U. S. NUCLEAR REGULATORY COMMISSION

REGION V

Report No. 50-206/OL-85-03

Docket No. 50-206

License No. DPR-13

Licensee: Southern California Edison Company
P. O. Box 800
2244 Walnut Grove Avenue
Rosemead, California 91770

Facility Name: San Onofre Nuclear Generating Station, Unit 1

Examination Conducted: July 9-12, 1985

Examiners: John O. Ele for — 8-23-85
Gary W. Johnston — Date Signed
Operator Licensing Examiner

John O. Ele for — 8-23-85
Clyde W. Shiraki — Date Signed
Operator Licensing Examiner

Approved By: John O. Ele for — 8-23-85
Robert J. Pate, Chief — Date Signed
Reactor Safety Branch

Summary:

During this requalification cycle, seven licensed operators of the staff of 57 were examined. This included four Reactor Operators and three Senior Reactor Operators. The NRC prepared a complete written examination and administered it to the operators on July 9, 1985. The examiners also administered orals to six of the participants (one operator was exempted from oral examination). All of the operators passed the written and oral examinations. The examiners also reviewed facility records of the requalification program to verify compliance with the present requalification program.

Based on the review of facility records and in accordance with NUREG-1021 criteria, the requalification program at San Onofre Nuclear Generating Station, Unit 1 is evaluated as satisfactory.

1. Persons Examined:

A. Reactor Operators:

James F. Cummings	55-02933
Robert E. Froiseth	55-09053
Richard L. Keefer	55-50033
Jerry W. Reynolds	55-08872

B. Senior Reactor Operators:

John H. Custer	55-08869
Mark B. McKinley	55-50031
Howard C. Schutter	55-08874

2. Persons Contacted:

Southern California Edison (SCE)

*J. Wambold, Training Manager
*J. Tate, Assistant Manager, Operations
*R. Mette, Supervisor of Operations Training
*M. Kirby, Nuclear Training Administrator

NRC

R. Huey, Senior Resident Inspector
A. D'Angelo, Resident Inspector
J. Tatum, Resident Inspector

*Denotes those present at exit meeting.

3. Program Evaluation

Required for Satisfactory:

The requalification program was evaluated based upon the criteria of Examiner Standard ES-601 of NUREG-1021. The requirement for a satisfactory program is more than 80 percent of the evaluated operators passed all oral examinations or sections of the written examination administered by the NRC.

Performance:

The NRC administered examinations to four reactor operators and three senior reactor operators. This included a complete written examination and oral walkthrough with six of the operators. One operator took only the written examination, he was exempted from the oral portion because he received an oral examination in October of 1984 from the NRC. The results of the examinations was a pass rate of 100 percent for all the operators for both the written and oral examinations.

Evaluation:

Satisfactory.

4. Review of Facility Records

The license examiners examined facility records associated with the requalification program since the last program evaluation by the NRC on October 15-19, 1984. This included a review of the schedule and the administration of monthly and annual examinations, and the completeness of the records for the licensed personnel. The review of the examinations included an evaluation of the level of difficulty and the appropriateness of the examinations based on the facility's requalification program. Also examined were the scheduled lectures and simulator training sessions. The review showed that the licensee is maintaining the program as described in the requalification program submitted November 8, 1984.

5. Examination Review

At the conclusion of the written examination, a review of the examination was conducted in accordance with NUREG-1021. The facility comments from this review were documented on the facility review copy of the examination key. These comments were discussed with the NRC examiner and were incorporated into the key prior to grading.

6. Exit Meeting

At the conclusion of the site visit, the examiners met with the facility representatives denoted in section 2 of this report to discuss the results of the examination to that point.

Resolution of Facility Comments

Reactor Operators Examination

1.0 Facility Comment: Question 1.02

(a) The reviewer stated that the control group insertion limits were to maintain the reactor subcritical following a reactor trip.

Resolution:

(a) Comment accepted.

2.0 Facility Comment: Question 1.03

(c) The reviewer stated that exial offset would become more positive rather than negative.

Resolution:

(c) Comment accepted.

7.0 Facility Comment: Question 3.05

(c) The reviewer pointed out that the power range instruments utilize the intermediate range instruments compensated ionization chambers for some channels. It was also pointed out that the portion of the key for this question covered the topic of compensation that does not relate to the operation of the chamber.

Resolution:

(c) The examiner will accept both comments, the key will be adjusted for the point value.

8.0 Facility Comment: Question 3.06

(b) The reviewer commented that the term "low differential pressure" is not communally used. The term typically used at the facility is "high reverse delta P".

Resolution:

(b) Comment accepted, will add "or high reverse delta P" to the key.

9.0 Facility Comment: Question 3.08

(a) "Should be 'Shutdown Rods Not Withdrawn'."

Resolution:

(a) The examiner recognizes that the term 'Shutdown bank deviation alarm' is not in the plant vernacular. Since this is so, the examiner will accept as an alarm function either case.

10.0 Facility Comment: Question 3.09

(a) For part 5 the reviewer felt that the feedback from Bistable TC-413B was appropriate to the question.

(b) Part 1 was also felt by the reviewer to indicate greater specificity than was appropriate as far as the Bistable TC-413B. In part 2 the reviewer felt the portion referring to the speed signal being fed through an auto/manual group selector switch to be more specific than required by the question.

Resolution:

For both (a) and (b) the comments are accepted.

11.0 Facility Comment: Question 3.10

(b) "The channel has no present control functions."

(c) The reviewer here again was concerned about the level of specificity of the question concerning the energization of a solenoid valve.

Resolution:

(b) The examiner will accept the answer of no control function.

(c) Agreed, will change key.

12.0 Facility Comment: Question 3.11

(a) The reviewers concern here was that the candidate may list each permissive and its purpose.

(b) For part 3 the reviewer stated that in this case the portion of the question pertaining to a decrease in power of five percent or greater being detected in any power range channel was not germain to the question.

Resolution:

(a) The examiner would concede that if a candidate answered in that fashion he would be answering the question appropriately, therefore, will accept such a response.

(b) Agreed, will strike that portion.

13.0 Facility Comment: Question 4.01

(b) "For a pressure controller Shift Superintendent cannot approve exception."

Resolution:

(b) Agreed, will change key.

14.0 Facility Comment: Question 4.03

(a) The reviewer noted a typographical error.

Resolution:

(a) That will be corrected.

15.0 Facility Comment: Question 4.04

For all of the parts the reviewers felt the answers the candidates might give could be other events that would correspond to the symptoms listed.

Resolution:

The examiner agrees and will consider other events as answers so long as they correspond to the symptoms.

16.0 Facility Comment: Question 4.05

(b) The reviewer noted that there are other systems that are required by the Technical Specification.

Resolution:

(b) If the candidate indicates that something in the Technical Specification would require the shutdown that will be sufficient as an answer.

Resolution of Facility Comments

Senior Reactor Operators Examination

1.0 Facility Comment: Question 5.2

Answers (a)(1),(3) are not in Technical Specifications.

Resolution:

Agreed, answer is in curve book, paragraph 4.3 page 45. Will change key reference.

2.0 Facility Comment: Question 5.3

(b) If the flux in the bottom is reduced the term associated with the reactivity in the bottom of the core decreases therefore the equation becomes more negative.

Resolution:

Comment accepted, key will be changed.

3.0 Facility comment: Question 5.5

(c) The steam would be saturated.

Resolution:

Agreed, key will be changed.

4.0 Facility Comment: Question 6.01

(b) Leak off past the thermal barrier to provide seal.

Resolution:

Accepted, key will be changed.

5.0 Facility Comment: Question 6.02

The reference may be a lesson plan vice study guide.

Resolution:

It is Study Guide 3 not 16 the key will reflect this.

6.0 Facility Comment: Question 6.06

(b) Use noun name instead of bistable number.

Resolution:

Either one will be accepted, key will be changed to reflect.

7.0 Facility Comment: Question 7.4

(a) If RCPs stopped, to restart with level greater than 80 percent, must be less than 50 deg. delta T secondary to primary. Reference S01-4-3.

(c) Or 1 PORV and block valve open is equivalent to required vent path.

Resolution:

Accepted, key will be changed.

8.0 Facility Comment: Question 7.6

(a) Or terminate boiling in the core.

(c) Instead of the auxiliary spray line through the loops via RHR loop inlet valves.

Resolution:

Accept both comments, key will be changed.

9.0 Facility Comment: Question 8.1

Station Order S01-0-100 superseded by TCN 6-1, refer to S01-14-13.

Resolution:

Items 3, 4, 5, and 6 will be accepted as answers any three of which will give full credit with 0.67 points each.

10.0 Facility Comment: Question 8.3

"Operators are not required to memorize Emergency Action Levels."

(b) This is only an Unusual Event if shutdown required. If shutdown is not required then no classification is required.

(d) Only if an alarm occurs.

Resolutions:

Comment about memorization taken into consideration.

(b) Accepted, will include in key.

(d) Accepted, will include in key.

11.0 Facility Comment: Question 8.4

3. May just say "Cognizant Functional Division Manager (CFDM) or his designee.

Resolution:

Accepted, will change key.

12.0 Facility Comment: Question 8.5

And at least hot shutdown within the following six hours.

Resolution:

Accepted, key will be changed and point value adjusted.

13.0 Facility Comment: Question 8.8

(d) This should be no, eight eight hour shifts would be 64 hours in 72.

Resolution:

Accepted, will change key.

~~FACILITY REVIEW~~ Corrected Key

U. S. NUCLEAR REGULATORY COMMISSION
SENIOR REACTOR OPERATOR EXAMINATION

Facility: SONGS 1 (Requal.)

Reactor Type: Westinghouse

Date Administered: July 9, 1985

Examiner: G. W. Johnston

Candidate: ____________

INSTRUCTIONS TO CANDIDATE:

Use separate paper for the answers. Write answers on one side only. Staple question sheet on top of the answer sheet. Points for each question are indicated in parenthesis 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 three hours and forty minutes after the examination starts.

Category Value	% of Total	Candidate's Score	% of Cat. Value	Category
15.0	25.0			5. Theory of Nuclear Power Plant Operation, Fluids, and Thermodynamics
15.0	25.0			6. Plant Systems Design, Control and Instrumentation
15.0	25.0			7. Procedures - Normal, Abnormal, Emergency and Radiological Control
15.0	25.0			8. Administrative Procedures, Conditions, and Limitations
60.0				TOTALS

Final Grade ________ %

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

____
Candidate's Signature

MICHAEL J KIRBY SECT 6
DENNIS WILCOCKSON SECT 7
JOHN TRAYNOR SECT 8
JOE BOCAVICH SECT 5

SECTION 5 THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS AND THERMODYNAMICS

5.1 (2.0)

Production and removal of xenon are described by the following relationship.

Production from fission + Decay from I-135 = Decay to Cs-135 + Burnup by neutron absorption

(a) Explain the change in this relationship following a reactor trip. (1.0)

(b) If a reactor has been at a stable power level for a number of weeks, an equilibrium concentration of xenon will exist. If a reactor trip then takes place, why will there be an increase in xenon concentration over the equilibrium concentration? (0.5)

(c) Explain the sharp decrease in xenon concentration that results from a return to full power operation simultaneously with the point in time at which peak xenon occurs. (0.5)

(a) The terms for Production from fission and Burnup by neutron absorption are nonexistent (0.5) since fission and neutron production cease. (0.5)

(b) The half-life of I-135 is shorter than that of Xe-135 (0.25), thus, more xenon is being produced than is decaying away. (0.25)

(c) The accumulated xenon burns out in the neutron flux (0.5)(as well as decaying off.)

Reference: WNTC Notes, Phase 1, Module B, B-1, page 52

5.2 (2.5)

(a) State three reasons for control group insertion limits. (1.5)

(b) State one reason why flux redistribution following a reactor trip from full power to hot zero power conditions may cause reactivity to increase. (0.5)

(c) Explain why Technical Specifications limit the average burnup of the core to 21,000 MWD/MTU. (0.5)

(a) Three required for full credit. 0.5 points each.

(1) To maintain an acceptable power distribution during normal operation, (2) Limit effect of a rod ejection accident, (3) Maintain reactor subcritical during the design accident (main steamline break), (4) Provide adequate shutdown margin.

(b) At zero power, the moderator in the top of the core is relatively cooler than at full power (0.25), so with negative MTC (0.25), reactivity is higher .

(c) Reactivity addition accident analyses assume a maximum negative value of MTC. This value may be exceeded (0.25) if the burnup specification is exceeded because MTC becomes more negative with increasing burnup (0.25).

Reference: Technical Specifications 3.9, 3.5.2, Curve Book pages 44-45, 66-67.

5.3 (2.0)

The Technical Specifications require that the average position of control bank 2 be at least 90% withdrawn after 20% of the core cycle.

(a) How would axial offset change immediately after this bank is rapidly borated out while maintaining power constant? Explain your answer. (1.25)

(b) Four hours after rod motion, what changes, if any, do you expect to have occurred to axial offset? Explain your answer. (0.75)

(a) Axial offset should become less negative or, perhaps, slightly positive (0.25), because the power distribution will shift from a peak in the lower two thirds of the core to a more symmetrical distribution (0.5) and because axial offset is (Ptop - Pbottom)/(Ptop + Pbottom). (0.5)

(b) Axial offset should become less negative (0.25) because xenon will buildup in the reduced flux in the lower two thirds of the core (0.25), and burnout in the increased flux in the upper one third of the core (0.25).

Reference: Curve book, Technical Specification 3.5.2

5.4 (2.5)

(a) The plant is operating at equilibrium steady state zero power conditions, all rods out, 2934 MWD/MTU when an alert operator notices that Tave has increased 5 degrees F in the last twenty four hours. Calculate, using Figure 1, the change in boron concentration that could have caused this increase. Show your work. (1.0)

(b) Considering each case independently, does differential boron worth increase or decrease as (1) Boron concentration increases, (2) Control rods are inserted, (3) Moderator temperature increases. (1.5)

(a) Change in Boron Concentration = MTC x Change in Temperature x Boron Worth (0.5). So Boron Concentration must have decreased by 7.0 x 5.0 x 0.15 = ~~52.5~~ 5.25 ppm (0.5)

(b) (1) Differential boron worth decreases (less negative) as boron concentration increases (0.5)
(2) Differential boron worth decreases as control rods are inserted (0.5)
(3) Differential boron worth decreases as moderator temperature increases (0.5)

Reference: Academic Program for Nuclear Plant Personnel, page 4-119, WNTC Notes, Reactor Theory, Page 65.

TABLE 1

END-POINT BORON CONCENTRATIONS, BORON WORTH, AND MODERATOR TEMPERATURE COEFFICIENTS

(No Redistribution Effects)

HZP

2934 MWD/MTU

Rod Condition	C_B (ppm)	Boron Worth (ppm/1000 pcm)	Moderator Temperature Coefficient (pcm/°F)
ARO	1355	150.36	-7.0
Control Group 2 In	991	147.43	-12.8
Control Groups 2+1 In	747	144.13	-17.2

Doppler Temperature Coefficient = -1.9 pcm/^{o}F

5.5 (1.5)

(a) The reactor is shutdown and pressurizer pressure is 1425 psig. At what temperature will the reactor coolant system be 35 degrees F subcooled, according to the Steam Tables? (0.5)

(b) A primary to containment atmosphere steam leak is occurring from the pressurizer with the pressurizer at 1600 psig. If containment pressure is 14.7 psia, what kind of steam (saturated, subcooled, or superheated) should be expected as soon as the steam from the break has depressurized to containment pressure? (0.5)

(c) For a leak identical to that in (b) above, if containment pressure is 50 psia, what kind of steam (saturated, subcooled, or superheated) should be expected as soon as the steam from the break has depressurized to containment pressure? (0.5)

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(a) ~~583~~ degrees F (0.5)

(b) At 14.7 psia, the steam will be superheated (0.5).

(c) At 50 psia, the steam will be saturated (0.5).

Reference: Steam Tables, Mollier Diagram

5.6 (2.5)

Refer to Figure 2 from Technical Specification 3.1.3.

(a) Explain why, in theory, the curves in the Figure should be expected to shift to the right for the second 6.0 effective full power years, the third 6.0 full power years, etc. (1.5)

(b) The reactor coolant system is at 1600 psig and 300 degrees F due to a rapid cooldown transient. How should pressure and/or temperature be changed (if at all) to minimize the probability of overstressing the reactor vessel? Explain your answer. (1.0)

(a) The shift in curves corresponds to a change in RTNDT (a measure of the fracture toughness of the vessel, or of its degree of embrittlement) during that period. (0.75) RTNDT is expected to increase with reactor vessel fluence, according to the Technical Specifications, which will steadily increase during these intervals, since fluence is neutron flux times time. So, the curves shift to the right. (0.75)

(b) Pressure should be decreased while maintaining temperature constant. (0.5) This will minimize thermal stress, and therefore total stress on the reactor vessel. (0.5)

Reference: Technical Specification 3.1.3

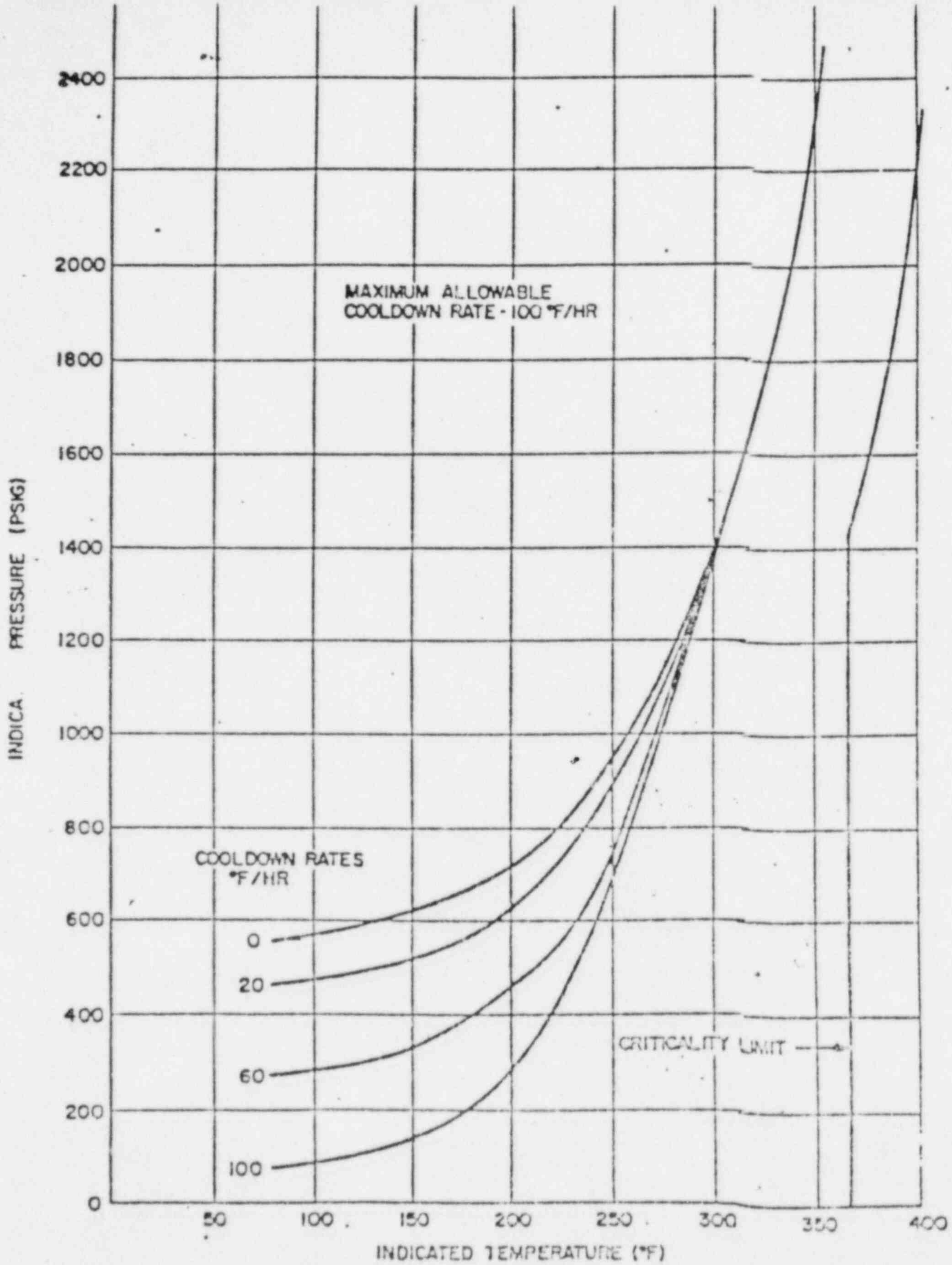

FIGURE 3.1.3b SCE REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS APPLICABLE FOR FIRST 6.0 EFFECTIVE FULL POWER YEARS.
T_{ERROR} = 10°F, P_{ERROR} = 60 PSIG

Change No. 14
Date: 4/12/74

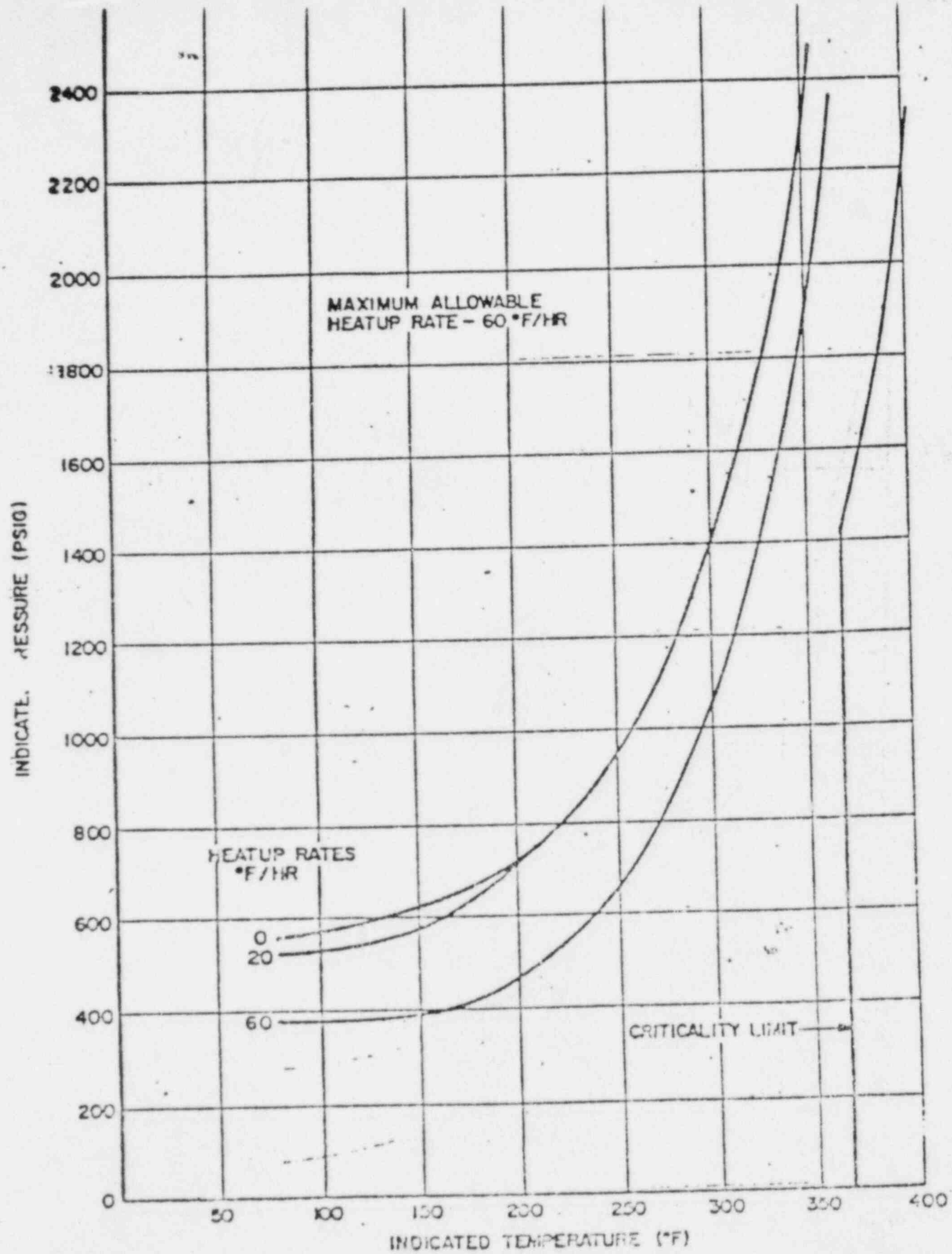

FIGURE 4. SCE REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE FOR FIRST 6.0 EFFECTIVE FULL POWER YEARS.
$T_{ERROR} = 10°F$, $P_{ERROR} = 60$ PSIG

Change No. 1
Date: 4/1?7

5.7 (2.0)

The reactor is operating at 85% reactor power when control bank 2 is shimmed out 10 steps. On the attached Figure 3, sketch reactor power, fuel temperature, Thot, and Tcold versus time until new final values are reached. Numerical estimates are not required, but carefully show and or state the relative relationships of the peaks and trends of these parameters. Assume no reactor trip occurs.

See attached figure (0.5 points per variable.)

Reference: Academic Program for Nuclear Power Plant Personnel, Volume III, Nuclear Power Plant Technology, Chapter 3, Section C

END OF SECTION 5

5.7

KEY

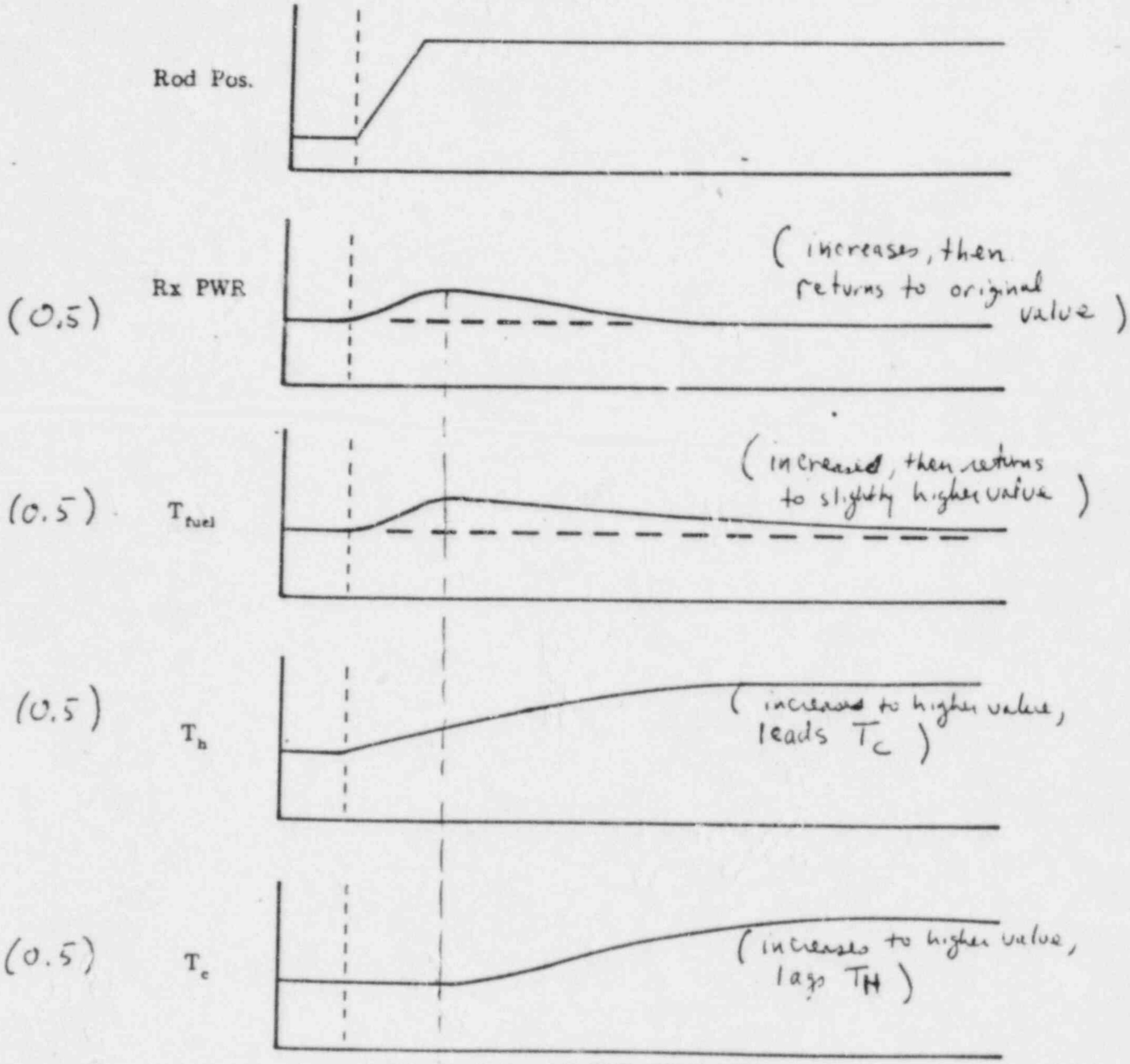

Figure Control Rod Withdrawal

SECTION 6

SENIOR REACTOR OPERATORS EXAMINATION

Plant System Design, Control and Instrumentation

6.01 (2.0)

For the Reactor Coolant Pump Seal Water Circuit:

(a) What is the primary function of the Reactor Coolant Pump Seal Water Circuit in conjunction with the pump seals it supplies? (1.0)

(b) In the event of a failure (loss of seal water flow) how can seal water be supplied? (1.0)

6.01 Answer:

(a) To effectively prevent the leakage of reactor coolant along the Reactor Coolant Pump shaft. (1.0)

(b) (Providing the loss of flow is not from a ruptured line) the Test pump in the CVCS is sized to supply adequate flow to all three RCP's. (1.0)

Reference: Study Guide No. 4, pages 13, 14, 17, and 24.

Leak off past Thermal Barrier to provide seal.

6.02 (2.0)

Regarding the Pressurizer:

(a) Give three reasons why a 1 gpm flow rate is maintained around the spray valves into the Pressurizer. (1.5)

(b) The Pressurizer level program has a ramp function from 25% at no load to 37% at full load. Why is the Level Control System provided with a programmed level? (0.5)

6.02 Answer:

(a) 1) To reduce thermal stresses and thermal shock when the spray valves open. (0.5)

2) To help maintain uniform water chemistry (boron concentration). (0.5)

3) And to maintain the desired temperature in the surge line. (0.5)

(b) To allow for surges into and out of the Pressurizer during load transients. (0.5)

Reference: Study Guide No. ~~16.~~ 3

Check Reference No.. It may be a Lesson Plan vice Study Guide

ok

6.03 (3.0)

For the Diesel Generators:

(a) List six trips for the Diesel Generator engines that are in service for a manual start. (3.0)

6.03 Answer:

(a) Any 6 (0.5) each.

ok

1) Engine Overspeed.
2) Generator Differential.
3) High Crankcase Pressure.
4) Low-low Engine Oil Pressure.
5) Low-low Turbo Oil Pressure.
6) High-high Jacket Water Temperature.
7) High Vibration Engine or Turbo.
8) High-high Lube Oil Temperature.
9) High Main Bearing Temperature.

Reference: Study Guide No. 98, page 9.

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6.04 (2.0)

Regarding the steam generator level control system:

(a) The steam generator level control system is commonaly referred to as a three element control system, what are those three elements? (1.5)

(b) If there are three elements used for control in this system, why then is steam pressure (not one of the three 'elements') also an input to the system? (0.5)

6.04 Answer

(a) 1. Feedwater flow. (0.5)

2. Steam flow. (0.5)

3. Steam generator level. (0.5)

(b) To provide input for density compensation in the Steam Flow Computer. (0.5)

Reference: OT-1045 and OT-1064 "Steam Generator Water Level Control", Study Guide No. 62.

dk

6.05 (3.0)

For the curve on the following page (Figure 3.2) identify the following parts of the curve:

(a) I (0.5)

(b) II (0.5)

(c) III (0.5)

(d) IV (0.5)

(e) V (0.5)

(f) VI (0.5)

6.05 Answer

(a) Recombination (0.5)

(b) Ionization (0.5)

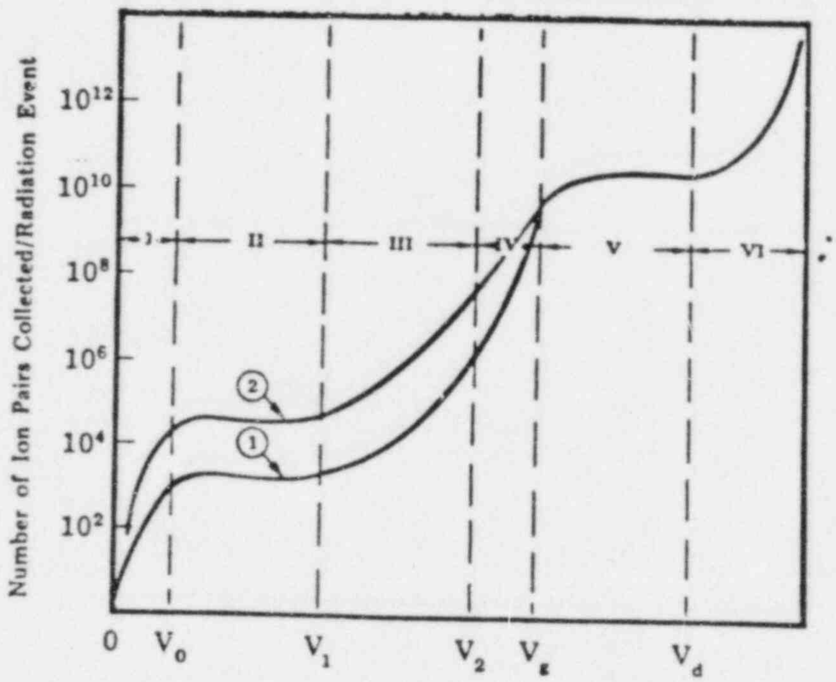
(c) Proportional (0.5)

(d) Limited Proportional (0.5)

(e) Geiger-Mueller (0.5)

(f) Continous Discharge (0.5)

Reference: General Physics Volume II, Chapter 3, Section E



Curve 1: Radiation event of lower specific ionization.
Curve 2: Radiation event of higher specific ionization.

Figure 3.2

6.06 (3.0)

For the Control Rod Drive Summing Computer:

(a) The computer receives inputs from 5 sources, name 4 of those sources. (2.0)

(b) The output from the computer goes to two places, what are those places and what function does the signal that is provided perform? (1.0)

6.06 Answer:

(a) Any 4 (0.5) each:

1) Average Tavg.

2) Tref.

3) Primary pressure.

4) Nuclear flux.

5) Feedback from Bistable TC-413B indicating demanded rod postion.

Rod Direction Control

(b) 1) (Bistable 413B)(0.25), to move rods in or out (0.25).

2) Rod speed controller (0.25. Which produces a demanded speed to be fed to the rod drive system through an auto/manual group selector switch (0.25).

Reference: Study Guide 12, page 4.

END OF SECTION 6

Use noun name instead of bistable Number.

SECTION 7 PROCEDURES: NORMAL, ABNORMAL EMERGENCY, AND RADIOLOGICAL CONTROL

7.1 (2.5)

Loss of the secondary heat sink can occur if the total feedwater flow capability decreases to less than 165 gpm.

(a) Under normal containment conditions, what two parameters are monitored to determine if reactor coolant system feed and bleed is required? (1.0)

(b) What two parameters are monitored to determine if adverse containment conditions exist? (1.0)

(c) If a loss of secondary heat sink has occurred, prior to initiating reactor coolant system feed and bleed, what two options are available to restore the secondary heat sink? (0.5)

(a) Steam generator wide range level (0.5), and RCS pressure (cycling on PORV setpoint) (0.5).

(b) Containment pressure (0.5) and containment radiation level (0.5)

(c) Restore auxiliary feedwater flow (0.25) and restore main feedwater flow (0.25).

Reference: EOI S01-1.3-1

7.2 (2.5)

The response to a potential loss of core cooling attempts to prevent the reactor core from reaching an inadequate cooling condition.

(a) State two symptoms of a loss of core cooling. (1.0)

(b) When responding to a potential loss of core cooling, to what source of water is the charging pump suction aligned? (0.5)

(c) What are the two reasons for using the source of water in (b) above? (1.0)

(a) Five or more core exit thermocouples greater that 680 degrees F. (0.5)
RCS subcooling margin less than 40 degrees F. (0.5)

(b) Refueling water storage tank. (0.5)

(c) Volume available (0.5) and elevated boron concentration (0.5)

Reference: EOI S01-1.2-2

7.3 (1.5)

During CCW System Operation, Radiation Monitoring Channel R-1217 should be operable to monitor activity level.

(a) If this channel should become inoperable, how may the presence of leakage from the CCW system to the saltwater cooling system be determined? (0.5)

(b) If CCW system leakage to the saltwater cooling system exists, what action is required? (1.0)

(a) Observe surge tank level over a period of time. (0.5)

(b) Sample the CCW system (0.5) and estimate the activity being released (0.5).

accept also isolate Hx & place stby Hx in service

Reference: SO1-4-19

7.4 (2.0)

When the reactor coolant system pressure is less than 400 psig and pressurizer water level is greater than 50%, state the condition or operating restrictions of the following with regard to the prevention of overpressurization.

(a) Reactor coolant pump operation (0.5)

(b) Centrifugal charging pumps (0.5)

(c) Overpressure mitigation system (1.0)

(a) Reactor coolant pumps must operate continuously. (0.5)
Also accept: In order to restart RCP's, if pressurizer level is <80%, the primary to secondary delta temperature must be <50 degrees F.

(b) A maximum of one of the two centrifugal charging pumps shall be operable. (0.5) (The charging pump which is to be declared inoperable shall have its motor breaker racked out and caution tagged.)

(c) At least one of the following overpressure mitigation systems shall be operable.

Two power operated relief valves (PORV's) (with a lift setting of less than or equal to 500 psig), (0.5) <u>or</u>

Reactor coolant system vent path(s) (0.5) (of greater than or equal to 1.75 square inches.)
Also accept: One PORV and its block valve open since this is equivalent to the required vent path.

Reference: OI S01-3-1

7.5 (2.5)

Reactor coolant pump and/or residual heat removal pump operation is required under certain circumstances.

(a) What are the requirements regarding reactor coolant pump and residual heat removal pump operation during boron dilution operations? (1.0)

(b) Why is the requirement in (a) imposed? (0.5)

(c) Give two reasons why the requirements for reactor coolant pump and residual heat removal pump operation change for a boron injection operation. (1.0)

(a) Either a reactor coolant pump (0.5) or a residual heat removal pump (0.5) shall be in operation.

(b) To prevent the formation of areas in the core of reduced boron concentration. (0.5)

(c) Thermal circulation will cause the boron to flow to the core. (0.5)
Lack of further mixing cannot result in areas of reduced boron concentration within the core. (0.5)

Reference: Technical Specification 3.1.2

7.6 (3.0)

After the occurrence of a LOCA in one of the cold legs, it becomes necessary to transfer to hot leg recirculation.

(a) What are the two conditions or occurrences that hot leg recirculation is designed to prevent? (1.0)

(b) State the components involved and describe the the flowpath used during hot leg recirculation. (1.0)

(c) After establishing the normal line-up for hot leg recirculation, no flow is evident. How may the line-up be changed in an effort to provide flow? State the components involved and describe the flow path. (1.0)

(a) To prevent or terminate boiling in the core (0.5), and prevent the precipitation of boric acid (0.5).

(b) From the charging pumps (0.5), through the auxiliary spray line to the pressurizer (0.25), through the surge line to the hot leg (0.25).

(c) From a refueling water pump (0.25), through the letdown line (0.25) through a cross tie (0.25), to the loops via residual heat removal loop inlet valves. (0.25)

References: EOI S01-1.0-24

7.7 (1.0)

The radiation dose standards for individuals in restricted areas are proscribed in 10 CFR 20.101.

(a) What are the radiation dose standards per calendar quarter for individuals for the following?

1) Whole body

2) Hands

3) Skin of the whole body

4) Lens of the eye

1) 1.25 rem

2) 18.75 rem

3) 7.5 rem

4) 1.25 rem

Reference: 10 CFR 20.101

END OF SECTION 7

SECTION 8 ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS

8.1 (2.0)

Station Order SO1-14-13 contains certain Operations Reporting Requirements.

(a) State four conditions of which the Unit 1 Superintendent or his designee and the on-duty Shift Technical Advisor must be notified by the Shift Supervisor as soon as practical. (2.0)

.67

(a) Any three required for full credit, 0.~~35~~ each.

1. An unplanned unit load reduction or reactor trip.

2. An unplanned or unexplained major system or component failure.

3. Any significant event which cannot be corrected by operators, or any abnormal operating event, which, in the judgment of the Shift Supervisor, warrants notification of the Unit 1 Superintendent and/or on-duty Shift Technical Advisor.

4. Any event that would be reported as a Reportable Occurrence.

Reference: Station Order SO1-0-100, paragraph IV.J

8.2 (1.0)

10 CFR 55 allows certain exemptions to the requirement to hold an operator's license.

(a) Under what conditions may a nonlicensed individual manipulate the controls of the reactor plant? (1.0)

(a) An unlicensed individual may manipulate the controls of the reactor plant as a part of his training program to qualify as an operator (0.5) under the direction of a licensed operator (0.5).

Reference: 10 CRF 55.9

8.3 (2.0)

In reference to Emergency Action Levels (EAL's), classify the following conditions as an Unusual Event, Alert, Site Area Emergency, or General Emergency.

(a) A loss of all vital DC power for more than 15 minutes. (0.4)

(b) Reactor Coolant Activity exceeds the Technical Specification limits. (0.4)

(c) Loss of physical security control of the facility. (0.4)

(d) A fuel handling accident with the release of radioactivity to the containment or to the fuel handling building. (0.4)

(e) Loss of both Emergency Diesel Generators for greater than two hours. (0.4)

(a) Site Area Emergency (0.4)

(b) Unusual Event (0.4)
This is an unusual event only if a shutdown is required. If no shutdown is required, then it is not classified. (EPIP S01-VIII-1, Tab C1-4)

(c) General Emergency (0.4)

(d) Alert (0.4)
This is an alert only if an alarm occurs. (EPIP S01-VIII-1, Tab A2-5)

(e) Unusual Event (0.4)

Reference: San Onofre Nuclear Generating Station Emergency Plan, Section 4.1

8.4 (2.5)

Technical Specification 6.8.1 reads as follows:

Written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the requirements and recommendations of Sections 5.1 and 5.3 and ANSI N18.7-1976, Administrative Controls for Nuclear Power Plants; Appendix "A" of USNRC Regulatory Guide 1.33, Rev.1, Quality Assurance Program Requirements (Operation); Paragraph 2.2.1 of <u>Fire Protection Program Review, BTP APCSB 9.5-1, San Onofre Nuclear Generating Station, Unit</u> 1, March, 1977; the OFFSITE DOSE CALCULATION MANUAL; the PROCESS CONTROL PROGRAM: and Quality Assurance Program for effluent and environmental monitoring using the guidance in Regulatory Guide 1.21 Revision 1 June 1974 and Regulatory Guide 4.1 Revision 1 April 1975; except as provided in 6.8.2 and 6.8.3 below.

(a) Temporary changes to procedures written in accordance with 6.8.1 above may be made provided three conditions are met. What are these conditions? (2.5)

(a)

1. The intent of the original procedure is not altered. (0.5)

2. The change is approved by two members of the plant management staff (0.5), at least one of whom holds a Senior Reactor Operator's License on the unit affected. (0.5)

3. Any four required for full credit. 0.25 each.
   In lieu of all of below, accept: Cognizant Functional Division Manager (CFDM).

The change is approved by the Station Manager; or by the Manager, Operations; the Manager, Technical; the Manager, Maintenance; the Deputy Station Manager; the Manager, Health Physics; the Manager, Station Security; the Manager, Configuration Control and Compliance; the Manager, Station Emergency Preparedness; or the Manager, Material and Administrative Services; (within 14 days of implementation.)

Reference: Technical Specification 6.8

8.5 (2.0)

Refer to the attached Technical Specification 3.4.3 on the Auxiliary Feed System.

With the reactor in Mode 1, the turbine driven auxiliary feed pump has been out of commission due to mechanical problems for 12 hours. At this time, the motor driven auxiliary feed pump is disabled by water spraying on its electrical controls.

(a) What action(s) is/are required regarding the condition of the plant? (2.0)

(a) Commence unit shutdown to hot standby (0.4) within one hour (0.4) and be in hot standby (0.4) within six hours (0.4), and at least hot ~~standby~~ shutdown within the following six hours. (0.4)

Reference: Lesson Plan OT-1190, Technical Specification 3.4.3

8.6 (2.5)

During the conduct of operations, certain definitions and manning levels are applicable.

(a) What is meant by the term <u>Control Room, at the Controls</u>? (0.5)

(b) What is the <u>normal</u> shift complement, during Modes 1-4, for the positions of Senior Reactor Operator, Reactor Operator, Plant Equipment Operator and Shift Technical Advisor? (2.0)

(a) The area bounded by the three vertical instrumentation boards (0.17), the red line on the floor (0.17), and the Technical Support Center wall (0.17).

(b) Senior Reactor Operator - 2 (0.5)
Reactor Operator - 2 (0.5)
Plant Equipment Operator - 2 (0.5)
Shift Technical Advisor - 1 (0.5)

Reference: Station Order SO1-0-100

8.7 (1.0)

Refer to the attached Technical Specification 3.1.5.

The reactor plant is in Mode 4 and one of the pressurizer PORV's is inoperable but its block valve is operable. The Shift Supervisor wants to shut the block valve and go to Mode 3.

(a) Is this allowed? (0.5)

(b) Why? (0.5)

(a) Yes (0.5)

(b) Action Statement C states that the provisions of Specification 3.0.4 are not applicable. (0.5)

Reference: Technical Specification 3.1.5

8.8 (2.0)

Technical Specification 6.2 provides guidelines in regard to the working of overtime. For each of the situations listed below, state whether the circumstances do or do not comply with those guidelines.

(a) An individual works four separate 8-hour shifts in a 48-hour period. (0.5)

(b) An individual works 14 hours straight. (0.5)

(c) An individual works an 8-hour shift followed by a 6 hour break and then returns to start another shift. (0.5)

(d) An individual works eight 8-hour shifts in a 72 hour period. (0.5)

(a) No (0.5)

(b) Yes (0.5)

(c) No (0.5)

(d) No (0.5)

Reference: Technical Specification 6.2

END OF SECTION 8

U.S. Nuclear Regulatory Commission
Reactor Operator License Examination

Key

Facility: SONGS 1 (Requal.)

Reactor Type: Westinghouse

Date Administered: July 9, 1985

Examiner: G. W. Johnston

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 three hours and forty minutes after the exam starts.

Category Value	% of Total	Candidate's Score	% of Category Value	Category
15.0	25.0			1. Principles of Nuclear Power Plant Operation, Thermodynamics, Heat Transfer and Fluid Flow
15.0	25.0			2. Plant Design Including Safety and Emergency Systems
15.0	25.0			3. Instruments and Controls
15.0	25.0			4. Procedures - Normal, Abnormal, Emergency, and Radiological Control
60.0				TOTALS

Final Grade ________%

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

____
Candidate's Signature

EQUATION SHEET

$f = ma$

$v = s/t$

Cycle efficiency = (Network out)/(Energy in)

$w = mg$

$s = V_o t + 1/2\ at^2$

$E = mc^2$

$KE = 1/2\ mv^2$

$a = (V_f - V_o)/t$

$A = \lambda N$

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

$PE = mgh$

$V_f = V_o + at$

$w = e/t$

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

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

$\Delta E = 931\ \Delta m$

$I = I_o e^{-Ex}$

$\dot{Q} = \dot{m}Cp\Delta t$

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

$Pwr = W_f \Delta n$

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

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

$TVL = +.3/\mu$

$HVL = -0.693/\mu$

$P = P_o 10^{sur(t)}$

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

$SUR = 26.06/T$

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

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

$SUR = 26\rho/\ell^* + (\beta - \rho)T$

$CR_1(1 - K_{eff1}) = CR_2(1 - K_{eff2})$

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

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

$T = \ell/(\rho - \beta)$

$M = (1 - K_{effo})/(1 - K_{eff1})$

$T = (\beta - \rho)/(\lambda\rho)$

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

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

$\ell^* = 10^{-4}$ seconds

$\bar{\lambda} = 0.1\ \text{seconds}^{-1}$

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

$I_1 d_1 = I_2 d_2$

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

$I_1 d_1{}^2 = I_2 d_2{}^2$

$\Sigma = \sigma N$

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

Water Parameters

1 gal. = 8.345 lbm.
1 gal. = 3.78 liters
1 ft^3 = 7.48 gal.
Density = 62.4 lbm/ft^3
Density = 1 gm/cm^3
Heat of vaporization = 970 Btu/lbm
Heat of fusion = 144 Btu/lbm
1 Atm = 14.7 psi = 29.9 in. Hg.

Miscellaneous Conversions

1 curie = 3.7×10^{10} dps
1 kg = 2.21 lbm
1 hp = 2.54×10^3 Btu/hr
1 mw = 3.41×10^6 Btu/hr
1 in = 2.54 cm
°F = 9/5°C + 32
°C = 5/9 (°F-32)

SECTION ONE

REACTOR OPERATOR EXAMINATION

1.01 (2.5)

A refueling is being conducted following removal of many fuel assemblies to locate a foreign object in the bottom of the vessel. As the fuel assemblies are replaced, the neutron count rate increases as shown in Fig. 1.

(a) Define "M" in Figure 1. (0.5)

(b) What is the significance of the point where 1/M = 0? (0.5)

(c) Give one reasonable explanation for the concave downward shape of the plot. (0.5)

(d) The count rate after loading one fuel assembly is observed to increase from 100 cps to 125 cps. If the final keff was .99, what was the keff prior to loading the assembly? (Show your work) (0.5)

(e) At what count rate can the reactor go critical, in theory? (0.5)

(a) Multiplication ratio (fractional neutron increase per generation)

(b) Estimated fuel loading for criticality

(c) A larger percentage of neutrons are being detected as more fuel is loaded nearer the detector, or the detector was located close to an installed source assembly (either acceptable, may be others).

(d) keff prior to load = 0.9875 using C1(1-keff1) = C2(1-keff2)

(e) At any count rate above the installed source level

Ref: WNTC Lesson Notes, Phase 1, #A-8, Rev 1

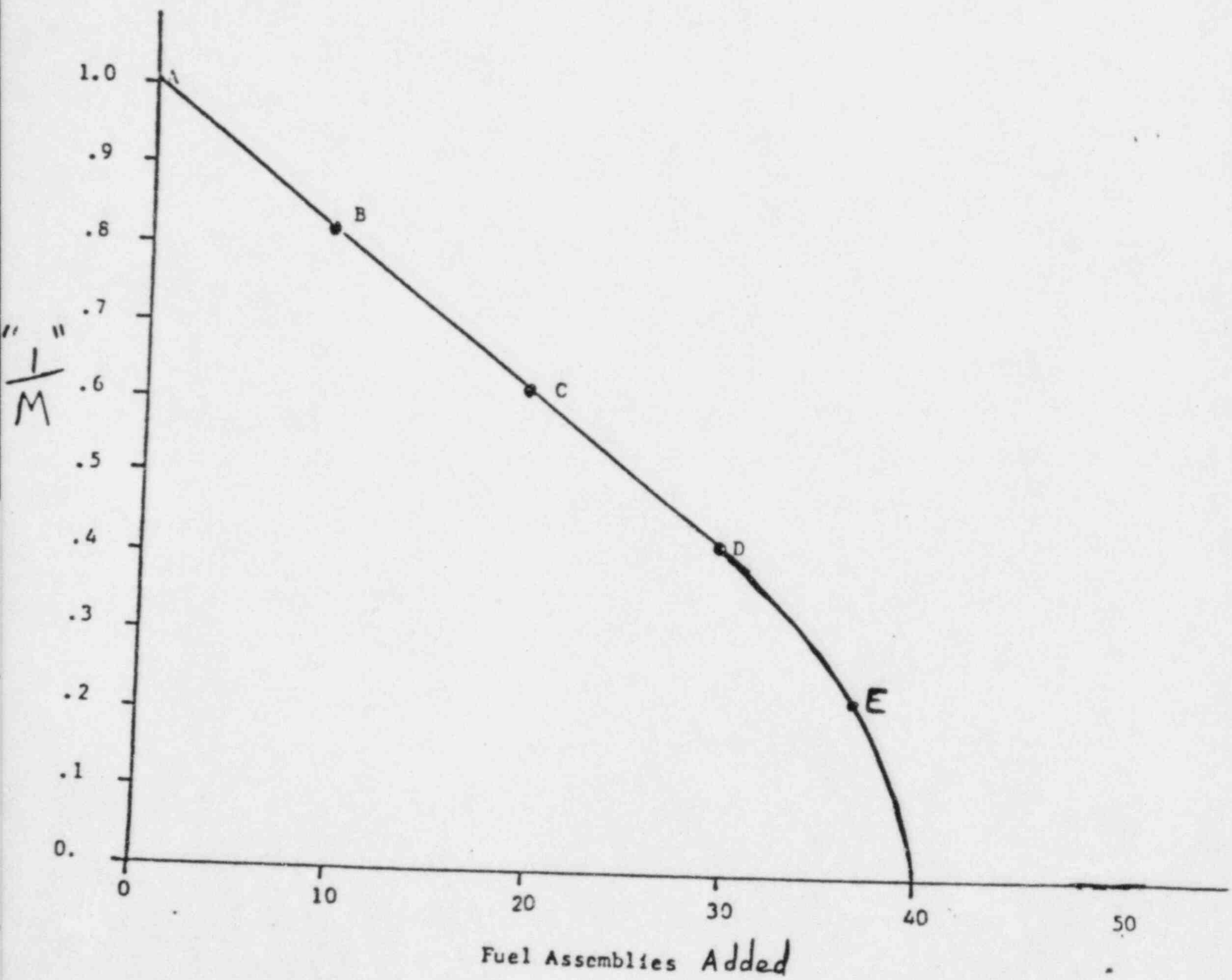


Figure 1: "$\frac{1}{M}$" Plot

1.02 (2.5)

(a) State three reasons for control group insertion limits. (1.5)

(b) State one reason why flux redistribution following a reactor trip from full power to hot zero power conditions may cause reactivity to increase. (0.5)

(c) Explain why the Technical Specifications limit the average burnup of the core to 21,000 MWD/MTU. (0.5)

(a) (1) To maintain an acceptable power distribution during normal operation (0.5), (2) Limit effect of a rod ejection accident (0.5), and (3) Maintain reactor subcritical ~~during design accident (steam break)~~ (0.5). reactor trip

(b) At zero power, the moderator in the top of the core is relatively cooler than at full power (0.25), so with negative MTC (0.25), reactivity is higher.

(c) Reactivity addition accident analyses assume a maximum negative value of MTC. This value may be exceeded (0.25) if the burnup specification is exceeded because MTC becomes more negative with increasing burnup (0.25). 01

Ref: Tech. Spec. 3.9, 3.5.2, "Curve Book" pp. 44-45, 66-67

1.03 (2.75)

(a) Do the Technical Specifications allow operation at full power with control bank 2 eighty per cent withdrawn for a month in the middle of the core cycle? Explain your answer. (1.0)

(b) How will axial offset change immediately after this bank is rapidly borated out while maintaining power constant? Explain your choice. (1.0)

(c) Four hours after the rod motion, what changes, if any, do you expect to have occurred to axial offset? Explain your answer. (0.75)

(a) No. (0.5) Technical Specifications (3.5.2) require that the average position of CB 2 be at least 90% withdrawn (0.25) after 20% of the core cycle (0.25)

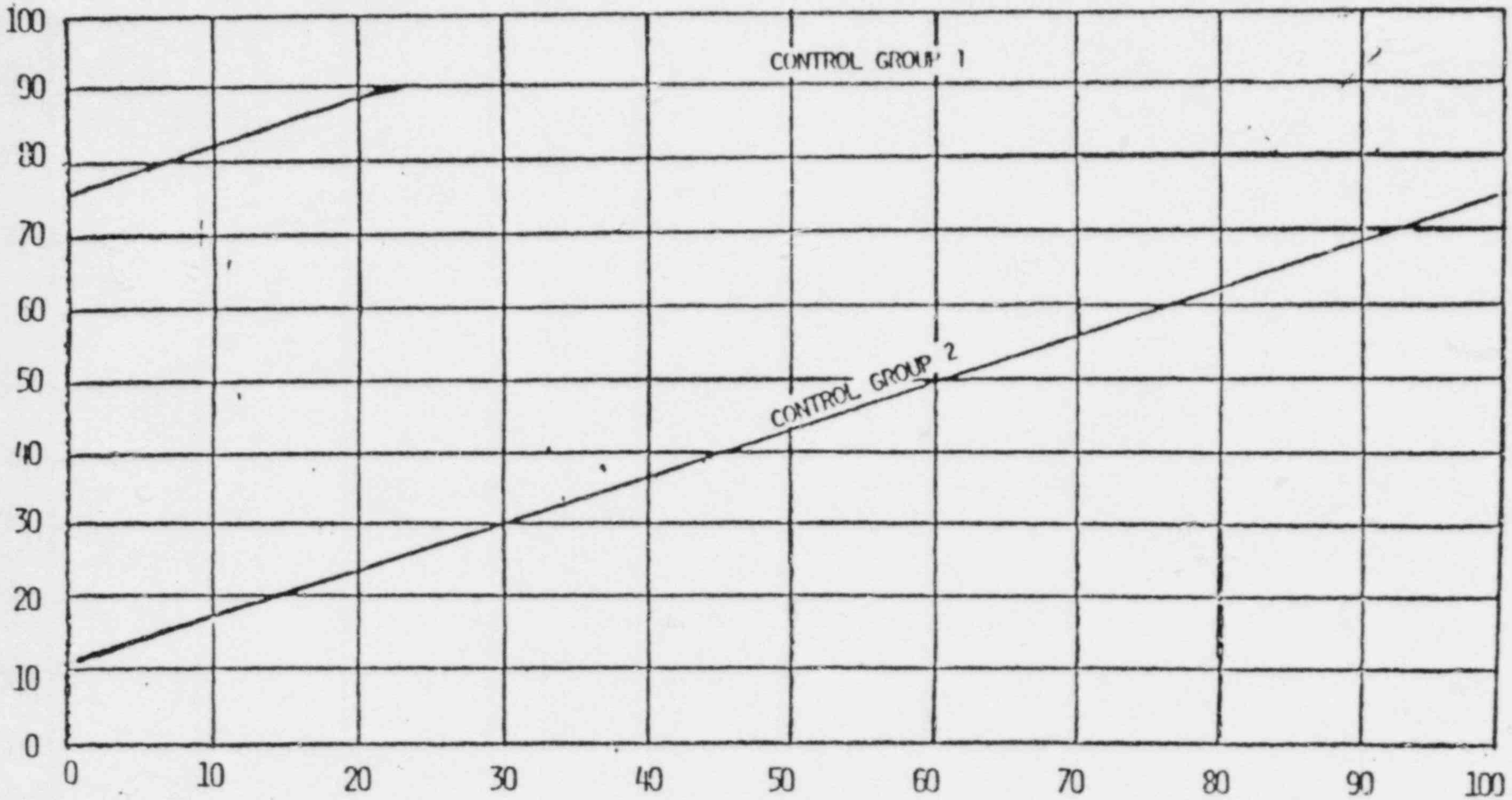
(b) Axial offset should become less negative or, perhaps, slightly positive (0.34), because the power distribution will shift from a peak in the lower two thirds of the core to a more symmetrical distribution (0.33) and because axial offset is (PTOP - PBOTTOM)/(PTOP + PBOTTOM). (0.33)

(c) Axial offset should become more positive ~~negative~~ (0.25) because xenon will build up in the reduced flux in the lower two thirds of the core (.25), and burnout in the increased flux in the upper one third of the core (.25) during this time period.

Ref: "Curve Book", TS 3.5.2

FIGURE 3

LIMITING CONDITION FOR OPERATION - CONTROL GROUP INSERTION LIMITS



POWER (PERCENT OF 1347 MWT)

Amendment No. 56

1.04 (2.5)

(a) The plant is operating at equilibrium steady state zero power conditions, all rods out, 2934 MWD/MTU when an alert operator notices that TAVE has increased 5 F in the last twenty four hours. Calculate, using Table 1, what change in boron concentration could have caused this increase. Show your work. (1.0)

(b) Considering each case independently, does differential boron worth increase or decrease as (1) Boron concentration increases, (2) Control rods are inserted, and (3) Moderator temperature increases. (1.5)

(a) Change in Boron Conc. = MTC X Change in Temperature X Boron Worth (0.5), so Boron concentration must have decreased by 7.0 X 5 X 0.15 = ~~52.5~~ ppm (0.5)
5.25

(b) (1) Differential boron worth decreases (less negative) as boron concentration increases (0.5)
(2) Differential boron worth decreases as control rods are inserted (0.5)
(3) Differential boron worth decreases as moderator temperature increases (0.5)

Ref: Academic Prog. for Nucl. Plant Personnel, p. 4-119, West. Nuclear Trng. Ctr. Reactor Theory Notes, Rev. III, p. 65

TABLE 1

END-POINT BORON CONCENTRATIONS, BORON WORTH, AND MODERATOR TEMPERATURE COEFFICIENTS

(No Redistribution Effects)

HZP

2934 MWD/MTU

Rod Condition	C_B (ppm)	Boron Worth (ppm/1000 pcm)	Moderator Temperature Coefficient (pcm/°F)
ARO	1355	150.36	-7.0
Control Group 2 In	991	147.43	-12.8
Control Groups 2+1 In	747	144.13	-17.2

Doppler Temperature Coefficient = -1.9 pcm/^{o}F

1.05 (2.5)

(a) Define net positive suction head (NPSH). (0.5)

(b) Explain why, for many centrifugal pumps, it is recommended that NPSH be some specified amount greater than zero. (0.5)

(c) The reactor is shutdown with only one reactor coolant pump operating. Then, a second reactor coolant pump is started. Describe what happens to loop flow in every loop and to flow through the core, once flow stabilizes. Calculations are not required. (1.5)

(a) NPSH is the difference between the suction pressure and the saturation pressure of the fluid being pumped (Mathematical explanation also OK)

(b) To prevent cavitation.

(c) Loop flow in the originally operating loop drops slightly (0.25), loop flow in the newly operating loop rises to the same value (0.25), loop flow in the idle loop is zero or slightly reversed (0.5), and flow through the reactor increases to slightly less than twice its original value (0.5).

Ref: General Physics HTFF Fundamentals Sect III, Pt. B, Chap 1

1.06 (2.25)

(a) State three conditions which indicate that natural circulation of the reactor coolant system is occurring (assume no forced circulation). (1.75)

(b) If a DNB ratio of 1.00 exists in the core, what is the probable effect, if any? (0.5)

END OF SECTION 1

(a) (1) RCS DELTA T < or = to Full Load DELTA T (0.5)
(2) RCS or CET temperatures constant or decreasing (0.5)
(3) Steam generator pressures constant or decreasing at a rate equivalent to the rate of decrease of RCS temperatures (0.5) while maintaining steam generator level constant with continuous auxiliary feedwater (0.25).

(b) Failure of some fuel cladding (0.5)

Ref: TS 2.1,, Westinghouse Mitigating Core Damage Training Manual, p. 51

SECTION 2

REACTOR OPERATORS EXAMINATION

Plant Design - Including Safety and Emergency Systems

2.01 (2.0)

For the Reactor Coolant Pump Seal Water Circuit:

(a) What is the primary function of the Reactor Coolant Pump Seal Water Circuit in conjunction with the pump seals it provides? (1.0)

(b) In the event of a failure (loss of seal water flow) how will seal water be supplied provided the loss of flow is not from a ruptured line? (1.0)

2.01 Answer:

(a) To effectively prevent the leakage of reactor coolant along the Reactor Coolant Pump shaft. (1.0)

(b) The Test pump in the CVCS is sized to supply adequate flow to all three RCP's. (1.0)

Reference: Study Guide No. 4, pages 13, 14, 17, and 24.

Will also accept leak off past thermal barrier to provide seal.

2.02 (2.0)

The pressurizer has two code sa.eties and two power operated relief valves.

(a) During operation while at power it is determined that there is leakage coming from a relief valve on the pressurizer. How can an operator determine which specific valve is leaking among the four valves? (1.0)

(b) If the leakage was determined to be past a power operated relief valve, and exceeded the Technical Specification limits, would the plant be required to be shutdown? Explain. (1.0)

2.02 Answer:

(a) Each safety valve has its own temperature element, an indication of higher than ambient temperature would allow the operator to determine the valve. (0.5)

The case for the power operated reliefs is different, they have a common temperature element. However they have individual isolation valves, which can be closed to determine which valve is leaking. (0.5)

(b) No (0.5), The power relief valves have block valves and can be isolated so that operation can continue (0.5).

Reference: Requalification Exam No. 0831, page 7.

2.03 (2.0)

Regarding the Pressurizer:

(a) Give three reasons why a 1 gpm flow rate is maintained around the spray valves into the Pressurizer. (1.5)

(b) The Pressurizer level program has a ramp function from 25% at no load to 37% at full load. Why is pressurizer level required to vary as a function of power? (0.5)

2.03 Answer:

(a) 1) To reduce thermal stresses and thermal shock when the spray valves open. (0.5)

2) To help maintain uniform water chemistry (boron concentration). (0.5)

3) And to maintain the desired temperature in the surge line. (0.5)

(b) To allow for surges into and out of the Pressurizer during load transients. (0.5)

Reference: Study Guide No. 16.

2.04 (3.0)

For the Diesel Generators:

(a) List six trips for the Diesel Generator <u>engines</u> that are in service for a manual start. (3.0)

2.04 Answer:

(a) Any 6 (0.5) each.

1) Engine Overspeed.
2) Generator Differential.
3) High Crankcase Pressure.
4) Low-low Engine Oil Pressure.
5) Low-low Turbo Oil Pressure.
6) High-high Jacket Water Temperature.
7) High Vibration Engine or Turbo.
8) High-high Lube Oil Temperature.
9) High Main Bearing Temperature.

Reference: Study Guide No. 98, page 9.

2.05 (1.0)

For the Reactor Coolant Pump No. 1 Seals:

(a) An operator is about to start a Reactor Coolant Pump, system pressure is 375 psig. An annunciator window is illuminated "No. 1 SEAL LO FLOW", can the pump be started? Explain. (1.0)

2.05 Answer:

(a) Yes (0.5). At pressures below 1000 psig the No. 1 seal flow may be too low to clear the alarm (a flow of at least 0.3 gpm is considered adequate for the condition) (0.5).

Reference: Study Guide No. 3, page 12.

2.06 (2.0)

For the Containment Spray System and the Containment Spray Actuation System (CSAS):

(a) What is the sequence (order) of events when an actuation signal (CSAS) comes in to start Containment Spray? (1.0)

(b) What setpoints and logic must be satisfied to initiate an actuation of Containment Spray from the CSAS (automatic only)? (1.0)

2.06 Answer:

(a) Refueling Water Pumps start first. (0.25)

The Spray Header valves open. (0.25)

The Hydrazine injection pump stars. (0.25)

The Hydrazine valves open to admit hydrazine to the Containment Spray header. (0.25)

(b) 2/2 Safety Injection (1/1 sequencer per train) (1.0)
and
2/3 High Sphere pressure - 10psig.

Reference: Study Guides No.s 16 and 17.

2.07 (3.0)

Concerning the Safeguards Load Sequencing System (SLSS):

(a) The SLSS recieves six signals from various sources including one from a test switch. What are the remaining five signals? (1.5)

(b) What actions must an operator take to <u>terminate</u> and <u>reset</u> an Automatic actuation of Safety Injection? (1.5)

2.07 Answer:

(a) 1) Pressurizer Pressure (0.3)

2) Containment Pressure (0.3)

3) Undervoltage signal from both 4160 busses (1C and 2C). (0.3)

4) Safety Injection Block (0.3)

5) Standby Diesel Generation (Voltage frequency signal and circuit breaker status signal.) (0.3)

(b) First he must ~~block~~ reset the automatic actuation signal by depressing the reset switches on the sequencer surveillance panel (0.5). and then <u>in order</u> turning off the feedwater pumps (0.5) and the safety injection pumps (0.5).

Reference: Study Guide No. 17 pages 11, 12, and 15.

END OF SECTION 2

SECTION 3

REACTOR OPERATORS EXAMINATION

Instrumentation and Controls

3.01 (2.0)

Regarding the steam generator level control system:

(a) The steam generator level control system is commonaly referred to as a three element control system, what are those three elements? (1.5)

(b) If there are three elements used for control in this system, why then is steam pressure (not one of the three 'elements') also an input to the system? (0.5)

3.01 Answer

(a) 1. Feedwater flow. (0.5)

2. Steam flow. (0.5)

3. Steam generator level. (0.5)

(b) To provide input for density compensation in the Steam Flow Computer. (0.5)

Reference: OT-1045 and OT-1064 "Steam Generator Water Level Control", Study Guide No. 62.

3.02 (1.5)

For the diagram (figure 3.1) on the following page describe:

(a) Which side of the manometer will rise when a fluid is flowing in the direction shown? (0.5)

(b) What equation is used to determine the flow rate in the pipe? (Noun name or description of relationship.) (0.5)

(c) What other type of instrumentation is normally provided in the plant to perform this function (of the manometer)? (0.5)

3.02 Answer

(a) The leg connected to the restriction of the venturi. (0.5)

(b) The Bernoulli equation (Flow rate is proportional to the square root of the differential pressure). (0.5)

(c) Typically a Bourdon tube type transmitter (D/P cell). (0.5)

Reference: General Physics "Heat Transfer Thermodynamics and Fluid Flow Fundamentals"

Venturi

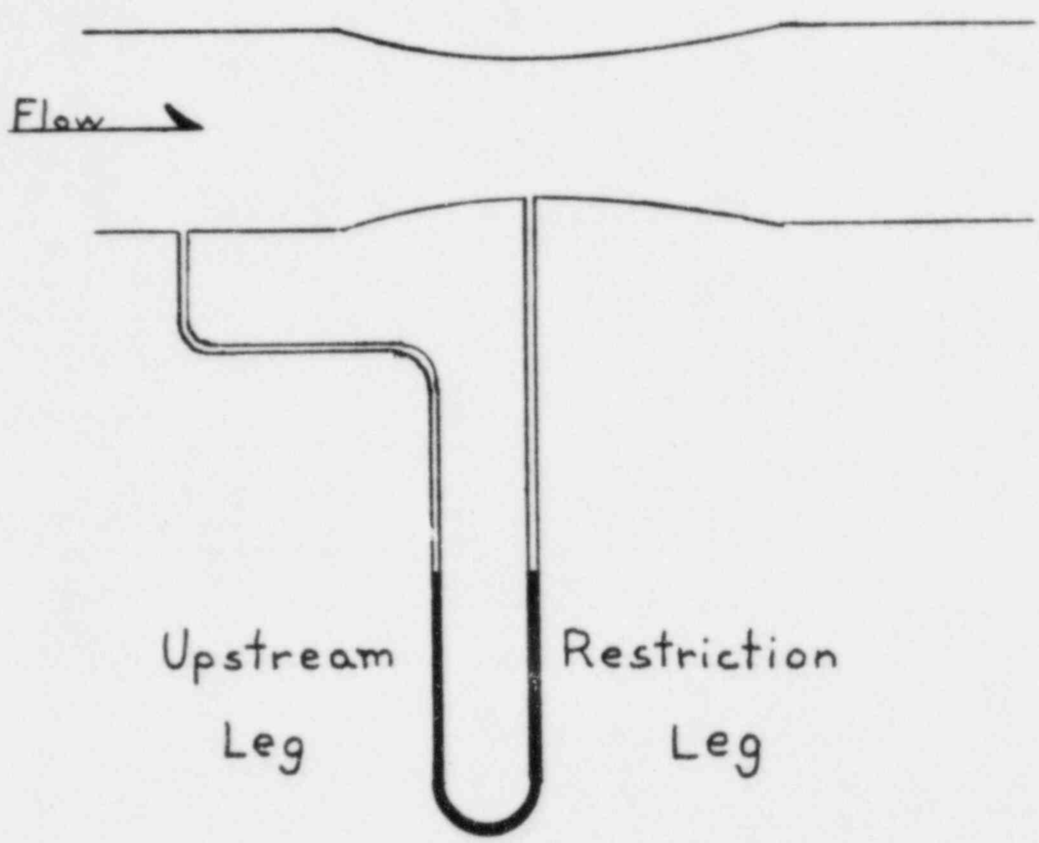


Figure 3.1

3.03 (3.0)

For the curve on the following page (Figure 3.2) identify the following parts of the curve:

(a) I (0.5)

(b) II (0.5)

(c) III (0.5)

(d) IV (0.5)

(e) V (0.5)

(f) VI (0.5)

3.03 Answer

(a) Recombination (0.5)

(b) Ionization (0.5)

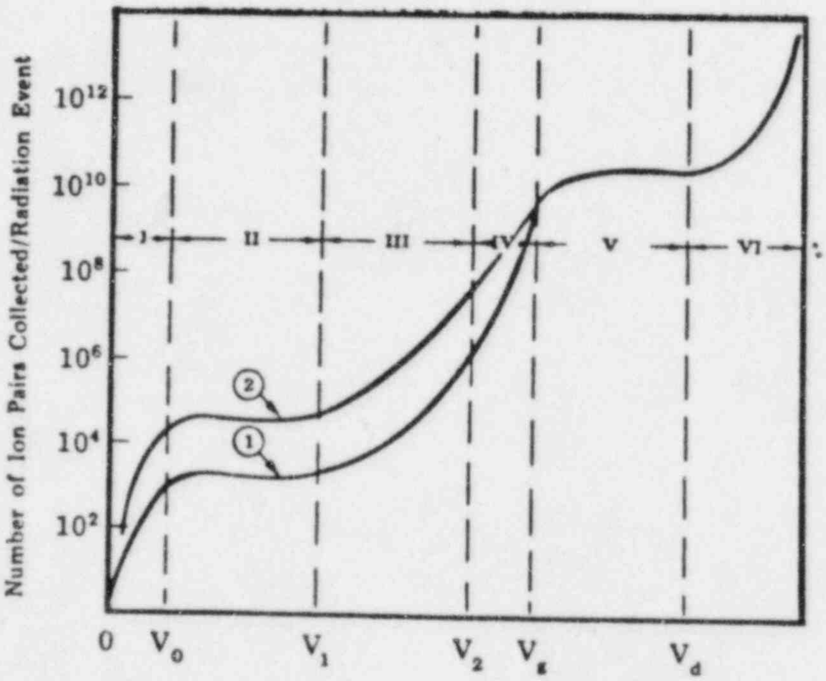
(c) Proportional (0.5)

(d) Limited Proportional (0.5)

(e) Geiger-Mueller (0.5)

(f) Continous Discharge (0.5)

Reference: General Physics Volume II, Chapter 3, Section E



Curve 1: Radiation event of lower specific ionization.
Curve 2: Radiation event of higher specific ionization.

Figure 3.2

3.04 (2.5)

During solid plant operations the following alarms come in:

"OMS Hi Pressure"

"Pressure Transient in Progress"

(a) What condition would actuate these alarms (i.e. System setpoints)? (0.5)

(b) What function does the Overpressure Mitigation System serve? (0.5)

(c) After securing charging flow as the first immediate action taken, what valves must an operator check to insure are open to complete the immediate actions in response to these alarms? (1.5)

3.04 Answer:

(a) The "OMS Hi Pressure" alarm comes in at 480 psig. (.25) ~~(0.5)~~

"Pressure Transient in Progress" at 500 psig. (.25) ~~(0.5)~~

(b) To protectect the RCS from potential overpressurization during solid operations. (.5)

(c) Insure LCV-1112, letdown isolation, open. (0.5)

PCV-1105, letdown control valve open. (0.5)

And, CV-525 and CV-526, containment isolation valves, are open. (0.5)

Reference: Line Diagrams, SO1-2.1-11 "Overpressurization Mitigation System Actuation", and Study Guide No. 6

3.05 (1.5)

For the Main Turbine:

(a) Besides the Overspeed trip that is provided what are three other trips associated with the Turbine. (1.5)

3.05 Answer:

(a) Any three (0.5) each.

Low bearing oil pressure.
Solenoid Trip.
Thrust bearing trip.
Low vacuum trip.

Reference: Study Guide 51, pages 29 and 30.

3.06 (3.0)

For the Control Rod Drive Summing Computer:

(a) The computer receives inputs from 5 sources, name 4 of those sources. (2.0)

(b) The output from the computer goes to two places, what are those places and what function does the signal that is provided perform? (1.0)

3.06 Answer:

(a) Any 4 (0.5) each:

1) Average Tavg.

2) Tref.

3) Primary pressure.

4) Nuclear flux.

5) Feedback from Bistable (TC-413B) indicating demanded rod postion.

Rod direction control

(b) 1) (Bistable 413B) (0.25), to move rods in or out (0.25).

2) Rod speed controller (0.25). Which produces a demanded speed to be fed to the rod drive system (through an auto/manual group selector switch) (0.25).

Reference: Study Guide 12, page 4.

3.07 (1.5)

For the following Operational Radiation Monitoring System channels what type of detector is used, and what control funtions do they provide? (Do not provide alarms.)

(a) Channel R-1211 - continuous air sample from either the vapor container or the plant stack. (0.5)

(b) Channel R-1214 Main Stack Gas Monitor. (0.5)

(c) Channel R-1216 Steam Generator Blowdown-Liquid Sample monitor. (0.5)

3.07 Answer:

(a) Scintillation detector. (0.25)

No direct control function. (0.25)

(b) Geiger-Mueller detectors. (0.25)

No control function

~~High level alarm deenergizes SV-99 which closes to stop all gas discharge form gas decay tanks.~~ (0.25)

(c) Scintillation detector. (0.25)

High radiation level alarm (energizes solenoid valve (SV-84)), causing CV-100, 100A, and 100B to close. This stops discharge to blowdown tank and to circulating water outfall. (Candidate only has to indicate that discharges to blowdown tank and outfall are isolated.) (0.25)

Reference: Study Guide 31; pages 2, 4, 8, 9, 11, and 13.

End of Section 3

SECTION FOUR

REACTOR OPERATOR EXAMINATION

4.01 (2.5)

(a) Under nonemergency conditions, give three general types of activities for which procedures are required to be used. (An example would be an evolution affecting plant reliability). (1.5)

(b) It is desired to perform minor troubleshooting of a pressure controller. One valve will be shut to perform the troubleshooting, and then it will be reopened. No in-place procedure is required for this activity. May this activity be performed without equipment control? Explain. (1.0)

(a) Any three of the following (0.5 each, max. of 1.5): manipulation of all safety related equipment, major (complex) evolutions, infrequently performed evolutions, manipulations of major (nonsafety) equipment, or evolutions affecting plant plant safety.

(b) In general, no, since equipment control is required for most evolutions not controlled by procedures. (0.5). ~~However,~~ The Shift Superintendent may not approve a minor maintenance item, ~~not important to safety, without equipment control.~~ (0.5) for a pressure controller.

Ref: SO!-14 -42, SO!-14-12 pp. 8-10

4.02 (3.0)

(a) List the four highest priority critical safety functions. (2.0)

(b) Explain how red and orange conditions in critical safety function trees require different priority sequences. (1.0)

integrity

(a) Maintenance or control of subcriticality, core cooling, reactor coolant ~~inventory~~, and heat sink (0.5 pts. each, correct order not required)

(b) Both require departure from any other EOI in use, but a red condition requires immediate implementation of the EOI (0.5), whereas for an orange condition it is required to complete the current pass through all of the status trees (to look for higher priority red conditions) (0.5).

Ref: SO1-1.0-1 pp. 11-14

4.03 (2.0)

According to the Emergency Operating Instructions, what events should the following symptoms correspond to?

(a) Reactor Trip and a Steam/Feed Mismatch Reactor Trip Alarm ON (0.5)

(b) Reactor Trip, Safety Injection, Electrical Power Available, Emergency Systems Operating as Required, and Steam Generator 'A' level is rising uncontrolled. (0.5)

(c) Six core exit thermocouples indicating 700 F (0.5)

(d) Same as (b) except steam generator levels are under control and containment radiation monitor R1255 is above its alarm setpoint. (0.5)

(a) Loss of Secondary Coolant

(b) Steam Generator Tube Rupture

(c) Potential Loss of Core Cooling

(d) Loss of Reactor Coolant

- consideration will be made for other events which correspond to symptoms.

Ref: S01-1.0-30, S01-1.0-40, S01-1.2-2, S01-1.0-20

4.04 (2.5)

(a) Upon confirmed loss of which vital or utility bus (or buses) is an immediate reactor trip NOT required? (0.9)

(b) Which AC Distribution buses are required to be OPERABLE by the Technical Specifications, assuming the plant is in Mode 1, and NOT in any Action statements of the Specifications? (1.1)

(c) What equipment is required other than fuel tanks by the Technical Specifications for the emergency diesel generators. (0.5)

(a) Vital buses 3A, 5 and 6 (0.3 each)

(b) 4160 Buses 1C and 2C, 480 Buses 1, 2, and 3, and Vital Buses 1, 2, 3, 3A, 4, 5, and 6. (0.1 pts each)

(c) Fuel transfer pumps

Ref: SO1-2.6-3, TS 3.7

4.05 (3.0)

The plant is being started up from hot standby to minimum load.

(a) What rod positions are procedurally required for criticality unless otherwise directed by the Shift Superintendent? (1.0)

(b) When, according to the procedure, should criticality be anticipated? (1.0)

(c) What is the limit on Start Up Rate during the startup? (0.5)

(d) What is the limit on turbine backpressure prior to rolling the turbine? (0.5)

(a) Shutdown Banks 1 and 2 and Control Bank 1 at 318 steps, Control Bank 2 at 100 steps (0.25 pts each).

(b) At any time when control rods are being withdrawn, or when boron dilution is in progress (0.5 pts each)

(c) 1.0 dpm

(d) 5.5" Hg

Ref: SO1-3-2

4.06 (2.0)

(a) Define a high radiation area. (1.0)

(b) What are the SONGS quarterly Administrative Limits for radiation exposure? (1.0)

END OF SECTION 4

(a) A high radiation area is any area which is accessible (0.1) in which a major portion of a person's body (0.1) could receiv in excess of 100 mrem in one hour (0.8)

(b) 900 mrem whole body (0.34), 3750 mrem skin (0.33), and 4700 mrem extremities (0.33)

Ref: SONGS Radiation Training Handout