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

EXAMINATION REPORT NO. 84-27

FACILITY DOCKET NO. 50-219

FACILITY LICENSE NO. DPR-16

LICENSEE: GPU Nuclear Corporation
P. O. Box 388
Forked River, New Jersey 08731

FACILITY: Oyster Creek Nuclear Generating Station

DATES: August 20 - 23, 1984

CHIEF EXAMINER: John Berry ______ 10-16-84
John Berry, Reactor Engineer Examiner — Date

APPROVED BY: ______ 10/16/84
Chief, Project Section 1D — Date

SUMMARY: Four Senior Reactor Operator candidates were examined at the Oyster Creek Nuclear Power Plant the week of August 20, 1984. All four candidates passed their respective oral and written examinations.

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REPORT DETAILS

TYPE OF EXAMS: Initial ____ Replacement X Requalification ____

EXAM RESULTS:

| | RO Pass/Fail | SRO Pass/Fail | Inst. Cert Pass/Fail | Fuel Handler Pass/Fail |
|---|---|---|---|---|
| Written Exam | / | 4/0 | / | / |
| Oral Exam | / | 4/0 | / | / |
| Simulator Exam | / | / | / | / |
| Overall | / | 4/0 | / | / |

CHIEF EXAMINER AT SITE: William Thomas, ORNL

PERSONS EXAMINED

| | |
|---|---|
| Boyle, Joseph | Upgrade (SRO) |
| Cropper, Gillis | Upgrade (SRO) |
| Reopell, Jeffrey | Upgrade (SRO) |
| Schaffer, David | Upgrade (SRO) |

1. Summary of generic strengths or deficiencies noted on oral exams:

   No generic strengths were noted on the oral examinations. The examiner noted a generic weakness concerning the placement of the Augmented Off-gas System in service during start up.

2. Summary of generic strengths or deficiencies noted from grading of written exams:

   On the written exam, all candidates did very well in category six. This section covers the design, control, and instrumentation of Plant Systems.

   A generic deficiency was noted in the candidates' ability to specifically define the BWR thermal limits.

3. Comments on interface effectiveness with plant training staff and plant operations staff during exam period.

   The courtesy and cooperation exhibited by the facility training staff and plant operations staff was greatly appreciated.

4. Personnel Present at Exit Meeting:

   NRC Personnel

   Curtis Cowgill - NRC Senior Resident Inspector

   NRC Contractor Personnel

   W. Thomas, Examiner (ORNL)

   Facility Personnel

   Pete Fiedler, Station Manager
   Rod Davidson, Training Supervisor
   Dan McMillan, Training
   Derric Willson, Training Instructor

5. Summary of NRC Comments made at exit interview:

   •- The facility was informed that the four (4) applicants passed the operating examination.

   •- Examiner stated that the new version of the Abnormal Operating Event Procedures was missing from the control room at the time of the examination. He was informed that those procedures are receiving final approval and will be available for reactor startup.

   •- Of the two (2) applicants asked, neither could find a statement in the startup procedures or the offgas procedures directing when to valve in the AOG during a startup. The station manager agreed to have someone review the existing procedures and, if an omission exists, to initiate appropriate procedure revisions to provide adequate operator guidance.

   •- From observation during the walk-through from control room to refueling floor, the plant appeared to be very clean and the levels of radioactivity in the posted areas were well below limits

   •- The courtesy and spirit of cooperation exhibited by the members of the facility training staff during the visit was also greatly appreciated.

6. Summary of facility comments and commitments made at exit interview:

   •- Management expressed the desire for minimum exam turn-around time since these operators will be needed for the upcoming startup.

7. CHANGES MADE TO WRITTEN EXAM

   Attached is a list of specific comments made by the licensee concerning the written exam administered on August 21, 1984. Mr. Thomas accepted all of the facility comments when he graded the exams and no formal response to these comments is required.

Attachment:

Written Examination(s) and Answer Key(s) (SRO)
Comments

attachment 1

OYSTER CREEK NUCLEAR GENERATING STATION

NRC EXAMINATION OF AUGUST 21, 1984

EXAMINER: WILLIAM THOMAS

FACILITY COMMENTS

SPECIFIC EXAMINATION QUESTION COMMENTS

SECTION 5

5.4.c Suggest that the correct answer is related to 25 days following restart of a clean core. Problem may occur with the definition of the term "Cycle"

5.8.a Suggest you accept the following

1) Core subcooling ( core inlet enthalpy)

2) Excessive moisture content of steam leaving the reactor due to "blow-by" of steam dryer seal.

5.11 Suggest other technically correct answers include:

a. Rx water level inside separators above the "turnaround point"

b. The magnitude of the heat source

c. The amount of feedwater flow rate (changes the density of the downcomer water)

SECTION 6

6.1.c From Procedure EMG - 3200.03 "Level Restoration" Step C1-3. another system included is the Liquid Poison System

6.10 Yes - The operator initially secures the Core Spray Pumps by pushing the overide buttons for each of the initiating signals and turning the control switch for each pump to stop. When the low level signal comes in again, the Primary Core Spray and Core Spray Booster Pumps will automatically start.

SEE DIAGRAM

1 & 2 Initiating signals

3 System start contacts

4 & 5 Overide buttons

SECTION 7

7.1 Entry condition for high Reactor Pressure is 1050 psig as per Standing Orders #1.
See attached copy of Standing Order .

SECTION 8

8.3 Answer b should include Red Tags Ref. 108 Procedure page 13 & 14 section 5.2.5 & 5.2.6

8.4 Answer a - the Fire Brigade leader could be a leadership qualified CRO Procedure 106 section 3.7.3

8.5 Answer b - By procedure the second SRO is the G.S.S. or G.O.S.

OVERALL EXAMINATION COMMENTS

The examination questions were clear and concise. The answers to the questions were well referenced and accurate.

Given the present restrictions on the examination review process, the examination review went well. This is atributed to the clearly written exam and the cooperative attitude of the examiner.

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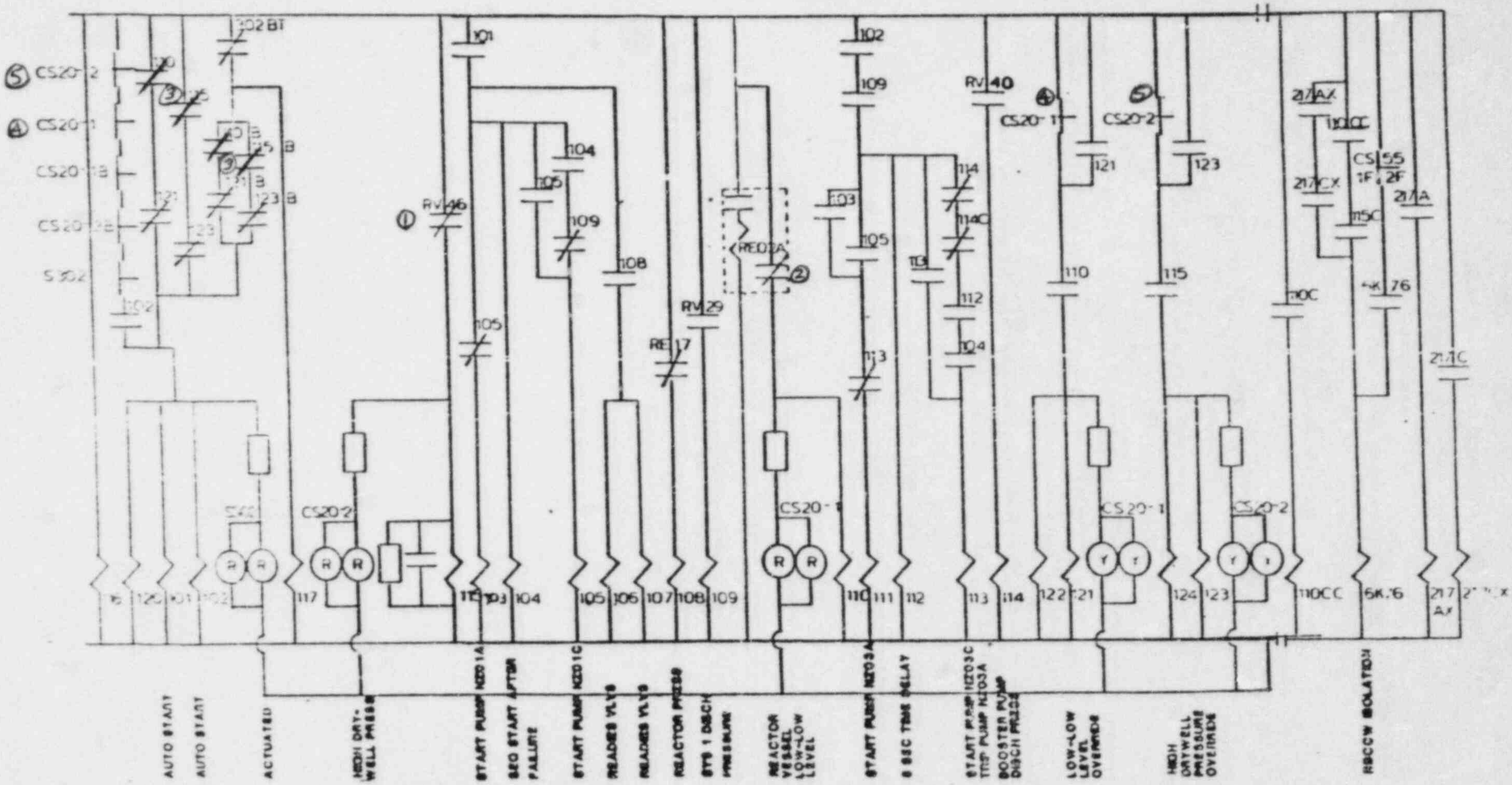
FOR TRAINING
USE ONLY



# CORE SPRAY SYSTEMS CONTROLS LOGIC




[illegible] represent any one of the four channels that start the pumps in core spray system 1 and 2. [illegible] only channels B and C have the [illegible] portion of the elementary. Since the last available contact from the drywell pressure [illegible] (115) is used in the RBCCW isolation, [illegible] (125) is placed in parallel with relay 115. Contacts from relay 125 are used in the [illegible] pressure override instead of [illegible] relay 115 which are used for channels A and D. All the relays and contacts shown with no [illegible] have a letter subscript corresponding to the channel represented. The letter subscripts shown should correspond to the proper cross-channel interchange.

## STANDING ORDER # 1 INSTRUMENT SETPOINTS

| FUNCTION | DEVICE | ACTION | TECH. SPEC. LIMIT | CORRECTED TECH. SPEC. LIMIT (See Note 1) | INSTRUMENT SETPOINT (See Note 2) | CORRECTED INSTRUMENT SETPOINT (See Note 3) |
|---|---|---|---|---|---|---|
| 1. High Reactor Pressure | Safety Valves | Valves Open | 4@ 1212 psig | 4@ 1212 psig | 4@ 1212 psig | 4@ 1212 psig |
| | | | 4@ 1221 psig ±12 psi | 4@ 1221 psig±12 psi | 4@ 1221 psig | 4@ 1221 psig |
| | | | 4@ 1230 psig | 4@ 1230 psig | 4@ 1230 psig | 4@ 1230 psig |
| | | | 4@ 1239 psig | 4@ 1239 psig | 4@ 1239 psig | 4@ 1239 psig |
| | Relief Valves NR108 A | Relief Valves Open | ≤ 1070 psig | ≤ 1079.15 psig | 1060 psig | 1069 ± 2.5 psig |
| | B | | ≤ 1090 psig | ≤ 1104.5 psig | 1080 psig | 1094 ± 2.5 psig |
| | C | | ≤ 1090 psig | ≤ 1096.6 psig | 1080 psig | 1086 ± 2.5 psig |
| | D | | ≤ 1070 psig | ≤ 1082.2 psig | 1060 psig | 1072 ± 2.5 psig |
| | E | | ≤ 1090 psig | ≤ 1102.2 psig | 1080 psig | 1092 ± 2.5 psig |
| | Relief Valves NR108 | Relief Valves Close | None | None | | |
| | A | | | | 1010 psig | 1019 ± 2.5 psig |
| | B | | | | 1058 psig | 1072 ± 2.5 psig |
| | C | | | | 1058 psig | 1065 ± 2.5 psig |
| | D | | | | 1010 psig | 1022 ± 2.5 psig |
| | E | | | | 1058 psig | 1070 ± 2.5 psig |
| | RE 03 | Scram | ≤ 1060 psig | | | |
| | A & B | | | ≤ 1068.35 psig | 1050 psig | 1058 ± 2.5 psig |
| | C & D | | | ≤ 1066.1 psig | 1050 psig | 1056 ± 2.5 psig |
| | RE 15 A & B | Isolation Condenser Initiation & Recirc Pump Trip (No Time Delay) | ≤ 1060 psig with time delay ≤ 3 sec. | ≤ 1068.35 psig | 1060 psig | 1068 psig |
| | | | | ≤ 3 sec. | 1.5 ± 1 sec. | 1.5 ± 1 sec. |
| | C & D | | | ≤ 1066.1 psig | 1060 psig | 1066 psig |
| | | | | ≤ 3 sec. | 1.5 ± 1 sec. | 1.5 ± 1 sec. |
| | ID 77 | Alarm | None | None | 1040 psig | 1040 psig |
| 2. Low Reactor Pressure | RE 23 A | MSIV Closure | ≥ 825 psig | ≥ 832.9 psig | 842 psig | 850 psig |
| | B | | | ≥ 833.7 psig | 841 psig | 850 psig |
| | C | | | ≥ 833.6 psig | 841 psig | 850 psig |
| | D | | | ≥ 834.6 psig | 840 psig | 850 psig |
| | RE 16 A | MSIV Closure and Low Vacuum Scram Bypass | ≤ 600 psig | ≤ 608.34 psig | 576 psig | 584 ± 5 psig |
| | B | | | ≤ 606.1 psig | 576 psig | 582 ± 5 psig |
| | RE 17 A&C | Core Spray Valves Open | ≥ 285 psig | ≥ 293.34 psig | 301 psig | 309 psig |
| | B&D | | | ≥ 291.1 psig | 300 psig | 306 psig |

NOTE 1: These values are obtained by adding to the Tech. Spec. limit the associated head correction factors for each instrument and represent maximum allowable trip values. During instrument calibration and surveillance testing observed operation outside these limits constitute a reportable occurrence.

NOTE 2: These values represent the magnitude of the process variable at which the instruments trip. The difference between each value and its associated Tech. Spec. limit accounts for any instrument drift or added conservatism included in the instrument setting.

NOTE 3: These values are obtained by adding to the instrument setpoint the associated head correction for each instrument as applicable, and specifies the value at which the instrument is to be set during calibration.

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U. S. NUCLEAR REGULATORY COMMISSION

SENIOR REACTOR OPERATOR LICENSE EXAMINATION

MASTER COPY

Rod Davidson
Dan McMillan
Derick Willson
Dave Fawcett

Facility: OYSTER CREEK

Reactor Type: BWR/2

Date Administered: 8/21/84

Examiner: W.THOMAS

Applicant: ____________

INSTRUCTIONS TO APPLICANT:

Use separate paper for the answers. Write answers on one side only. Staple questions 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 | Applicant's Score | % of Cat. Value | Category |
|---|---|---|---|---|
| 25 | 25 | | | 5. Theory of Nuclear Power Plant Operation, Fluids & Thermodynamics |
| 25 | 25 | | | 6. Plant Systems: Design, Control & Instrumentation |
| 24 | 25 | | | 7. Procedures-Normal, Abnormal, Emergency & Radiological Control |
| 24.5 | 25 | | | 8. Administrative Procedures, Conditions and Limitations |
| 98.5 | 100 | | | TOTALS |

Final Grade ______ %

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

____________________
Applicant's Signature

## 5. THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS & THERMODYNAMICS (25.0)

5-1 For each of the following conditions, state whether individual control rod worth will increase, decrease, or remain the same and briefly explain why.

a. Moderator temperature increase (1.0)
b. Increase in control rod density (1.0)
c. Core void fraction increase (1.0)

5-2 A minimum SRM count rate is required for reactor startup. What are three (3) neutron production reactions that occur in the Oyster Creek Reactor to provide that minimum neutron source strength? (1.5)

5-3 Give the reason (basis) why chloride ion concentration limits for reactor coolant are:

a. Lower during reactor startup than at full power operation. (1.5)

b. Less important during refueling than during a reactor startup. (1.5)

5-4 Regarding the effects of Fission Product Poisons:

a. Is the equilibrium $^{135}Xe$ concentration doubled when power level is raised from half to full power? Why? (0.75)

b. What effect (increase/decrease) will Xenon have on shutdown margin during the first six (6) hours following a reactor scram from extended full power operation? (0.5)

c. At what point in the operating cycle does $^{149}Sm$ reach its highest concentration? (0.5)

d. During a four to six hour period after a rapid power increase from half to full power the transient xenon effects will require control rods be progressively moved (IN/OUT) to maintain constant power? Choose one underscored word and EXPLAIN WHY. (0.5)

(continued on next page)

5-5 How does the MAGNITUDE of the VOID COEFFICIENT of REACTIVITY change (more negative, less negative, or unaffected) for each of the following changes in core condition? BRIEFLY EXPLAIN WHY.

a. Increase in core void fraction (1.0)
b. Decrease in fuel temperature (1.0)
c. Increase in core age (effective core size) (1.0)

5-6 Answer each part below TRUE/FALSE.

a. For a given reactivity addition, delayed neutron effects cause the SRM stable period to be shorter at BOL than at EOL. (0.5)

b. A reactor is exactly critical (k=1) when the neutron flux increases with a stable positive period without additional control rod movement. (0.5)

c. The presence of delayed neutrons cause the average neutron generation time ($\bar{\ell}$) to decrease. (0.5)

5-7 Figure 5.1 shows Oyster Creek core reactivity versus Core Age. Briefly, what are the main processes occurring that cause core reactivity to change with core life between the following points:

a. Points 1 to 2 (0.75)
b. Points 2 to 3 (0.75)
c. Points 3 to 4 (0.75)

5-8 a. Other than decreased plant efficiency, what are four (4) other undesirable consequences that may result from excessive CARRYUNDER? (1.5)

b. What can an operator do to minimize CARRYUNDER? (0.5)

5-9 Define the three (3) BWR thermal limits and state what each protects against. (3.0)

(continued on next page)

5-10 A precaution in the Feedwater System Procedure (317) states that whenever attempting to maintain vessel level the "Reactor water level shall be limited to 180 inches above TAF Yarway". What is the reason for this caution statement? (1.5)

5-11 The effectiveness of the heat sink is one factor which greatly affects NATURAL CIRCULATION. What are four (4) other factors? (2.0)

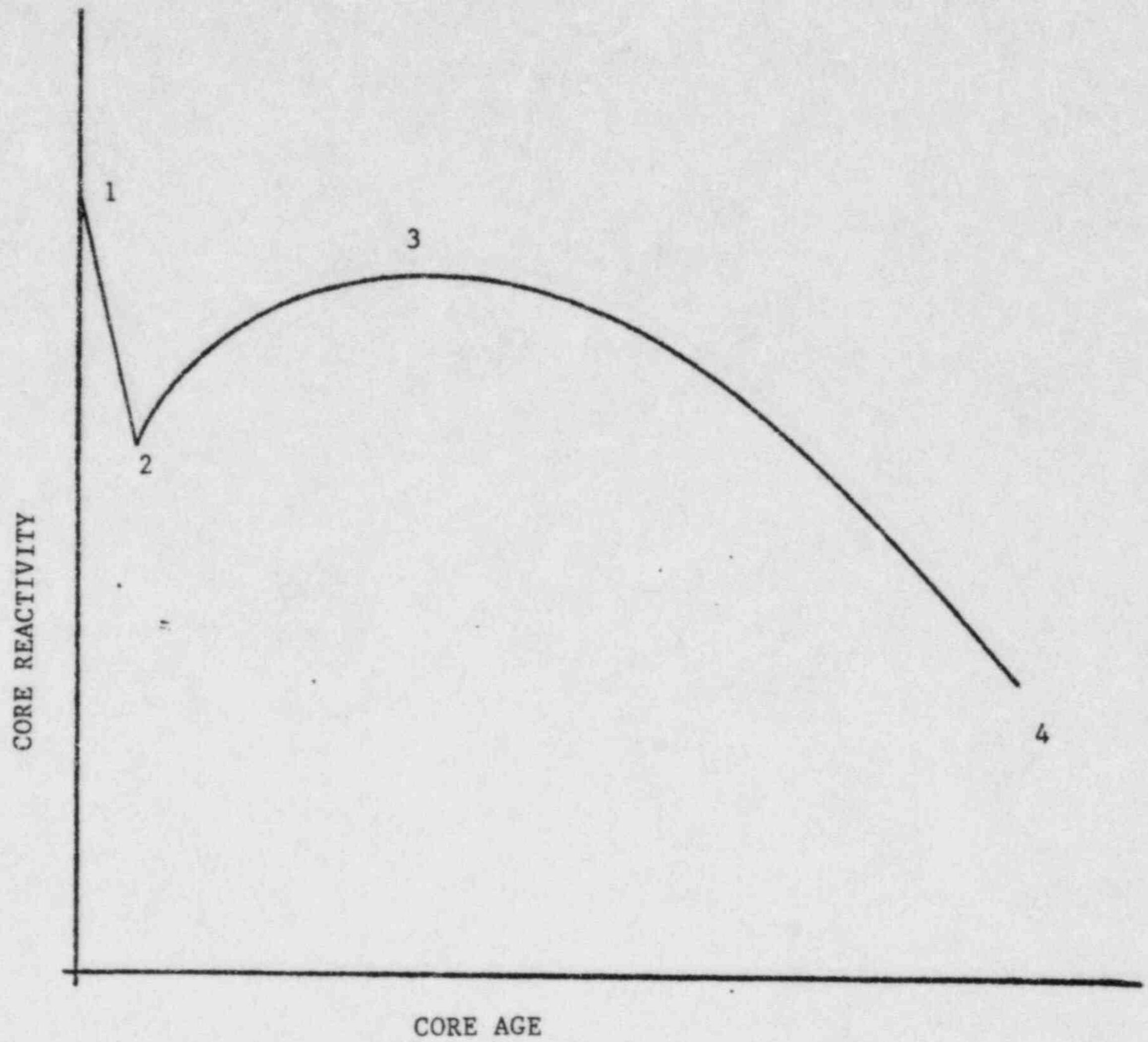


FIGURE 5.1

## 6. PLANT SYSTEMS: DESIGN, CONTROL & INSTRUMENTATION (25.0)

6-1 Concerning the Automatic Depressurization System (ADS):

a. Where are the vacuum breakers located in this system and why are they necessary? (1.0)

b. Which (sensed) auto-initiation signal seals-in and must be manually reset when it clears? (0.5)

c. ADS should not be manually initiated unless one pump is running in at least one of which three (3) systems? (1.0)

d. How many EMRV's should be open two seconds after the ADS timers have timed-out? (0.5)

6-2 For the Area Radiation Monitoring (ARM) system:

a. Why are radioactive sources installed in some ARM's? (1.0)

b. Which ARM's initiate automatic protective action and what actions do they initiate? (1.0)

c. What reference document lists the current alarm setpoints? (0.5)

6-3 Isolation of the ISOLATION CONDENSER may occur while it is in-service.

a. What signal(s) cause this isolation to occur? (1.0)

b. How is the isolation signal cleared if it is determined to be inadvertent? (1.0)

6-4 State the initiating signals and resulting actions, including setpoints, for both types of reactor cleanup isolations. (3.0)

6-5 Describe how the CONTAINMENT SPRAY SYSTEM could be used to reduce torus temperature (during power operation) without wetting down the drywell. Confine your answer to SYSTEM I and include the system valve positions, pumps operating, and control room switch positions, necessary for this operation. (2.5)

(continued on next page)

6-6 Which of the following RPS trips are <u>never</u> bypassed automatically: (2.0)

a. APRM Flow Unit INOP or Upscale
b. IRM Upscale or INOP
c. Reactor High Pressure Trip
d. Hi Drywell Pressure
e. Load Reject
f. Low condenser vacuum
g. Main Steam Line Isolated
h. Main Steam Line High Radiation

6-7 An air operated positioner is used to position the scoop tube of an MG set associated with a recirculation pump. Explain <u>what</u> happens to both the scoop tube <u>and</u> to the recirculation flow on loss of supplied air to the positioner. (3.0)

6-8 Regarding the reactor level and feedwater control system:

a. What two (2) conditions cause a lockup of the feed reg. valves? (1.0)

b. Three (3) feedwater pumps are running and level control is from the master controller. How does an operator balance the flows through the three (3) strings should they become uneven? (1.0)

c. Describe two (2) ways that a feed pump runout setpoint signal is reset. (1.0)

6-9 List four (4) of the six (6) plant systems that have the capability of being supplied with fire protection water. (2.0)

(continued on next page)

6-10 A LOCA condition exists and is of sufficient magnitude to cause one core spray pump and one core spray booster pump in each core spray system (1 and 2) to inject. Level then increase and when it reaches normal (+160") the operator secures both running booster pumps and both running main pumps. After the pumps are secured, level decreases. If the high drywell pressure signal is still present, will the core spray system automatically resume injection if level again decreases to the initiation setpoint? (2.0)

----------If YES, explain how the system functions to cause this result.

----------If NO, explain the operator and system actions that must be accomplished to resume injection.

7. PROCEDURES-NORMAL, ABNORMAL, EMERGENCY & RADIOLOGICAL CONTROL (24.0)

7-1 For the Reactor Pressure Vessel (RPV) Control Emergency Operating Procedure (EMG-3200.01):

a. Give the six (6) procedure entry conditions. (3.0)

b. Immediate use of Standby Liquid Control is required when what conditions exist? (0.5)

7-2 Give four (4) of the six (6) FUEL CLAD INTEGRITY SAFETY LIMITS. (2.5)

7-3 What constitutes Primary Containment Integrity? (2.0)

7-4 List four (4) situations, or conditions, for which a standard Radiation Work Permit (RWP) is required. (2.5)

7-5 Regarding the status of SRM's for refueling:

a. How many channels must be operable? (1.0)

b. What are the requirements on detector positioning and location in-core during refueling? (1.0)

c. What is the restriction on minimum SRM count rate. (1.0)

7-6 With the reactor initially at rated power, what are the main condenser vacuum related alarms and automatic actions that occur when vacuum decreases from normal to atmospheric pressure. Include setpoints. (2.5)

7-7 While operating at rated power, a relief valve (EMRV) fails open causing blowdown to torus (procedure 3200.02):

a. What two (2) operator actions should be initiated to close the valve prior to a manual scram? (1.0)

b. What systems are placed in operation to control torus temperature? (1.0)

c. Which torus parameter determines when to scram the reactor? Include setpoints. (0.5)

(continued on next page)

7-8 For a reactor startup from cold condition to rated power:

a. During heatup where is the mechanical pressure regulator setpoint maintained, and why? (1.0)

b. Describe the administrative constraints on rod motion when criticality is not achieved prior to withdrawing the first eight (8) rod groups. (1.0)

c. What are the containment inerting requirements or limitations for a startup? (1.0)

d. How does an operator determine that the nuclear instrumentation have "sufficient overlap"? (0.5)

7-9 With the reactor at rated power, what action is required if the Halon fire suppression system for a vital system (such as the A and B Battery Room) becomes inoperable? (2.0)

## 8. ADMINISTRATIVE PROCEDURES, CONDITIONS & LIMITATIONS (24.5)

8-1 According to the Administrative Control (Tech. Specs., Sec. 6.7.1), what actions must be taken in the event a safety limit is violated? Three (3) required. (3.0)

8-2 Regarding the Emergency Plan Implementing Procedures:

a. List the four (4) emergency classifications in the order of decreasing severity. (1.0)

b. Which emergency classifications require activation of the Operations Support Center (OSC)? (1.0)

8-3 Match one (1) of the six (6) types of tags to each statement below: (Red/White, Yellow, White, Blue, Orange, or Red tag).

a. This tag is recorded in the information tag log. (0.5)

b. An electrical tag which is used for testing an isolated circuit with an independent source of voltage. (0.5)

c. Mechanical equipment caution tag. (0.5)

d. Electrical caution tag. (0.5)

8-4 Pertaining to the Conduct of Operations (procedure 106):

a. The Fire Brigade is composed of what personnel? (1.0)

b. Are there any personnel excluded from the Fire Brigade? Explain. (1.0)

c. When is it required that a Shift Technical Advisor (STA) be available at the station? (i.e., plant operating conditions) (1.0)

8-5 According to the provisions of Tech. Specs. pertaining to temporary changes for safety related procedures:

a. When are temporary changes permitted? (1.5)

b. Who approves the temporary change before implementation? (1.5)

(continued on next page)

8-6 Regarding SBLC injection times and reactor poison concentration following system initiation:

a. The Tech. Specs. basis describes two (2) reactor boron concentrations, 600 and 750 ppm. What are the basis for each value? (2.0)

b. What is the minimum and maximum injection time (min) for SBLC injection and what are the two( 2) basis for choosing those times? (1.0)

8-7 a. A person having an SRO license (or SRO limited to fuel handling license) must be present on the refueling floor while "core alterations" are in progress. Define the term CORE ALTERATION. (2.0)

b. TRUE/FALSE: A person supervising refueling operations is permitted to have other collateral duties or responsibilities? (0.5)

8-8 A surveillance test for a safety related system is due, and the plant operating conditions will permit conducting the test, but testing is delayed because of scheduling difficulty. What are the Tech. Specs. provisions for extending the surveillance time interval (include in your answer any consideration to previous surveillance intervals)? (3.0)

8-9 a. What administrative requirements must be met to continue a plant startup from less than 10% of rated power with a bypassed (inoperable) rod worth minimizer? (1.5)

b. TRUE/FALSE. A plant startup is not permitted if the RWM is declared inoperable before withdrawing the first control rod. Explain your answer. (1.5)

5-1 a. Increase-Moderator density decrease causes thermal diffusion length to increase which increases the leakage of thermal neutrons from fuel bundles into control rod regions. (1.0)

b. Decrease-Increasing the number of notches in core decreases the region (zone) of control of a particular control rod. The rod is absorbing thermal neutrons leaked from fewer bundles because the additional inserted notches are "shadowing" it. (1.0)

c. Decrease-Increased voiding causes significant decrease in neutron moderation causing more fast and fewer thermal neutrons to leak from a bundle. Control rods are thermal neutron absorbers hence worth decreases. Also voiding allows thermal neutrons to migrate over longer distances which "spreads" the reactivity of one core region with another (a more coupled core). (1.0)

REF.: Reactor Theory Chapter 8 p. 56, 69

5-2 Oyster Creek Rx does not have Installed Neutron Sources. The intrinsic Neutron Sources are: (1.5)

1- Spontaneous fission from several isotopes ($^{242}Cm$, $^{244}Cm$, $^{238}U$, etc.).

2- The alpha-neutron reaction with $^{18}O$ in the $UO_2$.

3- The photo-neutron reaction with deuterium.

REF.: Reactor Theory Chapter 9 p. 7, 12, 13

5-3 a. Dissolved <u>oxygen content</u> in the coolant (water) is <u>lower at power</u> operation hence the effect of Cl ion concentration on stress corrosion cracking of SS is less at power operation. (1.5)

b. <u>Water temperature</u> necessary for stress corrosion to occur is not present with the reactor in a cold shutdown condition while refueling. (1.5)

REF.: T.S. basis 3.3.E Chemistry (p. 3.3-5, 3.3-6)

(continued on next page)

5-4 a. No. (0.25) The production rate is directly proportional to power level but removal rate is not since it is the sum of two terms, xenon decay (concentration dependent) and burnup (concentration and flux dependent). Since the burnup term becomes more significant with increased flux (power) the equilibrium value will be higher, but not twice as high. (0.5)

b. Increase (0.5)

c. During a shutdown (following power operation) after all promethium has decayed or approx. 12.5 days following shutdown. (0.5)

d. Control rods must move in to compensate for the initial xenon concentration decrease. (0.5)

REF.: Fission Product Poison L.P. #300.10, p. 10, 17, 22, 23

5-5 a. More negative. The increase in void fraction implies that more voids are being formed in the region of maximum thermal neutron flux (i.e., voids form lower in the core); therefore, a small change in void fraction has a larger effect on reactivity with increased voiding. (1.0)

b. Less negative. Decreasing fuel temp. decreases resonance capture (resonances not as broad) which results in an increase in resonance escape probability (i.e., for a given void fraction more neutrons can thermalize, interact with fuel and cause fission) (1.0)

c. Less negative. The more negative contribution caused by decreased resonance escape (increased resonance absorption in $^{240}Pu$ buildup with age) is overridden by the combined less negative contributions from 1) decreased thermal utilization (due to increased moderator-to-fuel ratio as fuel depletes) plus, 2) increased "core size" which decreases neutron leakage. (1.0)

REF.: Reactor Theory Chapter 8, p. 18-26

5-6 a. False (0.5/ea.)
b. False
c. False

REF.: Reactor theory chap. 11 (for a & c)
Approach to critical procedure 201.1 (for b)

(continued on next page)

5-7 a. Buildup of Samarium (0.75/ea.)
b. Burnup of Gadolinium faster than fuel burnup
c. Burnup of fuel greater than Gadolinium

REF.: Reactor Theory Chapter 7, p. 41-43

5-8 a. increased core average void content (1.5)
increased core pressure drop
reduced critical power ratio
possible recirc. pump cavitation

b. Maintain normal vessel water level (on separators) and avoid low water level which will promote carryunder (0.5)

REF.: Nuclear Steam Supply System L.P. #38, p.23

5-9 1- MCPR is the lowest critical power ratio where CPR= the ratio of that power required to produce boiling transition to actual bundle power. MCPR limits protect against fuel damage from boiling transitions. (1.0)

2- LHGR is heat generation rate per unit length of fuel (kw/ft). Limits on LHGR protect against exceeding 1% plastic strain on the zirc clad from overpowered fuel pellets. (1.0)

3- APLHGR -- Average LHGR per each 6" length of fuel bundle. Maintaining APLHGR limit (or maximum APLHGR limits) avoids exceeding 2200°F post-LOCA clad temperature. (1.0)

REF.: Heat Transfer and Fluid Flow LP p. 2-162 to 2-171

5-10 To avert water hammer if initiation of the Isolation condensers should occur. (1.5)

REF.: Procedure 317 (Feedwater System) pg. 3

5-11 1- The amount of decay heat available (0.5/ea.)

2- The height of the heat sinks (downcommers) above the heat source (core).

3- Loop flow resistance, which could be caused by core blockage (i.e, all recirc. valves closed).

4- Void formation

NOTE: Other technically correct answers are also acceptable.

REF.: Core Cooling Mechanics (Criteria for Natural Circulaton)
L.P. 400.14 p. 17 and 18

6-1 a. Located on each EMRV downcommer. Prevents drawing torus water up into the downcommers when the steam condenses following system operation. Minimizes steam vent clearing phenomena during subsequent valve operation. (1.0)

b. Hi drywell pressure (0.5)

c. Fire Water (valved into core spray). Condensate. Core spray. (1.0)

d. Three (3) (0.5)

REF.: ADS Lesson Plan (Rev. 0)

6-2 a. To prevent them from giving the downscale alarm unless a channel failure occurs i.e., the downscale alarm for mid and hi range detectors means channel failure. (1.0)

b. Either B-9 (Rx Operating floor equipment hatch area, 119' elevation) or C-9 (Fuel Pool low range, 119' elevation) upon alarm start a 2 minute timer. If the alarm has not cleared before the 2 minute timer times-out, the Rx building normal ventilation will be secured and the Standby Gas Treatment System will auto-initiate. (1.0)

NOTE: The Identification B-9 and C-9 not required for full credit.

c. Latest revision of Standing orders (0.5)

REF.: ARM L.P. #4 (Facility Q and A Bank)

6-3 a. High flow in either the steam or condensate legs of the isolation condenser (3 x normal) (1.0)

b. Use Emergency Condenser Reset Button on Panel 4F. (1.0)

PEF.: Isolation Condenser L.P. #21, p. 10 and 11

(continued on next page)

6-4 TYPE I (Signals) (3.0)

1- Low flow (80 gpm) or in-service filter outlet valve shut

2- Aux. pump cooling water outlet temp Hi (130°F)

3- Non Regen Hx outlet temp Hi (140°F)

4- Hi System, pressure (140 psig)

5- Poison System ON

ACTION:

1- Isolates three (3) valves 16-1, 16-2, and 16-14

2- Trips cleanup recirculation pump(s)

3- Aux. cleanup pump trip from 16-2 valve closure

TYPE 2 (Signals):

1- Hi Drywell pressure (2 psig)

2- Lo Lo Rx level (7'2" TAF)

ACTION:

1- Isolates four (4) valves 16-1, 16-2, 16-14, and 16-61

2- Trips cleanup recirculation pump(s)

3- Aux. cleanup pump trip from 16-2 valve closure

(NOTE: Valve 16-1 = inboard isolation, 16-2 and 16-14 are outboard isolation valves from Rx. 16-61 is return outboard isolation valve).

REF.: Reactor Cleanup System L.P. #43 (Facility Question CV-6)

6-5 The system is operated in the "dynamic test mode" by placing the mode selector switch to DYNAMIC TEST I. This will auto close the drywell isolation valve in System I, ensure the containment spray suction valves are open, and cause the 6" dynamics test valve in System I to open. Manually start pump (51A) by placing containment spray loop I manual start switch to "A" and by then placing its control switch to the "START" position. ESW pump (52A) auto starts appx. 45 seconds after the containment pump starts. (2.5)

REF.: Containment Spray and Emergency Service Water L.P.

(continued on next page)

6-6 a,c,d,h (0.5/ea.)

REF.: Reactor Protection System LP p. 14-23

6-7 An air failure brake is provided to lock up the scoop tube on loss of supplied air (2.0). The Recirc. flow will remain unchanged. (1.0) (3.0)

REF.: Recirculation System Flow Control LP #48 Sect. C.6

6-8 a. Low supply air pressure (<70 psi) (0.5/ea.)
Loss of instrument electrical signal

b. By adjusting the bias adj knobs on each of the individual M/A stations (1.0)

c. Automatically when Rx level reaches 174" indicated (NR GEMAC) (0.5/ea.)
Manual reset button on panel 5F/6F

REF.: Reactor Level and FW Control LP. p. 4, 5, & 7

6-9 Any four (4) at 0.5/ea.

1-Dilution Pump, CWP, and NRW service water pump seal water

2-Isolation condenser makeup

3-CST makeup

4-Air compressor cooling water

5-Core Spray Emergency Supply

6-Fire System

REF.: Facility Question & Answer (FN-10)

6-10 ~~NO - The main pumps that were secured are locked out from staring and subsequent core spray system operations require operation of the backup main pumps. To restart the backup pumps when hi drywell pressure is present requires dispatching an operator to the 4160V switchgear room to place their "69" breaker permissive switches to the trip position and then back to the closed position before a restart can occur. Once the main pumps start and develop sufficient discharge pressure, the operator can then manually restart the booster pumps.~~ (2.0)

Answer is revised to reflect plant modification. See facility Answer for grading.

REF.: Core Spray System LP #10

7-1 a. 1-RPV level below + 138" TAF (0.5/ea.)

2-Drywell pressure above 2.0 psig

3-A condition which requires a Rx scram and power above 2%

4-RPV pressure above ~~1060~~ psig (change to 1050 psig)

5-MSIV closure

6-A condition which requires a scram in the judgement of the operator to either conserve RPV inventory or reduce the release of radioactivity to the environment.

b. If the reactor cannot be shutdown before torus temperature reaches 110°F. (0.5)

REF.: EMG-3200.01, p.1 and 18

7-2 Any four (4) below for full credit (2.5)

1. When the reactor pressure is greater than 600 psia, the combination of reactor core flow and reactor thermal-power-to-water shall not exceed the limit shown on (Fig. 2.1.1) for any fuel type.

2. When the reactor pressure is less than 600 psia or reactor flow is less than 10 percent of design, the reactor thermal power shall not exceed 354 Mwt.

3. The neutron flux shall not exceed its scram setting for longer than 1.75 seconds

4. During all modes of reactor operation with irradiated fuel in the reactor vessel, the water level shall not be less than 4'-8" above the top of the normal active fuel zone.

5. The existence of a minimum critical power ratio (MCPR) less than (1.32 for 7 x 7 fuel and) 1.34 for 8 x 8 fuel shall constitute violation of the fuel cladding integrity safety limit.
NOTE: The 7 x 7 specification is optional since none is in-core.

6. During all modes of operation except when the reactor head is off and the reactor is flooded to a level above the main steam nozzles, at least two (2) recirculation loop suction valves and their associated discharge valves will be in the full open position.

REF.: Nuclear Steam Supply System, L.P. p.26 and Tech. Specs. Sec. 2.1

(continued on next page)

7-3 Primary containment integrity means that the drywell and adsorption chamber are closed and all of the following conditions are satisfied: (2.0)

1. All non-automatic-primary containment isolation valves which are not required to be open for plant operation are closed.

2. At least one (1) door in the airlock is closed and sealed.

3. All automatic containment isolation valves are operable or are secured in the closed position.

4. All blind flanges and manways are closed.

REF.: Tech. Specs. Definition 1.13

7-4 Any four (4) below for full credit (2.5)

1. Contamination Area

2. Airborne radioactivity areas requiring the use of respiratory protection equipment and where MPC-hours are to be recorded

3. Neutron radiation exposure

4. High radiation area exposure

5. Unknown conditions in an area to be entered

6. Maintenance of equipment, controls, or instrumentation which contain radioactive material

REF.: RWP Procedure 915.12, p. 2 of 21

(continued on next page)

7-5 a. At least two (2) (1.0)

b. Must be fully inserted, with one (1) detector in the quadrant where fuel/control rod moves are being performed and one (1) located in an adjacaent quadrant. (1.0)

c. Both channels must read at least 1 cps. (1.0)

REF.: Core off loading procedure 205.4, p4
Core reloading procedure 205.5, p2

7-6 Alarm at 24.9" Hg vac. (2.5)

Rx scram at 23.0"

Main turbine trip at 22"

Turbine bypass valves interlocked closed at 10"

REF.: ABN-3200.14 (Loss of Condenser Vacuum)

7-7 a. 1) Cycle the controller (closed, open, closed several times) for the stuck-open Rv. (0.5)

2) If RV fails to reset, remove the control power fuses in the 480V switchgear room for the stuck-open RV. (0.5)

b. Containment Spray System (1.0)

Emergency Service water system (one or two pump operation)

c. Torus water temperature. Scram at 110°F. (0.5)

REF.: EMG-3200.02 p. 1-4

(continued on next page)

7-8 a. Approximately 100 psig above reactor pressure to ensure the turbine bypass valves remain closed and avoid bypassing steam to the condenser. (201.2, p. 9) (1.0)

b. All rods, except peripheral rods, shall be notch withdrawn until the reactor is critical and adding heat to the coolant. (201.1, p. 5) (1.0)

c. Inert to less than 5% oxygen before mode switch to RUN unless authorized by Plant Operations Director. In any case less than 5% oxygen within 24 hours after mode switch to RUN. (201.2, p.5). (1.0)

d. Sufficient = at least one (1) decade (201.2, p.8) (0.5)
(Second instrument must come on scale while a full decade of range is indicated on first instrument)

REF.: Source procedure and page no. in ( ) above.

7-9 Within one (1) hour establish a continuous fire watch with backup fire suppression equipment as necessary. (2.0)

REF.: Plant Fire Protection System, procedure 333, 10.6.2 and Tech. Specs. 3.12.F.2

8-1 Any three (3) below for full credit: (3.0)

1. If any Safety Limit is exceeded, the reactor shall be shutdown immediately until the Commission authorizes the resumption of operation.

2. The Safety Limit violation shall be reported to the Commission and the Vice President and Director.

3. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the Plant Operations Review Committee and submitted to the Vice President and Director. This report shall describe: 1) applicable circumstances preceding the violation, 2) effects of the violation upon facility components systems or structures, and 3) corrective action taken to prevent recurrence.

4. The Safety Limit Violation Report shall be submitted to the Commission within 10 days of the violation. It shall also be submittted to the ISRC Coordinator.

REF.: Administrative Controls, T.S. 6.7.1

8-2 a. General → Site Area → ALERT → Unusual Event (1.0)

b. General, Site Area, Alert (1.0)

REF.: EPIP-01, EPIP-25, EPIP-27

8-3 a - Orange (information tag • only one recorded in info. log) (0.5/ea.)
b - Blue (electrical test tag)
c - White
d - Yellow

REF.: Equipment Control Procedure 108, p. 12-18 (Rev. 30)

(continued on next page)

8-4 a. One supervisor (GOS) and 4 additional personnel including Equipment operators, New Radwaste Operators, and Site Protection Officers all trained in Fire fighting and qualified by the Fire Protection Supervisor. (1.0)

b. Shall not include the minimum shift personnel necessary for safe shutdown of the unit or any personnel required for other essential functions during a fire emergency. (1.0)

c. The position of STA is required only when the reactor coolant temperature is greater than 212°F or when the reactor mode switch is in the STARTUP or RUN positions. (1.0)

REF.: Procedure 106 (Conduct of Operations) p. 7, 11, 14

8-5 a. When the intent of the original procedure is not altered. (1.5)

b. Two authorized members of GPUNC Management Staff knowledgeable in the area affected by the procedure and at least one of these individuals shall be a member of facility management or supervision holding an SRO license. (1.5)

REF.: Tech. Specs. 6.8.3 (Ammendment No. 69)

8-6 a. 600 ppm — Reason. To give a negative reactivity worth equal to the combined effects of 1) rated coolant voids, 2) fuel Doppler, 3) Xenon, 4) Samarium, 5) temperature change plus 6) shutdown margin. (1.5)

750 ppm — Reason. The additional 25% Boron provides margin for mixing uncertainties. (0.5)

b. 60-120 min. — Reason. 1) To provide for good mixing in the reactor and 2) to override the rate of reactivity insertion due to cooldown of the reactor following the Xenon peak. (1.0)

REF.: Tech. Spec. Basis for 3.2.C, p. 3.2.-6

(continued on next page)

8-7 a. A core alteration is the addition, removal, relocation or other manual movement of fuel or controls in the reactor core. (Control rod movement with the CRD hydraulic system is not defined as a core alteration). (2.0)

b. False (0.5)

REF.: Tech Specs. Definition 1.21

8-8 1) Maximum allowable extension not to exceed 25% of the surveillance interval. But, 2) the combined time interval for any three (3) consecutive surveillance intervals not to exceed 3.25 times the specified surveillance interval. (3.0)

[Example: For a daily (24 hr.) surveillance test, such as SBLC solution temperature and volume, the current interval may be extended 1) for six (6) hours provided 2) the combined delay for the current plus two (2) preceding intervals will not be exceeded by more than six (6) hours total].

REF.: Tech. Specs. Definition 1.24

8-9 a. After 12 rods have been withdrawn may continue if a second licensed operator verifies the rod program is being followed by the licensed operator at the console. (1.5)

b. False (0.5) Before 12 rods have been withdrawn (or prior to startup) one startup is allowed each calendar year if verification of the rod pull by the licensed console operator is made by a second licensed operator and by the Station Engineer from the technical group. (Also, the maximum rod worth must be <1.25%Δk after the core is critical). (1.0)

REF.: Tech. Specs. 3.2.B.2 (a) and (b), p. 3.2-1