ENCLOSURE 3

CP&L

Carolina Power & Light Company

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

December 17, 1987

File: NTS-4208(F) Serial: NO-87R317

Mr. Kenneth E. Brockman
Operator Licensing Section
U. S. Nuclear Regulatory Commission, Region II
Suite 2900
101 Marietta Street, N.W.
Atlanta, Georgia 30323

SUBJECT: Senior Reactor Operator Written Examinations

Dear Mr. Brockman:

On December 15, 1987, Mr. J. A. Arildsen of the USNRC administered written senior reactor operator examinations at H. B. Robinson. Enclosed, please find comments for the examination.

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

R. S. Allen
A. McGilvray
D. A. Neal
D. K. Seagle
R. R. Stebbins

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

Very truly yours,

REMorgan

R. E. Morgan
General Manager
H. B. Robinson SEG Plant

RSA:eaw

cc: J. L. Harness

Enclosures (2)

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NRC EXAMINATION COMMENTS

H. B. ROBINSON SRO CLASS 87-1

EXAMINATION DATE - DECEMBER 15, 1987

1. Section 5

<u>Question 5.04, Part 3</u>

Recommend accepting "a" or "b" as answer. Depends on interpretation of "vs.". All rods in is more negative than all rods out and all rods out is less negative than all rods in.

No reference necessary.

<u>Question 5.04, Part 4</u>

Recommend also accepting "a" or "c" as correct answers. Large commercial reactors have negligible leakage.

Reference: Rx Theory Handout, Session 25, Page 5

<u>Question 5.05</u>

Recommend also accepting a correct answer describing the relationship of "P" and "F" to MTC.

Reference: Rx Theory Handout, Session 25, Page 2
Rx Theory Handout, Session 18, Page 6

<u>Question 5.07, Part 1</u>

Recommend deleting question. The outer fuel always shields inner fuel but fuel self-shielding is reduced (as described in Part 2).

Reference: Rx Theory Handout, Session 28, Pages 2 and 3

<u>Question 5.10, Part 2</u>

Recommend deleting question. Three (3) basic concepts are not included in the theory program as such.

Reference: Rx Theory Handout, Session 32, Page 4

<u>Question 5.11, Part 1</u>

Recommend accepting as answer "The initial sharp drop is due to buildup of fission products". Last part of answer should not be required.

No reference necessary.

Question 5.12, Part 2

Delete question. The question does not test operator knowledge of the relationship between boron concentration and rod worth. Statement was taken out of context on a discussion of rod worth.

Reference: Rx Theory Handout, Session 35, Page 3

Question 5.13

Recommend also accepting as complete answer the equation:

Rod Worth $\propto$ KAL $\left(\frac{\phi local}{\phi avg}\right)^2$ with explanation of high local flux and low average flux.

Reference: Rx Theory Handout, Session 35, Page 5

Question 5.14

Recommend adding a band of ± 100 pcm (.001 $\Delta K/K$)

No reference necessary.

Question 5.15

Recommend also accepting as a alternate answer:

$\dot{Q} = \dot{M} cp \Delta T$

$$\dot{M} = \frac{(2700\ mw)\ (3.413 \times 10^6\ BTU/hr/mw)}{Cp\ (606 - 554)} = \frac{9.2151 \times 10^9}{(1.3625)\ (52)} = 1.3006 \times 10^8$$

- Cp Calculated:

  Cp based on a TAVG of 580°F

  From Steam Table

  H_f at 576°F = 583.7

  H_f at 584°F = 594.6

  $$Cp = \frac{594.6 - 583.7}{584 - 576} = \frac{10.9}{8} = 1.3625$$

  OR

  Cp Assumed (of ≈ 1.3)

Reference: HT&FF Lesson Plan 27, Page 4
HT&FF Lesson Plan 4, Page 9

Question 5.18, Part 2

Recommend accepting any power change, not just up power maneuver. Recommend accepting any evolution that results in rod movement.

Reference: Fuel Management Procedure, FMP-007, Page 5

Question 5.19

Recommend grading by general shape of curve without using given percentages. The reference given is from Shearon Harris Nuclear Power Plant and therefore is not plant specific.

No reference necessary.

Question 5.20, Part 1

This question is vague in nature and had to be explained by the proctor. Recommend that this question be reworded if used in future.

No reference necessary.

2. Section 6

Question 6.02, Part 1

Recommend change answer to read "Loss of power to bus E-1 will energize loads associated with blackout sequencer. Loss of power to bus E-1 with an SI will energize safeguards loads by the safeguards sequencer". NRC reference referred to difference in breaker lineups, not to difference in sequencers.

Reference: H. B. Robinson Logic Diagrams, Page 14 of 18, "Safeguards Sequence"

Question 6.02, Part 2

For the given answer to be correct, an SI signal must be present. This may or may not have been assumed by the candidates. The correct answer for only a blackout is "Normal supply and tie breakers (52/18B and 52/22B, and 52/28B and 52/29B number memorization not required) open. Recommend also using this answer.

Reference: H. B. Robinson Logic Diagrams, Page 13 of 18, "Emergency Generator Start and Bus Clearing"

Question 6.02, Part 3

Recommend also accepting as an answer "When voltage/power has been returned to the emergency bus". This is synonymous with generator output breakers closed.

Reference: H. B. Robinson Logic Diagrams, Pages 13 and 14 of 18

Question 6.03

Recommend not requiring exact temperature at which hydrogen recombination occurs. Recommend only accepting hydrogen recombination occurs at high temperatures.

No reference required.

Question 6.04

Recommend deleting question as:

1. System is cautioned tagged out and is used for information only.

2. Question is a double jeopardy question. (If you can not answer Part 1, then you can not answer Part 2 or Part 3.)

Reference: H. B. Robinson Caution Tag Log
NUREG 1021, ES107, Page 2

Question 6.05

Recommend accepting "words to the effect of" a function generator (i.e., comparator).

No reference required.

Question 6.06, Parts 1 and 2

Parts 1 and 2. Recommend accepting NIS rod drop signal or 5%/5 seconds since both terms are synonymous.

Part 2. Recommend accepting rod bottom signal from any rod or rod bottom signal from RPI.

Reference: System Description, SD-10, Page 26
H. B. Robinson Logic Diagrams, Page 4 of 18, "Nuclear Instrumentation Permissives and Blocks

Question 6.09

Total question is worth three (3) points. However, individual sections add up to 3.5 points. Recommend Part 2 answer be as follows with 1.5 points assigned. "If manual signal fails meter will deviate from zero (0.5). If automatic signal fails meter will deviate from zero (0.5). If meter deviates from zero, turbine will revert to manual (0.5)."

Reference: System Description, SD-033, Page 49

Question 6.12, Part 1

Recommend accepting for RC 519 A and B primary water makeup to PRT or primary water makeup to CV since these valves isolate all primary water to CV

Reference: H. B. Robinson Drawing Number 5379-1971, Page 2 of 2

Question 6.13, Part 1

Recommend accepting as correct answer "A reactor trip and low TAVG or a safety injection signal or S/G high level".

Reference: H. B. Robinson Logic Diagrams, Page 10 of 18, "Feed Water Isolation"

3. Section 7

Question 7.01

Recommend also accepting:

Selection number one (1) because H. B. Robinson Plant is allowed by Technical Specifications to have up to +5 pcm/°F moderator temperature coefficient up to 50% power.

Reference: H. B. Robinson Technical Specifications, Page 3.1-11

Question 7.04

Parts 1 and 2 have had the answers reversed. Recommend changing answer to:

Part 1 - 10%

Part 2 - 1%

Reference: Technical Specifications 3.10.8, Page 3.10-10

Question 7.08

Recommend also accepting "Maintain adequate SDM" verses "Diluting the RCS below the shutdown margin". Recommend redistribute point value to have each part worth 0.75 points.

No reference required.

Question 7.10, Part 2

Recommend for full credit accepting "Fire Protection Building". Specific location within building is not asked for.

No reference required.

Question 7.10, Part 3

Recommend for secondary control panel also accepting Turbine Building Mezzanine Level.

Reference: Abnormal Operating Procedure, AOP-004, Page 3

Question 7.13

Recommend accepting valve number or noun name for correct answer.

No reference required.

Question 7.15

Recommend not requiring "Stop all RCP's if both conditions listed below are met" since this is implied by question statement. Should accept for full credit:

1. SI pump, at least one running (0.5) and (0.5.)

2. RCS subcooling, less than 25°F (45°F) (0.5).

No reference required.

Question 7.16, Part 1

Recommend change answer to "Turn the pumps off". Guidance on use of foldouts requires action to be taken once recognized.

Reference: OMM-002, Page 10
EPP Foldouts (A), Page 4

Question 7.16, Part 2

NRC reference assumes RVLIS in service. This in not true of the H. B. Robinson Plant. Recommend accepting the answer for tripping pumps based on WOG background basis included in attached reference.

Reference: EOP Handout 1, Session 3, Page 5
ERG Executive Summary, Pages 60 and 61

Question 7.19

Recommend deleting question. The SCO/SF is not responsible for Radiation Control Supervisor's duties and responsibilities. Radiation Control Supervisor reports to Manager - Environmental and Radiation Control.

Reference: Administrative Procedure, AP-001, Pages 34 and 42

4. Section 8

Question 8.08, Part 5

Recommend also accept both as an answer. The SF or SCO can approve a work request.

Reference: Maintenance Management Procedure, MMM-003, Page 11

Question 8.14

Recommend also accepting positions noted in the attached reference (AP-002, Pages 15 and 16) for on call Manager and on call Supervisor as allowed in PEP-204 (ERO Directory).

Reference: PEP-204, Page 8
AP-002, Pages 15 and 16

Question 8.16, Part 2

Recommend answer to read "NRC is notified prior to deviation if time permits or as soon as possible afterwards (0.5). NRC is notified via red telephone (ENS) (0.5). "Criteria for notification" is a vague statement and is not addressed in the Plant Operating Manual.

Reference: AP-006, Page 7

Question 8.17, Part 2

Recommend also accept as correct answer:

1. Plant Technical Specifications or

2. E&RC Procedure, EMP-023, or

3. OMM-008 (Minimum Equipment List)

Reference: EMP-023, page 25
Technical Specifications, Page 3.16-2
OMM-008, Page 33

Question 8.18

Recommend deleting this question. Record retention is the responsibility of Document Control. Document Control is the responsibility of the Administrative Supervisor and the Manager - Control and Administration.

Reference: AP-001, Pages 24 and 38
Technical Specifications, Page 6.1-1

EXAMINATION GENERAL COMMENTS

1. It was noted in the examination that excessive verbatim knowledge of background documents were asked rather than actual procedures from the Plant Operating Manual. Recommend testing the operators for knowledge of procedures rather than E.R.G. documents (i.e., Section 7 had four (4) questions from E.R.G.'s which totaled 27.27% of Section 7).

2. Two (2) questions asked for memorization of abnormal operating procedures subsequent actions. Memorization of subsequent actions is not required.

3. Throughout the examination specific wording seemed to be required for answers. Recommend accepting reasonably worded correct responses.

4. The theme of the examination strayed from the operational oriented examinations that H. B. Robinson has received in the recent past.

REFERENCE FOR 5.04 PART 4

RxTh-TP-25.2

MODERATOR COEFFICIENT

$$\alpha_m = \frac{\Delta\left(\frac{\Delta K}{K}\right)}{\Delta T_m}$$

$$\alpha_m \approx \frac{1}{P}\frac{\Delta P}{\Delta T_m} + \frac{1}{f}\frac{\Delta f}{\Delta T_m} + \frac{1}{L}\frac{\Delta L}{\Delta T_m}$$

$$\frac{\Delta P}{\Delta T_m} < 0$$

$$\frac{\Delta f}{\Delta T_m} > 0$$

$$\frac{\Delta L}{\Delta T_m} \approx 0$$ **For Large Reactors**

MODERATOR TEMPERATURE COEFFICIENT, α_m

The moderator temperature coefficient relates changes in reactivity to changes in moderator temperature. Operationally, it is defined:

$$\alpha_m = \frac{\Delta\ (\Delta K/K)}{\Delta T_{mod}}$$

where ΔT_{mod} is the change in moderator temperature. Since reactivity is defined in terms of the multiplication factor, K, it is necessary to examine how moderator temperature changes affect the multiplication factor. The physical result of a temperature change is a change in density. For the moderator, an increase in temperature results in a decrease in density (for the temperature ranges in which the reactor normally operates), and so this effect must be translated into a change in K. This is most easily accomplished by determining the effect of a density change on each term in the six factor formula.

The six factor formula is $K = \epsilon L_{f} p L_{t} f \eta$. The moderator density affects K primarily through the thermal utilization factor, f, the resonance escape probability, p, and the non-leakage probabilities, L_t and L_f. RXTH-TP-25.2 illustrates the calculation of the moderator temperature coefficient in terms of the changes in these quantities.

An increase in temperature results in a decrease in density, with an accompanying increase in slowing down length and thermal diffusion length. This is to be expected since the moderator molecules are further apart, hence neutrons travel further between collisions. Since the slowing down length increased, the slowing down time also increased, which means that neutrons spend more time at resonance energies. This reduces their probability of resonance escape, and so $\Delta p / \Delta T_{mod}$ is negative, i.e., p decreases with increasing temperature.

On the other hand, thermal neutron absorption in the moderator decreases, due to the fact that there is less moderator in the core area and hence a reduction in the number of moderator molecules which can absorb neutrons. This increases the probability of thermal neutron absorption in the fuel, and so the thermal utilization increases, i.e., $\Delta f / \Delta T_{mod}$ is positive.

REFERENCE FOR 5.05

SIMPLIFIED THERMAL UTILIZATION EQUATIONS

$$f \simeq \frac{\Sigma_a^{Fuel}}{\Sigma_a^{Fuel} + \Sigma_a^{Mod} + \Sigma_a^{o}}$$

For one fuel type:

$$f \simeq \frac{\sigma_a^{f}}{\sigma_a^{f} + \left(\frac{N_m}{N_f}\right)\sigma_a^{m} + \left(\frac{N_o}{N_f}\right)\sigma_a^{o}}$$

REFERENCE FOR 5.07 PART 1

SELF-SHIELDING

The last section explained how fuel temperature affected resonance absorption peaks through Doppler broadening. It was pointed out that for one atom the total absorption probability does not change with different fuel temperatures. This section will show how Doppler broadening and fuel temperatures, when applied to many atoms, does effect the number of resonant energy neutrons absorbed. This is described by the process known as self-shielding. Thin and thick target examples will be presented to illustrate self-shielding.

The thin target example, RXTH-TP-28.2, shows neutrons with different energies incident on a target. The target is shown at two different temperatures. A thin target actually means a target one atom layer thick. The probabilities of microscopic cross section apply. At a low temperature (temp 1), the resonance absorption peaks are narrow. The neutrons possessing the resonant energy, E_3, have a very high probability for absorption and many are absorbed. The off-resonant energy neutrons have a low absorption probability and few of these are absorbed. At a high temperature (temp 2), Doppler broadening occurs. The resonant energy neutrons, E_3, with a decreased absorption probability, are absorbed less by the target than at the lower temperatures (100 vs 10). The off-resonant energy neutrons have a higher absorption probability and more of these are absorbed than at the lower temperatures. However, the total fraction of neutrons absorbed is the same for both temperatures. This is as expected since, as stated previously, microscopic cross section probabilities apply in this example -temperature does not effect the total absorption probability for one atom.

The thick target example is presented in RXTH-TP-28.3. A thick target is two atom layers thick or is the same as placing two thin targets together. Again, at a low temperature (temp 1), resonance absorption peaks are narrow. As seen in the thin target example most resonant energy neutrons, E_3, are absorbed in the first layer - about 190. In the next layer only about 9 are absorbed. There are much less available for absorption in the second layer. This is called self-shielding - the first layer is shielding the second layer from resonant energy neutrons. Most of the off-resonant energy neutrons are still available for absorption after the first layer, but still not many are absorbed in both layers. Now at a high target temperature, Doppler broadening occurs. More resonant energy neutrons are available for absorption after the first layer than in the low temperature case (about 90 more). Many E_3 neutrons are absorbed in the second layer - amount of self-shielding has been reduced. In this case more off-resonant energy neutrons are absorbed and the total fraction of all neutrons absorbed is increased. This fact is very important and is due to the reduction of self-shielding of resonant energy neutrons. For the thick target example, the temperature does affect the fraction of neutrons absorbed.

Now the concept of self-shielding will be applied to the reactor and the fuel pellets.

FUEL SELF-SHIELDING

At low fuel temperatures a neutron with the exact resonant energy has a very high probability of absorption and will most likely be absorbed in the outer edge of the fuel pellet. Neutrons of other than resonant energies will not slow significantly in the fuel and so they will likely pass directly through the pellet without being absorbed. (The outer fuel shields the inner fuel from the resonant energy neutrons at low fuel temperatures.)

At higher fuel temperatures resonant absorption peaks broaden. Neutrons possessing a resonant energy will have a slightly less probability to be absorbed near the pellet edge but still have a high probability of being absorbed somewhere in the pellet. Neutrons with energies slightly off the resonant value have an increased probability of absorption and some will most likely not make it all the way through the fuel pellet. Therefore there is a greater fraction of neutrons absorbed. (As fuel temperature increases the amount of fuel self-shielding is reduced and more neutrons are absorbed in the fuel pellet.)

DOPPLER COEFFICIENT

The Doppler coefficient, α_D, is defined by:

$$\alpha_D = \frac{\Delta \rho}{\Delta T_{fuel}} = \frac{\Delta (\Delta K/K)}{\Delta T_{fuel}}$$

To explain the Doppler coefficient, an examination of how an increase in fuel temperature effects the six factor is needed. Since increases in fuel temperature result in increased resonant absorption (Doppler broadening and reduction in self-shielding), the resonance

REFERENCE FOR 510 PART 2

has balanced out the initial positive reactivity added to the reactor by the control system. Obviously, a reactor which responds in this manner is self-limiting. Therefore, a large power defect is advantageous for safety reasons. However, one aspect which must be considered is the requirement for the control system to add sufficient positive reactivity to overcome the power defect and allow the reactor to increase to full power.

Power defect is determined as follows:

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

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 100% typical values are:

BOL ≈ -1700 pcm
EOL ≈ -2300 pcm

Example for the use of the power defect curve: A reactor is operating at 75% 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 500 ppm.

Use RXTH-TP-32.3. The reactivity change from 75% to 0% is +1375 pcm. Reactivity change from 0% to 50% is -925 pcm. Therefore the reactivity change from 80% to 50% = +1375 + (-925) = +450 pcm.

It must be pointed out that the power coefficient and defect are only valid when the power changes with T_{avg} on the normal program. These curves are calculated for specific $\Delta T_{mod}/\Delta$ % power. This factor would be different if T_{avg} is off the normal program. Or in other words, the ρ_{mod} contribution to defect would be different, changing ρ_{power}.

REFERENCE FOR 5.12 PART 2

Also to a smaller extent, only neutrons slowed within one slowing down length can be absorbed. Thermal diffusion length varies with several things, among them the concentration of nearby fuel atoms and the presence of the rod itself. The most important factor in thermal diffusion length, however, and the only one with an easily measured effect, is reactor coolant temperature. Increasing coolant temperature decreases moderator density, and increases thermal diffusion length. The rod will, therefore, have a higher worth at high temperature than at low temperature.

It seems reasonable that, since increasing boron concentration reduces the thermal diffusion length, boron concentration should affect rod worth. However, experiments have failed to verify a significant relationship between boron concentration and rod worth. Void fraction will affect L. The more voids in the moderator, L is longer, worth is greater. Fission product accumulation will also decrease worth because of competetion.

The most important variable in rod worth is the local-to-average flux ratio. Since rod worth is proportional to the square of this ratio, rod worth is seen to be particularly sensitive to the local flux. Average flux is a function of power level, so that the average flux is constant for a constant power level.

In an idealized reactor with all control rods withdrawn, the axial flux distribution assumes a cosine shape as shown in RXTH-TP-35.3. Thus the rod inserts the most reactivity when moving near the middle of their travel and least near the top and bottom of the core.

The worth of the control rod is also affected by its radial position in the core. Flux is peaked more toward the center, and lower toward the edges. Central control rods will have a higher worth than those near the edges.

Since control rod worth is proportional to $\phi_{local})^2$, the presence of other control rods, by decreasing ϕ_{local}, can decrease the worth of an individual rod. For example, with all rods inserted, the reactor is shutdown, and ϕ_{avg} is very small. If the central rod is fully withdrawn, the flux in the area of the withdrawn rod increases substantially, and core multiplication increases. Because this rod causes the value of $\phi_{local}/\phi_{avg})^2$ to be large, its worth for this condition is quite high. For a total rod worth (45 rods) of 7% Δ K/K, this central rod may have a worth of as much as 1.1% Δ K/K.

REFERENCE FOR 5.13

FACTORS AFFECTING ROD WORTH

$$\text{ROD WORTH} \propto K \cdot A \cdot L \cdot \left(\frac{\phi_{\text{Local}}}{\phi_{\text{Avg}}}\right)^2$$

K = Constant Involving Materials In Core, Moderator, And Control Rods

A = Area Of Control Rod Exposed to Flux

L = Thermal Diffusion Length and Slowing Down Length

ϕ = Thermal Neutron Flux
Local: At The Position Of Interest
Avg: Core Average

b) $\dot{Q} = \dot{m}(h_{out} - h_{in})$

$= (1\times10^6\ lb_m/hr)(1194-400)\ BTU/lb_m$

$= 7.9\times10^8\ BTU/hr$

b. $\dot{Q} = \dot{m}c_p\,\Delta T$

1) Used to detemine heat out of primary coolant

2) Cannot be used for phase change

3) Primary coolant with a flow rate of $7\times10^7\ lb_m/hr$ enters the steam generator at a temperature of 560°F and leaves at 551°F. What is the rate of heat transfer through the steam generator. The c_p for water is 1.28 BTU/lb_m-°F

o $\dot{Q} = \dot{m}c_p\,(T_{in} - T_{out})$
$= (7\times10^7)(1.28)(560-551)$
$= 8\times10^8\ BTU/hr$

c. $\dot{Q} = U_oA_o\,\Delta T_{lm}$

1) Can be used to calculate heat transfer through tubes

OUTLINE | KEY AIDS

$$h = \frac{H}{m}$$

REFERENCE FOR 5,15

$$\Delta h = \frac{\Delta H}{m} = \frac{1500\ \text{BTU}}{1000\ \text{lb}_m} = 1.5\ \text{BTU/lb}_m$$

(2) If the temperature of the water decreased 1°F, what is the specific heat capacity of water?

$$\Delta h = c_p\ \Delta T$$

$$c_p = \frac{\Delta h}{\Delta T}$$

$$c_p = \frac{1.5\ \text{BTU/lb}_m}{1°\text{F}}$$

$$c_p = 1.5\ \text{BTU/lb}_m\text{-°F}$$

III. SUMMARY

A. <u>OBJECTIVE 1:</u> From memory, correctly list the three characteristic types of energy (Mechanical, Thermal, and Electrical) and for each energy type the units as expressed in English system, mks, and cgs systems. — HT/FF-TP-4.1

B. <u>OBJECTIVE 2:</u> Using tables provided, work problems converting from one energy type to another with at least an 80% correct proficiency level. — HT/FF-TP-4.1

1. Energy equivalence proved by Joule

2. Joule's equivalence
   1 BTU = 777.9 ft-lb_f

5.0 GENERAL (Continued) REFERENCE FOR 5.18 PART 2

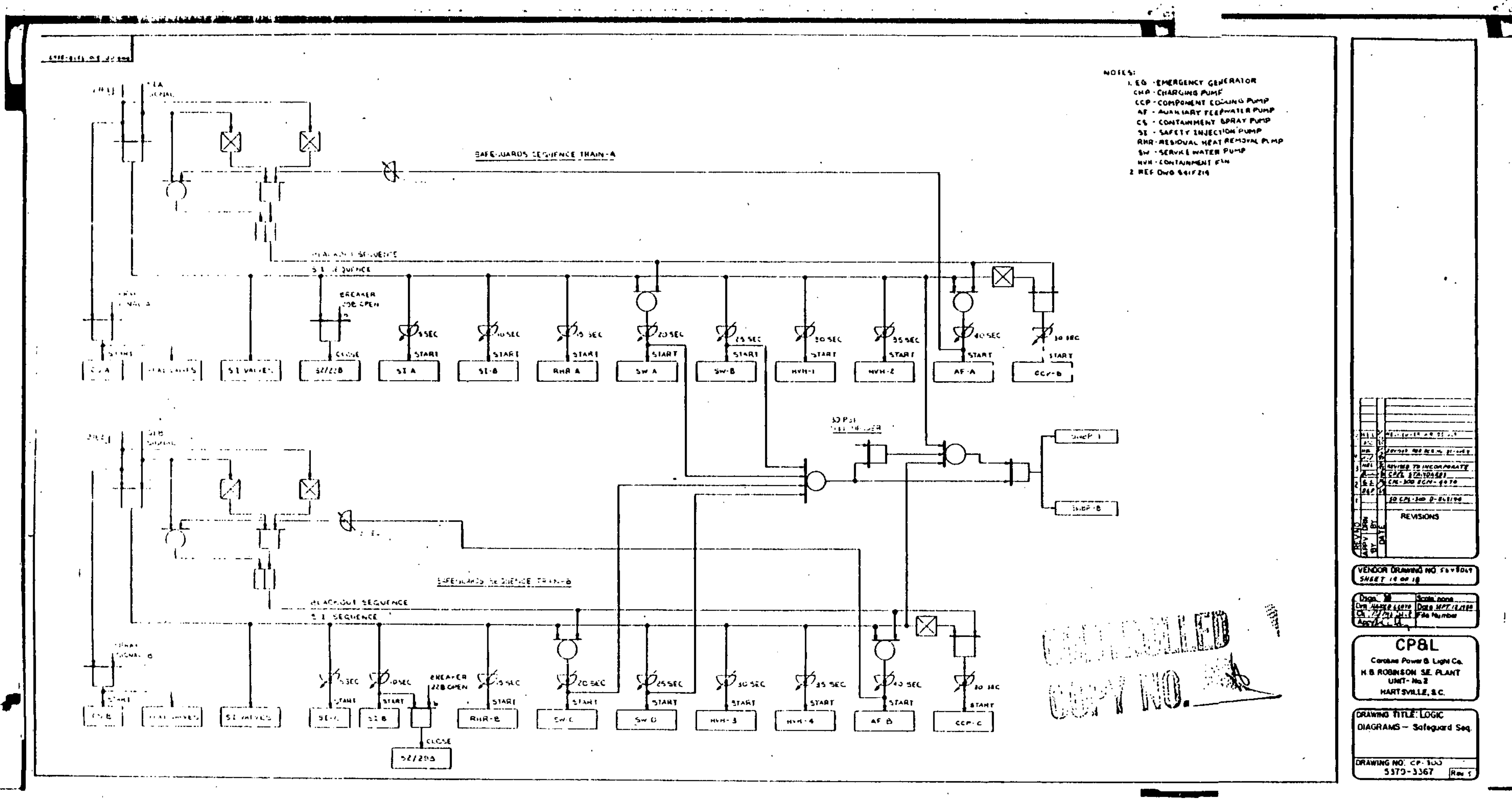
The most likely cause of a power tilt is a Dropped or Misaligned Control Rod. This causes an imbalance in the normal core symmetry and directly affects the power balance throughout the four quadrants. The most affected quadrant would be the one in which the misalignment occurs, with the quadrant diagonally across the core showing a significant effect of a lesser degree. This occurs because of the redistribution of power away from a flux perturbed region. Once the rod has been realigned, an oscillation may occur radially within the core. Normally it will oscillate diagonally across the core with a cycle of 26 hours. Little can be done to dampen a radial oscillation, but past experience has shown that large radial oscillations will diminish within 24 to 36 hours.

5.2 Monitoring System

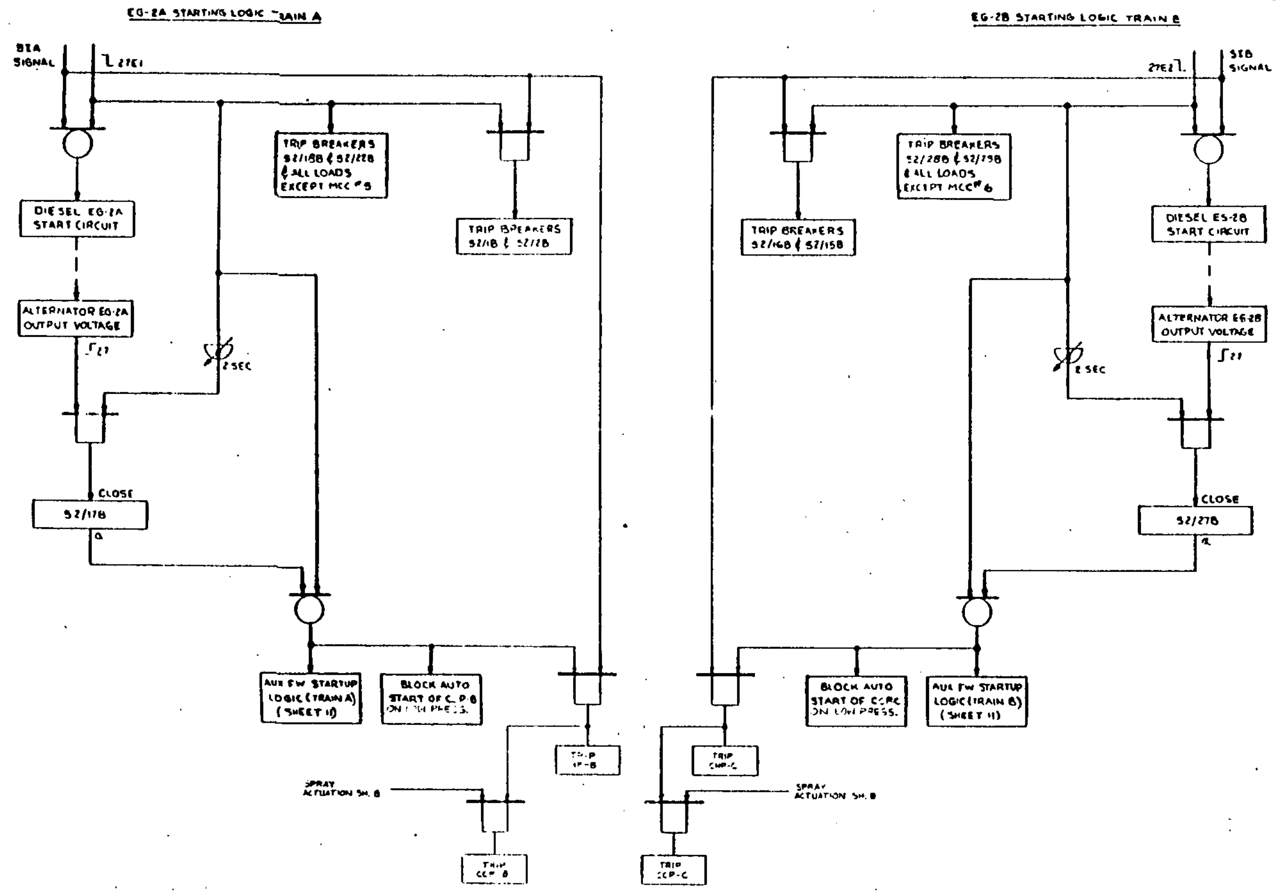
Monitoring of Quadrant Power Tilt is performed through the four Power Range Neutron Detectors located external to the core. Each detector is a long ion-chamber which provides signals related to power in the top and in the bottom of the core at a particular quadrant. The signals from all four quadrants are compared in order to warn the operator, if the power tilt ratio in either the top or the bottom of the core reaches 1.02 (2%). This condition results in either of two alarms to be actuated on the RTGB; P.R. UPPER DET HI FLUX DEV or AUTO DEFEAT, or P.R. LOWER DET HI FLUX DEV or AUTO DEFEAT. An additional alarm, P.R. CHANNEL DEVIATION, alerts the operator if any two of the four Power Range Channels disagree by as much as 2% of rated power.

During normal operation, manual calculations for tilt ratio using the excore detector raw currents are performed every two hours by the Control Operator. A value is determined for the top and the bottom halves of the core. Determining the tilt at this frequency will alert the Control Operator of slow moving trends which could lead to abnormal tilt conditions within the core.

REFERENCE FOR 6.02
PART 1
PART 3



REFERENCE FOR
6.02
PART II
PART 3



CONTROLLED
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NOTES:
1. EG - EMERGENCY GENERATOR
CHP = CHARGING PUMP
CCP - COMPONENT COOLING PUMP
AF - AUXILIARY FEEDWATER PUMP
CS - CONTAINMENT SPRAY PUMP
SI - SAFETY INJECTION PUMP
RHR - RESIDUAL HEAT REMOVAL PUMP
SW - SERVICE WATER PUMP
HVH - CONTAINMENT FAN
2. REF DWG. 5AIF 214.

REV NO	DRN BY	APPV BY	DATE	REVISIONS
6				DCR 95-1031
5				DCR 95-310
4				REVISED PER DCR No. 81-4465
3				REVISED TO INCORPORATE CP&L STANDARDS
2				CR-300 ECN-4410
1				30 CPL-300

VENDOR DRAWING NO. 5649D69
SHEET 18 OF 18

Dsgn.	Scale none
Drn. HAROLD LLOYD	Date SEPT. 19, 19??
Ck.	File Number:
Appr.	

CP&L
Carolina Power & Light Co.
H.B. ROBINSON S.E. PLANT
UNIT- No. 2
HARTSVILLE, S.C.

DRAWING TITLE: LOGIC DIAGRAMS — Emerg. Gen. Start & Bus. Clearing

DRAWING NO. CP-300 5379-3368 Rev.

CAUTION TAG LOG

REFERENCE FOR 6.84

TAG SERIAL NO.	LOCATION	* R&LIS not accepted. Can be selected to troubleshoot ICCM Channel Annunciator REASON	CONTINUATION SHEET USED (Check if used)	NO. OF TAGS	SF INIT.	ATTACHED BY DATE / INITIALS	REMOVED BY DATE / INITIALS
685	Control Rm (R&LIS Cooling Monitor's)	Not to be used Cal. only *	N/A	2	[illegible]	5/29/87 [illegible]	6/1/87 [illegible]
686	S/D AFW PUMP SUCTION VALVE	STOP LOSS OF WATER FROM CST THROUGH MISC. DR COLL. TK.	N/A	1	[illegible]	6-1-87 [illegible]	6-9-87 [illegible]
687	SGB-19C	ISOLATE PUMP FROM B/D SYS. PUMP RUPTURED	N/A	1	[illegible]	6/5/87 [illegible]	9-8-87 [illegible]
688	(WEST S/DUMP N2) is isolated CAP ON STM DUMP POTENTIOMETER	TO ALLOW USE OF STM DUMPS WHILE BOP IS IN USE WEST S/DUMP N2 IS ISOLATED	NA	1	[illegible]	6/6/87 [illegible]	6-11-87 [illegible]
689	AFW-4	prevent overflow of misc collecting tank and wasting good water	✓	~~4~~ 2 6-7-87	[illegible]	6-7-87 [illegible]	6/13/87 [illegible]
690	Rx Trip Reset Pushbutton on RTGB	replace fuses in Rod Control	NA	1	[illegible]	6-8-87 [illegible]	6-8-87 [illegible]
~~691~~	~~V12-9~~	V12-8 NOT CLOSING ALL THE WAY. IAW AOP-23 TO INSURE INTEGRITY FOR H/U	void 6-8-87 N/A	1	Not	Used	
692	Quench valve's isol. valves (Turb. & Aux Bldg)	Quench valves leak by & fill up heaters	✓	2	[illegible]	6-9-87 [illegible]	6/15/87 [illegible]
693	GS-36	THROTTLE AS NECESSARY TO MAINTAIN GLAND STEAM press below 18 PSIA	N/A	1	[illegible]	6-10-87 [illegible]	6/19/87 [illegible]
694	MS-161	THIS VALVE CLOSED DUE TO TRAP BLOWING BY	N/A	1	[illegible]	6-10-87 [illegible]	6/13/87 [illegible]

*7. Review all questions and answer keys to ensure there is no double jeopardy.

8. Review the answer key to ensure all questions are answered concisely and clearly. Each question should have numerical values assigned for partial credit; that is, when the question elicits a complex multifaceted response, a scheme should be enumerated for scoring each of these facets. For example, a single question worth 3 points of a 25-point category might have as many as 10 facets, each of which should be assigned a value.

9. Verify that there is a reference to the plant training material for each answer, if available.

10. Review questions and answers to ensure they correspond to the required level of knowledge (i.e., RO or SRO level) as described in Standard ES-202.

11. Ensure that "lone questions" of a section are flagged on a previous page by a "continued on next page" statement.

12. Ensure that each category is concluded with the statement "End of Category ______".

13. Verify proper distribution of topics within a category. For example, category 2 should include a variety of questions on major, auxiliary, engineered safety systems and electrical systems.

D. Documentation

When the review is completed, the "Written License Examination Quality Assurance Checkoff Sheet," attachment 1 of this standard, should be approved by regional section chief and filed with the record copy of the examination.

*See Standard ES-202, p. 4 of 6.

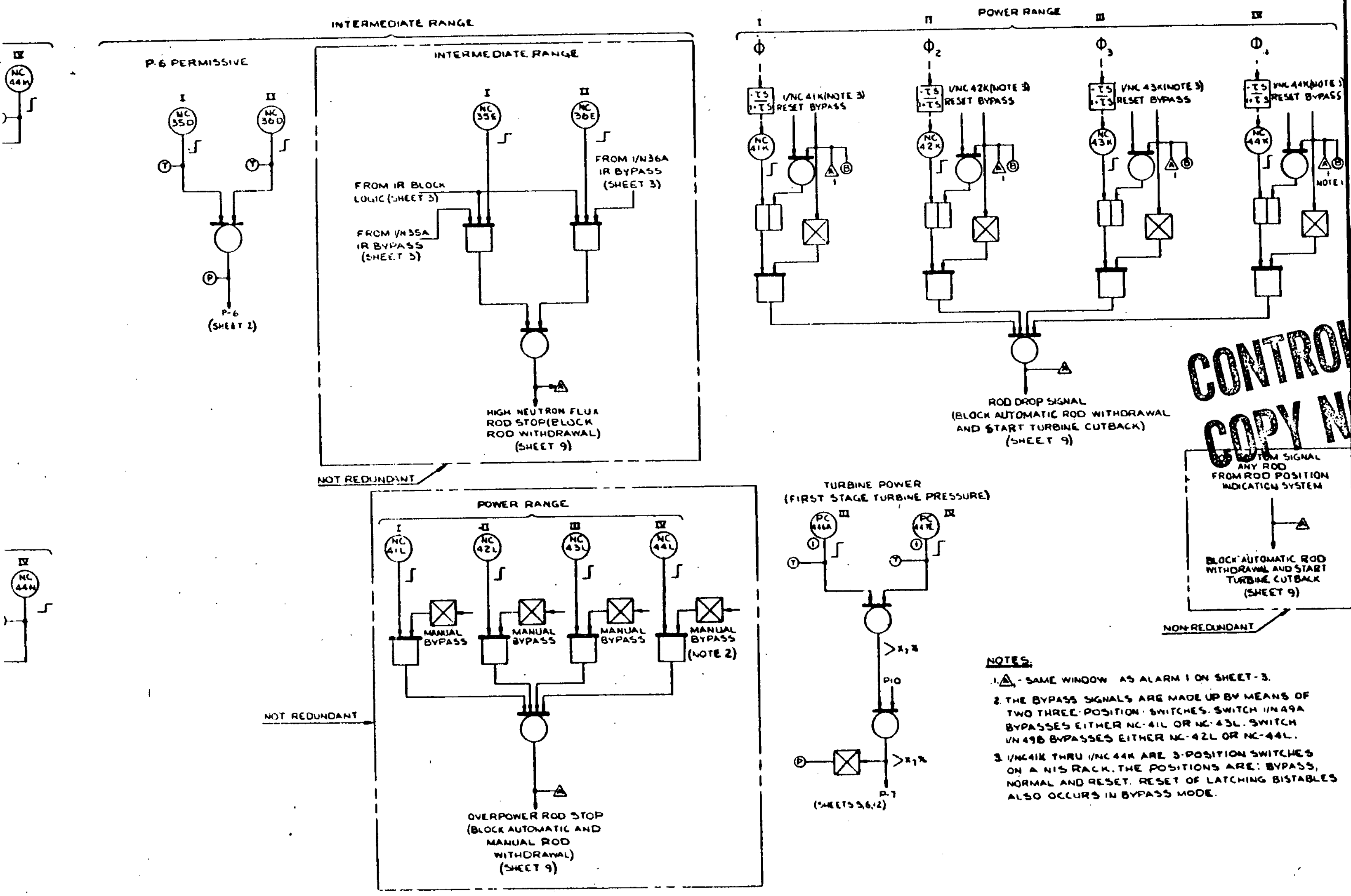
2.0 **COMPONENT DESCRIPTION** (Continued)

REFERENCE FM 6.06 PART 1 & 2

The dropped-rod sensor assembly is an operational amplifier unit which incorporates an adjustable lag network at one input and a nondelayed signal on the other. The unit compares the actual power signal with the delayed power signal received through the lag network and amplifiers the difference. This amplified differential signal is delivered to a bistable relay driver unit which trips when the level of this signal exceeds a preset amount. Tripping of this unit indicates a power level change over the lag period (5 seconds) of 5% power. This bistable unit is a latching type, ensuring that the necessary action will be initiated and carried to completion. Specifically, the unit controls relays which in two of four logic matrics, provide a rod-stop and turbine runback signal, a control board annunciation signal, and a computer input signal. A reset switch on the associated Power Range drawer must be operated manually to reset the rod stop and turbine runback functions. The bistable units which sense the power level signal as derived by the linear amplifier are nonlatching and perform the following functions:

- overpower rod-stop (blocks automatic and manual rod withdrawal);
- permissive functions (provisions for three are incorporated in the design, but are not required on all plants;
- low-range reactor trip;
- high-range reactor trip.

REFERENCE FOR 606 PART 1 + 2



CONTROLLED COPY NO 18

NOTES:

1. [A] - SAME WINDOW AS ALARM 1 ON SHEET-3.
2. THE BYPASS SIGNALS ARE MADE UP BY MEANS OF TWO THREE-POSITION SWITCHES. SWITCH I/N49A BYPASSES EITHER NC-41L OR NC-43L. SWITCH I/N49B BYPASSES EITHER NC-42L OR NC-44L.
3. I/NC41K THRU I/NC44K ARE 3-POSITION SWITCHES ON A NIS RACK. THE POSITIONS ARE: BYPASS, NORMAL AND RESET. RESET OF LATCHING BISTABLES ALSO OCCURS IN BYPASS MODE.

REV NO	DATE	BY	APPV	DRN	REVISIONS
7					REVISED PER ECN SL-6609
6					REVISED TO INCORPORATE CP&L STANDARDS
5					CP-300 ECN-4599
4					CP-300 ECN-4532
3					S.A. CP-300 ECN-300
2					S.A. CP-300 D-[illegible]
1					

VENDOR DRAWING NO. 5648D69 SHEET 4 OF 18

Dsgn: | Scale: none
Drn: HAROLD LLOYD | Date: SEPT 30/80
Chk: | File Number:
Appv:

CP&L
Carolina Power & Light Co.
H. B. ROBINSON S.E. PLANT UNIT- No. 2
HARTSVILLE, S.C.

DRAWING TITLE: LOGIC DIAG.
Nuclear Instr. Permissives & Blocks

DRAWING NO: CP-300-5379-2755 Rev 6

i. LOAD CHANNEL is lit when the impulse pressure transducer is greater than 5.5V or less than .5V, or the reference counter output is greater than the combined signals of impulse pressure and load comparison monitor setpoint by +20%. The system transfers to IMP OUT.

j. OVERSPEED PROTECTION CONTROL is lit when the kilowatt transducer is greater than 10 volts or less than .2 volts.

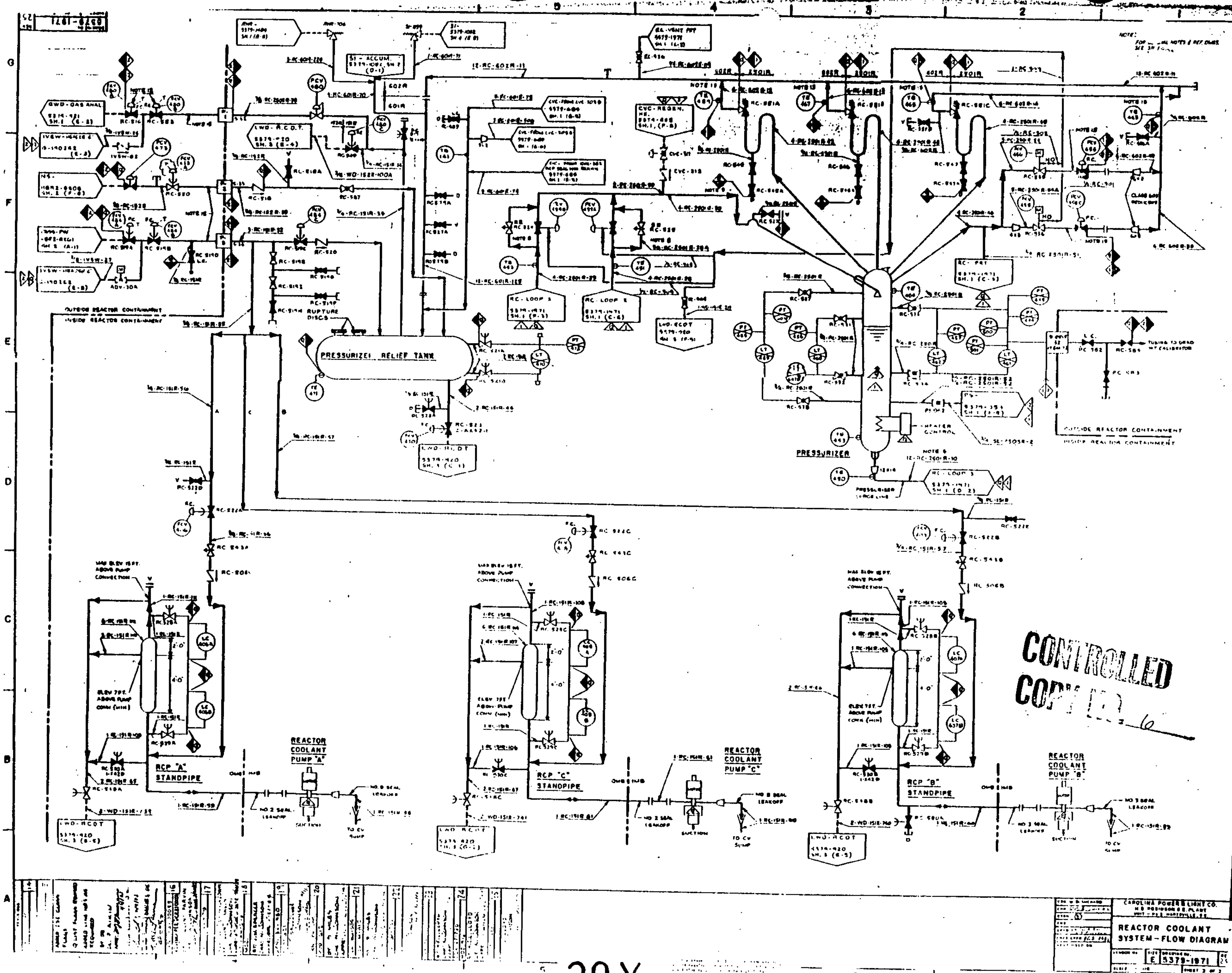
k. EMERGENCY POWER SUPPLY is lit when any one of the following power supply output voltages is lost: +15V "A", +15V "B", -15V "A", -15V "B", +48V "A", and +48V "B".

l. GOVERNOR VALVE TRACKING METER zero indicates that manual governor valve control (MGVC) and automatic governors valve control (AGVC) are tracking and that both signals are the same. During normal steady state conditions, this meter should always indicate zero. The meter will deflect from zero during Transients, such as a runback, but it should only be momentary. If the indicator moves to either the + or - side and remains, it means that one signal is not tracking the other and (if in AUTO) it should transfer to MANUAL. The load would increase or decrease, depending on the direction and magnitude the tracking meter is displaced from zero.

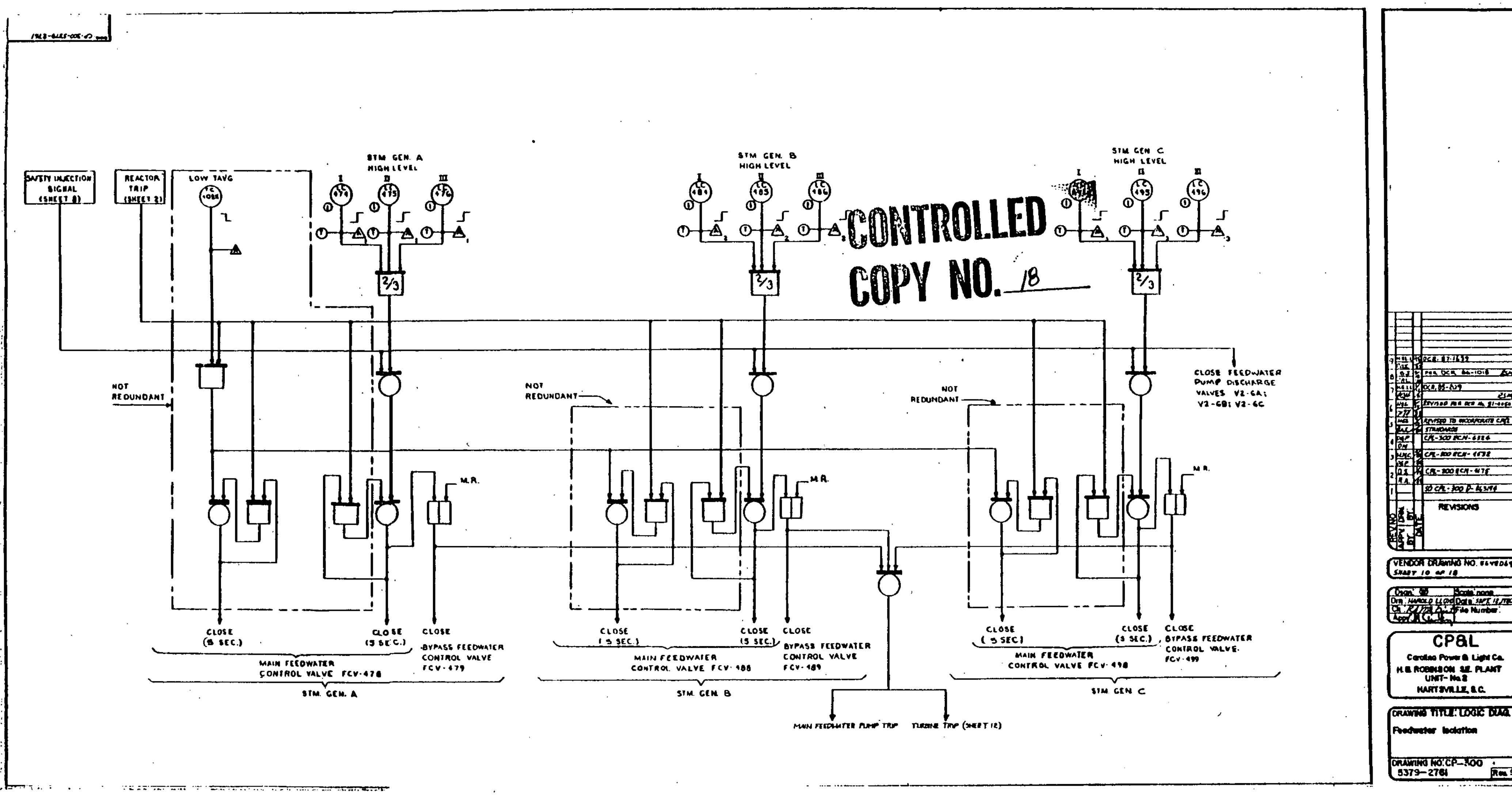
m. GOVERNOR VALVE POSITION indicates the position of the four (4) governor valves.

n. VALVE POSITION LIMIT indicates when the electronic valve position Limiter is set. When the governor valves open to the setting of the limiter, the valve position limit light will energize and the governor valves will not be permitted to open further electronically.

REFERENCE FOR 6.12 ... 1



REFERENCE FOR 6.13 PART I



3.1.3 Minimum Conditions for Criticality

REFERENCE FAN 7.01

3.1.3.1 Except during low power physics tests, the reactor shall not be made critical at any temperature, above which the moderator temperature coefficient is greater than:

a) +5.0 pcm/°F less than 50% of rated power, or

b) +5.0 pcm/°F at 50% of rated power and linearly decreasing to 0 pcm/°F at rated power.

3.1.3.2 In no case shall the reactor be made critical above and to the left of the criticality limit shown on Figure 3.1-1a or 3.1-1b (as appropriate per 3.1.2.1).

3.1.3.3 When the reactor coolant temperature is in a range where the moderator temperature coefficient is greater than as specified in 3.1.3.1 above, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization.

3.1.3.4 The reactor shall be maintained subcritical by at least 1% until normal water level is established in the pressurizer.

Basis

During the early part of fuel cycle, the moderator temperature coefficient may be slightly positive at low power levels. The moderator coefficient at low temperatures or powers will be most positive at the beginning of the fuel cycle, when the boron concentration in the coolant is the greatest. At all times, the moderator coefficient is calculated to be negative in the high power operating range, and after a very brief period of power operation, the coefficient will be negative in all circumstances due to the reduced boron concentration as Xenon and fission products build into the core. The requirement that the reactor is not to be made critical when the moderator coefficient is more positive than as specified in 3.1.3.1 above has been imposed to prevent any unexpected power excursion during normal operations as a result of either an increase of moderator temperature or decrease of coolant

REFERENCE FOR 7.04

3.10.8.2 When the reactor is in the cold shutdown condition, the shutdown margin shall be at least 1 percent $\Delta k/k$.

3.10.8.3 When the reactor is in the refueling operation mode, the shutdown margin shall be at least 10 percent $\Delta k/k$.

Basis

The reactivity control concept is that reactivity changes accompanying changes in reactor power are compensated by control rod motion. Reactivity changes associated with xenon, samarium, fuel depletion, and large changes in reactor coolant temperature (operating temperature to cold shutdown) are compensated by changes in the soluble boron concentration. During power operation, the shutdown groups are fully withdrawn and control of reactor power is by the control groups. A reactor trip occurring during power operation will put the reactor into the hot shutdown condition.

The control rod insertion limits provide for achieving hot shutdown by reactor trip at any time assuming the highest worth control rod remains fully withdrawn with sufficient margins to meet the assumptions used in the accident analysis. In addition, they provide a limit on the maximum inserted rod worth in the unlikely event of hypothetical rod ejection and provide for acceptable nuclear peaking factors. The solid lines shown in Figure 3.10-1 meet the shutdown requirement for the first 50 percent of the cycle. The end-of-cycle life limit is represented by the dotted lines. The end-of-cycle life limit may be determined on the basis of plant startup and operating data to provide a more realistic limit which will allow for more flexibility in plant operation and still assure compliance with the shutdown requirement. The maximum shutdown margin requirement occurs at end of core life and is based on the value used in analysis of the hypothetical steam break accident. Early in core life, less shutdown margin is required, and Figure 3.10-2 shows the shutdown margin required at end of life with respect to an uncontrolled cooldown. All other accident analyses are based on 1 percent reactivity

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Reference for 7.10
Part 3

1.0 <u>SYMPTOMS</u>

1.1 Smoke or toxic gas in the control room.

1.2 Control room area radiation alarm.

1.3 Bomb threat or other acts of sabotage.

2.0 <u>AUTOMATIC ACTIONS</u>

2.1 None Applicable

3.0 <u>OPERATOR ACTIONS</u>

3.1 <u>Immediate Actions</u>

3.1.1 All available operators should report to the Fire Equipment Building and await direction from the Shift Foreman.

****

CAUTION

IF AT ANY TIME DURING THE PERFORMANCE OF THIS PROCEDURE CONTROL OF THE PLANT CANNOT BE MAINTAINED DUE TO FIRE DAMAGE, THEN GO TO DSP-002.

****

3.2 <u>Subsequent Actions</u>

NOTE

Keys are maintained in a sealed cabinet to the right of doorway in the Fire Equipment Building. Copies of this procedure are located at the Fire Equipment Building, Secondary Control Panel and the Charging Pump Room Control Panel.

3.2.1 STATION Operators at local control stations as directed by the Shift Foreman <u>AND</u> OBTAIN two-way portable radios and keys for Secondary Control Panel.

1. Charging Pump Room.
2. Secondary Control Panel Turbine Building Mezzanine.

5.0 **PROCEDURE** (Continued)

Reference for 7.16 Part 1

Brackets are used to indicate values that should be used if adverse containment conditions are present. Adverse containment conditions exist when containment pressure is greater than 4 psig.

Example: Verify RCS Subcooling - GREATER THAN 25°F [45°F]

5.1.2 Foldouts

Several EOPs require the use of a FOLDOUT. The applicable FOLDOUT is intended to be available (visible) when the EOP is being performed. The FOLDOUT page contains several pieces of information or actions which are applicable at any step in the EOP being used. The most important of these actions are procedure transitions which allow immediate response to new symptoms as they appear. The placement of these transitions on the FOLDOUT allows prompt response to the appearance of subsequent symptoms. The contents of each FOLDOUT specify actions which must be taken when the symptoms associated with that action are recognized.

5.1.3 Supplements

The EPP Supplements are used to provide information too lengthy to include on a PATH, EPP or FRP. Specific uses are described in the following paragraphs.

In PATH 1, the Operator is expected to be able to verify proper SI, Phase A, Containment Ventilation Isolation, etc., valve alignments from the Safeguards Status Panel (pink & blue lights). If he is unable to verify proper alignment from this panel, he can then refer to Supplement "A". Supplement "A" contains a list of all valves and their required position which actuate on a Safety Injection actuation. Using Supplement "A" and other available indications the Operator can verify proper valve alignment independent from the Safeguards Status Panel.

FOR INFORMATION ONLY

1. Foldout A

a. RCP TRIP CRITERIA

REFERENCE FOR 7.16 PART 1

Stop all RCPs if BOTH conditions listed below are met:

1) SI pumps - AT LEAST ONE RUNNING

2) RCS Subcooling - LESS THAN 25°F [45°F]

b. SI ACTUATION CRITERIA

Actuate SI and go to PATH-1, Entry Point A if EITHER condition listed below occurs:

1) RCS Subcooling - LESS THAN 25°F [45°F]

2) PZR Level - CANNOT BE MAINTAINED GREATER THAN 10% [30%]

c. RED PATH SUMMARY

1) SUBCRITICALITY - Nuclear power greater than 5%

2) CORE COOLING - Core Exit T/Cs greater than 1100°F

3) HEAT SINK - Level in all S/Gs less than 25% AND total feedwater flow less than 300 GPM OR 0.2×10^6 PPH

4) INTEGRITY - Cold leg temperature decrease greater than 100°F in last 60 minutes AND RCS cold leg temperature less than 310°F

5) CONTAINMENT - Containment pressure greater than 42 PSIG

d. AFW SUPPLY SWITCHOVER CRITERION

Switch to alternate AFW water supply if CST level decreases to less than 10%.

 REFERENCE FOR 7.16 PART I

5. Open Fold-Out "A"

5. Fold outs contain a list of important items that must be continuously monitored so that specified actions are taken as required.

FOLD OUT "A"

a. Trip all RCP's if both conditions are met
   1) At least one SI pump running
   2) RCS subcooling less than 25°F

a. During a small break LOCA on a cold leg loop continued operation of the RCPs will result in a more rapid depletion of RCS inventory. If the RCPs are tripped after saturation conditions are reached then it is possible for the reactor coolant to settle out at a point beneath the level of the top of the core. Therefore saturation plus 25°F for instrument error is chosen. At least one SI pump running in order to insure make up to the RCS. If no SI pumps are running the RCPs should continue to run to provide core cooling until the RCS is depressurized to point of entry of the SI accumulators and RHR.

REFERENCE FOR 7.1.1

reactor trip should have expired such that any subsequent failure, beyond that causing the reactor trip, should not require RCP trip in order to ensure acceptable clad temperature even if it is a small LOCA of critical size. This is due to the reduction in decay heat generation with time and because the RCS cooldown will result in less time to SI accumulator injection for a small LOCA. Therefore, if an operator controlled RCS cooldown results in exceeding the RCP trip criteria, the RCPs should not be tripped.

If an RCP trip criteria step is missed by the operator and RCPs are not tripped when necessary, then appropriate contingency actions are provided in the ERGs.

A most probable best estimate analysis (see Reference 8) demonstrates that the RCPs can be tripped at any time after the RCP trip setpoint is reached with acceptable clad temperatures resulting for a spectrum of small LOCA sizes. Operator action times are not critical for this evaluation. Therefore, "missed RCP trip setpoint" steps are only necessary if an event is occurring with failures beyond those assumed in the most probable best estimate analysis. In general, if these failures occur and the RCPs are not tripped in a timely manner after the setpoint is reached, then a challenge to the Core Cooling Critical Safety Function may occur. Namely, significant core uncovery and clad heatup may occur if RCP trip is delayed and the RCPs are subsequently stopped.

For the reference plants, a vessel void fraction of 50%, as indicated by a RVLIS dynamic range setpoint, is used as the symptom of this Core Cooling challenge. If this setpoint is reached, the Function Restoration Guideline, FR-C.2, RESPONSE TO DEGRADED CORE COOLING, is implemented and directs the operator to <u>not</u> trip the RCPs in order to ensure continued core cooling from forced flow. Core cooling should be ensured as long as RCPs continue to run. Subsequent steps direct a controlled RCS cooldown using the steam generator to cause SI accumulator water injection and increase system inventory.

REFERENCE FOR 7.16 [illegible]

If the RCPs fail or are tripped after a 50% void fraction is reached, core uncovery and clad heatup may occur. If core exit thermocouple temperatures reach 700°F or greater and the RVLIS full range indication decreases to less than 3.5 feet in the core, then guideline FR-C.1, RESPONSE TO INADEQUATE CORE COOLING, is implemented and directs the operator to depressurize all steam generators at the maximum rate. If a further clad heatup occurs, exceeding 1200°F at the core exit, and SG depressurization is not effective, then RCPs are restarted one at a time to provide forced cooling flow.

In summary, the ERGs provide multiple levels of contingency actions that are symptom-based and function-related. In addition to the RCP trip parameter and setpoint, RVLIS and core exit thermocouples are used to direct operator action if a challenge to a Critical Safety Function is occurring. In this way the operator is provided with actions to maintain Critical Safety Functions that are dependent only on parameters available in the control room and that are independent of the specific event sequence. If an RCP trip criteria step is missed by the operator and conditions degrade to the point where core cooling may be challenged if RCPs are stopped, then the operator is provided with appropriate contingency actions.

For plants without the reference RVLIS design, the actions based on core exit thermocouple temperature are sufficient to ensure timely response to Critical Safety Function challenges since a challenge will still be identified and appropriate contingency actions provided.

5.0 PROCEDURE (Continued)

REFERENCE FOR 7.19

Directing a comprehensive reactor engineering program. This program includes reactor core monitoring for reactivity anomalies and reactor core performance, and the preparation of procedures utilized by operating personnel as related to core monitoring and performance. It also involves the appropriate reviews of information as required to follow core reactivity, identify anomalies (Technical Specification 4.9) in reactor characteristics and recommend corrective action, and initiate the required reports.

Accountability and record keeping of on-site Special Nuclear Material (SNM) as required by Volume 6, Part 5, Special Nuclear Material Accountability Plan.

5.4.4 Radiation Control Supervisor

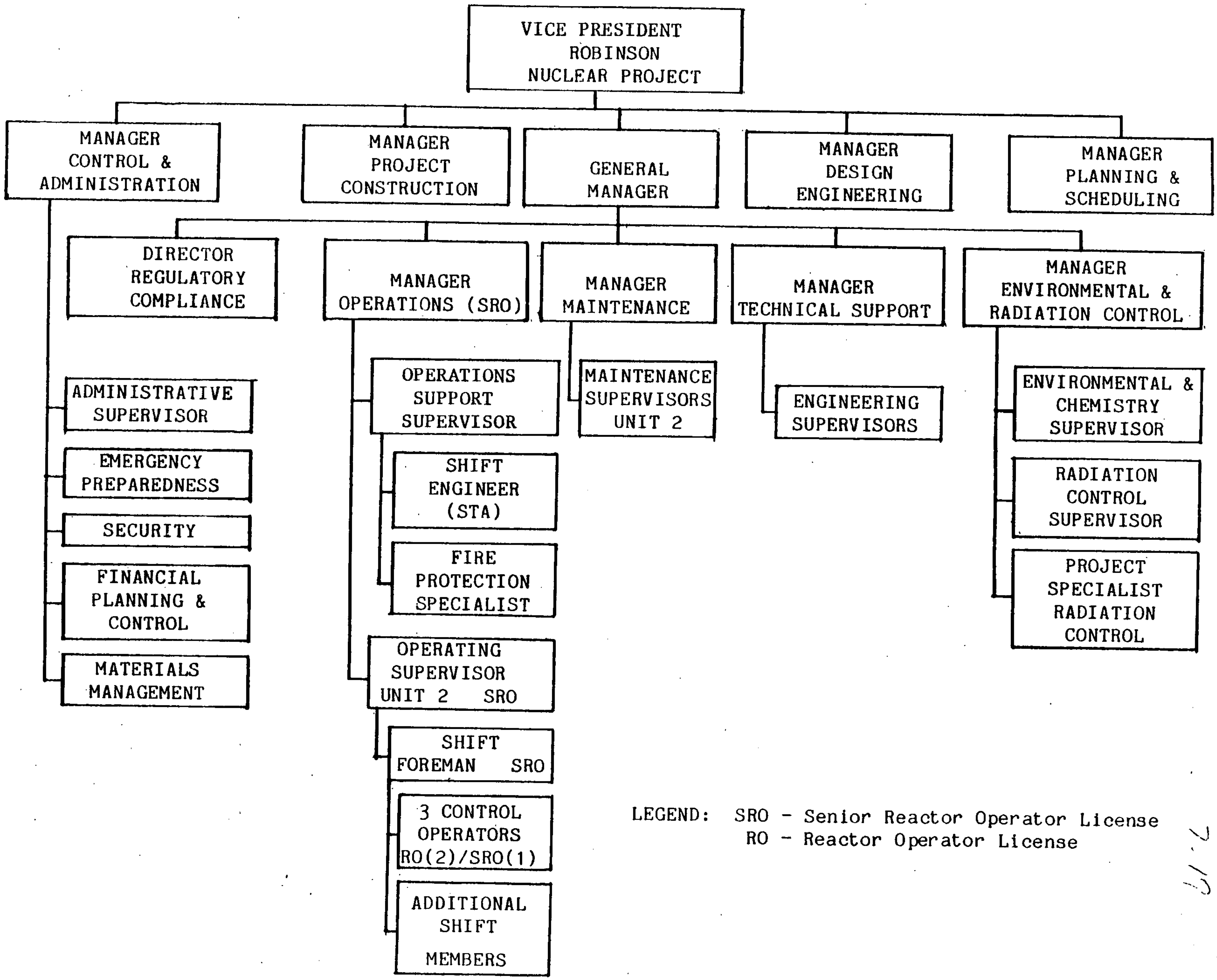
The Radiation Control Supervisor is responsible to the Manager - E&RC for the radiation control program. This work is performed by Radiation Control Technicians under the general supervision of the Radiation Control Foreman. The Radiation Control Supervisor's responsibilities include:

Directing the activities of the Radiation Control Subunit through the Radiation Control Foreman. This direction includes the instructions, checks, and reviews necessary to insure the implementation of Health Physics procedures and compliance with regulatory requirements by all personnel working at or visiting the project.

Ensuring that any plant operation or work in progress which does not comply with accepted Health Physics practices is stopped until corrective action is taken.

PROJECT ORGANIZATION

REFERENCE FOR 7.17



LEGEND: SRO - Senior Reactor Operator License
RO - Reactor Operator License

REFERENCE FOR
8.08 PART 5

PREPARATION AND PROCESSING OF AMMS WORK REQUESTS

- When satisfied that the work request is correct, approve the work request.
- A printout of the work request working copy may be obtained now, as desired.

5.0 OPERATIONS REVIEW (when necessary)

Shift Foreman or SRO signs on and approves the work request for work.

6.0 MAINTENANCE CRAFT PERSONNEL

- Obtain the Shift Foreman's or SRO's verbal permission prior to starting work which affects the operation of plant equipment.
- Maintenance on components, equipment, structures, or materials that do not affect plant operational systems or support systems involved in power production or safety (e.g., painting, housekeeping, insulation, repair of non-security doors, repair of lab equipment, vehicles, fabrications, shop equipment, cranes and gantries, portable survey equipment, security systems and equipment, lighting, etc.), do not require verbal permission from OPS.
- Obtain clearance (as necessary) and record clearance/LCTR number on the working copy.
- Job Details: Enter the following information on the working copy, as applicable, during performance of the work. (Information documented by the maintenance procedure or on affected data sheets, stores availability check forms, or other WR attachments shall remain with the WR and PM Route Workorder, and thus need not be duplicated on the WR or PM Route Workorder.)
  - As found condition.
  - As left condition.

REFERENCE FOR 8.14

RNPD EMERGENCY RESPONSE ORGANIZATION DIRECTORY

TECHNICAL SUPPORT CENTER (TSC)

ERO POSITION	ERO/ROLM	HOME	OFFICE
TSC - SITE EMERGENCY COORDINATOR	4100		
Primary - Dick Morgan		383-5947	1201
Alternate - Joe Curley		332-1517	1367
Alternate - Supervisor "On Call"		(Pager) 665-3895	
Alternate - Manager "On Call"		(Pager) 667-5847	
TSC - PLANT OPERATIONS DIRECTOR	4099		
Primary - Fred Lowery		332-0050	1204
Alternate - Duane Nelson		*383-4127	1300
Alternate - John Benjamin		383-4385	1298
TSC - RADIOLOGICAL CONTROL DIRECTOR	4096		
Primary - Dick Smith		665-0060	1205
Alternate - Stan Crocker		332-6068	1223
Alternate - Andy Eaddy		332-8054	1232
Alternate - James Harrison		*332-4652	1433
Alternate - Pat Harding		332-6420	1247
TSC - EMERGENCY REPAIR DIRECTOR	4098		
Primary - Don Quick		383-2822	1203
Alternate - Bill Gainey		332-7897	1271
Alternate - Rich Barnett		383-5953	1289
TSC - LOGISTICS SUPPORT DIRECTOR	4097		
Primary - Steve Zimmerman		332-0554	1206
Alternate - Danny Ingram		383-5365	1344

*UNLISTED NUMBERS

REFERENCE FOR

5.0 PROCEDURE (Continued) REFERENCE FOR 8.14

5.20 Jumper and Wire Removal

On occasion a condition may arise which requires installation or removal of electrical jumpers, wires, blind flanges or spool pieces which are not part of the system design. Usually the need is temporary in nature and may be required due to a malfunction, ground, need to simulate a condition, testing, or to provide a temporary power supply. For more detail refer to OMM-012, Jumper and Wire Removal.

5.21 On-Call Managers, Supervisors, and Backshift Inspection

As provided by Plant Technical Specification 6.1.1, the Plant General Manager shall delegate in writing the succession of responsibility for overall facility operation during his absence. Accordingly, periodically the Plant General Manager will issue a Manager and Supervisor On-Call Roster. This roster will designate the individuals who, on a rotating basis, will serve as "Acting General Manager" in the General Manager's absence. This would normally fall to the On-Call Manager; however, in the event he is unable to be contacted, the On-Call Supervisor will act in this capacity.

The On-Call Manager will normally rotate through the following positions; General Manager, Manager - Unit 2 Operations, Manager - Unit 2 Maintenance, Manager - Technical Support, Manager - Environmental & Radiation Control, and Director - Regulatory Compliance. The On-Call Manager should be trained and qualified to act in the capacity of the Site Emergency Coordinator during implementation of the Site Emergency Plan. This qualification is not mandatory providing the On-Call Supervisor at the time meets this qualification.

8.14

REFERENCE FOR 8.14

5.0 PROCEDURE (Continued)

The On-Call Supervisor will normally rotate through the following positions; Unit 2 Operating Supervisor, Unit 2 Mechanical Maintenance Supervisor, Unit 2 I&C Maintenance Supervisor, Radiation Control Supervisor, Environmental & Chemistry Supervisor, Engineering Supervisor - Plant Systems, Engineering Supervisor - Plant Performance, Unit 2 Operations Support Supervisor, Principal Specialist - Unit 2 Maintenance, or other supervisory level position as designated by the Plant General Manager. The On-Call Supervisor will be trained and qualified to act in the capacity of the Site Emergency Coordinator during implementation of the Site Emergency Plan.

Should a change in the rotation on the roster be required due to individual scheduling conflicts, these changes may be made without prior written approval providing appropriate personnel (Unit 2 Shift Foreman as minimum) are notified.

Each week the individual designated as On-Call Supervisor shall make a backshift check of the Plant. A specific check of the Plant housekeeping status shall be included and documented on ATTACHMENT 6.1 and forwarded to the General Manager. The On-Call Manager may accompany the On-Call Supervisor during this inspection or conduct other inspections as he or the General Manager directs. A special file is maintained in 14510 of the Plant Filing Index for the Weekend and Backshift Operations Assessment Form. Corrective action(s) will be initiated by the General Manager or his designee.

6.0 ATTACHMENTS

6.1 Weekend and Backshift Operations Assessment Form

REFERENCE FOR 8, 16 PART 2

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.

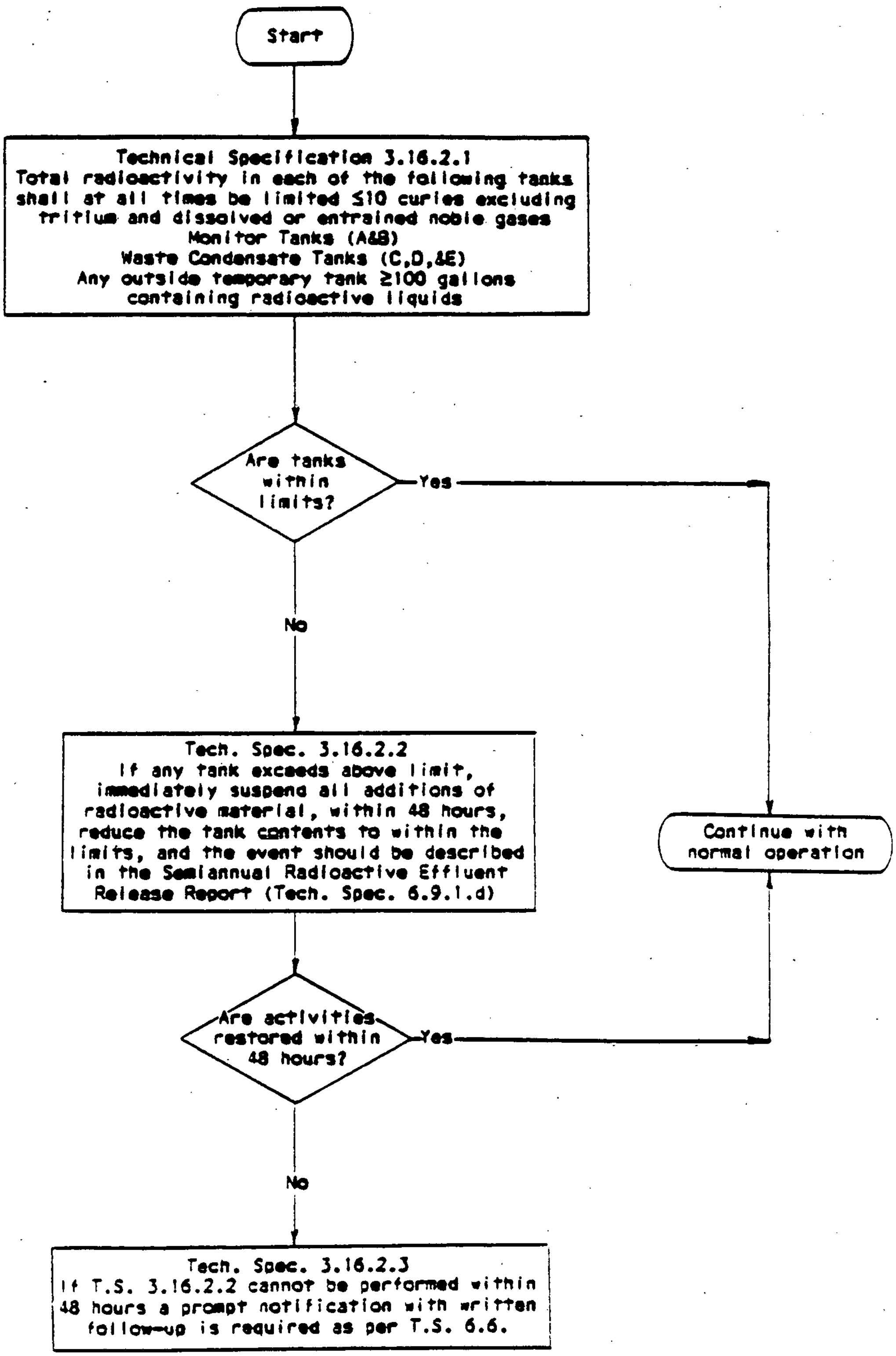
B. If sufficient time does not exist, the Shift Foreman shall approve of the deviation prior to performing the protective action.

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.

REFERENCE FOR 8.17 PART 2

.6 Compliance With Technical Specifications 3.16.2 (Tank Limits)



REFERENCE FOR 8,17 PART 2

Objective

To define the operating requirements for the liquid holdup tanks.

Specification

3.16.2.1 The quantity of radioactive material contained in each of the following tanks shall at all times be limited to $\leq$10 curies, excluding tritium and dissolved or entrained noble gases.

a. A monitor tank
b. B monitor tank
c. C Waste Condensate tank
d. D Waste Condensate tank
e. E Waste Condensate tank
f. Any Outside temporary tank*

3.16.2.2 With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and the event should be described in the Semiannual Radioactive Effluent Release Report, Specification 6.9.1.d.

3.16.2.3 If Specification 3.16.2.2 is not completed within 48 hours a notification must be made to the Commission in accordance with Specification 6.6.

3.16.3 Gaseous Radwaste and Ventilation Exhaust Treatment Systems

Applicability

Applies to the gaseous radwaste and ventilation exhaust treatment systems.

* A temporary tank is defined as any tank having a capacity of $\geq$ 100 gallons used for the receipt or transfer of radioactive liquids.

MINIMUM EQUIPMENT LIST
AUXILIARY OPERATOR
MISCELLANEOUS CHECKS

REFERENCE FOR 8.17 PART 2

INITIALS	INDICATOR	TECH. SPEC.
	LI-182 Monitor Tank "A" - LI-182*	Table 3.5-6 Item 4.b Table 4.19-1
	LI-183 Monitor Tank "B" - LI-183*	
	Waste Condensate Tank "C" LI - LI-603*	Table 3.5-6 Item 4.c Table 4.19-1
	Waste Condensate Tank "D" LI - LI-604*	
	Waste Condensate Tank "E" LI - LI-605*	
	S/G "A" Blowdown Barton - DPI-1328A	Table 3.5-6 Item 2.b Table 4.19-1
	S/G "B" Blowdown Barton - DPI-1328B	
	S/G "C" Blowdown Barton - DPI-1328C	
	Temporary Tank LI**	Table 3.5-6 Item 4.d 3.16.2
	R-19 Flow from S/G "A" - FIS-4360A	Table 3.5-6 Item 2.b Table 4.19-1
	R-19 Flow from S/G "B" - FIS-4360B	Table 3.5-6 Item 2.b Table 4.19-1
	R-19 Flow from S/G "C" - FIS-4360C	Table 3.5-6 Item 2.b Table 4.19-1

READING	INDICATOR	TECH. SPEC.
%	H_2 Concentration (GDT in Service)***	Table 3.5-7 Item 2 3.16.4 Table 4.19-2
%	O_2 Concentration (GDT in Service)***	

*Check during liquid additions to the tank per Tech. Spec. Table 4.19-1.

**A temporary tank is defined as any outdoor tank having a capacity of ≥100 gallons used for the receipt or transfer of radioactive liquids. May be marked N/A when not applicable.

***Implement requirements of Technical Specification Table 3.5-7, Item 2, if Gas Analyzer is not in service.

REFERENCE FOR 8.18

5.3.6 Manager - Control & Administration

The Manager - Control & Administration is responsible to provide for the administrative needs of the project.

Through administrative direction, the Manager - Control & Administration manages the functions of the Administration, Emergency Preparedness, Security, Financial Planning and Materials Control Units.

The Manager - Control and Administration's responsibilities include; maintaining a nuclear records management program to meet the requirements of the NRC and license regulations; managing the procurement and materials management requirements for the site; provide applicable training programs for the site, coordinating with the Operations Training and Technical Services Department; manage staff support to provide financial planning and cost control functions for the project; develop and maintain long-range plans and goals to provide manpower requirements for the project; manage office services and data processing functions to support project needs; managing the site's Emergency Preparedness Program in accordance with corporate requirements, license regulations, and technical specifications; directing the site's security program in concert with legal, plant Technical Specifications, and regulatory standards.

The Manager - Control and Administration will exercise general direction and guidance to Unit Heads in the selection of personnel for positions on the Project Staff.

7.12

5.0 PROCEDURE (Continued)

REFERENCE FOR 8.18

When relieved of Site Emergency Coordinator responsibilities, the E&C Supervisor will perform the duties of the Environmental Monitoring Team Leader. The Environmental Monitoring Team Leader is responsible to the Radiological Control Director for providing technical and administrative direction to the Environmental Monitoring Teams.

At the time of assignment, the E&C Supervisor shall have a minimum of five years experience in chemistry, of which one year shall be in radiochemistry. A maximum of four years of this five years experience may be fulfilled by related technical or academic training.

5.4.6 Administrative Supervisor

The Administrative Supervisor is responsible to the Manager - Control and Administration for the administrative services required for effective functioning of plant activities. These services are performed by the Administrative Supervisor and the Administrative Group under the supervision of the Administrative Supervisor. The Administrative Supervisor is responsible for:

Directing the activities of the Office Services Subunit.

Directing the Records Management, Document Control and Micrographics functions.

Maintaining custody of plant records as required by Company policies, state and federal regulations, and good office management. This general custody assignment excludes certain records and files specifically assigned to other supervisory personnel.

6.0 ADMINISTRATIVE CONTROLS

REFERENCE FOR 8.18

6.1 RESPONSIBILITY

6.1.1 The General Manager - Robinson Plant shall be responsible for overall facility operation except as described in Section 6.1.2 and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Manager - Control and Administration shall be responsible for facility physical security, emergency planning, and document control and shall delegate in writing the succession to this responsibility during his absence.

6.1.3 Disagreements between the General Manager - Robinson Plant and the Manager - Control and Administration in the areas of physical security, emergency planning, and document control will be resolved by the Vice President - Robinson Nuclear Project.

ENCLOSURE 4

SIMULATION FACILITY FIDELITY REPORT

Facility Licensee: H. B. Robinson Steam Electric Plant

Facility Licensee Docket No.: 50-261

Facility Licensee No.: DPR-23

Operating Tests administered at: H. B. Robinson facility

Operating Tests Given On: December 16-17, 1987

During the conduct of the simulator portion of the operating tests identified above, the following apparent performance and/or human factors discrepancies were observed.

- During a steam break of 10^5/bm/hr, the simulator gave little information that was relevant to a Steam Break (e.g., 5/6 levels did not respond, 5/6 pressure did not drop). RCS temperature did not respond. Some of the indications did change, but not to the extent that was expected.

- The simulator "locked up" following an inadvertent SI Train A actuation.

- The simulator "locked up" on a number of occasions during different scenarios. The facility was aware of this.