ENCLOSURE 4

CP&L

Carolina Power & Light Company

H. B. ROBINSON SEG PLANT
POST OFFICE BOX 790
HARTSVILLE, SOUTH CAROLINA 29550

March 20, 1986

File: NTS-4208(F) Serial: NO-86R076

Mr. John Munroe
U. S. Nuclear Regulatory Commission, Region II
Suite 2900
101 Marietta Street, N.W.
Atlanta, Georgia 30323

SUBJECT: Reactor Operator and Senior Reactor Operator Written Examinations

Dear Mr. Munroe:

On March 14, 1986, Mr. Gerry Douglas of the USNRC administered written reactor operator and senior reactor operator examinations at H. B. Robinson. Enclosed, please find comments for each examination.

Instructors from the H. B. Robinson Training Unit reviewed the examinations for comments. They included:

R. S. Allen
W. T. Hensley
A. McGilvray
D. A. Neal
V. L. Smith

If you have any questions, please contact me or Mr. C. A. Bethea.

Very truly yours,

R. E. Morgan
General Manager
H. B. Robinson S. E. G. Plant

RSA:eaw

Enclosures (6)

NRC EXAMINATION COMMENTS

H. B. ROBINSON RO CLASS 85-1

EXAMINATION DATE - MARCH 14, 1986

1. Section 1

Question 1.02, Part b

Accept the answer stated but also recommend accepting as an answer: "As the temperature decreases, the pressure will decrease; therefore, the valves will have to be opened further to maintain the same flow rate."

Question 1.03, Part c

On the answer key, change U235 to U238. This was probably a typographical error.

Reference: RXTH-HO-1, Session 47, Page 2 of 3.

Question 1.08, Part a

Accept the answer as stated but also recommend accepting as an answer: "SDM is increased (0.5) as the RCS is borated and rods are withdrawn, thus providing increased trippable reactivity. (0.5)

Reference: RXTH-HO-1, Session 50, Page 6.

Question 1.08, Part b

Accept all the answers stated but also recommend accepting "Power Defect" as an answer.

Reference: RXTH-HO-1, Session 50, Page 6.

Question 1.11, Part c

This question involved a calculation. This could lead to the rounding of absolute pressure from 1.9632 psia to 2 psia since the table did not have 1.9632 psia as a selection. Recommend also accepting 11.08°F as an answer.

Question 1.12

The fourth part of the answer stated is correct, but axial power distribution is maintained by operating as per PDC-II (Power Distribution Control Procedure). Therefore, recommend also accepting as an answer: "Operate per PDC-II."

Reference: FMP-009, Page 4.

2. Section 2

Question 2.10, Part b

Historically, it has been taught that the generator breaker delay was to:

1. Prevent turbine overspeed.
2. Gives additional one minute of forced circulation flow on loss of A.C.

However, no reference has been found at H. B. Robinson. Recommend accepting answer above or delete the question due to the lack of plant specific reference.

Question 2.14, Part a

Answers as stated are correct, but recommend accepting the following clarification:

1. Turbine building supply or V6-16A, V6-16B (valve numbers).
2. A diesel room or diesel cross-connect.
3. Intake structure of SW pump discharge or SW pump discharge.
4. Intake structure of SW to circ. water pump seals or SW to circ. water pump seals.

Reference: SD-004; SD-006; Drawing #G-190199, Sheet 10 of 12.

Question 2.16

Recommend changing correct answer from "c" to "a".

Reference: SD-025, Page 15.

Question 2.17, Part d

Recommend changing answer to read:

d. Scintillation - isolates S/G blowdown, sample valves, S/G blowdown flow control valves, and blowdown tank drain (or V1-31, or blowdown to circ. water isolation valves).

Reference: AOP-005, Page 13; Drawing G-190234, Sheet 2 of 2, Location D-2.

3. Section 3

Question 3.11, Part a

The answers provided on the key are correct, but an additional location for S/G wide range level exists. Recommend also accepting: "on stanchion (I-Beam) next to secondary control panel or near feedwater bypass valves on stanchion (I-Beam)."

Reference: Block Diagram #5379-3440.

Question 3.12, Part d

The letdown isolation valves will automatically reopen when the pressurizer level increases above the setpoint. Recommend changing the answer on Part "d" from "yes" to "no".

Reference: SD-021, Page 17, Part 3.1.1.

Question 3.13

Recommend deleting this question since no correct choice is given. Correct choice should be "two minutes have elapsed since SI initiation."

Reference: SD-006, Page 12.

Question 3.17, Part b

Answers stated are correct, but also recommend accepting as clarification:

1. Manual or inhibit switch on RTGB.
2. Control rod cabinet or control rod motion (since rod motion provides the signal).

Reference: SD-052, Page 18.

4. Section 4

Question 4.01

General Procedure (GP-007, Page 13) has a caution that mentions only SIS. Therefore, recommend accepting as the required answer "initiation of SIS."

Reference: GP-007, Page 13.

Question 4.11, Part b

Recommend accepting "core" to be synonymous with "fuel".

Question 4.15

See general comment, enclosed.

Question 4.17, Part b

The candidates were returned to shift due to the delay in the written portion of the examination and were exposed to procedure changes. The dose procedure has recently been changed. Recommend changing answer from "375 mRem" to "1250 mRem" for Part 1 (one).

Reference: DP-003, Page 8; DP-014, Page 5.

Question 4.19, Part b

Recommend also accept "AOP-002" as an answer since AOP-002 is the Emergency Boration Procedure.

Reference: AOP-002.

END OF RO EXAMINATION COMMENTS

Enclosure 2 To Serial: NO-86R076

NRC EXAMINATION COMMENTS

H. B. ROBINSON SRO CLASS 85-2

EXAMINATION DATE - MARCH 14, 1986

1. Section 5

Question 5.12

The differences of dew point temperatures, wet bulb, or dry bulb are not taught at H. B. Robinson. Also, the different readings are not normally obtainable by an operator. In addition, the answer reference (thermodynamics, K. Wark) is not used at H. B. Robinson for training. Recommend deleting this question.

Question 5.15

Recommend also accepting as an answer: "to overcome reactivity inserted from power defect" (as this is a combination of MTC and Doppler).

Reference: RXTH-HO-1, Session 32, Pages 2 and 3.

Question 5.19

Recommend also accepting as an answer "the machine is capacitive or inductive relative to load."

Reference: GEN-LP-1, Pages 12 and 13; GEN-HO-1, Session 1, Pages 3 and 4.

2. Section 6

Question 6.01

1. The question is worth three (3) points but the answer key has an individual total of two (2) points. Recommend point values for parts in "a" and "b" be changed to 0.75 each, instead of 0.5 each.
2. Part a: The answers listed in Part "a" are correct. "High steam line flow" may also be listed, as this signal is possible under the right conditions. Recommend requiring the answers listed in the answer key, but not lose credit if "high steam line flow" is also listed as an answer.

Question 6.02

1. The question is worth three (3) points but the answer key has an individual total of two (2) points. Recommend point value for Part "a" be two (2) points and Part "b" be one (1) point. This would make all the individual answers worth 0.5 points each.
2. Part a: The answers listed for Part "a" are correct. Recommend accepting additional clarification:

   a. Turbine building supply or V6-16A, V6-16B (valve numbers).
   b. A diesel room or diesel cross-connect.
   c. Intake structure of SW pump discharge or SW pump discharge.
   d. Intake structure of SW to circ. water pump seals or SW to circ. water pump seals.

Reference: SD-004; SD-006; Drawing #G-190199.

Question 6.04

1. Part "a": Recommend accepting "sensing lines", "transmitter supply lines", or "capillary lines" as synonymous terminology for "impulse lines."
2. Part "d": Answering this question requires memorization of a significantly sized table in the system description for RVLIS. Memorization of such a table seems beyond that which should be required knowledge to obtain a license. Recommend deleting Part "d" of this question.

Reference: SD-051, Page 13

Question 6.06, Part a

Recommend Part "a" be deleted as the provided reference does not contain any description of the indicated effect on RCS pressure for stopping a RHR pump. Also, the word "initially" implies short-term response while the answer indicates long-term response.

Question 6.10, Part 2

Technical Specifications and SD-011 state the reason for the OPΔT trip as "limit core power density (KW/FT)." We could not locate any reference stating OPΔT being a backup for the high neutron flux trip. Recommend deleting Part 2 of this question.

Reference: Technical Specifications Pages 2.3-5; SD-011, Page 18.

Question 6.11

Recommend changing "c" to "a" as the correct answer.

Reference: SD-025, Page 15.

Question 6.13, Case 2

A note in the stem of the question stated "conditions may be used more than once or not at all." This could lead the candidates to choose "none" as an answer. For Case 2, the pressure would rise to the S/G PORV setpoint of ~1035 psig (relating to ~550°F). This was not a choice given. Therefore, recommend accepting "none" as an additional response for Case 2.

Reference: GP-003, Page 13, Precaution 4.12.

Question 6.14

Recommend deleting this question as no such interlock exists on the H. B. Robinson manipulator crane.

Reference: Enclosed memorandum, Serial: RNPD/86-1192.

Question 6.17, Part b

The term "refueling cavity lower section" could be misleading to the candidates. The lowest section of the refueling cavity is drained to the CV sump and is referred to by the procedure as the "lower" section drain. Recommend also accepting as an answer: "the refueling cavity lower section is drained to the CV sump. From there it is pumped to the waste hold-up tank.

Reference: SD-014, Page 12; GP-009, Page 29.

3. Section 7

Question 7.15, Part b

Recommend accepting "fuel" or "core" as being synonymous terminology.

Question 7.17, Part b

The candidates were returned to shift due to the delay in the written portion of the examination and were exposed to procedure changes. The dose procedure was recently changed. Recommend changing answer from 375 mRem to 1250 mRem for Part 1.

Reference: DP-003, Page 8; DP-014, Page 5.

Question 7.22

See enclosed general comment.

Question 7.24

General Procedure (GP-007, Page 13) has a caution that mentions only SIS. Therefore, recommend accepting the required answer as "initiation of SIS."

Reference: GP-007, Page 13.

4. Section 8

Question 8.19

The candidates may have answered "no action required" if their train of thought was for immediate actions. They may have used the LCO's in Technical Specifications if their train of thought was for long-term actions. Recommend also accepting as an answer: "return the RHR to service or be in cold shutdown within 72 hours."

Reference: Technical Specifications 3.3.1.3.

Question 8.21, Part a

Initially, the Site Emergency Coordinator is the Shift Foreman. The Shift Foreman is the interim Site Emergency Coordinator until relieved per PEP-001. Recommend accepting either "Shift Foreman" or "Site Emergency Coordinator" as an answer.

Reference: PEP-001, Page 5; PLP-007, Page 29, Paragraph 2.

Question 8.22, Part b

10CFR50 lists SRO as approval. AP-006 lists the Shift Foreman as approval. AP-006 also states, if time permits, Plant Management should be consulted and the NRC notified prior to taking deviation. Recommend accepting "Shift Foreman" or "SRO" as answers. Also recommend not losing credit if consulting Plant Management and notifying the NRC is also listed in the answer.

Reference: 10CFR50.54(y); AP-006, Page 7.

END OF SRO EXAMINATION COMMENTS

Enclosure 3 To Serial: NO-86R076

GENERAL COMMENT

Please find enclosed the purpose of OMM-22 (Emergency Operating Procedures User's Guide). This states that operator memorization of EOP steps are no longer required. The purpose of this General Comment is to highlight this important concept.

Reference: OMM-022, Page 3, Section 1.

REFERENCE FOR GENERAL COMMENT

1.0 PURPOSE

The purpose of this procedure is to provide instructions for the use of Emergency Operating Procedures (EOP). Memorization, by the Operator, of EOP steps is no longer required. The need for Operator memorization of the "Immediate Action" EOP steps has been eliminated by the development of symptom-based EOP's and the "Flow Path" format of the initial recovery actions. The Control Room copies of PATH-1 and PATH-2 and the CSFST's are board mounted and kept readily available to the Operators. This results in timely, consistent response, by the Operators, to events requiring EOP use without dependence on memorization. To successfully use the EOP's, the Operator should be generally familiar with each EOP within the EOP Network and the "Rules of Usage" contained in this procedure.

2.0 REFERENCES

2.1 NUREG - 0660, NRC Action Plan Developed as a Result of the TMI-2 Accident.

2.2 NUREG -0737, Clarification of the TMI Action Plan Requirements.

2.3 OMM-013, Emergency Operating Procedures Writer's Guide.

3.0 RESPONSIBILITIES

3.1 The Shift Foreman is responsible for insuring that the Emergency Operating Procedures are used correctly and understood by the personnel on shift.

CP&L

Carolina Power & Light Company

Company Correspondence

ROBINSON NUCLEAR PROJECT DEPARTMENT
POST OFFICE BOX 790
HARTSVILLE, SOUTH CAROLINA 29550

MAR 18 1986

Robinson File No: 7095 Serial: RNPD/86-1192

MEMORANDUM TO: C. A. Bethea

FROM: A. R. Wallace

SUBJECT: SD-008 - Fuel Handling Tools
Error in Identification of Interlock Bypass Switches

A question has been raised concerning the location and purpose of the Rotation Interlock Bypass switch on the Manipulator Crane as described in Section 3.1.6g of SD-008. The switch does not exist on the H. B. Robinson Manipulator Crane. The description of this switch was inserted erroneously into SD-008 during the last update. The error was caused by review of the procedure in its new format against technical data relating to a more recent design control circuit than exists at HBR. The HBR design contains only six (6) Interlock Bypass switches as listed in Section 3.1.6a through f.

SD-008 will be revised to correct this error. Until it is, please use this memorandum as documentation of this error.

If you have any questions, please contact me.

A. R. Wallace

MFP:ac

cc: D. A. Neal

Reference for Question 1.03.c

RXTH-HO-1

FACTORS IN K_{eff}

As U-238 is converted to Pu-239, and as radiative capture in Pu-239 produces Pu-240 and Pu-241, the isotopes contributing to the fast fission factor, ϵ, change. Many changes take place in the individual contributions to ϵ, but the net effect on the value of ϵ is slight, and the direction of change of ϵ is uncertain at any particular time in life. As a result, ϵ is normally assumed to be constant over core life.

When Pu-239 undergoes radiative capture, Pu-240 is produced. This isotope will not fission with thermal neutrons. However, it has many resonance absorption peaks. The buildup of Pu-240 over core life increases the amount of resonance absorption, decreasing p, the resonance escape probability. In addition, many fission products have appreciable resonance absorption cross sections. As fission products build up, they cause a continuous decrease in p over core life.

The thermal utilization factor, f, is given by the expression in RXTH-TP-48.2. Since f is the only factor in K_{eff} over which the operator has control, the actual value of f is adjusted by the operator to give a K_{eff} equal to 1 at the desired power level. It is possible, and necessary, however, to look at the individual terms in f to see how they change with core burnup.

As fuel burns up, fission product poisons accumulate. This decreases Σ_a^{fuel} and increases Σ_a^{other}, since fission product poisons are included in Σ_a^{other}. Both these effects lower the value of f.

As U-238 is converted to Pu-239, this raises the value of Σ_a^{fuel}. However, since fuel is consumed faster than Pu-239 is produced, this effect does not increase f; it merely slows the rate of decrease of f.

Diluting boron concentration causes Σ_a^{other} to decrease, thus f increases. Diluting compensates for the effects mentioned above which decrease f. So the overall effect is that f increases due to operator control - lowering the boron concentration.

Reference for Question 1.03.c

Changes in fuel composition also change the value of the reproduction factor, η. For a mixture of fissile isotopes, the value of η must be appropriately averaged. RXTH-TP-47.3 shows the value of η for a mixture of U-235 and Pu-239. Since Σ_f^{235} decreases faster than Σ_f^{239} increases, the numerator will decrease over core life.
Σ_a^{235} decreases and Σ_a^{239} increases while Σ_a^{238} is essentially constant. Although less Pu-239 is produced than U-235 is consumed, Pu-239 has a larger absorption cross section, so the decrease in the demoninator is small. The net effect is a small decrease in η over core life.

As the reactor is operated, the flux and power distributions flatten in the center of the core and increase near the edges. (RXTH-TP-47.4.) In addition, the soluble boron concentration is reduced to compensate for fuel burnup and other effects.

The decreased boron concentration increases the thermal diffusion length, which increases thermal neutron leakage, lowering the themal non-leakage probability, L_t. The increased flux near the edges of the core means increased fast and thermal neutron leakage, lowering L_f and L_t. Since L_f and L_t are both so close to one, the fractional change in L_f and L_t is small, and the effect on K_{eff} is normally neglected.

REACTIVITY COEFFICIENTS

The moderator temperature coefficient, α_m, is a function of 1) moderator-to-fuel ratio which depends on moderator density and 2) soluble boron concentration. The moderator-to-fuel ratio is fixed by the dimensions of the fuel pin lattice. The moderator density is a function of temperature, which is constant for a given power level.

Soluble boron concentration changes appreciably over core life. The plot of moderator temperature coefficient is similar in shape to that of boron concentration. As shown in RXTH-TP-47.5, as boron concentration decreases, α_m becomes more negative. The more negative moderator temperature coefficient late in core life causes the reactor to respond more rapidly to moderator temperature changes.

The increase in Pu-240, discussed previously, has a significant effect on resonance escape probability. The buildup of Pu-240 and fission products with large resonance cross

Reference for Question 1.08.b
1.08.a

RXTH-TP-50.5

. B. Robinson Unit No. 2

DETERMINATION OF SHUTDOWN MARGIN BORON CONCENTRATION

Calculation For Hot (530 degrees F), N-1 Rods Inserted, Xe Decay Not Included Short Term S/D Margin.

1. Time plant has been at relatively constant power = _____days + _____hours.

2. Nuclear power level = _____%P.

3. RCS Tavg at above power level = _____°F.

4. Boron concentration at above power level = _____ppm.

5. Inserted control bank worth (in core) at above power level = Total Inserted Control Rod Worth (HFP Integral Worth of D Bank Curve 1.6) _____pcm.

6. Equilibrium Xenon worth at above power level (Curve 2.2) = _____pcm.

7. Power defect (Use item 2 and Curve 1.3) _____pcm.

8. Equilibrium Samarium Worth (Curve 2.5) _____pcm.

9. S/D Margin = (N-1 Rod Worth) - (Item 5) - (Item 7) - _____pcm.
Short term S/D Margin (Xe decay not included)
= (N-1 Rod Worth) - (Item 5) - (Item 7)
= (-5894 pcm) - (_____pcm) - (_____pcm) = _____pcm.

10. PEAK SAMARIUM WORTH _____ pcm

 Reference for Question 1.12

1.0 PURPOSE

1.1 The purpose of this procedure is to provide the necessary steps to minimize axial power distribution during power operation in order to comply with Technical Specification 3.10.2.4 to 3.10.2.11.

1.2 The steps in this procedure are necessary to control the effects of xenon shifts and power redistributions which result from routine load changes. These steps, termed Power Distribution Control (PDC), result in maintaining relatively constant power shapes based on equilibrium conditions encountered throughout a given core cycle. These controls, developed by Exxon Nuclear Comp. and known as PDC-II, insure that core peaking factor limits will not be exceeded.

1.3 To provide the steps necessary to update the penalty point logging.

1.4 To provide a guideline for identifying, monitoring, and controlling divergent axial oscillations.

2.0 REFERENCES

2.1 Technical Specification 3.10.2.4 to 3.10.2.11 and Figures 3.10.4 and 3.10.5

3.0 RESPONSIBILITIES

3.1 Performance Engineering will be responsible for providing the target values to Operations.

3.2 Operations will be responsible for performing this procedure.

Reference for Questions 2.16 & 6.11

PAGE	TITLE	REV.	PROC. NO.
15 OF 26	Main Steam System	2	SD-025

2.0 INSTRUMENTATION AND CONTROL (Continued)

Controller No.	Arms	Setpoint
(PM-447A)	*3 Condenser Dumps	15% of full load
(PM-447B)	*2 Condenser Dumps	35% of full load
(PM-447D)	**3 PORV's	70% of full load

*NOTE: The condenser dumps can only be armed if the condenser is available, i.e., at least one circulating water pump running and sufficient vacuum in condenser.

**NOTE: The PORV's can be armed only if the turbine is not tripped.

2.10.3 Temperature Bistables and Controllers

Temperature bistables are provided to trip open the valves if they are armed. The trip open feature is accomplished by a three way solenoid that bypasses the positioner and applies nitrogen directly to the condenser dump valves or instrument air directly to the PORV's. The bistables that are allowed to trip open the valves are determined by whether or not the turbine is tripped. If the signal is not sufficient to trip the bistables, the valves may be modulated by the turbine trip or load rejection controller.

A. Turbine not tripped (Load Rejection)

The bistables and controller for a load rejection receive their signals from auctioneered Taverage and first stage pressure (PT-446).

1. High (Tavg - Tref) three valves trip open (Bistable TC-408F) 12.1 °F

Reference for Question 2.2.d

RADIATION MONITORING SYSTEM

2.0 HIGH RADIATION LEVEL, PROCESS MONITOR SYSTEM (continued)

2.2 AUTOMATIC ACTIONS (continued)

2.2.4 R-18

2.2.4.1 Waste release valve RCV-018 closes.

2.2.5 R-19

2.2.5.1 Steam generator blowdown isolation valves close.

2.2.5.2 Steam generator blowdown flow control valves close.

2.2.5.3 Steam generator blowdown sample valves close.

2.2.5.4 Steam generator blowdown tank discharge valve V1-31 closes

2.2.6 R-21

2.2.6.1 HVE-15 stops.

2.3 OPERATOR ACTIONS

2.3.1 Immediate Actions

2.3.1.1 Source CHECK the alarming channel AND VERIFY alarm is valid.

2.3.1.2 NOTIFY the E&C/RC group to take samples AND PERFORM a background radiation check for the alarming channel for follow up action as required.

2.3.1.3 VERIFY the appropriate Automatic Actions have occurred as specified in 2.2 "Automatic Actions" have occurred.

Reference for Question 3.12.d

2.0 COMPONENT DESCRIPTION (Continued)

2.21 Discharge Pulsation Dampeners

The discharge pulsation dampeners are spherical type vessels installed in each of the charging pump discharge lines. Their internal baffles reduce discharge pressure pulsations.

3.0 INSTRUMENTATION AND CONTROLS

All valves discussed in this section are operated from the RTGB unless otherwise specified. The failed position for the air operated valves can be found in the system drawing. All instrumentation that provides local indication only can be found in the system drawing.

3.1 Valves

3.1.1 Letdown Stop Valves (LCV-460A and 460B)

Both valves are controlled by one three position switch. The switch positions are: OPEN, CLOSE and AUTOMATIC. These valves are closed automatically on a pressurizer low level alarm and will automatically re-open when the low level alarm clears. These valves are located in "A" reactor coolant pump bay.

3.1.2 Letdown Orifice Isolation Valves (CVC-200A, 200B, 200C)

Three air operated valves are provided to determine which letdown orifices are in service. One orifice will pass 45 gpm and the other two will pass 60 gpm each when the RCS is at normal pressure and letdown pressure is adjusted to approximately 300 psig. Care should be taken not to exceed design flow rate through the demineralizers. These valves are located next to the letdown orifices in the Regenerative Heat Exchanger cubicle. These valves will close on a Phase "A" Containment Isolation Signal (T signal).

Reference for Question 3.13

5.0 **OPERATION** (Continued)

NOTE

Status Light Panels on RTGB will indicate at a glance if all the safeguard valves are in their proper position. Lights will be pink when in the proper position.

NOTE

An SI signal can be reset two minutes after actuation if the status of equipment is to be changed. The initiating signal being present when reset occurs will cause annunciator APP-002-48 and status light AUTOMATIC S.I. SIGNAL OVERRIDDEN TRAIN A TRAIN B to illuminate indicating to operator that automatic Safety Injection is overridden. During this condition, if another automatic Safety Injection signal is generated, a manual SI must be initiated by the operator for Safety Injection to occur. When the initiating SI signal clears, the annunciator and status light will reset.

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CAUTION

UNTIL THE SAFEGUARD SIGNAL IS MANUALLY RESET, ANY SAFEGUARD EQUIPMENT STOPPED FROM RTGB CANNOT BE RESTARTED WITHOUT REMOVING THE CONTROL POWER FUSES AT THE BREAKER AND REINSTALLING THEM. THIS IS DUE TO ANTI-PUMP FEATURE IN THE BREAKERS.

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The Phase "A" Containment isolation, Containment Ventilation isolation, and Feedwater isolation must be reset individually after the SI signal is reset or cleared.

Reference for Question 3.17.b

2.0 COMPONENT DESCRIPTION (Continued)

2.3.7 System Inhibit - The System Inhibit receives two signals from outside of the MIM Equipment, and a manual input from the Utility Circuit Board. REMINH/ is received from RTGB Section 13, MIM selector switch and CRDM/ is received as a differential signal from the Control Rod Cabinet. Either signal true will cause a System Inhibit (SYSINH/ low) unless the Inhibit Override Switch, on the Utility Circuit Board, is closed. SYSINH/ is an input to the MIM Circuit Board and inhibits the detection of impacts by the Input Monitors.

2.3.8 Random Access Memory - One K bytes, of Random Access Memory, are located on the Utility Circuit Board. This memory is functionally a part of the CPU. The Random Access Memory (RAM) is addressed when the Decode Buffer decodes a message addressed to the Utility Circuit Board, RAMPRO from the Power Fail Timer is not true and the address bits from the CPU that define a RAM address are all high.

When RAMADOR/ is low, address bits ADDR0 through ADDR9 from the CPU will address a single location in RAM. If the Board Write Command (BDWRT/) is high, the contents of the RAM at the addressed location will appear on data lines DATA0 through DATA7. If BDWRT/ is low, the data impressed on the data lines by the CPU will be loaded into the addressed location in RAM.

2.3.9 Read-Only Memory - Sixteen K bytes, of Read-Only Memory, are located on the Utility Circuit Board. This memory is functionally a part of the CPU. The Read-Only Memory (ROM) is addressed when the Decode Buffer decodes a message addressed to the Utility Circuit Board and further decodes a PROM address, PROM0/ through PROM7/ and a BDRD/ (Board Read) command is received low from the CPU.

 Reference for Question 4.01 & 7.24

5.0 **PROCEDURE** (Continued) **INITIALS**

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CAUTION

MAINTAIN STEAM FLOW LESS THAN 0.64 x 10^6 LBS/HR PRIOR TO BLOCKING THE SI SIGNAL. A LO STEAM PRESSURE OR LO T-AVG, COINCIDENT WITH A HI STEAM FLOW, WILL INITIATE A SAFETY INJECTION.

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5.2.12 Below 543°F, manually block Safety Injection by placing the T-AVG Block/Unblock switch momentarily in the BLOCK position. _____

5.2.13 Place the Pressurizer Spray Valves in MANUAL. PCV-455A _____
PCV-455B _____

NOTE

Maintain Reactor Coolant System pressure within the pressure/temperature limitations as shown in Figure 3.4 of the Plant Curve Book. Operate RCP's in the combination of "A", "B", and "C", or "A" and "C", or "B" and "C" only. Operation of RCP "B" by itself does not give an effective spray flow.

5.2.14 Slowly OPEN Pressurizer Spray Valve, PCV-455A and/or PCV-455B, to begin cooldown and depressurization of the Pressurizer. PCV-455A _____
PCV-455B _____

Reference for questions 4.17.6 & 7.17.6

10.0 **PROCEDURE** (Continued)

NOTE

Under RIMS, quarterly doses are subject to the following administrative limit system:

Administrative Limit	Available Dose	Documentation Requirements
1250 mRem/qtr	1000 mRem/qtr	No NRC-4 required
3000 mRem/qtr	2400 mRem/qtr	Form NRC-4 required with approved Exposure Extension Authorization

NOTE

The administrative limit is the dose which shall not be exceeded. The available dose incorporates a buffer (20%) to avoid exceeding the administrative limits. The available dose is automatic. calculated by RIMS based on either the administrative limit the individual's employer limit (as provided in writing by the employer), whichever is lower, according to the following method:

Available Dose = (Applicable Limit x 0.8)-(Individual's Current Total Dose)

10.3 Issuance of SRPDs

10.3.1 Upon notification that an individual is to enter the RCA, Dosimetry/Records personnel will sign on RIMS.

10.3.2 Using key function F-2, Log In/Log Out, enter the individual's security badge number. Enter "I" for logging individual SRPD dose in, and press the return key.

10.3.3 If the individual is working under an RWP for steam generator inspection and maintenance, look at the expiration date for this training on the most recent steam generator inspection and maintenance qualification list to assure that the training is still valid prior to issuing SRPDs.

Reference for Questions 4.1.6 & 7.17.6

5.0 GENERAL (Continued)

5.2 Under RIMS, quarterly doses are subject to the following administrative limit system:

Administrative Limit	Available Dose	Documentation Requirements
1250 mrem/qtr.	1000 mrem/qtr.	No Form NRC-4
1500 mrem/qtr.	1200 mrem/qtr.	Form NRC-4 and approved
2000 mrem/qtr.	1600 mrem/qtr.	Exposure Extension
2500 mrem/qtr.	2000 mrem/qtr.	Authorization
3000 mrem/qtr.	2400 mrem/qtr.	

NOTE

The administrative limit is the dose which shall not be exceeded. The available dose incorporates a further buffer (20 percent) to avoid exceeding the administrative limits. The available dose is automatically calculated by RIMS based on either the administrative limit or the individual's employer limit (as provided in writing by the employer), whichever is lower, according to the following method:

Available Dose = (Applicable Limit X 0.8) - (Individuals Current Total Dose)

NOTE

Approval authority for quarterly dose limit extension authorization. The following management positions are authorized to approve the quarterly dose limit extension authorization.

Title	Approval Authority
Plant General Manager	3000 mrem
Manager - E&RC	2500 mrem
RC Supervisor	2000 mrem
RC Foreman	1500 mrem

6.0 PREREQUISITES

6.1 Dosimetry/Records personnel directly involved with data input into RIMS shall be qualified under the guidance of the RIMS Site Coordinator or his designee.

 Reference for Quiz 5.15

POWER COEFFICIENT, α_{power}

It is convenient to combine the various reactivity coefficients into a single coefficient. Although the coefficients are associated with fuel temperature, moderator temperature, pressure and voids, ultimately the quantity of concern is reactor power. Reactor power is easily measurable (as opposed to % voids or fuel temperature, for example) and thus the reactivity changes due to changes in reactor power can be analogous to other reactivity coefficients:

$$\text{power} = \frac{\Delta(\Delta K/K)}{\Delta(\%\ \text{power})}$$

For all practical purposes, the only coefficients which need be considered are the moderator temperature and the fuel temperature coefficients. The void and pressure coefficients are negligible. The power coefficient can be rewritten as:

$$\alpha_{power} = \frac{\alpha_D\, \Delta T_{fuel} + \alpha_m\, \Delta T_{mod}}{\Delta(\%\ \text{power})}$$

OR

$$\alpha_{power} = \alpha_D \frac{\Delta T_{fuel}}{\Delta\%\text{power}} + \alpha_m \frac{\Delta T_{mod}}{\Delta\%\text{power}}$$

OR

$$\alpha_{power} = \alpha_D \text{ only power} + \alpha_m \frac{\Delta T_{mod}}{\Delta\%\text{power}}$$

The Doppler and moderator temperature contributions to the power defect vary over the full range of power. At hot zero power, the moderator and fuel temperatures are at the same temperatures $\approx 547^{\circ}F$. As reactor power increases, T_{fuel} increases to $\approx 1500^{\circ}F$ at 100% power. However T_{mod} increases to $\approx 575.4^{\circ}F$ at 100% power. T_{fuel} changes $\approx 953^{\circ}F$ while T_{mod} changes $\approx 28.4^{\circ}F$. Therefore α_D is the dominant factor of the power coefficient. As seen in RXTH-TP-32.2, the shape of the α_{power} curve is similar to the α_D curve.

The power coefficient is **always negative.**

It becomes slightly less negative at higher powers, similar to α_D. Also α_{power} is more negative for lower boron concentrations, EOL. This is due to two effects. The α_m is more negative at EOL and α_D is slightly more negative at EOL. Typical values are ≈ -15 pcm/% power at BOL and ≈ -20 pcm/% power at EOL.

The power coefficient, always being negative, contributes to reactor control and safety. Negative reactivity will always be introduced if there is an abrupt increase in reactor power. This feedback will limit power excursions. α_D acts immediately, α_v next, but α_m takes several seconds to become effective.

POWER DEFECT

The power defect is used to find the reactivity changes for a power change.

$$\rho_{power} = \alpha_{power} \, \Delta \% \text{ power}$$

The power defect curve, RXTH-TP-32.3, shows the $\Delta\rho$ for a power change of zero power to some power level. Note that the power defect is always negative for a power increase.

The total power defect increases from BOL to EOL. For power increases of 0% to 80% typical values are:

BOL ≈ -1200 pcm
EOL ≈ -1900 pcm

Example for the use of the power defect curve: A reactor is operating at 80% power. The power must be decreased to 50% to perform maintenance on a feed pump. How much reactivity is added due to decreasing power. The original boron concentration is 300 ppm.

Use RXTH-TP-32.3. The reactivity change from 80% to 0% is +1720 pcm. Reactivity change from 0% to 50% is -1140 pcm. Therefore the reactivity change from 80% to 50% = +1820 + (-1140) = +580 pcm.

OUTLINE	KEY AIDS
a. Frequency	
1) Dependent on generator speed	$F = \frac{NP}{120}$
2) Generator speed controlled by EH System prior to closing generator output breakers	
3) Following closing of output breakers generator speed will match system frequency and LOAD will be controlled by EH System	
4) 60 Hz corresponds to a speed of 1800 RPM	
b. Megawatts	
1) Generator megawatts is displayed on the RTGB and indicates the loading on the Main Generator	
2) Controlled by the setting of the Electro-hydraulic System	
c. Megavars	
1) Displayed on RTGB	
2) Positive VAR's indicates an inductive load and that current lags voltage	

OUTLINE	KEY AIDS
3) Negative VAR's indicate a capactive load and that current leads voltage	
4) MVAR is product of apparent power (MVA) and sine of angle between true power (MW) and apparent power (MVA)	GEN-TP-1.7 $MVA = \sqrt{MW^2 + MVAR^2}$
5) Cosine θ = MW/MVA = Power Factor (PF)	
a) Power factor is an indication of how much of the generator output is being converted into useful work	
b) VAR's can be positive or negative but PF is always positive	
d. Current	
1) Indicted on RTGB for each output phase	
2) Armature current is representative of generator load	
3) Exciter field current controls the exciter field strength which in turn controls the main field current and voltage	
4) Exciter field current controlled by a base signal from base adjuster plus an	

Reference for Question 5.19

The generator is internally cooled by circulating hydrogen gas through the interior of the machine. The hydrogen is circulated by a blower mounted on the turbine end of the generator shaft. Four coolers mounted near the turbine end of the generator remove the heat absorbed by the hydrogen.

The hydrogen used to cool the generator must be maintained at high concentrations and purity levels to prevent an explosive mixture and/or moisture buildup in the generator. The hydrogen is contained within the generator and leakage along the shaft is prevented by a dual seal arrangement that receives oil from the generator Seal Oil System. The shaft seals are located at each end of the generator shaft where it penetrates the casing. The seals contain the hydrogen within the casing by forming a leak tight boundary and must be supplied with seal oil whenver the generator is filled with hydrogen.

Operational controls and indications for the generator and turbine are provided remotely on the control board and locally on the hydrogen and stator cooling control panel. Normally the generator is running loaded and supplying power to the 230KV Transmission System. The 230KV System is an infinite source of power with respect to the generator and thus any attempts to alter the generator terminal voltage or frequency will have negligible affect upon the voltage or frequency of the 230KV System. With the generator paralleled to the 230KV System, the generator megawatt output will be controlled by the setting of the Turbine Electro-Hydraulic System while power factor and reactive load will be controlled by varying the field current.

B. INSTRUMENTATION AND CONTROL

1. Instrumentation

MVAR's

VAR's (Volt-amps Reactive) are displayed on the control board and indicate the type of loading on the generator. Positive VARs indicate an inductive load on the generator and the machine will operate with a lagging power factor (current lags the voltage). Negative VARs indicate a capacitive loading and

the machine power factor will be leading. GEN-TP-1.7 is a vector representation of these two conditions. As shown in the GEN-TP-1.7, the angle θ is the angle between the generator MW output (true power) and the MVA output (apparent power). This angle θ is also the angle between machine terminal voltage and current. The cosine of θ is called the power factor and is the ratio of true power (MW) to apparent power (MVA). Cos θ = MW/MVA = Power Factor (PF). The power factor is an indicator of how much of the generator output is being converted into useful work. As can be seen in the GEN-TP-1.7. the VARs can be either positive (lagging) or negative (leading) but the power factor is always a positive number.

There are no Technical Specifications associated with the Main Generator System. However, it must be operated within the confines of a calculated capability curve.

Generator Capability Curve

GEN-TP-2.3 shows the capability curve for this generator. The curve is drawn so as to limit the hot spot temperatures in the stator and rotor windings, and in the stator core, to practical operating values, and to limit the temperature differential across the insulation of the windings. These capabilities are determined by calculation and shop test, and as explained above, are not directly determinable from the usual observed temperatures during commercial operation.

Operation in the range between zero P.F. overexcited and rated P.F. is limited by rotor winding temperature. This curve gives operation with the field amperes constant at the name plate value corresponding to the gas pressure under consideration.

The region between rated P.F. (overexcited) and 100 P.F. is limited by stator winding temperature. Operation on this portion of the curve corresponds to constant stator amperes. In this region, the field current will vary with load and power factor but will always be less than the maximum allowable value.

Reference for Question 6.04.d

4.0 SYSTEM AND COMPONENT DESIGN (Continued)

4.6 RTD - Minco Model S8809

100 ohm platinum 4-wire

4.7 Valves

Autoclave Engineers, Inc.
1/4" x 28I - 1/4" Magnetically Operated Ball Valve
A.E. Cat. No. 2.5 MV4061
No. 30-9480 Rev. 2

Rockwell - Edward Hermavalve
3/4 T78 - 3/4" Root Valve
Dwg. No. ACD-31602608

5.0 OPERATION

5.1 Normal Operation

5.1.1 The table below shows the expected normal level readings for various plant conditions.

PLANT CONDITIONS	UPPER RANGE LI-511AA/LI-511AB	FULL RANGE LI-511AB/LI-511BB	DYNAMIC HEAD LI-511AC/LI-511BC
HEATUP			
- No Pumps	106%	100%	33%
- 1 Pump	*	120% (off scale)	50%
- 2 Pumps	*	120% (off scale)	75%
- 3 Pumps	67% (off scale)	120% (off scale)	100%
TAVG - No Load	67% (off scale)	120% (off scale)	100%
Full Power	67% (off scale)	120% (off scale)	110%

* If the instrument loop RCP is not running, the upper range will read approximately 90%. If one of the pumps is in the instrumented loop, the reading is off scale low.

Reference for Question 6.10 Part 2.

3.0 INSTRUMENTATION AND CONTROLS (Continued)

3.1.5.6 Overpower ΔT Trip (OPΔT)

a. The Overpower ΔT Trip provides protection for the Reactor against exceeding the Linear Power Rating (KW/ft) of the fuel rods and trips the Reactor when 2 out of 3 Reactor Coolant Loop ΔT's (Th-Tc) exceeds the calculated T Setpoint. The OPΔT Setpoint is not a fixed number. It is calculated continuously with inputs from Tavg and Power Range Current Signals f(ΔI). To help prevent this trip from occurring, when the OP T nears the Reactor Trip Setpoint:

1. Control Rod withdrawal is blocked
2. A Turbine Runback is initiated

b. Setpoint -

$$\Delta T_0 \left[K_4 - K_5 \left(\frac{T_3 S}{1 + T_3 S}\right) T - K_6 (T - T') - f(\Delta I)\right]$$

ΔT_0 = indicated ΔT at nominal full power, in °F, for the channel being calibrated

T = Average temperature, °F

T' = indicated average temperature at nominal conditions and rated power, for the channel being calibrated

K_4 = 1.06

K_5 = 0 for decreasing average temperature
0.02 per °F for increasing average temperature

K_6 = 0.00277 per °F for T>T'
0 for T≤T'

(TM-412L, TM-422L, TM-432L) T_3 = 10 seconds

(NM-412B, NM-422B, NM-432B) f (ΔI)

The overtemperature ΔT reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution provided only that (1) the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds)[4], and (2) pressure is within the range between the high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors,[2] is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to Specification 2.3.1.2.d.

The overpower ΔT reactor trip prevents power density anywhere in the core from exceeding 118% of design power density as discussed in Section 7.2.2 of the FSAR and includes corrections for axial power distribution, change in density, and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified setpoints meet this requirement and include allowance for instrument errors.[2]

The setpoints in the Technical Specifications ensure the combination of power, temperature, and pressure will not exceed the core safety limits as shown in Figure 2.1-1.

The low flow reactor trip protects the core against DNB in the event of a sudden loss of power to one or more reactor coolant pumps. The setpoint specified is consistent with the value used in the accident analysis.[5] The undervoltage and underfrequency reactor trips protect against a decrease in flow caused by low electrical voltage or frequency. The specified setpoints assure a reactor trip signal before the low flow trip point is reached.

The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. Approximately 1150 ft^3 of water corresponds to 92% of span. The specified setpoint allows margin for instrument error[2] and transient level overshoot beyond this trip setting so that the trip function prevents the water level from reaching the safety valves.

 Reference to Question 6.13 Case 2.

4.0 PRECAUTIONS AND LIMITATIONS (Continued)

4.11.3 An Inverse Count Rate Plot (1/M) with a minimum of four (4) data points (including baseline data point) must accompany the Reactor startup.

4.11.4 Reactor Coolant temperature must be above the minimum criticality limit specified on the Pressure/Temperature Limitation Curve for heatup (Tech. Spec. Figure 3.1-1) before taking the Reactor critical.

4.11.5 The Reactor shall be maintained at least 1% Δ K/K shutdown until normal water level is established in the Pressurizer.

4.11.6 Do not bring the Reactor critical until a steam bubble has been formed in the Pressurizer.

4.11.7 Do not bring the Reactor critical with a positive moderator temperature coefficient greater than +5 PCM/°F up to 50% power, decreasing linearly to 0 PCM/°F at 100% rated power.

4.12 Set the Steam Dump PORV setpoint at 30%. This corresponds to a lift setting of 1035 psig.

4.13 Critical operation is allowed with the Main Steam Isolation Valves closed and under Clearance. This does not violate the intent of Technical Specifications.

4.14 Whenever possible, use Condenser Dump Valves for temperature control instead of PORV's.

4.15 Feedwater additions during Hot Shutdown should be controlled to minimize the thermal stress cycles on the feedwater nozzle.

Reference for Question 617 Part b.

4.0 OPERATION

4.1 Normal Operation

The SFP cooling loop will be in operation anytime there is spent fuel in the SFP. The SFP purification loop will normally be connected to the SFP cooling loop, utilizing the differential pressure across the SFP heat exchanger to meet flow requirements. The SFP purification system can be isolated from the SFP cooling system and lined up to purify the RWST using the refueling water purification pump. The SFP can also be filled from the RWST. The SFP water can also be used to heat the RWST water.

4.2 Refueling Operation

This system can be used during refueling in the lineups discussed under normal operation (Section 4.1).

Valves are available for using the mixed bed demineralizers in the CVCS system with the flow returning to the RCS via the high head safety injection header.

The draining of the refueling cavity lower section is accomplished by opening a valve to the RCDT and pumping with the refueling purification pump through the demineralizer and filter to the RWST. Finally, draining of the cavity is to the CV sump.

4.3 SFP Skimmer Operation

The SFP skimmer system can be operated at any time. It is normally operated to clean the surface of the pool prior to a fuel handling operation.

 Reference for Question 6.17 Part b.

5.0 PROCEDURE (Continued) INITIALS

5.4.2.18 Realign the RCDT and associated systems as follows:

1. CLOSE RCDT to Gas Analyzer Header Vent, WD-1609. _____
2. OPEN the RCDT Vent Isolation, WD-1716. _____
3. OPEN RCDT to Gas Analyzer Isolation, WD-1717. _____
4. OPEN Collection Tank Transfer Pumps Combined Discharge, CVC-409. _____
5. OPEN RCDT/Charging Pump Leakoff to CVCS HUT, CVC-1101. _____
6. CLOSE RWP Pump Suction from RCDT, SFPC-804. _____

5.4.2.19 Align the Spent Fuel Pit Cooling System for the desired mode in accordance with OP-910. _____

5.4.2.20 To drain the lower Refueling Cavity, OPEN the Lower Cavity Drain to Containment Sump, WD-1757C. _____

5.4.2.21 After draining is complete, CLOSE WD-1757C. _____

NOTE

Perform the following steps after notification from Maintenance that the Fuel Transfer Tube blind flange has been installed.

5.4.2.22 Verify CLOSED the Fuel Transfer Tube Isolation Valve. _____

5.4.2.23 OPEN Fuel Transfer Tube S-36 Blind Flange Water Seal Isolation, PP-111D. _____

5.4.2.24 Leak test the Fuel Transfer Tube blind flange for Penetration Pressurization System air leakage. _____

Reference for Question 8.19

3.3.1.3 When the reactor is in the hot shutdown condition, the requirements of 3.3.1.1 and 3.3.1.2 shall be met. Except that the accumulators may be isolated, and in addition, any one component as defined in 3.3.1.2 may be inoperable for a period equal to the time period specified in the subparagraphs of 3.3.1.2 plus 48 hours, after which the plant shall be placed in the cold shutdown condition utilizing normal operating procedures. The safety injection pump power supply breakers must be racked out when the reactor coolant system temperature is below 350°F and the system is not vented to containment atmosphere.

3.3.1.4 When the reactor is in the cold shutdown condition (except refueling operation when Specification 3.8.1.e applies), both residual heat removal loops must be operable. Except that either the normal or emergency power source to both residual heat removal loops may be inoperable.

a. If one residual heat removal loop becomes inoperable during cold shutdown operation, within 24 hours verify the existence of a method to add make-up water to the reactor coolant system such as charging pumps, safety injection pumps (under adequate operator control to prevent system overpressurization), or primary water (if the reactor coolant system is open for maintenance) as back-up decay heat removal method. Restore the inoperable RHR loop to operable status within 14 days or prepare and submit a Special Report to the Commission within the next 30 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the loop to operable status.

b. If both residual heat removal loops become inoperable during cold shutdown operation, close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere prior to the reactor coolant average temperature exceeding 200°F, restore at least one residual

Reference for Question 8.21.a.

5.0 **PLAN** (Continued)

Finally, the position of Site Emergency Coordinator is established to be activated immediately on declaration of an emergency. To that individual is delegated the immediate and unilateral authority to act on behalf of the Company to manage and direct all emergency operations involving the facility. Upon activation of the EOF, the Emergency Response Manager assumes responsibility of overall emergency response and performs those requirements for all off-site related activities. The Site Emergency Coordinator maintains overall on-site emergency response responsibilities and reports to the Emergency Response Manager.

Initially the Site Emergency Coordinator would be the Shift Foreman. He would act in that capacity until formally relieved by the designated Site Emergency Coordinator - the General Manager (or his alternate). In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but the Plant General Manager is expected to manage the emergency response as soon as he is available to do so--in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency.

This section of the plan delineates the various emergency actions and separates them into groups of related functions. These functions are then assigned to emergency "teams" with designated leaders who are responsible to the Site Emergency Coordinator for the performance of the activities required to fulfill those functions.

Upon the declaration of an emergency, specified on-shift individuals are assigned as interim leaders (i.e., a designated interim leader is always available on site). Such individuals assume the responsibility for performing the required emergency response actions until properly relieved by the assigned team leader or one of his alternates. All team leaders, alternates, and interim leaders are trained as described in Section 5.6.1.1.

If necessary, the Site Emergency Coordinator will allocate available resources based on existing plant conditions. Where necessary, additional personnel will be notified and requested to augment on-site personnel.

A current call list of the Emergency Response Organization is maintained in the Plant Emergency Procedures and is available to make this notification. Since most of the Robinson plant management staff and substantial numbers of its support personnel live in the site vicinity (i.e., Hartsville and surrounding areas) additional assistance can be quickly provided.

 Reference for Question [illegible], b

5.0 PROCEDURE (Continued)

A deviation is permitted that departs from a license condition, a Technical Specification, or the Plant Operating Manual in an emergency when the action is immediately needed to protect the public health and safety and no action consistent with license conditions and Technical Specifications that can provide adequate or equivalent protection is immediately apparent. Further, this type of deviation is required to be taken by 10CFR50.54 as interpreted in the Federal Register if, in an emergency, protective action is needed and no action consistent with the license that can provide adequate or equivalent protection is immediately apparent.

The approvals required and the order of notification depend on the urgency of the protective action required. The guidelines listed below are to be followed:

A. If sufficient time exists, the Shift Foreman shall consult with another member of the Plant Management staff prior to approval of the deviation and subsequent implementation.

B. If sufficient time does not exist, the Shift Foreman shall approve of the deviation prior to performing the protective action.

C. The NRC must be notified, via the red phone, if the protective action would violate a technical specification or license condition. The NRC must be notified prior to performing the protective action if time permits; otherwise, the notification must be made as soon as possible thereafter.

The approved deviation will be entered in the Shift Foreman's Log and reported to the Cognizant Supervisor. The departure and circumstances surrounding the departure will be submitted to the Plant Nuclear Safety Committee by the Cognizant Supervisor.

OUTLINE	KEY AIDS
function of those structures, systems, and components important to plant safety.	
2) Any event resulting in manual or automatic actuation of Engineered Safety features, including the reactor protection system.	
3) Transport of a radioactively contaminated, injured individual to an off site hospital.	
4) Any event meeting the criteria of 10CFR20.403 for notification.	
5. Deviations from Technical Specifications	PROC-TP-25.2
a. New paragraph added to 10CFR50:	
50.54 Conditions of Licenses	
(X) A licensee may take reasonable action that departs from a license condition or a technical specification in an emergency when this action is immediately needed to protect the public health and safety and no action consistent with license conditions and technical specifications that can provide adequate or equivalent protection is immediately apparent.	
(Y) Licensee action permitted by paragraph (X) of this section shall be approved, as a minimum, by a licensed senior operator prior to taking the action.	Approval is needed by a shift foreman by the Administrative Procedures