U.S. NUCLEAR REGULATORY COMMISSION REGION I
OPERATOR LICENSING EXAMINATION REPORT

EXAMINATION REPORT NO. 88-16 (OL)

FACILITY DOCKET NO. 50-352

FACILITY LICENSE NO. NPF-39

LICENSEE: Philadelphia Electric Company
2301 Market Street
Philadelphia, Pennsylvania 19101

FACILITY: Limerick 1

EXAMINATION DATES: June 7-9, 1988

CHIEF EXAMINERS: Terry E. Walker 8/3/88
T. Walker, Senior Operations Engineer — Date

Allen D. Howe 8/3/88
A. Howe, Senior Senior Operations Engineer — Date

APPROVED BY: David J. Lange 8-3-88
David J. Lange, Chief, BWR Section, Operations Branch, Division of Reactor Safety — Date

SUMMARY: Written examinations and operating tests were administered to three senior reactor operator (SRO) and two reactor operator (RO) candidates. Two SROs and one RO passed the written examinations. One SRO and one RO failed the written examinations. Three SROs and one RO passed the operating examinations. One RO failed the operating examination.

DETAILS

TYPE OF EXAMINATIONS: Replacement

EXAMINATION RESULTS:

	RO Pass/Fail	SRO Pass/Fail
Written	1/1	2/1
Operating	1/1	3/0
Overall	1/1	2/1

1. CHIEF EXAMINERS AT SITE: T. Walker, Senior Operations Engineer
   A. Howe, Senior Operations Engineer

2. OTHER EXAMINERS: S. Pullani, Senior Operations Engineer
   T. Fish, Operations Engineer
   J. Hanek, EG&G (Examiner)
   M. Parrish, EG&G (Examiner)
   D. Lange, Chief, BWR Section (Observer)
   N. Conicella, Operations Engineer (Observer)
   T. Easlick, Operations Engineer (Observer)

3. The following is a summary of generic strengths and deficiencies noted on the operating tests. This information is being provided to aid the licensee in upgrading license and requalification training programs. No licensee response is required.

   3.1 Strengths

   - Knowledge of Remote Shutdown Panel and associated procedure
   - Familiarity with Piping and Instrumentation Drawings (P&IDs)
   - Security awareness

3.2 Deficiencies

- A general unfamiliarity to the Fuel Handling Casualty procedures were noted among the candidates. The facility was requested to check into the adequacy of training in this area.

- A general unfamiliarity with Administrative Procedures (for example, Shift Turnover Sheets) were noted among some candidates. The facility was requested to verify the adequacy of training received by the candidates in this area during their 3 months on-the-job training.

- Annunciator Response Procedures (ARPs) filed in a single binder were not easily accessible to the candidates during the simulator examination. The "required" category of ARPs, identified by a triangle on the annunciator windows were required to be referred to, once such annunciators are received. However, the candidates were not referring to such procedures during the simulator scenarios.

- Inadequate communication among the operating crew were noted during certain events of the simulator test scenarios. Examples of this were: (1) EHC Pressure Regulator Oscillation event and (2) Turbine Trip with Failure to Scram event. The details of these deficiencies were discussed with the facility during the exit interview.

4. The following is a summary of generic strengths and deficiencies noted from the grading of the written examinations. This information is being provided to aid the licensee in upgrading license and requalification training programs. No licensee response is required.

4.1 Strengths

- Knowledge of the effect of changes in related core parameters on Shutdown Margin (SDM). (Question 1.02)

- Knowledge of meaning of "Prompt Jump" and "Prompt Criticality". (Question 1.04)

- Knowledge of the effect on Control Rod Worth by changes in related core parameters. (Question 1.06)

- Knowledge of the effect of Xenon transients due to a scram from full power operation and subsequent startup on the radial flux shape and resultant change in Control Rod Worth of peripheral rods. (Questions 1.07 and 5.03)

- Knowledge of "Critical Power" and how it changes due to changes in related core parameters. (Question 1.11)

- Knowledge of automatic actions that occur on a LOCA signal while one loop of RHR is in Suppression Pool cooling and the reactor is operating at power. (Question 2.02)

- Knowledge of Diesel Generator trip signals while connected to the normal supply. (Question 2.03)

- *Knowledge of trip and interlock functions of SRM, IRM and APRM. (Question 3.04)

  *A deficiency for SRO candidates. (Question 6.10)

- Knowledge of how RCIC speed controller controls in the turbine speed in various modes of control. (Question 3.11)

- Knowledge of the guidance (in Administrative Procedure A7, "Shift Operation") concerning when a plant shutdown or scram shall be initiated by SLO or LO. (Questions 4.01 and 8.03)

- Knowledge of the requirements in 10 CFR 55 concerning the limitation on who may manipulate the controls of the reactor. (Question 4.01)

- Knowledge of the entry conditions and immediate operator actions for Special Event Procedure SE-1, "Remote Shutdown" (Questions 4.02 and 7.05)

- Ability to work thermodynamic problems using steam tables. (Question 5.05)

- Knowledge of Tech. Spec. for core thermal limits. (Question 5.09)

- Knowledge of plant response due to loss of feedwater flow transients. (Question 5.10)

- Knowledge of automatic actions initiated by Process Radiation Monitors. (Question 6.07)

- Knowledge of plant response due to instrument/signal failures in the Feedwater Level Control System. (Question 6.09)

- Ability to distinguish situations requiring an RWP. (Question 7.01)

- Knowledge of station administrative dose limits. (Question 7.02)

- Knowledge of entry condition to TRIP procedure. (Question 7.08)

- Knowledge of Safety Limits and Thermal Limits in Tech. Spec. (Question 8.01)

- Knowledge that Tech. Spec. could be violated to comply with a TRIP procedure. (Question 8.08)

- Knowledge and ability to identify Tech. Spec. LCOs and Actions for Primary Containment Integrity. (Question 8.09)

4.2 <u>Deficiencies</u>

- Ability to identify when the reacto [illegible] ing range after it is critical and on a steady per[illegible]d [illegible]n 1.05)

- Knowledge of the effect of Doppler [illegible] mitigation of a power transient. (Question 1.08(a))

- Knowledge of the contributing factors fo[illegible] Required" and "Available" NPSH. (Question 1.10)

- Knowledge of the automatic response of the Core Spray system pump and valves during a Small Break LOCA. (Question 2.05)

- Knowledge of automatic response of RHR (valve realignment) from its shutdown cooling mode due to a LPCI initiation. (Question 2.06(e))

- Knowledge of the valve actions on a RFP trip. (Question 2.09 (b))

- Knowledge of the response of the recirc. pump speed on a condensate pump trip while operating at power. (Question 3.01(b))

- Knowledge of the source of error in the indicated level of the reactor level instruments. (Question 3.02)

- Knowledge of the initiation signals for MSIV isolation during reactor startup. (Question 3.03(a))

- Knowledge of the response of the FWCS on a loss of level signal and its effects on plant. (Question 3.08)

- Knowledge that a Group I isolation will be received if the reactor pressure is reduced to 900 psig in an attempt to reseat a stuck open SRV. (Question 4.03(e))

- Knowledge and general understanding of Administrative Procedure A41, "Control of Plant Equipment".

- Knowledge of the requirements for entering a high radiation area. (Question 4.08(b))

- Knowledge of the requirements for implementing a temporary change to an approved procedure. (Question 4.09(b))

- Knowledge of relative worth of shallow and deep control rods. (Question 5.04)

- Knowledge of plant response on a loss of feedwater system. (Question 5.10)

- Knowledge of the requirements for a procedure to be considered valid and for implementing temporary changes to an approved procedure. (Question 8.04)

- Knowledge of the time limit for release of Tech Spec. equipment for surveillance testing. (Question 8.06(d))

5. Personnel Present at Exit Interview, June 10, 1988

5.1 NRC Personnel

N. Conicella, Operations Engineer
T. Easlick, Operations Engineer
A. Howe, Senior Operations Engineer
T. Kenny, Senior Resident Inspector
S. Pullani, Senior Operations Engineer

5.2 Facility Personnel

E. Firth, Superintendent - Training
G. Leitch, Vice President - Limerick Generating Station
D. Neff, Licensing Engineer
R. Nunez, Operations Training Supervisor
D. Weiksner, Instructor
S. Wilhelmson, Lead Instructor
A. Yarmer, Simulator Instructor

6. Summary of NRC Comments Made at Exit Interview

- The written examination was conducted on Tuesday, June 7, 1988. The candidates had very few questions during the written examination.

- The facility review of the written examination was conducted on Tuesday, June 7, 1988, immediately after the examination. No significant problems were identified during the examination review. The facility was reminded to send their written comments to the NRC and EG&G within five working days.

- The Training Department was cooperative during the examination process.

- The operating tests were conducted on Wednesday and Thursday, June 8 and 9, 1988. The generic strengths and weaknesses noted on the operating tests (see Section 3. of this report) were presented.

- Distractions to the candidates from outside personnel through the glass doors and windows of the simulator during the operating test were noted by the examiners which were corrected by the Training Department later.

- The need for isolating the candidates from outside personnel in between the simulator test scenarios were noted. A separate room could be arranged for candidates to wait in between scenarios. However, no significant problems were noted during the tests.

- Significant on-the-spot changes were required to be made in the simulator scenarios originally prepared by the NRC consultant before they are administered to the candidates on June 8 and 9, 1988. This was because the Simulator Malfunction Book, sent to the NRC and the consultant and used for the preparation of the original scenarios, was subsequently revised in the numbering and scope of malfunctions.

- Several problems concerning the simulator fidelity were encountered during the execution of the simulator scenarios. The details of these problems were discussed with the facility during the exit meeting (see Attachment 5).

- There were no problems with access to the plant. The operations staff was cooperative during the plant walk through portions of the examinations.

- The results of both the written and operating examinations would not be discussed at the exit meeting but would be contained in the Examination Report. Every effort would be made to send the candidates' results in approximately 30 working days.

- The reference materials provided to NRC for preparation of the written examination were generally adequate. However, the lack of specific Learning Objectives (LOs) to match several Knowledge and Abilities (K/As) in the K/A Catalog (NUREG-1123) were noted by the examiners. The facility personnel stated that they have plans to correct the deficiency before the next license examination is conducted.

ttachments:
1. Written Examination and Answer Key (RO)
2. Written Examination and Answer Key (SRO)
3. Facility Comments on Written Examinations after Facility Review
4. NRC Resolution of Facility Comments
5. Simulation Facility Fidelity Report

ATTACHMENT 1

U. S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR LICENSE EXAMINATION

FACILITY: Limerick

REACTOR TYPE: BWR-GE4

DATE ADMINSTERED: 88/06/06

MASTER COPY

EXAMINER: NRC REGION I

CANDIDATE

INSTRUCTIONS TO CANDIDATE:

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

CATEGORY VALUE	% OF TOTAL	CANDIDATE'S SCORE	% OF CATEGORY VALUE		CATEGORY
24.50	24.81 ~~24.44~~			1.	PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW
23.25 ~~24.25~~	23.54 ~~24.19~~			2.	PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS
26.25	26.58 ~~26.18~~			3.	INSTRUMENTS AND CONTROLS
24.75 ~~25.25~~	25.06 ~~25.19~~			4.	PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
98.75 ~~100.2~~		Final Grade	%		Totals

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

Candidate's Signature

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

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

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

3. Use black ink or dark pencil only to facilitate legible reproductions.

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

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

6. Use only the paper provided for answers.

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

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

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

10. Skip at least three lines between each answer.

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

12. Use abbreviations only if they are commonly used in facility literature.

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

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

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

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

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

18. When you complete your examination, you shall:

a. Assemble your examination as follows:

(1) Exam questions on top.

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

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

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

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

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

QUESTION 1.01 (2.50)

Concerning Prompt and Delayed Neutrons state whether EACH of the following TRUE or FALSE:

a. The percentage of delayed neutrons produced from fission increases as the age of the core increases. (0.5)

b. The energy level at which delayed neutrons are produced categorizes them as thermal neutrons. (0.5)

c. Neutrons produced from the moment of fission until 10 E-14 seconds are considered prompt. (0.5)

d. Delayed neutrons are produced as a result of both thermal fission of U-235 and fast fission of U-238. (0.5)

e. Delayed neutrons are the major factor in determining the rate of reactor power decrease immediately (after 1 second) following a scram. (0.5)

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

QUESTION 1.02 (2.00)

Will Shutdown Margin (SDM) INCREASE or DECREASE for each of the following?

a. Poison concentration increases (0.5)

b. A number of control rods are inserted into the core (0.5)

c. Moderator Temperature increases (0.5)

d. Plutonium 240 concentration increases. (0.5)

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

QUESTION 1.03 (1.50)

You are performing a normal reactor startup; as K-eff of the reactor increases, STATE the effect (INCREASE or DECREASE) on each of the following parameters for equal reactivity additions.

a. The magnitude of the change in count rate (0.5)

b. The rate of rise in the count rate (0.5)

c. The time it takes to reach a new equilibrium count rate (0.5)

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

QUESTION 1.04 (1.00)

a. During a rod withdrawal the operator will observe a sudden increase in reactor period initially. This sudden increase is referred to as _______________________. (0.5)

b. If the reactor were critical on prompt neutrons alone power would increase uncontrollably. This condition is called _________________. (0.5)

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

QUESTION 1.05 (2.00)

You have just taken the reactor critical from a cold condition and are increasing power on an 80 second period.

a. Utilizing control room instrumentation, STATE two methods which will tell you the heating range has been reached? Rod position and recirculation flow have been held constant. (1.0)

b. In which ONE of the following intervals was the heating range entered? (1.0)

(1) Interval 1 - reactor power increased by a factor of 6 in 143.3 seconds.

(2) Interval 2 - reactor power increased by a factor of 3 in 99.0 seconds.

(3) Interval 3 - reactor power increased by a factor of 5 in 128.8 seconds.

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

QUESTION 1.06 (2.00)

Will control rod worth INCREASE, DECREASE, or REMAIN THE SAME for each of the following?

a. Increasing moderator temperature (0.5)

b. Increasing the percent voids (0.5)

c. Increasing the fuel temperature (0.5)

d. Increase in Xenon concentration following a power change (0.5)

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

QUESTION 1.07 (1.00)

You are performing a reactor startup 12 hours following a scram which occurred after 30 days of full power operation.

WHICH statement below describes the expected effects of Xenon concentration when performing the startup?

CHOICES:

1. Thermal neutron flux will be highest in the same areas where the flux was highest during the previous operational period.
2. Thermal neutron flux will be higher in areas of high Xenon concentration than during the previous operational phase to maintain the same reactor power level.
3. Thermal flux will be pushed to the periphery of the core, making the periphery rods have a high rod worth.
4. Xenon burnup during the reactor startup will make the reactor go critical earlier in the rod sequence than is normal.

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

QUESTION 1.08 (3.00)

During operation at 100% power a feedwater train automatically isolates due to high water level in a heater.

STATE how each of following coefficients of reactivity will respond to MITIGATE or INCREASE the severity of this transient. Include a brief reason in your answer and consider the entire transient UNTIL the scram occurs.

a. Doppler Coefficient (1.0)

b. Moderator Temperature Coefficient (1.0)

c. Void Coefficient (1.0)

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

QUESTION 1.09 (2.00)

A reactor heat balance was performed (by hand) during the 00-08 shift due to the Process Computer being OOC. The GAF's were computed, but the APRM GAIN ADJUSTMENTS HAVE NOT BEEN MADE.

DETERMINE if each of the following statements is TRUE or FALSE.

a. If the feeuwater flow rate used in the heat balance calculation was LOWER than the actual feedwater flow rate, then the actual power is HIGHER than the currently calculated power. (0.5)

b. If the reactor recirculation pump heat input used in the heat balance calculation was OMITTED, then the actual power is HIGHER than the currently calculated power. (0.5)

c. If the steam flow used in the heat balance calculation was LOWER than the actual steam flow, then the actual power is HIGHER than the currently calculated power. (0.5)

d. If the RWCU return temperature used in the heat balance calculation was LOWER than the actual RWCU return temperature, then the actual power is HIGHER than the currently calculated power. (0.5)

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

QUESTION 1.10 (2.50)

a. LIST FOUR parameters which contribute to AVAILABLE NPSH (Net Positive Suction Head) for a recirculation pump. Limit your answer to those parameters which are DIRECTLY indicated in the CONTROL ROOM. (1.0)

b. Consider TWO Reactor Plant conditions:

Low Power and Low Flow (<10%)
OR
High Power and High Flow (>85%).

1. During which condition is the REQUIRED NPSH for a recirculation pump greater? (0.5)

2. During which condition is AVAILABLE NPSH for a recirculation pump greater and WHY is it greater? (1.0)

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

QUESTION 1.11 (3.00)

a. Define critical power. (1.0)

b. For EACH condition (a.-d.) given below, INDICATE whether it will cause an INCREASE, a DECREASE, or have NO EFFECT on CRITICAL POWER.

1. Local peaking factor (LPF) INCREASES (0.5)

2. DECREASE in inlet subcooling (0.5)

3. INCREASE in reactor pressure (0.5)

4. Axial power peak shifts from BOTTOM to TOP of channel (0.5)

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

QUESTION 1.12 (2.00)

With regard to MAPRAT:

a) WHAT is the relationship between MAPRAT and MAPLHGR? (1.0)

b) WHAT physical consequence could occur if the MAPLHGR Technical Specification limit is exceeded? (1.0)

(***** END OF CATEGORY 1 *****)

QUESTION 2.01 (3.00)

Answer the following questions concerning operation of the High Pressure Coolant Injection (HPCI) system.

a. STATE the minimum speed that the the HPCI turbine should be operated at AND TWO reasons for NOT operating the turbine below the minimum speed. (1.5)

b. The operator is observing the automatic initiation of the HPCI system. The minimum flow valve should close at what system flow? (0.5)

c. HPCI is being operated in the test mode when a high drywell pressure initiation signal is received. LIST two signals that will cause the HPCI Test Bypass to CST to close. (1.0)

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

QUESTION 2.02 (3.00)

Reactor is operating at power with the B loop of RHR in suppression pool cooling. An inadvertent LOCA signal is generated.

STATE if the operator WOULD or WOULD NOT observe each of the following conditions.

a. The ESW pumps are running 30 seconds after the diesels start. (0.5)

b. The CRD pumps are running. (0.5)

c. The RHR SW pump continues to operate to supply cooling to the RHR heat exchanger. (0.5)

d. Turbine building equipment compartment exhaust fan is tripped. (0.5)

e. The control room chillers are operating 2 minutes after the diesels start. (0.5)

f. The load center Transformer Breakers D114, D124, D134, and D144 are closed 5 seconds after the signal is generated. (0.5)

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

QUESTION 2.03 (2.00)

A diesel generator is operating for a surveillance in parallel with the normal supply. The diesel trips during operation. List 6 possible causes that may have caused the diesel engine to trip. (Setpoints not required)

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

QUESTION 2.04 (3.25)

Answer the following questions concerning automatic initiation of the SBLC system.

a. A reactor high pressure signal of 1093 psig exists. STATE two additional conditions that must exist prior to automatic initiation occurring. (1.0)

b. State the reactor level at which an initiation signal for automatic initiation of SBLC is generated. (0.25)

c. State the four actions that occur if an automatic injection of SBLC occurs. Do not include redundant components as separate actions. (2.0)

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

QUESTION 2.05 (2.50)

A small leak develops which results in depressurization of the reactor at a rate of 30 psi per minute. HPCI maintains reactor water level but drywell pressure increases to 3.2 psig.

a. For each component in the Core Spray system listed STATE the pressure or flow rate at which the component will operate during the depressurization.

1. Minimum flow valve closes. (0.5)

2. Injection valves open. (0.5)

3. Testable check valve disk opens. (0.5)

4. Core spray pump starts. (0.5)

b. What is CS rated flow and at what pressure should the operator observe this flow. (0.5)

QUESTION 2.06 (3.00)

The reactor is in cold shutdown with loop A of RHR in shutdown cooling. Loop B is inoperable. A break results in a loss of reactor inventory.

a. At what reactor level will the S/D cooling isolation occur? (0.25)

b. LIST the RHR system valves that will close or receive a close signal when the S/D cooling isolation occurs. (1.0)

c. At what reactor level will RHR receive a LPCI initiation signal? (0.25)

d. WHAT action(s) would the operator have to take other than arming and depressing the manual LPCI initiation pushbuttons in order for loop A of RHR to inject in the LPCI mode? (0.5)

e. LIST the RHR loop A valves that will automatically reposition when the LPCI initiation signal is received. (1.0)

QUESTION 2.07 (2.00)

a. LIST two conditions that will result in an off gas isolation. (Setpoints not required) (1.0)

b. LIST two automatic actions that occur due to an off gas isolation. (1.0)

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

QUESTION 2.08 (2.50)

A complete loss of RECW has occurred due to a rupture in the suction line from the head tank.

a. Assuming no operator action, WHAT 3 actions will occur in the RWCU system due to the loss of RECW? (1.5)

b. WILL the RECW pumps automatically trip due to the loss of suction? (0.5)

c. WHAT system can the operators use to supply flow to the RECW system? (0.5)

QUESTION 2.09 (3.00)

The reactor is operating at 98% power when a RFP trips.

a. LIST 4 of the possible causes for the trip other than manual trips. Only a single RFP has tripped and no other plant equipment has tripped. (Setpoints are not required) (2.0)

b. LIST 5 valve actions associated with the tripped RFP that the operator will observe. (1.0)

(***** END OF CATEGORY 2 *****)

QUESTION 3.01 (3.00)

STATE what EFFECT each of the conditions will have on 1A and 1B Reactor Recirculation Pump speed. (Values for speed are required.)

a. Both recirc pumps are operating at 35% flow. The master controller is set at minimum. An operator inadvertently places the 1A M/A Transfer Station in Auto. ~~(1.0)~~ 1.5

b. Both recirc pumps are operating at 95% flow. Condensate pump 'A' trips. ~~(1.0)~~ 1.5

c. 1B recirc pump is started with its M/A transfer station in automatic and master controller is at 75%. 1A is at 28% speed with its M/A transfer station in manual. (1.0)

Deleted.

point values
should be: Total
2.0
a. 1.0
b. 1.0

QUESTION 3.02 (2.50)

For each of the failures listed below state whether indicated level monitored by the operator is HIGHER THAN ACTUAL level, LOWER THAN ACTUAL level or the SAME AS ACTUAL level.

a. A break occurs in the reference leg. (0.5)

b. A wide range instrument is used when reactor pressure is 1000 psig. (0.5)

c. A 50 F increase in drywell temperature occurs. (0.5)

d. Narrow range level instrument is used when reactor pressure is 75 psig. (0.5)

e. Reactor pressure is below saturation temperature for the drywell. (0.5)

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.03 (2.75)

a. The mode switch is in startup with reactor pressure equal to 800 psig when a MSIV isolation occurs. LIST the possible signals that could have caused the isolation. (Include setpoints) (2.25)

b. (TRUE or FALSE) A scram will occur if MSIVs close in only two Main Steam lines. (0.5)

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.04 (3.00)

For each condition listed below SELECT the action(s) (listed below) that will occur.

STATE no action will occur.

a. A reactor startup is in progress with IRMs on range 2. The operator withdrawing SRM detectors also has channel A of the IRM selected. (.5)

b. The operator adjusts recirc flow such that there is a 15% mismatch between loops. (.5)

c. Reactor is in the RUN mode and an APRM fails downscale. All IRM are withdrawn and indicating 25 on range 3. (.5)

d. Reactor is in startup mode and control rods are withdrawn to 15% power. (.5)

e. An approach to criticality begins with IRM C on range 2. (.5)

f. APRM channel 'A' Mode switch is placed to standby. (.5)

Actions:

1. IRM 'A' detector will not withdraw.

2. A scram signal is generated

3. A rod block is generated.

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.05 (2.50)

a. Reactor water level has decreased to below -129 inches. Drywell pressure is 1.2 psig. RHR and CS pump interlock is satisfied. ADS valves should open in ______ seconds. (0.5)

b. Reactor water level has decreased to below -129 inches. Drywell pressure is 4.6 psig. RHR and CS pump interlock is satisfied. ADS valves should open in ______ seconds. (0.5)

c. Reactor water level decreased to below -129 inches but recovered to greater than -129 inches prior to either ADS timer timing out. Will an ADS actuation occur? (0.5)

d. An ADS blowdown is in progress. Will the blowdown stop if the operator places t' ADS Auto Inhibit Switches to INHIBIT? (0.5)

e. An operator observ- that both the green and the amber lights are lit for the acoust.. monitor for an SRV. Describe what information this provides to the operator concerning the SRV. (0.5)

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.06 (3.00)

The reactor is at 50% power with the load limit set at 65% and Maximum combined flow limiter at 115%. An electrical failure occurs that causes the pressure set signal to decrease 10 psi.

DETERMINE the final control valve flow rate and bypass valve flow rate. Refer to the attached drawing of the Electro-Hydraulic Control Logic (LOT-0590-6). DESCRIBE how you determined your answer.

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.07 (2.50)

STATE the automatic action(s) that will occur when each of the process radiation monitors exceed the condition listed.

a. Refueling Area Ventilation Exhaust Duct High Radiation. (0.5)

b. Reactor enclosure radiation monitors exceed the Hi-Hi setpoint. (1.5)

c. RHR ~~Heat~~ Service Water combined loop monitor Hi radiation. (0.5)

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.08 (3.00)

DESCRIBE the response of the FWCS that will occur if an instrument technician places NR A level instrument in test when it is the selected input to FWCS. The instrument technician simulates 0 inches level. Include the expected response of the Narrow Range level indications, RFP speed, feed flow, and any output signals to other systems (i.e. Recirc, RPS, etc.). Limit discussion to the response until either a Rx scram or RFP trip occurs. Assume initial power is 75% power.

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.09 (1.00)

An ATWS has occurred and reactor power is 25% on APRM. Reactor water level is 20 inches and drywell pressure is 1.2 psig. All scram valves opened and the SDV vent and drain valves shut.

DESCRIBE why the scram cannot be reset in any position of the reactor mode switch.

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.10 (1.50)

DETERMINE if the following statements are TRUE or FALSE.

a. A reactor startup is in progress using the A-2 withdraw sequence. The operator can continuously withdraw a Group 3 rod from position 04 to 08. (0.5)

b. RSCS will NOT allow rods to be moved in a manner that would cause the RWM to cause a rod block. (0.5)

c. The RWM will NOT allow the operator to continue insertion of control rods with 3 insert errors present. Power is below the LPSP and no withdraw errors are present. (0.5)

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

QUESTION 3.11 (1.50)

For each mode of operation of the RCIC speed control listed below briefly discuss how speed of the turbine is controlled.

a. During system startup

b. During automatic flow control

c. During manual flow control

(***** END OF CATEGORY 3 *****)

QUESTION 4.01 (2.50)

a. Per "Shift Operations (A-7)" procedure LIST the four statements of guidance concerning when a plant shut down or scram shall be initiated by a Senior Licensed or Licensed operator. (2.0)

b. Per 10CFR55 "Operator Licenses" who may manipulate the controls of the reactor other than Licensed Operators and Senior Operators. STATE what condition(s) must be met to allow this person to manipulate the controls. (0.5)

QUESTION 4.02 (2.50)

Answer the following concerning Special Event Procedure (SE-1) "Remote Shutdown."

a. LIST three areas of the plant in which a fire could necessitate the use of the Remote Shutdown procedure. (1.0)

b. The SRO determines that the Control Room must be abandoned. WHAT immediate actions are required prior to exiting the C.R.? (1.5)

QUESTION 4.03 (2.50)

The operator has observed indications of a stuck open relief valve during power operation. Answer the following questions in accordance with "Inadvertent Opening of a Relief Valve (OT-114)."

a. The operator is required to place _____________ (one/both) loops of suppression pool cooling in service. (0.5)

b. If the suppression pool temperature reaches ___________ F then the operator shall place the mode switch in "Shutdown." (0.5)

c. In addition to the Operational Transient procedure for a Stuck Open Relief Valve, the operator shall enter which procedure at 95 F suppression pool temperature. (0.5)

d. If the stuck open relief valve cannot be shut within _______________ the operator shall place the mode switch in "Shutdown". (0.5)

e. The operator is directed in the followup steps to "Reduce turbine inlet pressure to 900 psig" to attempt to reseat the valve. WHY does the instruction specify "Turbine inlet pressure" instead of "Reactor Pressure"?. (0.5)

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

QUESTION 4.04 (3.00)

For each set of conditions below LIST which Trip Procedure(s) would be entered.

a. A loss of drywell cooling occurs. Operators vent the drywell to maintain pressure < 1.2 psig. Drywell temperature is 150 F. (0.5)

b. A reactor scram occurs due to a turbine trip from 45% power. Reactor level decreases to -10 inches following the scram but is automatically recovered by feedwater. (0.5)

c. A MSIV isolation occurs due to improper testing by instrumentation technicians. The reactor scrams due to the isolation. (0.5)

d. During a reactor shutdown the operator places the mode switch in startup at 20% power. Reactor power decreases to 2% due to PARTIAL insertion of control rods. (0.5)

e. A small leak in the drywell causes drywell pressure to increase to 3.2 psig. (0.5)

f. A failure of the EHC system results in a pressure increase which causes high pressure scram. (0.5)

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

QUESTION 4.05 (2.50)

Answer the following concerning "Control of Plant Equipment (A-41)."

a. Who's responsibility is it to determine if independent verification of blocking is required? (0.5)

b. What two classes of equipment are required to have independent verification performed when removed from service or restored? (1.0)

c. A valve is determined to require independent verification because it is in one of the two classes listed in part b. If the valve is located in a high radiation area is an independent verification required? (0.5)

d. The Shift Supervision released a surveillance for testing at 0900 June 6, 1988. How long is permission granted to perform the surveillance ? (0.5)

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

QUESTION 4.06 (2.00)

Discuss the reasons for each of the following cautions concerning operation of the Feedwater system.

a. The A RFP should be placed in service first. (0.5)

b. Maintaining the RFPT speed less than 2000 rpm for at least one hour following turbine roll. (0.5)

c. Avoid excessive throttling of the RFP Discharge valve that causes the MGU controller to automatically raise RFP speed (to maintain RPV water level). (Two reasons required) (1.0)

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

QUESTION 4.07 (2.50)

Answer the following per "Shutdown Cooling Operation (S51.8.B)" procedure.

a. Placing the RHR system in Shutdown Cooling requires that the operator shut and tag the minimum flow valve. LIST the two purposes for performing this action. (1.0)

b. When operating RHR in the shutdown cooling mode the minimum pump flow that is allowed by the procedure is 1500 gpm. EXPLAIN why flow must be maintained greater than this value. (0.5)

c. EXPLAIN why flow through a RHR heat exchanger is limited to less than 11000 gpm. (0.5)

d. EXPLAIN why reactor water level must be maintained above 60 inches as read on the Shutdown range indicator (LI-42-R605) or 78 inches on the Upset range recorder (LR-42-R608). (Do not explain why the values are different.) (0.5)

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

QUESTION 4.08 (2.50)

You are directed to operate a valve in a high radiation area.

a. (TRUE or FALSE) The expected dose rate in the area will be >100 mr/hr. (0.5)

b. In order to enter the area the operator is required to have one of three items. LIST these three items. (1.5)

c. A RWP _________________ (WOULD/WOULD NOT) be required for access to the area. (0.5)

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

QUESTION 4.09 (2.00)

Answer the following in accordance with "Procedure for Temporary Changes to Approved Procedures (A-3)."

a. Maintenance has been completed on a valve in the RHR system. The SRO on shift has determined that only part of the RHR valve operability surveillance must be performed in order to determine if the valve is operable. IS a Temporary Procedure Change required per "Procedure for Temporary Changes to Approved Procedures (A-3)" in order to perform the procedure? (0.5)

b. An operator who is reviewing a recently revised procedure prior to performance determines that a step necessary to complete the procedure has been deleted. WHAT requirements must be met to implement a temporary change before the procedure can be performed? (1.5)

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

QUESTION 4.10 (3.25)

The operator observes confirming indications that a loss of MCC D114-R-G has occurred.

a. WHAT event procedure would the operator refer to for actions. (.25)

b. While the operator is performing the actions for the loss of MCC D114-R-G a loss of off-site power occurs.

   1. LIST 2 plant components that the operator is required to verify as starting. (Do not list redundant components in separate divisions) (1.0)

   2. WHAT 2 system lineups must be performed by the operator in order to supply cooling to drywell coolers. (1.0)

c. Subsequent to the loss of off-site power a loss of all AC occurs. LIST the 2 actions that the operator is required to verify. (1.0)

(***** END OF CATEGORY 4 *****)
(********** END OF EXAMINATION **********)

ANSWER 1.01 (2.50)

a. False

b. False

c. True

d. True

e. True [5 @ 0.5 ea.] (2.5)

REFERENCE

LGS: LOT-0860, PP. 4 & 5 Lesson Objective 5. LOT 0870 PP. 4 & 5, Lesson Objectives 3, 4 & 5
KAI 3.2

292001K102 ..(KA's)

ANSWER 1.02 (2.00)

a. SDM Increases

b. SDM Increases

c. SDM Increases

d. SDM ~~Decreases~~ Increases [4 @ 0.5 ea.] (2.0)

REFERENCE

LGS: LOT-0950, PP. 6 & 11 Lesson Objective 6.
KAI 2.6

29002K114 ..(KA's)

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

ANSWER 1.03 (1.50)

a. Increase

b. Increase

c. Increase [3 @ 0.5 ea.] [1.50)

REFERENCE

LGS: LOT-0970 P. 10. Learning Objective 1, 3.
KAI 2.9

292003K101 ..(KA's)

ANSWER 1.04 (1.00)

a. Prompt Jump. (.5)

b. Prompt Criticality. (.5)

REFERENCE

LGS: LOT-1430, P. 20, Lesson Objective 1.
KAI 3.3

292003K107 ..(KA's)

ANSWER 1.05 (2.00)

a. 1. Period indication becomes longer.
2. Indicated power on SRMs/IRMs is leveling off (due to power overshoot). [2 @ 0.5 ea.] (1.0)

b. Interval 2 From P=Poe(t/T). (In interval 2 the period has lengthened from 80 seconds. The other intervals have 80 second periods.) (1.0)

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

REFERENCE

LGS: LOT-1430 Learning Objective 4.
LGS: Normal Plant Startup GP-2 Appendix I PP. 4 & 5. LOT-1430 P. 5.
KAI 3.6 3.8

292008K113 292008K112 ..(KA's)

ANSWER 1.06 (2.00)

a. Increase

b. Decrease.

c. Remains the same.

d. Decrease [4 @ 0.5 ea.] (2.0)

REFERENCE

LGS: LOT-1490 PP. 5-12, Lesson Objectives 4 & 5.
KAI 2.5

292005K109 ..(KA's)

ANSWER 1.07 (1.00)

3 (1.0)

REFERENCE

LGS: LOT-1510 PP. 7-9, Lesson Objective 6. GE BWR Acdemic Series
Chapter 6, PP. 6-10a to 6-12a.
KAI 2.8

292006K108 ..(KA's)

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

ANSWER 1.08 (3.00)

a. Doppler will mitigate the transient [0.5] because of the increase in fuel temperature. [0.5] (1.0)

b. Moderator Temperature Coefficient will increase the severity, [0.5] due to increase in core inlet subcooling. [0.5] (1.0)

a. Void coefficient will increase the severity of the transient [0.5] because of the increase in core inlet subcooling. [0.5] (1.0)

REFERENCE

LGS: FSAR PP. 15.1-1 thru 15.1-4, Figure 15.1-2
KAI 3.3 3.2 3.4

295014K206 295014K204 295014K203 ..(KA's)

ANSWER 1.09 (2.00)

a. False

b. False

c. True

d. False

[4 @ .5 each] (2.0)

REFERENCE

LGS: LOT-1300 PP. 1-6.
KAI 2.7

293007K111 ..(KA's)

ANSWER 1.10 (2.50)

a. Feedwater temperature
Feedwater flow
RPV pressure
RPV water level [4 @ 0.25 ea.] (1.0)

b. 1. High flow, High power (0.5)
2. High flow, High power [0.50], due to the increased inlet subcooling from the increased feedwater flow. [0.50] (1.0)

REFERENCE

LGS: LOT-1290 PP.8-9
KAI 3.9 3.3

291004K106 202001K402 ..(KA's)

ANSWER 1.11 (3.00)

a. The assembly power that would cause DTB at some point in the assembly. accept bundle interchangably with assembly (1.0)

b. 1. Decreases

2. Decreases

3. Decreases

4. Decreases [4 @ 0.5 ea.] (2.0)

REFERENCE

LGS: 1370 PP. 8 Learning Objective 1, 2.
KAI 2.9 2.7 2.7 2.6

293009K126 293009K125 293009K124 293009K122 ..(KA's)

(***** CATEGORY 1 CONTINUED ON NEXT PAGE *****)

ANSWER 1.12 (2.00)

a. MAPRAT is the ratio of APLHGR TO APLHGR Limit OR the ratio of MAPLHGR(act) to MAPLHGR(LCO) (Either answer acceptable for full credit.) (1.0)

b. The clad temperature can exceed 2200 degrees F. during a DBA LOCA (1.0)

REFERENCE

LGS: LOT 1410 P.4 Lesson Objectives 3 & 4.
KAI 2.9 2.8
293009K111 293009K112 ..(KA's)

(***** END OF CATEGORY 1 *****)

ANSWER 2.01 (3.00)

a. 2200 rpm (0.5) Minimize water hammer in exhaust line (0.5) and proper operation of hydraulically operated valves which the attached lube oil pump supplies (0.5)

b. 600 gpm (0.5)

c. Injection valve to CS (F006) is not fully shut. (0.5)
Initiation signal. (0.5)
Accept drywell pressure and low level for the initiation signal.

REFERENCE

LGS: LOT-340 P. 6, 14, AND 15.
Lesson Objectives 6.
KA (3.2) (3.4) (3.2)

206000K418 206000K411 206000K407 ..(KA's)

ANSWER 2.02 ~~(3.00)~~ 2.5

a. Would not

b. Would not.

~~c. Would not.~~ deleted

d. Would.

e. Would not.

f. Would.

[~~6~~ 5 @ .5 each] ~~(3.0)~~ 2.5

REFERENCE

LGS: LOT-660 P. 12, 13.
Learning Objectives 5 and 6.
KA (3.3) (3.2) (3.1)

262001K403 264000K405 264000K408 ..(KA's)

ANSWER 2.03 (2.00)

Jacket Coolant Temp High
Jacket Coolant Low Press
Generator Ground Neutral Overcurrent
Lube Oil Low Pressure
Lube Oil High Temperature
Fire Protection Actuation
Engine Overspeed
Diesel Generator Differential Overcurrent

(2.0) (6 @ .33 each)

REFERENCE

LGS: LOT-670 P 11 and 12
Learning Objective 5.
KA (3.5) (4.0)

264000K402 264000K401 ..(KA's)

ANSWER 2.04 (3.25)

a. No APRM downscale (0.5)
118 ~~58~~ second time delay (0.5)

b. -38" (0.25)

c. All pumps start (0.5)
All squib valves open (0.5) or accept continuity lights extinguish.
RWCU isolates (0.5)
10 minute reset timer actuates. (0.5)
Decreasing tank level
Decreasing Reactor Power

REFERENCE

LGS: LOT-0310 p. 16 and 17.
Learning objectives 9, 10, and 11.
KA (4.2)

211000A308 ..(KA's)

ANSWER 2.05 (2.50)

a. 1. 775 gpm
2. 455 ~~445~~ psig
3. 330 psig (accept 310 to 350)
4. 455 psig

(2.0) (.5 for each setpoint)

b. 3175 gpm (.25) at 250 psig (.25)

REFERENCE

LGS: LOT-0350 p.6, 8, 10, 11
Learning Objectives 5, 7, 9, 11.
KA (3.8) (3.6) (3.8) (3.5) (3.7)

209001A304 209001A303 209001A302 209001A301 209001K408
..(KA's)

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

ANSWER 2.06 (3.00)

a. 12.5 inches (.25)

b. Inboard Shutdown Cooling Suction Valve (F009) (.25)
Outboard Shutdown Cooling Suction Valve (F008) (.25)
Shutdown Cooling Injection Valves (F015 A,B) (.25)
Head Spray Valves (F023, F022) (.25)
(Accept either the valve names or numbers)

c. -129 inches (.25)

d. The suction of RHR pumps will have to be aligned to the suppression pool (.5).

e. RHR HX bypass valve (F048) would open (.5) and the LPCI injection valves (F017) would open (.5)

REFERENCE

LGS: LOT-0370 p. 19, 22, and 23
Learning Objective 6, 7, 8, 10.
KA (4.2) (4.4) (4.1) (3.8) (3.6)

205000A205 205000K403 203000A308 203000A216 203000K401
..(KA's)

ANSWER 2.07 (2.00)

a. Low Process flow (0.5)
High Recombiner Outlet temp (0.5)

b. Isolates steam supply to first stage air ejectors. (.5)
Isolates suction piping from condenser (.5)

(***** CATEGORY 2 CONTINUED ON NEXT PAGE *****)

REFERENCE

LGS: LOT-0510 p. 9, 10.
Learning objective 2h, 4.
KA (3.1)

271000K408 ..(KA's)

ANSWER 2.08 2.0 ~~(2.50)~~

a. Outboard isolation valve shuts. (.5)
RWCU pumps trip (.5)
Demin hold pumps start (.5)

b. No (.5)

~~c. ESW (.5)~~ deleted

REFERENCE

LGS: LOT-0460 p. 8. 11, 12.
Learning objective 9.
KA (3.5) (3.1) (3.3) (3.2)

204000A201 295018A101 295018K301 295018K101 ..(KA's)

ANSWER 2.09 (3.00)

a. Low suction pressure
Low bearing oil pressure
Inactive thrust bearing wear high
Active thrust bearing wear high
RFPT High vibration
Overspeed
Low condenser Vacuum.
(~~1~~ 2.0) (4 @ .5 each)

b. RFP Discharge Check valve will shut (.2~~5~~)
HP and LP Stopvalves (.2~~5~~) and control valves (.2~~5~~) will shut
HP and LP below seat drains will open (~~.125~~) (0.2)
LP Steam supply valve upstream drain valve opens (~~.125~~) (0.2)
Min Flow recirc valve will shut if open (CAF if this valve would be open at the given power level) the recirc valve would not be open at this power per discussion during exam review with facility
(5 @ 0.2 each)

REFERENCE

LGS: LOT-0540 p. 13, 22.
Learning Objective 4.
KA (3.4)

259001A310 ..(KA's)

(***** END OF CATEGORY 2 *****)

ANSWER 3.01 2.00 (~~3.00~~)

a. 1A recirc pump will speed up to 45% (0.75) ~~1.125~~ ~~(0.75)~~. 1B will be unaffected ~~(0.25)~~. ~~.375~~ (0.25)

b. Both recirc pumps will decrease to 60 ~~75%~~. ~~1.5~~ (1.0) ~~(1.0)~~
allow .75 for stating 75% core flow which is equivalent to 60% speed

~~c.~~ When the discharge valve for 1B recirc pump is >90% the pump will speed up to 75% (0.75) 1A will be unaffected (0.25).
Deleted

REFERENCE

LGS: LOT-0040 figure 2, p. 6, 8.
Learning objectives 2, 3, 4 and 12.
KA (3.0) (3.5)

202002K604 202002K402 ..(KA's)

ANSWER 3.02 (2.50)

a. higher than actual

b. same as actual or higher than actual

c. higher than actual

d. higher than actual

e. higher than actual

(2.5) (0.5 each)

REFERENCE

LGS: LOT-0050 p. 28 and 29
Learning objective 7
KA (3.0) (3.4) (3.2) (3.2)

216000A211 216000A208 216000A207 216000A203 ..(KA's)

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

ANSWER 3.03 (2.75)

a. High Steam Line flow 140% of rated in a steam line
Low reactor vessel level -129 inches
MSL High temperature Tunnel Temp 192 or Turbine enclosure temp 165
MSL High Radiation 3 x NFPBG
Manual
(2.25)(.25 for each signal and .2~~5~~0 for each setpoint)

b. False (0.5)

REFERENCE

LGS: LOT-0120 p. 18
Learning Objective 9.
KA (4.0) (3.8)

239001K401 239001K127 ..(KA's)

ANSWER 3.04 (3.00)

a. 3 (.5)

b. No action occurs (.5)

c. 3 (.5)

d. 2 (~~.25~~ .5) ~~and 3 (.25)~~

e. 3 (.5)

f. 2 (.25) and 3 (.25)

(3.0) (6 @ .5 each)

REFERENCE

LGS: LOT-0250 p. 10. Learning Objective 10.
LOT-0270 p. 11, 12, and 13. Learning Objective 7.
KA (3.7) (3.7) (4.1)

215005K402 215005K401 215003K401 ..(KA's)

ANSWER 3.05 (2.50)

a. ~~420~~ 525 (0.5)

b. 105 (0.5)

c. no (0.5)

d. no (0.5)

e. The valve has opened (0.25) but is presently closed (0.25).

REFERENCE

LGS: LOT-0330 p. 9, 12, 13. Figure 6.
Learning Objectives 2, 5, and 6.
KA (3.7) (3.8) (3.8) (3.7)

218000A303 218000K501 218000K402 218000K401 ..(KA's)

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

ANSWER 3.06 (3.00)

Output from pressure summer A increases to ~~40~~ 25 psi (.5). The HVG passes this signal to the Pressure/Flow converter (.5). The Pressure/Flow converter output will increase to 83% (.5). The load limit will limit the signal to the governor valves to 65% (.5). Total flow is less than the maximum combined limiter (.5) so the bypass valves will open to pass 17% flow ~18% (.5).

*REFERNCE
LGS: LOT-0590 p. 6, 7, 8, 13, 15, 16.
Learning objectives 3, 8.
KA (3.7) (4.1) (3.7)

REFERENCE
241000K308 241000K306 241000K305 ..(KA's)

ANSWER 3.07 (2.50)

a. Refuel floor supply and exhaust isolates. (~~.5~~ .25)
SGT starts if alighed to the refuel floor (.25)

b. Supply and exhaust ventilation isolates. (.5)
Standby gas treatment starts. (.5)
Reactor enclosure recirculation starts (.5)

c. Trips RHR service water pumps. (.5)

REFERENCE

LGS: LOT-0720 p. 10, 11, 18.
Learning Objective 2.
LGS: LOT-0180 p. 23
Learning Objective 2.
KA (3.6) (3.2) (3.6) (3.6) (3.7)

272000K402 272000K109 272000K108 272000K106 272000K101
..(KA's)

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

ANSWER 3.08 (3.00)

NR A (.375) and the NR Recorder (.375) will indicate 0 inches.
All remaining level instruments will track upward as feedflow increases (.375).
The RFP will accelerate to the high speed setting (.375).
Feedflow will increase (.375).
Recirc pumps will runback to 28% (.375).
The RFP (.375) and Main turbine trip on high level (0.375).

REFERENCE

LGS: LOT-0550 p. 18, 19.
Learning Objective 6, 7c.
KA (3.2) (3.8) (3.7) (3.4) (3.5) (3.6)

259002A203 259002K605 259002K307 259002K302 259002K301
259002K115 ..(KA's)

ANSWER 3.09 (1.00)

The SDV scram bypass will only operate in Shutdown or Refuel. (.5)
The scram cannot be reset in startup, refuel or shutdown because of the high APRM flux. (.5)

REFERENCE

LGS: LOT-0300 p. 8, 9, 14.
Learning objective 7, 8.
KA (3.9) (3.8)

212000K219 212000A404 ..(KA's)

(***** CATEGORY 3 CONTINUED ON NEXT PAGE *****)

ANSWER 3.10 (1.50)

a. False (.5) (Continuous Withdraw Inhibit will prevent continuous withdrawl.)

b. False (.5) (RWM divides RSCS group into more than one group)

c. True (.5)

REFERENCE

LGS: LOT-0100 p. 10, 16.
Learning Objective 6.
LGS: LOT-0090 p. 12, 16.
Learning Objective 3.
KA (3.1) (3.3) (3.3) (3.3) (3.4) (3.5) (3.3) (3.2)
201006A301 201006K403 201006K402 201006K401 201006K106
201006K406 201004K403 201004K402 ..(KA's)

ANSWER 3.11 (1.50)

a. a ramp generator controls turbine speed. (0.5)

b. Flow controller produces an electrical signal proportional to the difference between desired flow and actual flow. (0.5)

c. Speed is adjusted by the operator. (0.5)

REFERENCE

LGS: LOT-0380 p. 12
Learning Objective 10.
KA (3.7) (3.6) (3.6) (3.7)

217000A401 217000A304 217000A302 217000A105 ..(KA's)

(***** END OF CATEGORY 3 *****)

ANSWER 4.01 (2.50)

a. observations of plant conditions and equipment indicates a safety hazard. (0.5)
doubt as to whether safe conditions exist (0.5)
when RPS parameters have been exceeded without a scram (0.5)
approved procedures so direct (0.5)

b. License trainees (0.25) under direct supervision of a licensed operator. (0.25)

REFERENCE

10CFR55.13
LGS: Shift Operations (A-7) p. 11, 15 and 16.
LGS: LOT-1570 Learning Objective 2 and 3.
NRC Information Notice No. 88-20.
KA (3.7)

201001G001 ..(KA's)

ANSWER 4.02 (2.50)

a. Main Control Room (.33)
Aux. Equipment Room (.33)
Cable Spreading Room (.33)

b. Scram the reactor (.5)
Trip the main turbine (.5)
Close the MSIVs (.5)

REFERENCE

LGS: LOT-1563 p. 3
Learning objectives 1, 2.
LGS: Remote Shutdown (SE-1) p. 1 and 2.
KA (3.8) (4.1)

295016G011   295016G010   ..(KA's)

ANSWER   4.03   (2.50)

a. both (.5)

b. 110 (.5)

c. T-102 (Primary Containment Control) (.5) (Accept either procedure name or number)

d. 2 minutes (.5)

e. Reactor pressure CANNOT be reduced to 900 psig without receiving a GP I isolation (.5)

REFERENCE

LGS: LOT-1540 p. 30, 32, and 33. Learning objectives 1 and 2.
LGS: Inadvertent Opening of a Relief Valve (OT-114) p. 1.
LGS: Inadvertent Opening of a Relief Valve (OT-114) Bases p. 3.
KA (3.0) (3.8) (4.1) (4.2) (3.3) (3.8)

239001K401   241000K504   239[illegible]015   239002G014   239002G001
239002A201   ..(KA's)

ANSWER 4.04 (3.00)

a. T-102 (Primary Containment Control) (.5)

b. T-100 (Scram) (.5)

c. T-101 (RPV Control) (.5)

d. T-100 (Scram) (.5)

e. T-101 (RPV Control) (.25) and T-102 (Primary Containment Control) (.25)

f. T-101 (RPV Control) (0.5)

(Accept procedure name or number)

REFERENCE

LGS: LOT-1560 p. 7
Learning Objective 3
LGS: RPV Control (T-101)
LGS: Primary Containment Control (T-102)
LGS: Scram (T-100)
KA (4.3) (4.2) (4.2) (4.3) (4.2) (4.4) (4.3)

295037G011	295031G011	295030G011	295029G011	295028G011
295006G011	295024G011	..(KA's)		

ANSWER 4.05 2.0 ~~(2.50)~~

a. Chief Operator (.5)

b. Safety related (.5) or tech spec related (.5)
accept Q listed equipment for safety related.

deleted ~~c. No. (.5)~~

d. Until 0800 June 7, 1988. (.5)

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

REFERENCE

LGS: Procedure for Control of Plant Equipment (A-41) p. 6, 8, 12.
LGS: LOT-1570 p. 44. Learning Objective 3.
KA (3.9)

294001K102 ..(KA's)

ANSWER 4.06 (2.00)

a. Makes it possible to make up to the RPV using the Startup Bypass valve. (.5)

b. Allows for sufficient RFPT pre-warming. (.5)

c. Can cause unstable flow (.5) and excess valve wear (.5).

REFERENCE

LGS: Placing a Standby RFP in Service (S06.1.C) p. 3
LGS: Removing RFPs from service to a Standby Condition (S06.2.C)
KA (3.6) (3.9) (3.2)

259001G010 259001A402 259001A401 ..(KA's)

ANSWER 4.07 (2.50)

a. ~~Ensure all flow is to the vessel (0.5) and~~ prevent inadvertant draining of the vessel ~~(0.5)~~ (1.0)

b. Prevent pump overheating (0.5)

c. Prevent damage to the heat exchanger (0.5)

d. Ensure proper natural circulation (0.5)

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

REFERENCE

LGS: Shutdown Cooling Operation (S51.8.b) p. 3 and 5.
KA (3.2) (3.6)

205000K102 205000G010 ..(KA's)

ANSWER 4.08 (2.50)

a. True (0.5)

b. Radiation monitoring instrument continuously operating (0.5)
An alarming dosimeter (0.5)
A qualified HP technician (0.5)

c. would (0.5)

REFERENCE

LGS: Technical Specification 6.12
LGS: LOT-1760 p. 7, 13, and 15.
Learning Objective 1, 3, and 4.
KA (3.3)

294001K103 ..(KA's)

ANSWER 4.09 (2.00)

a. No. (.5)

b. 1. Determine that the change does not change the intent of the procedure. .25 (~~.5~~)
2. Document the change .25 (~~.5~~)
Have the change approved by the shift superintendent (.5) and an individual knowledgeable in the areas affected by the procedure (.25) who is a member of PORC or previously designated (.25).

(***** CATEGORY 4 CONTINUED ON NEXT PAGE *****)

REFERENCE

LGS: Procedure for Temporary Changes to Approved Procedures (A-3) p. 3.
LGS: Technical Specifications 6.8.3.
LGS: LOT-1570 p. 16. Learning objective 2.
KA (4.2).

294001A102 ..(KA's)

ANSWER 4.10 (3.25)

a. E-D114-R-G (.25)

b. 1. Diesels (.5)
ESW pumps (.5)

2. Line up ESW to RECW (.5)
Line up RECW to Drywell coolers (.5)

c. Rx scram (.5)
MSIV isolation (.5)

REFERENCE

LGS: LOT-1566 p. 2, 4, 6. Learning Objectives 2 and 3.
LGS: Loss of All AC Power (Station Blackout) (E-1) p. 1.
LGS: Loss of Off-Site Power (E-10/20) p. 1 and 2.
LGS: Loss of MCC D114-R-G.
KA (3.9)
295003G010 ..(KA's)

(***** END OF CATEGORY 4 *****)
(********** END OF EXAMINATION **********)

ATTACHMENT 2

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

FACILITY: LIMERICK 1

REACTOR TYPE: BWR-GE4

DATE ADMINISTERED: 88/06/07

EXAMINER: NRC REGION I

CANDIDATE: MASTER

INSTRUCTIONS TO CANDIDATE:

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

CATEGORY VALUE	% OF TOTAL	CANDIDATE'S SCORE	% OF CATEGORY VALUE	CATEGORY
25.00	25.00			5. THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS
24.50 ~~25.00~~	24.50 ~~25.00~~			6. PLANT SYSTEMS DESIGN, CONTROL, AND INSTRUMENTATION
25.50 ~~25.00~~	25.50 ~~25.00~~			7. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
25.00	25.00			8. ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS
100.00		Final Grade	%	Totals

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

Candidate's Signature

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

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

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

3. Use black ink or dark pencil only to facilitate legible reproductions.

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

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

6. Use only the paper provided for answers.

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

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

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

10. Skip at least three lines between each answer.

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

12. Use abbreviations only if they are commonly used in facility literature.

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

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

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

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

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

18. When you complete your examination, you shall:

a. Assemble your examination as follows:

(1) Exam questions on top.

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

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

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

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

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

QUESTION 5.01 (2.50)

For each of the following events, STATE which coefficient of reactivity would act FIRST to change reactivity.

a. Control rod drop at 100% power (0.5)

b. SRV opening at 100% power (0.5)

c. Loss of shutdown cooling while in Cold Shutdown (0.5)

d. One recirc pump trips while at 50% power (0.5)

e. Loss of one feedwater heater (extraction steam isolated) (0.5)

QUESTION 5.02 (2.25)

Following a normal increase in power from 75% to 100% with recirculation flow, HOW will each of the following parameters change (INCREASE, DECREASE, or REMAIN THE SAME) and WHY?

a. The pressure difference between the reactor and the turbine steam chest (0.75)

b. Condensate subcooling at the exit of the main condenser (0.75)

c. Feedwater temperature (at inlet to the reactor vessel) (0.75)

QUESTION 5.03 (3.00)

Reactor power was decreased from 100% to 50%.

a. Briefly EXPLAIN WHY the xenon concentration will increase following the manuever. (1.00)

b. How will peripheral control rod worth be affected (INCREASE, DECREASE, or REMAIN THE SAME) during the xenon peak? Briefly EXPLAIN your answer. (1.50)

c. Will the new (50% power) equilibrium xenon reactivity be MORE THAN, LESS THAN, or EQUAL TO one half the 100% equilibrium value? (0.50)

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

QUESTION 5.04 (2.00)

The reactor is operating at 75% rated power and the operator is withdrawing control rods.

WILL the withdrawal of a central control rod from notch 04 to notch 08 have a LARGER or SMALLER affect than withdrawal of the same rod from notch 36 to notch 40 on EACH of the following parameters?

a. Overall core thermal power (0.50)

b. Axial flux distribution (0.50)

c. Radial flux distribution (0.50)

d. Local power surrounding the rod (0.50)

QUESTION 5.05 (2.00)

Using the Steam Tables or Mollier Diagram, calculate HOW LONG it will take to cooldown from 1000 psig to 0 psig at the maximum allowable cooldown rate allowed per GP-3," Normal Plant Shutdown". (2.00)

QUESTION 5.06 (2.25)

Following an AUTO INITIATION of HPCI at a reactor pressure of1000 psig, reactor pressure decreases to 500 psig.

HOW are each of the following parameters affected (INCREASES, DECREASES, REMAINS CONSTANT) by the change in reactor pressure? BRIEFLY EXPLAIN your choice.

ASSUME the HPCI System is operating in automatic as designed.

a. HPCI flow to the reactor. (0.75)

b. HPCI pump discharge head (assuming NPSH remains constant). (0.75)

c. HPCI turbine RPM. (0.75)

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

QUESTION 5.07 (2.00)

EXPLAIN HOW it is possible to produce an increase in power as control rods are inserted into the core (Reverse Power Effect). Include in your answer under WHAT conditions this is possible. (2.00)

QUESTION 5.08 (2.00)

A reactor heat balance was performed (by hand) during your shift due to the Process Computer being out of service.

STATE whether each of the following statements is TRUE or FALSE.

a. If the feedwater flow rate used in the heat balance calculation was LOWER than the actual feedwater flow rate, then the actual power is HIGHER than the currently calculated power. (0.50)

b. If the reactor recirculation pump heat input used in the heat balance calculation was OMITTED, then the actual power is HIGHER than the currently calculated power. (0.50)

c. If the steam flow used in the heat balance calculation was LOWER than the actual steam flow, then the actual power is HIGHER than the currently calculated power. (0.50)

d. If the RWCU return temperature used in the heat balance calculation was LOWER than the actual RWCU return temperature, then the actual power is HIGHER than the currently calculated power. (0.50)

QUESTION 5.09 (2.00)

A periodic core performance edit (P-1) has just been completed by the Process Computer. After reviewing the output, you notice a MAPRAT value equal to 1.002.

a. WHAT is the relationship between MAPRAT and MAPLHGR ? (0.50)

b. Have any thermal limits been exceeded? If so, WHICH one(s)? EXPLAIN your answer. (1.00)

c. WHAT physical consequence could occur if the MAPRAT limit is exceeded? (0.50)

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

QUESTION 5.10 (3.00)

When the reactor is at full power, a spurious trip of all feedwater pumps occurs. Using the attached FSAR Figure 15.2-9, EXPLAIN WHY the following parameters respond for the periods stated below.

a. WHY does the reactor water level decrease between 0 seconds and 7 seconds? (0.25)

b. WHY does reactor water level continue to decrease following the scram at approximately 7 seconds? (2 reasons) (0.50)

c. WHY does the reactor pressure decrease between 0 seconds and 7 seconds and WHY does the rate of pressure decrease INCREASE after 7 seconds? (0.75)

d. WHY does reactor pressure increase following the MSIV closure at approximately 18 seconds and WHY does the increase stop at approximately 32 seconds? (0.75)

e. WHY does the core (inlet) flow SLOWLY decrease between 0 seconds and 15 seconds and WHY does it decrease at a FASTER rate after 15 seconds? (0.75)

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

QUESTION 5.11 (2.00)

a. Choose WHICH ONE of the following events is the most likely to produce water hammer. (1.00)

1. Core Spray pumps A and C are running in full flow test lineup, CS pump C is stopped

2. Core Spray pumps A and C are running in full flow test lineup, the test line orifice becomes blocked with debris

3. Core Spray pump A is running in full flow test lineup, Core Spray pump C is started

4. Core Spray pump A is started with the suction valve closed

b. Choose WHICH ONE of the pairs listed below will complete the following statement. (1.00)

With Core Spray pump A running in full flow test lineup, starting Core Spray pump C will _____________ CS pump A flow and ___________ CS pump A discharge pressure.

1. increase, increase

2. increase, decrease

3. decrease, increase

4. decrease, decrease

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

QUESTION 6.01 (1.00)

HPCI has automatically initiated, taking a suction from the suppression pool and injecting to the reactor vessel. The suction screen becomes partially clogged, causing the pump suction to reach 18" Hg vacuum.

Assuming no operator action and the initiation signal present, the response of HPCI will be to: (CHOOSE ONE)

a. Continue to inject

b. Trip

c. Isolate

d. Trip then restart

(1.00)

QUESTION 6.02 (2.50)

Reactor water level has decreased to below -129 inches. Drywell pressure is 1.2 psig. The RHR and CS pump interlock is satisfied.

a. WHEN will the ADS blowdown commence? (0.50)

b. WHY is a low level (12.5") signal used along with a low level initation signal (-129") to initiate ADS? (0.50)

c. During blowdown the operator depresses the ADS DIV I Logic Reset Push Button. DESCRIBE the response of the ADS system. (0.50)

d. During blowdown the operator turns ADS DIV I and II Auto Inhibit Switches to the INHIBIT position. DESCRIBE the response of the ADS system. (0.50)

e. An operator observes that both the green and amber lights are lit for the acoustic monitor for an SRV. DESCRIBE what information this provides to the operator concerning the SRV. (0.50)

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

QUESTION 6.03 (2.00)

The plant is operating at 23% power and both Recirc Pump M/A Transfer Stations are in MANUAL with the Master Controller set at 50% speed demand. The "Recirc Flow B Limit" annunciator is CLEAR.

For each of the following situations, STATE HOW the speed of Recirc Pump "B" will change (INCREASE, DECREASE, or REMAIN THE SAME), WHICH COMPONENT(S) of the control system will limit or prevent the speed change and WHAT SPEED the Recirc Pump is limited to.

NOTE: Figures T-LOT-0040-2 & 8 are provided for reference.

a. Vessel pressure oscillations cause feedwater flow (actual and indicated) to oscillate plus and minus 5%. (1.00)

b. Recirc Pump "B" M/A Transfer Station manual decrease pushbutton ~~potentiometer~~ is pushed ~~turned~~ fully in continuously as far as it will go. ~~the counter clockwise direction.~~ (1.00)

QUESTION 6.04 (3.00)

A reactor high pressure signal of 1093 psig exists.

a. WHAT two (2) additional conditions must exist for an automatic initiation of SBLC to occur? (1.00)

b. STATE the four (4) actions that occur upon an automatic initiation of SBLC. Do not include redundant components as separate actions. (2.00)

QUESTION 6.05 2.50 ~~(3.00)~~

The mode switch is in STARTUP with reactor pressure equal to 800 psig when a MSIV isolation occurs.

a. LIST the possible signals that could have caused the isolation. (Setpoints not required) (1.00)

b. The MSIVs required closing time is three (3) to five (5) seconds. WHAT are two (2) reasons for each the minimum and maximum time requirements? (1.00)

c. TRUE or FALSE? A scram will occur if MSIVs close in only two of the Main Steam lines. (0.50)

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

QUESTION 6.06 (3.00)

Under LOCA conditions with a LOSS OF OFFSITE POWER, Diesel Generator D11 output breker has just closed on to Safeguard Bus D11.

a. WHAT four (4) loads will be automatically sequenced on to the bus? LIST the loads in the sequencing order. (Times are not required) (2.0)

b. LIST two (2) other loads on the bus that will not automatically sequence onto the bus. (0.5)

c. WHAT operator action is required to restart the loads that do not automatically sequence onto the bus? (0.5)

QUESTION 6.07 (2.50)

STATE the automatic action(s) that will occur when each of the process radiation monitors exceed the condition listed.

a. Refueling Area Ventilation Exhaust Duct High Radiation. (0.50)

b. Reactor enclosure radiation monitors Hi-Hi setpoint. (1.50)

c. RHR ~~Heat~~ Service water combined loop monitor Hi radiation. (0.50)

QUESTION 6.08 (3.00)

The reactor is at 50% power with the Load Limit set at 65% and Maximum Combined Flow Limiter at 115%. An electrical failure occurs that causes the pressure set signal to decrease by 10 psi.

DETERMINE the final control valve flow rate and bypass valve flow rate. Refer to the attached drawing of the Electro-Hydraulic Control Logic (LOT-0590-6). DESCRIBE how you determined your answer. (3.00)

QUESTION 6.09 (2.00)

The Feedwater Control System is being operated in 3-Element Control using reactor level detector channel "A". Reactor power is at 85%.

For each of the following instrument or control signal failures, STATE HOW reactor water level will INITIALLY respond (INCREASE, DECREASE, or REMAIN CONSTANT) and briefly EXPLAIN what happens in the Feedwater Control System to cause the response.

NOTE: A block diagram of the Feedwater Control System is attached.

a. Channel "A" reactor level detector signal fails downscale. (1.0)

b. "B" reactor feed pump speed controller fails low. (1.0)

QUESTION 6.10 (3.00)

For each condition SELECT the action(s) from the list below that will occur. If no action will occur, state NONE.

a. A reactor startup is in progress with IRMs on range 2. The operator withdrawing SRM detectors also has channel A of the IRMs selected. (0.50)

b. The operator adjusts recirc flow such that there is a 15% mismatch between loops. (0.50)

c. Reactor is in the RUN mode and APRM 'D' fails downscale. All IRMs are withdrawn and indicating 25 on range 3. (0.50)

d. Reactor is in STARTUP mode and control rods are withdrawn to 15% power. (0.50)

e. An approach to criticality begins with IRM C on range 2. (0.50)

f. Reactor is in RUN mode and APRM 'A' is bypassed, APRM 'F' mode switch is placed in the "STANDBY" position. (0.50)

ACTIONS:

1. Half Scram

2. Full Scram

3. Rod Block

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

QUESTION 7.01 (2.00)

In accordance with HP-310, "Radiation Work Permits", STATE (YES/NO) whether an RWP is required for each of the following conditions:

a. Whole body radiation level = 25 millirem/hr (0.50)

b. Average removable surface contamination of 5000 dpm/100 sq.cm , beta-gamma (0.50)

c. Average removable surface contamination of 500 dpm/ 100 sq.cm, alpha (0.50)

d. Neutron dose of 25 millirem/hr (0.50)

QUESTION 7.02 (2.00)

According to Health Physics Procedure HP-102, "Administrative Dose Limits, Dose Extensions, and Notification Requirements":

a. WHAT is the PECO administrative whole body dose limit for 1 year with a current NRC Form-4 on file? (0.50)

b. Based upon 10CFR20, WHAT is the maximum allowable whole body accumulated dose for a 30 year old person? (1.00)

c. WHAT whole body exposure, if exceeded, requires immediate notification (1 hour) to the NRC? (0.50)

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

QUESTION 7.03 (2.00)

a. Concerning procedure GP-5, "POWER OPERATIONS":

1. Power is decreased to 8% rated thermal power. The reactor operator informs you that the control rod pattern is NOT latched in the RSCS. What action per GP-5 do you take? (0.50)

2. Why are you cautioned to avoid prolonged recirculation pump operation at resonance speeds? (0.50)

b. Concerning GP-3, "NORMAL PLANT SHUTDOWN":

1. Why are recirculation pump speeds required to be maintained within 5% of each other? (Assume power is 50%) (0.50)

2. At about 60% power when shutting down the first condensate pump, WHY should the operator be prepared to take manual control of the individual Reactor Feed Pump MGU controllers? (0.50)

QUESTION 7.04 (2.50)

In accordance with procedure S51.8.b, "Shutdown Cooling Operation":

a. Placing the RHR system in Shutdown Cooling requires that the operator shut and tag the minimum flow valve. LIST the two purposes for performing this action. (1.0)

b. When operating RHR in the shutdown cooling mode the minimum pump flow that is allowed by the procedure is 1500 gpm. EXPLAIN why flow must be maintained greater than this value. (0.5)

c. EXPLAIN why flow through a RHR heat exchanger is limited to less than 11000 gpm. (0.[illegible])

d. EXPLAIN why reactor water level must be maintained above 60 inches as read on the Shutdown range indicator ([illegible]-42-R605) or 78 inches on the Upset range [illegible] (LR-42-R608). (Do not explain why the values are differe[illegible]) (0.5)

(***** CATEGORY [illegible] NUED ON NEXT PAGE *****)

QUESTION 7.05 (3.00)

In accordance with Special Event Procedure SE-1, "Remote Shutdown":

a. For each of the following situations STATE (YES/NO) whether entry into SE-1 is required.

1. Fire in a Diesel Generator Room (0.50)

2. Fire in Auxiliary Equipment Room (0.50)

3. Cable spreading Room uninhabitable due to noxious fumes (No fire) (0.50)

b. LIST the three (3) immediate operator actions to be performed in the Control Room, prior to evacuation, when a remote shutdown is required. (1.50)

QUESTION 7.06 ~~(2.50)~~ 3.00

During a reactor startup, the unit is at 28% power and 50% core flow when reactor water level begins to oscillate (+ 5 inches).

a. WHICH Operational Transient Procedure(s) should be entered? If none, state NONE. (1.00)

b. WHAT ACTIONS would you, as the Shift Supervisor, direct the operators to take? Include any immediate actions required by procedure. (1.00)

c. If reactor water level reaches 100" the operator, by procedure, is directed to scram the reactor and close the MSIVs. WHAT is the bases for each of these actions? (1.00)

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

QUESTION 7.07 (2.50)

For each of the following conditions, determine whether entry into the Off Normal (ON) Procedures is required. If entry is not required, state NONE. STATE any ON(s) that would be entered. The ON Procedure Index is attached for your reference.

a. An unexplained increase in reactor power accompanied by an unexplained decrease in core flow indication (0.50)

b. ROD OVERTRAVEL alarm when rod is fully withdrawn (0.50)

c. CRD CHARGING WATER LOW PRESS alarm (0.50)

d. Total loss of SRMs in STARTUP mode with IRMs in range 3 (0.50)

e. Standby Gas Treatment System surveillance test results indicate that SGTS can maintain the Reactor Enclosure at -0.18 inches of water with a flow rate of 1350 SCFM (0.50)

QUESTION 7.08 (3.00)

For each set of conditions below, STATE which, if any, Trip Procedures (100 Series) should be entered. If none, state NONE.

a. A loss of drywell cooling occurs. Operators vent the drywell to maintain pressure below 1.2 psig. Drywell temperature is 150 degrees F. (0.50)

b. A reactor scram occurs due to a turbine trip from 45% power. Reactor level decreases to -10 inches following the scram but is automatically recovered by feedwater. (0.50)

c. A MSIV isolation occurs due to improper testing by I&C technicians. The reactor scrams due to the isolation. (0.50)

d. During a reactor shutdown, the operator places the mode switch in STARTUP at 20% power. Reactor power decreases to 2% due to partial insertion of control rods. (0.50)

e. A small leak in the drywell causes drywell pressure to increase to 3.2 psig. (0.50)

f. A failure of the EHC system results in a pressure increase which causes a high pressure scram. (0.50)

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

QUESTION 7.09 (2.50)

Procedure T-113, "Blowdown Cooling", directs the operator to steam cool the reactor. This is accomplished by opening one SRV. If RPV pressure drops below 700 psig during steam cooling, the procedure directs the operator to T-112, "Emergency Blowdown".

a. WHY must Emergency Blowdown be performed in place of steam cooling when pressure drops below 700 psig? (0.50)

b. HOW MANY SRVs are required for performing the Emergency Blowdown? (0.50)

c. After Emergency Blowdown is complete, it is assumed that injection from at least one system will be successful. WHAT is the preferred system and WHY is this system preferred over other systems? (1.50)

QUESTION 7.10 (3.00)

Using the attached Emergency Plan Implementing Procedure EP-101, "Classification of Emergencies", CLASSIFY the following events.

a. A cable fire started in the HPCI Room and was promtly exinquished by the Fire Brigade. Damage to HPCI control cables suspected. (0.75)

b. During steady state operations, SJAE Discharge radiation monitor levels increased from 20 R/hr to 210 R/hr over 30 minutes. (0.75)

c. A total loss of Control Room annunciators occurred concurrent with a reactor scram. (0.75)

d. All Diesel Generators started and picked up emergency loads during a blizzard which caused a loss of all off-site power. Sustained wind speeds of 75 mph are indicated on OBC699. (0.75)

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

QUESTION 8.01 (2.50)

STATE whether a SAFETY LIMIT or a THERMAL LIMIT would be violated for EACH of the following operating conditions. If none, state NONE.

1. Core flow is 20%, thermal power is 33%, pressure is 755 psig. (0.50)
2. All relief valves open, reactor pressure is 1315 psig. (0.50)
3. A P-1 edit shows CMFLPD is 0.^9. (0.50)
4. MODE switch in run, all rods in, level is -150 in. (0.50)
5. Reactor pressure is 900 psig, core flow is 30%. MCPR is 1.03. (0.50)

QUESTION 8.02 (1.50)

For each of the following automatic protective functions, STATE the Tech Spec BASES for the function. Include any applicable Safety Limits.

a. Turbine Stop Valve Closure SCRAM (0.75)

b. Reactor Vessel Water Level-Low SCRAM (0.75)

QUESTION 8.03 (2.50)

a. In accordance with Administrative Procedure A-7, "Shift Operations", LIST the four (4) situations that require a Senior Licensed Operator or Licensed Operator to scram or shutdown the plant. (1.00)

b. Following an unscheduled shutdown of the plant:

1. WHOSE (by title) approval is required for a restart? (0.50)
2. WHO shall direct the return to power? (0.50)
3. An unlicensed Shift Technical Advisor wishes to operate the controls to withdraw rods prior to criticality under the direction of a the Chief Operator. Is he allowed to do this? WHY or WHY NOT? (0.50)

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

QUESTION 8.04 (3.00)

While on shift in the Control Room as the Shift Supervisor, you are reviewing the procedure to perform an operability surveillance following completion of maintenance on a valve in the RHR System.

a. LIST the four (4) requirements that must be met for the procedure to be considered valid for use. (1.00)

b. You have determined that only part of the RHR valve operability surveillance must be performed for post maintenance testing. Is a temporary change required per A-3, "Procedure for Temporary Changes to Approved Procedures" in order to perform the partial surveillance? (0.50)

c. Assuming a temporary change is required, WHAT requirements must be met to implement a temporary change before the procedure can be performed? (1.50)

QUESTION 8.05 (2.00)

In accordance with Administrative Procedure A-43, "Surveillance Testing Program":

a. LIST two (2) of the three (3) responsibilities of Shift Supervision in regards to Surveillance Testing (ST). For each responsibility, STATE whether or not it may be delegated to the Chief Operator (CO) or Assistant Control Operator (ACO) (1.00)

b. LIST two (2) responsibilities of Shift Supervision, if a ST performed on the shift failed and Technical Specification requirements cannot be met. (1.00)

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

QUESTION 8.06 (2.50)

In accordance with Administrative Procedure A-41, "Procedure for Control of Plant Equipment":

a. LIST the two (2) classes of equipment that are controlled by this procedure. (1.00)

b. WHOSE (by title) permission is required to remove equipment from service? (0.50)

c. STATE one instance where an independent verification is NOT required when returning equipment to service. (0.50)

d. A piece of equipment was released for surveillance testing at 0900 June 6, 1988. How long is permission granted to perform the surveillance? (0.50)

QUESTION 8.07 (2.50)

In accordance with Administrative Procedure A-31, "Procedure for Notification of NRC", STATE (YES/NO) whether the following events WOULD or WOULD NOT be a 1 hour reportable event to the NRC. Use the attached Emergency Plan Implementing Procedure EP-101 for classification of the event, if required.

a. Thermal Power of 3290 MWt (0.50)

b. A maintenance worker gets injured and contaminated while repairing a feedwater turbine and must be taken to the hospital. (0.50)

c. A spurious initiation of Core Spray at 100% power during testing of the system logic (0.50)

d. Operations resulting in a condition requiring a change in an approved procedure (0.50)

e. Operations resulting in an Unanalyzed Condition (0.50)

QUESTION 8.08 (1.00)

A step of a TRIP procedure violates a Technical Specification requirement. Should you perform the TRIP procedure step or comply with the Technical Specifications? EXPLAIN YOUR CHOICE. (1.00)

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

QUESTION 8.09 (3.00)

The plant is operating at 85% power. During a surveillance cycling test of the suppression chamber to drywell Vacuum Breakers, you are informed that the "A/B/C/D/Drywell VACUUM RELIEF VALVE OPEN" alarm will not clear, and the OPEN AND CLOSED valve position indication lights are lit for the "A" vacuum breaker.

a. Are Primary Containment integrity requirements met in accordance with Technical Specifications? (Consider your answer in terms of Tech. Spec. definitions) (1.00)

b. Can the plant continue to operate? If yes, under WHAT conditions? If not, WHY not? (2.00)

NOTE: USE THE ATTACHED SECTIONS OF THE TECHNICAL SPECIFICATIONS (TS) TO ANSWER THIS QUESTION. FULLY REFERENCE ALL APPLICABLE SECTIONS OF THE TS THAT YOU USE TO DEVELOP YOUR ANSWER.

QUESTION 8.10 (2.50)

When you assume the midnight to eight A.M. shift, the plant is at 85% power and all conditions are normal with the following exceptions:

- APRM channel 'A' is bypassed for maintenance

- APRM channel 'B' is failed low and bypassed.

Two hours into the shift, IRM channel 'C' fails downscale.

In accordance with the Technical Specifications:
- Are the applicable LCO's satisfied ? (1.00)
- Can the plant continue to operate ? (0.50)
- If so, under what conditions ? If not, why not ? (1.00)

*** Justify your answers ***

NOTE: USE THE ATTACHED SECTIONS OF THE TECHNICAL SPECIFICATIONS (TS) TO ANSWER THIS QUESTION. FULLY REFERENCE ALL APPLICABLE SECTIONS OF THE TS THAT YOU USE TO DEVELOP YOUR ANSWER.

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

QUESTION 8.11 (2.00)

Limerick Unit 1 is in COLD SHUTDOWN. A unit startup is scheduled to commence on your shift. The Maintenance Supervisor reports the failure of the outboard blower for the MSIV Leakage Control System. He estimates that it will take two (2) days to return the blower to service.

In view of the above malfunction, CAN you proceed with the startup? If you can, under WHAT conditions? If you cannot, WHY not? (2.00)

NOTE: USE THE ATTACHED SECTIONS OF THE TECHNICAL SPECIFICATIONS (TS) TO ANSWER THIS QUESTION. FULLY REFERENCE ALL APPLICABLE SECTIONS OF THE TS THAT YOU USE TO DEVELOP YOUR ANSWER.

(***** END OF CATEGORY 08 *****)

(************* END OF EXAMINATION ***************)

ANSWER 5.01 (2.50)

a. Doppler or fuel temperature

b. Void

c. Moderator temperature

d. Void

e. Moderator temperature

(0.5 each) (2.5)

REFERENCE
LOT-1440, p. 1, LOs 1, 2 & 3
K/As 292004: K1.01 (3.2/3.2), K1.05 (2.9/2.9), K1.10 (3.2/3.2), & K1.14 (3.3/3.3)
292004K101 292004K105 292004K110 292004K114 ...(KA'S)

ANSWER 5.02 (2.25)

a. Increase [+0.25]. Higher steam flow results in a higher pressure drop [+0.50].

b. decrease [+0.25]. Higher heat input to condenser with same cooling [0.50]

c. Increase [+0.25]. Extraction steam energy to feedwater heaters increases faster than feed flow [+0.50].

REFERENCE
LOT-1270, LO 7; p. 10-13
LOT-1190, LO 3; p. 7-21
K/As 293006 K1.03 (2.4/2.5), 293005 K1.05 (2.7/2.8) & 292008 K1.20 (3.3/3.4)
292008K120 293005K105 293006K103 ...(KA'S)

ANSWER 5.03 (3.00)

a. The decrease in the burnout term [0.5] with the production of xenon from iodine still at the higher power rate dominates [0.5] causing the xenon concentration to increase. (1.0)

b. Peripheral rod worth will increase [0.5] because the highest xenon concentration will be in the center of the core where the highest flux existed previously [0.5]. This will suppress the flux in the center of the core and increase the flux in the area of the peripheral rods, thereby, increasing their worth [0.5]. (1.5)

c. More than half the value at 100%. (0.5)

REFERENCE
LOT-1510, LOs 3, 5 & 6; p. 6-10
LOT-1490, LO 4; p. 10
K/As 292006 K1.11 (2.6/2.7); 292005 K1.09 (2.5/2.6)
292005K109 292006K111 ...(KA'S)

ANSWER 5.04 (2.00)

a. Larger
b. Smaller
c. Larger
d. Smaller (0.50 Each)

REFERENCE
LOT-1490 LO 4; p. 4-12
KA 292005 K1.12 (2.6/2.9)
292005K112 ...(KA'S)

ANSWER 5.05 (2.00)

0 psig = 14.7 psia = 212 deg F (0.50)

1000 psig = 1014.7 psia = 546.3 deg F (0.50)

GP-3 Limit= 100 deg F/hr (0.50)

546.3 - 212 = 334.3 deg F (0.25)

334.3 deg F / 100 deg F/hr = 3.3 hours (0.25)

REFERENCE
LOT-1160, LO 1; p. 4-6
Steam Tables
GP-3, p. 12
K/A 293003 K1.23 (2.8/3.1)
293003K123 ...(KA'S)

ANSWER 5.06 (2.25)

a. Remains constant (0.25). Flow is controlled by the HPCI flow controller which will attempt to maintain a constant output flow regardless of reactor pressure (0.50).

b. Decreases (0.25). The flow controller functions to maintain a constant flow, thus pump discharge pressure is decreased along with the decreasing reactor pressure to maintain constant flow. OR Since the flow controller maintains a constant flow to the reactor, as reactor pressure decreases, the pump discharge head must decrease to maintain a constant flow. (0.50).

c. Decreases (0.25). To maintain a constant flow, turbine RPM must also decrease (0.50).

REFERENCE
LOT-1290, LO 5; p. 13
LOT-0340, LO 5; p. 6
K/As 291004 K1.05 (2.8/2.9); 293006 K1.08 (2.5/2.6)
291004K105 293006K108 ...(KA'S)

ANSWER 5.07 (2.00)

If power were reduced from a high power (>90%) condition (0.5) by the insertion of shallow rods (rods below the core midplane) (0.5), the resulting positive reactivity from local decrease in void formation in the lower portion of the control cell (0.5) would more than offset the negative reactivity of the low-worth rod insertion (0.5).

REFERENCE
LOT-1490, LO 9; p. 5-13
K/A 292005 K1.12 (2.6/2.9)
292005K112 ...(KA'S)

ANSWER 5.08 (2.00)

a. FALSE
b. FALSE
c. TRUE
d. ~~TRUE~~ FALSE
(0.50 Each)

REFERENCE
LOT-1300, LO 2; p. 4-6
K/A 293007 K1.13 (2.3/2.9)
293007K113 ...(KA'S)

ANSWER 5.09 (2.00)

a. MAPRAT = MAPLHGR actual/MAPLHGR-LCO ~~(1.00)~~ (0.50)
(MAPRAT is the Maximum Average Planar Ratio. It is the comparison between the actual APLHGR to the MAPLHGR limit as programmed into the Process Computer.)

b. Yes (0.25), we have exceeded the Tech Spec limit for APLHGR (0.50). The value of MAPRAT should never be greater than 1.0 (0.25).

c. The clad temperature can exceed 2200 F during a DB LOCA (0.50)

REFERENCE
LOT-1410, LOs 2-4; p. 4
TS 3/4.2.1, Average Planar Linear Heat Generation Rate
K/As 293009 K1.10 (3.3/3.7) & K1.13 (3.1/3.6)
293009K113 ...(KA'S)

ANSWER 5.10 (3.00)

a. Reactor water level decreases because the feedwater pumps are no longer feeding the reactor and steam is still being removed (0.25).
b. The level continues to decrease following the scram because of void collapse (0.25) and because steam is being removed via BPVs and SRVs for decay heat removal. (0.25)
c. Rx pressure decreases as reactor power decreases (0.25). Power decreases initially due to the decrease in inlet subcooling (0.25) and decreases at a faster rate when the reactor scrams (0.25).
d. After the MSIVs close decay heat causes the pressure to increase (0.375) until the relief valves open (0.375)
e. Core inlet flow decreases slowly due to loss of feedwater (0.375) The rate of decrease increases due to the trip of the recirculation pumps (on low level) (0.375)

REFERENCE
LOT-0040, LOs 3, 4; p. 6-10
LOT-0550, LOs 6, 7; p.8-21
LOT-0580, LO 4; 17-19
K/As 295001 K3.01 (3.9/3.9)
295001 K3.12 (3.8/3.9)
295001 K4.11 (3.5/3.5)
259001K301 259001K312 259001K411 ...(KA'S)

ANSWER 5.11 (2.00)

a. 2 (1.00) or 1
b. 3 (1.00)

REFERENCE
LOT-1260, p. 16 - 18, LO 4
K/As 293006: K1.05 (3.2/3.3), K1.13 (2.6/2.7)
293006K105 293006K113 ...(KA'S)

ANSWER   6.01   (1.00)

d. (1.0)

REFERENCE
LOT-340, Objective 9
KA 206000 K4.01 (3.8/3.9) 206000 K4.03 (4.2/4.1)
206000K401   206000K403   ...(KA'S)

ANSWER   6.02   (2.50)

a. after the 420 second timer times out (0.5)
b. Low level (12.5") signal is a confirmatory signal that prevents inadvertant initiation following a single instrument failure (0.50)
c. Blowdown will continue. (0.5) (If both DIV I and II were reset, blowdown would be interrupted.)
d. Blowdown will continue. (0.5) (These switches do not terminate blowdown once initiated)
e. The valve has opened (0.25) but is presently closed (0.25)

REFERENCE
LOT-0330, LO 2, 5 & 6; p. 9-13 & Figure 6
K/As 218000: K4.01 (3.7/3.9), K4.02 (3.8/4.0), K4.03 (3.8/4.0)
K5.01 (3.8/3.8), A3.03 (3.7/3.8)
21800K403   ...(KA'S)

ANSWER   6.03   (2.00)

a. Decrease (0.5); <20% feedwater flow places the 28% speed limiter in service (0.25) reducing speed to 28% (0.25).
(or no effect (1.00))
b. Decrease (0.5); Scoop Tube Positioner Stops (Electrical or Mechanical) (0.25) limit the decrease to 20% (0.25)

REFERENCE
LOT-0040, LOs 11 & 12; p. 5, 6, 10 & 11
K/A 202002 SG.09 (3.8/3.5)
202002G009   ...(KA'S)

ANSWER 6.04 (3.00)

a. No APRM downscale (0.5)
~~50~~ 118 second time delay (0.5)

b. All SBLC pumps start (0.5)
All squib valves open (0.5) or continuity lights extinguish
RWCU isolates (0.5)
10 minute reset timer actuates. (0.5)
Also accept: decreasing tank level or power decreasing

REFERENCE
LGS: LOT-0310 p. 16 and 17.
Learning objectives 9, 10, and 11.
K/A 211000 A3.08 (4.2/4.2)
211000A308 ...(KA'S)

ANSWER 6.05 [illegible])

a. High Steam Line flow (0.2~~5~~)
Low reactor vessel level (0.2~~5~~)
MSL High temperature (Tunnel or Turbine enclosure) (0.2~~5~~)
MSL High Radiation (0.2~~5~~)
Manual (0.2)

b. Fast enough to:
- Limit coolant loss during SLB outside containment
- Limit radioactive release during gross fuel failure
- Limit radioactive release during a major leak inside containment
(Any 2, @ 0.25 each)
Slow enough to:
- Minimize damage to valve and piping
- minimize transient in boiler
(0.25 each)

c. FALSE (0.5)

REFERENCE
LOT-0120, LO 2.f & 9; p. 13 & 18
TS BASES 3/4.4.7
K/As 223002 SG.06 (2.9/3.9); 239001 K1.27 (4.0/4.1) & K4.01 (3.8/3.8)
223002G006 239001K127 239001K401 ...(KA'S)

ANSWER 6.06 (3.00)

a. 1. RHR pump 1A (0.4)
   2. Load Center Bus/Transformer D114 (0.4)
   3. CS pump 1A (0.4)
   4. ESW pump 0A (0.4)
   (0.4 for correct sequence)

b. 1. RHRSW pump 0A (0.25)
   2. TB Equip. Compartment Exhaust Fan 1A (0.25)

c. Manual reset (0.5)

REFERENCE
LOT-0660, LO 6; p. 14
K/As 264000 K3.03 (4.1/4.2), K4.05 (3.2/3.5)
264000K303 264000K405 ...(KA'S)

ANSWER 6.07 (2.50)

a. Refuel floor supply and exhaust isolate. (0.25)
   SBGTS starts if aligned to RFVS via slide gate dampers (0.25)

b. Supply and exhaust ventilation isolate. (0.5)
   Standby gas treatment starts. (0.5)
   Reactor enclosure recirculation starts (0.5)

c. Trips RHR service water pumps. (0.5)

REFERENCE
LGS: LOT-0720 p. 10, 11, 18. Learning Objective 2.
LGS: LOT-0180 p. 23. Learning Objective 2.
KAs 272000: K1.01 (3.6/3.8), K1.06 (3.2/3.3), K1.08 (3.6/3.9),
K1.09 (3.6/3.8), K4.02 (3.7/4.1)
272000K101 272000K106 272000K108 272000K109 272000K402
...(KA'S)

ANSWER 6.08 (3.00)

Output from pressure summer A increases to ~~40~~ 25 psi (0.5). The HVG passes this signal to the Pressure/Flow converter (0.5). The Pressure/Flow converter output will increase to ~~83~~ 82% (0.5). The load limit will limit the signal to the control valves to 65% (0.5). Total flow is less than the maximum combined limiter (0.5) so the bypass valves will open to pass ~~17~~ 18% flow (0.5).

REFERENCE
LGS: LOT-0590 p. 6, 7, 8, 13, 15, 16.
Learning Objectives 3, 8.
KAs 241000: K3.05 (3.7/3.7), K3.06 (4.1/4.1) & K3.08 (3.7/3.7)
241000K305 241000K306 241000K308 ...(KA'S)

ANSWER 6.09 (2.00)

a. Causes reactor level to increase (0.5) due to the level control system having a level error, (level set > indicated level) (0.25) resulting in an increase in the speed of the reactor feed pump turbines (0.25).

b. Reactor level decreases (0.5) because the RFP turbine "B" decelerates to low speed setting (0.5). (The level decrease causes a flow error which causes the master controller to raise the speed of turbines "A" & "C". The level will stabilize at the original setpoint.)

REFERENCE
LOT-0550, LO 7; p. 18-20
K/As 259002 K6.05 (3.5/3.5); K4.06 (3.1/3.2)
259002K406 259002K605 ...(KA'S)

ANSWER 6.10 (3.00)

a. 3 (0.5) or NONE
b. NONE (0.5)
c. 3 (0.5)
d. 2 (0.~~7~~5) (and 3) ~~(0.25)~~
e. 3 (0.5)
f. 1 (0.25) and 3 (0.25)

REFERENCE
LGS: LOT-0250 p. 10. Learning Objective 10.
LOT-0270 p. 11, 12, and 13. Learning Objective 7.
KAs 215003 K4.01 (3.7/3.7), 215005: K4.01 (3.7/3.7) & K4.02 (4.1/4.2)
215003K401 215005K401 215005K402 ...(KA'S)

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ANSWER 7.01 (2.00)

a. NO (0.50)

b. NO (0.50)

c. YES (0.50)

d. YES (0.50)

REFERENCE
LOT-1760, LO 5 & p. 21, 22
K/A 294001 K1.03 (3.3/3.8)
294001K103 ...(KA'S)

ANSWER 7.02 (2.00)

a. 4500 milli-Rem/year (0.50)

b. (5(N-18) = 5(30-18)) = 60 Rem ~~(0.50 for formula & 0.50 for calculation)~~ (1.00)

c. 25 Rems (0.50)

REFERENCE
LOT-1760, LOs 1, 2, SRO-2 & p. 4,5
K/A 294001 K1.03 (3.3/3.8)
294001K103 ...(KA'S)

ANSWER 7.03 (2.00)

a.1. (Immediate) SCRAM (0.5)
a.2 To prevent damage to the pump internals (0.5)
b.1. To ensure adequate core flow coastdown on a LOCA (0.5)
b.2 Because there is a potential for the loss of the master level control signal. (0.5)

REFERENCE
LOT-1530, LO 2
GP-5 p. 4, 5; GP-3 p. 4, 5
T/S B3/4 4-1
K/As 201004 G.1 (3.9/4.1), 202001 G.10 (3.5/3.7) & G.6 (3.0/4.1), 259002 G.10 (3.3/3.4)
201004G001 202001G006 202001G010 259002G010 ...(KA'S)

ANSWER 7.04 (2.50)

a. Ensure all flow is to the vessel ~~(0.5)~~ and prevent inadvertant draining of the vessel ~~(0.5)~~ (1.00)

b. Prevent pump overheating (0.5)

c. Prevent damage to the heat exchanger (0.5)

d. Ensure proper natural circulation (0.5)

REFERENCE
LGS: Shutdown Cooling Operation (S51.8.b) p. 3 and 5.
KA 205000 G.10 (3.2/3.3) 205000 K1.02 (3.6/3.6)
205000G010 205000K102 ...(KA'S)

ANSWER 7.05 (3.00)

a. 1. NO (0.50)
2. YES (0.50)
3. NO (0.50)

b. 1. Scram the reactor (0.50)
2. Trip the main turbine (0.50)
3. Close the MSIVs (0.50)

REFERENCE
LOT-1563, LOs 1, 2 & p. 3
SE-1, "Remote Shutdown", p. 1 and 2
K/As 295016 SG.11 (4.1/4.2) & SG.10 (3.8/3.6)
295016G010 295016G011 ...(KA'S)

djP.

ANSWER 7.06 3.00 ~~(2.50)~~

a. (OT-100, Reactor Low Level ~~(0.5)~~ ~~and~~ OT-110, Reactor High Level (1.0) ~~(0.5)~~

b. ~~Runback recirc. flow (0.30)~~
(If a RFP controller malfunction exists,) take manual control of the RFP(s) to control level ~~(0.40)~~ Reduce flow until normal level is restored (0.5)
If a scram condition occurs, enter Procedure T-100 ~~(0.30)~~ (0.5)

c. Scram: Turbine Stop Valve Closure Scram is bypassed when power is less than 30% (as sensed by the first stage shell pressure) so, a turbine trip (@ +54") would not have scrammed the reactor (0.25) and a scram is necessary to limit the heat input to the suppression pool (0.25)
Close MSIVs: To protect aginst unnecessary flooding of MSLs (MSL supports downstream of outboard MSIVs are not designed for flooding) (0.5)

REFERENCE
LOT-1540 LOs 1, 2, 3 & p. 3, 4, 25, 26
OT-100, "Reactor Low Level", p. 1
OT-110, "Reactor High Level", p. 1, 2, 3
K/A 295031 SG.10 (4.0/3.8) & 295008 AK3.01 (3.4/3.5)
295031G010 ...(KA'S)

ANSWER 7.07 (2.50)

a. None (0.5)
b. ON-104, "Control Rod Problems" (0.5)
c. ON-107, "Control Rod Drive System Problems (0.5)
d. ~~None (0.5)~~ ON-109, "Total Loss of SRM, IRM or APRM Systems" (0.5)
e. ON-111, "Loss of Secondary Containment" (0.50)

REFERENCE
LOT-1550 LO 1 & p. 3-20
ON-100, "Failure of a Jet Pump", p. 1
ON-104, "Control Rod Problems", p. 1
ON-107, "Control Rod Drive System Problems", p. 1
ON-109, "Total Loss of SRM, IRM, or APRM Systems", p. 1
ON-111, "Loss of Secondary Containment", p. 1
K/As 202001 K3.03 (3.9/3.9)
201003 A2.02 (3.7/3.8)
201003 A2.08 (3.8/3.7)
215004 K3.04 (3.7/3.7)

ANSWER 7.08 (3.00)

a. T-102, "Primary Containment Control" (0.50)
b. T-100, "Scram" (0.50)
c. T-101, "RPV Control" (0.50) (T-100, "Scram")
d. T-100, "Scram" (0.~~2~~5)
e. T-101, "RPV Control" (0.25) and T-102, "Primary Containment Control" (0.25) (T-100, "Scram")
f. T-101, "RPV Control" (0.50) (T-100, "Scram")

REFERENCE
LOT-1560, LO 3 & p. 7
K/As 295006 SG.11 (4.3/4.5), 295024 SG.11 (4.3/4.5), 295026 SG.11 (4.2/4.5)
295028 SG.11 (4.2/4.4), 295029 SG.11 (4.2/4.5), 295030 SG.11 (4.3/4.5)
295037 SG.11 (4.4/4.7)
295006G011 295024G011 295028G011 295029G011 295030G011
295031G011 295037G011 ...(KA'S)

ANSWER 7.09 (2.50)

a. insufficient steam flow for adequate core cooling exists with only one SRV open below 700 psig (0.50)
b. Five SRVs (0.50) or three SRVs
c. One of the Core Spray systems (0.50). Spray heat transfer is preferred because the core temperature is elevated during steam cooling (to provide the temperature differential so steam can carry away core heat) (0.50). Core Spray spargers (system) can safely lower core temperature with a reduced possibility of core damage (0.50).

REFERENCE
LOT-1560 LO 5 & p. 11, 12
K/A 295031 SG.12 (3.9/4.5)
295031SG12 ...(KA'S)

ANSWER 7.10 (3.00)

a. Alert (0.75)
b. Alert (0.75) or Unusual Event (0.65)
c. Site Emergency (0.75)
d. Alert (0.75)

REFERENCE
LOT 1520, LO SRO-1, p. 5-7
EP-101, p. 2, 8, 10, 13, 14

K/A 294001 A1.16 (2.9/4.7)
294001A116 ...(KA'S)

ANSWER 8.01 (2.50)

1. Safety ~~Thermal~~ Limit (0.5)
2. NONE (0.5)
3. NONE (0.5)
4. NONE (0.5)
5. Safety Limit (0.5)

REFERENCE
T/S 2.0, 3.0
LOT-1820 Objective 2, LOT-1840 Objective 3.b

ANSWER 8.02 (1.50)

a. Anticipates the pressure, neutron flux and heat flux increase which result from the closure of the stop valves. (0.4 ~~5~~) ~~Protects the reactor vessel pressure and fuel thermal/hydraulic safety limits. (0.25)~~
Ensures adequate thermal margins are maintained (0.35)

b. ~~Used in transient analysis dealing with coolant inventory decrease. (0.5) Protects the fuel thermal/hydraulic and vessel pressure safety limits.~~ ~~(0.25)~~
Far enough below normal operating level to avoid spurious trips (0.35) but high enough above the fuel to assure adequate protection for fuel and pressure limits (0.40)

REFERENCE
T/S BASES Sec. 2.0, LOT-1830 Objective 4
KA 212000 G.6 (3.4/4.3)
212000G006 ...(KA'S)

ANSWER 8.03 (2.50)

a. 1. Safety hazards indicated
   2. Doubt of safe condition
   3. RPS parameters exceeded without a scram
   4. By procedure direction (0.25 Each)

b. 1. Station Superintendent or Plant Manager or designated alternate (0.5)
   2. Senior Licensed Operator (SLO) (0.50 each) or Shift Supervision
   3. No. Only licensed operators or unlicensed operators in a licensed operator training program may operate the controls (0.5)

REFERENCE
LOT-1570, LOs 2, 3, SRO-1, 2
A-7, "Shift Operations", p. 11, 15, & 16
10 CFR 55.13(a)(2)
K/As 294001 A1.09 (3.2/4.3) & A1.11 (3.3/4.3)

ANSWERS -- LIMERICK 1 -88/06/07-NRC REGION I

294001A109 294001A111 ...(KA'S)

ANSWER 8.04 (3.00)

a. 1. Must be stamped in red, "Controlled Copy" (0.25)
   2. Station Supt. or Alternate signature and date (0.25)
   3. QA Supt. or Alternate signature and date (0.25)
   4. Date of use must be later than effective date (0.25)

b. No. (0.5)

c. Determine that the change does not change the intent of the procedure (0.25)
   Document the change (0.25)
   Have the change approved by the Shift Superintendent (0.5) and an individual knowledgeable in the areas affected by the procedure (0.25) who is a member of PORC or previosly designated (0.25)

REFERENCE
LOT-1570, LOs 1 & 2; p. 5, 6 & 16
A-3, "Procedure for Temporary Changes to Approved Procedures", p. 3
Technical Specifications 6.8.3
K/As 294001: A1.01 (2.9/3.4) & A1.02 (4.2/4.2)
294001A101 294001A102 ...(KA'S)

ANSWER 8.05 (2.00)

a. 1. Permission to perform ST - may not be delegated
   2. Signs off completed ST - may be delegated
   3. Comply with Tech. Spec., (if ST failed) - may NOT be delegated
   (Any 2 @ 0.50 Each - 0.3 for responsibility, 0.2 for delegation)

b. 1. Determine system operability
   2. Determine if LCOs satisfied
   3. Scheduling of any additional tests required
   4. Notify agencies required
   5. Notify Plant Staff (Any 2, 0.50 Each)

REFERENCE
LOT-1570, LOs 3, SRO-1, 2; p. 48
A-43, p. 2, 7
K/A 294001 A1.03 (2.7/3.7)
294001A103 ...(KA'S)

ANSWER 8.06 (2.50)

a. 1. Safety-Related equipment (0.50) (Q List)
   2. Other equipment required to be operable or in surveillance by Tech. Spec. (TS equipment) (0.50)

b. Shift Supervision (0.50)

c. High radiation areas (0.50) or if check off list will be performed

d. 0800 June 7, 1988 (0.50)

REFERENCE
LOT-1570, LOs 2, 3, SRO-1 & p. 44
A-41, "Procedure for Control of Plant Equipment", p. 1, 6, 7, 8, 9, 12
K/A 294001 K1.02 (3.9/4.5)
294001K102 ...(KA'S)

ANSWER 8.07 (2.50)

a. no (0.50)

b. yes (0.50) (Unusual Event declared)

c. no (0.50)

d. No (0.50)

e. yes (0.50)

REFERENCE
LOT-1570, LOs 3, SRO-1, 2 & p. 39
A-31, Attach. 2
K/A 294001 A1.03 (2.7/3.7)
294001A103 ...(KA'S)

ANSWER 8.08 (1.00)

Follow TRIP (0.5) TRIPs are the governing document. (0.5)

REFERENCE
LOT-1560 Intro To TRIPS, Objective #1, SRO1
KA 295025 G.7 (3.5/3.7) 295025 G.12 (3.9/4.5)
295025G007 295025G012 ...(KA'S)

ANSWER 8.09 (3.00)

a. Containment integrity requirements are met. (1.0)

b. Yes (0.5)

T/S 3/4.6.4 ACTION b. applies. (1.0)

Conditions: (ACTION b.) the other valve in the pair must be verified closed within 2 hrs. and the open valve must be restored to closed within 72 hrs or be in hot shutdown within the next 12 hrs and cold S/D within the following 24 hrs. (0.5)

(The valve cannot be verified closed. Thus, the more conservative approach is to assume that the valve is open. Until some means other than the position indication shows the actual valve position, the position indication must be used since there is no information given to place the indication in doubt.)

REFERENCE
Technical Specifications 3/4.6.4
LOT-1840 Objective 3.b, 3.c
KA 223001 G.11 (3.3/4.2) 223001 K6.09 (3.4/3.6)
223001G011 223001K609 ...(KA'S)

ANSWER 8.10 (2.50)

TS, Table 3.3.1-1, requires that there shall be a minimum of two (2) operable channels per trip system. In this situation, RPS Channel A does not meet the specification (1.0) (for downscale trips).

(The ACTION statement requires that you be in STARTUP within 6 hours. BUT) - LCO 3.3.1 allows that if the minimum number of operable channels is not met in one channel, then that channel may be placed in the tripped condition within one hour.
THEREFORE - in this condition the plant may continue to operate (0.5) with a half-scram inserted on RPS Channel 'A'.(1.0)

REFERENCE
LOT-1840 Objective 3.b, 3.c
TS 3/4.3.1
K/A 215005 SG.11 (3.4/4.1)
215005G011 ...(KA'S)

ANSWER 8.11 (2.00)

NO (0.5). (A reactor startup would represent a violation of the Tech Specs.)
TS 3.6.1.4 requires that two MSIV LCS be operable in Op Con 1-2-3.
The action statement for TS 3.6.1.4 allows 30 days of continued operation, (0.75)
but Tech Spec 3.0.4 does not allow entry into an Operational Condition while relying on an action statement. (0.75)

REFERENCE
Technical Specifications 3/4.6.1.4, 3.0.4
LOT-1840 Objective 3.b, 3.c
K/A 223001 SG.11 (3.3/4.2)
223001SG11 ...(KA'S)

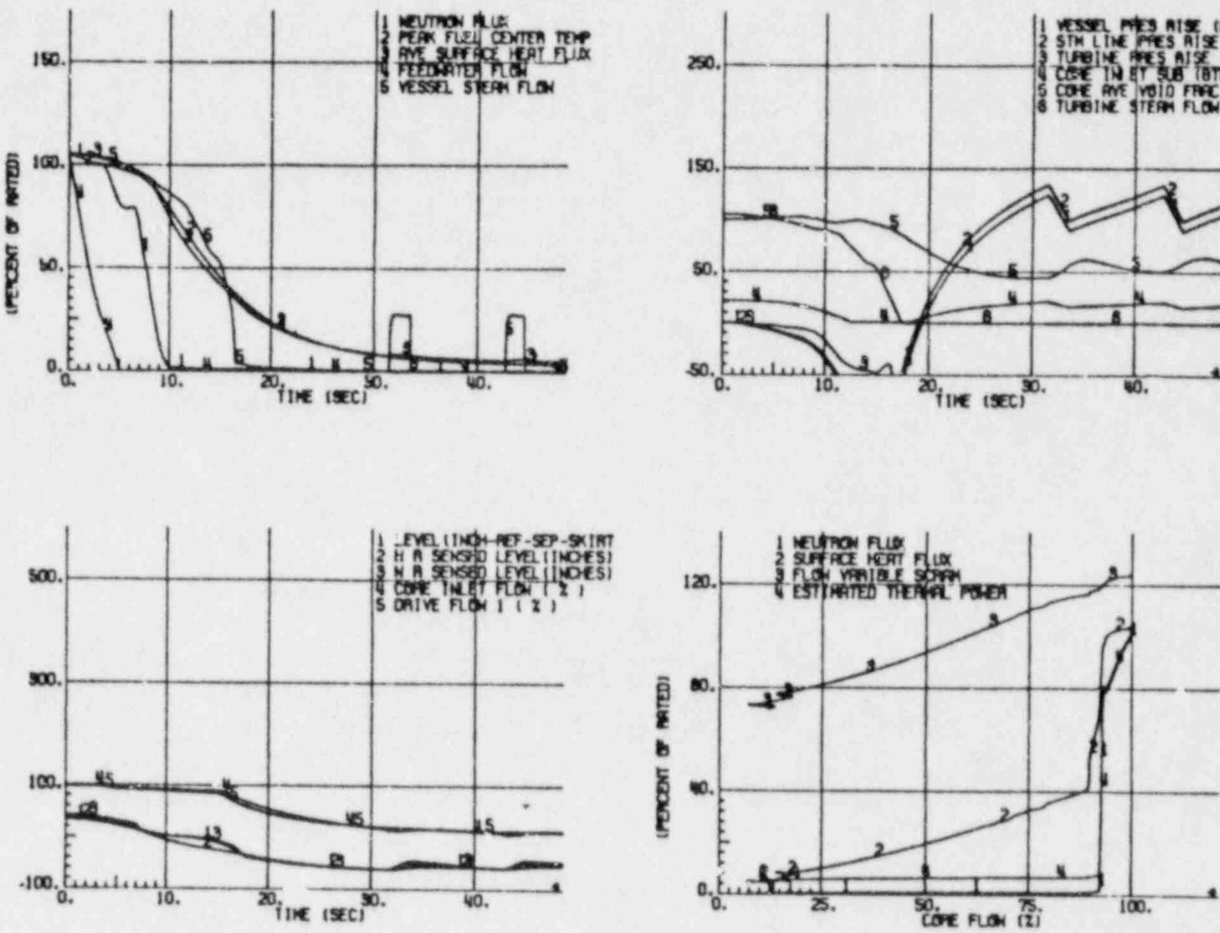
QUESTION	VALUE	REFERENCE
05.01	2.50	SVP0000435
05.02	2.25	SVP0000438
05.03	3.00	SVP0000441
05.04	2.00	SVP0000439
05.05	2.00	SVP0000444
05.06	2.25	SVP0000445
05.07	2.00	SVP0000440
05.08	2.00	SVP0000442
05.09	2.00	SVP0000443
05.10	3.00	SVP0000437
05.11	2.00	SVP0000436
	25.00	
06.01	1.00	SVP0000476
06.02	2.50	SVP0000456
06.03	2.00	SVP0000457
06.04	3.00	SVP0000458
06.05	~~3.00~~ 2.5	SVP0000464
06.06	3.00	SVP0000459
06.07	2.50	SVP0000462
06.08	3.00	SVP0000463
06.09	2.00	SVP0000461
06.10	3.00	SVP0000460
	~~25.00~~ 24.5	
07.01	2.00	SVP0000455
07.02	2.00	SVP0000454
07.03	2.00	SVP0000447
07.04	2.50	SVP0000446
07.05	3.00	SVP0000450
07.06	~~2.50~~ 3.0	SVP0000448
07.07	2.50	SVP0000449
07.08	3.00	SVP0000451
07.09	2.50	SVP0000452
07.10	3.00	SVP0000453
	~~25.00~~ 25.5	
08.01	2.50	SVP0000472
08.02	1.50	SVP0000475
08.03	2.50	SVP0000466
08.04	3.00	SVP0000465
08.05	2.00	SVP0000469
08.06	2.50	SVP0000468
08.07	2.50	SVP0000467
08.08	1.00	SVP0000471
08.09	3.00	SVP0000470
08.10	2.50	SVP0000474

QUESTION	VALUE	REFERENCE
08.11	2.00	SVP0000473

	25.00	

	100.00	

DOCKET NO 352

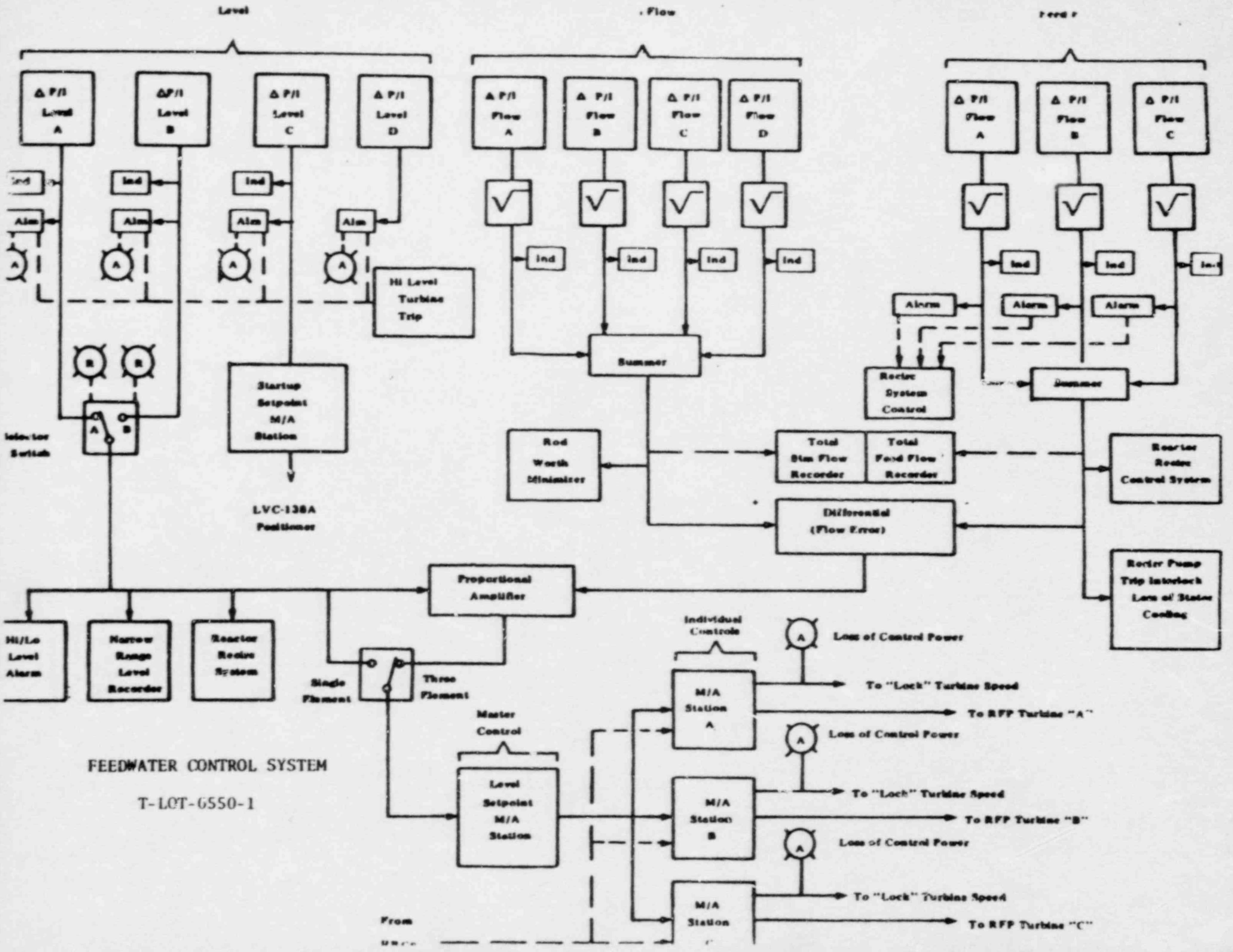


SEE NOTE ON TABLE 15.2-11

LIMERICK GENERATING STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

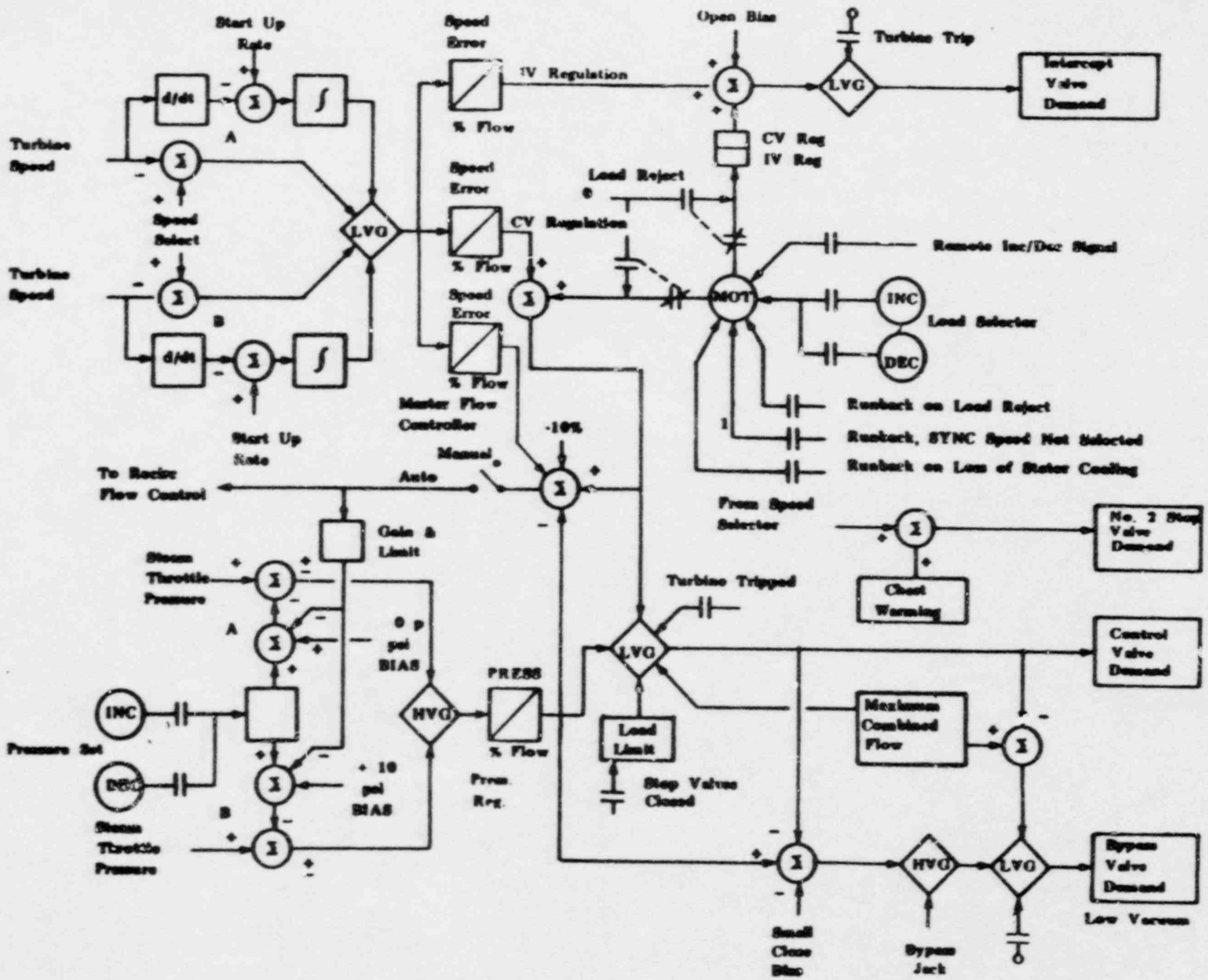
LOSS OF ALL FEEDWATER FLOW

FIGURE 15.2-9 REV. 32, 05/84



FEEDWATER CONTROL SYSTEM

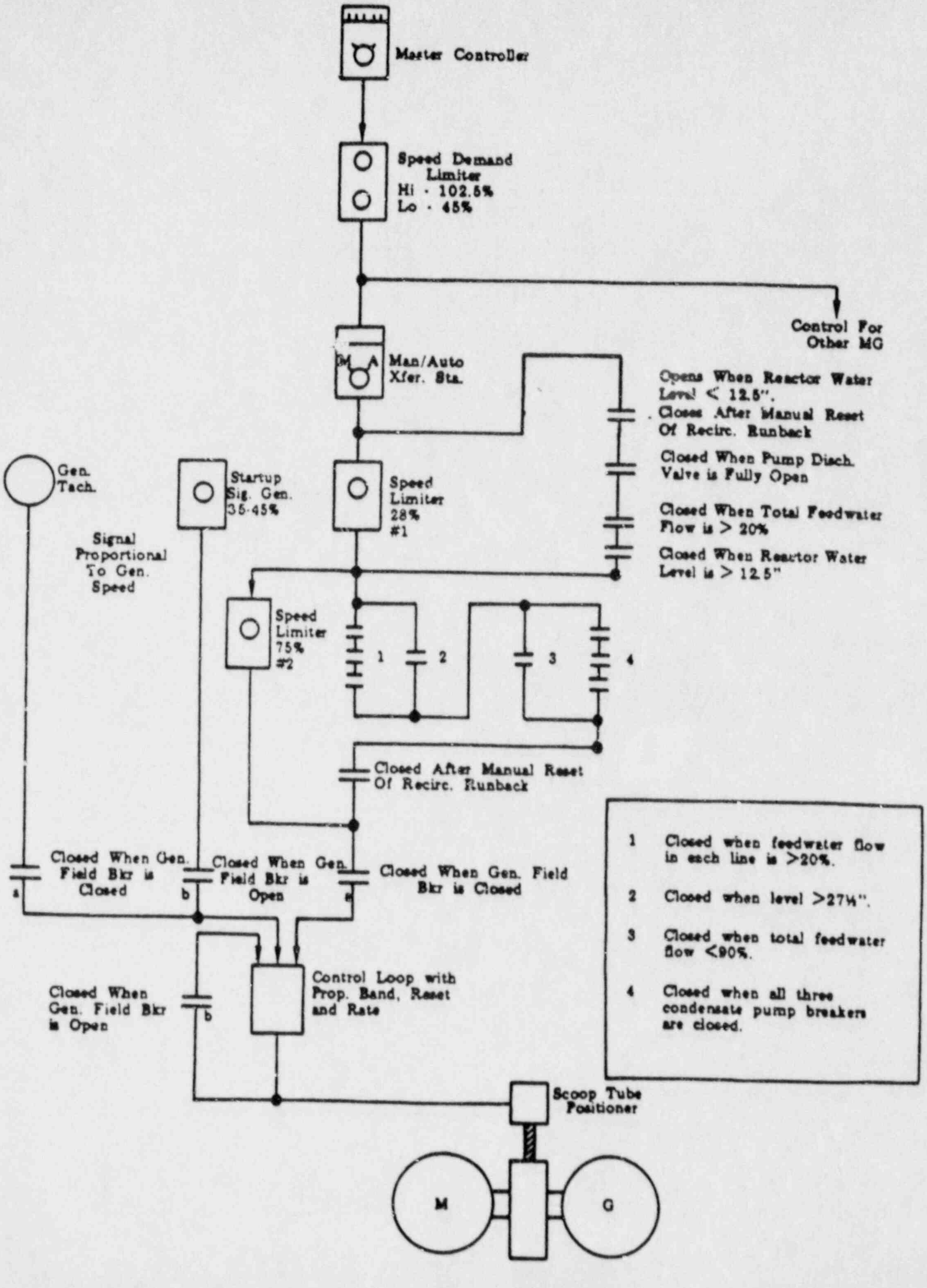
T-LOT-6550-1

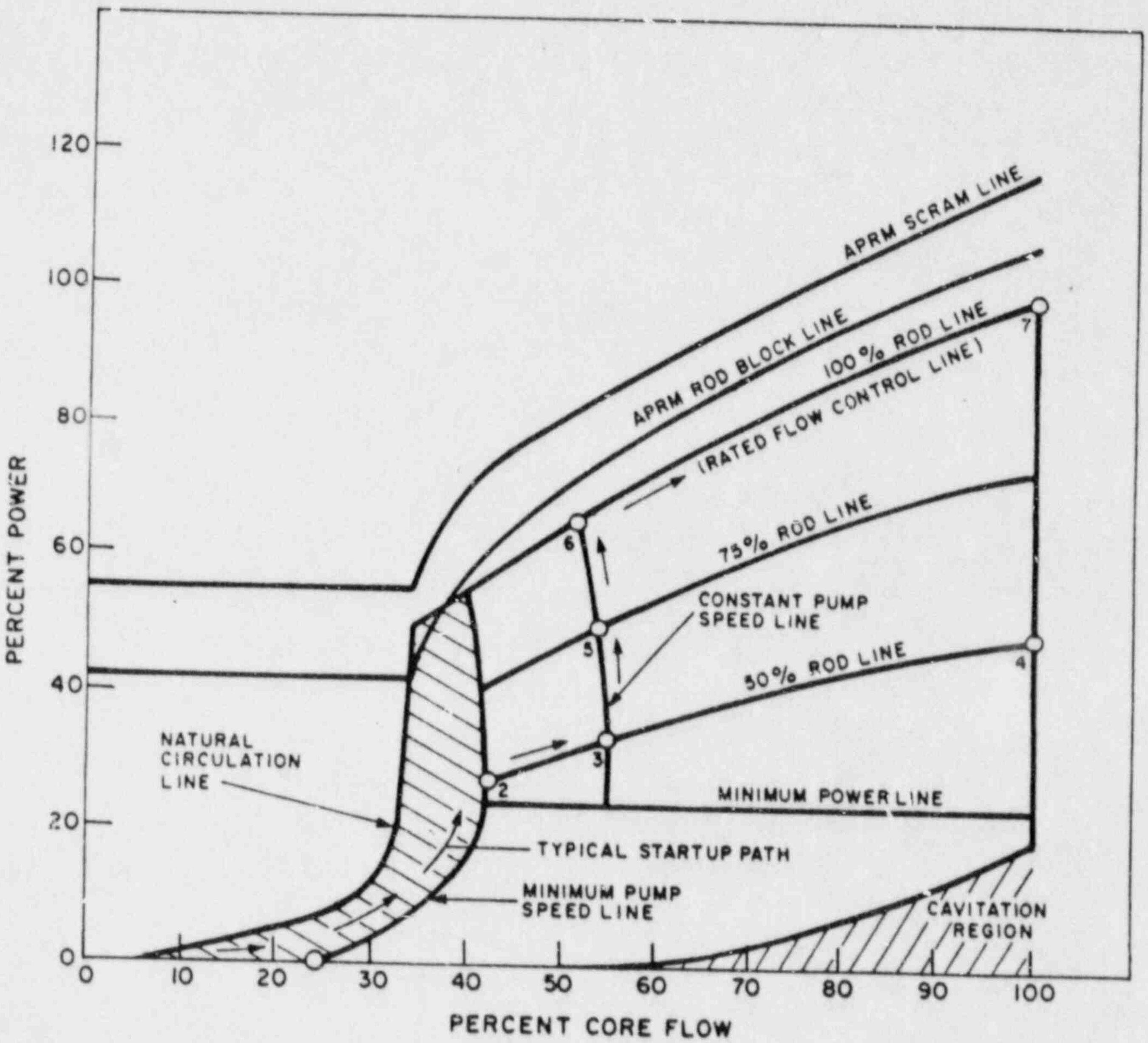


ELECTRO-HYDRAULIC CONTROL LOGIC

T-LOT-0590-6

2887


Master Controller
Speed Demand Limiter
Hi - 102.5%
Lo - 45%
Control For Other MG
Man/Auto Xfer. Sta.
M
A
Opens When Reactor Water Level < 12.5".
Closes After Manual Reset Of Recirc. Runback
Closed When Pump Disch. Valve is Fully Open
Closed When Total Feedwater Flow is > 20%
Closed When Reactor Water Level is > 12.5"
Gen. Tach.
Startup Sig. Gen. 35-45%
Speed Limiter 28% #1
Signal Proportional To Gen. Speed
Speed Limiter 75% #2
1
2
3
4
Closed After Manual Reset Of Recirc. Runback
Closed When Gen. Field Bkr is Closed
a
Closed When Gen. Field Bkr is Open
b
Closed When Gen. Field Bkr is Closed
a
Closed When Gen. Field Bkr is Open
b
Control Loop with Prop. Band, Reset and Rate
1 Closed when feedwater flow in each line is >20%.
2 Closed when level >27¼".
3 Closed when total feedwater flow <90%.
4 Closed when all three condensate pump breakers are closed.
Scoop Tube Positioner
M
G



OPERATING MAP

T-LOT-0040-8

LIMERICK GENERATING STATION

OFF-NORMAL PROCEDURES INDEX

PROCEDURE NUMBER	REV. NO.	TITLE	DATE SIGNED BY SUPER.	DATE OF LAST PERIODIC REVIEW
ON-100	2	Failure of a Jet Pump	12/01/87	12/01/87
ON-101	2	Loss of Isolated Phase Bus Cooling	12/01/87	12/01/87
ON-102	4	Air Ejector Discharge High Radiation	02/25/88	02/25/88
ON-103	1	Control of Sustained Combustion in the Off-gas System	12/22/86	12/22/86
ON-104	2	Control Rod Problems	03/07/88	03/07/88
ON-105		Cancelled		
ON-106		Cancelled		
ON-107	4	Control Rod Drive System Problems	03/07/88	03/07/88
ON-108		Cancelled		
ON-109	1	Total Loss of the SRM, IRM, or APRM Systems	02/25/88	02/25/
ON-110	3	Loss of Primary Containment	03/02/88	03/02/88
ON-111	3	Loss of Secondary Containment	02/25/88	02/25/88
ON-113	4	Loss of RECW	02/25/88	02/25/
ON-114	3	Loss of Stator Water Cooling Runback	02/25/88	02/25
ON-115	4	Loss of Control Enclosure Cooling	03/02/88	03/02/88
ON-116	1	High Reactor Water Conductivity	02/19/88	12/30/8
ON	1	Loss of TECW	02/25/88	02/25/8
ON-118	0	North Stack High Radiation	12/31/87	12/31/8

Jfrom
10/28/87

PHILADELPHIA ELECTRIC COMPANY
LIMERICK GENERATING STATION
EMERGENCY PLAN IMPLEMENTING PROCEDURE

EP-101 CLASSIFICATION OF EMERGENCIES

1.0 PARTICIPANTS

1.1 Shift Superintendent or designated alternate shall assume the role of Emergency Director and implement this procedure, until relieved.

1.2 Plant Manager or designated alternate shall relieve the Emergency Director, assume the role of Emergency Director, and continue implementing this procedure, if necessary.

2.0 ACTIONS-IMMEDIATE

CAUTION:
IMPLEMENTATION OF THIS PROCEDURE DOES NOT CONSTITUTE IMPLEMENTATION OF THE EMERGENCY PLAN

CAUTION:
THE JUDGEMENT OF THE EMERGENCY DIRECTOR IS VITAL IN PROPER CONTROL OF AN EMERGENCY AND TAKES PRECEDENCE OVER GUIDANCE IN THIS EMERGENCY PLAN PROCEDURE

2.1 Emergency Director shall:

2.1.1.1 Select categories related to station events or conditions.

	Page Number
Hazards to Station Operation	6
Environmental	7
Loss of Power	8
Personnel Injury	9
Fire	10
Radioactive Release	11
Evacuation of Control Room	12
Damage of Fuel	13
Instrument Failure	14
Scram Failure	15
Boundary Degradation/LOCA	16
Unusual Shutdown	18
Loss of Hot or Cold Shutdown Capability	19
Security	20

2.1.2 Beginning at the indicated page in Appendix EP-101, review the Emergency Action Levels (EALS) for categories selected.

2.1.3 Classify the event based on the selected category and EALS.

IF EVENT TRIGGER IS KNOWN TO BE SPURIOUS, DO NOT CLASSIFY EVENT I.E., FALSE HI READING, FALSE CHLORINE MONITOR READINGS ETC.

2.1.4 If the most severe events or conditions are classified as an Unusual Event, implement EP-102, "Unusual Event Response."

2.1.5 If the most severe events or conditions are classified as an Alert, implement EP-103, "Alert Response."

2.1.6 If the most severe events or conditions are classified as a Site Emergency, implement EP-104, "Site Emergency Response."

2.1.7 If the most severe events or conditions are classified as a General Emergency, implement EP-105, "General Emergency Response."

3.0 ACTIONS-FOLLOW-UP

3.1 If the event is classified as an Unusual Event, the Emergency Director shall have a written summary sent to the NRC within twenty-four hours of closeout in accordance with EP-106, Written Summary Notification.

3.2 If event is classified as Alert, Site Emergency, or General Emergency, the Emergency Director shall:

3.2.1 Periodically evaluate the event classification as listed on attached Appendix EP-101. Based upon results of corrective action taken to recover from the emergency situation, escalate or de-escalate the emergency classification. (It is preferable, but not mandatory, to obtain concurrance from the Site Emergency Coordinator and Corporate Headquarters prior to classification reduction). The NRC and appropriate off-site authorities shall be informed of the decision to move from one emergency class to the next. As appropriate, agencies or personnel listed in phone lists of Appendix 1 of EPs 102, 103, 104, and 105 shall be notified within 15 minutes once the emergency level is declared.

3.2.2 Have a written summary sent to the NRC within eight hours of closeout or de-escalation of the emergency classification in accordance with EP-106, Written Summary Notification.

3.2.3 When the emergency has been controlled and the power plant and auxiliaries have been placed in a safe shutdown condition, only then will a decision be made as to whether a recovery phase is justified. Enter the recovery phase after the emergency or accident situation is considered no longer in effect, obtain the concurrence of the Site Emergency Coordinator and the Emergency Support Officer at Corporate Headquarters as required per EP-410, Recovery Phase Implementation. The recovery phase is a departure from an emergency situation. The Site Emergency Coordinator and yourself should evaluate plant operating conditions as well as the in-plant and out-of-plant radiological conditions when making this decision. Notifications to the various individuals and agencies that the recovery phase has been implemented is the responsibility of the Site Emergency Coordinator.

4.0 APPENDICES

4.1 EP-101-1 Hazards to Station Operation

4.2 EP-101-2 Environmental

4.3 EP-101-3 Loss of Power

4.4 EP-101-4 Personnel Injury

4.5 EP-101-5 Fire

4.6 EP-101-6 Radioactive Release

4.7 EP-101-7 Evacuation of Control Room

4.8 EP-101-8 Damage of Fuel

4.9 EP-101-9 Instrument Failure

4.10 EP-101-10 Scram Failure

4.11 EP-101-11 Boundary Degradation/LOCA

4.12 EP-101-12 Unusual Shutdown

4.13 EP-101-13 Loss of Hot or Cold Shutdown Capability

4.14 EP-101-14 Security

5.0 SUPPORTING INFORMATION

5.1 Purpose

The purpose of this procedure is to provide guidelines for classifying an event or condition into one of four emergency classifications as described in the Emergency Plan. Additionally this procedure details the method to change from one emergency action level to another and to enter the recovery phase, if applicable.

5.2 Criteria For Use

5.2.1 This procedure shall be implemented whenever the Shift Superintendent becomes aware of conditions which meet or exceed the Emergency Action Levels in EP-101, Classification Tables.

5.3 Special Equipment

None

5.4 References

5.4.1 Limerick Generating Station Emergency Plan

5.4.2 NUREG 0654 Rev. 1 Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants

5.4.3 EP-102 Unusual Event Response

5.4.4 EP-102 Appendix 1 Unusual Event Notification Message

5.4.5 EP-103 Alert Response

5.4.6 EP-103 Appendix 1 Alert Notification Message

5.4.7 EP-104 Site Emergency Response

5.4.8 EP-104 Appendix 1 Site Emergency Notification

5.4.9 EP-105 General Emergency Response

5.4.10 EP-105 Appendix 1 General Emergency Notification

5.4.11 EP-106 Written Summary Notification

5.4.12 EP-410 Recovery Phase Implementation

HAZARDS TO STATION OPERATION

UNUSUAL EVENT

1. Aircraft crash on-site or unusual aircraft activity over the site.

2. Train derailment within the site boundary.

3. Explosion within or near the site boundary.

4. Nearby or on-site release of potentially harmful quantities of toxic, flammable gas or chlorine.

ALERT

1. Entry of toxic, flammable gases or chlorine into the power block with subsequent habitability problem indicated by:
   Visual observation, direct measurement or notification received by Control Room.

WHEN BOTH UNITS ARE IN COLD SHUTDOWN

2. Aircraft crash or missile impact on the Reactor Enclosure, Control Enclosure, Turbine Enclosure, Diesel Generator Enclosure or Spray Pond Pump House.

3. Known explosion damage affecting plant operation.

SITE EMERGENCY

1. Entry of toxic, flammable gases or chlorine into vital areas, where lack of access constitutes a reactor safety problem, indicated by:

   A. Shift Supervision evaluation
      AND

   B. Visual observation, direct measurement, notification received by Control Room.

WHEN EITHER UNIT IS NOT IN COLD SHUTDOWN

2. Aircraft crash or missile impact on the Reactor Enclosure, Control Enclosure, Turbine Enclosure, Diesel Generator Enclosure or Spray Pond Pump House.

3. Known explosion damage affecting plant operation.

GENERAL EMERGENCY

ENVIRONMENTAL

UNUSUAL EVENT

1. An actual earthquake detected by the Seismic Monitoring System (00C693) at or below operating basis earthquake (.075g).

2. A tornado is observed within or near site boundary.

3. A National Weather Service hurricane warning is issued for Montgomery County.

ALERT

1. An actual earthquake detected by the Seismic Monitoring System (00C693) beyond the operating basis earthquake (.075g).

2. Tornado strikes the Reactor Enclosure, Turbine Enclosure, Spray Pond Pump House, Control Enclosure or Diesel Generator Enclosure.

3. Sustained high winds greater than 70 mph as indicated on 0BC699.

SITE EMERGENCY

1. Sustained high winds greater than 90 mph as indicated on 0BC699 if either unit is not in Cold Shutdown.

2. An actual earthquake detected by the Seismic Monitoring System (00C693) beyond the safe shutdown earthquake (.15g) if either unit is not in Cold Shutdown.

GENERAL EMERGENCY

1. Earthquake beyond the safe shutdown earthquake (.15g) or other natural disaster which causes massive damage leading to other General Emergencies.

LOSS OF POWER

UNUSUAL EVENT	ALERT
1. Loss of all off-site power or loss of all on-site AC power for greater than 60 seconds.	N/A

SITE EMERGENCY	GENERAL EMERGENC
1. Loss of all on-site AC power and loss of off-site power 2. Loss of all safety-related DC power as indicated by: a) 250VDC MCC Out of Service alarms AND b) 1PPA1 D3 Distribution Panel undervoltage alarms	N/A

PERSONNEL INJURY

UNUSUAL EVENT	ALERT
1. Transportation of contaminated injured individual from site to off-site hospital.	N/A

SITE EMERGENCY	GENERAL EMERGENCY
N/A	N/A

FIRE

UNUSUAL EVENT	ALERT
1. Fires involving permanent plant structures within the protected area lasting 10 minutes or more after initial attempts to extinguish it.	1. Fire which could make an ECCS inop as indicated by observation.

SITE EMERGENCY	GENERAL EMERGENCY
1. Fire which makes an ECCS inop and requires or causes immediate plant shutdown as indicated by observation.	1. Fire which causes massive damage leading to other General Emergencies.

RADIOACTIVE RELEASE

UNUSUAL EVENT

1. Report indicates liquid effluent release exceeds technical specification 3.11.1.1 or 3.11.1.2.

2. Report indicates gaseous effluent release exceeds technical specification 3.11.2.1 or 3.11.2.2 or 3.11.2.3

ALERT

1. Radiological effluents release greater than 0.5 mR/hr at site boundary as indicated by an uncontrollable release for greater than 20 minutes with:

   a) North stack effluent radiation monitor exceeds 1.0N2 uCi/cc or

   b) South stack effluent radiation monitor exceeds 1.2N2 uCi/cc.

SITE EMERGENCY

1. Radiological effluent release greater than 50 mR/hr at site boundary as indicated by an uncontrollable release for greater than 20 minutes with:

   a) North stack effluent radiation monitor exceeds 1.0 uCi/cc.

2. Projected whole body dose greater than .1 rem or thyroid dose greater than .5 Rem at or beyond the site boundary over course of the event utilizing RMMS procedure calculating offsite doses.

GENERAL EMERGENCY

1. Radiological effluent release greater than 500 mR/hr at site boundary as indicated by an uncontrollable release for greater than 20 minutes with:

   a) North stack effluent radiation monitor exceeds 10 uCi/cc.

2. Projected whole body dose greater than 1 Rem or thyroid dose greater than 5 Rem at or beyond the site boundary over course of the event utilizing RMMS procedure calculating offsite doses.

EVACUATION OF CONTROL ROOM

UNUSUAL EVENT	ALERT
N/A	1. Evacuation of Control Room anticipated or required with control established .t remote shutdown panel.

SITE EMERGENCY	GENERAL EMERGENCY
1. Evacuation of Contol Room and control of shutdown systems <u>not</u> established from remote shutdown panel in 15 minutes.	N/A

DAMAGE OF FUEL

UNUSUAL EVENT

1. Steam Jet Air Ejector Discharge radiation monitor exceeds 2.1P4 mR/hr.

2. Steam Jet Air Ejector Discharge radiation monitor has an unexpected increase of 4000 mR/hr over 30 minutes.

3. I-131 dose equivalent in the reactor coolant exceeds 0.2 uCi/g from sample analysis.

ALERT

1. Steam Jet Air Ejector Discharge radiation monitor exceeds 2.1P5 mR/hr

2. I-131 dose equivalent in the reactor coolant exceeds 300 uCi/g from sample and main steam line high-high radiation with resultant scram.

3. Spent fuel damage resulting in a refueling floor area ventilation exhaust monitor alarm.

4. Containment Post LOCA Radiation Monitors greater than 1P2 R/hr.

SITE EMERGENCY

1. Major damage to spent fuel:

   a) Observation of major damage to spent fuel

   <u>OR</u>

   b) Water loss below fuel level in spent fuel pool.

2. Containment Post LOCA Radiation Monitors greater than 1P3 R/hr.

GENERAL EMERGENCY

1. Containment Post LOCA Radiation Monitors greater than 1P4 R/hr.

INSTRUMENT FAILURE

UNUSUAL EVENT	ALERT
1. Complete loss of all Main Control Room communication equipment. 2. Significant loss of assessment capability in the Main Control Room as indicated by: Simultaneous loss of the plant process computer and the ERFDS for a period greater than 24 hours.	1. Loss of all annunciators.

SITE EMERGENCY	GENERAL EMERGENCY
1. All annunciators lost and plant transient initiated or in progress.	N/A

SCRAM FAILURE

UNUSUAL EVENT	ALERT
N/A	1. Failure of the Reactor protection system to automatically initiate and complete a scram **AND** Scram fails to bring Reactor subcritical as indicated by APRM's greater than 4%, one minute after scram initiates.

SITE EMERGENCY	GENERAL EMERGENCY
1. Transient requiring standby liquid control system to initiate with failure to scram. Failure to Scram is indicated by APRM'S greater than 4% one minute after a scram initiates.	1. Transient requiring standby liquid control system to initiate with failure to scram **AND** Reactor does not become sub-critical as indicated by APRM'S greater than 4% 10 minutes after scram initiates.

BOUNDARY DEGRADATION/LOCA
PAGE 1 of 2

UNUSUAL EVENT	ALERT
1. Failure of a main steam relief valve or ADS valve to close following reduction of applicable pressure.	1. Scram with small leak as indicated by: a) Scram alarm **AND** b) Reactor level less than -129" **AND** c) Containment pressure greater than 1.68 psig and pressure is increasing.
	2. Reactor coolant leak rate exceeds 60 gpm total leakage averaged over any 24 hour period as indicate by surveillance test report.
2. Reactor coolant leak rate exceeds 30 gpm total leakage as indicated by surviellance test report.	3. High airborne contamination in the Reactor Enclosure as indicated by: a) Reactor Enclosure vent exhaust RAD monitor A/B or C/D Hi-Hi ALARM on 10C800 (20C800) OR b) 1000 fold increase of airborne radiation in a major area of the reactor enclosure as determined by health physics.

BOUNDARY DEGRADATION/LOCA
PAGE 2 of 2

SITE EMERGENCY

1. Scram with LOCA as indicated by:

   a) Scram alarm
   AND

   b) Reactor level less than -129"
   AND

   c) Containment pressure greater than 10 psig

2. Main steam line break outside containment without isolation as indicated by:

   a) High Main Steam Line Flow (108.7 psid)
   AND

   b) High Steam Tunnel Temp (165 deg F)
   AND

   c) Main Steam Line Low Pressure (756 psig)

GENERAL EMERGENCY

1. Scram with LOCA & no ECCS as indicated by:

   a) Scram alarm
   AND

   b) Reactor level less than -129"
   AND

   c) Failure to bring Reactor level above -129" after 3 minutes
   AND

   d) Containment pressure greater than 20 psig

2. Scram with LOCA & Containment Failure as indicated by:

   a) Scram with Reactor level less than -129"
   AND

   b) Reactor Enclosure Vent Exhaust Rad Monitor A/B or C/D Hi-Hi alarm on 10C800 (20C800)

UNUSUAL SHUTDOWN

UNUSUAL EVENT	ALERT
1. Controlled shutdown due to failure to meet limiting condition of operation. 2. Shutdown other than normal controlled shutdown AND for the purpose of placing the the plant in a safer condition. 3. Cooldown rate exceeds technical specification limits.	N/A

SITE EMERGENCY	GENERAL EMERGENCY
N/	N/A

LOSS OF HOT OR COLD SHUTDOWN CAPABILITY

UNUSUAL EVENT

N/A

ALERT

1. Complete loss of the ability to establish and maintain the plant in a Cold Shutdown condition. Symptomized by:

   a) Inability to establish Reactor Coolant Temperature of less than 200 degrees F in a timely manner with the mode switch in Shutdown

   OR

   b) Loss of all means of Primary and Alternate Decay Heat Removal when shutdown such that Reactor Coolant Temperature cannot be maintained below 200 degrees F.

SITE EMERGENCY

1. Actual inability to reduce Reactor Coolant System Temperature while the Plant is Shutdown. Indicated by:

   a) Reactor mode switch in Shutdown

   AND

   b) Reactor Coolant Temperature greater than 200 degrees F and rising

   AND

   c) Suppression Pool Temperature greater than 120 degrees F and rising.

GENERAL EMERGENCY

1. Inability to reduce Reactor Coolant System Temperature with the potential for the release of large amounts of Radioactivity. Indicated by:

   a) Inability to maintain Reactor Level greater TAF

   OR

   substantial Fuel Damage has occurred

   AND

   b) Inability to maintain Suppression Pool Temperature below the upper limit of the Heat Capacity Curve (SP/T-21 on T102 Procedure.

SECURITY

UNUSUAL EVENT	ALERT
1. Security threat OR Attempted entry OR Attempted sabotage as illustrated by: Event 1 - Sabotage or Bomb Threat Event 2 - Intrusion and Attack Threat Event 7 - Suspected Intrusion Event 8 - Actual Intrusion Event 9 - Suspected Bomb or Sabotage Device Discovered Event 15 - Guard Strike Event 16 - Onsite Hostage Situation	1. Ongoing security compromise

SITE EMERGENCY	GENERAL EMERGENCY
1. Imminent loss of physical control of the plant.	1. Loss of physical control of the facility.

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3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding Specifications is required during the OPERATIONAL CONDITIONS or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

3.0.2 Noncompliance with a Specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified tim intervals. If the Limiting Condition for Operation is restored prior to e: piration of the specified time intervals, completion of the Action requirei ents is not required.

3.0.3 When a Limiting Condition 'or Operation is not met, except as provided in the associated ACTION requirem its, within one hour action shall be initiated to place the unit in an OPERATIONA CONDITION in which the Specification does not apply by placing it, as applica le, in:

a. At least STARTUP within the next 6 hours,
b. At least HOT SHUTDOWN within the following 6 hours, and
c. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications.

This Specification is not applicable in OPERATIONAL CONDITION 4 or 5.

3.0.4 Entry into an OPERATIONAL CONDITION or other specified condition shall not be made unless the conditions for the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION requirements. This provision shall not prevent passage through or to OPERATIONAL CONDITIONS as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual Specifications.

APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL CONDITIONS or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirements.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval, but

b. The combined time interval for any 3 consecutive surveillance intervals shall not exceed 3.25 times the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the specified time interval shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specificatons. Surveillance requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL CONDITION or other specified applicable condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the applicable surveillance interval or as otherwise specified.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, & 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g) (6) (i).

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel(s) and/or that trip system in the tripped condition* within 1 hour. The provisions of Specification 3.0.4 are not applicable.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within 1 hour and take the ACTION required by Table 3.3.1-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip functional unit shown in Table 3.3.1-2 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip system.

---

*An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.1-1 for that Trip Function shall be taken.

**The trip system need not be placed in the tripped condition if this would cause the Trip Function to occur. When a trip system can be placed in the tripped condition without causing the Trip Function to occur, place the trip system with the most inoperable channels in the tripped condition; if both systems have the same number of inoperable channels, place either trip system in the tripped condition.

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
1. Intermediate Range Monitors[b]:			
a. Neutron Flux - High	2	3	1
	3, 4	3	2
	5(c)	3(d)	3
b. Inoperative	2	3	1
	3, 4	3	2
	5	3(d)	3
2. Average Power Range Monitor[e]:			
a. Neutron Flux - Upscale, Setdown	2	2	1
	3, 4	2	2
	5(c)	2(d)	3
b. Neutron Flux - Upscale			
1) Flow Biased	1	2	4
2) High Flow Clamped	1	2	4
c. Inoperative	1, 2	2	1
	3, 4	2	2
	5(c)	2(d)	3
d. Downscale	1(g)	2	4
3. Reactor Vessel Steam Dome Pressure - High	1, 2(f)	2	1
4. Reactor Vessel Water Level - Low, Level 3	1, 2	2	1
5. Main Steam Line Isolation Valve - Closure	1(g)	1/valve	4

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
6. Main Steam Line Radiation - High	1, 2(f)	2	5
7. Drywell Pressure - High	1, 2(h)	2	1
8. Scram Discharge Volume Water Level - High			
a. Level Transmitter	1, 2	2	1
	5(i)	2	3
b. Float Switch	1, 2	2	1
	5 (i)	2	3
9. Turbine Stop Valve - Closure	1(j)	4(k)	6
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	1(j)	2(k)	6
11. Reactor Mode Switch Shutdown Position	1, 2	2	1
	3, 4	2	7
	5	2	3
12. Manual Scram	1, 2	2	1
	3, 4	2	8
	5	2	9

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION STATEMENTS

ACTION 1 - Be in at least HOT SHUTDOWN within 12 hours.

ACTION 2 - Verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within 1 hour.

ACTION 3 - Suspend all operations involving CORE ALTERATIONS and insert all insertable control rods within 1 hour.

ACTION 4 - Be in at least STARTUP within 6 hours.

ACTION 5 - Be in STARTUP with the main steam line isolation valves closed within 6 hours or in at least HOT SHUTDOWN within 12 hours.

ACTION 6 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure until the function is automatically bypassed, within 2 hours.

ACTION 7 - Verify all insertable control rods to be inserted within 1 hour.

ACTION 8 - Lock the reactor mode switch in the Shutdown position within 1 hour.

ACTION 9 - Suspend all operations involving CORE ALTERATIONS, and insert all insertable control rods and lock the reactor mode switch in the SHUTDOWN position within 1 hour.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) This function shall be automatically bypassed when the reactor mode switch is in the Run position and the associated APRM is not downscale.

(c) The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn* and shutdown margin demonstrations performed per Specification 3.10.3.

(d) The noncoincident NMS reactor trip function logic is such that all channels go to both trip systems. Therefore, when the "shorting links" are removed, the Minimum OPERABLE Channels Per Trip System is 4 APRMs, 6 IRMs and 2 SRMs.

(e) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 14 LPRM inputs to an APRM channel.

(f) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(g) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(h) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(j) This function shall be automatically bypassed when turbine first stage pressure is equivalent to a THERMAL POWER of less than 30% of RATED THERMAL POWER.

(k) Also actuates the EOC-RPT system.

*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High	N.A.
b. Inoperative	N.A.
2. Average Power Range Monitor*:	
a. Neutron Flux - Upscale, Setdown	N.A.
b. Neutron Flux - Upscale	
1) Flow Biased	$\leq$0.09
2) High Flow Clamped	$\leq$0.09
c. Inoperative	N.A.
d. Downscale	N.A.
3. Reactor Vessel Steam Dome Pressure - High	$\leq$ 0.55
4. Reactor Vessel Water Level - Low, Level 3	$\leq$ 1.05
5. Main Steam Line Isolation Valve - Closure	$\leq$ 0.06
6. Main Steam Line Radiation - High	N.A.
7. Drywell Pressure - High	N.A.
8. Scram Discharge Volume Water Level - High	
a. Level Transmitter	N.A.
b. Float Switch	N.A.
9. Turbine Stop Valve - Closure	$\leq$ 0.06
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	$\leq$ 0.08**
11. Reactor Mode Switch Shutdown Position	N.A.
12. Manual Scram	N.A.

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel.

**Measured from start of turbine control valve fast closure.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	S/U,S(b)	S/U(c), W	R	2
	S	W(j)	R	3, 4, 5
b. Inoperative	N.A.	W(j)	N.A.	2, 3, 4, 5
2. Average Power Range Monitor[f]:				
a. Neutron Flux - Upscale, Setdown	S/U,S(b)	S/U(c), W	SA	2
	S	W(j)	SA	3, 5
b. Neutron Flux - Upscale				
1) Flow Biased	S,D(g)	S/U(c), W	W(d)(e),SA,	1
2) High Flow Clamped	S	S/U(c), W	W(d)(e), SA	1
c. Inoperative	N.A.	W(j)	N.A.	1, 2, 3, 5
d. Downscale	S	W	SA	1
3. Reactor Vessel Steam Dome Pressure - High	S	M	R	1, 2(h)
4. Reactor Vessel Water Level - Low, Level 3	S	M	R	1, 2
5. Main Steam Line Isolation Valve - Closure	N.A.	M	R	1
6. Main Steam Line Radiation - High	S	M	R	1, 2(h)
7. Drywell Pressure - High	S	M	R	1, 2

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High				
a. Level Transmitter	S	M	R	1, 2, 5(i)
b. Float Switch	N.A.	M	R	1, 2, 5(i)
9. Turbine Stop Valve - Closure	N.A.	M	R	1
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	N.A.	M	R	1
11. Reactor Mode Switch Shutdown Position	N.A.	R	N.A.	1, 2, 3, 4, 5
12. Manual Scram	N.A.	M	N.A.	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) The IRM and SRM channels shall be determined to overlap for at least ½ decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least ½ decades during each controlled shutdown, if not performed within the previous 7 days.

(c) Within 24 hours prior to startup, if not performed within the previous 7 days.

(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER > 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.

(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.

(g) Verify measured core flow (total core flow) to be greater than or equal to established core flow at the existing loop flow (APRM % flow).

(h) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(j) If the RPS shorting links are required to be removed per Specification 3.9.2, they may be reinstalled for up to 2 hours for required surveillance. During this time, CORE ALTERATIONS shall be suspended, and no control rod shall be moved from its existing position.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6. The control rod block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2..

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

a. With a control rod block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, take the ACTION required by Table 3.3.6-1.

SURVEILLANCE REQUIREMENTS

4.3.6 Each of the above required control rod block trip systems and instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1.

TABLE 3.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. ROD BLOCK MONITOR(a)			
a. Upscale	2	1*	60
b. Inoperative	2	1*	60
c. Downscale	2	1*	60
2. APRM			
a. Flow Biased Neutron Flux - Upscale	4	1	61
b. Inoperative	4	1, 2, 5	61
c. Downscale	4	1	61
d. Neutron Flux - Upscale, Startup	4	2, 5	61
3. SOURCE RANGE MONITORS ***			
a. Detector not full in(b)	3 2	2 5	61 61
b. Upscale(c)	3 2	2 5	61 61
c. Inoperative(c)	3 2	2 5	61 61
d. Downscale(d)	3 2	2 5	61 61
4. INTERMEDIATE RANGE MONITORS			
a. Detector not full in	6	2, 5	61
b. Upscale	6	2, 5	61
c. Inoperative	6	2, 5	61
d. Downscale(e)	6	2, 5	61
5. SCRAM DISCHARGE VOLUME			
a. Water Level-High	2	1, 2, 5**	62
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW			
a. Upscale	2	1	62
b. Inoperative	2	1	62
c. Comparator	2	1	62
7. REACTOR MODE SWITCH SHUTDOWN POSITION	2	3, 4	63

TABLE 3.3.6-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

ACTION STATEMENTS

ACTION 60 - Declare the RBM inoperable and take the ACTION required by Specification 3.1.4.3.

ACTION 61 - With the number of OPERABLE channels one or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.

ACTION 62 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour.

ACTION 63 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, initiate a rod block.

NOTES

* With THERMAL POWER $\geq$ 30% of RATED THERMAL POWER.

** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

*** These channels are not required when sixteen or fewer fuel assemblies, adjacent to the SRMs, are in the core.

(a) The RBM shall be automatically bypassed when a peripheral control rod is selected or the reference APRM channel indicates less than 30% of RATED THERMAL POWER.

(b) This function shall be automatically bypassed if detector count rate is > 100 cps or the IRM channels are on range 3 or higher.

(c) This function is automatically bypassed when the associated IRM channels are on range 8 or higher.

(d) This function is automatically bypassed when the IRM channels are on range 3 or higher.

(e) This function is automatically bypassed when the IRM channels are on range 1.

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TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. **ROD BLOCK MONITOR**		
a. Upscale		
i. flow biased	≤ 0.66 W + 41%, with a maximum of,	≤ 0.66 W + 44%, with a maximum of,
ii. high flow clamped	≤ 107%	≤ 110%
b. Inoperative	N.A.	N.A.
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
2. **APRM**		
a. Flow Biased Neutron Flux - Upscale	≤ 0.58 W + 50%*	≤ 0.58 W + 53%*
b. Inoperative	N.A.	N.A.
c. Downscale	≥ 4% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale, Startup	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
3. **SOURCE RANGE MONITORS**		
a. Detector not full in	N.A.	N.A.
b. Upscale	≤ 1 x 10^5 cps	≤ 1.6 x 10^5 cps
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 3 cps**	≥ 1.8 cps**
4. **INTERMEDIATE RANGE MONITORS**		
a. Detector not full in	N.A.	N.A.
b. Upscale	≤ 108/125 divisions of full scale	≤ 110/125 divisions of full scale
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 divisions of full scale
5. **SCRAM DISCHARGE VOLUME**		
a. Water Level-High	≤ 257' 5 9/16" elevation***	≤ 257' 7 9/16" elevation
a. Float Switch		

TABLE 3.3.6-2 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	$\leq$ 111% of rated flow	$\leq$ 114% of rated flow
b. Inoperative	N.A.	N.A.
c. Comparator	$\leq$ 10% flow deviation	$\leq$ 11% flow deviation
7. REACTOR MODE SWITCH SHUTDOWN POSITION	N.A.	N.A.

*The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

**May be reduced to 0.7 cps provided the signal-to-noise ratio is $\geq$ 2.

***Equivalent to 13 gallons/scram discharge volume.

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TABLE 4.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	N.A.	S/U[b][c], M[c]	SA	1*
b. Inoperative	N.A.	S/U[b][c], M[c]	N.A.	1*
c. Downscale	N.A.	S/U[b][c], M[c]	SA	1*
2. APRM				
a. Flow Biased Neutron Flux - Upscale	N.A.	S/U[b], M	SA	1
b. Inoperative	N.A.	S/U[b], M	N.A.	1, 2, 5
c. Downscale	N.A.	S/U[b], M	SA	1
d. Neutron Flux - Upscale, Startup	N.A.	S/U[b], M	SA	2, 5
3. SOURCE RANGE MONITORS				
a. Detector not full in	N.A.	S/U[b], W	N.A.	2, 5
b. Upscale	N.A.	S/U[b], W	SA	2, 5
c. Inoperative	N.A.	S/U[b], W	N.A.	2, 5
d. Downscale	N.A.	S/U[b], W	SA	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	N.A.	S/U[b], W	N.A.	2, 5
b. Upscale	N.A.	S/U[b], W	SA	2, 5
c. Inoperative	N.A.	S/U[b], W	N.A.	2, 5
d. Downscale	N.A.	S/U[b], W	SA	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	N.A.	M	R	1, 2, 5**
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	N.A.	S/U[b], M	SA	1
b. Inoperative	N.A.	S/U[b], M	N.A.	1
c. Comparator	N.A.	S/U[b], M	SA	1
7. REACTOR MODE SWITCH SHUTDOWN POSITION	N.A.	R	N.A.	3, 4

TABLE 4.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) Within 24 hours prior to startup, if not performed within the previous 7 days.

(c) Includes reactor manual control multiplexing system input.

* With THERMAL POWER $\geq$ 30% of RATED THERMAL POWER.

** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2*, and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

a. After each closing of each penetration subject to Type B testing, except the primary containment air locks, if opened following Type A or B test, by leak rate testing the seals with gas at P_a, 44.0 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Surveillance Requirement 4.6.1.2d. for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 L_a.

b. At least once per 31 days by verifying that all primary containment penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except as provided in Table 3.6.3-1 of Specification 3.6.3.

c. By verifying the primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. By verifying the suppression chamber is in compliance with the requirements of Specification 3.6.2.1.

*See Special Test Exception 3.10.1

**Except valves, blind flanges, and deactivated automatic valves which are located inside the containment, and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed when the primary containment has not been deinerted since the last verification or more often than once per 92 days.

PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to L_a, 0.500 percent by weight of the containment air per 24 hours at P_a, 44.0 psig.

b. A combined leakage rate of less than or equal to 0.60 L_a for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves* and valves which are hydrostatically tested per Table 3.6.3-1, subject to Type B and C tests when pressurized to P_a, 44.0 psig.

c. *Less than or equal to 11.5 scf per hour for any one main steam line through the isolation valves when tested at P_t, 22.0 psig.

d. A combined leakage rate of less than or equal to 1 gpm times the total number of containment isolation valves in hydrostatically tested lines which penetrate the primary containment, when tested at 1.10 P_a, 48.4 psig.

APPLICABILITY: When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.6.1.1.

ACTION:

With:

a. The measured overall integrated primary containment leakage rate exceeding 0.75 L_a, or

b. The measured combined leakage rate for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves* and valves which are hydrostatically tested per Table 3.6.3-1, subject to Type B and C tests exceeding 0.60 L_a, or

c. The measured leakage rate exceeding 11.5 scf per hour for any one main steam line through the isolation valves, or

d. The measured combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment exceeding 1 gpm times the total number of such valves,

restore:

a. The overall integrated leakage rate(s) to less than or equal to 0.75 L_a, and

*Exemption to Appendix J of 10 CFR Part 50.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

b. The combined leakage rate for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves* and valves which are hydrostatically tested per Table 3.6.3-1, subject to Type B and C tests to less than or equal to 0.60 L_a, and

c. The leakage rate to less than or equal to 11.5 scf per hour for any one main steam line through the isolation valves, and

d. The combined leakage rate for all containment isolation valves in hydrostatically tested lines which pentrate the primary containment to less than or equal to 1 gpm times the total number of such valves,

prior to increasing reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The primary containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50 using the methods and provisions of ANSI 45.4-1972 and BN-TOP-1 and verifying the result by the Mass Point Methodology described in ANSI N56.8-1981:

a. Three Type A Overall Integrated Containment Leakage Rate tests shall be conducted at 40 ± 10 month intervals during shutdown at P_a, 44.0 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

b. If any periodic Type A test fails to meet 0.75 L_a, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 L_a, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 L_a, at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

1. Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within 0.25 L_a. The formula to be used is: $[L_o + L_{am} - 0.25\ L_a] \leq L_c \leq [L_o + L_{am} + 0.25\ L_a]$ where L_c = supplemental test result; L_o = superimposed leakage; L_{am} = measured Type A leakage.

2. Has duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be between 0.75 L_a and 1.25 L_a.

*Exemption to Appendix "J" to 10 CFR Part 50.

SURVEILLANCE REQUIREMENTS (Continued)

d. Type B and C tests shall be conducted with gas at P_a, 44.0 psig*, at intervals no greater than 24 months** except for tests involving:

   1. Air locks,

   2. Main steam line isolation valves,

   3. Containment isolation valves in hydrostatically tested lines which penetrate the primary containment, and

e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

f. Main steam line isolation valves shall be leak tested at least once per 18 months.

g. Containment isolation valves in hydrostatically tested lines which penetrate the primary containment shall be leak tested at least once per 18 months.**

h. The provisions of Specification 4.0.2 are not applicable to Specifications 4.6.1.2a., 4.6.1.2b., 4.6.1.2c., 4.6.1.2d., and 4.6.1.2e.

*Unless a hydrostatic test is required per Table 3.6.3-1.

**A Type C test interval extension to May 26, 1986 is permissible for primary containment isolation valves identified by an asterisk in the inboard and outboard isolation barrier columns of Table 3.6.3-1, Part A, as discussed in Application for Amendment of Facility Operating License dated December 18, 1985.

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MSIV LEAKAGE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.4 Two independent MSIV leakage control system (LCS) subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With one MSIV leakage control system subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 Each MSIV leakage control system subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Starting the blower(s) from the control room and operating the blower(s) for at least 15 minutes.

2. Energizing the heaters and verifying a temperature rise indicating heater operation on downstream piping.

b. During each COLD SHUTDOWN, if not performed within the previous 92 days, by cycling each motor operated valve through at least one complete cycle of full travel.

c. At least once per 18 months by:

1. Performance of a functional test which includes simulated actuation of the subsystem throughout its operating sequence, and verifying that each interlock and timer operates as designed, each automatic valve actuates to its correct position and the blower starts.

2. Verifying that the blower(s) develops at least the below required vacuum at the rated capacity:

a) Inboard valves, 15" H_2O at 100 scfm.

b) Outboard valves, 15" H_2O at 200 scfm.

d. By verifying the operating instrumentation to be OPERABLE by performance of a:

1. CHANNEL CHECK at least once per 24 hours,
2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and
3. CHANNEL CALIBRATION at least once per 18 months.

PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.5 The structural integrity of the primary containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.5.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the structural integrity of the primary containment not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5.1 The structural integrity of the exposed accessible interior and exterior surfaces of the primary containment, including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test by a visual inspection of those surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.5.2 Reports Any abnormal degradation of the primary containment structure detected during the above required inspections shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days. This report shall include a description of the condition of the liner and concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.6 Drywell and suppression chamber internal pressure shall be maintained between 0.0 and +2.0 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the drywell and/or suppression chamber internal pressure outside of the specified limits, restore the internal pressure to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The drywell and suppression chamber internal pressure shall be determined to be within the limits at least once per 12 hours.

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DRYWELL AVERAGE AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.7 Drywell average air temperature shall not exceed 135°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the drywell average air temperature greater than 135°F, reduce the average air temperature to within the limit within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The drywell average air temperature shall be the volumetric average of the temperatures at the following locations and shall be determined to be within the limit at least once per 24 hours:

	Elevation	Azimuth*
a.	330'	45°, 90°, 225°
b.	320'	105°, 225°, 345°
c.	260'	50°, 165°, 285°
d.	248'	11°, 74°, 150°, 182°, 252°, 337°

*At least one reading from each elevation is required for a volumetric average calculation.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The primary containment isolation valves and the reactor instrumentation line excess flow check valves shown in Table 3.6.3-1 shall be OPERABLE with isolation times less than or equal to those shown in Table 3.6.3-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or more of the primary containment isolation valves shown in Table 3.6.3-1 inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours either:

1. Restore the inoperable valve(s) to OPERABLE status, or
2. Isolate each affected penetration by use of at least one de-activated automatic valve secured in the isolated position,* or
3. Isolate each affected penetration by use of at least one closed manual valve or blind flange.*
4. The provisions of Specification 3.0.4 are not applicable provided that within 4 hours the affected penetration is isolated in accordance with ACTION a.2. or a.3. above, and provided that the associated system, if applicable, is declared inoperable and the appropriate ACTION statements for that system are performed.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one or more of the reactor instrumentation line excess flow check valves shown in Table 3.6.3-1 inoperable, operation may continue and the provisions of Specifications 3.0.3 and 3.0.4 are not applicable provided that within 4 hours either:

1. The inoperable valve is returned to OPERABLE status, or
2. The instrument line is isolated and the associated instrument is declared inoperable.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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*Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under administrative control.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each primary containment isolation valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.

4.6.3.2 Each primary containment automatic isolation valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE during COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that on a containment isolation test signal each automatic isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each primary containment power operated or automatic valve shown in Table 3.6.3-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 Each reactor instrumentation line excess flow check valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE at least once per 18 months by verifying that the valve checks flow.

4.6.3.5 Each traversing in-core probe system explosive isolation valve shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying the continuity of the explosive charge.

b. At least once per 18 months by removing the explosive squib from the explosive valve, such that each explosive squib in each explosive valve will be tested at least once per 90 months, and initiating the explosive squib. The replacement charge for the exploded squib shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life and/or operating life, as applicable.

TABLE 3.6.3-1

PART A - PRIMARY CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX. ISOL. TIME. IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
003B	CONTAINMENT INSTRUMENT GAS SUPPLY - HEADER 'B'	59-1005B (CK)		NA			59
			HV59-129B	7	C,H,S		
003D-2	CONTAINMENT INSTRUMENT GAS SUPPLY TO ADS VALVES E & K	59-1112*(CK)		NA			
			HV59-151B*	45	M		59
007A(B,C,D)	MAIN STEAM LINE 'A'(B,C,D)	HV41-1F022A (B,C,D)		5*	C,D,E,F,P,Q	6	41
			HV41-1F028A (B,C,D)	5*	C,D,E,F,P,Q	6	
			HV40-1F001B (F,K,P)	45	EA	6	
			(XV40-101B (F,K,P) SEE PART B, THIS TABLE)	NA		6,1	
008	MAIN STEAM LINE DRAIN	HV41-1F016		30	C,D,E,F,P,Q	4	41
			HV41-1F019	30	C,D,E,F,P,Q		
009A	FEEDWATER	41-1F010A(CK)		NA			41
			HV41-1F074A(CK)	NA			
			41-1036A(CK)	NA			
			HV41-130B	45			
			HV41-133A	45			
			HV41-109A	NA		32	
			HV41-1F032A(CK)	NA			
			HV55-1F105	30		7	
			HV44-1F039(CK) (X-9B)	NA			
			41-1016(X-9B, X-44)	NA		31	

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TABLE 3.6.3-1 (Continued)

PART A - PRIMARY CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX. ISOL. TIME, IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
009B	FEEDWATER	41-1F010B(CK)		NA			41
			HV41-1F074B(CK)	NA			
			41-1036B(CK)	NA			
			HV41-130A	45			
			HV41-133B	45			
			HV41-109B	NA		32	
			HV41-1F032B(CK)	NA			
			HV49-1F013	23	LFCC		
			HV44-1F039(CK) (X-9A)	NA			
			41-1016(X-9A, X-44)	NA		31	
010	RCIC STEAM SUPPLY	HV49-1F007		7.2*	K, KA	5	49
			HV49-1F008	7.2*	K, KA		
			HV49-1F076	45	K, KA		
011	HPCI STEAM SUPPLY	HV55-1F002		12*	L, LA	5	55
			HV55-1F003	12*	L, LA		
			HV55-1F100	45	L, LA		
012	RHR SHUTDOWN COOLING SUPPLY	HV51-1F009		100	A,V	9,22	51
		PSV51-155		NA			
			HV51-1F008	100	A,V		
013A(B)	RHR SHUTDOWN COOLING RETURN	HV51-1F050A*(B*) (CK)		NA	A,V	9,22	51
		HV51-151A*(B*)		20	A,V		
			HV51-1F015A(B)	45	A,V		
014	RWCU - SUCTION	HV44-1F001*		10*	B,J,Y		44
			HV44-1F004*	10*	B,J,Y		

TABLE 3.6.3-1 (Continued)

PART A - PRIMARY CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX. ISOL. TIME, IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
016A	CORE SPRAY INJECTION	HV52-1F006A(CK)		NA		9,22	52
		HV52-1F039A		7		9,22	
			HV52-1F005	18			
016B	CORE SPRAY INJECTION	HV52-1F006B(CK)		NA		9,22	52
		HV52-1F039B		7		9,22	
			HV52-108(CK)	NA			
017	RPV HEAD SPRAY	HV51-1F022		60	A,V	4,9,22	51
		PSV51-122		NA		9,22	
			HV51-1F023	135	A,V		
021	SERVICE AIR TO DRYWELL	15-1140		NA			15
			15-1139	NA			
022	DRYWELL PRESSURE INSTRUMENTATION		HV42-147C	45		10	42
023	RECW SUPPLY TO RECIRC PUMPS	HV13-106*		40		11,28, 29	13
			HV13-108*	30		11,28 29	
			HV13-109*	NA		11,13	
024	RECW RETURN FROM RECIRC PUMPS	HV13-107*		40		11,28, 29	13
			HV13-111*	30		11,28, 29	
			HV13-110*	NA		11,13	

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TABLE 3.6.3-1 (Continued)

PART A - PRIMARY CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX. ISOL. TIME, IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
025	DRYWELL PURGE SUPPLY	HV57-121(X-201A)		5**	B,H,S,U,W,R,T	3,11,14	57
		HV57-123		5**	B,H,S,U,W,R,T	3,11,14	
		HV57-163		9	B,H,R,S	3,11,14	
			HV57-109 (X-201A)	6**	B,H,S,U,W,R,T	11	
			HV57-131 (X-201A)	5**	B,H,S,U,W,R,T	11	
			HV57-135	6**	B,H,S,U,W,R,T	11	
026	DRYWELL PURGE EXHAUST	HV57-114		5**	B,H,S,U,W,R,T	3,11,14,33	57
		HV57-111		15**	B,H,S,U,R,T	5,11	
		HV57-161		5	B,H,R,S	3,11,14	
		SV57-139		5		10	
			HV57-115	6**	B,H,S,U,W,R,T	11,33	
			HV57-117	5**	B,H,S,U,R,T	11	
			SV57-145	5	B,H,R,S	11	
027A	CONTAINMENT INSTRUMENT GAS SUPPLY TO ADS VALVES H,M,&S	59-1128(CK)		NA			59
			HV59-151A	45	M		
028A-1	RECIRC LOOP SAMPLE	HV43-1F019		10	B,D		43
			HV43-1F020	10	B,D		
028A-2	DRYWELL H2/O2 SAMPLE	SV57-132		5	B,H,R,S	11	57
			SV57-142	5	B,H,R,S	11	
028A-3	DRYWELL H2/O2 SAMPLE	SV57-134		5	B,H,R,S	11	57
			SV57-144	5	B,H,R,S	11	

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TABLE 3.6.3-1 (Continued)

PART A - PRIMARY CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX. ISOL. TIME. IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
028B	DRYWELL H2/02 SAMPLE	SV57-133		5	B,H,R,S	11	57
			SV57-143	5	B,H,R,S	11	
			SV57-195	5	B,H,R,S	11	
030B-1	DRYWELL PRESSURE INSTRUMENTATION		HV42-147A	45		10	42
035A	TIP PURGE	59-1056(CK) (DOUBLE "O" RING)		NA			59
			HV59-131	7	B,H,S	16	
035C-G	TIP DRIVES	XV59-141A-E (DOUBLE "O" RING)		NA	B,H	11,16,21	59
			XV59-140A-E	NA		11,16	
037A-D	CRD INSERT LINES	BALL CHECK		NA		12	47
			HCU	NA		12	
038A-D	CRD WITHDRAW LINES SDV VENTS & DRAINS		HCU	NA		12	47
			XV47-1F010	25		30	
			XV47-1F180	30		30	
			XV47-1F011	25		30	
			XV47-1F181	30		30	
039A(B)	DRYWELL SPRAY	HV51-1F021A(B)		160		4,11	51
			HV51-1F016A(B)	160		11	
040E	DRYWELL PRESSURE INSTRUMENTATION		HV42-147D	45		10	42
040F-2	CONTAINMENT INSTRUMENT GAS -SUCTION	HV59-101		45	C,H,S	5	59
			HV59-102	7	C,H,S		

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TABLE 3.6.3-1 (Continued)

PART A - PRIMARY CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX. ISOL. TIME. IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
040G-1	ILRT DATA ACQUISITION	60-1057		NA		5,11	60
			60-1058	NA		11	
040G-2	ILRT DATA ACQUISITION	60-1071		NA		5,11	60
			60-1070	NA		11	
040H-1	CONTAINMENT INSTRUMENT GAS SUPPLY - HEADER 'A'	59-1005A(CK)		NA			59
			HV59-129A	7	C,H,S		
042	STANDBY LIQUID CONTROL	48-1F007(CK) (X-116)		NA			48
			HV48-1F006A	60		29	
043B	MAIN STEAM SAMPLE	HV41-1F084		10	B,D		41
			HV41-1F085	10	B,D		
044	RWCU ALTERNATE RETURN	41-1017		NA		5,31	41
			41-1016(X-9A, X-9B)	NA			
			PSV41-112	NA			
045A(B,C,D)	LPCI INJECTION 'A'(B,C,D)	HV51-1F041A*(B,C*, D*)(CK)			NA		9,22
		HV51-142A*(B,C*, D*)		7		9,22	
			HV51-1F017A* (B,C*,D*)	38			
050A-1	DRYWELL PRESSURE INSTRUMENTATION		HV42-147B	45		10	42
053	DRYWELL CHILLED WATER SUPPLY - LOOP 'A'	HV87-128*		60	C,H	11	87
			HV87-120A*	60		11,28, 29	
			HV87-125A*	60		11,28,29	

TABLE 3.6.3-1 (Continued)

PART A - PRIMARY CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX. ISOL. TIME, IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
054	DRYWELL CHILLED WATER RETURN - LOOP 'A'	HV87-129*		60	C,H	11	87
			HV87-121A*	60		11,28, 29	
			HV87-124A*	60		11,28, 29	
055	DRYWELL CHILLED WATER SUPPLY - LOOP 'B'	HV87-122*		60	C,H	11	87
			HV87-120B*	60		11,28, 29	
			HV87-125B*	60		11,28,29	
056	DRYWELL CHILLED WATER RETURN - LOOP 'B'	HV87-123*		60	C,H	11	87
			HV87-121B*	60		11,28,29	
			HV87-124B*	60		11,28,29	
061-1	RECIRC PUMP 'A' SEAL PURGE	43-1004A(CK)		NA		15	43
			(XV43-103A - SEE PART B, THIS TABLE)	NA		1	
061-2	RECIRC PUMP 'B' SEAL PURGE	43-1004B*(CK)		NA		15	43
			(XV43-103B - SEE PART B, THIS TABLE)	NA		1	
062	DRYWELL H2/O2 SAMPLE RETURN, N2 MAKE-UP	SV57-150(X-220A)		5	B,H,R,S	11	57
			SV57-159 (X-220A)	5	B,H,R,S	11	
			HV57-116 (X-220A)	30**	B,H,R,S	11	
			SV57-190 (X-220A)	5	B,H,R,S	11	

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TABLE 3.6.3-1 (Continued)

PART A - PRIMARY CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX. ISOL. TIME, IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
			SV57-191 (X-220A)	5	B,H,R,S	11	
116	STANDBY LIQUID CONTROL	48-1F007(CK) (X-42)		NA			48
			HV48-1F006B	60		29	
117B-1	DRYWELL RADIATION MONITORING SUPPLY	SV26-190A		5	B,H,R,S	11	26
			SV26-190B	5	B,H,R,S	11	
117B-2	DRYWELL RADIATION MONITORING RETURN	SV26-190C		5	B,H,R,S	11	26
			SV26-190D	5	B,H,R,S	11	
201A	SUPPRESSION POOL PURGE SUPPLY	HV57-124		5**	B,H,S,U,W,R,T	3,11,14	57
		HV57-131(X-25)		5**	B,H,S,U,W,R,T	3,11,14	
		HV57-164		9	B,H,R,S	3,11,14	
			HV57-109(X-25)	6**	B,H,S,U,W,R,T	11	
			HV57-147	6**	B,H,S,U,W,R,T	11	
			HV57-121(X-25)	5**	B,H,S,U,W,R,T	11	
202	SUPPRESSION POOL PURGE EXHAUST	HV57-104		5**	B,H,S,U,W,R,T	3,11,14,33	57
		HV57-105		15**	B,H,S,U,R,T	5,11	
		HV57-162		9	B,H,R,S	3,11,14	
			HV57-112	6**	B,H,S,U,W,R,T	11, 33	
			HV57-118	5**	B,H,S,U,R,T	11	
			SV57-185	5	B,H,R,S	11	
203A(B,C,D)	RHR PUMP SUCTION		HV51-1F004A(B, C,D)	240		4,22, 19,29	51
			PSV51-1F030A(B, C,D)	NA		22	

TABLE 3.6.3-1 (Continued)

PART A - PRIMARY CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX. ISOL. TIME. IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
204A(B)	RHR PUMP TEST LINE AND CONTAINMENT COOLING		HV51-125A(B)	180		4,22,29	51
205A(B)	SUPPRESSION POOL SPRAY		HV51-1F027A*(B)	45	C,G	11	51
206A(B,C,D)	CS PUMP SUCTION		HV52-1F001A (B,C,D)	160		4,22,29	52
207A(B)	CS PUMP TEST AND FLUSH		HV52-1F015A(B)	23	C,G	5,22	52
208B	CS PUMP MINIMUM RECIRC		HV52-1F031B	45	LFCH	5,22,29	52
209	HPCI PUMP SUCTION		HV55-1F042	160	L,LA	4,22	55
210	HPCI TURBINE EXHAUST		HV55-1F072	120		4,22,29	55
212	HPCI PUMP TEST AND FLUSH		HV55-1F071	40	B,H	4,22	55
214	RCIC PUMP SUCTION		HV49-1F031	60		4,22,29	49
215	RCIC TURBINE EXHAUST		HV49-1F060	80		4,22,29	49
216	RCIC MINIMUM FLOW		HV49-1F019	8	LFRC	5,22	49

TABLE 3.6.3-1 (Continued)

PART A - PRIMARY CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX. ISOL. TIME, IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
217	RCIC VACUUM PUMP DISCH	HV49-1F002		60		5,29	49
			49-1F028(CK)	NA			
218	INSTRUMENT GAS TO VACUUM RELIEF VALVES	59-1001(CK)		NA			59
			HV59-135	7	C,H,S		
219A	INSTRUMENTATION - SUPPRESSION POOL LEVEL	--	HV55-121	45		10	55
219B	INSTRUMENTATION - SUPPRESSION POOL LEVEL	--	HV55-120	45		10	55
220A	H2/02 SAMPLE RETURN	SV57-191(X-62)		5	B,H,R,S	11	57
			SV57-190(X-62)	5	B,H,R,S	11	
			HV57-116(X-62)	30**	B,H,R,S	11	
			SV57-150(X-62)	5	B,H,R,S	11	
			SV57-159(X-62)	5	B,H,R,S	11	
220B	INSTRUMENTATION - SUPPRESSION POOL PRESSURE SUPPRESSION POOL LEVEL	--	SV57-101	5		10	57
221A	WETWELL H2/02 SAMPLE	SV57-181		5	B,H,R,S	11	57
			SV57-141	5	B,H,R,S	11	
			SV57-184	5	B,H,R,S	11	
221B	WETWELL H2/02 SAMPLE	SV57-183		5	B,H,R,S	11	57
			SV57-186	5	B,H,R,S	11	

TABLE 3.6.3-1 (Continued)

PART A - PRIMARY CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX. ISOL. TIME. IF APP. (SEC)(36)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
225	RHR VACUUM RELIEF SUCTION	HV51-130		60	B,H	4,11	51
			HV51-131	60	B,H	11	
226A	RHR MINIMUM RECIRC		HV51-105A	40		4,22,29	51
226B	RHR MINIMUM RECIRC		HV51-105B	40		4,22,29	51
227	ILRT DATA ACQUISITION SYSTEM	60-1073		NA		5	60
			60-1074	NA			
228D	HPCI VACUUM RELIEF	HV55-1F095		40	H,LA	4,11,24	55
			HV55-1F093	40	H,LA	11,24	
230B	INSTRUMENTATION - DRYWELL SUMP LEVEL	--	HV61-102	45		1,23,29	61
			HV61-112	45		23,29	
			HV61-132	45		23,29	
231A	DRYWELL FLOOR DRAIN SUMP DISCHARGE	HV61-110		30	B,H	11,22	61
			HV61-111	30	B,H	11,22	
231B	DRYWELL EQUIPMENT DRAIN TANK DISCHARGE	HV61-130		30	B,H	11,22	61
			HV61-131	30	B,H	11,22	
235	CS PUMP MINIMUM RECIRC		HV52-1F031A	45		5,22,29	52
236	HPCI PUMP MINIMUM RECIRC		HV55-1F012	15	LFHP	5,22	55

TABLE 3.6.3-1 (Continued)

PART A - PRIMARY CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX. ISOL. TIME, IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
237-1	SUPPRESSION POOL CLEANUP PUMP SUCTION	HV52-127		60	B,H	4,11,22	52
			PSV52-127	NA		11,22	
			HV52-128	60	B,H	11,22	
237-2	SUPPRESSION POOL LEVEL INSTRUMENTATION		HV52-139	45		10	52
			SV52-139	6		10	
238	RHR RELIEF VALVE DISCHARGE		HV-C-51-1F104B	18	C,G		51
			PSV51-106B	NA		19	
			PSV51-1F055B	NA		19	
			PSV51-101B	NA		19	
239	RHR RELIEF VALVE DISCHARGE		HV-C-51-1F103A	18	C,G		51
			PSV51-106A	NA		19	
			PSV51-1F055A	NA		19	
			PSV51-101A	NA		19	
240	RHR RELIEF VALVE DISCHARGE		PSV51-1F097	NA		19	51
241	RCIC VACUUM RELIEF	HV49-1F084		40	H,KA	4,11,24	49
			HV49-1F080	40	H,KA	11,24	

TABLE 3.6.3-1 (Continued)

PART B - PRIMARY CONTAINMENT ISOLATION EXCESS FLOW CHECK VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX.ISOL. TIME.IF APP (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
003A-1	INSTRUMENTATION - 'D' MAIN STEAM LINE FLOW	--	XV41-1F070D XV41-1F073D			1	41
003A-2	INSTRUMENTATION - 'A' RECIRC PUMP SEAL PRESSURE	--	XV43-1F003A			1	43
003C-1	INSTR. - HPCI STEAM FLOW	--	XV55-1F024A			1	55
003C-2	INSTR. - HPCI STEAM FLOW	--	XV55-1F024C			1	55
003D-1	INSTR. - 'A' MAIN STEAM LINE FLOW	--	XV41-1F070A XV41-1F073A			1	41
007A(B,C,D)	INSTR - 'A'(B,C,D) MAIN STEAM LINE PRESSURE	(HV41-1F022A(B, C,D) SEE PART A THIS TABLE)		5	C,D,E,F,P,Q	6	41
			(HV41-1F028A (B,C, D) SEE PART A THIS TABLE)	5	C,D,E,F,P,Q	6	
			(HV40-1F001B (F,K,P) SEE PART A THIS TABLE)			1	
			XV40-101B(F, K,P)			1	
020A-1	INSTR - RPV LEVEL	--	XV42-1F045B			1	42
020A-2	INSTR - 'B' LPCI DELTA P	--	XV51-102B			1	51
020A-3	INSTR - 'D' LPCI DELTA P	--	XV51-103B			1	51
020B-1	INSTR - RPV LEVEL	--	XV42-1F045C			1	42
020B-2	INSTR - 'C' LPCI DELTA P	--	XV51-102C			1	51

TABLE 3.6.3-1 (Continued)

PART B - PRIMARY CONTAINMENT ISOLATION EXCESS FLOW CHECK VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX. ISOL. TIME. IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
027B-1	INSTR - HPCI FLOW	--	XV55-1F024B			1	55
027B-2	INSTR - HPCI FLOW	--	XV55-1F024D			1	55
029A	INSTR - RPV FLANGE LEAKAGE	--	XV41-1F009			1,27	41
029B	INSTR - CS DELTA P	--	XV52-1F018A			1	52
030A	INSTR - 'D' MAIN STEAM FLOW	--	XV41-1F071D XV41-1F072D			1	41
030B-2	INSTR - 'C' MAIN STEAM LINE FLOW	--	XV41-1F071C XV41-1F072C			1	41
031A	INSTR - JET PUMP FLOW	--	XV42-1F059B (JP1) XV42-1F059D (JP2) XV42-1F059F (JP3)			1	42
031B	INSTR - JET PUMP FLOW	--	XV42-1F059H (JP4) XV42-1F051B (JP5) XV42-1F053B (JP6)			1	42

TABLE 3.6.3-1 (Continued)

PART B - PRIMARY CONTAINMENT ISOLATION EXCESS FLOW CHECK VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX. ISOL. TIME, IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&I
032A	INSTR - JET PUMP FLOW	--	XV42-1F059M (JP6) XV42-1F059P (JP7) XV42-1F059S (JP8)			1	42
032B	INSTR - JET PUMP FLOW	--	XV42-1F059U (JP9) XV42-1F051D (JP10) XV42-1F053D (JP10)			1	42
033[illegible]	INSTR-PRESSURE ABOVE CORE PLATE	--	XV42-1F055 XV42-1F076			1	42
033A-2	INSTR-PRESSURE BELOW CORE PLATE	--	XV42-1F061			1	42
033B	INSTR-RCIC STEAM FLOW	--	XV49-1F044A,C			1	49
034A	INSTR - 'C' MAIN STEAM LINE FLOW	--	XV41-1F070C XV41-1073C			1	42
034B-1	INSTR - RECIRC FLOW	--	XV43-1F009C XV43-1F010D			1	43
034B-2	INSTR - RECIRC FLOW	--	XV43-1F009D XV43-1F010C			1	43

TABLE 3.6.3-1 (Continued)

PART B - PRIMARY CONTAINMENT ISOLATION EXCESS FLOW CHECK VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX. ISOL. TIME, IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
040A	INSTR - JET PUMP FLOW	--	XV42-1F059L (JP15) XV42-1F059N (JP17) XV42-1F059R (JP18)			1	42
040B	INSTR - JET PUMP FLOW	--	XV42-1F059G (JP14) XV42-1F051A (JP16) XV42-1F053A (JP16)			1	42
040C	INSTR - JET PUMP FLOW	--	XV42-1F059A (JP11) XV42-1F059C (JP12) XV42-1F059E (JP13)			1	42
040D-1	INSTR - PRESSURE BELOW CORE PLATE	--	XV42-1F057			1	42
040D-2	INSTR - RWCU BOTTOM DRAIN FLOW	--	XV44-170 XV44-171			1	44

TABLE 3.6.3-1 (Continued)

PART B - PRIMARY CONTAINMENT ISOLATION EXCESS FLOW CHECK VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX. ISOL. TIME. IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
040F-1	INSTR - RCIC STEAM FLOW	--	XV49-1F044B XV49-1F044D			1	49
040H-2	INSTR 'B' RECIRC PUMP COOLER FLOW	--	XV87-156B XV87-157B			17	87
041-1	INSTR - RWCU FLOW	--	XV44-102A,B			1	44
041-2	INSTR - 'A' LPCI DELTA P	--	XV51-103A			1	51
043A	INSTR - RECIRC LOOP 'A' DELTA P	--	XV43-1F040A,C			1	43
047	INSTR - RWCU FLOW	--	XV44-102D			1	44
048A-1	INSTR - RPV LEVEL	--	XV42-1F065B XV42-1F047B			1	42
048A-2	INSTR - CS DELTA P	--	XV52-1F018B			1	52
048B	INSTR - RPV LEVEL	--	XV42-1F055A XV42-1F047A			1	42
049A,B	INSTR - 'A' AND 'B' MAIN STEAM LINE FLOW	--	XV41-1F071A,B XV41-1F072A,B			1	41
050A-2	INSTR 'B' RECIRC FLOW	--	XV43-1F011A XV43-1F012B			1	43

TABLE 3.6.3-1 (Continued)

PART B - PRIMARY CONTAINMENT ISOLATION EXCESS FLOW CHECK VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX.ISOL. TIME.IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
050A-3	INSTR 'B' RECIRC FLOW	--	XV43-1F011B XV43-1F012A			1	43
050B-1	INSTR - 'A' RECIRC PUMP SEAL PRESSURE	--	XV43-1F004A			1	43
050B-2	INSTR - 'A' RECIRC PUMP COOLER FLOW	--	XV87-156A XV87-157A			17	87
051A-1	INSTR - 'A' RECIRC LINE FLOW	--	XV43-1F009A XV43-1F010B			1	43
051A-2	INSTR - 'A' RECIRC LINE FLOW	--	XV43-1F009B XV43-1F010A			1	43
051B	INSTR - JET PUMP FLOW	--	XV42-1F059T (JP19) XV42-1F051C (JP20) XV42-1F053C (JP20)			1	42
052A	INSTR - 'B' MAIN STEAM LINE FLOW	--	XV41-1F070B XV41-1F073B			1	41
052B-1	INSTR - 'B' RECIRC LINE FLOW	--	XV43-1F011C,D			1	43
052B-2	INSTR - 'B' RECIRC LINE FLOW	--	XV43-1F012C,D			1	43
057	INSTR - RWCU FLOW	--	XV44-102C			1	44

TABLE 3.6.3-1 (Continued)

PART B - PRIMARY CONTAINMENT ISOLATION EXCESS FLOW CHECK VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX. ISOL. TIME. IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
058A	INSTR - RECIRC LOOP 'B' DELTA P	--	XV43-1F040B			1	43
061-1	RECIRC PUMP SEAL PURGE	(43-1004A(CK) - See Part A of this table)	XV43-103A			15 1	43
061-2	RECIRC PUMP SEAL PURGE	(43-1004B(CK) See Part A of this table)	XV43-103B			15 1	43
063-1	INSTR - RECIRC LOOP 'B' DELTA P	--	XV43-1F040D			1	43
063-2	INSTR - 'B' RECIRC PUMP SEAL PRESSURE	--	XV43-1F004B XV43-1F003B			1	43
065A	INSTR - RPV PRESSURE	--	XV42-1F043B			1	42
065B	INSTR - RPV PRESSURE	--	XV42-1F049A			1	42
066A-1	INST-RPV LEVEL	--	XV42-1F045D			1	42
066A-2	INSTR - 'B' LPCI DELTA P	--	XV51-102D XV51-103D			1	1
066B-1	INST - RPV LEVEL	--	XV42-1F045A			1	42
066B-2	INST - 'A' LPCI DELTA P	--	XV51-102A XV51-103C			1	51
067A	INSTR - RPV PRESSURE	--	XV42-1F049B			1	42

TABLE 3.6.3-1 (Continued)

PART B - PRIMARY CONTAINMENT ISOLAION EXCESS FLOW CHECK VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX. ISOL. TIME. IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
067B-1	INSTR - RPV PRESSURE	--	XV42-1F043A			1	42
067B-2	INSTR - RPV LEVEL	--	XV42-1F041			1	42
102A	INST - JET PUMP, REACTOR LEVEL	--	XV42-185A(JP16)			1	42
107	INST. - JET PUMP, REACTOR LEVEL	--	XV42-185B(JP5)			1	42

TABLE 3.6.3-1 (Continued)

PART C - PRIMARY CONTAINMENT PENETRATIONS (TYPE B)

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX. ISOL. TIME, IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
NA	DRYWELL HEAD FLANGE	DOUBLE O-RING	--	--	--	2	60
001	EQUIPMENT ACCESS DOOR	DOUBLE O-RING	--	--	--	2	60
002	EQUIPMENT ACCESS DOOR AND PERSONNEL LOCK	DOUBLE O-RING	--	--	--	2,18	60
004	HEAD ACCESS MANHOLE	DOUBLE O-RING	--	--	--	2	60
006	CRD REMOVAL HATCH	DOUBLE O-RING	--	--	--	2	60
100A-D	NEUTRON MONITORING SYSTEM	CANISTER	--	--	--	8	60
101A-D	RECIRC PUMP POWER	CANISTER	--	--	--	8	60
103A,B	TEMPERATURE AND LOW LEVEL SIGNALS	CANISTER	--	--	--	8	60
104A-D	CRD POSITION INDICATOR	CANISTER	--	--	--	8	60
105A-E	MISCELLANEOUS LOW-VOLTAGE CONTROL POWER	CANISTER	--	--	--	8	60
106A-C	LOW-VOLTAGE CONTROL	CANISTER	--	--	--	8	60

TABLE 3.6.3-1 (Continued)

PART C - PRIMARY CONTAINMENT PENETRATIONS (TYPE B)

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX. ISOL. TIME. IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
200A,B	ACCESS HATCH	DOUBLE O-RING	--	--	--	2	60
222	INDICATION AND CONTROL	CANISTER	--	--	--	8	60
230A	STRAIN GAUGE INSTR.	CANISTER	--	--	--	8	60

TABLE 3.6.3-1
PRIMARY CONTAINMENT ISOLATION VALVES
NOTATION

NOTES

1. Instrumentation line isolation provisions consist of an orifice and excess flow-check valve or remote manual isolation valve. The excess flow-check valve is subjected to operability testing, but no Type C test is performed or required. The line does not isolate during a LOCA and can leak only if the line or instrument should rupture. Leaktightness of the line is verified during the integrated leak rate test (Type A test).

2. Penetration is sealed by a blind flange or door with double O-ring seals. These seals are leakage rate tested by pressurizing between the O-rings.

3. Inboard butterfly valve tested in the reverse direction.

4. Inboard gate valve tested in the reverse direction.

5. Inboard globe valve tested in the reverse direction.

6. The MSIVs and this penetration are tested by pressurizing between the valves. Testing of the inboard valve in the reverse direction tends to unseat the valve and is therefore conservative. The valves are Type C tested at a test pressure of 22 psig.

7. Gate valve tested in the reverse direction.

8. Electrical penetrations are tested by pressurizing between the seals.

9. The isolation provisions for this penetration consist of two isolation valves and a closed system outside containment. Because a water seal is maintained in these lines by the safeguard piping fill system, the inboard valve may be tested with water. The outboard valve will be pneumatically tested.

10. The valve does not receive an isolation signal but remains open to measure containment conditions post-LOCA. Leaktightness of the penetration is verified during the Type A test. Type C test is not required.

11. All isolation barriers are located outside containment.

12. Leakage monitoring of the control rod drive insert and withdraw line is provided by Type A leakage rate test. Type C test is not required.

13. The motor operators on HV-13-109 and HV-13-110 are not connected to any power supply.

14. Valve is provided with a separate testable seal assembly, with double concentric O-ring seals installed between the pipe flange and valve flange facing primary containment. Leakage through these seals is included within the Type C leakage rate for this penetration.

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TABLE 3.6.3-1
PRIMARY CONTAINMENT ISOLATION VALVES
NOTATION

NOTES (Continued)

15. Check valve used instead of flow orifice.

16. Penetration is sealed by a flange with double O-ring seals. These seals are leakage rate tested by pressurizing between the O-rings. Both the TIP Purge Supply (Penetration 35A) and the TIP Drive Tubes (Penetration 35C-G) are welded to their respective flanges. Leakage through these seals is included in the Type C leakage rate total for this penetration. The ball valves (XV-141A-E) are Type C tested. It is not practicable to leak test the shear valves (XV-140A-E) because squib firing is required for closure. Shear valves (XV-140A-G) are normally open.

17. Instrument line isolation provisions consist of an excess flow check valve. Because the instrument line is connected to a closed cooling water system inside containment, no flow orifice is provided. The line does not isolate during a LOCA and can leak only if the line or instrument should rupture. Leaktightness of the line is verified during the integrated leak rate test (Type A test).

18. In addition to double "O" ring seals, this penetration is tested by pressurizing volume between doors per Specification 4.6.1.3.

19. The RHR system safety pressure relief valves will be exempted from the initial LLRT. The relief valves in these lines will be exposed to containment pressure during the initial ILRT and all subsequent ILRTs. In addition, modifications will be performed at the first refueling to facilitate local testing or removal and bench testing of the relief valves during subsequent LLRTs. Those relief valves which are flanged to facilitate removal will be equipped with double O-ring seal assemblies on the flange closest to primary containment by the end of the first refueling outage. These seals will be leak rate tested by pressurizing between the O-rings, and the results added into the Type C total for this penetration.

20. See Specification 3.3.2, Table 3.3.2-1, for a description of the PCRVICS isolation signal(s) that initiate closure of each automatic isolation valve. In addition, the following non-PCRVICS isolation signals also initiate closure of selected valves:

    EA Main steam line high pressure, high steam line leakage flow, low MSIV-LCS dilution air flow

    LFHP With HPCI pumps running, opens on low flow in associated pipe, closes when flow is above setpoint

    LFRC With RCIC pump running, opens on low flow in associated pipe, closes when flow is above setpoint

    LFCH With CSS pump running, opens on low flow in associated pipe, closes when flow is above setpoint

    LFCC Steam supply valve fully closed or RCIC turbine stop valve fully closed

All power operated isolation valves may be opened or closed remote manually.

TABLE 3.6.3-1
PRIMARY CONTAINMENT ISOLATION VALVES
NOTATION

NOTES (Continued)

21. Automatic isolation signal causes TIP to retract; ball valve closes when probe is fully retracted.

22. Isolation barrier remains water filled or a water seal remains in the line post-LOCA. Isolation valve may be tested with water. Isolation valve leakage is not included in 0.60 La total Type B & C tests.

23. Valve does not receive an isolation signal. Valves will be open during Type A test. Type C test not required.

24. Both isolation signals required for valve closure.

25. Deleted

26. Valve stroke times listed are maximum times verified by testing per Specification 4.0.5 acceptance criteria. The closure times for isolation valves in lines in which high-energy line breaks could occur are identified with a single asterisk. The closure times for isolation valves in lines which provide an open path from the containment to the environs are identified with a double asterisk.

27. The reactor vessel head seal leak detection line (penetration 29A) excess flow check valve is not subject to OPERABILITY testing. This valve will not be exposed to primary system pressure except under the unlikely conditions of a seal failure where it could be partially pressurized to reactor pressure. Any leakage path is restricted at the source; therefore, this valve need not be OPERABILITY tested.

28. Automatic isolation logic to be added by the end of the first refueling outage.

29. Valve may be open during normal operation; capable of manual isolation from control room. Position will be controlled procedurally.

30. Valve normally open, closes on scram signal.

31. Valve 41-1016 is an outboard isolation barrier for penetrations X-9A, B and X-44. Leakage through valve 41-1016 is included in the total for penetration X-44 only.

32. Feedwater long-path recirculation valves are sealed closed whenever the reactor is critical and reactor pressure is greater than 600 psig. The valves are expected to be opened only in the following instances:

    a. Flushing of the condensate and feedwater systems during plant startup.

    b. Reactor pressure vessel hydrostatic testing, which is conducted following each refueling outage prior to commencing plant startup.

    Therefore, valve stroke timing in accordance with Specification 4.0.5 is not required.

33. Valve also constitutes a Refueling Area Secondary Containment Automatic Isolation Valve as shown in Table 3.6.5.2.2-1.

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3/4.6.4 VACUUM RELIEF

SUPPRESSION CHAMBER - DRYWELL VACUUM BREAKERS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Each pair of suppression chamber - drywell vacuum breakers shall be OPERABLE and closed.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or more vacuum breakers in one pair of suppression chamber - drywell vacuum breakers inoperable for opening but known to be closed, restore the inoperable pair of vacuum breakers to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one suppression chamber - drywell vacuum breaker open, verify the other vacuum breaker in the pair to be closed within 2 hours; restore the open vacuum breaker to the closed position within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With one position indicator of any suppression chamber - drywell vacuum breaker inoperable:

   1. Verify the other vacuum breaker in the pair to be closed within 2 hours and at least once per 15 days thereafter, or

   2. Verify the vacuum breaker(s) with the inoperable position indicator to be closed by conducting a test which demonstrates that the ΔP is maintained at greater than or equal to 0.7 psi for one hour without makeup within 24 hours and at least once per 15 days thereafter.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each suppression chamber - drywell vacuum breaker shall be:

a. Verified closed at least once per 7 days.

b. Demonstrated OPERABLE:

1. At least once per 31 days and within 2 hours after any discharge of steam to the suppression chamber from the safety/relief valves, by cycling each vacuum breaker through at least one complete cycle of full travel.

2. At least once per 31 days by verifying both position indicators OPERABLE by observing expected valve movement during the cycling test.

3. At least once per 18 months by;

a) Verifying each valve's opening setpoint, from the closed position, to be 0.5 psid ± 5%, and

b) Verifying both position indicators OPERABLE by performance of a CHANNEL CALIBRATION.

c) Verifying that each outboard valve's position indicator is capable of detecting disk displacement $\geq$0.050", and each inboard valve's position indicator* is capable of detecting disk displacement $\geq$0.120".

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3/4.6.5 SECONDARY CONTAINMENT

REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.1.1 REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

Without REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY, restore REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1.1 REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

a. Verifying at least once per 24 hours that the pressure within the reactor enclosure secondary containment is greater than or equal to 0.25 inch of vacuum water gauge.

b. Verifying at least once per 31 days that:

   1. All reactor enclosure secondary containment equipment hatches and blowout panels are closed and sealed.

   2. At least one door in each access to the reactor enclosure secondary containment is closed.

   3. All reactor enclosure secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers/valves and required to be closed during accident conditions are closed by valves, blind flanges, slide gate dampers or deactivated automatic dampers/valves secured in position.

c. At least once per 18 months:

   1. Verifying that one standby gas treatment subsystem will draw down the reactor enclosure secondary containment to greater than or equal to 0.25 inch of vacuum water gauge in less than or equal to 121 seconds with the reactor enclosure recirc system in operation, and

   2. Operating one standby gas treatment subsystem for one hour and maintaining greater than or equal to 0.25 inch of vacuum water gauge in the reactor enclosure secondary containment at a flow rate not exceeding 1250 cfm.

JUL 8 1987

ATTACHMENT 3

PHILADELPHIA ELECTRIC COMPANY

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GRAHAM M. LEITCH
VICE PRESIDENT
LIMERICK GENERATING STATION

June 13, 1988

Mr. Robert M. Gallo, Chief
Operations Branch
Division of Reactor Safety
U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

Subject: LO/SLO Written Exam Comments for NRC Inspection #88-16

Dear Mr. Gallo:

Attached you will find the Limerick Operations/Training Section responses to selected questions and answers associated with the NRC RO/SRO written examinations recently administered on 06/07/88 at the Limerick Generating Station. These comments are submitted in the interests of clarifying those areas where alternate correct answers should be considered as well as providing updated Limerick specific information which can be used in the process of evaluation of these exams. Wherever possible, references have been indicated or added which justify the comment made or question/answer clarified. Also, a separate sheet is attached to indicate discrepancies in the point values of exam questions and/or answer keys.

The written examination for both the RO and SRO candidates did appear to be quite lengthy in that all of the candidates took the entire six (6) hours or more with little time for review of their answers. Compared to the last Hot License exams, which were administered on 10/20/86, it appears that these recent exams had many more subparts within each question. For example, the

10/20/86 RO exam had 47 questions with 81 total subparts, whereas the 06/07/88 RO exam had 42 questions with 136 total subparts. Now, although there is absolutely no concern about exam comprehensiveness, there is a concern that the many question subparts in both exams may have caused undue stress on several candidates to complete the exam within the six (6) hours allotted.

If there are any questions concerning these comments, please contact E. G. Firth, LGS Superintendent-Training, at 327-1200 (x2080).

Very truly yours,

EGF:mgd

Attachments

c.c. M. J. McCormick, Jr.
J. Doering
E. G. Firth
R. A. Nunez
J. F. Hanek - EG&G Idaho, Inc.

NRC SENIOR REACTOR OPERATOR EXAM (06/07/88) POINT VALUE DISCREPANCIES

EXAM

Section 5 - None

Section 6 - Question 6.05

Total Point Value Given	=	(3.00)
Breakdown Point Value	=	(2.50)
Category 6 Total Points would	=	(25.00) if worth (3.00)

Section 7 - Question 7.06

Total Point Value Given	=	(2.50)
Breakdown Point Value	=	(3.00)
Category 7 Total Point would	=	(25.00) if worth (2.50)

Section 8 - None

ANSWER KEY

Section 5 - Question 5.09 part (a) states (1.00), should state (0.50)

Section 6 - Question 6.05 - same as exam

Section 7 - Question 7.06 - same as exam
Question 7.08-part (d) should be worth (0.50) vice (0.25)

NRC REACTOR OPERATOR EXAM (06/07/88) RESPONSES

Category 1: Principles of Nuclear Power Plant Operations, Thermodynamics, Heat Transfer and Fluid Flow

1.01 d. Answer can be either TRUE or FALSE. The phrase "as a result of" is vague, and can be interpreted two ways by the candidate.

1) Delayed neutrons are produced upon decay of delayed neutron precursors, and not from fission of U-235 and U-238 directly, therefore the answer is FALSE.

2) Delayed neutrons are eventually produced after the fission of U-235 and U-238, therefore the answer is TRUE.

REF: LOT 0870 I.C.2.a

1.02 d. Answer should be SDM INCREASES. Increasing Pu-240, an absorber of epithermal neutrons, reduces k_{eff} in the core, which results in an increased SDM.

REF: LOT 0950 Typical Question 6.d

1.05 a. Additional correct answer: INCREASING TEMPERATURE OF RECIRC PUMP SUCTION. During startup, Recirc Pump Suction Temperature is monitored, and is considered to be Reactor Water Temperature for an operating Recirc pump. Reactor Water temperature is recorded and monitored to the heating range per GP-2.

REF: GP-2, Sect. 3.3.25

1.05 b. Interval 2 and Interval 3 are both correct answers.

The period for Interval 2 is 90.11 sec.
The period for Interval 3 is 80.03 sec.

Both are greater than 80 seconds, and either could be an interval in which the heating range was entered.

REF: LOT 1430, Obj. 4

08 Candidates may discuss how the reactivity coefficients change when feedwater temperature is reduced, as well as how the coefficients affect the transient, and should not be penalized for this.

REF: LOT 1450; LOT 1460; LOT 1480

1.08 c. FSAR analysis assumes a decrease in Feedwater temperature of 100 degrees F. The isolation of a feedwater train would result in less of a change in temperature than 100 degrees F, and with appropriate operator action, a scram on high flux can be avoided. Operator action per OT-104 "Unexplained Reactivity Insertion" would result in voids decreasing followed by voids increasing.

REF: OT-104; LOT-1460, Typical Question #1; FSAR pp 15.1-1 through 15.1-4

1.09 This question is similar to SRO 5.08. The wording is awkward and extremely confusing. The question is not based on a Learning Objective

REF: LOT 1300

1.10 a. Additional correct answers are:

REACTOR POWER - Class discussion included graphing NPSH vs Reactor Power for Recirc pumps. Reactor Power is a parameter available to the operator on the Process Computer.

RECIRC SUCTION TEMPERATURE - Saturation temperature is directly related to NPSH, and is a parameter available to the operator on the Process Computer.

RECIRC PUMP SPEED - Friction loss is affected by pump speed and is directly related to NPSH. It is available on a meter on the C601 panel in the main control room.

REF: LOT 1290, pp 8 & 9

1.11 a. "Bundle" and "Assembly" are used interchangeably, and both should be accepted.

REF: LOT 1370, p 5

1.12 a. Additional correct answer:

$MAPLHGR/APLHGR_{LCO}$

Although LOT 1410 uses $MAPLHGR_{LCO}$, Tech Spec 3.2.1 defines the LCO for APLHGR, and not MAPLHGR.

REF: Tech Spec LCO 3.2.1

Category 2: Plant Design Including Safety and Emergency Systems

2.02 c. Both WOULD or WOULD NOT are correct answers, depending on candidate's assumptions.

1) If B RHRSW is providing cooling water to the B RHR heat exchanger, it will trip on a LOCA signal, since it is powered by the Unit 1 D-12 Bus, and WOULD NOT is the correct answer.

2) If D RHRSW is providing cooling water to the B RHR heat exchanger, it will not trip on a LOCA signal, since it is powered by the Unit 2 D-22 Bus, and WOULD is the correct answer.

REF: LOT 0400, p 9

2.04 a. Correct answer is 118 seconds. MOD 86-189 changed this time delay from 58 seconds to 118 seconds. This is referenced in S36.1.B, section 8.2 NOTE: "116 to 120 seconds after manual initiation a 10 minute timer is enabled."

REF: S36.1.B, step 8.2 Note, step 8.3

2.04 c. Additional correct answers:

1) Decreasing tank level

2) Decreasing reactor power

3) Continuity Lights extinguish

REF: LOT 0310, p 17

2.05 a. 2. Correct answer is 455 psig.

REF: LOT 0350, p 8

2.05 a. 3. Question should be deleted. Check valve will open when discharge pressure is greater than reactor pressure. Discharge pressure is not required knowledge for RO.

REF: LOT 0350, p 8

2.05 b. Design pressure should be deleted from this question. Design flow is required for RO knowledge. Design pressure for rated flow is not required knowledge for RO.

REF: LOT 0350, Obj. 5a

2.06 e. Suppression pool spray and Full flow test valves also close. Candidates should not be penalized for including these in the answer.

REF: LOT 0370, p 14

2.08 c. Answer should be either NO SYSTEM or ESW, depending on candidate's interpretation of question. ESW can be used to cool the RECW heat exchanger, and can provide cooling to the Recirc pump seal and motor oil coolers which are normally cooled by RECW, but ESW cannot supply flow to the entire RECW system. Depending on assumptions made by the candidate, either NO SYSTEM or ESW is an appropriate answer.

REF: LOT 0460, pp 11-14

2.09 a. Additional correct answers:

OVERSPEED

LOW CONDENSER VACUUM

Each overspeed trip is independent of the other feedpumps, and could have caused one feedpump to trip.

Each RFP turbine exhausts to a different shell of the main condenser, independent of the other turbines, and could have caused one feedpump to trip.

REF: LOT 0540, pp 6 & 13

Category 3: Instruments and Controls

3.01 a. Delete this question. The Recirc Pump Master Controller is not used at LGS, and the M/A Transfer Station is not used in auto. This question should be deleted since it does not apply to LGS.

REF: S43.1.a, step 8.20; S43.2.A, step 8.1; LOT 0040, pp 6 & 7

3.01 b. 75% flow is equivalent to approximately 60% pump speed. Either is an acceptable answer.

REF: LOT 0040, p 8

3.01 c. Delete this question. The Recirc Pump Master Controller is not used at LGS, and the M/A Transfer Station is not used in auto. This section should be deleted since it does not apply to LGS.

REF: S43.1.A, step 8.20; S43.2.a, step 8.1; LOT 0040, pp 6 & 7

3.02 b. Both HIGHER THAN ACTUAL and SAME AS ACTUAL are correct answers. Wide range is calibrated with no jet pump flow, and would read the same as actual assuming no jet pump flow. Assuming jet pump flow, wide range would read higher than actual.

REF: LOT 0050, pp 9 & 10

3.03 a. (Similar to SRO 6.05) Candidate should not be penalized for including LOW CONDENSER VACUUM 10.5 PSIA as an additional answer, since this could be bypassed or active in startup, depending on assumptions made. Candidate should not be penalized for including MANUAL, due to the wording of question.

REF: Tech Spec 3.3.2 notation ** on page 3/4 3-16; LOT 0120, p 18

3.04 d. (Similar to SRO 6.10.d)

Action 3 is not a correct answer. If the reactor is in startup mode and power reaches 15%, a scram signal is expected. A rod block should have been generated at 12% power. If rods can be withdrawn to allow power to reach 15%, the rod block signal has failed, and a scram signal is now expected. Action 2 is the only correct answer.

REF: LOT 0270 pp 11-13, Obj. 7

3.04 e. NO ACTION is also a correct answer. Since criticality can be reached at any time, a rod block will not always be generated in this condition.

REF: GP-2, App 1, steps 3.1.5; 3.1.6; 3.1.7; on pp 3 & 4

3.06 Numbers and calculations on LOT 0590-6 should be accepted in lieu of discussions. Candidate should not be penalized for continuing the transient until the plant stabilizes.

REF: None

3.07 a. Standby Gas Treatment System will also start if aligned to the Refuel Floor Ventilation system.

REF: P&ID M-76, sh 5 & 6

Category 4: Procedures - Normal, Abnormal, Emergency, and Radiological control

4.01 a. Directions in A-7 for an RO to shutdown the plant or scram are covered in two sections of A-7, using different wording. Section 5.1.2.4 is as appropriate as section 5.2.8 as an answer.

REF: A-7, sections 5.2.8; 5.1.2.4

4.05 b. Safety Related Equipment is equipment contained on the Q List or QAD, and is commonly referred to as "Q-listed equipment".

REF: A-41, section 4.1; SRO exam 8.06 a.

4.05 d. The permission to release equipment for the surveillance is an SRO responsibility. This question should be deleted from this RO exam.

REF: A-41, section 5.1.2

4.05 c. YES is the correct answer. This question is vague. A-41 allows IVOR "need not be performed in cases where such verification would result in significant radiation exposure, as determined by Shift Supervision." A "valve in a high radiation area" may or may not result in significant radiation exposure. This would be determined by Shift Supervision, and would not be automatically excluded from an IVOR.

REF: A-41, section 5.3

4.07 a. Answers on answer key are redundant:

Additional correct answer:

TO REMIND OPERATOR THAT THE MIN FLOW VALVE IS BLOCKED CLOSED.

This is because of the basic purpose of the Blocking Procedures, i.e., assure safety of workers and avoid equipment damage.

REF: LOT 1860 Obj. 1; p 2

NRC SENIOR REACTOR OPERATOR EXAM (06/07/88) RESPONSES

Category 5: Theory of Nuclear Power Plant Operation, Fluids and Thermodynamics

5.04 a. Answer can be either LARGER or SMALLER depending on assumptions (which were not required by the question). When dealing with how moving a control rod will effect overall thermal power, many variables come into play and the final answer will be dependent on assumptions.

REF: LOT 1490, pgs 4-12

5.05 SROs should not lose credit if the problem was worked both at 100°F/hr and at less than 100°F/hr cooldown rates since GP-3 note states cooldown is limited to ≤ 100 degrees F per hour.

REF: GP-3, p 16

5.08 c. Can be confusing to the candidates since steam flow is not directly inputted in a manual core thermal power evaluation as per RE-101. Consider alternate answer depending on assumptions.

REF: RE-101, Core Thermal Power Evaluation

5.08 d. Answer is incorrect. The correct answer should be FALSE vice TRUE. The Reactor Operator Exam question 1.09 part (d) is the same and it's answer is given as FALSE. This answer is correct and supported by the heat balance calculation.

REF: Reactor Operator Exam, Question 1.09 (d)

5.09 a. Additional correct answer:

$MAPLHGR/APLHGR_{LCO}$

Although LOT-1410 uses $MAPLHGR_{LCO}$, Technical Specification 3.2.1 defines the LCO for APLHGR, and not MAPLHGR.

REF: Technical Specification 3.2.1

5.10 Answers should be accepted depending on SRO candidates supporting information. The candidates were not provided the note from Table 15.2-11 which is referenced on Figure 15.2-9 which they were given. In addition, Table 15.0-2 was not provided to the candidates which lists the input parameters and initial conditions for transients. Also, in accordance with the objectives for LOT-1580, BWR Transient Analysis, the candidates were not held responsible for a Loss of All Feedwater Flow Transient. Due to these reasons, a range of answers should be accepted based on student's assumptions/justifications.

REF: FSAR, Figure 15.2-9, Table 15.0-2, Table 15.2-11; LOT-1580, p 1

5.11 a. Subjective question dealing with water hammer. Candidates could justify choices 1, 2, or 3. Many flow changes (starting flow, cessation of low) can be linked to cases of water hammer. For this reason, acceptable choices are 1, 2, or 3.

REF: LOT 1291, pgs 1-7

Category 6: Plant Systems Design, Control, and Instrumentation

6.01 Question is worded such that it could be misinterpreted since the HPCI turbine trip is at 15" Hg Vac and the question implies that the HPCI pump is still working with a suction of 18" Hg Vac. Therefore, answer (a) "Continue to inject" can be justified due to the fact the pump did not trip at 15" Hg Vac.

REF: LOT 0340, pgs 11, 12

6.03 a. Alternate answer is NOTHING. Given the conditions stated in the question, recirc pump M/A stations may already be at 28% or lower. The operating map referenced in the question does not specify "speed" only flow which can be due to natural circulation and/or forced circulation. For this reason, if the recirc pumps are already at or below 28% speed, nothing will occur when feedflow drops below 20%.

REF: LOT 0040, pgs 25, 26, 27

6.04 a. Correct answer is 118 seconds. MOD 86-189 changed this time delay from 58 seconds to 118 seconds. This is referenced in S36.1.B section 8.2 note, "116 to 120 seconds after manual initiation, a 10 minute time is enabled".

REF: S36.1.B section 8.2 NOTE, step 8.3

6.04 b. Additional correct answers:

1) Decreasing tank level

2) Decreasing reactor power

3) Continuity lights extinguish

REF: LOT 0310, pgs 10, 17

6.05 a. Additional acceptable answers are:

Condenser vacuum low
Manual

Since the question did not specifically set other conditions or state "Automatic", these answers are acceptable in accordance with Technical Specifications depending on candidates interpretation of the question.

REF: Technical Specifications LCO 3.3.2, Table 3.3.2-1, Table 3.3.2-2

6.07 a. Standby Gas Treatment System will also start if aligned to the Refuel Floor Ventilation System via the Slide Gate Dampers.

REF: P&ID M-76, shts 5 and 6

6.08 During post exam review, the examiner was made aware of the error in the answer key dealing with the EHC numbers. Answer should state 25 vice 40 and 18% vice 17%.

6.10 a. Question was worded such that interpretation of the phrase "...also has channel A of the IRM's selected" could be the IRM/APRM selector switch on the C603 panel. This is due to the use of the word "channel" instead of the word "detector". Alternate answer in this case would be NONE.

REF: LOT 0250, pgs 10-14; LOT 0270, pgs 11, 14

6.10 d. Alternate acceptable answer would not include the rod block as part of the answer. This is due to the fact that the candidate may interpret the question as those actions which occur at 15% power. Since the rod block occurs at 12%, they may only say scram signal (which is generated at 15% in startup).

REF: Technical Specifications, Table 2.2.1-1, Table 3.3.6-2

<u>Category 7</u>: Procedures - Normal, Abnormal, Emergency, and Radiological Control

7.02 b. Answer key breaks down point values into credit for calculation and credit for formula. The question does not ask the SRO candidate to show this in his answer. Request that the formula and calculation not be required for full credit.

REF: SRO Exam Question 7.02, Answer Key 7.02

7.04 a. The question asks the candidate for two purposes. The caution in S51.8.B shows that the two purposes given in the answer are in essence saying the same. Either should be counted for full credit and other answers given by the SROs should not cause a loss of credit since the question asked to give two different purposes.

REF: S51.8.B, pg 5

7.06 a. and b. Confusion exists as to what the question was asking for versus what the answer key was answering. If the question meant to state level was oscillating ± 5 inches, did this mean from normal to +5 and back to normal; or did it mean from normal level ± 5 inches (question only lists +5 inches, which would not be an oscillation so it's assumed it was meant to be ± 5 inches).

In either case, the following OT procedures could be entered based on assumptions.

OT-100 Low Level
OT-110 Hi Level
OT-104 Unexplained Reactivity Insertion

The SRO's answer to part (b) will then be in accordance with part (a). The answer key lists two OTs in part (a) but only addresses one in part (b).

Acceptable answers will be according to question interpretation.

Also, part (b) question states "What actions would you, as the Shift Supervisor, direct the operator to take?" The Shift Supervisor would not direct the operator to enter T-100 if a scram occurs but rather would enter T-100 and direct the operators accordingly. Therefore, this step should not be counted as part of the answer to part (b).

In addition, the SRO candidate may enter OT-104, Unexplained Reactivity Insertion, in response to the power increase which the reactor may undergo when level oscillates so this answer should be accepted.

REF: OT-100, pg 1; OT-104, pg 1; OT-110, pg 1

7.07 c. Answer key typographical error. Answer should read ON-107 vice ON-105, for "Control Rod Drive System Problems".

REF: Off Normal Procedure Index

7.07 d. Acceptable answer is entry into ON-109, "Total Loss of the SRM, IRM, or APRM Systems". This is because Technical Specifications require the SRM Upscale and Inoperative TRIPS be operable until the IRMs are on range 8 or higher in order to meet LCO 3.3.6 "Control Rod Block Instrumentation".

REF: Technical Specifications 3.3.6, Table 3.3.6-1, Table 3.3.6-2

7.09 b. Alternate acceptable answer is 3. In T-112, steps EB-7, EB-8 and EB-9 ask for 5 ADS valves or ADS/SRV valves. However, in step EB-10, it asks the operator "Are at least 3 ADS/SRVs open". For this reason either 3 or 5 are acceptable answers.

REF: T-112, Emergency Blowdown

7.10 b. Acceptable answer according to EP-101 is classifying this as an Unusual Event since for an Alert classification, Steam Jet Air Ejector Discharge Radiation monitor exceeds 2.1P5 mR/hr. Question states 210 R/hr. For conservatism, the SRO candidate may answer the same as the answer key, which states Alert. However, strict interpretation could lead the SRO candidate to only classify as an Unusual Event. Both answers should be accepted.

REF: EP-101, pg 13, Damage of Fuel

Category 8: Administrative Procedures, Conditions, and Limitations

8.01 1. Answer key is incorrect. This is a safety limit violation in accordance with Technical Specifications for Thermal Power, Low Pressure or Low Flow.

REF: Technical Specification 2.1.1

8.03 a. Alternate correct answer should include the portion of A-7 dealing with job responsibilities of the Control Supervisor (section 5.1.2.2) and Control Room Operators (section 5.1.2.4). Although the wording is different than section 5.2.8 of A-7 dealing with Initiation of a Scram or Shutdown, the intent is the same and the SROs should be given credit.

REF: A-7, pgs 9, 10, 11, 17

8.03 b. (1) Alternate acceptable answers should be his designated alternate or Plant Manager. Due to the reorganization within PECo, the title will change to Plant Manager vice Station Superintendent. A-7 states Station Superintendent or his designated alternate.

REF: A-7, 5.2.12

8.03 b. (2) Alternate acceptable answer should be Shift Supervision. They are Senior Licensed Operators.

REF: A-7, 5.2.12

8.05 a. Answer key states that permission to perform an ST may be delegated. In accordance with A-43, this may not be delegated. Shift Supervision's permission to perform an ST must be given as well as having the responsible CO/ACO's permission. This permission can not be delegated.

REF: A-43, p 2, part 2.4

8.06 c. An acceptable alternate answer is (COL) in that an independent verification of a permit does not have to be done if the equipment check off list (COL) is going to be performed. This is in accordance with procedure A-41 sections 5.4.1 and 5.4.4

REF: A-41, pg 12, parts 5.4.1 and 5.4.4

8.10 Candidate answers should be graded based upon what interpretative Tech Spec logic they used. The answer key takes the conservative approach by placing the "A" channel in the tripped condition. However, a strict interpretation of Technical Specifications would lead the candidate to come to the conclusion that no actions are required since Tech Spec Table 3.3.1-1 does not require the IRM neutron flux high trip to be OPERABLE in OPCON 1, and table notation (b) states that the function is automatically bypassed when the mode switch is in RUN and the associated APRM is not downscale (given in the question).

REF: Technical Specification 3.3.1

ATTACHMENT 4

NRC Resolution of Facility Comments on SRO Examination Administered on June 7, 1988

Category 5: Theory of Nuclear Power Plant Operation, Fluids and Thermodynamics

5.04(a). Answer can be either LARGER or SMALLER depending on assumptions (which were not required by the question). When dealing with how moving a control rod will effect overall thermal power, many variables come into play and the final answer will be dependent on assumptions.

REF: LOT 1490, pgs. 4-12

NRC Resolution: Rejected (This question compares deep rod movement with shallow rod movement. Movement of a deep rod has larger effect on overall core thermal power. No questions were asked about other assumptions during the examination).

5.05. SROs should not lose credit if the problem was worked both at 100°F/hr and at less than 100°F/hr cooldown rates since GP-3 note states cooldown is limited to ≤ 100 degrees F per hour.

REF: GP-3, p 16

NRC Resolution: Rejected per GP-3 note, maximum cool down rate is 100°F/hr. However, partial credit will be given for method used for assumptions other than 100°F/hr.

5.08(c). Can be confusing to the candidates since steam flow is not directly inputted in a manual core thermal power evaluation as per RE-101. Consider alternate answer depending on assumptions.

REF: RE-101, Core Thermal Power Evaluation

NRC Resolution: Rejected (Candidate should know that steam flow enters into the heat balance, although RE-101 assures steam flow = feedwater flow).

5.08(d). Answer is incorrect. The correct answer should be FALSE vice TRUE. The Reactor Operator Exam question 1.09 part (d) is the same and it's answer is given as FALSE. This answer is correct and supported by the heat balance calculation.

REF: Reactor Operator Exam, Question 1.09 (d)

NRC Resolution: Accepted (changed answer key)

5.09(a) Additional correct answer:

$MAPLHGR/APLHGR_{LOC}$

Although LOT-1410 uses $MAPLHGR/APLHGR_{LOC}$, Technical Specification 3.2.1 defines the LCO for APLHGR, and not MAPLHGR.

REF: Technical Specification 3.2.1

NRC Resolution: Partially accepted (The answer key changed to the correct definition in TS, i.e., MAPLHGR/APLHGR-LCO.

5.10. Answers should be accepted depending on SRO candidates supporting information. The candidates were not provided the note from Table 15.2-11 which is referenced on Figure 15.2-9 which they were given. In addition, Table 15.0-2 was not provided to the candidates which lists the input parameters and initial conditions for transients. Also, in accordance with the objectives for LOT-1580, BWR Transient Analysis, the candidates were not held responsible for a Loss of All Feedwater Flow Transient. Due to these reasons, a range of answers should be accepted based on student's assumptions/justifications.

REF: FSAR, Figure 15.2-9, Table 15.0-2, Table 15.2-11; LOT-1570, p 1

NRC Resolution: Accepted (Exact wording not required, candidate's correct assumption/justification will be considerate in the grading)

5.11(a). Subjective question dealing with water hammer. Candidates could justify choices 1, 2, or 3. Many flow changes (starting flow, cessation of low) can be linked to cases of water hammer. For this reason, acceptable choices are 1, 2, or 3.

REF: LOT 1291, pgs. 1-7

NRC Resolution: Partially accepted (Question asks to identify most likely cause of water hammer. Answers 1 and 2 are more likely than Answers 3 and 4. Changed Answer Key to 1 or 2).

Category 6: Plant Systems Design, Control, and Instrumentation.

6.01. Question is worded such that it could be misinterpreted since the HPCI turbine trip is at 15" Hg Vac and the question implies that the HPCI pump is still working with a suction of 18" Hg Vac. Therefore, answer (a) "Continue to inject" can be justified due to the fact the pump did not trip at 15" Hg Vac.

REF: LOT 0340, pgs. 11, 12

NRC Resolution: Rejected (the wording of the question is clear to indicate pertinent parameters. An assumption of a failure to trip is not indicated on the question).

6.03(a). Alternate answer is NOTHING. Given the conditions stated in the question, recirc pump M/A stations may already be at 28% or lower. The operating map referenced in the question does not specify "speed" only flow which can be due to natural circulation and/or forced circulation. For this reason, if the recirc pumps are already at or below 28% speed, nothing will occur when feedflow drops below 20%.

REF: LOT 0040, pgs. 25, 26, 27

NRC Resolution: Accepted (Changed Answer Key)

6.04(a). Correct answer is 118 seconds. Mod 86-189 changed this time delay from 58 seconds to 118 seconds. This is referenced in §36.1.B section 8.2 note, "116 to 120 seconds after manual initiation, a 10 minute time is enabled".

REF: §36.1B section 8.2 NOTE, step 8.3

NRC Resolution: Accepted (Changed Answer Key)

6.04(b). Additional correct answers:

1) Decreasing tank level

2) Decreasing reactor power

3) Continuity lights extinguish

REF: LOT 0310, pgs. 10, 17

NRC Resolution: See resolution to similar RO Question

Changed the Answer Key to include the additional answer

6.05(a). Additional acceptable answers are:

Condenser vacuum low
Manual

Since the question did not specifically set other conditions or state "Automatic", these answers are acceptable in accordance with Technical Specifications depending on candidates interpretation of the question.

REF: Technical Specifications LCO 3.3.2, Table 3.3.2-1, Table 3.3.2-2

NRC Resolution: Partially accepted (Condenser vacuum low is not acceptable because safety valves, which bypass the scram signal, are not open until after 2 BPVs are open. This would not occur until the pressure is greater than 920 psig. Reference: F01.1.a and GP-2. However, Manual is an additional acceptable answer. Answer Key modified to include this and revised point value distribution).

6.07(a). Standby Gas Treatment System will also start if aligned to the Refuel Floor Ventilation System via the Slide Gate Dampers.

REF: P&ID M-76, shts 5 and 6

NRC Resolution: Accepted (Changed Answer Key)

6.08. During post exam review, the examiner was made aware of the error in the answer key dealing with the EHC numbers. Answer should state 25 vice 40 and 18% vice 17%

NRC Resolution: Accepted (Changed Answer Key)

6.10(a). Question was worded such that interpretation of the phrase, "... also has channel A of the IRM"s selected" could be the IRM/APRM selector switch on the C603 panel. This is due to the use of the word "channel" instead of the word "detector". Alternate answer in this case would be NONE.

REF: LOT 0250, pgs. 10-14; LOT 0270, pgs. 11, 14

NRC Resolution: Accepted (Changed Answer Key)

6.10(d). Alternate acceptable answer would not include the rod block as part of the answer. This is due to the fact that the candidate may interpret the question as those actions which occur at 15% power. Since the rod block occurs at 12%, they may only say scram signal (which is generated at 15% in startup).

REF: Technical Specifications, Table 2.2.1-1, Table 3.3.6-2

NRC Resolution: Accepted (Changed Answer Key)

Category 7: Procedures - Normal, Abnormal, Emergency, and Radiological Control

7.02(b). Answer key breaks down point values into credit for calculation and credit for formula. The question does not ask the SRO candidate to show this in his answer. Request that the formula and calculation not be required for full credit.

REF: SRO Exam Question 7.02, Answer Key 7.02

NRC Resolution: Accepted (Changed Answer Key).

7.04(a) The question asks the candidate for two purposes. The caution in S51.8.B shows that the two purposes given in the answer are, in essence, saying the same. Either should be counted for full credit and other answers given by the SROs should not cause a loss of credit since the question asked to give two different purposes.

REF: §51.8.B, pg. 5

NRC Resolution: Accepted (Changed Answer Key to reflect this comment. To prevent inadvertent draining of the vessel is the main idea which will get full credit. Other reasonable answers will not cause a loss of credit.

7.06(a). and (b). Confusion exists as to what the question was asking for versus what the answer key was answering. If the question meant to state level was oscillating ±5 inches, did this mean from normal to ±5 and back to normal; or did it mean from normal level ±5 inches (question only lists ±5 inches, which would not be oscillation so it's assumed it was meant to be ±5 inches).

In either case, the following OT Procedures could be entered based on assumptions.

OT-100 Low Level
PT-110 Hi Level
OT-104 Unexplained Reactivity Insertion

The SRO's answer to part (b) will then be in accordance with part (a). The answer key lists two OTs in part (a) but only addresses one in part (b).

Acceptable answers will be according to question interpretation.

Also, part (b) question states "What actions would you, as the Shift Supervisor, direct the operator to take?" The Shift Supervisor would not direct the operator to enter T-100 if a scram occurs but, rather, would enter T-100 and direct the operators accordingly. Therefore, this step should not be counted as part of the answer to part (b).

In addition, the SRO candidate may enter OT-104, Unexplained Reactivity Insertion, in response to the power increase which the reactor may undergo when level oscillates so this answer should be accepted.

REF: OT-100, pg. 1: OT-104, pg. 1; OT-110, pg. 1

NRC Resolution: Practically accepted (a typographical error occurred in preparation, omitting - sign in ±5 inches and leaving it as +5 inches. Answer Key is modified to an oscillation of + 5 inches. Entry to OT-104 is not warranted by the condition, but no credit will be removed if it is mentioned. Also, for the question as asked entry into OT-100 is not warranted because no low level is expected.

7.07(c). Answer key typographical error. Answer should read ON-107 vice ON-105, for "Control Rod Drive System Problems".

REF: Off Normal Procedure Index

NRC Resolution: Accepted (Changed Answer Key)

7.07(d). Acceptable answer is entry into ON-109, "Total Loss of the SRM, IRM, or APRM Systems". This is because Technical specifications require the SRM Upscale and Inoperative TRIPS be operable until the IRMs are on range 8 or higher in order to meet LCO 3.3.6 "Control Rod Block Instrumentation".

REF: Technical Specifications 3.3.6, Table 3.3.6-1, Table 3.3.6-2

NRC Resolution: Accepted (Changed Answer Key)

7.09(b). Alternate acceptable answer is 3. In T-112, steps EB-7, EB-8 and EB-9 ask for 5 ADS valves or ADS/SRV valves. However, in step EB-10, it asks the operator "Are at least 3 ADS/SRVs open". For this reason either 3 or 5 are acceptable answers.

REF: T-112, Emergency Blowdown

NRC Resolution: Accepted (Changed Answer Key)

Category 8: Administrative Procedures, Conditions, and Limitations

8.01.1. Answer key is incorrect. This is a safety limit violation in accordance with Technical Specifications for Thermal Power, Low Pressure or Low Flow.

REF: Technical Specification 2.1.1

NRC Resolution: Accepted (Changed Answer Key)

8.03(a). Alternate correct answer should include the portion of A-7 dealing with job responsibilities of the Control Supervisor (section 5.1.2.2) and Control Room Operators (section 5.1.2.4). Although the wording is different than section 5.2.8 of A-7 dealing with Initiation of a Scram or Shutdown, the intent is the same and the SROs should be given credit.

REF: A-7, pgs. 9, 10, 11, 17

NRC Resolution: Accepted (Exact wording not required)

8.03(b)(1). Alternate acceptable answers should be his designated alternate or Plant Manager. Due to the reorganization within PECo, the title will change to Plant Manager vice Station Superintendent. A-7 states Station Superintendent or his designated alternate.

REF: A-7, 5.2.12

NRC Resolution: Accepted (Changed Answer Key)

8.03(b)(2). Alternate acceptable answer should be Shift Supervision. They are Senior Licensed Operators.

REF: A-7, 5.2.12

NRC Resolution: Accepted (Changed Answer Key)

8.05(a). Answer key states that permission to perform an ST may be delegated. In accordance with A-43, the may not be delegated. Shift Supervision's permission to perform an ST must be given as well as having the responsible CO/ACO's permission. This permission can not be delegated.

REF: A-43, p. 2, part 2.4

NRC Resolution: Rejected (Step 2.4 of A-43 states Shift Supervision and or the ACO or CO shall be responsible for giving permission to start an ST. Further, Paragraph 5.5.3 again states that the ACO/CO can give this permission. If it is the policy at Limerick not to delegate this responsibility, the Procedure A-43 needs revision to remove the flexibility currently allowed.

8.06(c). An acceptable alternate answer is (COL) in that an independent verification of a permit does not have to be done if the equipment check off list (COL) is going to be performed. This is in accordance with procedure A-41 sections 5.4.1 and 5.4.4.

REF: A-41, pg. 12, parts 5.4 1 and 5.4.4

NRC Resolution: Accepted (Changed Answer Key)

8.10. Candidate answers should be graded based upon what interpretative Tech Spec logic they used. The answer key takes the conservative approach by placing the "A" channel in the tripped condition. However, a strict interpretation of Technical Specifications would lead the candidate to come to the conclusion that no actions are required since Tech Spec Table 3.3.1-1 does not require the IRM neutron flux high trip to be OPERABLE in OPCON 1, and table notation (b) states that the function is automatically bypassed when the mode switch is in RUN and the associated APRM is not downscale (given in the question).

REF: Technical specification 3.3.1

NRC Resolution: Comment not accepted. The APRM Downscale Scram function of APRM C would not be operable with the companions IRM inoperable. No change to answer key.

Question Number: 1:01 d

Facility Comment:

Answer can be either TRUE or FALSE. The phrase "as a result of" is vague, and can be interpreted two ways by the candidate.

1) Delayed neutrons are produced upon decay of delayed neutron precursors, and not from the fission of U-235 and U-238 directly, therefore the answer is FALSE.
2) Delayed neutrons are eventually produced after fission of U-235 and U-238, therefore the answer is TRUE.

NRC Resolution:

Disagree with comment. Delayed neutrons can only result if a fission has occurred. The delayed neutron precursors occur through beta minus decay of the fission products produced from fission therefore the neutrons result only from fission occurring. Answer key was not changed.

Question Number: 1.02 d

Facility Comment:

Answer should be SDM INCREASES. Increasing PU-240, an absorber of epithermal neutrons, reduces keff in the core, which results in an increased SDM.

NRC Resolution:

Agree with comment. Answer key is modified to accept SDM Increases as the correct answer.

Question Number: 1.05 a.

Facility Comment:

Additional correct answer: INCREASING TEMPERATURE OF RECIRC PUMP SUCTION. During startup, Recirc Pump Suction Temperature is monitored, and is considered to be Reactor Water Temperature for an operating Recirc pump. Reactor water temperature is recorded and monitored to the heating range per GP-2.

NRC Resolution:

Agree with comment. Increasing Recirc Pump suction temperature. Will be accepted as an alternate answer.

Question Number: 1.05 b

Facility Comment:

Interval 2 and Interval 3 are both correct answers.

The period for Interval 2 is 90.11 sec.
The period for Interval 3 is 80.03 sec.

Both are greater than 80 seconds, and either could be an interval in which the heating range was entered.

NRC Resolution:

Disagree with comment. The answer for Interval 3,80.03 sec, is beyond the significant digits that are used when referring to reactor period. Also the question indicated that only ONE answer was correct and the 10 second change in period in Interval 2 is the most correct indicator of entry into the heating range.

Question Number: 1.08

Facility Comment:

Candidates may discuss how the reactivity coefficients change when feedwater temperature is reduced, as well as how the coefficients affect the transient, and should not be penalized for this.

NRC Resolution:

Agree with comment, however candidates will be penalized for incorrect additional information. The candidates were informed of this during the examination briefing.

Question Number: 1.08 c

Facility Comment

FSAR analysis assumes a decrease in Feedwater temperature of 100 F. The isolation of a feedwater train would result in less of a change in temperature than 100 degrees F, and with appropriate operator action, a scram on high flux can be avoided. Operator action per OT-104 "Unexplained Reactivity Insertion" would result in voids decreasing followed by voids increasing.

NRC Resolution:

Partially agree with comment. Partial credit will be given for discussions concerning reducing recirculation flow.

Question Number: 1.09

Facility Comment:

The question is similar to SRO 5.08. The wording is awkward and extremely confusing. The question is not based on a Learning Objective.

NRC Resolution:

Disagree with comment. Learning objective LOT-1300 #2 indicates that the candidate shall be able to calculate a reactor heat balance. If the candidate can calculate a reactor heat balance then the candidate should be able to analyze the situations in the question to determine if a correct heat balance was performed.

Question Number: 1.10 a.

Facility Comment:

Additional correct answers are:

REACTOR POWER - Class discussion included graphing NPSH vs Reactor Power for Recirc pumps. Reactor Power is a parameter available to the operator on the Process Computer.

RECIRC SUCTION TEMPERATURE - Saturation temperature is directly related to NPSH, and is a parameter available to the operator on the Process Computer.

RECIRC PUMP SPEED - Friction loss is affected by pump speed and is directly related to NPSH. It available on a meter on the C601 panel in the main control room.

NRC Resolution:

Disagree with comment. The question asked for factors that CONTRIBUTED to the NPSH. Reactor power does not contribute to NPSH, rather the increase in feedwater flow does. Recirc Suction Temperature is a result of feedwater temperature and feedwater flow. The facility's reference included with comment did not differentiate between REQUIRED and AVAILABLE NPSH. Recirc pump speed is considered in determining REQUIRED NPSH but has negligible effect on AVAILABLE NPSH. Answer key was not changed.

Question Number: 1.11 a

Facility Comment:

"Bundle" and "Assembly" are used interchangeably, and both should be accepted.

NRC Resolution:

Agree with comment.

Question Number: 1.12 a

Facility Comment:

Additional correct answer:

MAPLHGR/APLHGR (LCO)

Although LOT-1410 uses MAPLHGR (LCO), Tech Spec 3.2.1 defines the LCO for APLHGR, not MAPLHGR.

NRC Resolution:

Answer key is modified to accept MAPLHGR/APLHGR (LCO) as correct answer. The answer key is modified to NOT accept MAPLHGR/MAPLHGR (LCO) as a correct answer. The answer was changed because the reference provided by the facility's comments is more accurate or disagrees with the reference material provided for use by the examiner in developing the written examination. The originally referenced material should be corrected or destroyed.

Question Number: 2.02 c

Facility Comment:

Both WOULD and WOULD NOT are correct answers, depending on candidate's assumptions.

1) If D RHRSW is providing cooling water to the B RHR heat exchanger, it will trip on a LOCA signal, since it is powered by the Unit 1 D-12 Bus, and WOULD NOT is the correct answer.

2) If D RHRSW is providing cooling water to the B RHR head exchanger, it will not trip on a LOCA signal, since it is powered by the Unit 2 D-22 Bus, and WOULD is the correct answer.

NRC Resolution:

Delete part c of the question. Value for the question has been decreased from 3.0 points to 2.5 points.

Question Number: 2.04 a

Facility Comment:

Correct answer is 118 seconds. MOD 86-189 changed this time delay from 58 seconds to 118 seconds. This is referenced in 836.1.b, section 8.2 NOTE: "116 to 120 seconds after manual initiation a 10 minute timer is enabled."

NRC Resolution:

Answer key is modified to accept 118 seconds as the correct answer. The answer was changed because the reference provided by the facility's comments is more accurate or disagrees with the reference material provided for use by the examiner in developing the written examination. The originally referenced material should be corrected or destroyed.

Question Number: 2.04 c

Facility Comment:

Additional correct answers:

1) Decreasing tank level

2) Decreasing reactor power

3) Continuity Lights extinguish

NRC Resolution:

Agree with comment. Answer key is modified to accept decreasing tank level and decreasing reactor power as the additional correct answers. Continuity lights extinguish will be accepted in lieu of squib valves firing.

Question Number: 2.05 a 2

Facility Comment:

Correct answer is 455.

NRC Resolution:

Agree with comment. Answer key is modified to accept 455 as the correct answer. This was a typographical error on the answer key.

Question Number: 2.05 a 3

Facility Comment:

Question should be deleted. Check valve will open when discharge pressure is greater than reactor pressure. Discharge pressure is not required knowledge for RO.

NRC Resolution:

Disagree with comment. The check valve will open when reactor pressure is less than shutoff head of the pump. Learning objective 5.b of LOT-0350 requires that the RO know the shut off head of the pump. Answer key was not changed. Partial credit was given for understanding the concept of where the valve will open.

Question Number: 2.05 b

Design pressure should be deleted from this question. Design flow is required for RO knowledge. Design pressure for rated flow is not required knowledge for RO.

NRC Resolution:

Agree with comment. Full credit will be given for design flow.

Question Number: 2.06 e.

Facility Comment:

Suppression pool spray and full flow test valves also close. Candidates should not be penalized for including these in the answer.

NRC Resolution:

Disagree with comment. These valves should not be open on a loop that is being operated in the Shutdown cooling mode as it would result in draining the vessel, therefore they would not be expected to reposition on the LPCI initiation. Points will be deducted if these are included as repositioning.

Question Number: 2.08 c

Facility Comment:

Answer should be either NO SYSTEM or ESW, depending on candidate's interpretation of the question. ESW can be used to cool the RECW heat exchanger, and can provide cooling to the Recirc pump seal and motor oil coolers which are normally cooled by RECW. Depending on the assumptions made by the candidate, either NO SYSTEM or ESW is an appropriate answer.

NRC Resolution:

Part c of the question will be deleted because question did not specify components that can be cooled. Point value for the question has been decreased from 2.5 points to 2.0 points.

Question Number: 2.09 a

Facility Comment:

Additional correct answers:

OVERSPEED

LOW CONDENSER VACUUM

Each overspeed trip is independent of the other feedpumps, and could have caused one feedpump to trip.

Each RFP turbine exhausts to a different shell of the main condenser, independent of the other turbines, and could have caused one feedpump to trip.

NRC Resolution:

Agree with comment. Answer key is modified to also accept Overspeed and Low Condenser Vacuum as correct answers.

Question Number: 3.01 a

Facility Comment:

Delete this question. The Recirc Pump Master Controller is not used at LBS, and the M/A Transfer Station is not used in auto. This question should be deleted since it does not apply to LGS.

NRC Resolution:

Disagree with comment. The equipment is installed and training material contains information concerning its operation. The action described in the question could occur and the candidate should be aware of the consequences of the action.

Question Number: 3.01 b

Facility Comment:

75% flow is equivalent to approximately 60% pump speed. Either is an acceptable answer.

NRC Resolution:

Agree with comment, however Figure 2 of LOT-0040 indicates that the limiter is based on 75% speed and should be corrected. The question requested speed, therefore the answer key is changed to 60% speed. Partial credit will be given for stating 75% core flow.

Question Number: 3.01 c

Facility Comment:

Delete this question. The Recirc Pump Master Controller is not used at LGS, and the M/A Transfer Station is not used in auto. This question should be deleted since it does not apply to LGS.

NRC Resolution:

Agree with comment. Part c has been deleted. Point value for question decreased from 3.0 to 2.0 points.

NOTE: Part C was deleted because the possibility of this occuring, due to the fact that the Facility does not use the Master Controller is minimal due to procedural constraints, however the possibility of inadvertant operation of a switch on the individual recirc pump controller is significant enough to warrant the operator understanding the response of the system if this occurs. Operators use buttons on the individual recirc pump controllers where the transfer switch referred to in part a is located.

Question Number: 3.02 b

Facility Comment:

Both HIGHER THAN ACTUAL and SAME AS ACTUAL are correct answers. Wide range is calibrated with no jet pump flow, and would read the same as actual assuming no jet pump flow. Assuming jet pump flow, wide range would read higher than actual.

NRC Resolution:

Agree with comment. Answer key is modified to also accept HIGHER THAN ACTUAL.

Question Number: 3.03 a

Facility Comment:

Candidate should not be penalized for including LOW CONDENSER VACUUM 10.5 PSIA as an additional answer, since this could be bypassed or active in startup, depending on assumptions made. Candidate should not be penalized for including MANUAL, due to the wording of the question.

NRC Resolution:

Partially agree with comment. Low Condenser Vacuum is not acceptable because the stop valves, which bypass the scram signal, are not opened until after 2 Bypass Valves are open. This would not occur until pressure is greater than 920 psig. REF: S01.1.A and GP-2.
Answer key is modified to also accept Manual as a correct answer.

Question Number: 3.04 d

Facility Comment:

Action 3 is not a correct answer. If the reactor is in startup mode and power reaches 15%, and scram signal is expected. A rod block should have been generated at 12% power. If rods can be withdrawn to 15%, the rod block signal has failed, and a scram signal is now expected. Action 2 is the only correct answer.

NRC Resolution:

Agree with comment. Answer key is modified to only accept Action 2 as the correct answer. Point value was changed from 0.25 for Action 2 to 0.5 point.

Question Number: 3.04 e

Facility Comment:

NO ACTION is also a correct answer. Since criticality can be reached at any time, a rod block will not always be generated in the condition.

NRC Resolution:

Disagree with comment. "Approach to Criticality" is referred to in GP-2 at the point at which rod withdrawal commences. Criticality could not be reached with all rods in or SDM requirement could not be met. A rod block is generated because with all rods IRM's are downscale. The rod block is bypassed only if all IRM's are on range 1. Answer key was not changed.

Question Numb[illegible]r: 3.05 a

During ex[illegible]n grading it was determined that the answer to part a should be 525 [illegible]ead of 420. The timers act sequentially on a low level signal w[illegible] drywell pressure. Refer to figure LOT-0330-6. Answer key is modifi[illegible]t 525 seconds as the correct answer.

Question N[illegible] 3.06

Facility Comment:

Numbers and calculation on LOT 0590-6 should be accepted in lieu of discussions. Candidate should not be penalized for continuing the transient until the plant stabilizes.

NRC Resolution:

Partially agree with comment. Instructions for the examination require that the answer be on answer sheets. Credit will be given for answer on diagram. Candidate will not be penalized for continuing the transient if correct information is provided.

Question Number: 3.07 a

Facility Comment

Standby Gas Treatment System will also start if aligned to the Refuel Floor Ventilation System.

NRC Resolution:

Agree with comment. Answer key is modified to accept SBGT start if aligned to the refuel floor as an additional required answer. Comment could not be verified by reference material provided because it was illegible, however comment was verified in the Secondary Containment LP.

Question Number: 4.01 a

Facility Comment:

Directions in A-7 for an RO to shutdown the plant or scram are covered in two sections of A-7, using different wording. Section 5.1.2.4 is appropriate as section 5.2.8 as an answer.

NRC Resolution:

Comment noted. The conditions specified in 5.1.2.4 are conceptually equivalent to 3 of the conditions specified in 5.2.8 and credit will be given if the correct concept is specified, however section 5.2.8 gives additional guidance to the operators which will be required in order to obtain full credit. Answer key was not changed.

Question Number: 4.07 a

Facility Comment:

Answers on answer key are redundant:

Additional correct answer:

TO REMIND OPERATOR THAT THE MIN FLOW IS BLOC CLOSED.

This is because the basic purpose of the Blocking Procedures, i.e. assure safety of workers and avoid equipment damage.

NRC Resolution:

Partially agree with comment. Answer key is modified to accept prevent draining the vessel as the correct answer for full credit. "To remind operator that the min flow is blocked closed" will not be accepted as an additional correct answer because the reference material supplied with the facility comments did not support the additional answer.

ATTACHMENT 5

SIMULATION FACILITY FIDELITY REPORT

Facility Licensee:	Philadelphia Electric Company 2301 Market Street Philadelphia, Pennsylvania 19101
Facility Licensee Docket No.:	50-352
Facility License No.:	NPF-39
Operating Test administered at:	Limerick Simulator
Operating Tests Given On:	June 8 and 9, 1988

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

1. In IC-20 at 75% power, the conditions are such that the APRM upscale alarms and rod block setpoints are exceeded and the recirculation flow is in a critical speed instability area.

2. Malfunction 72 (SJAE steam supply valve fails closed) had no effect on condenser vacuum. The simulator operator suggested that this effect was due to a plant modification which added an automatic swap to an alternate steam supply, had been implemented in the software but the malfunction book and panel hardware had not been updated.

3. Malfunction 44A (reactor pressure recorder XR-6233 failing upscale) did not produce a half-scram and other actions that were listed in the malfunction book.

4. Malfunction 140E (stuck open safety relief valve) would cause the valve to open but allowed it to close when the operator manipulated the switch. It should have remained open regardless of operator actions.

5. Malfunction 110 (ATWS with failure of the RRCS, main turbine trip, and the main condenser available) reactor pressure remained at greater than 1190 psig regardless of power. Also, reactor level oscillated between 20 and 40 inches then dropped to -300 inches. An attempt to reproduce this transient caused similar effects.