, Reactor Core Control For Large Pressurized Water Reactors, 1983, p.7-21 thru 7-23.
192002K113 3.5 ...(KA'S)

ANSWER 1.10 (2.00)

a. Moderator Temperature Coefficient (MTC) (0.5) due to the increase (more negative) in MTC as boron concentration is reduced over core life. (0.5)

b. Doppler coefficient (FTC) (0.5) since fuel temperature changes before other parameters change. (0.5)

REFERENCE
SHEARON HARRIS RT-LP- 3.9 p.13
192004K108 3.1 ...(KA'S)

ANSWER 1.11 (1.00)

d.

REFERENCE
SHEARON HARRIS RT-LP-3.7 p.34
Westinghouse, Fundamentals of Nuclear Reactor Physics, 1983, p. 8-54
192008K103 3.9 ...(KA'S)

ANSWER 1.12 (1.00)

d.

REFERENCE
SHEARON HARRIS RT-LP-3.6 p.33 to 40
Westinghouse, Fundamentals of Nuclear Reactor Physics, 1983, p.7-10.
192008K123 2.9 ...(KA'S)

ANSWER 1.13 (1.00).

c.

REFERENCE
SHEARON HARRIS RCTEMP-LP-3.0 File No. 10.11 p. 5
Westinghouse, Mitigating Core Damage, 1984, p.6.9
191002K114 2.8 ...(KA'S)

ANSWER 1.14 (1.00)

d.

REFERENCE
SHEARON HARRIS PZRLC-LP 3.0 p.6
Westinghouse, Mitigating Core Damage, 1984, p.7.16.
191002K108 2.8 ...(KA'S)

ANSWER 1.15 (1.00)

b.

REFERENCE
SHEARON HARRIS RADIATION PROTECTION MANUAL p.5-56 and 5-58
191002K119 3.1 ...(KA'S)

ANSWER 1.16 (1.00)

b.

REFERENCE
SHEARON HARRIS RT-LP-3.15 p. 4 to 6
Westinghouse, Reactor Core Control for Large Pressurized Water Reactors, 1983, p. 7-31 thru 7-33.
192008K102 2.8 ...(KA'S)

ANSWER 1.17 (1.00)

To prevent the formation of abnormally high flux peaks, OR PREVENT FUEL DAMAGE, OR PREVENT UNACCEPTABLE. POWER PEAKS, OR MORE UNIFORM RADIAL FLUX DISTRIBUTION, OR MAKE UNIFORM DIFFERENTIAL CONTROL ROD WORTH.

REFERENCE
SHEARON HARRIS REACTOR THEORY MANUAL p.13-31
Westinghouse, Reactor Core Control for Large PWRs, 1983, p.6-28
192005K108 2.7 ...(KA'S)

ANSWER 1.18 (1.00)

1. Erratic or low flow indication. (ANY 2 AT 0.5 EACH)
2. Pump motor current fluctuating. ~~(0.25 each)~~
3. Excessive pump vibration.
~~4. Abnormal noise.~~

REFERENCE
SHEARON HARRIS FF-LP- 3.2 p.15; FFM File 12.3 No. p. 3-33
Westinghouse, Thermal-Hydraulic Principles and Applications to the PWR, Vol.2, 1982, p. 10-54.
193008K117 2.9 ...(KA'S)

ANSWER 1.19 (1.50)

a. TRUE
b. FALSE
c. FALSE (0.5 each)

REFERENCE
SHEARON HARRIS RT-LP-3.10 p. 7 and 9
192006K102 3.0 ...(KA'S)

ANSWERS -- SHEARON HARRIS 1&2 -88/04/25-VICTOR, F.

ANSWER 1.20 (1.00)

c. OR a. WHO 5/24/88

REFERENCE
SHEARON HARRIS REACTOR THEORY MANUAL p.13-31
Westinghouse, Reactor Core Control for Large PWRs, 1983, p.6-32.
192005K115 3.4 ...(KA'S)

ANSWER 1.21 (1.00)

b.

REFERENCE
SHEARON HARRIS RT-LP-3.12 p.7
Westinghouse, Reactor Core Control for Large PWRs, 1983, p.6-14.
192005K105 2.8 ...(KA'S)

ANSWER 1.22 (1.00)

a.

REFERENCE
SHEARON HARRIS PTS-LP-3.0 p.4
Westinghouse, Thermal-Hydraulic Principles and Applications to the PWR, Vol. 2, p. 13-62.
193010K105 2.9 ...(KA'S)

ANSWER 1.23 (1.00)

b.

REFERENCE
SHEARON HARRIS HEAT TRANSFER MANUAL p.3-23
Westinghouse, Thermal-Hydraulic Principles and Applications to the PWR, Vol. 2, 1982, p. 13-24
193008K105 3.4 ...(KA'S)

ANSWERS -- SHEARON HARRIS 1&2 -88/04/25-VICTOR, F.

ANSWER 1.24 (1.00)

c.

REFERENCE
SHEARON HARRIS MCD-LP-2.6 p.7,8
Westinghouse, Mitigating Core Damage, 1984, p.9.8.
191002K117 3.3 ...(KA'S)

~~ANSWER 1.25 (1.00)~~ DELETED

~~a.~~

REFERENCE
SHNPP: RT-LP-3.14, L.O. 1.1.3, 1.1.11
HBR RXTH-HO-1 Session [CAF]

3.2/3.5
192005K114 ...(KA'S)

ANSWER 1.26 (1.00)

b. OR d.

REFERENCE
NUS, Vol 4, pp 2.2-4
Surry 1-PT-35
SHNPP: HT-LP-3.2, L.O. 1.1.5
GP-LP-3.5
TS 3.3.1
OST 1004

2.6/3.1 3.1/3.4
015000K504 193007K108 ...(KA'S)

ANSWER 1.27 (1.00)

d.

REFERENCE
CNTO, "Thermal/Hydraulic Principles and Applications, II", pp 10-45/48
SHNPP: FF-LP-3.2, L.O. 1.1.5

2.9/3.1 2.3/2.4 2.4/2.5 2.3/2.4 3.1/3.3
006050K501 191004K105 191004K109 193006K102 193006K115
...(KA'S)

ANSWER 2.01 (1.00)

1. Engine Overspeed
2. DG differential (87 relay)
3. Emergency bus differential (0.25 each ans.)
4. Emergency voltage regulator shutdown pushbutton OR LOSS OF POTENTIAL TRANSFORMER.

REFERENCE
SHEARON HARRIS SD-155.01 p.11
3.9 06400K402 ...(KA'S)

ANSWER 2.02 (2.00)

a. TRUE
b. TRUE (0.5 each)
c. FALSE
d. TRUE

REFERENCE
SHEARON HARRIS SD-107 p.28,29,34 and 35
SHEARON HARRIS CVCS-LP-3.0 File No. 4.1 p.21
2.7 004010K606 ...(KA'S)

ANSWER 2.03 (1.50)

1. Primary Makeup Water,(0.5) Emergency makeup.
2. Demineralized Water. (0.5) Prefered makeup.
(0.5 for correct order)

REFERENCE
SHEARON HARRIS SD-145 p.6 and 14
3.0 008010K101 ...(KA'S)

ANSWER 2.04 (1.00)

1. Seal Water Return (ICS-310)
2. Letdown Line Downstream of Pressure Control Valve (1CS-38) or Relief valve ICS-47.

REFERENCE
SHEARON HARRIS CVCS-LP-3.0- File No. 4.1 p. 26,27
3.1 004010K403 ...(KA'S)

ANSWERS -- SHEARON HARRIS 1&2 -88/04/25-VICTOR, F.

ANSWER 2.05 (1.00)

a.

REFERENCE
SHEARON HARRIS SD-156 p.11
3.4 012000K101 ...(KA'S)

ANSWER 2.06 (1.50)

A pressure control valve (0.5) is located at the discharge of each pump (0.5) which throttles flow to maintain adequate discharge pressure. (0.5)

REFERENCE
SHEARON HARRIS SD-137 p. 5 and 9
3.1 061000K404 ...(KA'S)

ANSWER 2.07 (1.00) ~~(2.00)~~

a. 1.Two motor operated valves (0.25) in series. (0.25)
2.Two check valves (0.25) in series. (0.25)

~~b. Each relief valve is sized to pass the combined flow of three charging pumps/SI pumps(1.0) (operating against relief valve set pressure of 450 psig.)~~

REFERENCE
SHEARON HARIS SD-111 p.4 and 7
3.6 005000K109 ...(KA'S)

~~ANSWER 2.08 (1.50)~~ ~~DELETED~~ Leave in WMD 5/24/88

1. Loss of a detection circuit.
2. Loss of an activation circuit.
3. Loss of an alarm circuit.
4. Water not flowing 5 seconds after deluge activated.
5. Operation of water flow detection device.
6. Loss of supervisory air pressure.
7. Operation of a Fire Protection System valve away from normal.

(Any 5 at 0.3 each)

REFERENCE
SHEARON HARRIS SD-149 p.18,19
SHEARON HARRIS L.O. 1.1.4 FP-LP-3.0 File No. 4.14 p.4
086000K403 086000K604 ...(KA'S)

ANSWER 2.09 ~~(2.50)~~ (2.00)

a. 1. ~~The Gross Failed Fuel detector.~~ RCDT HX
~~2. The Sample System Heat Exchanger.~~ (0.5 each ans.)
2. ~~3.~~ The Excess Letdown Heat Exchanger.

b. 1. One RHR Heat Exchanger
2. One RHR Pump ~~Oil~~ SEAL Cooler (0.5 each ans.)

REFERENCE
SHEARON HARRIS SD-145 p.5 and 16
3.3 008000K102 ...(KA'S)

ANSWER 2.10 (1.50)

One of two fans for each cooler unit (0.25) starts in SLOW speed. (0.25) The normal discharge damper (parallel-blade damper) remains open (0.25) and the post accident discharge damper opens. (0.25) The normal damper is locked open (0.25) while the post accident discharge damper fails open (when its instrument air supply is isolated). (0.25)

REFERENCE
SHEARON HARRIS SD-169 p.8,14,15
3.1 022000K402 ...(KA'S)

ANSWER 2.11 (2.50)

a. All loads are stripped from the ESF bus as a result of the LOSP. (0.5) The LOCA loads are blocked (will not restart) for 10 seconds.(0.5) After the Diesel Generator breaker closes, the block 1 LOCA loads start immediately (0.5) and the remaining loads are sequenced on to the ESF bus at 5 second intervals.(0.5)
b. A manual load permissive signal is initiated after the last LOCA (block 8) loads have sequenced on. (0.5)

REFERENCE
SHEARON HARRIS SD-155.02 p.4
3.5 064000K411 , 064000K410 ...(KA'S)

ANSWER 2.12 (2.00)

a. Ensures an extended coastdown of coolant flow through the reactor core (0.5) upon a loss of electrical power to the pump.(0.5).
b. Prevents transfer of heat from the reactor coolant to the pump bearings and seals (0.5) during a loss of seal injection (0.5).

REFERENCE
SHEARON HARRIS SD-100.01 p. 10 and 19

ANSWER 2.13 (1.50)

1. RCS H2 inventory.
2. Zirc-water reaction. (Any 3 at 0.5 each)
3. Radiolysis decomposition of core and sump water.
4. Corrosion of metal/paint (aluminum NaOH reaction)

REFERENCE
SHEARON HARRIS SD-125 p.4
2.9 028000K503 ...(KA'S)

ANSWER 2.14 (1.50)

a. Prevent feeding a faulted Steam Generator. (0.5)

b. Main Steam Isolation signal (0.5) combined with a High Steam Line Differential Pressure signal. (0.5)

REFERENCE
SHEARON HARRIS SD-137 p.11
3.5 061000K414 ...(KA'S)

ANSWER 2.15 (1.00)

d.

REFERENCE
SHEARON HARRIS SD-151 p.10
SHEARON HARRIS ISA-LP-3.0 File No. 5.5 p.8
3.2 078000K402 ...(KA'S)

ANSWER 2.16 (1.00)

1. Storage cells have neutron absorbing material. (0.5)
2. Center-to-center cell spacing (sufficient to maintain a subcritical array even if the pool is filled with unborated water).(0.5)

REFERENCE
SHEARON HARRIS SD-115 p.15
3.1 033000K405 ...(KA'S)

ANSWER 2.17 (1.50)

a. 2. Delayed neutron (1.0)
b. FALSE (0.5)

REFERENCE
SHEARON HARRIS SD-117 p.4
2.5 000076A203 ...(KA'S)

ANSWER 2.18 (1.00)

a. AUX BUS 1B
b. AUX BUS 1D2 (0.25 each)
c. EMERG BUS 1A-SA
d. Either EMERG BUS 1A-SA or 1B-SB

REFERENCE
SHEARON HARRIS SD-134 p.9
SD-100.03 p.10
SD-107

SD-139 p.11
3.3 062000K201 ...(KA'S)

ANSWER 2.19 (1.00) ~~(1.50)~~

a. 1. The ESW booster pumps start on an SI signal.
2. The containment air cooler orifice bypass valves close. (0.5 each)

DELETED
~~b. To prevent sluicing water from the auxiliary reservoir (preferred source) to the main reservoir (backup source). (0.5)~~

REFERENCE
SHNPP: ESWS-LP-3.0, p. 13, 17-19, L.O. 1.1.6, 1.1.3, 1.1.5
076000K402 076000K119 ...(KA'S)

ANSWER 2.20 (1.00)

1. Turbine building stack
2. Plant vent stack
3. WPB stack 5
4. WPB stack 5A (0.25 each)

REFERENCE
SHNPP: RMS-LP-3.0, p. 15, L.O. 1.1.4
071000K106 3.1 ...(KA'S)

ANSWERS -- SHEARON HARRIS 1&2 -88/04/25-VICTOR, F.

ANSWER 3.01 (2.00)

When 1PT-445 fails high, PORVs 445A(1RC-118) and 445B(1RC-116) (0.4) will open.(0.4) (System pressure will rapidly decrease however the PORV will not shut at 2315 psig since the failed detector signal exceeds the reset value.) When pressure decreases to 2000 psig as sensed on 2/3 pressure detectors (1PT-455,456,457)(0.4) the P-11 permissive (0.4) will block the automatic open signal and shut the PORVs. (0.4)

REFERENCE
SHEARON HARRIS SD-100.3 p.16
002000K410 4.2 ...(KA'S)

ANSWER 3.02 (1.00)

1. Regulation failure
2. Phase failure
3. Logic error
4. Multiplexing error
5. Loose card (Any 4 at 0.25 each)

REFERENCE
SHEARON HARRIS RODCS-LP-3.0 File No. 10.6 p.29
001010K605 2.9 ...(KA'S)

ANSWER 3.03 (1.00)

1. Off-site power must be available (0.5)
2. The main generator output breakers must be closed (at the time of the trip). (0.5)

REFERENCE
SHEARON HARRIS SD-156 p.10
062000K403 2.8 ...(KA'S)

ANSWER 3.04 (1.50)

1. Source range reactor trip
2. RCP bus undervoltage
3. RCP bus underfrequency
4. Pressurizer low pressure
5. Pressurizer high level
6. Low flow trip
7. Turbine trip (6 at 0.25 each)

REFERENCE
SHEARON HARRIS RPS-LP-3.0 File No. 10.2 p.24,25
012000K610 3.3 ...(KA'S)

ANSWER 3.05 (1.00)

c.

REFERENCE
SHEARON HARRIS RODCS-LP-3.1 File No. 10.6 p.12
001000K504 4.3 ...(KA'S)

ANSWER 3.06 (2.00)

a. In-(0.2) Loop A Tave will become the auctioneered high Tave and will be higher that Tref.(0.8)

b. In-(0.2) Power mismatch circuit senses turbine power decreasing at a faster rate than nuclear power.(0.8) (Will also accept temperature mismatch.)

REFERENCE
SHEARON HARRIS RODCS-LP-3.0 File No. 10.6 p.20 to 23
001000K403 001050K501 3.3 3.5 ...(KA'S)

ANSWER 3.07 (1.50)

a. 1. RCS pressure <363 psig +/- 5 psig.
   2. RHR discharge to CSIP suction valves (RH-25/RH-63) shut.
   3. Suction from RWST must be shut. (0.33 each ans.)

b. Automatically OPEN when RHRS flow is between ~~725~~ 700 and ~~775~~ 806 gpm.
   Automatically CLOSE when RHRS flow is between ~~1375~~ 1333 and ~~1425~~ 1425 gpm.
   (0.25 each ans.)

REFERENCE
SHEARON HARRIS RHRS-LP-3.0 File No. 2.2 p.20,21
005000K407 3.2 ...(KA'S)

ANSWER 3.08 (1.00)

d.

REFERENCE
SHEARON HARRIS SD-126.01 p.12
041020K414 2.5 ...(KA'S)

ANSWER 3.09 (1.00)

a. 6 (0.25 each ans.)
b. 5
c. 2
d. 1

REFERENCE
SHEARON HARRIS NIS-LP-3.0 File No. 10.2 p.27,28
001050K401 3.4 ...(KA'S)

ANSWER 3.10 (2.00)

a. Actual SG Water Level (0.25) is subtracted from Programmed Level of 66% (0.25) to give Level Error signal. (0.25)

b. Inputs:
   Steam Generator Pressure (0.25)
   Steam Flow (0.25)
   Feed Flow (0.25)

   Feed Flow is subtracted from Steam Flow which has been compensated by Steam Pressure(0.25) to generate a flow Flow Error signal.(0.25)

REFERENCE
SHEARON HARRIS SGWLC-LP-3.0 File No. 10.7 p.6,7
059000K104 059000K408 2.5 3.4 ...(KA'S)

ANSWER 3.11 (1.00)

a. FALSE
b. TRUE

REFERENCE
SHEARON HARRIS SD-137 p.12
061000K103 061000K201 3.2 3.5 ...(KA'S)

ANSWER 3.12 (1.00)

a.

REFERENCE
SHEARON HARRIS NIS-LP-3.0 File No.10.1 p.17,18
015000K601 2.9 ...(KA'S)

ANSWER 3.13 (1.00)

a. Each upper/lower detector reading is compared to the average of the upper/lower detectors.(0.5)
b. An alarm is generates at greater than 2% deviation.(0.25)
c. The circuit is in operation above 50% power on any channel (0.25).

REFERENCE
SHEARON HARRIS SD-105 p.23
015000K604 3.1 ...(KA'S)

ANSWER 3.14 (2.00)

a. 1. RIL Programmer.
   2. Rod Control System.
   3. Steam Dump.
   4. Programed Pressurizer Level (0.25 each)

b. 1. OTdT
   2. OPdT
   3. Lo Tavg
   4. Lo Lo Tavg (0.25 each)

REFERENCE
SHEARON HARRIS RCTEMP-LP-3.0 File No. 10.11 p.8,9
016000K101 3.4 ...(KA'S)

ANSWER 3.15 (1.50)

a. (1) VCT (2) 135
b. (1) both (2) 5%
c. (1) >17% (2) open (0.25 each ans.)

REFERENCE
SHEARON HARRIS SD-107 p.25; p.35; p.37
00401K403/ 004000K403 004000K123 ...(KA'S)

ANSWER 3.16 (1.50)

a. Upper range will indicate minimum level.

b. ~~Dynamic head will read higher than 100%.~~ no change / changes from 100% to slightly o[illegible]
uwo 5/24/88

c. Yes (accident Monitoring Instrumentation) (0.5 each)

REFERENCE
SHEARON HARRIS ICCM-LP-3.0 File No. 10.16 p.14,15,21 and 27
016000A302 016000K101 2.9 3.4 ...(KA'S)

ANSWER 3.17 (2.00)

1. Starting air solenoids close.
2. Keep warm jacket water pumps shut off.
3. Keep warm lube oil pumps shut off.
4. DC current is flashed to the field. (0.5 each)

REFERENCE
SHEARON HARRIS SD-155.01 p.14
064000A306 3.3 ...(KA'S)

ANSWER 3.18 (2.00)

OR 58.5 HZ

a. 1. 260 VAC, 3 phase, 58.3Hz (0.3)
From the rod drive MG sets. (0.3)

OR 58.5 HZ

2. 120 VAC, 1 phase, 58.3Hz (0.3)
From the rod drive MG sets. (0.3)

b. 125 VDC for latching(0.3)
70 VDC for holding rods (0.3) (0.2 for correct association)

REFERENCE
SHEARON HARRIS SD-104 p.8; RODCS-LP-3.0 File No.10.6 p.31
001050G007 3.2 ...(KA'S)

ANSWER 3.19 (1.00)

The temperature response across the thermowell wall is not fast enough for protection grade instrumentation requirements.

REFERENCE
SHEARON HARRIS RCTEMP-LP-3.0 File No.10.11 p.11
016000K101 ...(KA'S)

ANSWER 3.20 (1.00)

~~a~~ d.

REFERENCE
SHNPP: SD-126.01, p. 11, 29
ESFAS-LP-3.0, p.14-15, L.O. 1.1.5
3.7 039000K405 ...(KA'S)

ANSWER 3.21 (2.00)

a. 1. The 60 - second time delay relay must time out
   2. A reactor trip signal (P-4) must be in effect
   3. The operator must manually reset / block the SI signal(s) Any 3@
   4 ϕA RESET (0.5 each)

b. The reactor trip breakers must be shut (0.5)

REFERENCE
SHNPP: ESFAS-LP-3.0, p. 11-12, 26; L.O. 1.1.6, 1.1.7
3.9 013000K401 ...(KA'S)

ANSWERS -- SHEARON HARRIS 1&2 -88/04/25-VICTOR, F.

ANSWER 4.01 (1.50)

1. Increased frequency of RCS makeup.
2. Increasing Containment Pressure.
3. Increasing reactor vessel cavity sump level or pump operation.
4. Increased reactor coolant drain tank temperature.
5. Increase in PRT parameters.
6. Reactor vessel flange leak-off temperature increasing.
7. PORV discharge temperature indication increasing.
8. Pressurizer Safety Valve discharge line temperature increasing.
9. Increasing Containment Temperature.
10. NOTIFICATION OF LEAKAGE BY PLANT PERSONNEL. (any 6 at 0.25 each)

REFERENCE
SHEARON HARRIS AOP-16, p. 3,4
000028A106 3.3 ...(KA'S)

ANSWER 4.02 (1.50)

a. 1. From the RWST (or through LCV-115B, 115D)
   2. Into the top of the VCT (or through FCV-113A and FCV-114A)
   3. Bypass the Boric Acid Blender (or through FCV-113A and 1CS-287)
   (NOTE: ONLY 1 AND 3 REQUIRED FOR FULL CREDIT IF ASSUMPTION MADE THAT FLOW THROUGH BLENDER IS NOT AVAILABLE.) (0.25 each ans.)

b. 1. Seal water supply lines to RCPs.
   2. Auxiliary spray to the pressurizer.
   (0.25 each ans./0.25 correct order)

REFERENCE
SHEARON HARRIS AOP-LP-3,2 File No. 16.12 p.7,8
004000K104 004000K117 004000K609 3.4 4.4
...(KA'S)

ANSWER 4.03 (1.00)

To limit the consequences of a dilution accident (while in cold shutdown).

REFERENCE
SHEARON HARRIS GP-007 p.20
004000A206 4.2 ...(KA'S)

ANSWER 4.04 (1.50)

- Containment pressure (0.25) greater than (or equal to) 3 psig (Hi-1) (0.25)
  or
- Containment radiation (0.25) greater than (or equal to) 100,000 R/hr (0.25)
  or
- Integrated containment radiation dose (0.25) greater than 1,000,000 R (determined by TSC staff) (0.25)

REFERENCE
SHEARON HARRIS EOP-LP-3.16 File No. 16.4 p.60
000011G011 4.3 ...(KA'S)

ANSWER 4.05 (1.00)

(1) 3
(2) 1
(3) 2
(4) 4 (0.25 each ans.)

REFERENCE
SHEARON HARRIS T.S. 3.3.1; SD-105 p.32,38
015020G005 3.3 ...(KA'S)

ANSWER 4.06 (1.00)

a. TRUE
b. TRUE (0.5 each ans.)

REFERENCE
SHEARON HARRIS EOP-Users'Guide p.10,12,15,18
000011G012 4.0 ...(KA'S)

ANSWERS -- SHEARON HARRIS 1&2 -88/04/25-VICTOR, F.

ANSWER 4.07 (1.00)

a. 24 hours after LOCA initiation.

b. To prevent boron precipitation at the top of the core where coolant may be boiling.

(0.5 each)

REFERENCE
SHNPP: EOP-Guide-1, p. 87;
SD-110, p. 17
SIS-LP-3.0, L.O. 1.1.15
EOP-LP-3.3, L.O. 1.1.2

3.8/4.2
000011K313 ...(KA'S)

ANSWER 4.08 (1.00)

d.

REFERENCE
SHEARON HARRIS RADIATION PROTECTION MANUAL p. 3-5, Fig. 3.4
194001K103 2.8 ...(KA'S)

~~ANSWER 4.09 (1.50)~~ DELETED

a. When installing and removing temporary jumpers and lifting electrical leads (0.5)

b. If a component will be frequently cycled during a shift (in which case final position is independently verified). (0.5)

c. Must receive written approval (0.25) by the manager responsible for the procedure in use. (0.25)

REFERENCE
SHEARON HARRIS PLP-702 p. 5 to 7
194000K101 3.6 ...(KA'S)

ANSWER 4.10 (1.50)

1. Check PZR PORVs - shut.
2. Verify LTDN Isolation Valves - shut. (0.5 each answer)
3. Verify Excess LTDN Isolation Valves - shut.

REFERENCE
SHEARON HARRIS EOP-EPP-001 p.4
000056G013 3.4 ...(KA'S)

ANSWER 4.11 (2.00)

a. 1. Condensate Pump discharge temperature increasing.
   2. Increasing Turbine Exhaust Hood temperature. (ANY 4 AT 0.25 EACH)
   3. Abnormal Gland Seal Steam pressure. ~~(0.25 each ans.)~~
   4. Increase in Turbine vibration.
   5. CONDENSER VACUUM BREAKER VALVES NOT CLOSED.

b. 1. Verify tripped Circulating Water Pump Discharge valve closes. (0.5)
   2. Reduce Turbine load. (0.5)

REFERENCE
SHEARON HARRIS AOP-012 p. 3,4
000051G010 2.6 ...(KA'S)

ANSWER 4.12 (1.50)

To isolate flow from the ruptured SG (0.5) which will effectively minimize any release of radioactivity from the ruptured SG (0.5) and allow the establishment of a differential pressure between the ruptured SG and non-ruptured SG. (0.5)

REFERENCE
SHEARON HARRIS EOP-LP-3.2 File No. 16.4 p. 13, 14
000038K306 4.2 ...(KA'S)

ANSWER 4.13 (1.00)

a. True

b. True

(0.5 each)

REFERENCE
SHNPP: EOP User's Guide, p. 15
EOP-LP-3.0, L.O. 1.1.46m
COMP-LP-3.0, L.O. 1.1.1

4.3/4.1
194001A113 ...(KA'S)

ANSWER 4.14 (1.50)

Core Exit T/C
RCS Hot Leg Temperature (0.5 each)
RCS Subcooling

REFERENCE
SHEARON HARRIS EOP-EPP-005 p.14
000074A102 3.9 ...(KA'S)

ANSWER 4.15 (1.00)

1. Gross Failed Fuel Detector High Alarm (0.5)
2. High area radiation levels/alarms (in vicinity of pipes or components containing RCS coolant.) (0.5)

REFERENCE
SHEARON HARRIS AOP-032 p.3
000076G011 3.4 ...(KA'S)

ANSWER 4.16 (2.00)

a. An associated NOTE or the step will state that the task must be completed prior to proceeding. (0.5)

b. Return to next step or sub-step on the left side. (0.5)

c. Monitor all remaining trees for a RED terminus (0.5) and if not encountered, suspend any PATH in progress and perform the applicable FRP for the MAGENTA terminus, AFTER COMPLETION OF THE PATH-1 ~~(0.5)~~ IMMEDIATE ACTIONS. (0.5)

REFERENCE
SHEARON HARRIS EOP-User's Guide, Vol.3 Part 4, p.8 and 11
000011G012 4.0 ...(KA'S)

ANSWER 4.17 (1.00)

d.

REFERENCE
SHEARON HARRIS GP-007 p.6
001050G010 3.3 ...(KA'S)

ANSWER 4.18 (1.00)

To alert the operator that flashing gives a high flow signal (0.5) which could cause 1CC-252 to shut. (0.5)

REFERENCE
SHEARON HARRIS EOP-LP-3.1 File No. 16.4 p.36
000026A106 2.9 ...(KA'S)

ANSWER 4.19 (2.50)

1. Manually insert control rods.
2. Have operator locally trip reactor trip breakers.
3. Verify Turbine Tripped.
4. Verify both MDAFW or TDAFW or one MFW pump running.
5. Initiate Emergency Boration of RCS. (0.5 each ans.)

REFERENCE
SHEARON HARRIS EOP-LP-3.15 File No. 16.4 p. 3 and 12; Path-1 TREE
RPS-LP-3.0 File No. 10-2 p. 4,32
000029G010 4.5 ...(KA'S)

ANSWER 4.20 (1.00)

To prevent overpressurization of the suction line.

REFERENCE
SHNPP: OP-107, p. 9
CVCS-LP-3.0

3.1/3.4 L.O. 1.1.8
004000G010 ...(KA'S)

ANSWER 4.21 (1.00)

To prevent the backflow of potentially dirty water from the VCT.

REFERENCE
SHNPP: OP-100, p. 7
RCS-LP-3.0, L.O. 1.1.20

3.3/3.6
003000G010 ...(KA'S)

ANSWER 4.22 (1.00)

Verifies the Low Temperature Overpressure System is not armed.

REFERENCE
SHNPP: GP-LP-3.2, L.O. 1.1.5
GP-002

3.3/3.6 3.5/3.7 3.8/4.1
010000G010 010000G013 010000K403 ...(KA'S)

ANSWER 4.23 (1.00)

c

REFERENCE
SHNPP: AOP-018, p. 18

2.7/3.1 4.0/3.9
003000A203 003000G014 ...(KA'S)

U. S. NUCLEAR REGULATORY COMMISSION
SENIOR REACTOR OPERATOR LICENSE EXAMINATION

FACILITY: SHEARON HARRIS 1&2

REACTOR TYPE: PWR-WEC3

DATE ADMINISTERED: 88/04/25

EXAMINER: PAYNE, C.

CANDIDATE: MASTER

INSTRUCTIONS TO CANDIDATE:

Use separate paper for the answers. Write answers on one side only. Staple question sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires at least 70% in each category and a final grade of at least 80%. Examination papers will be picked up six (6) hours after the examination starts.

CATEGORY VALUE	% OF TOTAL	CANDIDATE'S SCORE	% OF CATEGORY VALUE		CATEGORY
[illegible]	[illegible]	________	________	5.	THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS
30.00	25.00	________	________	6.	PLANT SYSTEMS DESIGN, CONTROL, AND INSTRUMENTATION
30.00	25.00	________	________	7.	PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
30.00	25.00	________	________	8.	ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS
[illegible]		________ Final Grade	________%		Totals

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

____
Candidate's Signature

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

1. Cheating on the examination means an automatic denial of your application and could result in more severe penalties.

2. Restroom trips are to be limited and only one candidate at a time may leave. You must avoid all contacts with anyone outside the examination room to avoid even the appearance or possibility of cheating.

3. Use black ink or dark pencil <u>only</u> to facilitate legible reproductions.

4. Print your name in the blank provided on the cover sheet of the examination.

5. Fill in the date on the cover sheet of the examination (if necessary).

6. Use only the paper provided for answers.

7. Print your name in the upper right-hand corner of the first page of <u>each</u> section of the answer sheet.

8. Consecutively number each answer sheet, write "End of Category __" as appropriate, start each category on a <u>new</u> page, write <u>only on one side</u> of the paper, and write "Last Page" on the last answer sheet.

9. Number each answer as to category and number, for example, 1.4, 6.3.

10. Skip at least <u>three</u> lines between each answer.

11. Separate answer sheets from pad and place finished answer sheets face down on your desk or table.

12. Use abbreviations only if they are commonly used in facility <u>literature</u>.

13. The point value for each question is indicated in parentheses after the question and can be used as a guide for the depth of answer required.

14. Show all calculations, methods, or assumptions used to obtain an answer to mathematical problems whether indicated in the question or not.

15. Partial credit may be given. Therefore, ANSWER ALL PARTS OF THE QUESTION AND DO NOT LEAVE ANY ANSWER BLANK.

16. If parts of the examination are not clear as to intent, ask questions of the <u>examiner</u> only.

17. You must sign the statement on the cover sheet that indicates that the work is your own and you have not received or been given assistance in completing the examination. This must be done after the examination has been completed.

18. When you complete your examination, you shall:

a. Assemble your examination as follows:

(1) Exam questions on top.

(2) Exam aids - figures, tables, etc.

(3) Answer pages including figures which are part of the answer.

b. Turn in your copy of the examination and all pages used to answer the examination questions.

c. Turn in all scrap paper and the balance of the paper that you did not use for answering the questions.

d. Leave the examination area, as defined by the examiner. If after leaving, you are found in this area while the examination is still in progress, your license may be denied or revoked.

QUESTION 5.01 (1.00)

The reactor trips from full power, equilibrium xenon conditions. Six hours later the reactor is brought critical at 10E-8 amps on the Intermediate Range. If power level is maintained at 10E-8 amps, WHICH one of the following statements concerning rod motion requirements for the next two hours is correct.

a. Rods will have to be rapidly withdrawn since the critical reactor will cause a higher than normal rate of xenon build-in.

b. Rods will have to be rapidly inserted since the critical reactor will cause a high rate of xenon burnout.

c. Rods will have to be withdrawn since xenon will closely follow its normal build-in rate following a trip.

d. Rods will have to be inserted since xenon will closely follow its normal decay rate following a trip.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.02 (1.00)

Initially, one centrifugal pump is in operation when a second centrifugal pump (in parallel with the first pump) is also put into operation. SELECT the one (1) statement below which correctly describes the effect on system volumetric flow rate and system head loss.

a. Same flow rate, same head loss

b. Higher flow rate, same head loss

c. Same flow rate, higher head loss

d. Higher flow rate, higher head loss

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.03 (1.00)

SELECT the one statement below that is correct if the Power Range instruments have been adjusted to 100% based on a calculated calorimetric.

a. If the feedwater temperature used in the calorimetric calculation was HIGHER than actual feedwater temperature, actual power will be LESS than indicated power.

b. If the reactor coolant pump heat input used in the calorimetric calculation is OMITTED, actual power will be LESS than indicated power.

c. If the steam flow used in the calorimetric calculation was LOWER than actual steam flow, actual power will be LESS than indicated power.

d. If the steam pressure used in the calorimetric calculation is LOWER then actual steam pressure, actual power will be LESS than indicated power.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

~~QUESTION 5.04 (1.00)~~

WHICH one of the following situations will the insertion of control rods cause Delta I to become MORE positive?

a. Burnout of Xenon in the top of the core with rods intially fully withdrawn.

b. Positive MTC during a reactor startup.

c. Bank D control rods inserted toward the core midplane.

d. Excessively negative MTC at EOL.

Deleted

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.05 (1.50)

Indicate whether EACH of the following will INCREASE, DECREASE, or have NO EFFECT on the available (actual) Net Positive Suction Head (NPSH).

a. Increasing pump speed.

b. Increasing pump suction temperature.

c. Increasing system pressure.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.06 (2.00)

The reactor is operating at 30% power when one RCP trips. Assuming no reactor trip or turbine load change occur, INDICATE whether EACH of the following parameters will INCREASE, DECREASE, or REMAIN THE SAME.

a. Flow in operating reactor coolant loops

b. Core delta T

c. Reactor vessel delta P

d. Operating loop steam generator pressure

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.07 (2.50)

a. If steam goes through a throttling process, (e.g. a leak from the high pressure main steam header), STATE whether the following parameters will INCREASE, DECREASE, or REMAIN THE SAME. (2.0)

1. Enthalpy
2. Pressure
3. Entropy
4. Temperature

b. STATE whether the steam will become subcooled, saturated or superheated as it leaks out. (0.5)

QUESTION 5.08 (2.50)

Compare the CALCULATED Estimated Critical Position (ECP) for a startup to be performed 4 hours after a trip from 100% power, equilibrium conditions, to the ACTUAL Critical Position (ACP) for EACH of following events/situations. Consider each independently. STATE whether the ACP is HIGHER THAN, LOWER THAN, or the SAME as the ECP.

a. One reactor coolant pump is stopped two minutes prior to criticality.

b. The startup is delayed until 8 hours after the trip.

c. The steam dump pressure setpoint is increased to a value just below the Steam Generator PORV setpoint.

d. Condenser vacuum is reduced by 4 inches of Mercury (24 to 20 in.).

e. All Steam Generator levels are being raised by 5% as the ACP (criticality) is reached.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.09 1.00 ~~(2.00)~~

INDICATE whether EACH of the following fuel loading situations would result in a 1/M plot that was CONSERVATIVE (under predicts criticality) or NONCONSERVATIVE (over predicts criticality).

~~a. Detector located too far from core (source).~~
~~b. Detector located too near core (source).~~ } deleted.

c. Loading core from center (source) towards detector.

d. Loading highest worth assemblies first; lowest worth last.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.10 (1.50)

You have just completed a reactor startup and power level is at the Point Of Adding Heat (POAH). For EACH of the following situations, INDICATE if final stable power level will be HIGHER THAN, LOWER THAN or the SAME AS the power level before the situation occurred. ASSUME the core is at at mid-life. Consider each situation independently.

a. Steam dump pressure setting is raised by 20 psig while dumping steam.

b. A 1% steam leak develops outside of containment.

c. An inadvertant 20 ppm boron addition is made.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.11 (1.50)

Answer EACH of the following statements concerning rod worth TRUE or FALSE:

a. One reason for overlapping rod groups is to minimize the effects of rod shadowing on total rod worth.

b. Both an RCS temperature increase and a buildup of fission product poisons will DECREASE rod worth.

c. The maximum differential rod worth occurs at the point where the integral rod worth is maximum.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.12 (2.00)

For the Moderator Temperature Coefficient (MTC), MATCH the parameter change in Column A to the direction it will change the MTC in Column B. CONSIDER EACH CASE SEPARATELY.

COLUMN A	COLUMN B
1. Moderator temperature increases	a. More Negative
2. Boron concentration increases	b. Less Negative
3. All rods in from an all rods out condition	c. No Effect
4. Flux shape shifting towards edge [illegible]	

QUESTION 5.13 (2.00)

Hot channel factors are normally only measured periodically (by performing incore flux maps). This is sufficient to ensure the core is operated as designed provided four (4) key operational limitations are monitored and maintained. LIST these four (4) key operational limitations.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.14 (1.00)

LIST the four (4) factors that cause the Doppler Power Coefficient to change over core life and indicate whether each of these factors make the Doppler Power Coefficient MORE or LESS NEGATIVE.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.15 (1.00)

LIST the two parameters by which the Technical Specification Heat Flux Hot Channel Factor limit varies.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.16 (1.50)

a. STATE the primary factor at BOL that causes reditribution of the axial flux as power is INCREASED. (0.5)

b. DESCRIBE how the axial flux will shift as power is REDUCED from full to zero power at EOL. STATE the main cause of this behavior. (1.0)

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.17 (1.00)

Given the following. CALCULATE the required boron change to increase reactor power from 75% to 100% while maintaining constant rod position. STATE whether this change will be a boration or a dilution.

Moderator Temperature Coefficient	-15 pcm/degree
Doppler-only Power Coefficient	-12 pcm/%power
Void Reactivity change	-25 pcm
Xenon change	-50 pcm
Boron Coefficient	- 9 pcm/ppm

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.18 (2.50)

a. The plant is currently in Mode 5 with one train of RHR in operation. Assume a nominal RHR flow of 4000 gpm and a reduction in temperature of 8 deg F across the RHR heat exchanger. The reactor engineer informs you that his calculated decay heat load is 0.3% of rated power. With the above plant conditions, STATE whether you CAN or CANNOT control the heat load. SHOW YOUR WORK and state any assumptions. (1.5)

b. LIST two (2) actions that can be taken if the RHR system can not handle the heat load. (1.0)

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.19 (1.50)

A shutdown reactor has a Source Range reading of 270 cps. The operator begins to pull one of three equal worth rod banks. The count rate rises to 450 cps when one bank is fully withdrawn. The operator now pulls the second of the three banks and the count rate rises to 1350 cps when it is fully withdrawn. STATE whether the reactor will go critical on the third bank. (Assume an initial Keff of 0.96) SHOW YOUR WORK!

(***** END OF CATEGORY 05 *****)

QUESTION 6.01 (1.00)

WHICH one of the following statements correctly describes the operation of the Main Steam Line isolation logic?

a. Any ESFAS signal which isolates the MSIVs will also isolate the steam supplies to the turbine driven auxiliary feedwater pump.

b. A low steam line pressure signal in one channel of 2/3 main steam lines will initiate an isolation signal.

c. A trip signal to an MSIV causes redundant solenoid valves to energize and bleed air from the MSIV pilot valves.

d. A retentive memory in the isolation logic prevents the MSIVs from being reset with the actuation signal still present.

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

~~QUESTION 6.02 (1.00)~~

WHICH one of the following statements correctly describes the operation of the reactor trip breaker shunt trip coils?

a. They provide the primary mechanism for tripping the reactor in response to automatic and manual trip signals.

b. They deenergize in response to a reactor trip signal thereby operating a lever which strikes the breaker trip bar to open the breaker.

c. They are ONLY on the main trip breakers and not on the bypass breakers.

d. They energize ONLY in response to automatic reactor trip signals.

Deleted

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.03 (1.00)

The plant is operating normally at 100% power with all control systems in AUTOMATIC, when Power Range nuclear instrumentation Channel N-44 fails upscale. SELECT the one statement below which correctly describes the rod control system's response to this failure.

a. The upscale failure of N-44 will have no effect, since N-45 provides input to the rod control system.

b. CB-D rods will step in due to the mismatch in nuclear versus secondary power and stop at a lower level when the rate of change signal decays.

c. CB-D rods will initially step in due to the mismatch in nuclear versus secondary power and then step out due to the induced Tavg/Tref mismatch.

d. The rods will not move because of an automatic rod motion stop imposed when N-44 failed above 103% power.

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.04 (1.00)

The plant is operating normally at 100% power with all control systems in AUTOMATIC. A normal load reduction to 90% power is initiated, but the controlling feedwater flow transmitter for the "A" steam generator remains stuck at the 100% value. SELECT the one (1) statement below which correctly describes the effects of this malfunction if NO ACTION is taken to correct the problem.

a. Steam generator level will stabilize at a level sufficiently LESS than the original level to offset the flow error.

b. Steam generator level will stabilize at a level sufficiently MORE than the original level to offset the flow error.

c. Steam generator level will remain stable at 66% because of the constant level program regardless of power level.

d. Steam generator level will oscillate around the 66% program setpoint as flow and level errors rise and fall.

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.05 (2.00)

MATCH each RPS trip function in Column A with ALL applicable characteristics in Column B. Items in Column B may be used more than once or not at all.

COLUMN A	COLUMN B
a. Source Range High Flux	1. DNB protection
b. Power Range High Positive Flux Rate	2. rod ejection/withdrawal prot.
c. Overpower Delta-T	3. 1/2 logic
d. Low RCS Loop Flow	4. 2/3 logic below P-8
	5. rod stop
	6. P-7 block
	7. P-6 block

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.06 (1.50)

Answer EACH of the following with regard to the Auxiliary Feedwater System:

a. STATE the design feature that protects the motor-driven AFW pumps from runout or cavitation. (0.5)

b. FILL IN THE BLANKS:

During startup, shutdown and low power operations, main feedwater is introduced to the steam generators via the AFW lines/nozzles in order to _______(1)___________. During power operations, some feedwater is introduced to the steam generators via the AFW lines/nozzles in order to _______(2)________. (1.0)

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.07 1.00 ~~(1.50)~~

Answer EACH of the following with regard to the Emergency Service Water System:

a. LIST two (2) design features of the ESW system that prevent the escape of radioactivity from containment via the ESW header during a loss of coolant accident.

b. ~~A valve interlock prevents opening the ESW pump backup suction supply valves while the preferred supply valves are still open. STATE the purpose of this interlock.~~ Deleted

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.08 (2.50)

The pressurizer protection circuits generate several signals that feed the reactor protection or safeguards initiation circuits. LIST the five (5) protection signals - INCLUDING SETPOINTS - generated by pressurizer pressure.

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.09 (1.50)

Answer EACH of the following with regard to the Post Accident Hydrogen Purge System:

a. LIST two (2) Main Control Board annunciators associated with the Post Accident Hydrogen Purge System. (1.0)

b. Answer the following TRUE or FALSE:

Post-LOCA Containment hydrogen mixing by mass diffusion and natural convection will ensure that no local areas will exceed a 4% H_2 concentration, i.e., no active system is required. (0.5)

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.10 (1.50)

Answer EACH of the following with regard to the Reactor Vessel Level Indication System (RVLIS):

a. LIST two (2) design features common to each RVLIS range that enhance system accuracy during adverse containment environmental conditions. (1.0)

b. LIST the RVLIS range(s) which is/are valid during forced flow conditions. (0.5)

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.13 (1.00)

Four (4) plant atmospheric release points are equipped with wide range gas monitors to satisfy post-TMI requirements. LIST these four locations.

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.12 (2.00)

The plant is operating normally at 100% power with all control systems in AUTOMATIC and channels 459/460 selected for Pressurizer Level Control. LIST four (4) immediate component actuations - NOT ALARMS - that will be initiated as a DIRECT result of a downscale failure in level channel 459.

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.13 (2.00)

Answer EACH of the following with regard to 118 volt AC Uninterruptable Instrument Panel 1DP-1A-S1:

a. LIST the normal, backup and bypass power sources for this instrument panel. INCLUDE the bus designation. ([illegible])

b. TRUE or FALSE:

If the ESF inverter (7.5 KVA Channel I) were to malfunction, power to the instrument panel would automatically transfer to the backup source. (0.5)

QUESTION 6.14 (2.00)

Answer EACH of the following with regard to the Engineered Safety Features Actuation System (ESFAS):

a. STATE three (3) conditions/interlocks which must be satisfied or actions taken to restore the operators' ability to rearrange ESF equipment lineups after a safety injection has occurred. (1.5)

b. STATE any additional action(s) which must be taken to re-enable (unblock) subsequent automatic ESF initiation signals. (0.5)

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.15 (1.00)

STATE the two (2) conditions/interlocks which must be satisfied for the Containment Spray Pump suctions to automatically shift from the Refueling Water Storage Tank (injection mode) to the Containment Sump (recirculation mode). ASSUME all applicable controls are in their AUTOMATIC positions.

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.16 (1.00)

Refueling operations are in progress, and the Traverse Control Switch on the reactor side console for the fuel transfer car is in the "ON" position. STATE two (2) other interlocks which must be satisfied before fuel transfer car movement may be initiated.

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.17 (1.50)

Answer EACH of the following with regard to the Residual Heat Removal System:

a. LIST two (2) interlocks - OTHER THAN SYSTEM PRESSURE - which must be satisfied to permit manual opening of the RHR inlet isolation valves (1RH-1, 2, 39, 40). (1.0)

b. Answer the following TRUE or FALSE:

Each inlet isolation valve is interlocked to close automatically when RHR pump suction pressure exceeds 700 psig. (0.5)

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.18 (1.50)

The Containment Leak Detection System gas/particulate monitor utilizes function pushbuttons for "Purge", "Filter", and "C/S". Briefly DESCRIBE what happens (e.g., flow path/configuration changes) when each of these pushbuttons is momentarily depressed.

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.19 (1.50)

EXPLAIN how the Component Cooling Water System responds to EACH of the following signals.

a. A Safety Injection (SI) signal (1.0)

b. A Containment Phase "B" isolation signal (assume a Phase "A" signal has already occurred) (0.5)

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.20 (2.00)

STATE what actions must be taken and conditions/interlocks met to trip the Emergency Diesel Generators (EDGs) from EACH of the following locations. BE SPECIFIC!

a. Diesel Engine Control Panel (DECP)

b. Auxiliary Control Panel (ACP)

(***** END OF CATEGORY 06 *****)

QUESTION 7.01 (2.50)

a. WHICH one of the following is NOT a condition requiring Emergency Boration according to AOP-002? (1.0)

1. Excessive control rod motion indicated by bank step counter showing the control bank is below its insertion limit.

2. Uncontrolled cooldown following a reactor trip indicated by decreasing PZR level and/or pressure.

3. Unexplained or uncontrolled reactivity increase indicated by abnormal control rod insertion.

4. One or more rod position indicators failing to indicate rod(s) inserted after a reactor trip.

b. WHICH one of the following flow paths is the one required by AOP-002 if the low-low insertion limit is reached? (1.0)

1. Boric acid pump to blender to CSIP suction.

2. Boric acid pump to emergency boration valve.

3. Boric acid pump to blender to VCT.

4. RWST suction to CSIP.

c. According to AOP-002, STATE when Emergency Boration may be stopped. (0.5)

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.02 (1.00)

The plant is operating at full power and all systems are functioning within their normal operating bands. WHICH one of the following conditions/malfunctions would require an IMMEDIATE trip of the affected Reactor Coolant Pump, per AOP-018, "Reactor Coolant Pump Abnormal Conditions".

a. Thrust bearing temperature increases to 180 degrees F.

b. Seal inlet temperature increases to 210 degrees F.

c. Motor winding temperature increases to 310 degrees F.

d. Seal injection flow is lost.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.03 (1.00)

Into WHICH one of the following areas is a worker allowed entry or a GRWP?

a. Neutron radiation area.

b. Airborne radioactivity area.

c. Beta hazard area.

d. High radiation area.

e. Restricted high radiation area.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.04 (1.00)

A 25 year old radiation worker's total lifetime dose is 34 rem and his Form NRC-4 is up to date. His personal monitoring device is processed after the first week of the quarter with a whole body reading of 500 mrem. WHICH one of the following represents the worker's allowable whole body exposure for the remainder of the quarter given the above information?

a. none

b. 500 mrem

c. 750 mrem

d. 1.25 rem

e. 2.5 rem

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.05 (2.50)

During Mode 1 operation and a Loss of Instrument Air, INDICATE whether EACH of the following valves will fail OPEN, CLOSED or is NOT AFFECTED.

a. MSIV's

b. PZR PORV's

c. MFWRV's

d. Charging Flow Control (1CS-231)

e. Letdown Orifice Isolation valves

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.06 (1.00)

Answer EACH of the following TRUE or FALSE:

a. When directed to check a trended parameter by the EOPs, the operator should use the computer CRTs as the primary indication and the recorder panels in the back of the Control Room as a backup.

b. Under adverse containment conditions, the Reactor Coolant Pumps do NOT have to be tripped if ONLY one pressure instrument indicates less than the value contained in brackets for RCP TRIP CRITERIA.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.07 (.50)

Answer the following TRUE or FALSE:

In accordance with EPP-009, "Post LOCA Cooldown and Depressurization", if RCS subcooling is lost during depressurization after a LOCA, RCS depressurization must be stopped until after subcooling is restored.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.06 2.00 (~~2.00~~)

LIST the four (4) Safety Injection termination criteria per EOP Path-1 and their coincidence criteria. INCLUDE setpoints or trends.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.09 (2.00)

LIST four parameters/conditions which are monitored to ensure the existence of natural circulation flow per EPP-002, "Loss of All AC Power Recovery Without SI Required" and EPP-009."Post-LOCA Cooldown and Depressurization". INCLUDE setpoints or trends for each.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.10 (1.50)

In accordance with Section 6.12 of the SHNPP Technical Specifications and HPP-020, "Radiation Work Permits," at least one of three conditions shall be met prior to any individual or group of individuals entering into a high radiation area (i.e., must accompany those entering). STATE the three (3) available options.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.11 (2.50)

STATE the five possible symptoms/indications of a gas release in the Waste Process Building (WPB) in accordance with AOP-009, "Accidental Release of Waste Gas".

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.12 (1.00)

Briefly STATE the basis/reason for EACH of the following, with regard to the transfer to Cold Leg Recirculation, per EOP-EPP-010.

a. The CAUTION which states to perform steps 1 through 5 without delay.

b. Shutting the CSIP alternate miniflow isolation valves (1CS-746, 752) before opening the RHR discharge valves to the CSIP suction (1RH-25, 63).

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.13 (1.00)

Answer EACH of the following with regard to Hot Leg Recirculation:

a. Per EOP-Guide-1, STATE when, after a loss of coolant accident initiation, the Safety Injection System is realigned for Hot Leg Recirculation.

b. EXPLAIN why this realignment is necessary.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.14 (1.00)

EXPLAIN how the EOPs differentiate between subtasks which must be performed in sequence and those which can be performed in any order.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.15 (1.50)

PWRs have on several occasions in the past suffered complete losses of RHR cooling flow during operations with a Steam Generator and parts of the associated RCS loop drained for maintenance (mid-loop operations). STATE three (3) methods/precautions employed at SHNPP to preclude RHR pump vortexing and loss of suction during mid-loop operation.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.16 (1.00)

OP-107, "Chemical and Volume Control System," cautions that, when isolating a charging pump for maintenance, the discharge isolation valve must be closed prior to closing the suction isolation valve. STATE the basis for this precaution.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.17 (1.00)

OP-100, "Reactor Coolant System," cautions that seal leakoff valves must be closed when the seal injection water is not supplied and the RCS pressure is less than 100 psig. STATE the basis for this operating precaution.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.18 (2.50)

A Control Room area fire has occurred and you order evacuation of the Control Room with transfer of essential safe shutdown equipment to the Auxiliary Control Panel (ACP).

a. STATE the major steps for actuating transfer to the ACP from the MCB in accordance with AOP-004, "Safe Shutdown In Case of Fire or Control Room Inaccessibility". INCLUDE any contingency steps and the location where each step would be performed. (2.0)

b. EXPLAIN why this transfer must be accomplished as soon as possible in the above situation. (0.5)

QUESTION 7.19 (1.00)

In accordance with GP-002, "Normal Plant Heatup From Cold Solid to Hot Subcritical (Mode 5 to Mode 3)", when RCS temperature is >350 deg F and before increasing pressure above 400 psig, the PORV Isolation Valves (1RC-113, 1RC-117) are cycled closed. Annunciators ALB-09-1-5 and ALB-09-3-5 (LOW AUC RCS TEMP & PZR RELF ISOL VLV A (C) SHUT) are verified NOT to alarm, then the isolation valves are cycled back open again. STATE the basis for performing these actions.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.20 (1.00)

In accordance with GP-007, "Normal Plant Cooldown (Mode 3 to Mode 5)", the Precautions and Limitations section states that all RCP's and/or RHR pumps may be de-energized for up to one hour when RCS temperature is <350 deg F. STATE the two addtitional restrictions that allow stopping all RCP's and RHR pumps.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.21 (1.00)

In reference to Emergency Operating Procedures, DEFINE "Optimal Eno State".

(***** END OF CATEGORY 07 *****)

QUESTION 8.01 (1.00)

WHICH one of the following statements concerning Shutdown Margin (SDM) considerations is correct?

a. With Tavg less than 200 deg F. the SDM requirements are increased because of the possibility of a positive MTC.

b. The most restrictive condition for SDM requirements occurs at EOL, with Tavg at no load temperature, and is associated with a rod ejection accident.

c. When in Mode 2 with Keff less than 1.0, adequate SDM is ensured by verifying the predicted critical rod position is above the rod insertion limits.

d. If one rod is known to be partially inserted and untrippable, an increased allowance for the entire rod worth shall be made to the SDM requirements.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.02 (1.00)

Unit 1 is at 90% power with no INOP equipment.

Ten minutes into the shift, two level instrument channels associated with the "RWST Level - Low Low" function of ESFAS Instrumentation fail their CHANNEL FUNCTIONAL TESTS. There is no estimate of repair time.

WHICH one of the following actions correctly details the allowances and/or limitations imposed by the Technical Specifications in this instance?

NOTE: APPLICABLE TS ARE ENCLOSED FOR REFERENCE.

a. Operation may proceed provided the inoperable channels are restored to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. Operation may proceed provided the inoperable channels are placed in the bypassed condition and the other Channels are demonstrated OPERABLE within 1 hour.

c. Within one hour action shall be initiated to place the unit in at least HOT STANDBY within the next 6 hours, and at least HOT SHUTDOWN within the following 24 hours.

d. Within one hour action shall be initiated to place the unit in at least HOT STANDBY within the next 6 hours, and at least HOT SHUTDOWN within the following 6 hours, and at least COLD SHUTDOWN within the subsequent 24 hours.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.03 (1.00)

Unit 1 has a Tavg of 250 deg F and is in the process of raising temperature to the normal operating range for plant startup. Twelve hours ago, RHR Heat Exchanger A was declared INOPERABLE. The maintenance supervisor now reports that the suction valve from the Containment Sump to RHR Pump B is INOPERABLE. Upon review, you concur. From the following statements, SELECT the one that correctly describes the allowances and/or limitations imposed by the Technical Specifications that apply in this situation.

NOTE: APPLICABLE TS ARE ENCLOSED FOR REFERENCE.

a. Suspend all operations involving reductions in Reactor Coolant System (RCS) boron concentration and immediately initiate corrective action to return loop to operation.

b. Within 1 hour, action shall be initiated to place the unit in at least COLD SHUTDOWN within the next 24 hours.

c. Restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System Tavg less than 350 deg F by by use of alternate heat removal methods.

d. Restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 24 hours.

QUESTION 8.04 (1.00)

WHICH one of the following conditions ALWAYS requires a Temporary Bypass, Jumper or Wire Removal per AP-024?

a. Pipe caps installed to stop remote manual drain valve leakage.

b. Electrical equipment such as motor leads, transmitter leads, or relays which can be de-energized.

c. Power cables from receptacles to temporary or portable equipment.

d. A temporary bypass, jumper or wire removal for an operable safety system.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.05 (1.00)

The reactor is operating at 20% power, normal operating temperature with all systems in AUTOMATIC. WHICH one of the following situations does NOT have an associated 1-hour Technical Specification action item?

a. One shutdown rod is found to be partially inserted.

b. One of three Overpower Delta T indications has failed.

c. One isolation valve on an RCS accumulator is found closed.

d. The RWST solution temperature is 35 deg F.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.06 (.50)

Answer the following TRUE OR FALSE:

If a conflict is created between a Shift Note initiated by the Operations Supervisor and a plant operating procedure, the Shift Note takes precedence.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.27 (2.50)

The unit is operating at full power when it is determined that a Technical Specification (TS) 3/4.52, "ECCS Subsystems - Tavg Greater Than or Equal to 350 Deg F" and its associated ACTION statements cannot be met, thereby placing the unit into TS 3.0.3.

Answer EACH of the following TRUE or FALSE regarding the actions/notifications required by PGO-040, "Implementation of Technical Specification 3.0.3":

a. The Shift Foreman (SF) shall document the time TS 3.0.3 is entered and the time by which the unit must be in COLD SHUTDOWN in the SF log.

b. Within one hour, the load dispatcher (LD) shall be notified of the shutdown requirement and requested to schedule an orderly shutdown of the unit to be completed within 6 hours of the LD being notified.

c. Except in the case of a declared System Emergency, a uniform shutdown at a rate of < 10 MW/min should be conducted.

d. In the case of a System Emergency, it is acceptable to continue to operate the unit, maximizing the use of TS 3.0.3 time limits to supply the grid, and then trip the unit.

e. If this event had occurred in Mode 3, no NRC notifications would be required.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.02 (1.75)

FILL IN THE BLANKS:

a. Per OMM-001, STATE the MINIMUM operations shift staffing levels for EACH of the following positions during normal power operations.

SF ________
SRO ________
RO ________
AO ________
STA ________

b. STATE which positions above are NOT required to be manned during Mode 5 operations.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.09 (1.50)

The Control Operator has just satisfactorily completed an operations surveillance test and submitted it to you, as Shift Foreman, for disposition. STATE the three (3) actions per OMM-001, "Conduct of Operations" you are required to take with regard to the completed test.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.10 (1.00)

STATE the two (2) conditions under which a Tracking EIR (Equipment Inoperable Record) is used.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.11 (1.00)

STATE the information provided in the Daily Batch Report, which is delivered to the control room every Monday through Friday (holidays excepted) in accordance with PLP-103, "Surveillance and Periodic Test Program.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.12 (2.00)

The unit is operating normally at full power with only one significant inoperable component - the 1B CSIP - which is not expected to be repaired for three days. While performing a periodic surveillance test on the 1A emergency diesel generator, it trips unexpectedly and is declared inoperable at 11:00 a.m. The EDG is repaired, satisfactorily tested and restored to operability at 8:00 p.m., that evening. LIST all the LCO compensatory actions that were required to have been completed as a result of this equipment failure. INCLUDE the time/day by which each must be completed.

NOTE: APPLICABLE TS ARE ENCLOSED FOR REFERENCE

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.13 (2.00)

The plant is operating normally in Mode 2, when a transient causes Reactor Coolant System pressure to exceed the safety limit of 2735 psig. LIST four actions required by Technical Specifications as a result of this situation.

NOTE: APPLICABLE TS ARE ENCLOSED FOR REFERENCE

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.14 (1.50)

Technical Specification 3.1.1.1 requires that the SDM be greater than or equal to 1770 pcm for 3-loop operation in Modes 1 - 4. The basis for this minimum SDM requirement postulates a particular accident occurring under the most restrictive plant conditions. STATE the postulated accident and LIST the two most restrictive plant conditions.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.15 (1.75)

Temporary changes to procedures required by Technical Specification 6.8.1 may be made provided three conditions are met. STATE those three (3) conditions.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.16 (1.50)

In order to perform core alterations near the "A" Reactor Hot Leg, the operating RHR loop is shutdown at 0900. Reactor water level is 24' above the reactor vessel flange. The other RHR loop is unavailable due to maintenance activities. One hour later, the core alterations are completed and the RHR loop is restarted. At 1030, the RHR loop is again shutdown to perform core alterations near "B" Reactor Hot Leg and then restarted at 1130.

STATE whether Technical Specifications have been violated. JUSTIFY your response.

NOTE: APPLICABLE TS ARE ENCLOSED FOR REFERENCE.

QUESTION 8.17 (2.00)

Technical Specification 3.4.1.2 requires that, when in Mode 3, at least two reactor coolant loops be operable with two Reactor Coolant Pumps in operation when the reactor trip system breakers are closed; however, only one Reactor Coolant Pump need be in operation when the reactor trip system breakers are open.

STATE the basis for the difference in Reactor Coolant Pump operability requirements depending on the Reactor Trip Breaker position.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.18 (2.00)

OMM-001, "Conduct of Operations", establishes a color coding scheme for MCR annunciators that will be on for an extended period of time. EXPLAIN what EACH of the following coded colors indicate regarding the status of the affected annunciator.

a. Green

b. Blue

c. Yellow

d. White

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.19 (1.00)

STATE the conditions under which the Shift Foreman may waive the requirement to independently verify a valve's position per PLP-702, "Independent Verification".

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.20 (3.00)

The concentration of the boric acid solution in the RWST shall be verified once per 7 days in accordance with Technical Specification 3.5.4. The plant chemist sampled the RWST on the following schedule:

April 1 --- April 8 --- April 15 --- April 23 --- May 1

ASSUME all samples were taken at 1200 hours. Answer EACH of the following with regard to the above conditions.

a. STATE whether the surveillance time interval requirements WERE or WERE NOT exceeded on April 23. JUSTIFY your response.

b. STATE whether the surveillance time interval requirements WERE or WERE NOT exceeded on May 1. JUSTIFY your response.

(***** END OF CATEGORY 08 *****)

(************* END OF EXAMINATION ***************)

$F = ma$

$v = s/t$

Cycle efficiency = (Net work out)/(Energy in)

$w = mg$

$s = V_o t + 1/2\ at^2$

$E = mc^2$

$KE = 1/2\ mv^2$

$a = (V_f - V_o)/t$

$A = \lambda N$

$A = A_o e^{-\lambda t}$

$PE = mgh$

$V_f = V_o + at$

$w = \theta/t$

$\lambda = \ln 2/t_{1/2} = 0.693/t_{1/2}$

$W = v\ \Delta P$

$A = \dfrac{\pi D^2}{4}$

$t_{1/2}eff = \dfrac{[(t_{1/2})(t_b)]}{[(t_{1/2}) + (t_b)]}$

$\Delta E = 931\ \Delta m$

$\dot{m} = V_{av}A\rho$

$\dot{Q} = \dot{m}\Delta h$

$I = I_o e^{-\Sigma x}$

$\dot{Q} = \dot{m}Cp\Delta t$

$\dot{Q} = UA\Delta T$

$I = I_o e^{-\mu x}$

$Pwr = W_f \Delta h$

$I = I_o\ 10^{-x/TVL}$

$TVL = 1.3/\mu$

$HVL = -0.693/\mu$

$P = P_o 10^{sur(t)}$

$P = P_o e^{t/T}$

$SUR = 26.06/T$

$SCR = S/(1 - K_{eff})$

$CR_x = S/(1 - K_{effx})$

$SUR = 26\rho/\ell^* + (\beta - \rho)T$

$CR_1(1 - K_{eff1}) = CR_2(1 - K_{eff2})$

$T = (\ell^*/\rho) + [(\beta - \rho)/\bar{\lambda}\rho]$

$M = 1/(1 - K_{eff}) = CR_1/CR_o$

$T = \ell/(\rho - \beta)$

$M = (1 - K_{effo})/(1 - K_{eff1})$

$T = (\beta - \rho)/(\bar{\lambda}\rho)$

$SDM = (1 - K_{eff})/K_{eff}$

$\rho = (K_{eff}-1)/K_{eff} = \Delta K_{eff}/K_{eff}$

$\ell^* = 10^{-4}$ seconds

$\bar{\lambda} = 0.1\ \text{seconds}^{-1}$

$\rho = [(\ell^*/(T\ K_{eff})] + [\bar{\beta}_{eff}/(1 + \bar{\lambda}T)]$

$I_1 d_1 = I_2 d_2$

$I_1 d_1^{\ 2} = I_2 d_2^{\ 2}$

$P = (\Sigma\phi V)/(3 \times 10^{10})$

$\Sigma = \sigma N$

$R/hr = (0.5\ CE)/d^2 (meters)$

$R/hr = 6\ CE/d^2\ (feet)$

Water Parameters

1 gal. = 8.345 lbm.
1 gal. = 3.78 liters
1 ft^3 = 7.48 gal.
Density = 62.4 lbm/ft^3
Density = 1 gm/cm^3
Heat of vaporization = 970 Btu/lbm
Heat of fusion = 144 Btu/lbm
1 Atm = 14.7 psi = 29.9 in. Hg.
1 ft. H_2O = 0.4335 lbf/in.

Miscellaneous Conversions

1 curie = 3.7×10^{10} dps
1 kg = 2.21 lbm
1 hp = 2.54×10^3 Btu/hr
1 mw = 3.41×10^6 Btu/hr
1 in = 2.54 cm
°F = 9/5°C + 32
°C = 5/9 (°F-32)
1 BTU = 778 ft-lbf

e = 2.718

1.0 PURPOSE/ENTRY CONDITIONS

This procedure provides the necessary instructions for transferring the safety injection system and containment spray system to the recirculation mode.

This procedure along with all the procedures making up the SHNPP EOPs contain items used to satisfy regulatory commitments in the FSAR and Tech Specs.

2.0 OPERATOR ACTIONS

Instructions	Response Not Obtained

****

CAUTION

- o Do Steps 1 through 5 without delay.
- o SI recirculation flow to RCS must be maintained at all times.
- o Manual action may be required to restart safeguards equipment following a loss of offsite power after SI reset.

****

NOTE: Foldout applies.

---

1. Reset SI

Instructions | Response Not Obtained

****

CAUTION

One CCW cooling train non-essential safety related load will be isolated when CCW system is divided into two separate headers in Step 2. Maintain one operable flow path to the following: Excess letdown heat exchangers, RCDT heat exchanger, CCW to RCP thermal barriers and oil coolers, letdown heat exchanger, seal return heat exchanger, recycle evaporator, spent fuel pool heat exchanger.

****

2. Establish CCW Flow To The RHR Heat Exchangers:

   a. Open the following CCW valves:

      1CC-147
      1CC-167

   b. Verify CCW pumps – BOTH RUNNING

   c. Verify CCW to the RHR heat exchangers.

   d. Perform one of the following to establish two independent CCW systems:

      o Shut CCW pump A nonessential supply AND return valves:

        1CC-128
        1CC-99

      o Shut CCW pump B nonessential supply AND return valves:

        1CC-127
        1CC-113

Instructions	Response Not Obtained

****

CAUTION

The following sequence of steps to transfer to cold leg recirculation assumes operability of all safeguards equipment. The sequence may have to be revised to establish recirculating SI flow depending on equipment operability.

****

3. Establish Recirculation Suction Flowpath:

   a. Verify the CNMT sump isolation valves - OPEN:

      1SI-300
      1SI-301
      1SI-310
      1SI-311

   b. Shut the RHR suction valves from RWST:

      1SI-322
      1SI-323

Instructions	Response Not Obtained
c. Shut Low head SI Train A To Cold Leg Valve 1SI-340	c. Shut Low head SI Train B To Cold Leg Valve 1SI-341

AC 2/11

   d. Shut the CSIP alternate miniflow isolation valves:

      1CS-746
      1CS-752

   e. Open RHR discharge to CSIP suction valves:

      1RH-25
      1RH-63

   f. Shut RWST to CSIP suction valves:

      LCV-115B
      LCV-115D

Instructions	Response Not Obtained
4. Establish Recirculation Injection Flowpath:	
a. Open the alternate cold leg injection valve 1SI-52.	
b. Check CSIP A <u>AND</u> B - IN SERVICE	GO TO Step 4e.
c. Shut discharge cross connects: 1CS-217 1CS-219	c. Shut discharge cross connects: 1CS-218 1CS-220
d. GO TO Step 5.	
e. Check CSIP A <u>AND</u> C - IN SERVICE	GO TO Step 4h.
f. Shut discharge cross connects: 1CS-217 1CS-219	
g. GO TO Step 5.	
h. Shut discharge cross connects: 1CS-218 1CS-220	
5. Verify Flow On Both SI High Pressure Injection Headers	

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1 for 3-loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

b. Operation with less than 3 loops is governed by Specification 3.4.1.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig except during hydrostatic testing.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4, and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlock Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

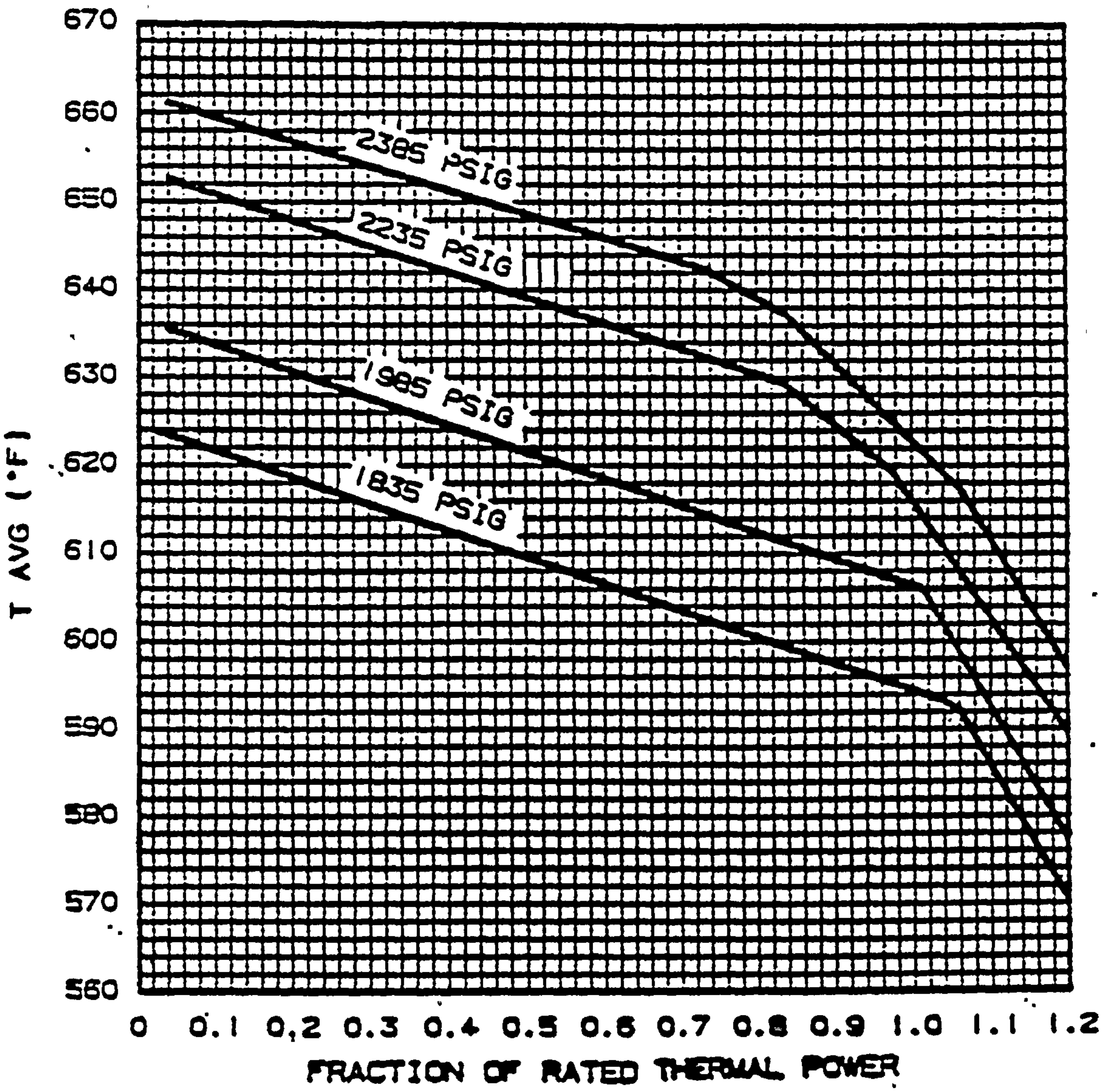

FIGURE 2.1-1

REACTOR CORE SAFETY LIMITS - THREE LOOPS IN OPERATION

APPLICABILITY (Continued)

ACTION:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 2.2-1, adjust the Setpoint consistent with the Trip Setpoint value.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, either:

1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 2.2-1 and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

2. Declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

Equation 2.2-1 $$Z + R + S \leq TA$$

Where:

Z = The value from Column Z of Table 2.2-1 for the affected channel,

R = The "as measured" value (in percent span) of rack error for the affected channel,

S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Error) of Table 2.2-1 for the affected channel, and

TA = The value from Column TA (Total Allowance) of Table 2.2-1 for the affected channel.

c. With a Reactor Trip System Instrumentation Channel or Interlock inoperable, take the appropriate ACTION shown in Table 3.3-1.

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding specifications is required during the OPERATIONAL MODES or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

3.0.2 Noncompliance with a specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required unless otherwise noted in the ACTION statement.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within 1 hour action shall be initiated to place the unit in a MODE in which the specification does not apply by placing it, as applicable, in:

a. At least HOT STANDBY within the next 6 hours,
b. At least HOT SHUTDOWN within the following 6 hours, and
c. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the action may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications.

This specification is not applicable in MODE 5 or 6.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the conditions for the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION requirements. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual specifications.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION: As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements specified in Table 4.3-1.

4.3.1.2 The REACTOR TRIP SYSTEM RESPONSE TIME of each Reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific Reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	9
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2#
b. Low Setpoint	4	2	3	1###, 2	2#
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2#
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2#
5. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
6. Source Range, Neutron Flux					
a. Startup	2	1	2	2##	4
b. Shutdown	2	1	2	3, 4, 5	5
7. Overtemperature ΔT	3	2	2	1, 2	6#
8. Overpower ΔT	3	2	2	1, 2	6#
9. Pressurizer Pressure--Low (Above P-7)	3	2	2	1	6#(1)
10. Pressurizer Pressure--High	3	2	2	1, 2	6#
11. Pressurizer Water Level--High (Above P-7)	3	2	2	1	6#

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
12. Reactor Coolant Flow--Low					
a. Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	6#
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop in each operating loop	1	6#
13. Steam Generator Water Level--Low-Low	3/stm. gen.	2/stm. gen. in any operating stm. gen.	2/stm. gen. each operating stm. gen.	1, 2	6#(1)
14. Steam Generator Water Level--Low Coincident With Steam/Feedwater Flow Mismatch	2 stm. gen. level and 2 stm./feedwater flow mismatch in each stm. gen.	1 stm. gen. level coincident with 1 stm./feedwater flow mismatch in same stm. gen.	1 stm. gen. level and 2 stm./feedwater flow mismatch in same stm. gen. or 2 stm. gen. level and 1 stm./feedwater flow mismatch in same stm. gen.	1, 2	6#
15. Undervoltage--Reactor Coolant Pumps (Above P-7)	2/pump	2/train	2/train	1	6#

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
16. Underfrequency--Reactor Coolant Pumps (Above P-7)	2/pump	2/train	2/train	1	6#
17. Turbine Trip (Above P-7)					
a. Low Fluid Oil Pressure	3	2	2	1	6#
b. Turbine Throttle Valve Closure	4	4	1	1	10#
18. Safety Injection Input from ESF	2	1	2	1, 2	8
19. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2##	7
b. Low Power Reactor Trips Block, P-7					
1) P-10 Input or	4	2	3	1	7
2) P-13 Input	2	1	2	1	7
c. Power Range Neutron Flux, P-8	4	2	3	1	7
d. Power Range Neutron Flux, P-10	4	2	3	1, 2	7
e. Turbine Impulse Chamber Pressure, P-13	2	1	2	1	7

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
20. Reactor Trip Breakers	2	1	2	1, 2	8, 11
	2	1	2	3*, 4*, 5*	9
21. Automatic Trip and Interlock Logic	2	1	2	1, 2	8
	2	1	2	3*, 4*, 5*	9
22. Reactor Trip Bypass Breakers	2	1	1	**	12

TABLE 3.3-1 (Continued)

TABLE NOTATIONS

*When the Reactor Trip System breakers are closed and the Control Rod Drive System is capable of rod withdrawal.

**Whenever Reactor Trip Breakers are to be tested.

#The provisions of Specification 3.0.4 are not applicable.

##Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

###Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

(1)The applicable MODES and ACTION Statement for these channels noted in Table 3.3-3 are more restrictive and, therefore, applicable.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours,

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1, and

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint, and

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - a. With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip System breakers, and verify compliance with the shutdown margin requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

b. With no channels OPERABLE, open the Reactor Trip System breakers within 1 hour and suspend all operations involving positive reactivity changes. Verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.

ACTION 7 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip System breakers within the next hour.

ACTION 10 - With the number of OPERABLE channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within 6 hours.

ACTION 11 - With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 8. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

ACTION 12 - No additional corrective actions are required.

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS Instrumentation or Interlock Trip Setpoint trip less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 3.3-4, adjust the Setpoint consistent with the Trip Setpoint value.

b. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Allowable Value column of Table 3.3-4, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 3.3-4, and determine within 12 hours that Equation 3.3-1 was satisfied for the affected channel, or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-3 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

   Equation 3.3-1 $Z + R + S \leq TA$

   Where:

   Z = The value from Column Z of Table 3.3-4 for the affected channel,

   R = The "as measured" value (in percent span) of rack error for the affected channel,

   S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Error) of Table 3.3-4 for the affected channel, and

   TA = The value from Column TA (Total Allowance) of Table 3.3-4 for the affected channel.

c. With an ESFAS instrumentation channel or interlock inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the ESFAS Instrumentation Surveillance Requirements specified in Table 4.3-2.

4.3.2.2 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" column of Table 3.3-3.

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Ventilation Isolation, Phase A Containment Isolation, Start Auxiliary Feedwater System Motor-Driven Pumps, Start Containment Fan Coolers, Start Emergency Service Water Pumps, Start Emergency Service Water Booster Pumps)					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure--High-1	3	2	2	1, 2, 3, 4	15*
d. Pressurizer Pressure--Low	3	2	2	1, 2, 3#	15*
e. Steam Line Pressure--Low	3/steam line	2/steam line in any steam line	2/steam line	1, 2, 3#	15*

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
2. Containment Spray					
a. Manual Initiation	2	1 with 2 coincident switches	2	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure--High-3	4	2	3	1, 2, 3	16
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	18
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
b. Phase "B" Isolation					
1) Manual Containment Spray Initiation	See Item 2.a. above for Manual Containment Spray initiating functions and requirements.				

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. Containment Isolation (Continued)					
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Containment Pressure--High-3	See Item 2.c. above for Containment Pressure High-3 initiating functions and requirements.				
c. Containment Ventilation Isolation					
1) Manual Containment Spray Initiation	See Item 2.a. above for Manual Containment Spray initiating functions and requirements.				
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4, 6**	17, 25
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
4) Containment Radioactivity					
a. Area Monitors (both preentry and normal purges)	4	See Table 3.3-6, Item 1a, for initiating functions and requirements.			
b. Airborne Gaseous Radioactivity					

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. Containment Isolation (Continued)					
(1) RCS Leak Detection (normal purge)	1	See Table 3.3-6, Item 1b1, for initiating functions and requirements.			
(2) Preentry Purge Detector	1	See Table 3.3-6, Item 1b2, for initiating functions and requirements.			
c) Airborne Particulate Radioactivity					
(1) RCS Leak Detection (normal purge)	1	See Table 3.3-6, Item 1C1, for initiating functions and requirements.			
(2) Preentry Purge Detector	1	See Table 3.3-6, Item 1C2, for initiating functions and requirements.			
5) Manual Phase "A" Isolation	See Item 3.a.1) above for Manual Phase "A" Isolation initiating functions and requirements.				
4. Main Steam Line Isolation					
a. Manual Initiation					
1) Individual MSIV Closure	1/steam line	1/steam line	1/operating steam line	1, 2, 3, 4	23
2) System	2	1	2	1, 2, 3	22

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Main Steam Line Isolation (Continued)					
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	21
c. Containment Pressure--High-2	3	2	2	1, 2, 3	15*
d. Steam Line Pressure--Low	See Item 1.e. above for Steam Line Pressure--Low initiating functions and requirements.				
e. Negative Steam Line Pressure Rate--High	3/steam line	2 in any steam line	2/steam line	3***, 4***	15*
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2	24
b. Steam Generator Water Level--High-High (P-14)	4/stm. gen.	2/stm. gen. in any stm. gen.	3/stm. gen. in each stm. gen.	1, 2	19*
c. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. Auxiliary Feedwater					
a. Manual Initiation	1/pump	1/pump	1/pump	1, 2, 3	23
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	21
c. Steam Generator Water Level--Low-Low					
1) Start Motor-Driven Pumps	3/stm. gen.	2/stm. gen. in any stm. gen.	2/stm. gen. in each stm. gen.	1, 2, 3	15*
2) Start Turbine-Driven Pump	3/stm. gen.	2/stm. gen. in any 2 stm. gen.	2/stm. gen. in each stm. gen.	1, 2, 3	15*
d. Safety Injection Start Motor-Driven Pumps	See Item 1. above for all Safety Injection initiating functions and requirements.				
e. Loss-of-Offsite Power Start Motor-Driven Pumps and Turbine-Driven Pump	See Item 9. below for Loss of Offsite Power initiating functions and requirements.				
f. Trip of All Main Feedwater Pumps Start Motor-Driven Pumps	1/pump	1/pump	1/pump	1, 2	15*

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. Auxiliary Feedwater (Continued)					
g. Steam Line Differential Pressure--High	3/steam line	2/steam line twice with any steamline low	2/steam line	1, 2, 3	15*
Coincident With Main Steam Line Isolation (Causes AFW Isolation)	See Item 4. above for all Steam Line Isolation initiating functions and requirements				
7. Safety Injection Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
b. RWST Level--Low-Low	4	2	3	1, 2, 3, 4	16
Coincident With Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
8. Containment Spray Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
8. Containment Spray Switchover to Containment Sump (Continued)					
b. RWST--Low Low	See Item 7.b. above for all RWST--Low Low initiating functions and requirements.				
Coincident With Containment Spray	See Item 2 above for all Containment Spray initiating functions and requirements.				
9. Loss-of-Offsite Power					
a. 6.9 kV Emergency Bus--Undervoltage Primary	3/bus	2/bus	2/bus	1, 2, 3, 4	15*
b. 6.9 kV Emergency Bus--Undervoltage Secondary	3/bus	2/bus	2/bus	1, 2, 3, 4	15*
10. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure,					
P-11	3	2	2	1, 2, 3	20
Not P-11	3	2	2	1, 2, 3	20
b. Low-Low T_{avg}, P-12	3	2	2	1, 2, 3	20
c. Reactor Trip, P-4	2	2	2	1, 2, 3	22
d. Steam Generator Water Level, P-14	See Item 5.b. above for all P-14 initiating functions and requirements.				

TABLE 3.3-3 (Continued)

TABLE NOTATIONS

*The provisions of Specification 3.0.4 are not applicable.

#Trip function may be blocked in this MODE below the P-11 (Pressurizer Pressure Interlock) Setpoint.

**During CORE ALTERATIONS or movement of irradiated fuel in containment, refer to Specification 3.9.9.

***Trip function automatically blocked above P-11 and may be blocked below P-11 when Safety Injection on low steam line pressure is not blocked.

ACTION STATEMENTS

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 16 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ACTION 17 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the Containment Purge Makeup and Exhaust Isolation valves are maintained closed while in MODES 1, 2, 3 and 4 (refer to Specification 3.6.1.7). For MODE 6, refer to Specification 3.9.4.

ACTION 18 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 19 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour, and

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.2.1.

ACTION 20 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 21 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 22 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 23 - With the number of OPERABLE channels one less than the Total Number of Channels, declare the associated equipment inoperable and take the appropriate ACTION required in accordance with the specific equipment specification.

ACTION 24 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 25 - During CORE ALTERATIONS or movement of irradiated fuel within containment, comply with the ACTION statement of Specification 3.9.9.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 All reactor coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2.*

ACTION:

With less than the above required reactor coolant loops in operation, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1 The above required reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

*See Special Test Exceptions Specification 3.10.4.

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least two of the reactor coolant loops listed below shall be OPERABLE with two reactor coolant pumps in operation when the Reactor Trip System breakers are closed or with one reactor coolant pump in operation when the Reactor Trip System breakers are open:*

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,

b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,

APPLICABILITY: MODE 3.**

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With only one reactor coolant loop in operation and the Reactor Trip System breakers in the closed position, within 1 hour open the Reactor Trip System breakers.

c. With no reactor coolant loop in operation, immediately open the Reactor Trip System breakers, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required reactor coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

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*All reactor coolant pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**See Special Test Exception 3.10.4.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying narrow range secondary side water level to be greater than or equal to 10% at least once per 12 hours.

4.4.1.2.3 The required reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loops listed below shall be OPERABLE and at least one of these loops shall be in operation:*

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,**

b: Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,**

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,**

d. RHR Loop [A], or

e. RHR Loop [B].

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is an RHR loop, be in COLD SHUTDOWN within 24 hours.

b. With no loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required loop to operation.

*All reactor coolant pumps and RHR pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 335°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required reactor coolant pump(s), if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying narrow range secondary side water level to be greater than or equal to 10% at least once per 12 hours.

4.4.1.3.3 At least one reactor coolant or RHR loop shall be verified in operation and circulating reactor coolant at least once per 12 hours.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation*, and either:

a. One additional RHR loop shall be OPERABLE**, or

b. The narrow range secondary side water level of at least two steam generators shall be greater than 10%.

APPLICABILITY: MODE 5 with reactor coolant loops filled***.

ACTION:

a. With one of the RHR loops inoperable and with less than the required steam generator water level, immediately initiate corrective action to return the inoperable RHR loop to OPERABLE status or restore the required steam generator water level as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The narrow range secondary side water level of at least two steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

*The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

***A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 335°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.2 Two residual heat removal (RHR) loops shall be OPERABLE* and at least one RHR loop shall be in operation.**

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

a. With less than the above required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.2 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

*One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

**The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer Code safety valve shall be OPERABLE with a lift setting of 2485 psig ± 1%.*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer Code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

REACTOR COOLANT SYSTEM

OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting of 2485 psig ± 1%.*

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one pressurizer Code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE with:

a. The isolation valve open with power supply circuit breaker open,

b. A contained borated water volume of between 66 and 96% indicated level,

c. A boron concentration of between 2000 and 2200 ppm, and

d. A nitrogen cover-pressure of between 585 and 665 psig.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying, by the absence of alarms, the contained borated water volume and nitrogen cover-pressure in the tanks, and

2. Verifying that each accumulator isolation valve is open.

*RCS pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 76 gallons, which is equivalent to an indicated level change of 9% by verifying the boron concentration of the accumulator solution; and

c. At least once per 31 days when the RCS pressure is above 1000 psig by verifying that the circuit breaker supplying power to the respective isolation valve operator is open.

4.5.1.2 Each accumulator water level and pressure channel shall be demonstrated OPERABLE at least once per 18 months by the performance of a CHANNEL CALIBRATION.

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE charging/safety injection pump,

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and, upon being manually aligned, transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

SURVEILLANCE REQUIREMENTS (Continued)

CP&L Valve No.	EBASCO Valve No.	Valve Function	Valve Position*
1SI-107	2SI-V500SA-1	High Head Safety Injection to Reactor Coolant System Hot Legs	Closed-1
1SI-86	2SI-V501SB-1	High Head Safety Injection to Reactor Coolant System Hot Legs	Closed-1
1SI-52	2SI-V502SA-1	High Head Safety Injection to Reactor Coolant System Cold Legs	Closed-1
1SI-340	2SI-V579SA-1	Low Head Safety Injection to Reactor Coolant System Cold Legs	Open-1
1SI-341	2SI-V578SB-1	Low Head Safety Injection to Reactor Coolant System Cold Legs	Open-1
1SI-359	2SI-V587SA-1	Low Head Safety Injection to Reactor Coolant System Hot Legs	Closed-1

b. At least once per 31 days by:

1. Verifying that the ECCS piping is full of water by venting accessible discharge piping high points, and

2. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

*Closed-1 and Open-1--The Control Power Disconnect Switch shall be maintained in the "OFF" position and the valve control switch shall be maintained in the valve position noted above.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months by:

1. Verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant System by ensuring that:

   a) With a simulated or actual Reactor Coolant System pressure signal greater than or equal to 425 psig the interlocks prevent the valves from being opened, and

   b) With a simulated or actual Reactor Coolant System pressure signal less than or equal to 750 psig the interlocks will cause the valves to automatically close.

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

e. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on safety injection actuation test signal and on safety injection switchover to containment sump from an RWST Lo-Lo level test signal, and

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection actuation test signal:

   a) charging/safety injection pump,
   b) RHR pump.

f. By verifying that each of the following pumps develops the required differential pressure when tested pursuant to Specification 4.0.5:

1. charging/safety injection pump (Refer to Specification 4.1.2.4)
2. RHR pump $\geq$ 100 psid at a flow rate of at least 3663 gpm.

g. By verifying that the locking mechanism is in place and locked for the following ECCS throttle valves:

1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE, and

2. At least once per 18 months.

SURVEILLANCE REQUIREMENTS (Continued)

HPSI SYSTEM EBASCO Valve No.	HPSI SYSTEM CP&L Valve No.
2SI-V440SA-1	1SI-5
2SI-V439SB-1	1SI-6
2SI-V438SA-1	1SI-7
2SI-V437SA-1	1SI-69
2SI-V436SB-1	1SI-70
2SI-V435SA-1	1SI-71
2SI-V434SA-1	1SI-101
2SI-V433SB-1	1SI-102
2SI-V432SA-1	1SI-103
2SI-V431SA-1	1SI-124
2SI-V430SB-1	1SI-125
2SI-V429SA-1	1SI-126

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

1. For charging/safety injection pump lines, with a single pump running:

   a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 379 gpm, and

   b) The total pump flow rate is less than or equal to 685 gpm.

2. For RHR pump lines, with a single pump running, the sum of the injection line flow rates is greater than or equal to 3663 gpm.

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} LESS THAN 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE charging/safety injection pump,*

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank upon being manually realigned and transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the charging/safety injection pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 24 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the residual heat removal heat exchanger or RHR pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

*A maximum of one charging/safety injection pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 335°F.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable requirements of Specification 4.5.2.

4.5.3.2 All charging/safety injection pumps, except the above allowed OPERABLE pump, shall be demonstrated inoperable* by verifying that the motor circuit breakers are secured in the open position prior to the temperature of one or more of the RCS cold legs decreasing below 335°F and at least once per 31 days thereafter.

*An inoperable pump may be energized for testing provided the discharge of the pump has been isolated from the RCS by a closed isolation valve with the power removed from the valve operator or by a manual valve secured in the closed position.

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained borated water volume of 436,000 gallons, which is equivalent to 92% indicated level.

b. A boron concentration of between 2000 and 2200 ppm of boron,

c. A minimum solution temperature of 40°F, and

d. A maximum solution temperature of 125°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank, and
2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is less than 40°F or greater than 125°F.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two separate and independent diesel generators, each with:

1. A separate day tank containing a minimum of 2670 gallons of fuel, which is equivalent to 85% indicated level,

2. A separate main fuel oil storage tank containing a minimum of 100,000 gallons of fuel, and

3. A separate fuel oil transfer pump.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one offsite circuit of 3.8.1.1.a inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If either emergency diesel generator (EDG) has not been successfully tested within the 24 hours preceding entry into this ACTION, demonstrate its OPERABILITY by performing Surveillance Requirement 4.8.1.1.2.a.4 and a.6 separately for each such EDG within 24 hours. Restore the offsite circuit to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

b. With one diesel generator of 3.8.1.1.b inoperable, demonstrate the OPERABILITY of the A.C. offsite sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; and if the EDG became inoperable due to any cause other than preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE EDG performing Surveillance Requirement 4.8.1.1.2.a.4 and a.6 within 24 hours*#; restore the

*This test is required to be completed regardless of when the inoperable EDG is restored to OPERABILITY.

#Activities that normally support testing pursuant to 4.8.1.1.2.a.4 and a.6, which would render the diesel inoperable (e.g., air roll), shall not be performed for testing required by this ACTION statement.

LIMITING CONDITION FOR OPERATION

ACTION (Continued):

diesel generator to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. See also ACTION d. below.

c. With one offsite circuit of 3.8.1.1.a and one diesel generator inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; and if the EDG became inoperable due to any cause other than preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE EDG by performing Surveillance Requirement 4.8.1.1.2.a.4 and a.6 within 8 hours*#; restore one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. See also ACTION d. below. Restore the other A.C. power source (offsite circuit or diesel generator) to OPERABLE status in accordance with the provisions of Specification 3.8.1.1 ACTION a or b, as appropriate with the time requirement of that ACTION based on the time of initial loss of the remaining inoperable A.C. power source. A successful test of diesel OPERABILITY per Surveillance Requirement 4.8.1.1.2.a.4 and a.6 performed under this ACTION for an OPERABLE diesel or a restored to OPERABLE diesel satisfies the EDG test requirement of ACTION a or b.

d. With one diesel generator inoperable, in addition to ACTION b and c above, verify that:

   1. All required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE. If these conditions are not satisfied within 2 hours, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours, except as provided for in ACTION d.2 below.

   2. If in MODES 1, 2, or 3 and the result of the inoperable diesel generator is that three auxiliary feedwater pumps are inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

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*This test is required to be completed regardless of when the inoperable EDG is restored to OPERABILITY.

#Activities that normally support testing pursuant to 4.8.1.1.2.a.4 and a.6, which would render the diesel inoperable (e.g., air roll), shall not be performed for testing required by this ACTION statement.

LIMITING CONDITION FOR OPERATION

ACTION (Continued):

e. With two of the required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by sequentially performing Surveillance Requirement 4.8.1.1.2.a.4 and a.6 on both diesels within 8 hours#, unless the diesel generators are already operating; restore one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. Following restoration of one offsite source, follow ACTION a. with the time requirement of that ACTION based on the time of initial loss of the remaining inoperable offsite A.C. circuit. A successful test(s) of diesel OPERABILITY per Surveillance Requirement 4.8.1.1.2.a.4 and a.6 performed under this ACTION for the OPERABLE diesels satisfies the EDG test requirement of ACTION a.

f. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; restore one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Following restoration of one diesel generator unit, follow ACTION b. with the time requirement of that ACTION based on the time of initial loss of the remaining inoperable diesel generator. A successful test of diesel OPERABILITY per Surveillance Requirement 4.8.1.1.2.a.4 and a.6 performed under this ACTION for a restored-to-OPERABLE diesel satisfies the EDG test requirement of ACTION b.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignment and power availability, and

b. Demonstrated OPERABLE at least once per 18 months by manually transferring the onsite Class 1E power supply from the unit auxiliary transformer to the startup auxiliary transformer.

#Activities that normally support testing pursuant to 4.8.1.1.2.a.4 and a.6, which would render the diesel inoperable (e.g., air roll), shall not be performed for testing required by this ACTION statement.

ELECTRICAL POWER SYSTEMS

A.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator with:

   1. Day tank containing a minimum volume of 2670 gallons of fuel, which is equivalent to 85% indicated level,

   2. A separate main fuel oil storage tank containing a minimum volume of 100,000 gallons of fuel, and

   3. A fuel oil transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel, or crane operation with loads over irradiated fuel and within 8 hours, depressurize and vent the Reactor Coolant System through a vent of greater than or equal to 2.9 square inches. In addition, when in MODE 5 with the reactor coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the requirements of Specifications 4.8.1.1.1 and 4.8.1.1.2.

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following D.C. electrical sources shall be OPERABLE:

a. 125-volt Emergency Battery Bank 1A-SA and either full capacity charger, 1A-SA or 1B-SA, and,

b. 125-volt Emergency Battery Bank 1B-SB and either full capacity charger, 1A-SB or 1B-SB:

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one of the required D.C. electrical sources inoperable, restore the inoperable D.C. electrical source to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt Emergency Battery and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The parameters in Table 4.8-2 meet the Category A limits, and

2. The total battery terminal voltage is greater than or equal to 129 volts on float charge.

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 110 volts, or battery overcharge with battery terminal voltage above 150 volts, by verifying that:

1. The parameters in Table 4.8-2 meet the Category B limits,

2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150×10^{-6} ohm, and

3. The average electrolyte temperature of 10 connected cells is above 70° F.

ELECTRICAL POWER SYSTEMS

D.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, one 125-volt Emergency Battery (either 1A-SA or 1B-SB) and at least one associated full-capacity charger shall be OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With the required Emergency Battery or full-capacity charger inoperable, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel; initiate corrective action to restore the required Emergency Battery and full-capacity charger to OPERABLE status as soon as possible, and within 8 hours, depressurize and vent the Reactor Coolant System through a vent of $\geq$ 2.9 square inches.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The above required 125-volt Emergency Battery and full-capacity charger shall be demonstrated OPERABLE in accordance with Specification 4.8.2.1.

ELECTRICAL POWER SYSTEMS

3/4.8.3 ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following electrical buses shall be energized in the specified manner with tie breakers open between redundant buses within the unit:

a. Division A ESF A.C. Buses consisting of:

   1. 6900-volt Bus 1A-SA.
   2. 480-volt Bus 1A2-SA.
   3. 480-volt Bus 1A3-SA.

b. Division B ESF A.C. Buses consisting of:

   1. 6900-volt Bus 1B-SB.
   2. 480-volt Bus 1B2-SB.
   3. 480-volt Bus 1B3-SB.

c. 118-volt A.C. Vital Bus 1DP-1A-SI energized from its associated inverter connected to 125-volt D.C. Bus DP-1A-SA*,

d. 118-volt A.C. Vital Bus 1DP-1A-SIII energized from its associated inverter connected to 125-volt D.C. Bus DP-1A-SA*,

e. 118-volt A.C. Vital Bus 1DP-1B-SII energized from its associated inverter connected to 125-volt D.C. Bus DP-1B-SB*,

f. 118-volt A.C. Vital Bus 1DP-1B-SIV energized from its associated inverter connected to 125-volt D.C. Bus DP-1B-SB*,

g. 125-volt D.C. Bus DP-1A-SA energized from Emergency Battery 1A-SA and charger 1A-SA or 1B-SA, and

h. 125-volt D.C. Bus DP-1B-SB energized from Emergency Battery 1B-SB and charger 1B-SB or 1A-SB

APPLICABILITY: MODES 1, 2, 3, and 4.

*Two inverters may be disconnected from their 125-volt D.C. bus for up to 24 hours as necessary, for the purpose of performing an equalizing charge on their associated Emergency Battery provided: (1) their vital buses are energized and (2) the vital buses associated with the other Emergency Battery are energized from their associated inverters and connected to their associated 125-volt D.C. bus.

LIMITING CONDITION FOR OPERATION

ACTION:

a. With one of the required divisions of A.C. ESF buses not fully energized, reenergize the division within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one 118-volt A.C. vital bus not energized from its associated inverter, reenergize the 118-volt A.C. vital bus within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one 118-volt A.C. vital bus not energized from its associated inverter connected to its associated D.C. bus, re-energize the 118-volt A.C. vital bus through its associated inverter connected to its associated D.C. bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With either 125-volt D.C. bus 1A-SA or 1B-SB not energized from its associated Emergency Battery, reenergize the D.C. bus from its associated Emergency Battery within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified buses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the buses.

ONSITE POWER DISTRIBUTION

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, one of the following divisions of electrical buses shall be energized in the specified manner:

a. Division A, consisting of:

1. 6900-volt Bus 1A-SA and
2. 480 volt Buses 1A2-SA and 1A3-SA, and
3. 118-volt A.C. Vital Buses 1DP-1A-SI and 1DP-1A-SIII energized from their associated inverter connected to 125-volt D.C. Bus DP-1A-SA, and
4. 125-volt D.C. Bus DP-1A-SA energized from Emergency Battery 1A-SA and chargers 1A-SA or 1B-SA, or

b. Division B, consisting of:

1. 6900-volt Bus 1B-SB and
2. 480-volt Buses 1B2-SB and 1B3-SB, and
3. 118-volt AC Vital Buses 1DP-1B-SII and 1DP-1B-SIV energized from their associated inverter connected to 125-volt D.C. Bus. DP-1B-SB, and
4. 125-volt D.C. Bus DP-1B-SB energized from Emergency Battery 1B-SB and chargers 1B-SB or 1A-SB.

APPLICABILITY MODES 5 and 6.

ACTION:

With any of the above required electrical buses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel; initiate corrective action to energize the required electrical buses in the specified manner as soon as possible; and within 8 hours, depressurize and vent the RCS through a vent of $\geq$ 2.9 square inches.

SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified buses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the buses.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6, with irradiated fuel in the vessel when the water level above the top of the reactor vessel flange is greater than or equal to 23 feet.

ACTION:

With no RHR loop OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.1 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 2500 gpm at least once per 12 hours.

*The RHR loop may be removed from operation for up to 1 hour per 2-hour period during the performance of CORE ALTERATIONS and core loading verification in the vicinity of the reactor vessel hot legs.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent residual heat removal (RHR) loops shall be OPERABLE, and at least one RHR loop shall be in operation.*

APPLICABILITY: MODE 6, with irradiated fuel in the vessel when the water level above the top of the reactor vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status or to establish greater than or equal to 23 feet of water above the reactor vessel flange as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 2500 gpm at least once per 12 hours.

*The operating RHR loop may be removed from operation for up to 1 hour per 2-hour period during the performance of CORE ALTERATIONS and core loading verification in the vicinity of the reactor vessel hot legs.

3/4.9.10 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor vessel flange.

APPLICABILITY: MODE 6, during movement of fuel assemblies or control rods within the containment when either the fuel assemblies being moved or the fuel assemblies seated within the reactor vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or control rods within the reactor vessel or containment (after placing assemblies in transit in a safe condition).

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or control rods.

3/4.9.11 WATER LEVEL - NEW AND SPENT FUEL POOLS

LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in a pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the affected pool area and restore the water level to within its limit within 4 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 At least once per 7 days, when irradiated fuel assemblies are in a pool, the water level in that pool shall be determined to be at least its minimum required depth.

6.2.3 ONSITE NUCLEAR SAFETY (ONS) UNIT

FUNCTION

6.2.3.1 The ONS Unit shall function to examine unit operating characteristics, NRC issuances, industry advisories (including information forwarded by INPO from their evaluation of all industry LERs), and other sources of unit design and operating experience information, including units of similar design, which may indicate areas for improving unit safety. The ONS Unit shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities, or other means of improving unit safety, to appropriate levels of management, up to and including the Senior Vice President-Operations Support, if necessary.

COMPOSITION

6.2.3.2 The ONS Unit shall be composed of at least five, dedicated, full-time engineers located on site. Each shall have a baccalaureate degree in engineering or related science and at least 2 years professional level experience in his field, at least 1 year of which experience shall be in the nuclear field.

RESPONSIBILITIES

6.2.3.3 The ONS Unit shall be responsible for maintaining surveillance of unit activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

RECORDS

6.2.3.4 Records of activities performed by the ONS Unit shall be prepared, maintained, and forwarded each calendar month to the Manager-Nuclear Safety and Environmental Services.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall provide advisory technical support to the Shift Foreman in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. The Shift Technical Advisor shall have a baccalaureate degree or equivalent in a scientific or engineering discipline and shall have received specific training in the response and analysis of the unit for transients and accidents, and in unit design and layout, including the capabilities of instrumentation and controls in the control room.

6.3 Deleted

*Not responsible for sign-off function.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Manager-Harris Training Unit and shall meet or exceed the requirements and recommendations of the September 1979 draft of ANS 3.1, with the exceptions and alternatives noted on FSAR pages 1.8-8 (Am.20), 1.8-9 (Am.26), 1.8-10 (Am.27), 1.8-11 (Am.27), 1.8-12 (Am.27), and 1.8-13 (Am.27), and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.

6.5 REVIEW AND AUDIT

6.5.1 SAFETY AND TECHNICAL REVIEWS

6.5.1.1 General Program Control

6.5.1.1.1 A safety and a technical evaluation shall be prepared for each of the following:

a. All procedures and programs required by Specification 6.8, other procedures that affect nuclear safety, and changes thereto;

b. All proposed tests and experiments that are not described in the Final Safety Analysis Report; and

c. All proposed changes or modifications to plant systems or equipment that affect nuclear safety.

6.5.1.2 Technical Evaluations

6.5.1.2.1 Technical evaluations will be performed by personnel qualified in the subject matter and will determine the technical adequacy and accuracy of the proposed activity. If interdisciplinary evaluations are required to cover the technical scope of an activity, they will be performed.

6.5.1.2.2 Technical review personnel will be identified by the responsible Manager or his designee for a specific activity when the review process begins.

6.5.1.3 Qualified Safety Reviewers

6.5.1.3.1 The Plant General Manager shall designate those individuals who will be responsible for performing safety reviews described in Specification 6.5.1.4.

Qualified Safety Reviewers (Continued)

These individuals shall have a baccalaureate degree in an engineering or related field or equivalent, and 2 years of related experience. Such designation shall include the disciplines or procedure categories for which each individual is qualified. Qualified individuals or groups not on the plant staff (as shown on Figure 6.2-2) may be relied upon to perform safety reviews if so designated by the Plant General Manager.

6.5.1.4 Safety Evaluations and Approvals

6.5.1.4.1 The safety evaluation prepared in accordance with Specification 6.5.1.1.1 shall include a written determination, with basis, of whether or not the procedures or changes thereto, proposed tests and experiments and changes thereto, and modifications constitute an unreviewed safety question as defined in Paragraph 50.59 of 10 CFR Part 50, or whether they involve a change to the Final Safety Analysis Report, the Technical Specifications, or the Operating License.

6.5.1.4.2 The safety evaluation shall be prepared by a qualified individual. The safety evaluation shall be reviewed by a second qualified individual.

6.5.1.4.3 A safety evaluation and subsequent review that conclude that the subject action may involve an unreviewed safety question, a change to the Technical Specifications, or a change to the Operating License, will be referred to the Plant Nuclear Safety Committee (PNSC) for their review in accordance with Specification 6.5.2.6. If the PNSC recommendation is that an item is an unreviewed safety question, a change to the Technical Specifications, or a change to the Operating License, the action will be referred to the Commission for approval prior to implementation. Implementation may not proceed until after review by the Corporate Nuclear Safety Section in accordance with Specification 6.5.3.9.

6.5.1.4.4 If a safety evaluation and subsequent review conclude that the subject action does not involve an unreviewed safety question, a change to the Technical Specification, or a change to the Operating License, the action may be approved by the Plant General Manager or his designee or, as applicable, by the Manager of the primary functional area affected by the action. The individual approving the action shall assure that the reviewers collectively possess the background and qualification in all of the disciplines necessary and important to the specific review for both safety and technical aspects.

6.5.1.4.5 A safety evaluation and subsequent review that conclude that the subject action involves a change in the Final Safety Analysis Report shall be referred to the Corporate Nuclear Safety Section for review in accordance with Specification 6.5.3.9, but implementation may proceed prior to the completion of that review.

6.5.1.4.6 The individual approving the procedure, test, or experiment or change thereto shall be other than those who prepared the safety evaluation or performed the safety review.

6.5.2 PLANT NUCLEAR SAFETY COMMITTEE (PNSC)

FUNCTION

6.5.2.1 The PNSC shall function to advise the Plant General Manager on all matters related to nuclear safety.

COMPOSITION

6.5.2.2 The PNSC shall be composed of the:

Chairman:	Plant General Manager
Member:	Assistant Plant General Manager
Member:	Manager-Operations
Member:	Manager-Technical Support
Member:	Manager-Maintenance
Member:	Manager-Environmental and Radiation Control
Member:	Director-Plant Programs and Procedures
Member:	Director-Regulatory Compliance
Member:	Director-QA/QC-Harris Plant

6.5.2.3 The Chairman may designate in writing other regular members who may serve as Acting Chairman of PNSC meetings. All alternate members shall be appointed in writing by the PNSC Chairman. Alternates shall be designated for specific regular PNSC members and shall have expertise in the same general area as the regular member they represent. No more than two alternates shall participate as voting members in PNSC activities at any one time.

MEETING FREQUENCY

6.5.2.4 The PNSC shall meet at least once per calendar month and as convened by the PNSC Chairman or his designated alternate. The PNSC must meet in session to perform its function under these Technical Specifications.

QUORUM

6.5.2.5 The quorum of the PNSC necessary for the performance of the PNSC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and four members including alternates.

RESPONSIBILITIES

6.5.2.6 The PNSC shall be responsible for:

a. Review of proposed procedures or changes thereto that have been initially determined to constitute an unreviewed safety question or involve an unreviewed change to the Technical Specifications;

RESPONSIBILITIES (Continued)

b. Review of all proposed tests and experiments that affect nuclear safety and that have been initially determined to appear to constitute an unreviewed safety question or involve an unreviewed change to the Technical Specifications;

c. Review of all proposed changes to Appendix "A" Technical Specifications;

d. Review of all proposed changes or modifications to unit systems or equipment that affect nuclear safety and that have been initially determined to appear to constitute an unreviewed safety question as defined in 10 CFR 50.59 or involve a change to the Technical Specifications;

e. Investigation of all violations of the Technical Specifications, including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence, to the Vice President-Harris Nuclear Project and to the Manager-Corporate Nuclear Safety Section;

f. Review of all REPORTABLE EVENTS;

g. Review of unit operations to detect potential hazards to nuclear safety;

h. Performance of special reviews, investigations, or analyses and reports thereon as requested by the Plant General Manager or the Manager-Corporate Nuclear Safety Section;

i. Review of the Security Plan;

j. Review of the Emergency Plan;

k. Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations, and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Vice President-Harris Nuclear Project and the Manager-Nuclear Safety and Environmental Services;

l. Review, prior to implementation, of changes to the PROCESS CONTROL PROGRAM, the OFFSITE DOSE CALCULATION MANUAL, the Radwaste Treatment Systems, and the Technical Specification Equipment List Program.

6.5.2.7 The PNSC shall:

a. Render determinations in writing with regard to whether or not each item considered under Specification 6.5.2.6a. through e. constitutes an unreviewed safety question; and

RESPONSIBILITIES (Continued)

b. Provide written notification within 24 hours to the Vice President-Harris Nuclear Project and the Manager-Nuclear Safety and Environmental Services of disagreement between the PNSC and the Plant General Manager. However, the Plant General Manager shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

RECORDS

6.5.2.8 The PNSC shall maintain written minutes of each PNSC meeting that, at a minimum, document the results of all PNSC activities performed under the responsibility provisions of these Technical Specifications. Copies shall be provided to the Vice President-Harris Nuclear Project and the Manager-Nuclear Safety and Environmental Services.

6.5.3 CORPORATE NUCLEAR SAFETY SECTION

FUNCTION

6.5.3.1 The Corporate Nuclear Safety Section (CNSS) of the Nuclear Safety and Environmental Services Department shall function to provide independent review of plant changes, tests, and procedures; verify that REPORTABLE EVENTS are investigated in a timely manner and corrected in a manner that reduces the probability of recurrence of such events; and detect trends that may not be apparent to a day-to-day observer. They shall also evaluate all CP&L LERs for their potential applicability to other CP&L nuclear power plants.

ORGANIZATION

6.5.3.2 The individuals assigned responsibility for independent reviews shall be technically qualified in a specified technical discipline or disciplines. These individuals shall collectively have the experience and competence required to review activities in the following areas:

a. Nuclear power plant operations,
b. Nuclear engineering,
c. Chemistry and radiochemistry,
d. Metallurgy,
e. Instrumentation and control,
f. Radiological safety,
g. Mechanical and electrical engineering,
h. Administrative controls,
i. Quality assurance practices,
j. Nondestructive testing, and
k. Other appropriate fields associated with the unique characteristics.

ORGANIZATION (Continued)

6.5.3.3 The Manager-Corporate Nuclear Safety Section shall have a baccalaureate degree in an engineering or related field and, in addition, shall have a minimum of 10 years' related experience, of which a minimum of 5 years shall be in the operation and/or design of nuclear power plants.

6.5.3.4 The independent safety review program reviewers shall each have a baccalaureate degree in an engineering or related field or equivalent and, in addition, shall have a minimum of 5 years' related experience.

6.5.3.5 An individual may possess competence in more than one specialty area. If sufficient expertise is not available within the Corporate Nuclear Safety Section, competent individuals from other Carolina Power & Light Company organizations or outside consultants shall be utilized in performing independent reviews and investigations.

6.5.3.6 At least three individuals, qualified as discussed in Specification 6.5.3.2 above shall review each item submitted under the requirements of Specification 6.5.3.9.

6.5.3.7 Independent safety reviews shall be performed by individuals not directly involved with the activity under review or responsible for the activity under review.

6.5.3.8 The Corporate Nuclear Safety Section independent safety review program shall be conducted in accordance with written, approved procedures.

REVIEW

6.5.3.9 The Corporate Nuclear Safety Section shall perform reviews of the following:

a. Written safety evaluations for all procedures and programs required by Specification 6.8 and other procedures that affect nuclear safety and changes thereto, and proposed tests or experiments and proposed modifications, any of which constitute a change to the Final Safety Analysis Report. Implementation may proceed prior to completion of the review;

b. All procedures and programs required by Specification 6.8 and other procedures that affect nuclear safety and changes thereto that constitute an unreviewed safety question as defined in Paragraph 50.59 of 10 CFR Part 50 or involve a change to the Technical Specifications;

c. All proposed tests or experiments that constitute an unreviewed safety question as defined in Paragraph 50.59 of 10 CFR Part 50 or involve a change to the Technical Specifications prior to implementation;

d. All proposed changes to the Technical Specifications and Operating License;

REVIEW (Continued)

e. Violations, which require written notification to the Commission, of applicable codes, regulations, orders, Technical Specifications, license requirements, internal procedures or instructions having nuclear safety significance, significant operating abnormalities or deviations from normal and expected performance of plant safety-related structures, systems, or components;

f. All REPORTABLE EVENTS;

g. All proposed modifications that constitute an unreviewed safety question as defined in Paragraph 50.59 of 10 CFR Part 50 or involve a change to the Technical Specifications;

h. Any other matter involving safe operation of the nuclear power plant that the Manager-Corporate Nuclear Safety Section deems appropriate for consideration or which is referred to the Manager-Corporate Nuclear Safety Section by the onsite operating organization or other functional organizational units within Carolina Power & Light Company;

i. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety; and

j. Reports and minutes of the PNSC.

6.5.3.10 Review of items considered under Specification 6.5.3.9.e, h and j above shall include the results of any investigations made and the recommendations resulting from these investigations to prevent or reduce the probability of recurrence of the event.

RECORDS

6.5.3.11 Records of Corporate Nuclear Safety Section reviews, including recommendations and concerns, shall be prepared and distributed as indicated below:

a. Copies of documented reviews shall be retained in the CNSS files.

b. Recommendations and concerns shall be submitted to the Plant General Manager and Vice President-Harris Nuclear Project within 14 days of completion of the review. A report summarizing the reviews encompassed by Specification 6.5.3.9 shall be provided to the Plant General Manager and the Vice President-Harris Nuclear Project every other month.

c. A summation of recommendations and concerns of the Corporate Nuclear Safety Section shall be submitted to the Chairman/President and Chief Executive Officer and other appropriate senior management personnel at least every other month.

6.5.4 CORPORATE QUALITY ASSURANCE AUDIT PROGRAM

AUDITS

6.5.4.1 Audits of unit activities shall be performed by the Quality Assurance Services Section of the Corporate Quality Assurance Department. These audits shall encompass:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions, at least once per 12 months;

b. The training, qualifications, and performance as a group, of the entire unit staff, at least once per 12 months;

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems, or method of operation that affect nuclear safety, at least once per 6 months;

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months;

e. The fire protection programmatic controls including the implementing procedures, at least once per 24 months, by qualified licensee QA personnel;

f. The Radiological Environmental Monitoring Program and the results thereof, at least once per 12 months;

g. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures, at least once per 24 months;

h. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes, at least once per 24 months;

i. The performance of activities required by the Quality Assurance Program for effluent and environmental monitoring, at least once per 12 months;

j. The Emergency Plan and implementing procedures, at least once per 12 months;

k. The Security Plan and implementing procedures, at least once per 12 months; and

l. Any other area of unit operation considered appropriate by the Manager-Corporate Nuclear Safety or the Vice President-Harris Nuclear Project.

6.5.4.2 Personnel performing the quality assurance audits shall have access to the plant operating records.

RECORDS

6.5.4.3 Records of audits shall be prepared and retained.

6.5.4.4 Audit reports encompassed by Specification 6.5.4.1 shall be prepared, approved by the Manager—Quality Assurance Services, and forwarded, within 30 days after completion of the audit, to the Senior Executive Vice President-Power Supply and Engineering and Construction, Senior Vice President-Nuclear Generation, Vice President-Harris Nuclear Project, Manager—Nuclear Safety and Environmental Services, Plant General Manager, and the management positions responsible for the areas audited.

AUTHORITY

6.5.4.5 The Manager—Quality Assurance Service Section, under the Manager—Corporate Quality Assurance Department, shall be responsible for the following:

a. Administering the Corporate Quality Assurance Audit Program.

b. Approval of the individuals selected to conduct quality assurance audits.

6.5.4.6 Audit personnel shall be independent of the area audited.

6.5.4.7 Selection of personnel for auditing assignments shall be based on experience or training that establishes that their qualifications are commensurate with the complexity or special nature of the activities to be audited. In selecting audit personnel, consideration shall be given to special abilities, specialized technical training, prior pertinent experience, personal characteristics, and education.

6.5.4.8 Qualified outside consultants, or other individuals independent from those personnel directly involved in plant operation, shall be used to augment the audit teams when necessary.

6.5.5 OUTSIDE AGENCY INSPECTION AND AUDIT PROGRAM

6.5.5.1 An independent fire protection and loss prevention inspection and audit shall be performed at least once per 12 months using either qualified offsite licensee personnel or an outside fire protection firm.

6.5.5.2 An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at intervals no greater than 36 months.

6.5.5.3 Copies of the audit reports and responses to them shall be forwarded to the Vice President-Harris Nuclear Project and the Manager—Corporate Quality Assurance.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

REPORTABLE EVENT ACTION (Continued)

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the PNSC, and the results of this review shall be submitted to the Manager-Corporate Nuclear Safety Section and the Vice President-Harris Nuclear Project.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Vice President-Harris Nuclear Project and the Manager-Corporate Nuclear Safety Section shall be notified within 24 hours;

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PNSC. This report shall describe: (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems, or structures, and (3) corrective action taken to prevent recurrence;

c. The Safety Limit Violation Report shall be submitted, within 14 days of the violation, to the Commission, the Manager-Corporate Nuclear Safety Section, and the Vice President-Harris Nuclear Project; and

d. Operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Generic Letter No. 82-33;

c. Security Plan implementation;

d. Emergency Plan implementation;

e. PROCESS CONTROL PROGRAM implementation;

f. OFFSITE DOSE CALCULATION MANUAL implementation;

ANSWERS -- SHEARON HARRIS 1&2 -86/04/25-PAYNE, C.

ANSWER 5.01 (1.00)

c

REFERENCE
WNTO, pp. I-5.63 - 77
SHNPP: RT-LP-3.10, L.O. 1.1.1
RT-LP-3.15, L.O. 1.1.7

3.4/3.4 3.3/3.4 3.2/3.3 3.6/3.5
192006K107 192006K108 192006K114 192008K118 ...(KA'S)

ANSWER 5.02 (1.00)

d

REFERENCE
CNTO, "Thermal/Hydraulic Principles and Applications, II", pp 10-45/48
SHNPP: FF-LP-3.2, L.O. 1.1.5

2.9/3.1 2.3/2.4 2.4/2.5 2.3/2.4 3.1/3.3
006050K501 191004K105 191004K109 193006K102 193006K115
...(KA'S)

ANSWER 5.03 (1.00)

b or d

REFERENCE
NUS, Vol 4, pp 2.2-4
Surry 1-PT-35
SHNPP: HT-LP-3.2, L.O. 1.1.5
GP-LP-3.5
TS 3.3.1
OST 1004

2.6/3.1 3.1/3.4
015000K504 193007K108 ...(KA'S)

ANSWERS -- SHEARON HARRIS 1&2 -88/04/25-PAYNE, C.

~~ANSWER 5.04 (1.00)~~

a.

Deleted

REFERENCE
SHNPP: RT-LP-3.14, L.O. 1.1.3, 1.1.11
HBR RXTH-HO-1 Session [CAF]

3.2/3.5
192005K114 ...(KA'S)

ANSWER 5.05 (1.50)

a. DECREASE
b. DECREASE
c. INCREASE (0.5 each)

REFERENCE
General Physics, HT&FF, p. 320
SHNPP: FF-LP-3.2, L.O. 1.1.7, 1.1.8

3.2/3.3 2.6/2.8
191004K106 191004K115 ...(KA'S)

ANSWER 5.06 (2.00)

a. INCREASE
b. INCREASE
c. DECREASE
d. DECREASE (0.5 each)

REFERENCE
General Physics, HT & FF - Fluid Flow Applications for Systems and Components
SHNPP: AOP-LP-3.11, L.O. 1.1.1.A

3.1/3.4
002000K501 ...(KA'S)

ANSWER 5.07 (2.50)

a. 1. REMAIN THE SAME
2. DECREASE
3. INCREASE
4. DECREASE (0.5 each)

b. Superheated. (0.5)

REFERENCE
Steam Tables
SHNPP: FF-LP-3.6. L.O. 1.1.21
THERMO-LP-3.1. L.O. 1.1.14

2.8/2.8
193004K115 ...(KA'S)

ANSWER 5.08 (2.50)

a. SAME

b. HIGHER THAN

c. HIGHER THAN

d. SAME

e. LOWER THAN (0.5 each)

REFERENCE
SHNPP: RT-LP-3.13, L.O. 1.1.11, 1.1.12
RT-LP-3.15, L.O. 1.1.3
SONGS Lesson B-7, p 10-13; Curve book Fig. A.13.

3.4/3.5
192008K101 ...(KA'S)

ANSWERS -- SHEARON HARRIS 1&2 -88/04/25-PAYNE, C.

ANSWER 5.09 1.00 ~~(2.00)~~

~~a. NONCONSERVATIVE~~
~~b. NONCONSERVATIVE~~ } Deleted
c. NONCONSERVATIVE
d. CONSERVATIVE (0.5 each)

REFERENCE
Westinghouse Nuclear Training Operations, pp. I-4.19 - 21
SHNPP: RT-LP-3.7, L.O. 1.1.6
AOP-LP-3.7, L.O. 1.1.1.A

4.0/4.3 3.4/4.1 2.9/3.1
000036A202 000036K103 192008K106 ...(KA'S)

ANSWER 5.10 (1.50)

a. LOWER THAN

b. HIGHER THAN

c. LOWER THAN (0.5 each)

REFERENCE
VCS Reactor Theory I-5
SHNPP: RT-LP-3.6, L.O. 1.2
RT-LP-3.15, L.O. 1.1.7

3.3/3.4
192008K117 ...(KA'S)

ANSWER 5.11 (1.50)

a. FALSE
b. FALSE
c. FALSE (0.5 each)

REFERENCE
Westinghouse Nuclear Training Operations, p. I-5.36 - 43
SHNPP: RT-LP-3.12, L.O. 1.1.6, 1.1.8, 1.1.9

ANSWERS -- SHEARON HARRIS 1&2 -88/04/25-PAYNE, C.

2.8/3.1 2.6/2.9 2.5/2.8
192005K105 192005K106 192005K107 ...(KA'S)

ANSWER 5.12 (2.00)

1. a
2. b
3. a
4. a (0.5 each)

REFERENCE
Westinghouse Nuclear Training Operations, pp. I-5.6 - 16
SHNPP: RT-LP-3.9, L.O. 1.1.7, 1.1.8, 1.1.9, 1.1.10

3.4/3.7 3.3/3.6 3.4/3.7 2.9/3.4 2.9/3.1 3.1/3.1
001000K515 001000K526 001000K549 001010K529 191004K106
192004K103 ...(KA'S)

ANSWER 5.13 (2.00)

1. Rod to group alignment (+/-12 steps)
2. Groups sequenced and overlapped.
3. Rod Insertion Limits maintained.
4. Axial Flux Difference limits maintained. (0.5 each)

REFERENCE
SHNPP: Technical Specifications, B 3/4 p 2-2 to 2-4.
HT-LP-3.2, L.O. 1.1.4
RT-LP-3.14, L.O. 1.1.11

3.1/3.5 2.9/3.3
193009K105 193009K107 ...(KA'S)

ANSWERS -- SHEARON HARRIS 1&2 -88/04/25-PAYNE, C.

ANSWER 5.14 (1.00)

1) Fuel conversion (Pu-240 buildup) -- More Negative
2) Fuel temp. change with power (FP gases, gap thermal conductivity coefficient) -- More Negative
3) Fuel Densification--More Negative
4) Clad Creep--Less Negative (0.25 each)

REFERENCE
TPT Requal Lesson Plan, Cycle I, 1985 "Core Life Changes", pp 16
CNTO, "Reactor Core Control", pp 2-44/45
Surry, ND-86.2-LP-1 & LP-10
SHNPP: RT-LP-3.8, L.O. 1.1.20

3.4/3.7
001000K54- ...(KA'S)

ANSWER 5.15 (1.00)

1. Core height (0.5)

2. Fraction of rated power (0.5)

REFERENCE
SUN/SHNPP: TS 3/4.2.2

2.3/3.6
001000K546 ...(KA'S)

ANSWER 5.16 (1.50)

a. Density changes of the moderator with core height. (0.5)

b. Flux will shift significantly towards the top of the core. (0.5)
This is due to uneven fuel burnup (higher density fuel at the top). and moderator temperature defect reduction (as moderator temperature becomes more uniform at low powers). (0.5)

REFERENCE
WESTINGHOUSE REACTOR CORE CONTROL, pp. 3-51 to 3-53
SHNPP:

ANSWERS -- SHEARON HARRIS 1&2 -88/04/25-PAYNE, C.

3.7/3.9 2.9/3.1
001000K529 001000K530 ...(KA'S)

ANSWER 5.17 (1.00)

Tave :	31.8 X 0.25 X -15 =	-119 pcm (+/- 1)	(0.5)
Power:	25 X -12 =	-300 pcm	
Void :		- 25 pcm	
Xenon:		- 50 pcm	
Total:		-494 pcm	
Boron:	-494 / -9 =	54.9 ppm (54-56)	(0.25)
		Dilution	(0.25)

REFERENCE
Westinghouse Nuclear Training Operations, pp. I-5.27 - 5.36
SHNPP: RT-LP-3.11, L.O. 1.1.11

3.5/3.8
001000K52L ...(KA'S)

ANSWER 5.18 (2.50)

a. m = 4000 gpm x 60 min/hr x 1 cu. ft/7.48 gal x 1 lb./.0166 cu.ft
= 1.93 x 10E6 lbs/hr (+/- 10,000 lbs/hr) (0.5)

Q = mc(delta-T)
= 1.93 x 10E6 lbs/hr x 1 BTU/lb-deg F x 8 deg F
= 1.544 x 10E7 BTU/hr / 3.413 x 10E6 BTU/hr/MW
= 4.52 MW (0.5)

% = 4.52 MW/2775 MW
= 0.16 % (0.25)

(Since 0.16 % < 0.3%) Cannot maintain heat load. (0.25)

NOTE: ECF will be applied and comparable solutions accepted.

b. Increase mc by starting a second RHR pump
Increase delta-T by increasing CCW flow
Increase Emergency Service Water (ESW) flow *(any 2* @ ~~X~~0.5 each)

REFERENCE
BVPS Thermodynamics Manual Chapter 3
BVPS System Description Chapter 10
SHNPP: HT-LP-3.1, L.O. 1.1.2.3

2.2/2.3 2.5/2.7 2.4/2.4
191006K103 191006K104 191006K108 ...(KA'S)

ANSWER 5.19 (1.50)

```
      C1(1 - k1) = C2(1 - k2)                                  (0.25)
  270 (1 - 0.96) = 450 (1 - k2)
              k2 = .976                                        (0.25)

      C1(1 - k1) = C3(1 - k3)
  270 (1 - 0.96) = 1350 (1 - k3)
              k3 = .992                                        (0.25)
```

Third bank worth = (.992 - .976) = 0.016
PCN = .992 + .016 = 1.008 (0.25)
Since Keff > 1, the reactor will be critical on the third bank. (0.5)

REFERENCE
HBR Reactor Theory RXTH-HO-1 Session 42 pp. 2, 3
SHNPP: R1-LP-3.7, L.O. 1.1.8, 1.1.11

3.1/3.1 2.6/2.6 3.9/4.0 3.8/3.8
192002K107 192002K108 192008K103 192008K104 ...(KA'S)

ANSWER 6.01 (1.00)

d

REFERENCE
SHNPP: SD-126.01, p. 11, 29
ESFAS-LP-3.0, p.14-15, L.O. 1.1.5

3.7/3.7
039000K405 ...(KA'S)

~~ANSWER 6.02 (1.00)~~

c

REFERENCE
SHNPP: SD-1[illegible], [illegible]
RPS-LP-3.0, L.O. 1.1.6

3.7/4.[illegible]
0[illegible]000K603 ...(KA'S)

Deleted

ANSWER 6.03 (1.00)

b

REFERENCE
SHNPP: RODCS-LP-3.0, p. 20-21, 37; L.O. 1.1.7; 1.1.8

3.3/3.5
015000K302 ...(KA'S)

ANSWER 6.04 (1.00)

d

REFERENCE
SHNPP: SGWLC-LP-3.0, p. 5-6, L.O. 1.1.4;
SD-126.02, p. 9

3.4/3.4
059000K104 ...(KA'S)

ANSWER 6.05 (2.00)

a. 2, 3, 7
b. 1, 2
c. 4, 5
d. 1, 4, 6

(0.2 each)

REFERENCE
SHNPP: RPS-LP-3.0, p. 12-25, L.O. 1.1.11, 1.1.12, 1.1.13, 1.1.14

3.9/4.3
012000K402 ...(KA'S)

ANSWER 6.06 (1.50)

a. Automatic pressure control valves in the pump discharge lines (0.5)

b. 1. Reduce the potential for water hammer
 2. Reduce steam generator (preheater) tube vibration (0.5 each)

REFERENCE
SHNPP: SD-137, p. 5
 SD-134, p. 21
 AFS-LP-3.0, L.O. 1.1.1, 1.1.4

4.1/4.2 3.1/3.4
061000K101 061000K404 ...(KA'S)

ANSWER 6.07 1.00 ~~(1.50)~~

a. 1. The ESW booster pumps start on an SI signal.
 2. The containment air cooler orifice bypass valves close. (0.5 each)

Deleted
~~b. To prevent sliding water from the auxiliary reservoir (preferred source) to the main reservoir (backup source).~~ ~~(0.5)~~

REFERENCE
SHNPP: ESWS-LP-3.0, p. 13, 17-19, L.O. 1.1.6, 1.1.3, 1.1.5

3.6/3.7 2.9/3.2
076000K119 076000K402 ...(KA'S)

ANSWER 6.06 (2.50)

1. Low pressure reactor trip [0.4] - 1960 psig [0.1]
2. Low pressure SI [0.4] - 1850 psig [0.1]
3. P-11 permissive bistable [0.4] - 2000 psig [0.1]
4. High pressure reactor trip [0.4] - 2385 psig [0.1]
5. Overtemperature delta-T [0.4] - variable [0.1] or 109% (+/-penalties)

REFERENCE
SHNPP: SD-100.03, p. 12
PZRPC-LP-3.0, p. 12-13, L.O. 1 1 4, 1.1.5

3.9/4.1 3.9/4.1 3.8/4.1
010000K101 010000K102 010000K403 ...(KA'S)

ANSWER 6.09 (1.50)

a. 1. Containment H2 Purge Filter/System High DP (ALB-28, 7-2)
2. Containment H2 Purge System High Relative Humidity (ALB-28, 7-3)
3. Containment H2 Purge System Charcoal Filter Trouble (ALB-28, 7-4)
4. Containment H2 Purge System Exhaust Fan E4 Low Flow - O/L (ALB-28, 7-75)

(Any 2 @ 0.5 each)

b. True (0.5)

REFERENCE
SHNPP: SD-125, p. 6, 10; APP-ALB-028
APP-ALB-028
PAHP-LP-3.0, L.O. 1.1.4, 1.1.6

3.1/3.3 3.3/3.5 3.3/3.4
028000A403 028000G004 028000G008 ...(KA'S)

ANSWER 6.10 (1.50)

a. 1. the impulse line temperatures are monitored and auto-compensated
   2. the DP cells are located outside containment
(0.5 each)

b. The Dynamic Head Range (0.5)

REFERENCE
SHNPP: ICCM-LP-3.0, p. 8-9, 14-15, L.O. 1.1.2, 1.1.3

4.4/4.6 3.5/3.7 3.5/3.8 3.1/3.6
002000A306 002000K107 002000K402 002000K603 ...(KA'S)

ANSWER 6.11 (1.00)

1. Turbine building stack
2. Plant vent stack
3. WPB stack 5
4. WPB stack 5A (0.25 each)

REFERENCE
SHNPP: RMS-LP-3.0, p. 15, L.O. 1.1.4

3.1/3.1 3.6/3.9
071000K106 ...(KA'S)

ANSWER 6.12 (2.00)

1. All pressurizer heaters will deenergize
2. All orifice isolation valves will close or LCV-459
3. Letdown isolation valve (1CS-2) will close
4. Charging flow will increase (FCV-122 will open) (0.5 each)

REFERENCE
SHNPP: PZRLC-LP-3.0, p. 3, 10, L.O. 1.1.3

3.4/3.6
011000A211 ...(KA'S)

ANSWER 6.13 (2.00)

a. Normal - 480 V AC Emerg Bus 1A3-SA (1A21-SA)
Backup - 125 V DC Emerg Bus DP1A-SA
Bypass - 480 V AC Emerg Bus 1A3-SA (MCC 1A21-SA; PP 1A211-SA)
(0.5 each)

b. False (0.5)

REFERENCE
SHNPP: SD-156, p. 11, 27
120VUPS-LP-3.0, p. 7-8, L.O. 1.1.4, 1.1.7

3.1/3.5 2.7/3.2
062000K410 063000K102 ...(KA'S)

ANSWER 6.14 (2.00)

a. 1. The 60 - second time delay relay must time out
2. A reactor trip signal (P-4) must be in effect
3. The operator must manually reset / block the SI signal(s)
(0.5 each)

b. The reactor trip breakers must be shut (0.5)

REFERENCE
SHNPP: ESFAS-LP-3.0, p. 11-12, 26; L.O. 1.1.6, 1.1.7

3.9/4.3
013000K401 ...(KA'S)

ANSWER 6.15 (1.00)

1. An SI signal must be present (not reset)
2. The RWST must reach the low-low level setpoint (0.5 each)

REFERENCE
SHNPP: CSS-LP-3.0, p. 12, 25; L.O. 1.1.6, 1.1.12

4.2/4.3
026000K401 ...(KA'S)

ANSWER 6.16 (1.00)

1. The valve on the pit side must be fully open
2. Both lifting arms on the upenders must be down (0.5 each)

REFERENCE
SHNPP: FHS-LP-3.0, p. 18, L.O. 1.1.6

2.5/3.3 2.9/3.5
000036K201 034000K402 ...(KA'S)

ANSWER 6.17 (1.50)

a. 1. The respective loop RHRS to CSIP suction valve must be closed.

2. The respective RWST to RHRS isolation valve must be closed. (0.5 each)

b. False

REFERENCE
SHNPP: SD-111, p. 10
RHRW-LP-3.0, p. 12-13, L.O. 1.1.6

3.2/3.5
006000K408 ...(KA'S)

ANSWER 6.18 (1.50)

Purge - (solenoid) valves open to substitute room air for the normal sample flow to the detector (for approximately 1 - 2 minutes)

Filter - the filter paper will advance in fast speed (for approx. one minute)

C/S - the check source will be exposed (for approx. 30-60 seconds)

(0.5 each)

REFERENCE
SHNPP: RMS-LP-3.0, p. 23-25, 40, L.O. 1.1.7

3.7/3.7

ANSWERS -- SHEARON HARRIS 1&2 -88/04/25-PAYNE, C.

073000A402 ...(KA'S)

ANSWER 6.19 (1.50)

a. 1. Idle CCW pumps start
   2. GFFD isolates
   3. Sample system heat exchangers isolate
   4. Excess letdown/reactor coolant drain tank heat exchangers isolate (0.25 each)

b. Isolation valves to and from the RCPs close (0.5)

REFERENCE
SHNPP: CCWS-LP-3.0, p. 20-21, L.O. 1.1.4
SD-145, p. 16-17

3.3/3.4 3.4/3.6 3.1/3.3 2.9/3.2
008000G007 008000G015 008000K401 008010A303 ...(KA'S)

ANSWER 6.20 (2.00)

a. 1. The MCSS must be in LOCAL (0.25)
   2. Simultaneously (0.25) depress the EMERGENCY STOP (0.25) and the EMERGENCY STOP THINK pushbuttons (0.25)

b. Simultaneously (0.33) depress the EMERGENCY STOP (0.33) and the ACP TRANSFER CONTROL pushbuttons (0.33)

REFERENCE
SHNPP: SACP-LP-3.0, p. 16, L.O. 1.1.2

3.9/4.2
064000K402 ...(KA'S)

ANSWERS -- SHEARON HARRIS 1&2 -88/04/25-PAYNE, C.

ANSWER 7.01 (2.50)

a. 4 (1.0)

b. 2 (1.0)

c. Safe reactor conditions are established. (0.5)

REFERENCE
SHNPP: AOP-LP-3.2, L.O. 1.1.1, 1.1.3, 1.1.5
AOP-002

3.8/3.9 4.1/4.4
000024G011 000024K301 ...(KA'S)

ANSWER 7.02 (1.00)

c

REFERENCE
SHNPP: AOP-018, p. 18

2.7/3.1 4.0/3.9
003000A203 003000G014 ...(KA'S)

ANSWER 7.03 (1.00)

d

REFERENCE
SHNPP: AP-503
RP-LP-3.3,L.O. 1.1.11

2.8/3.4 3.1/3.4
194001K103 194001K105 ...(KA'S)

ANSWERS -- SHEARON HARRIS 1&2 -88/04/25-PAYNE, C.

ANSWER 7.04 (1.00)

c

REFERENCE
SHNPP: RP-LP-3.3, L.O. 1.1.7, 1.1.8
10 CFR 20

2.8/3.4
194001K103 ...(KA'S)

ANSWER 7.05 (2.50)

a. Closed

b. Not affected

c. Closed

d. Open

e. Closed (0.5 each)

REFERENCE
SHNPP: AOP-LP-3.10, L.O. 1.1.1
ISA-LP-3.0, L.O. 1.1.2

2.9/3.3 3.1/3.1 3.4/3.6
000065A208 078000G015 078000K302 ...(KA'S)

ANSWER 7.06 (1.00)

a. True

b. True (0.5 each)

REFERENCE
SHNPP: EOP User's Guide, p. 15
EOP-LP-3.0, L.O. 1.1.46m
COMP-LP-3.0, L.O. 1.1.1

4.3/4.1

ANSWERS -- SHEARON HARRIS 1&2 -88/04/25-PAYNE, C.

194001A113 ...(KA'S)

ANSWER 7.07 (.50)

False

REFERENCE
SHNPP: EOP-LP-3.3, L.O. 1.1.1
EPP-009

4.2/4.7 3.7/3.9 4.4/4.6
000011A201 000011K310 000011K312 ...(KA'S)

ANSWER 7.08 (~~2.50~~) 2.00

1. RCS subcooling (0.4), > 10 deg F (C) (0.1) [>20 deg F (M)]

2. Total FF (0.4), > 222.5 kpph (0.1) ~~or~~ — OR —
Level in one S/G (0.4), > 10% (0.1)

3. RCS pressure (0.4), stable or increasing (0.1)

4. Pressurizer level (0.4), > 10% (0.1)

REFERENCE
SHNPP: EOP Path-1, p. 57
SIS-LP-3.0, L.O. 1.1.15
EOP-LP-3.0, L.O. 1.1.46a
EOP-LP-3.1, L.O. 1.1.1

4.1/4.3 4.1/4.6
000009G012 000009K324 ...(KA'S)

ANSWERS -- SHEARON HARRIS 1&2 -88/04/25-PAYNE, C.

ANSWER 7.09 (2.00)

1. RCS subcooling (0.4) - >10 degrees F (C), 20 degrees F (M) (0.1)
2. Steam pressure (0.4) - stable or decreasing (0.1)
3. RCS hot leg temperature (0.4) - stable or decreasing (0.1)
4. Core exit thermocouples (0.4) - stable or decreasing (0.1)
5. RCS cold leg temperature (0.4) - trending to or at T-sat for steam pressure (0.1)

(any 4 @ 0.5 each)

REFERENCE
SHNPP: EOP-EPP-009, p. 8
EOP-EPP-031, p. 26
EOP-LP-3.3, L.O. 1.1.2
EOP-LP-3.7, L.O. 1.1.2

4.2/4.5 4.4/4.6 4.3/4.5
000009A237 000017K101 002000A402 ...(KA'S)

ANSWER 7.10 (1.50)

1. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

2. A radiation monitoring device which continuously integrates the dose rate in the area and alarms when a preset dose is reached.

3. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device.

(0.5 each)

REFERENCE
SHNPP: TS 6.12
HPP-020, p. 8-9
TS-LP-3.0, L.O. 1.1.8

2.8/3.4
194001K103 ...(KA'S)

ANSWER 7.11 (2.50)

1. Particulate or gas alarms on the RM-11.
2. WPB Stack 5 rad. monitor alarm or increasing levels.
3. Verbal notification (of gas system leak/rupture).
4. Increase in CAM readings or PAS.
5. "WPB EFFLUENT RAD MONITOR TROUBLE" alarm (ALB-10-3-4)

(0.5 each)

REFERENCE
SHNPP: AOP-LP-3.5. L.O. 1.1.2.A
AOP-009

2.9/3.8 3.8/3.8 3.7/3.9
000060G008 000060G010 000060G011 ...(KA'S)

ANSWER 7.12 (1.00)

a. To prevent pumping the RWST dry.

b. To prevent injecting radioactive containment sump water into the RWST.

(0.5 each)

REFERENCE
SHNPP: ERG, ES-1.3, p. 3, 10
EOP-LP-3.3, L.O. 1.1.2

4.4/4.6
000011K312 ...(KA'S)

ANSWERS -- SHEARON HARRIS 1&2 --88/04/25-PAYNE, C.

ANSWER 7.13 (1.00)

a. 24 hours after LOCA initiation.

b. To prevent boron precipitation at the top of the core where coolant may be boiling.

(0.5 each)

REFERENCE
SHNPP: EOP-Guide-1, p. 87;
SD-110, p. 17
SIS-LP-3.0, L.O. 1.1.15
EOP-LP-3.3, L.O. 1.1.2

3.8/4.2
000011K313 ...(KA'S)

ANSWER 7.14 (1.00)

If the sequence is important then subtasks are designated by letters/numbers; if sequence is not important then they are designated by bullets.

REFERENCE
SHNPP: EOP User's Guide, p. 6
EOP-LP-3.0, L.O. 1.1.26

4.1/3.9 3.3. L.O. 1.1.2
194001A102 ...(KA'S)

ANSWERS -- SHEARON HARRIS 1&2 -88/04/25-PAYNE, C.

ANSWER 7.15 (1.50)

1. RHR flow is throttled (to 1500 gpm) when draining steam generator tubes.

2. The running RHR pump is stopped before starting the standby pump.

3. The RHR pumps are not operated when RCS level is low (<82" below head flange).

(0.5 each)

REFERENCE
SHNPP: OP-111, p. 6-8
LER 323-87-23 (SER 15-87)
RHRS-LP-3.0

2.2/2.5
005000K409 ...(KA'S)

ANSWER 7.16 (1.00)

To prevent overpressurization of the suction line.

REFERENCE
SHNPP: OP-107, p. 9
CVCS-LP-3.0

3.1/3.4 L.O. 1.1.8
004000G010 ...(KA'S)

ANSWER 7.17 (1.00)

To prevent the backflow of potentially dirty water from the VCT.

REFERENCE
SHNPP: OP-100, p. 7
RCS-LP-3.0, L.O. 1.1.20

3.3/3.6
003000G010 ...(KA'S)

ANSWERS -- SHEARON HARRIS 1&2 -88/04/25-PAYNE, C.

ANSWER 7.18 (2.50)

a. TP-A : Place both key switches to TRANSFER
TP-B : Place both key switches to TRANSFER
ACP : Place all four XFER ACTUATE switches to XFER and release (RED light on, GREEN light off)
TP-A&B: Use switch handle to actuate relays that fail to transfer (de-energize, than re-energize control power as necessary) (0.5 each)

b. Minimize spurious equipment actuations (due to fire) (0.5)

REFERENCE
SHNPP: AOP-LP-3.0, L.O. 1.1.5
AOP-004

4.4/4.4 3.9/4.1 4.1/4.2 3.9/4.4 4.1/4.5 4.2/4.5
000068A112 000068A121 000068G010 000068K309 000068K312
000068K318 ...(KA'S)

ANSWER 7.19 (1.00)

Verifies the Low Temperature Overpressure System is not armed.

REFERENCE
SHNPP: GP-LP-3.2, L.O. 1.1.5
GP-002

3.3/3.6 3.5/3.7 3.8/4.1
010000G010 010000G013 010000K403 ...(KA'S)

ANSWERS -- SHEARON HARRIS 1&2 -88/04/25-PAYNE, C.

ANSWER 7.20 (1.00)

a. No operations are permitted that would cause dilution of the RCS boron concentration.

b. Core outlet temperature is maintained at least 10 deg F below saturation temperature.

(0.5 each)

REFERENCE
SHNPP: GP-LP-3.7, L.O. 1.1.3
GP-007

3.1/3.2 3.7/3.9 3.7/3.8 3.5/3.6
000017G007 003000A202 003000G001 005000G001 ...(KA'S)

ANSWER 7.21 (1.00)

a. Radiation release and equipment damage minimized.

b. Plant conditions stable with plant equipment operating in long term alignments. (0.5 each)

REFERENCE
SHNPP: EOP-LP-3.0, L.O. 1.1.2

3.1/4.4
194001A116 ...(KA'S)

ANSWERS -- SHEARON HARRIS 1&2 -88/04/25-PAYNE, C.

ANSWER 8.01 (1.00)

c

REFERENCE
McG/CAT/FNP/SQNP/SHNPP: TS 3/4.1.1 and p. B 3/4 1-1
SHNPP: TS-LP-3.0, L.O. 1.1.7

3.2/3.8 3.4/3.9 3.9/4.4 3.5/4.1
001000A204 001000G011 001000K508 001010A404 ...(KA'S)

ANSWER 8.02 (1.00)

d

REFERENCE
SQN/SHNPP: TS 3.0.3, 3.3.2.1, 3.5.2, 3.5.3
TS-LP-3.0, L.O. 1.1.7

3.5/4.2 3.6/4.2
006000G005 006000G011 ...(KA'S)

ANSWER 8.03 (1.00)

d or b

REFERENCE
SHNPP: TS 3/4.5.3
TS-LP-3.0, L.O. 1.1.7.

3.5/4.2 3.6/4.2
006000G005 006000G011 ...(KA'S)

ANSWER 8.04 (1.00)

a

REFERENCE
SHNPP: ADOP-LP-3.0, L.O. 1.1.7
AP-024

3.7/4.1 3.6/3.7
194000K102 194000K107 ...(KA'S)

ANSWER 8.05 (1.00)

c (Requires IMMEDIATE action)

REFERENCE
SHNPP: TS 3.1.3.5
TS 3.3.1, Table 3.3.1
TS 3.5.1
TS 3.5.4
TS-LP-3.0, L.O. 1.1.7

3.5/4.2 3.6/4.2
006000G005 006000G011 ...(KA'S)

ANSWER 8.06 (.50)

FALSE

REFERENCE
SHNPP: OMM-004, p. 4
PP-LP-3.0, L.O. 1.1.5

2.5/3.4
194001A103 ...(KA'S)

ANSWER 8.07 (2.50)

a. False

b. False

c. True

d. True

e. False

(0.5 each)

REFERENCE
SHNPP: PGO-040
CVCS-LP-3.0, L.O. 1.1.9
TS 3.0.3, 3/4.5.2

2.9/3.6 2.6/3.8 3.3/3.8

ANSWERS -- SHEARON HARRIS 1&2 -88/04/25-PAYNE, C.

0040006002 0040006003 0040006005 ...(KA'S)

ANSWER 8.08 (1.75)

a. SF - 1
   SRO - 1
   RO - 2
   AO - 4
   STA - 1 (0.25 each)

b. SRO
   STA (0.25 each)

REFERENCE
SHNPP: OMM-001, p. 23
PP-LP-3.0, L.O. 1.1.3

2.5/3.4
194001A103 ...(KA'S)

ANSWER 8.09 (1.50)

1. Review it (for completeness and accuracy).

2. Sign and date the procedure.

3. Route the completed test to the Operating Supervisor/designee (and ISI, as required, for review and Document Control for retention).

4. log Completion ~~in SF log~~ on Control Room Surveillance Test Schedule. (0.5 each) any 3

REFERENCE
SHNPP: OMM-001, p. 64
PP-LP-3.0, L.O. 1.1.4

2.5/3.4
194001A103 ...(KA'S)

ANSWER 8.10 (1.00)

1. To track inoperable equipment not required in the existing plant mode/condition.

2. To track inoperable redundant equipment when LCO requirements are met by remaining operable equipment.

(0.5 each)

REFERENCE
SHNPP: OMM-003, p. 12
PP-LP-3.1, L.O. 1.1.1, 1.1.2

3.4/3.4
194001A106 ...(KA'S)

ANSWER 8.11 (1.00)

It lists all surveillance and periodic tasks which are pastdue, overdue or tests with exceptions.

REFERENCE
SHNPP: PLP-103, p. 13

2.6/3.1
194001A108 ...(KA'S)

ANSWER 8.12 (2.00)

1. Demonstrate operability of offsite ac sources by 12:00 noon, same day.

2. Verify operability of all redundant components by 1:00 p.m., same day.

3. *Unit in Hot Standby by 7:00 p.m., same day.*

4. ~~3~~. Demonstrate operability of offsite AC sources by 8:00 p.m., same day.

5. ~~4~~. Test the 1B EDG by 11:00 a.m., next day.

0.4 ~~(0.5~~ each)

REFERENCE
SHNPP: TS 3.8.1.1
TS-LP-3.0, L.O. 1.1.7
SACP-LP-3.0, L.O. 1.1.7

3.4/3.9

ANSWERS -- SHEARON HARRIS 1&2 -88/04/25-PAYNE, C.

0640006005 ...(KA'S)

ANSWER 8.13 (2.00)

1. Be in hot standby and reduce RCS pressure to within its limit within one hour.

2. Notify the NRC Operations Center by telephone as soon as possible and in all cases within one hour. (The VP-Harris Nuclear Project and the Manager Corporate Nuclear Safety Section shall be notified within 24 hours.)

3. Prepare a Safety Limit Violation Report.

4. Submit the Safety Limit Violation Report to the NRC within 14 days.

5. Do not resume unit operations until authorized by the NRC.

(any 4 @ 0.5 each)

REFERENCE
SHNPP: TS 2.1.2
TS 6.7.1
PVRPC-LP-3.0, L.O. 1.1.7

2.6/3.3 3.1/3.6
000027G002 000027G003 ...(KA'S)

ANSWER 8.14 (1.50)

Accident : A steam line break (uncontrolled RCS cooldown)

Conditions: 1. EOL
2. T-avg at no load

(0.5 each)

REFERENCE
SHNPP: TS 3/4.1.1.1 Bases
TS-LP-3.0, L.O. 1.1.7

2.9/3.8
001000G006 ...(KA'S)

ANSWER 8.15 (1.75)

1. The intent of the original procedure is not altered. (0.5)

2. The change is approved by two members of the plant management staff (0.5), at least one of whom holds an SRO license on the unit affected (0.25). (0.75)

3. The change is documented, reviewed and approved within 14 days by the Plant General Manager (or by the manager of the functional area affected). (0.5)

REFERENCE
SHNPP: TS 6.8.3
TS-LP-3.0, L.O. 1.1.8

2.5/3.4
194001A103 ...(KA'S)

ANSWER 8.16 (1.50)

Yes (TS have been violated) (0.5)

(Per TS 3.9.8.1) RHR loops can be shutdown only 1 hour per 2 hour period while in Mode 6. (Was S/D 1-1/2 hrs) (1.0)

REFERENCE
FNP: TS 3.9.8.1
SHNPP: TS 3.9.8.1

3.2/3.6 3.1/3.8 3.2/3.6
005000G005 005000G011 005000K307 ...(KA'S)

ANSWER 8.17 (2.00)

When the trip breakers are open a control rod bank withdrawal accident is prevented (1.0), thereby reducing the required heat removal capacity (1.0).

REFERENCE
SHNPP: TS 3/4.4.1 Bases
RCS-LP-3.0, L.O. 1.1.19

2.7/3.8

003000G006 ...(KA'S)

ANSWER 8.18 (2.00)

a. The system is inoperable.

b. The system is under test.

c. The alarm is not working properly (work request/deficiency tag written).

d. The alarm is a normal condition for 100% power. (0.5 each)

REFERENCE
SHNPP: OMM-001, p. 45

4.3/4.1
194001A113 ...(KA'S)

ANSWER 8.19 (1.00)

Only if the valve is to be cycled repeatedly during the shift (then IV is only required in its final position).

REFERENCE
SHNPP: PLP-702, p.6
PP-LP-3.0, L.O. 1.1.8

3.6/3.7
194001K101 ...(KA'S)

ANSWER 8.20 (3.00)

a. Were not exceeded (0.5). Eight days does not exceed 1.25 times the specified time interval (1.0).

b. Were exceeded (0.5). The last 3 consecutive intervals exceed 3.25 times the specified time interval (1.0).

REFERENCE
SHNPP: TS-LP-3.0, L.O. 1.1.1, 1.1.6
TS 4.0.1

3.5/4.2 3.6/4.2 3.1/3.4 2.1/2.5
006000G005 006000G011 006020A105 194001K117 ...(KA'S)

TEST CROSS REFERENCE

QUESTION	VALUE	REFERENCE
05.01	1.00	DCP0001450
05.02	1.00	DCP0001464
05.03	1.00	DCP0001473
05.04	1.00	DCP0001477
05.05	1.50	DCP0001449
05.06	2.00	DCP0001453
05.07	2.50	DCP0001460
05.08	2.50	DCP0001462
05.09	2.00	DCP0001466
05.10	1.50	DCP0001485
05.11	1.50	DCP0001448
05.12	2.00	DCP0001455
05.13	2.00	DCP0001458
05.14	1.00	DCP0001493
05.15	1.00	DCP0001600
05.16	1.50	DCP0001599
05.17	1.00	DCP0001457
05.18	2.50	DCP0001465
05.19	1.50	DCP0001479
	30.00	
06.01	1.00	DCP0001519
06.02	1.00	DCP0001527
06.03	1.00	DCP0001580
06.04	1.00	DCP0001583
06.05	2.00	DCP0001521
06.06	1.50	DCP0001528
06.07	1.00 ~~1.50~~	DCP0001517
06.08	2.50	DCP0001520
06.09	1.50	DCP0001522
06.10	1.50	DCP0001524
06.11	1.00	DCP0001525
06.12	2.00	DCP0001529
06.13	2.00	DCP0001530
06.14	2.00	DCP0001579
06.15	1.00	DCP0001582
06.16	1.00	DCP0001584
06.17	1.50	DCP0001598
06.18	1.50	DCP0001518
06.19	1.50	DCP0001526
06.20	2.00	DCP0001586
	30.00	
07.01	2.50	DCP0001573
07.02	1.00	DCP0001588
07.03	1.00	DCP0001589
07.04	1.00	DCP0001590
07.05	2.50	DCP0001574

TEST CROSS REFERENCE

QUESTION	VALUE	REFERENCE
07.06	1.00	DCP0001512
07.07	.50	DCP0001577
07.08	2.50	DCP0001509
07.09	2.00	DCP0001510
07.10	1.50	DCP0001513
07.11	2.50	DCP0001578
07.12	1.00	DCP0001507
07.13	1.00	DCP0001508
07.14	1.00	DCP0001511
07.15	1.50	DCP0001514
07.16	1.00	DCP0001515
07.17	1.00	DCP0001516
07.18	2.50	DCP0001571
07.19	1.00	DCP0001572
07.20	1.00	DCP0001576
07.21	1.00	DCP0001575
	30.00	
08.01	1.00	DCP0001592
08.02	1.00	DCP0001593
08.03	1.00	DCP0001594
08.04	1.00	DCP0001595
08.05	1.00	DCP0001596
08.06	.50	DCP0001496
08.07	2.50	DCP0001498
08.08	1.75	DCP0001505
08.09	1.50	DCP0001494
08.10	1.00	DCP0001495
08.11	1.00	DCP0001497
08.12	2.00	DCP0001499
08.13	2.00	DCP0001501
08.14	1.50	DCP0001502
08.15	1.75	DCP0001503
08.16	1.50	DCP0001591
08.17	2.00	DCP0001500
08.18	2.00	DCP0001504
08.19	1.00	DCP0001506
08.20	3.00	DCP0001597
	30.00	
	120.00	

DOCKET NO 400