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

EXAMINATION REPORT NO. 87-28(OL)

FACILITY DOCKET NO. 50-423

FACILITY LICENSE NO. NPF-49

LICENSEE: Northeast Nuclear Energy Company
P.O. Box 270
Hartford, Connecticut 06141-0270

FACILITY: Millstone Unit 3

EXAMINATION DATES: December 14-18, 1987

CHIEF EXAMINER: Noel F Dudley FOR 3-16-88
Barry S. Norris Date
Senior Operations Engineer

APPROVED BY: P W Eselgroth 3-16-88
Peter W. Eselgroth, Chief PWR Section Date
Operations Branch, Division of Reactor Safety

SUMMARY: Written and operating examinations were administered to four Senior Reactor Operator (SRO) candidates and three Reactor Operator (RO) candidates. Three SRO candidates and two RO candidates received their licenses. One SRO candidate failed both the written and operating examinations, and one RO candidate failed the operating examination.

REPORT DETAILS

TYPE OF EXAMINATIONS: Replacement

EXAMINATION RESULTS:

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

CHIEF EXAMINER AT SITE: B. S. Norris (USNRC)

OTHER EXAMINERS:
E. Yachimiak (USNRC)
R. M. Keller (USNRC)
L. E. Briggs (USNRC)
P. H. Bissett (USNRC)
N. C. Jensen (EG&G)
P. T. Isaksen (EG&G)

1.0 Summary of Generic Strengths and Deficiencies on the Operating Exams

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

1.1 Strengths

a. Communications between the members in one of the groups of candidates that were examined at the simulator were clear and precise. Orders were clear and direct and responses by the operators were of equal clarity.

b. Supervisory skills, as demonstrated by one group of candidates during the simulator portion of the examination, were effectively used when directing the operators during abnormal and emergency plant conditions. Time was taken to summarize the plant's status so that each operating crew member understood his responsibilities during the ensuing recovery efforts.

emergency plant conditions. Time was taken to summarize the plant's status so that each operating crew member understood his responsibilities during the ensuing recovery efforts.

1.2 Deficiencies

a. In contrast to paragraph 1.1.b, communications between the members in another group of candidates on the simulator were poor. Plant conditions and parameters were often described without adequate clarification. Requests for information were often non-specific and responses to these requests equally vague.

b. The candidates were unfamiliar with the expected plant response in the simulator while implementing section 5.4 of OP 3204, "Reduced Temperature Return to Power."

c. Som SRO candidates had difficulty verifying, via the control roo logs, that scheduled surveillance tests were completed.

d. Some RO candidates had difficulty performing a plant calorimetric by hand.

2.0 Summary of Generic Strengths and Deficiencies on the SRO Written Exams

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

2.1 Strengths

a. Overall knowledge of plant systems design, control, and instrumentation, with the exception of paragraphs 2.2.c and 2.2.d below.

b. Overall knowledge of procedures - normal, abnormal, emergency, and radiological control.

2.2 Deficiencies

a. The effects of xenon oscillations on primary plant conditions.

b. The effects of a dropped rod on primary and secondary plant conditions.

c. The basis for WHY the auto start signals are different for the auxiliary feedwater pumps.

d. HOW the arming of the cold overpressure protection system (COPS) affects the PORV block valves.

3.0 Summary of Generic Strengths and Deficiencies on the RO Written Exams

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

3.1 Strengths

Overall knowledge of plant system features in both safety and emergency systems.

3.2 Deficiencies

a. The effects of core age on the Doppler Only Power coefficient.

b. HOW to locally start the diesel using the manual start lever on the air start valves.

c. HOW the arming of COPS effects the pressurizer block valves and power operated relief valves (PORV) and PORV block valves.

d. Radiological exposure control procedures.

4.0 Summary of Simulator Discrepencies

During the conduct of the simulator portion of the operating exams, the following human factors discrepancies were observed:

a. The Critical Safety Function Status Tree computer displays for Subcriticality and Heat Sink do not always identify the correct path to the operators. For Subcriticality, a red path is continously displayed when at power. For Heat Sink, the path displayed is not in agreement with the procedurally derived path.

b. Black and White tape is in use on the control room benchboards to identify electrical bus configurations. This has not been done at the simulator.

c. White masking tape has been placed on the handles of the switches for buses 32B and 32N (power supplies for the rod drive MG sets) in the simulator. The tape has been removed in the control room.

d. Barrel switch guards are in use around various pushbuttons in the control room. This has not been accomplished in the simulator.

5.0 Personnel Present at the Exit Meeting

5.1 NRC Personnel
B. S. Norris - Chief Examiner

5.2 Facility Personnel
J. M. Black - Director, NTD
C. H. Clement - Superintendent, MP3
J. S. Harris - Operations Supervisor, MP3
R. F. Martin - ATS, Simulator Training
M. J. Moehlman - ATS, Operator Training
R. G. Stotts - Supervisor, Operator Training

6.0 Summary of Comments Made at the Exit Meeting

6.1 The NRC discussed the generic deficiencies noted on the operating examinations, see paragraph 1.0 for details.

6.2 The NRC noted the discrepancies between the simulator and the control room as discussed in paragraph 4.0.

6.3 The NRC noted that the posted telephone numbers for contacting the NRC are not consistent with those listed in EPIP-4112.

The facility stated that they would determine which telephone numbers were correct and would ensure that the procedure and the posting were consistent.

7.0 Review of Exam

The written examinations were reviewed by the utility and discussed with the examiners after all candidates had completed the examination on December 14, 1987. The facility's comments (Attachment 3) and the NRC resolution (Attachment 4) are enclosed.

Attachments:
1. RO Written Examination and Answer Key
2. SRO Written Examination and Answer Key
3. Facility Comments on Written Examinations
4. NRC Response to Facility Comments

ATTACHMENT 4

NRC RESPONSE TO FACILITY COMMENTS

The following statements address the NRC's resolution of comments on the written examinations submitted by the facility (see Attachment 3) and changes made to the answer key during the grading process.

REACTOR OPERATOR EXAMINATION

1.01.b Comment accepted, answer key modified.

1.05.c Comment accepted, answer key modified.

1.06.b Comment accepted, answer key modified.

1.07.c Comment accepted, answer key modified to accept "cold leg temperature decreases" as an alternate answer.

1.09(2) Comment not accepted. The question was intended to test the knowledge of the magnitude of the change of equilibrium Xenon levels. The candidates' responses will be evaluated on a case-by-case basis.

2.01(3) Comment not accepted. It was not agreed that either TRUE or FALSE was an acceptable answer. Rather, it was determined to delete the question. Question point value was reduced by 0.50 points.

2.02.b Comment accepted, answer key modified to accept "Containment Instrument Air Compressors" as an alternate answer.

2.03.a Answer key changed to include an additional acceptable answer during the grading of the examinations.

2.07 Comment noted, answer key not changed. Consideration will be given to candidates' responses which are appropriately justified.

2.08.a Comment noted. The answer key and point values were not changed because the level of design basis knowledge required to answer the question was considered appropriate for the RO.

2.08.c Comment accepted, answer key changed to reflect "RCS overpressurization" as the correct answer.

2.09.c Comment accepted, answer key modified to accept "Diesel has lost control power" as an alternate answer.

3.03.b Comment accepted, the reference material provided for exam preparation was in error. The answer key was modified to delete the word HIGHEST.

3.05.c Comment noted. Consideration will be given to any properly explained response which assumes the steam dumps are in a Mode which is normal for the stated plant conditions.

3.06.c The answer key was revised during the grading process to allow another correct response.

3.07.a Answer key corrected during the grading process.

3.07.c Comment accepted, answer key modified.

3.08.c Comment noted, answer key modified and points redistributed.

3.09.b Comment accepted, answer key modified.

3.11 Comment accepted, answer key modified.

4.05.a Comment not accepted, nor was there an agreement at the examination review. Departmental Instruction OPS-3.07 pages 3 and 4 "Handwheel Closure of Motor Operated Valves" stresses the requirement that Cheater Bars not be used.

4.07.a The answer key was revised and points were redistributed during grading.

4.08.a The answer key was revised and points were redistributed to include adverse containment values that were not included in originally supplied plant reference material.

4.09.a Comment accepted, answer key modified. The reference material provided for examination preparation was in error.

4.09.b Comment not accepted, nor was there an agreement at the examination review. Subsequent information supplied by the facility states that the 233 gpm flow limit is "... to ensure adequate NPSH to the charging pump..." The answer key was revised to indicate this reason.

4.10 The answer key revised to delete "at § 350 F" as a required part of the answer.

SENIOR REACTOR OPERATOR EXAMINATION

5.02.b See the response for question 1.06.b.

5.04 Comment noted. The answer key was modified to incorporate the changes made to the reference data during the administration of the exam. The change to the reference data, however, did not make it necessary for the candidates to have a Samarium curve. The candidates were asked for the reference value at 100% equilibrium power, not the time dependent value.

5.05 Comments noted. The answer key for part a.2 was changed to allow a candidate to provide a written response of power increasing and then stabilizing at a new, higher value. The answer key for part b.3 was changed to C. instead of B. The answer key for part b.2 was changed to G. instead of H. In addition, the following additional answers were changed: Parts a.1 and a.4 were changed to add F. as an alternate acceptable answer due to similarities between the two choices. Parts c.1 and c.2 were changed to "H. or D." for the same reason.

5.07.d Comment not accepted. A condenser circulating water pump does display the characteristics of a centrifugal pump when starting. If the discharge valve were to be open during the pump start, the starting current would be higher because of the low resistance to flow within the discharge piping. This can be shown to be true by referencing the circulating pump start procedure and the pump's head versus flow curve.

5.09.b Comment accepted, answer key modified.

6.01.c See the response for question 2.08.c.

6.03.b See the response for question 3.09.b.

6.08.d Comment accepted, answer key modified.

6.10 See the response for question 3.11.

7.01.d Comment not accepted. The caution is applicable throughout the entire procedure and should be understood prior to performing any procedural actions.

7.03.c Comment not accepted. The reference material supported the answer key response.

7.05.a See the response for question 4.09.a.

7.05.b See the response for question 4.09.b.

7.06.c Comment accepted, answer key changed.

7.07.c Comment not accepted. The facility reference material supported the answer key response.

8.01.d Comment accepted. The answer key was modified to accept "if equipment is deemed inoperable" as an alternate answer.

8.05.d Comment accepted, answer key modified.

8.06.a Comment accepted. Any superintendent will be accepted as an alternate response for duty officer.

8.07 Comments noted. The answer key responses were modified to allow reasonable candidate responses, as long as justification was provided. In addition, higher classifications were deemed acceptable with appropriate candidate justification.

8.08.b Comment not accepted, based on the assumptions stated within the question.

8.08.c Comment accepted, answer key changed.

8.08.d Comment accepted, answer key changed.

8.09 Comments noted. The answer key for 8.09.b was changed, the answer key for 8.09.c was not changed because both parts of the answer are needed to fully explain why the limit exists.

ATTACHMENT 1

U. S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR LICENSE EXAMINATION

FACILITY: MILLSTONE 3

REACTOR TYPE: PWR-WEC4

DATE ADMINISTERED: 87/12/15

EXAMINER: BRIGGS/BISSETT

CANDIDATE: Master

INSTRUCTIONS TO CANDIDATE:

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

CATEGORY VALUE	% OF TOTAL	CANDIDATE'S SCORE	% OF CATEGORY VALUE		CATEGORY
25.00	25.00	________	________	1.	PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW
24.50 * ~~25.00~~	25.00	________	________	2.	PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS
25.00	25.00	________	________	3.	INSTRUMENTS AND CONTROLS
25.00	25.00	________	________	4.	PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
99.50 ~~00.00~~		________ Final Grade	________%		Totals

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

____
Candidate's Signature

* Question 2.01 part 3 was deleted. - 0.5 points JRB

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

1. Cheating on the examination means an automatic denial of your application and could result in more severe penalties.
2. Restroom trips are to be limited and only one candidate at a time may leave. You must avoid all contacts with anyone outside the examination room to avoid even the appearance or possibility of cheating.
3. Use black ink or dark pencil only to facilitate legible reproductions.
4. Print your name in the blank provided on the cover sheet of the examination.
5. Fill in the date on the cover sheet of the examination (if necessary).
6. Use only the paper provided for answers.
7. Print your name in the upper right-hand corner of the first page of each section of the answer sheet.
8. Consecutively number each answer sheet, write "End of Category __" as appropriate, start each category on a new page, write only on one side of the paper, and write "Last Page" on the last answer sheet.
9. Number each answer as to category and number, for example, 1.4, 6.3.
10. Skip at least three lines between each answer.
11. Separate answer sheets from pad and place finished answer sheets face down on your desk or table.
12. Use abbreviations only if they are commonly used in facility literature.
13. The point value for each question is indicated in parentheses after the question and can be used as a guide for the depth of answer required.
14. Show all calculations, methods, or assumptions used to obtain an answer to mathematical problems whether indicated in the question or not.
15. Partial credit may be given. Therefore, ANSWER ALL PARTS OF THE QUESTION AND DO NOT LEAVE ANY ANSWER BLANK.
16. If parts of the examination are not clear as to intent, ask questions of the examiner only.
17. You must sign the statement on the cover sheet that indicates that the work is your own and you have not received or been given assistance in completing the examination. This must be done after the examination has been completed.

18. When you complete your examination, you shall:

   a. Assemble your examination as follows:

      (1) Exam questions on top.

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

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

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

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

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

QUESTION 1.01 (2.50)

a. State whether the parameters listed below would INCREASE, DECREASE or STAY THE SAME as a result of a normal power change from 75% load to 85% load. Consider each parameter separately and assume the control rod system is in AUTOMATIC. Assume no operator action with the exception of increasing turbine load.

1. Tavg

2. RCS Delta T

3. Reactor Power

4. Fuel Temperature

5. Shutdown Margin

b. State whether the parameters listed below would INCREASE, DECREASE or STAY THE SAME as a result of a normal power change from 75% load to 85% load. Consider each parameter separately and assume the control rod system is in MANUAL. Assume no operator action with the exception of the increase in turbine load.

1. Tavg

2. RCS Delta T

3. Reactor Power

4. Fuel Temperature

5. Shutdown Margin

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.02 (2.00)

Attachment 1.1 shows characteristic curves for two centrifugal pumps. Each pump is equipped with a check valve on its discharge. Using Attachment 1.1 determine the system pressure and flow rate for:

a. Two pumps operating in parallel.

b. Two pumps operating in series.

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.03 (2.00)

a. What is the subcooling margin of the plant if the following conditions exist: SHOW ALL WORK (1.50)

Thot = 617 F Tcold = 557 F
Ppzr = 2235 psig Psg = 990 psig

b. If power is lowered from 100% to 50%, how will the subcooling margin change (increase, decrease, or remain the same)? (0.50)

QUESTION 1.04 (1.50)

Answer the following TRUE or FALSE:

a. A neutron population at equilibrium always means that the reactor is critical.

b. Delayed neutrons are less likely to escape resonance capture than prompt neutrons.

c. Delayed neutrons have a greater effect on reactor period after a negative reactivity addition than after a positive reactivity addition.

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.05 (2.75)

HOW (Increase, Decrease, No Change) and WHY will an INCREASE in each of the following affect RCCA (differential) worth? Include effects and results on neutrons and materials. Consider each case separately and state any assumptions made.

a. Moderator Density (1.00)

b. Boron Concentration (0.75)

c. Core Age (1.00)

QUESTION 1.06 (2.00)

HOW (More Negative, Less Negative, No Change) does the Doppler Only Power Coefficient change if the below parameters change as follows. JUSTIFY WHY.

a. Reactor power Increases from 50% to 100% power. (0.75)

b. Core age Increases from BOL to EOL. (1.25)

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.07 (1.50)

With all systems in manual and no operator action, WHAT EFFECT (increase, decrease, no change) and WHY will decreasing the circulating water temperature have INITIALLY on the following?

a. Condensate temperature
b. Steam generator pressure
c. Reactor power

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.08 (3.00)

a. How does DNBR change (increase, decrease, no change) as each of the following is increased? (Consider each separately) (2.00)

1. RCS pressure
2. RCS flow
3. Reactor power
4. Core inlet temperature

b. What causes Beta bar to change over core life? (1.00)

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.09 (3.25)

After operation at 50% for FOUR (4) days, power is increased to 100% using control rods only (i.e.,boron concentration is not changed).From the beginning of the transient until steady state conditions are reached, explain HOW and WHY Xenon concentration will change in terms of production and removal mechanisms. Include approximate times and relative magnitudes (50% power vs 100% power).

QUESTION 1.10 (1.50)

A reactor at BOL is critical at 10^{-9} amps. Rods are withdrawn at 30 steps per minute for 10 seconds in bank sequence.

What is the SUR IMMEDIATELY AFTER the rod motion stops? Assume differential rod worth is 20 pcm per step, effective delayed neutron fraction is 0.006 and the effective precursor decay constant is 0.1 sec-1.

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.11 (3.00)

Indicate whether the value of the following reactivity parameters will become MORE NEGATIVE, LESS NEGATIVE, or REMAIN THE SAME (no significant change) for their respective condition changes below. Consider each case separately. Briefly explain your answer.

a. MTC: Beginning of life to end of life.

b. Doppler only power coefficient: 1% to 100% power

c. Total power defect. Beginning of life to end of life.

(***** END OF CATEGORY 01 *****)

UES .ON 2.01 (1.50) ~~(2.00)~~

Indicate whether the following statements, which pertain to the plant air system, are TRUE or FALSE. (2.00)

1. The cold shutdown air compressors use TPCCW for cooling water supply?

2. The instrument air compressors use TPCCW for cooling water supply?

~~3. The containment instrument air compressors use RPCCW for cooling water supply?~~ Deleted.

4. The containment instrument air compressors normally supply air to the containment header ring?

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)

QUESTION 2.02 (2.75)

The reactor plant component cooling water (RPCCW) is comprised of three separate subheaders, two of which are safety-related. Answer the following questions in regard to these two subheaders.

a. Given below is a list of heat loads. State whether each is supplied by RPCCW Train A, RPCCW Train B, both trains, or neither train. (1.75)

1. Reactor coolant pump thermal barrier
2. Letdown heat exchanger(HX)
3. Diesel generator lube oil cooler
4. Residual Heat Removal(RHR) HX
5. Service air compressor
6. Seal water HX
7. Safety injection pumps cooling surge tank

b. What TWO RPCCW loads would be cooled in the event of a station blackout or CIA condition? (1.00)

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)

QUESTION 2.03 (2.00)

a. List FOUR (4) signals that will automatically trip the Motor driven main feedwater pump. (1.60)

b. Why is moisture separator reheat steam not used to drive the Turbine driven main feedwater pumps during startup? (0.40)

QUESTION 2.04 (1.50)

a. What is the purpose of the crossover line from the suction of the safety injection pump 'A' to the suction of the centrifugal charging pumps. (1.00)

b. Explain the purpose of the "power lockout" feature associated with valve MV 8813 (SI pumps common miniflow header isolation valve)? (0.50)

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)

QUESTION 2.05 (2.75)

a. Explain the Technical Specification Bases for the following requirements for the SI Accumulator Tanks when at 100% power. (2.00)

1. Power is removed from the accumulator tank isolation valve.
2. Minimum water volume.
3. Maximum water volume.
4. Maximum nitrogen pressure.

b. List 3 occurrences that could cause accumulator tank pressure to rise. (0.75)

QUESTION 2.06 (2.00)

Answer the following questions concerning the steam generator outlet nozzles.

a. What are the 3 reasons why it is desirable to limit steam line break flow rate by the use of the S/G outlet nozzles. (1.50)

b. What other basic function, besides those referenced in a. above, is provided for by the design of the steam generator outlet nozzle?(0.50)

QUESTION 2.07 (1.50)

A large break LOCA concurrent with a loss of off site power has occurred. The emergency diesel generator has started on the loss of power and a containment depressurization actuation signal has initiated. Approximately 60 seconds elapse before the Quench Spray pumps deliver water spray to containment. State THREE reasons why there is a time delay before the Quench Spray pumps are energized and deliver water spray to containment.

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)

deleted

QUESTION 2.08 (3.00)

a. The turbine auxiliary feedwater (TDAFW) pump has THREE (3) main steamline supply lines. TWO (2) are required to meet single failure crieria considerations. WHY is the third supply line required? (0.50)

b. WHAT are the THREE (3) sources of water for the AFW system in their ORDER OF PREFERRED USAGE? (1.00)

c. WHAT are the FOUR (4) signals which can automatically start the motor driven AFW pumps? (1.00)

d. WHY are the automatic start signals of the TDAFW pump different from the MDAFW pumps? (0.50)

ANSWER

a. Because one RCS loop can be isolated.[0.50]

b. Demineralized water storage tank. [0.25]
Condensate storage tank. [0.25]
Service water (Long Island Sound) [0.25]
(Listed in the right order) [0.25]

c. SIS [0.25]
LOP [0.25]
CDA [0.25]
Lo-Lo S/G level [0.25]

d. Reduce the possibility of S/G overfill [0.50]

see page 20a
for replacement question

QUESTION ~~6.01~~ 2.08 ~~(2.00)~~ (3.00)

a. For a large break LOCA, WHAT are the minimum number of emergency core cooling system pumps required to cover exposed fuel and limit possible core damage? (1.20) ~~(0.75)~~

b. Following SI reset, WHAT operator action(s) must be performed in order to reinstate automatic re-initiation of SI? ~~(0.55)~~ (0.80)

c. If TWO (2) charging pump are OPERABLE in MODE 4, WHAT RCS system safety limit can be violated if BOTH are operated? Include TWO (2) significant parameters. ~~(0.70)~~ (1.00)

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

QUESTION 2.09 (2.00)

a. WHY would increasing battery charger output voltage above 143 volts be a concern when charging a discharged battery?

b. WHAT prevents standby battery charger 301A-3 from being used to feed more than one 125 VDC bus?

c. If the plant is at 100% power and experiences a loss of 125 VDC bus 301A-1, WHAT component DIRECTLY prevents emergency diesel generator "A from starting, if required, from the control room?

d. Will a loss of any ONE (1) class 1E 125 VDC bus cause a DIRECT reactor trip? If so, briefly explain HOW? Assume the plant is at 100% power.

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)

QUESTION 2.10 (2.50)

a. Besides LIQUID monitors, there are THREE (3) basic types of process radiation monitor sampling channels at Millstone 3. WHAT ARE THEY? (1.00)

b. List FIVE (5) of the liquid process monitoring points. Also, state any automatic actions associated with them when high activity levels are detected. (1.50)

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)

QUESTION 2.11 (2.00)

Answer the following questions concerning charging pump recirculation flow.

a. What is the primary reason for having charging pump recirculation flow? (0.80)

b. Where is recirculation flow directed to during normal operation? (0.60)

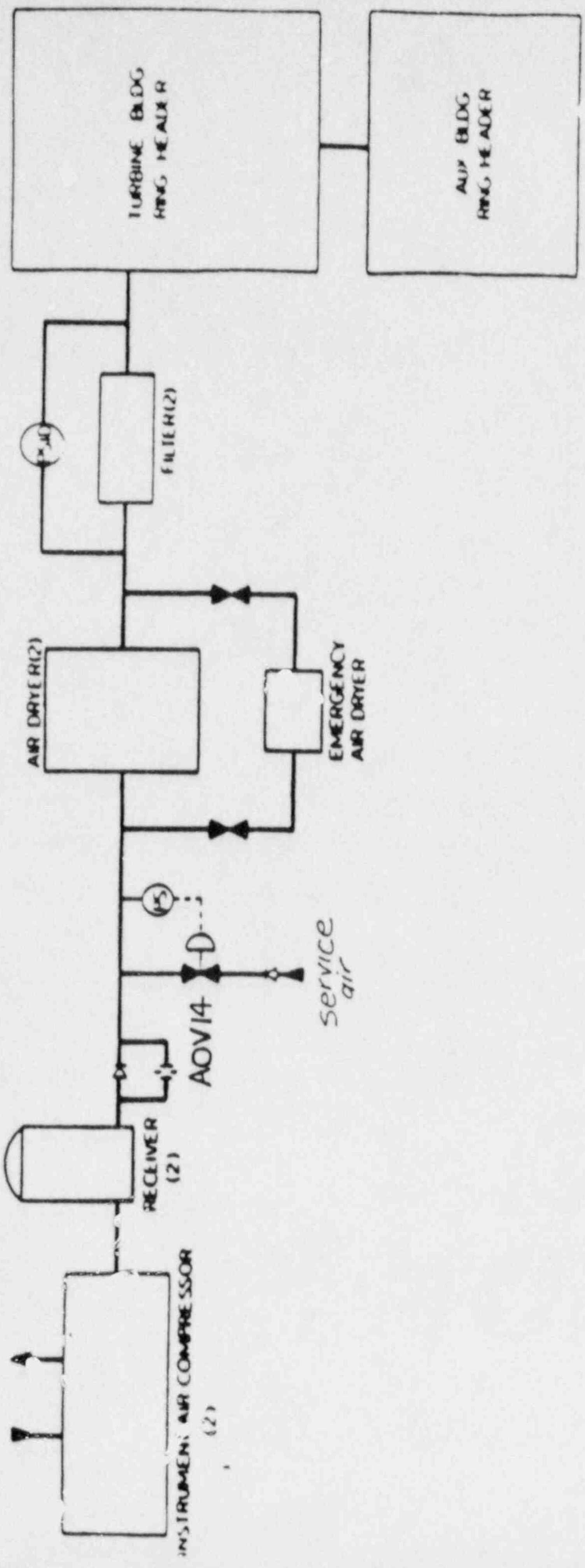
c. Where is recirculation flow directed to during a small break LOCA with RCS pressure at 2300 psia? (0.60)

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)

QUESTION 2.12 (1.00)

Using Attachment 2.1, STATE the purpose of valve AOV14. EXPLAIN WHY this valve does not tap directly into the turbine building ring header. (1.00)

(***** END OF CATEGORY 02 *****)



Attachment 2.1

QUESTION 3.01 (2.20)

The reactor is at 100% with all control systems in automatic. The controlling level instrument for number one steam generator fails high. WHAT happens to the actual level in the affected steam generator (increase, decrease or no change), WHY. EXPLAIN the transient to the first automatic protective action. Assume no operator action.

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.02 (2.70)

a. WHY is pressurizer level manually controlled during a plant cooldown. (1.00)

b. WHY is level controlled at 50 to 60% during a cooldown. (0.50)

c. HOW is pressurizer level program maintained during a power increase from no load to full load. (1.20)

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.03 (2.00)

The Reactor Vessel Level Monitoring System (RVLMS) consists of two safety related instrument channels with eight level detectors per channel.

a. WHAT type detectors are used and HOW do they detect level. (1.25)

b. What three (3) parameters are displayed on the control room CRT for the RVLMS SYSTEM. (0.75)

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.04 (.50)

TRUE or FALSE
When the last (No. 8) indicator on the RVLMS does not indicate any level in the reactor vessel, the core is definitely uncovered.

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.05 (3.00)

WHAT is steam dump response/action during the following plant evolutions. Identify the measured and reference parameters in each example. Steam dumps are aligned normally for plant conditions. Include setpoints where applicable. Consider each case separately. Assume no operator action.

a. Reactor trip from 20% power.

b. Turbine trip with plant at 45% power.

c. Plant at 559F, zero power.

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.06 (1.80)

For each of the below rod control interlocks, state:

1. The setpoint
2. Whether rod withdrawal is prevented in auto, manual or both
3. The reason for the rod block

a. C-1, Intermediate range neutron flux.

b. C-3, OT-Delta T

c. C-5, Turbine First Stage Pressure

QUESTION 3.07 (3.40)

Give the BASIS for each of the following Reactor Proctection and Safeguards Actuation System protective actions. INDICATE which of the actions can be blocked. Identify HOW and WHEN the block is accomplished. Include SETPOINTS where appropriate.

a. Source range High Flux

b. High pressurizer Pressure

c. OP Delta T

d. Pressurizer level High

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.08 (2.80)

Answer each of the following concerning the Emergency Diesel Generators.

a. When the Barring Device is installed can the diesel still be started. Explain your answer. (1.00)

b. After an Emergency start what three (3) automatic protective signals can stop the diesel generator. (0.60)

c. The solid state emergency generator load sequencer prevents overloading the diesel generator on a loss of power to the emergency bus. List automatic actions and equipment affected when this occurs. (1.20)

QUESTION 3.09 (2.20)

a. Briefly explain HOW neutrons produce current in a Source Range (SR) Excore Nuclear Instrumentation (NI) system detector. Include both nuclear and electrical reactions within the detector. (1.00)

b. A reactor shutdown is in progress with the SR detectors reading about 10,000 cps and both Intermediate Range (IR) detectors reading 1X10E-11 amps. Ten minutes later the SR detectors read about 1,000 cps but the IR detectors still read 1X10E-11 amps. WHY does the IR detector output remain the same? (0.60)

c. The plant is operating at 100% power with N44 out-of-service. If an automatic reactor trip occurs and N43 is failed as is, WHAT affect, if any, will this have on the NI system's ability to monitor neutron flux as the plant is stabilized in Mode 3. No action is taken. (0.60)

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.10 (2.40)

MATCH the ESF Status Panel group description from the right hand column with its respective Group number from the left hand column. Only ONE (1) description matches each Group number.

a. Group I
b. Group II
c. Group III
d. Group IV
e. Group V
f. Group VI

1. Consists of lights for those components whose status is changed during a CDA
2. Consists of lights covering the steam system
3. Consists of lights that only light during the injection phase
4. Most of these lights should always be off; however, some may illuminate during special or infrequent operation
5. Consists of lights for those components whose status only changes when in the cold leg recirc phase
6. Consists of lights that are normally off and will come on after an SIS
7. Consists of lights for those components whose status only changes for the hot leg recirc phase
8. Consists of lights for those components whose status is changed during the cold leg recirc phase and remains in the hot leg recirc phase

QUESTION 3.11 (2.00)

HOW WILL the Cold Overpressure Protection (COP) system respond if an operator were to arm both Trains while hot leg wide-range RTD TE-413A is failed LOW? Include in your answer WHY each Train is OR is not affected because of the RTD failure, and WHAT automatic actions take place. Assume the plant is shutdown at 300 F and 700 psig.

(***** END OF CATEGORY 03 *****)

QUESTION 4.01 (2.50)

Answer the following questions regarding "Loss of Instrument Air" procedure AOP 3562.

a. If instrument air pressure was decreasing, WHAT THREE (3) air compressors could be started in an attempt to restore pressure? (1.00)

b. Under WHAT TWO (2) conditions must the reactor be tripped? (1.00)

c. HOW would Tavg be controlled after a reactor trip due to a loss of instrument air? (0.50)

QUESTION 4.02 (2.00)

TRUE or FALSE?

Answer the following true or false questions relating to the use conventions associated with EOP's.

a. If a caution statement occurs before step one of an EOP it will apply only to the first step.

b. Unless otherwise specified, a required task need not be fully completed before proceeding to the next instruction; it is enough to begin the task and have some assurance that it is progressing satisfactorily.

c. Even after a transition to another procedure, the steps begun before the transition was made must still be completed.

d. Once a Functional Response Procedure (FRP) is entered due to a RED or ORANGE condition, that FRP must be completed, unless prompted by a higher priority condition.

QUESTION 4.03 (2.50)

List the FIVE conditions that support or indicate Natural Circulation (NC) Flow as stated in EOP 35 ES-0.1 Reactor Trip Response (step 9). Include parameters and trends.

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.04 (2.00)

List the six (6) Critical Safety Functions IN ORDER FROM HIGHEST TO LOWEST PRIORITY.

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.05 (2.50)

The following concern Department Instruction No. 3-OPS-3.07 on valve operation.

a. What are the FOUR requirements regarding manual seating (handwheel closure) of motor operated valves. (0.80)

b. How are manual valves VERIFIED OPEN during performance of a valve lineup. Include all required valve operations. (0.90)

c. WHAT action must be performad prior to plant cooldown for a backseated valve? WHY is this action necessary? (0.50)

d. WHO can authorize backseating a valve. (0.30)

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.06 (3.00)

A condition arises which requires entry into containment while critical at 40% power. The operator entering will receive a whole body dose of 40 mrem. Data is available on the following persons:

Candidate	1	2	3	4
Sex	male	male	female	male
Age	27	38	24	20
Qtr/exposure	270 mrem	970 mrem	475 mrem	900 mrem
Life exposure	-	54730 mrem	5200 mrem	9770 mrem
Remarks	history unavailable	-	4 months pregnant	-

Each candidate is technically competent and physically capable of performing the task. Emergency limits do not apply but time constraints do not permit obtaining authorization for an exposure limit increase. For each person indicate if that person could or could not be selected to perform the task based on exposure requirements only and justify your response. Use Millstone exposure limits. Include all requirements or limits that apply.

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.07 (2.00)

State the ACTION required by OP 3202-Reactor Startup, if the following conditions are observed during a Startup. Include times as appropriate.

a. Lowest loop Tave less than 551 F (0.75)

b. Unexpected increase in source range count rate (0.25)

c. Reactor critical below Rod Insertion Limit (0.50)

d. Reactor not critical at MAXIMUM rod position per ECP (0.50)

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.08 (3.00)

The plant has experienced a reactor trip and procedure EOP 35 E-0, REACTOR TRIP AND SAFETY INJECTION, has been entered. Per the EOP 35 E-0 FOLDOUT page answer the following, include appropriate setpoint numbers and adverse containment.

a. State the reactor coolant pump trip criteria. (1.00)

b. WHAT are the SI actuation criteria. (2.00)

QUESTION 4.09 (3.00)

Answer the following questions concerning procedure AOP-3566, "Immediate Boration"

a. WHAT are the FIVE (5) entry conditions for AOP-3566? (2.00)

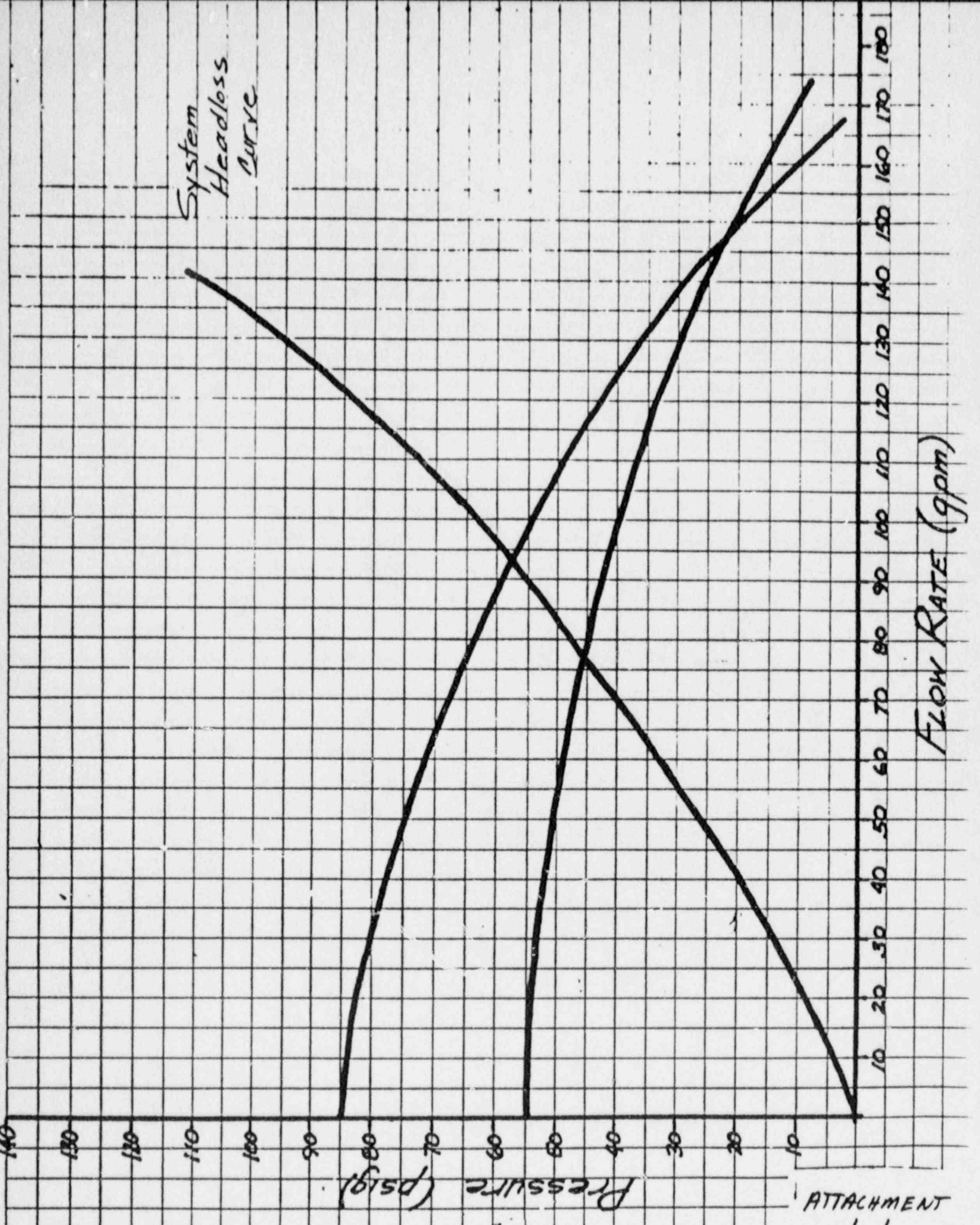
b. WHY is Immediate Boration charging flow required to be < 233 gpm? (0.50)

c. WHY is Immediate Boration charging flow required to be > 33 gpm? (0.50)

QUESTION 4.10 (2.50)

A plant heatup in preparation for startup is in progress. Plant temperature is 325 F with a heatup rate of 50 F/hr. When the operator attempts to open the "D" safety injection accumulator discharge valve it will not open. The Maintenance Supervisor informs the Shift Supervisor that the valve is mechanically damaged but repairs can be completed within 24 hours. The Shift Supervisor declares the "D" accumulator inoperable and tells the Reactor Operator to continue the heatup.
Explain WHY the heatup SHOULD or SHOULD NOT continue and WHAT ACTION, if any, is necessary.

(***** END OF CATEGORY 04 *****)
(************* END OF EXAMINATION ***************)



ATTACHMENT 1.1

$f = ma$

$v = s/t$

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

$w = mg$

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

$E = mc^2$

$KE = 1/2\ mv^2$

$a = (V_f - V_o)/t$

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

$PE = mgh$

$V_f = V_o + at$

$\omega = \theta/t$

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

$W = v\ \Delta P$

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

$\Delta E = 931\ \Delta m$

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

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

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

$Pwr = W_f \Delta h$

$I = I_o e^{-ux}$

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

$TVL = 1.3/u$

$HVL = -0.693/u$

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

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

$SUR = 26.06/T$

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

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

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

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

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

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

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

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

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

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

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

$\ell^* = 10^{-5}$ seconds

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

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

$I_1 d_1 = I_2 d_2$

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

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

$\Sigma = \sigma N$

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

$R/hr = 6\ CE/d^2\ (\text{feet})$

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.
1 ft. H_2O = 0.4335 $lbf/in.^2$

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)
1 BTU = 778 ft-lbf

ANSWER 4.09 (3.00)

Ey

a. - control bank height below the low-low limit
- failure of ~~one~~ TWO or more control rods to fully insert following a reactor trip of shutdown
- uncontrolled cooldown of the RCS following a reactor trip of shutdown
- uncontrolled or unexplained reactivity addition (indicated by abnormal control bank insertion, increasing Tavg, or increasing nuclear power)
- failure of the reactor makeup control system

[5 X 0.40]

b. ~~limit cold water addition (CAF)~~ [0.50]
ensure adequate NPSH to charging pumps. Ey

c. provide a minimum rate of negative reactivity insertion (CAF) [0.50]

REFERENCE
Millstone 3 EOs A66-02-C-001,002,004
Millstone 3 AOP-3566 pages 2,3
K/A 000024 EK3.01 4.1
K/A 000024 EK3.02 4.2
000024K302 000024K301 ...(KA'S)

ANSWER 4.10 (2.50)

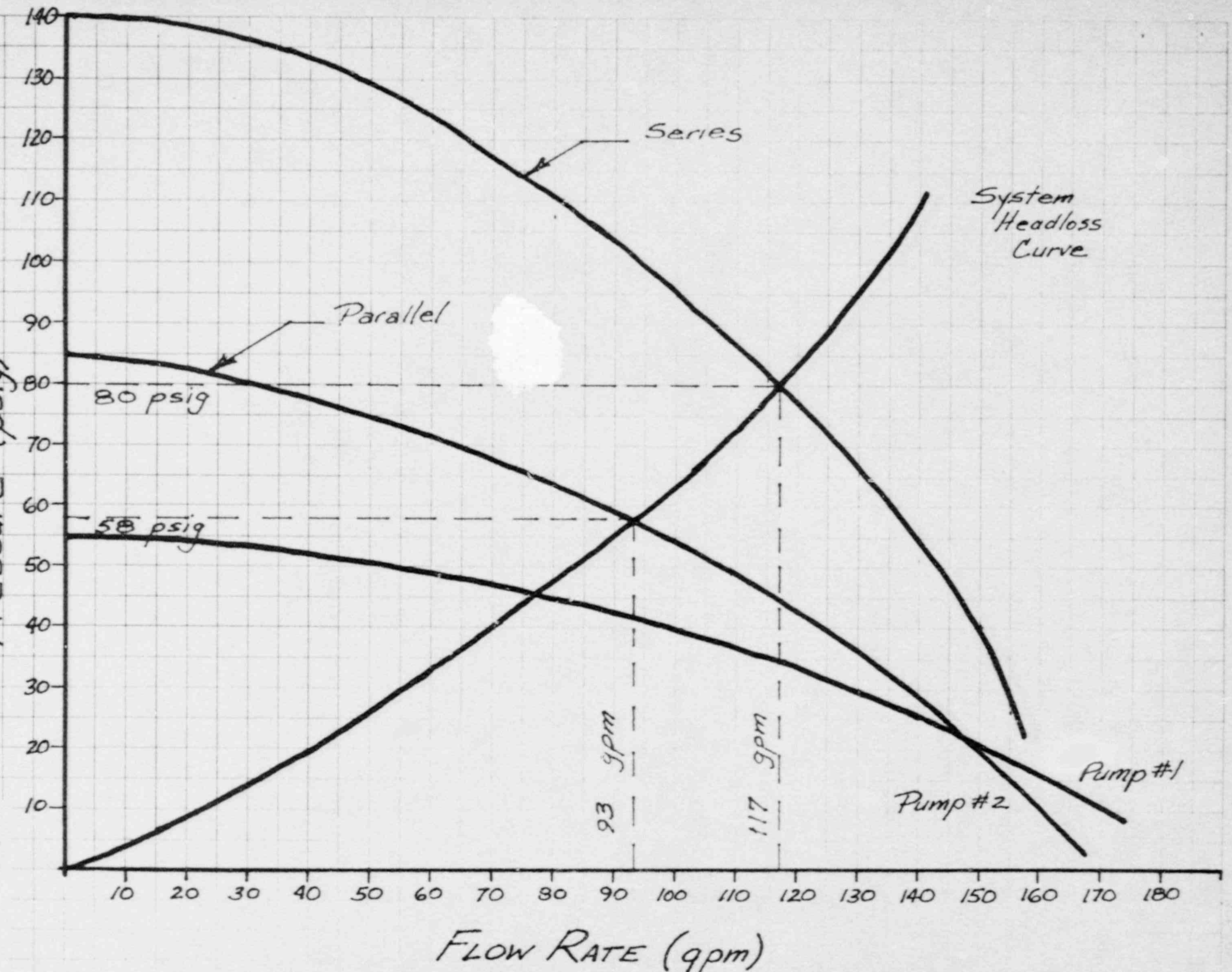
A mode change cannot be made with reliance on an LCO ACTION STATEMENT [1.00].

The plant will enter MODE 3 at 350 F [0.50]

The heatup must be stopped (at < 350 F) [0.50] until the valve can be made operable [0.50].

Ey

REFERENCE
TS 3.5.1 pg. 3/4 5-1
TS 3.0.4 pg. 3/4 0-1
K/A 006 000 G5 3.5 AND G11 3.6 SYSTEM GENERIC
Enabling Objective TSG-01-C-002, 012
006000G5 ...(KA'S)



Attachment 1.1

ANSWER 1.01 (2.50)

a. 1-INCREASE
   2-INCREASE
   3-INCREASE [5 X 0.25]
   4-INCREASE
   5-STAY THE SAME

b. 1-DECREASE
   2 INCREASE
   3-INCREASE [5 X 0.25]
   4-INCREASE
   5-~~INCREASE~~ STAY THE SAME ZY

REFERENCE
Millstone 3 "Reactivity Operations" Lesson Plan
Millstone 3 EOs RTJ-01-C-006,7; RTF-01-C-007; RTE-01-C-006
K/A 192008K121 3.6
K/A 192008K124 3.5
192008K121 192008K124 ...(KA'S)

ANSWER 1.02 (2.00)

a. Press = 58 psig [0.50]
   Flow = 93 gpm [0.50]

b. Press = 80 psig [0.50]
   Flow = 117 gpm [0.50]

REFERENCE
HTFF Fundamentals Ch 12 "Fluid Flow Applications", pages 33-36
Millstone 3 EOs HFE-01-C-005,006
Millstone 3 "Fluid Movement" Lesson Plan
K/A 004000A4.02 3.2
K/A 004000A4.03 2.7
K/A 004010A3.02 3.9
004000A402 004000A403 004010A304 ...(KA'S)

ANSWER 1.03 (2.00)

a. Tsat for 2250 psia (2235 psig) [0.50]
   = 652.67 F [0.50]
   SCM = Tsat - Thot = 652.~~76~~ 67 - 617 = 35.~~76~~ 67 [0.50] Ey

b. SCM will increase [0.50]

REFERENCE
Steam Tables
Millstone 3 Text "RCS" pg 33, "PZR & PRT" pg 16
HTFF Fund, Ch 3 "Thermodynamics", pgs 43-51
Ch 13 "Natural Circulation", pg 4
Millstone 3 EOs HFE-010C-006
K/A 191004K114 2.4
001000A106 001000K503 015000A102 ...(KA'S)

ANSWER 1.04 (1.50)

a. FALSE [0.50]
b. FALSE [0.50]
c. TRUE [0.50]

REFERENCE
MILLSTONE 3 EOs RTH-01-C-006; RTB 01-C-001
MS 3 Delayed Neutrons Lesson Plan, page 23
MS 3 "Neutron Sources and Subcritical Multiplication" Lesson Plan, pages 10,13,22
K/A 01010K508 2.9
001010K508 ...(KA'S)

ANSWER 1.05 (2.75)

a. RCCA worth decreases [0.50]; diffusion length decreases(thus neutrons travel less) [0.25] so are less likely to be absorbed in a control rod [0.25].

b. RCCA worth decreases [0.50]; more competition betweeen boron and RCCAs [0.25].

c. RCCA worth ~~increases~~ decreases [0.50]; fuel depletes over core cycle ~~[0.25]~~ YSB and boron concentration decreases as well ~~[0.25]~~ ~~[0.10]~~ YSB [0.05] Power flux moves towards the outside edges of the core ~~[0.40]~~ YSB [0.45]

REFERENCE
MS 3 Neutron Poisons Lesson Plan, pages 11-16
MS 3 EOs RTI-01-C-001,
K/A 001000K502 2.9
K/A 001010K504 2.2
K/A 001000K530 2.9
001000K502 001000K530 001010K504 ...(KA'S)

EY

ANSWER 1.06 (2.00)

a. Less Negative [0.25] because the fuel temperature coefficient (FTC) becomes less negative as fuel temperature (Rx power) increases [0.50].

b. ~~No Change [0.25] because plutonium buildup [0.25] which makes the FTC become more negative [0.25] is offset [0.25] by the decreasingly less negative fuel temperature change per percent power change [0.25].~~

REFERENCE
Millstone 3 "Reactivity Coefficients & Defects" Lesson Plan, pg 9
Millstone 3 Reactor Theory EO 85
K/A 192004K107 2.9
001000K549 ...(KA'S)

EY

b. Less negative [0.25] because the buildup of Pu 240 and Pu242 [0.25] causes FTC to become more negative [0.25] but the fuel temperature change per percent power change becomes smaller [0.25] overall, the two compete so that the coefficient becomes less negative over core life [0.25]

ANSWER 1.07 (1.50)

a. Decrease [0.25]; more heat is extracted [0.25]
b. Decrease [0.25]; decrease in feedwater temperature [0.25]
c. Increase [0.25]; moderator density decreases [0.25] Eg
of cold leg temperature decreases [0.25]

REFERENCE
Millstone 3 "Plant Cycles" Lesson Plan
Millstone 3 Text "Circulating Water"
Millstone 3 EOs CWS-01-C-015,16
K/A 000003EK103 3.8
K/A 000003EK105 4.1
000003K103 000005K105 ...(KA'S)

ANSWER 1.08 (3.00)

a. 1. Increase [0.50]
2. Increase [0.50]
3. Decrease [0.50]
4. Decrease [0.50]

b. Depletion of U-235 [0.50]
Buildup of Plutonium (Pu 239 & 241) [0.50]

REFERENCE
Millstone 3 "Boiling Processes" Lesson Plan, page 24
Millstone 3 "Delayed Neutrons" Lesson Plan, page 9
Millstone 3 EO's HFI-01-C-004,005
K/A 001000K520 3.2
K/A 001000K528 3.8
K/A 001000K530 3.1
001000K518 001000K528 001000K530 ...(KA'S)

ANSWER 1.09 (3.25)

After the power increase, xenon concentration initially decreases [0.50] since the removal of Xenon by decay [0.25] and burnout [0.25] is greater than the production of Xenon from fission [0.25] and the decay of Iodine [0.25]. After approximaterly 5 (4-6) hours [0.25], the production rate is greater than the removal rate and Xenon concentration increases [0.50] until equilibrium is reached after about 50 hours [0.50]. The new equilibrium Xenon is approximately 1.25 (1.15 - 1.35) times the original value (0.50)

REFERENCE
Millstone 3 "Xenon & Samarium" Lesson Plan, pages 19
Millstone 3 EOs RTK-01-C-004,005,
K/A 001000K533 3.2
K/A 001000K535 2.1
001000K533 001000K535 ...(KA'S)

ANSWER 1.10 (1.50)

a. In 10 sec, delta rho = (20pcm/step) (30 steps/min) (1/6 min) =100 pcm [0.50]

$$SUR = 26(0.1\ sec^{-1}\ (0.001))/(6E-3 - E-3) \quad [0.50]$$
$$= 26\ (1E-4)/5E-3$$
$$= 0.52\ DPM\ [0.50]$$

REFERENCE
Millstone 3 "Reactor Operations" Lesson Plan
Millstone 3 EOs RTG-01-C-006
K/A 001000A106 4.1
K/A 015000A102 3.5
001000A106 015000A102 ...(KA'S)

ANSWER 1.11 (3.00)

a. MORE NEGATIVE [0.50] There is less boron to leave the core area per degree change of coolant temperature. [0.50]

b. LESS NEGATIVE [0.50] The changes in resonant absorption by U238 become less negative as temperature increases. [0.50]

c. MORE NEGATIVE [0.50] Boron concentration decreases, resulting in a more negative MTC. [0.50]

REFERENCE
Millstone 3 "Reactivity Coefficients & Defects" pages 10-26
Millstone 3 EOs RTJ-01-C-003,006; RT1-01-C-001
K/A 001000K515
K/A 001000K538
001000K515 001000K538 ...(KA'S)

ANSWER 2.01 ~~(2.00)~~ [1.50]

1. FALSE [0.5]
2. TRUE [0.5]
3. ~~FALSE [0.5]~~ deleted
4. FALSE [0.5]

EJ

REFERENCE
Millstone 3 EOs PAS-01-C-003,4,5,7
Millstone 3 Text "Instrument Air" pages 2-9
K/A 078000K104 2.6
K/A 078000K103 3.3
K/A 079000K401 2.9
078000K103 078000K104 079000K401 ...(KA'S)

ANSWER 2.02 (2.75)

a. 1. BOTH
2. TRAIN A
3. NEITHER
4. BOTH
5. NEITHER
6. TRAIN B
7. TRAIN A [0.25 each]

b. 1. Containment air recirculation coolers ~~(0.50)~~
2. Neutron shield tank coolers ~~(0.50)~~
3. Containment instrument air compressors [2X0.50]

EJ

REFERENCE
Millstone 3 Text "RPCCW", pages 2-3
Millstone 3 EOs CCP-005,015
K/A 008000K401 3.4
K/A 008030A304 3.6
008000K401 008030A304 ...(KA'S)

ANSWER 2.03 (2.00)

a. 1.Low suction pressure [0.40]
2.Low lube oil pressure [0.40]
3.Sustained undervoltage on bus 34C [0.40]
4.Feedwater isolation trip signal [0.40]
5. Motor protection (amps, volts) [0.40]

[4 X 0.40]

b. Pressure is not high enough. [0.40]

Ey

REFERENCE
Millstone 3 Text "Feedwater", pages 2,6,15
Millstone 3 EOs FWS 01-C-03,10
K/A 059000K416 3.1
K/A 059000K103 3.1
003000A202 ...(KA'S)

ANSWER 2.04 (1.50)

a. Ensures that the SIPs and the CCPs have a suction supply of water [0.50] during the cold leg and hot leg recirculation modes of operation [0.50].

b. Prevents spurious or inadvertent valve movement (due to control circuit problems or unintentional operator action). [0.50]

REFERENCE
Millstone 3 Text "ECCS" pages 15,34,41,
Millstone 3 EOs C-01-ECC-005,032
K/A 006000K406 3.9
K/A 006000A402 4.0
005000K407 005000K509 ...(KA'S)

ANSWER 2.05 (2.75)

a.1 Unable to meet single failure criteria.[0.50]
.2 Sufficient inventory for core reflood. [0.50]
.3 Less volume for nitrogen gas [0.25], therefore sufficient water injection may not occur [0.25].
.4 High pressure gas forces more water into the RCS before blowdown is complete, thus less water is available for core flooding.[0.50]

b. Accumulator tank being filled. [0.25]
RCS inleakage. [0.25]
Faulty pressure regulator. [0.25]
(there may be others)

REFERENCE
Millstone 3 Text "ECCS" pages 46-48
Millstone 3 Technical Specifications
Millstone 3 EOs C-01-ECC-005,015,040
K/A 006000K502 2.8
K/A 006020K603 3.0
K/A 006020A107 3.3
006000K502 006020A107 006020K603 ...(KA'S)

ANSWER 2.06 (2.00)

a. Minimize reaction forces (reduce pipe whip). [0.50]
Minimize mass flow rate that a main steam isolation valve must close against. [0.50]
Reactor coolant system cooldown rate is limited.[0.50]

b. Measure steam flow rate. [0.50]

REFERENCE
Millstone 3 Text "Steam Generator" pages 16,17
Millstone 3 EO MSS-01-C-001
K/A 000040EK106 3.7
K/A 000040EK202 2.6
000040K106 000040K202 ...(KA'S)

ANSWER 2.07 (1.50)

EDG has to come up to speed (10 seconds) [0.50]
QS pump breaker time delay (5 seconds) [0.50]
QS header has to be filled (45 seconds) [0.50]

REFERENCE
MILLSTONE 3 TEXT "CDA" pages 24,25
MILLSTONE 3 TEXT "QS" page 8
MILLSTONE 3 EOs CDA-004,008,023.
K/A 026000K301 4.3
K/A 026000K401 4.5
026000K301 026000K401 ...(KA'S)

ANSWER 2.08 (3.00) deleted replaced with attached question/answer/ref

REFERENCE
Millstone 3 EOs FWA-02-C-000,001,004
Millstone 3 Text "AFW" pages 9,12,17
K/A 061000K101 4.2
K/A 061000K107 3.8
K/A 061000K402 4.6
061000K101 061000K107 061000K402 ...(KA'S)

EG

ANSWER 2.09 (2.00)

a. Significant hydrogen production occurs (increased chance for explosion) [0.50]
b. Kirk Key interlock. [0.50]
c. Air start solenoid valve. [0.50] or Diesel has lost control power
d. No [0.50]

REFERENCE
Millstone 3 EOs 125-02-C-000,003
Millstone 3 Text "125 VDC System" pages 3,4;"Diesel Generator and Support Systems" page 5
Millstone 3 AOP 3563 Attachments A,B,C,D page 1
K/A 063000K103 3.5
K/A 063000K301 4.1
K/A 063000K302 3.7
K/A 063000K502 2.6
063000K103 063000K301 063000K302 063000K502 ...(KA'S)

ANSWER ~~6.01~~ 2.08 ~~(2.00)~~ (3.00) Ey

a. 1 charging pump ~~[0.25]~~ [0.40]
   1 safety injection pump ~~[0.25]~~ [0.40]
   1 residual heat removal pump ~~[0.25]~~ [0.40]
b. close the reactor trip breakers ~~[0.55]~~ [0.80]
c. RCS overpressurization ~~[0.35] at reduced RCS temperatures [0.35]~~ ~~[0.70]~~ [1.00] Ey

REFERENCE
Millstone 3 EO ECC-02-C-010; RPS-02-C-018
Millstone 3 Text "RPSAS" page 65
Millstone 3 Technical Specifications pages B 3/4 3-3,4-15,5-1
K/A 006000 K6.02 3.9
K/A 006000 K6.03 3.9
K/A 006020 K4.06 4.2
006000K602 006000K603 006020K406 ...(KA'S)

ANSWER 6.02 (2.40) N/A

a. 4.
b. 6.
c. 2. [6 X 0.40]
d. 1.
e. 8.
f. 7.

Ey

REFERENCE
Millstone 3 EO ECC-02-C-006
Millstone 3 Text "ECCS" pages 86,87,120
K/A 013000 A3.02 4.2
013000A302 ...(KA'S)

ANSWER 2.10 (2.50)

a.1 gas [0.33]
 2 particulate [0.33]
 3 iodine [0.33)

b.1 S/G blowdown -process stream is isolated
 2 TB drains -process stream is diverted to LWS
 3 Liquid waste -process stream is isolated
 4 Auxiliary condensate -process stream is diverted to aerated drains
 5 RPCCW -none
 6 CVCS -isolates letdown
 7 Containment recirc cooler service water-none
 5 X (0.30)=1.50

REFERENCE
Millstone 3 Text "Radiation Monitoring" pages 2-4,14-15,43-44
Millstone 3 EOs RMS 01-C-007
K/A 073000K101 3.6
K/A 073000K201 2.3
073000K101 073000K201 ...(KA'S)

ANSWER 2.11 (2.00)

a. Prevent charging pump from overheating during periods of low flow operation.[0.80]

b. Seal water heat exchanger.[0.60]

c. RWST.[0.60]

REFERENCE
Millstone 3 Text "CVCS" pages 13,14
Millstone 2 EOs C-01-CHS-022,023
K/A 004010K606 2.7
004010K606 ...(KA'S)

ANSWER 2.12 (1.00)

Supplies service air during a loss or decrease of instrument air pressure [0.50].Maintain the quality of air by first directing service air thorough the instrument air dryer.[0.50]

REFERENCE
Millstone 3 Text "Instrument Air" pg 9
Millstone 3 EO PAS-01-C 005
K/A 079000K401 2.9
079000K401 ...(KA'S)

ANSWER 3.01 (2.20)

Level will decrease [0.50]. Feedwater flow will decrease [0.30] when a large level error signal is generated [0.30]. The reactor will trip on low steam generator level [0.50] because feed flow remains less than steam flow [0.50].

REFERENCE
MP3 NSSS Vol.5 I&C Failure Analysis pgs. 80 & 81
K/A 016000K112 3.5*
K/A 016000K312 3.4*
SGC-01-C-003 & 004
016000K312 016000K112 ...(KA'S)

ANSWER 3.02 (2.70)

a. The automatic controller uses a hot calibrated level instrument [1.00]

b. Ensures adequate water inventory. [0.50]

c. Pressurizer level is programmed to vary based on auctioneered high loop Tave signal [0.40]. As power increases from no load the programed plant Tave increases which generates a level error signal [0.40]. The level error signal will control the charging pump flow control valve to increase level to the desired value [0.40].

REFERENCE
MS3 Sys. Description, Vol.4, Pressurizer Press. & Level pgs. 20-23.
K/A 011000K407 2.9 011000K403 2.6
Pressurizer Press. & Level Control EO PPL-01-C-015, 026 & 027
011000K407 011000K403 ...(KA'S)

ANSWER 3.03 (2.00)

a. Heated junction thermocouples (HJTC)[0.25]. The output of the HJTC is compared to the output of an unheated Reference Thermocouple (tc)[0.50] If the output of the HJTC is sigificantly higher than the reference tc the system considers the sensor to be uncovered (0.50).

b. Level in the vessel head [0.25 each]
Level in the upper plenum
~~Highest~~ temperature of the unheated thermocouples

REFERENCE
MP3 NSSS Vol.4, Incore Tc System and RVLMS, pgs.4-10.
EO ICC-01-C-013 AND 002
K/A 002000K107 3.5* 002000K402 3.5*
002000K402 002000K107 ...(KA'S)

ANSWER 3.04 (.50)

FALSE [0.50]

REFERENCE
MP3 NSSS Vol 4, Incore Tc System and RVLMS, pgs 4-10
EO ICC-01-C-013 and 002
K/A 002000K107 3.5* 002000K402 3.5*
002000K107 002000K402 ...(KA'S)

ANSWER 3.05 (3.00)

a. Steam dumps will arm and actuate to reduce Tave to no load Tave of 557F [0.75].
Tave - Tno load (fixed signal when reactor trip is sensed) [0.25]

b. Steam dumps will arm and actuate to reduce Tave to Tref (no load this example) [0.75]
Tave - Tref (first stage pressure) [0.25]

c. Steam dumps are armed in the steam pressure mode and will actuate to reduce pressure to 1092 psig [0.75].
P pt 507 - P setpoint [0.25].

REFERENCE
MS3 NSSS Vol.3, Steam Dump, pgs. 10-13.
C-01-SDS-010 & 015
K/A 041020K417 3.7 041020K418 3.4

041020K418 041020K417 ...(KA'S)

ANSWER 3.06 (1.80)

a. 20 percent power [0.20 each]
Auto and manual
Prevent nuclear overpower Intr. Range

b. 3 percent below OT-Delta T variable setpoint
Auto and manual
Eliminate cause of impending trip

c. 15 percent power
Prevent auto rod withdrawal below 15 percent power due to secondary Plant instabilities
Auto only

Ey

REFERENCE ← Cf 3203 pg 7
MS3 NSSS Vol. 4, Rod Control System, pgs. 57 and Table 2.
EO C-01-ROD-009
K/A 001000K407 3.7 001000K408 3.2*
001000K408 001000K407 ...(KA'S)

ANSWER 3.07 (3.40)

a. Basis: Uncontrolled RCCA bank withdrawal from subcritical. [0.50]
Block: Manual [0.20] when ~~both~~ 1 of 2 Int. Range are above 10e-10 Amps (P-6) [0.30].

Ey

b. Basis: Protect RCS and its components from overpressure. [0.50]
Block: Cannot be blocked. [0.20]

c. Basis: Protects against excess KW/FT (and high fuel temp.) [0.50]
Block: Cannot be blocked. [0.20]

Ey

d. Basis: Backup overpressure protection and to prevent water discharge through safety valves. [0.50]
Block: Auto blockad (0.20) at less than 10% (P-7) [0.30]

REFERENCE
MP3 NSSS Vol. 5 RPSAS pgs. 39-61
MP3 T.S. pgs. B2-5&6
EO RPS-01-C-020, 040
K/A 012000K402 3.9 012000K406 3.2 012000K604 3.3
012000K604 012000K406 012000K402 ...(KA'S)

ANSWER 3.08 (2.80)

a. Yes [0.20]. The diesel can be started by using the manual start lever on the Air Start Valves [0.80]

b. Overspeed [0.20]
Low lube oil pressure [0.20]
Generator differential current [0.20]

Ey

[0.20] [0.1]
c. Strips all loads except (480 volt) load center and (motor control center supply breakers) ~~and~~ the running charging pumps [0.10]. Selected loads are started in a predetermined sequence [0.40]. Generates a manual start block signal [0.40].

REFERENCE
MS3 BOP Vol. 1 EDG & Support Systems pgs. 14 & 19
MS3 BOP Vol. 1 Diesel Generator Sequencer pgs. 1 & 2
K/A 064000K105 3.4 06400K402 3.9
EO EGS-01-C-007,002,005
EO 4KV-01-C-013
064000K402 064000K105 ...(KA'S)

ANSWER 3.09 (2.20)

a. a neutron and a boron interact to yield an ionized (+) lithium nucleus and an ionized (+) alpha particle [0.50] these ions create additional ion pairs [0.25] which migrate to the detector's charged electrodes [0.25]

OR "LIVE ZERO" Ey

b. 1X10E-11 amp signal is used as a reference for gamma compensation [0.60]

c. SR detectors willnot be energized [0.60]

REFERENCE
Millstone 3 EOs NIS-01-C-700-009,011 & NIS-01-C-702-014
Millstone 3 Text "Excore N.I." pages 6,16,20,26
K/A 015000 K4.01 3.1
K/A 015000 K5.01 2.9
K/A 015000 K6.02 2.6
015000K602 015000K501 015000K401 ...(KA'S)

ANSWER 3.10 (2.40)

a. 4.
b. 6.
c. 2. [6 X 0.40]
d. 1.
e. 8.
f. 7.

REFERENCE
MS3 NSS3 Vol. 2 ECCS, pages 86,87,120
K/A 013000 A3.02 4.2
EO C-01-ECC-007,010,011,012,013 & 014
013000A302 ...(KA'S)

ANSWER 3.11 (2.00)

Train B block valve (MV 8000B) will receive an open signal [0.40] Ey
PORV (PCV-456) will not open [0.40]

Train A block valve (MV 8000A) will receive an open signal [0.40]
PORV (PCV-455A) will open [0.40] because train "A" of COP system
uses autioneered LOW temperature for a pressure setpoint [0.40]

REFERENCE
MS3 NSSS VOL. 4, Pzr. Press. & Level Control, pages 13-15
K/A 010000 K4.03 4.1
EO PPL-01-C-013 & 014
010000K403 ...(KA'S)

ANSWER 4.01 (2.50)

a. standby instrument air
shutdown instrument air [3 X 0.33]
service

b. air pressure decreasing rapidly [0.50]
loss of turbine/feedwater control [0.50]

c. use of S/G atmospheric steam bypass valves [0.50]

REFERENCE
Millstone 3 EOs A62-01-C-000.001
Millstone 3 AOP 3562 page 3
K/A 000065 EK3.08 3.7
K/A 000065 EA2.06 3.6*
000065K308 000065A206 ...(KA'S)

ANSWER 4.02 (2.00)

a. FALSE [0.50 each]

b. TRUE

c. TRUE

d. TRUE

REFERENCE
MS3 EOP Format and Use pgs. 2,4 & 6
Enabling Objective for EOP format and use No.2,3 & 5
K&A 194001A102 Plant-Wide Generic 4.1*.
194001A102 ...(KA'S)

ANSWER 4.03 (2.50)

1. RCS Subcooling - > 30F [0.50 each]

2. Core Exit temperatures stable or decreasing

3. S/G Pressures stable or decreasing

4. RCS (loop) Hot Leg Temperatures are stable or decreasing.

5. RCS (loop) Cold Leg Temperatures are near saturation temperature for S/G Pressure.

REFERENCE
EOP 35 ES-0.1 Reactor Trip Response Enabling Objective No. 5
K&A 193008K122 4.2*
2193008K12 ...(KA'S)

ANSWER 4.04 (2.00)

S - Subcriticality
C - Core Cooling
H - Heat Sink
P - Integrity
Z - Containment
I - Inventory
[6 at 0.3 each, 0.20 for correct order]

REFERENCE
MS3 EOP Development pgs. 5-7
EOP Format and Use Enabling Objective No. 1
EOP Development Enabling Objective No. 6
K/A 000029 G012 4.1*/4.2*

ANSWER 4.05 (2.50)

a. 1. Shift Supervisor approval.
2. Notification of Operations Supervisor/Duty Officer.
3. Identification on Shift Turnover Sheet.
4. Do not use cheater bars. [4 at 0.20 each]

b. OPEN - Partially close [0.30] open to backseat [0.30] then close one-quarter turn. [0.30]

c. Take valve off its backseat [0.25] to prevent undue stress to the valve.[0.25]

d. Shift Supervisor [0.30]

REFERENCE
Department Inst. 3-OPS-3.07, pgs. 3,4 & 6
K/A 194001 K1.01 3.6/3.7
194001K101 ...(KA'S)

ANSWER 4.06 (3.00)

candidate #1: rejected (0.25) since he has no history on file and will exceed 300 MREM/QTR whole body exposure (0.5)

candidate #2: rejected (0.25) since he will exceed the quarterly limit of 1000 mrem whole body without Health Physics Supervisor's approval (0.5)

candidate #3: rejected (0.25) since she will exceed 500 mrem whole body during the term of her pregnancy (0.5)

candidate #4: accepted (0.25) since he will not exceed the admin limit of 1000 mrem/qtr (0.25) or the whole body limit of 10000 mrem lifetime exposure. (.25)

REFERENCE
SHP 4902 pgs. 9,10 & HP Form 4902-1
HP Enabling Objectives 14 and 21.
K/A 194001 K1.03 2.8/3.4
194001K103 ...(KA'S)

ANSWER 4.07 (2.00)

a. Increase Tave above 551 F in 15 minutes [0.45] ~~[0.25]~~ or Hot Standby ~~Shutdown~~ in next 15 minutes [0.35] ~~[0.25]~~. ~~If uncontrolled Borate (per AOP 3566) [0.25]~~

b. Stop and investigate reason. [0.25]

c. Terminate SU and insert control rods [0.25] Borate [0.25].

d. Insert Control Banks [0.25] Recalculate ECP [0.25]

REFERENCE
OP 3202 Reactor Startup, pos. 6.7 & 10
Enabling Objective B02-01-C-007,008,013 and 019
K/A 001010A101 3.7 001010A207 3.6
001010A101 001010A207 ...(KA'S)

ANSWER 4.08 (3.00) (1735 ~~1738~~ psia for adverse Cont.)

a. RCS pressure less than 1435 psia, and
At least one charging or SI pump running [3 at 0.33 ~~0.50~~ each]

b. - RCS subcooled less than 30 F per core exit Tc
- (less than 90 F for adverse containment) OR

- Cannot maintain PZR LEVEL > 7% [4 at 0.50 each]
- (> 50% for adverse containment)

REFERENCE
Procedure EOP 35 E-0 Foldout pg.
EOP 35 E-0 Enabling obj. #3
K/A 000011 EA1.03 4.0
K/A 000011 EA2.11 3.9
000011A103 000011A211 ...(KA'S)

ATTACHMENT 2

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

FACILITY: MILLSTONE 3

REACTOR TYPE: PWR-WEC4

DATE ADMINISTERED: 87/12/15

EXAMINER: YACHIMIAK, E.

CANDIDATE: Master Key

INSTRUCTIONS TO CANDIDATE:

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

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

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

Candidate's Signature

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

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

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

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

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

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

6. Use only the paper provided for answers.

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

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

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

10. Skip at least three lines between each answer.

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

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

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

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

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

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

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

18. When you complete your examination, you shall:

a. Assemble your examination as follows:

(1) Exam questions on top.

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

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

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

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

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

QUESTION 5.01 (2.50)

a. WHAT are the TWO (2) mechanisms by which Moderator Temperature Coefficient (MTC) becomes MORE NEGATIVE as core life ages from BOL to EOL? (1.00)

b. The limits for Maximum and Minimum NEGATIVE MTC values are based on TWO (2) postulated FSAR accidents. WHAT are these TWO (2) accidents, WHEN during core life are they assumed to occur, and for which limit (maximum or minimum) do they apply? (1.50)

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

QUESTION 5.02 (2.00)

HOW (More Negative, Less Negative, No Change) does the Doppler Only Power Coefficient change if the below parameters change as follows. JUSTIFY WHY. Include both positive and negative reactivity effects.

a. Reactor power Increases from 50% to 100% power. (0.75)

b. Core age Increases from BOL to EOL. (1.25)

QUESTION 5.03 (2.40)

In WHAT direction would the following parameters be changing (Increasing, Decreasing, More Negative, Less Negative, Not Changing) if xenon oscillations were induced by a 20 step insertion of control rod bank D? Assume the plant is at 90% power with rods in manual at 210 steps, with all other systems are in their normal at power line up. Assume no operator action is taken after T=0 hours, the time when the rods are inserted.

For parameters a through d, assume the xenon oscillation is at T=4 hours into a 28 hour period.

a. core power
b. AFD
c. T(hot)
d. T(cold)

For parameters e through h, assume the xenon oscillation is at T=18 hours into a 28 hour period.

e. core power
f. AFD
g. T(hot)
h. T(cold)

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

QUESTION 5.04 (3.75)

Given Millstone 3 procedure OP-3209A, "Estimated Critical Conditions" (Attachment 2), FULLY complete OPS Form 3209A-1 using the supplied data, EXCEPT for those items marked N/A.

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

QUESTION 5.05 (3.00)

For EACH of the THREE (3) transients listed below, match the stated parameters with the expected trend. NOTE: small perturbations and variations are not shown.

a. A steam generator atmospheric relief valve fails OPEN with reactor power at 65%, EOL equilibrium xenon conditions, rods in AUTO with bank D rods at 170 steps.

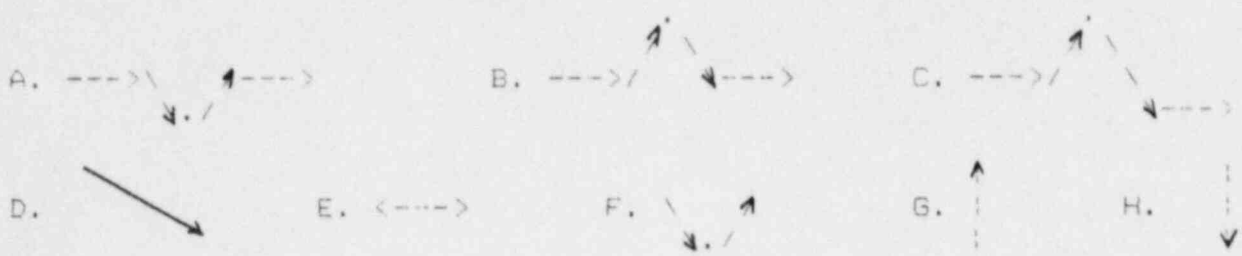
1. Tavg
2. Reactor Power
3. MWe
4. Pressurizer Level

b. A 10% step DECREASE in turbine load is performed with reactor power at 100%, EOL equilibrium xenon conditions, rods in AUTO with bank D rods at 180 steps.

1. Tavg
2. Steam Pressure
3. Pressurizer Level
4. Pressurizer Pressure

c. A single Shutdown Bank rod drops into the core with reactor power at 90% BOL equilibrium xenon conditions, rods in MANUAL with bank D rods at 210 steps. A reactor trip DOES NOT occur.

1. Tavg
2. MWe
3. Pressurizer Level
4. Pressurizer Pressure



QUESTION 5.06 (2.50)

Given Millstone 3 ENG Form 31002-5, "Core Heat Balance" (Attachment 3), use Steam Tables and steam/water properties tables (Attachment 4) to fill in all EMPTY spaces and calculate Core Power in %.

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

QUESTION 5.07 (2.20)

Answer the following questions TRUE or FALSE:

a. If a centrifugal pump's speed is DOUBLED, its flow will DOUBLE only if it is running against a zero discharge head.

b. If a centrifugal charging pump is in service and the operator INCREASES its flowrate, the pump's available Net Positive Suction Head (NPSH) will DECREASE because of greater head losses in the suction piping.

c. If a condensate pump was operating at "RUNOUT" conditions, cavitation would be present.

d. When starting a condenser circulating water pump, motor starting current is REDUCED by CLOSING the pump's discharge valve.

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

QUESTION 5.08 (2.40)

HOW (Increase, Decrease, No Change) and WHY would each of the following parameters affect the margin to DNB. Assume no change in power.

a. Pressurizer temperature increase 5 degrees
b. Mass flow rate through the core increases 10%
c. AFD increases to +10%

QUESTION 5.09 (2.25)

HOW (Increase, Decrease, No Change) does Differential Rod Worth (DRW) change for the following conditions? JUSTIFY your answer. Consider each case separately.

a. RCS average temperature increases from 557 F. to 587 F.

b. Core Age increases from BOL to EOL.

c. Bank D control rods are withdrawn from 100 steps to 228 steps.

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

QUESTION 5.10 (2.00)

Upon a loss of offsite power, list FOUR (4) indications (parameter and trend) that natural circulation has been established.

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

QUESTION 6.01 (2.00)

a. For a large break LOCA, WHAT are the minimum number of emergency core cooling system pumps required to cover exposed fuel and limit possible core damage? (0.75)

b. Following SI reset, WHAT operator action(s) must be performed in order to reinstate automatic re-initiation of SI? (0.55)

c. If TWO (2) charging pump are OPERABLE in MODE 4, WHAT RCS system safety limit can be violated if BOTH are operated? Include TWO (2) significant parameters. (0.70)

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

QUESTION 6.02 (2.40)

MATCH the ESF Status Panel group description from the right hand column with its respective Group number from the left hand column. Only ONE (1) description matches each Group number.

a. Group I

b. Group II

c. Group III

d. Group IV

e. Group V

f. Group VI

1. Consists of lights for those components whose status is changed during a CDA
2. Consists of lights covering the steam system
3. Consists of lights that only light during the injection phase
4. Most of these lights should always be off; however, some may illuminate during special or infrequent operation
5. Consists of lights for those components whose status only changes when in the cold leg recirc phase
6. Consists of lights that are normally off and will come on after an SIS
7. Consists of lights for those components whose status only changes for the hot leg recirc phase
8. Consists of lights for those components whose status is changed during the cold leg recirc phase and remains in the hot leg recirc phase

QUESTION 6.03 (1.70)

a. Briefly explain HOW neutrons produce current in a Source Range (SR) Excore Nuclear Instrumentation (NI) system detector. (0.70)

b. A reactor shutdown is in progress with the SR detectors reading about 10,000 cps and both Intermediate Range (IR) detectors reading 1X10E-11 amps. Ten minutes later the SR detectors read about 1,000 cps but the IR detectors still read 1X10E-11 amps. WHY does the IR detector output NOT decrease below 1X10E-11 amps? (0.50)

c. The plant is operating at 100% power with N44 out-of-service. If an automatic reactor trip occurs AND N43 is failed as is, WHAT affect, if any, will this have on the NI system's ability to monitor neutron flux as the plant stabilizes to Mode 3 conditions? Assume no manual action is taken to restore N43. (0.50)

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

QUESTION 6.04 (2.40)

a. WHY does letdown pressure control valve PCV-131 maintain pressure downstream of the letdown heat exchanger at about 350 psig? (0.60)

b. WHY must letdown heat exchanger 3CHS*E2 cool letc wn flow to less than 135 degrees F during Mode 1 operations? (0.60)

c. WHY is Volume Control Tank (VCT) temperature limited to less than 135 degrees F during Mode 1 operations? (0.60)

d. WHAT cover gas is used in the VCT during normal operations and WHY is it used? (0.60)

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

QUESTION 6.05 (2.50)

A large break LOCA has occured and the Quench Spray system (QSS) is in operation.

a. WHAT are TWO (2) functions for the QSS concerning containment pressure? Include any applicable time constraints. (1.00)

b. WHAT is the design purpose for using NAOH? (0.50)

c. HOW long after a CDA signal is generated do the Containment Recirculation system (RSS) pumps automatically start? (0.50)

d. WHY do the RSS pumps require a time delay before running? (0.50)

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

QUESTION 6.06 (3.00)

a. The turbine driven auxiliary feedwater (TDAFW) pump has THREE (3) main steamline supply lines. TWO (2) are required to meet single failure criteria considerations. WHY is the third supply line required? (0.50)

b. WHAT are the THREE (3) sources of water for the AFW system in their order of preferred usage? (1.00)

c. WHAT are the FOUR (4) signals which can automatically start the motor driven AFW pumps? (1.00)

d. WHY are the automatic start signals for the TDAFW pump different from the MDAFW pumps? (0.50)

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

QUESTION 6.07 (2.00)

a. WHY would increasing battery charger output voltage above 143 volts be a concern when charging a discharged battery?

b. WHAT prevents standby battery charger 301A-3 from being used to feed more than one 125 VDC bus?

c. WHAT emergency diesel generator starting system component can NOT be energized when either a loss of 125 VDC bus 301A-1 occurs OR the barring gear is engaged?

d. Will a loss of any ONE (1) class 1E 125 VDC bus cause a DIRECT reactor trip? If so, briefly explain HOW. Assume the plant is at 100% power.

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

QUESTION 6.08 (4.00)

For EACH of the following RPS trips, STATE its design basis and WHEN, if at all, the trip may be bypassed or blocked:

a. Power Range High Neutron Flux (low setpoint)
b. Low Pressurizer Pressure
c. Power Range High Negative Flux Rate
d. Low-Low S/G Level
e. Turbine Trip Reactor Trip
f. Power Range High Neutron Flux (high setpoint)

QUESTION 6.09 (3.00)

For EACH of the following situations, explain HOW Tavg is used by the indicated control system. Include in your answer the specified information requested by each part.

a. While at 100% power, a turbine load reduction at 5% per minute is started with ROD CONTROL in AUTO. Include all applicable programmed setpoint values.

b. PRESSURIZER (PZR) LEVEL CONTROL of PZR level when reactor power is increased from 50% power to 100% power. Include all applicable programmed setpoint values.

c. FEDDWATER VALVE CONTROL after a turbine trip from 60% power occurs. Include all applicalbe setpoint values, logic, and coincidences.

QUESTION 6.10 (2.00)

Explain HOW the Cold Overpressure Protection system (COPS) will respond if an operator were to arm both Trains with hot leg wide-range RTD TE-413A failed LOW? Justify your answer by explaining WHY each Train is OR is not affected because of the RTD failure, and WHAT actions takes place. Assume the plant is shutdown with the RCS at 300 F. and 700 psig.

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

QUESTION 7.01 (2.50)

Answer the following questions regarding Residual Heat Removal (RHR) system operation per procedure OP-3310A, "Residual Heat Removal."

a. WHAT is the reason for the maximum RHR suction pressure limit of 765 psia? (0.50)

b. WHY is the operator cautioned not to initiate RHR system operation until RCS temperature is less than 350 F? (0.50)

c. WHY should the RHR system NOT be isolated with the RCS solid? (0.50)

d. If a Safety Injection signal (SIS) occurs while the RHR system is aligned for RCS cooldown, WHAT are THREE (3) actions an operator should perform before realigning the RHR system injection flowpath for the SI mode? (1.00)

QUESTION 7.02 (2.00)

Answer the following questions regarding Circulating Water (CW) system operation:

a. After starting a CW pump, WHY is an operator required to wait 2 minutes before starting the another pump? (0.50)

b. WHAT is the minimum number of CW pumps required to be running when a radioactive effluent discharge is in progress? (0.50)

c. WHAT is the maximum temperature rise of the discharge water above the intake water temperature and WHY does this limit exist? (1.00)

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

QUESTION 7.03 (1.75)

Answer the following questions concerning Fuel Transfer System procedure OP-3303C.

a. WHAT major component is used for emergency retrieval of the fuel transfer car from inside containment when a traverse drive failure occurs due to a GEARBOX FREEZUP? (0.50)

b. WHAT major component is used for emergency retrieval of the fuel transfer car from inside containment when a traverse drive failure occurs due to a MOTOR BURNOUT? (0.50)

c. WHAT are TWO (2) resons for performing an emergency retrieval of the transfer car from inside containment when a traverse drive failure occurs? (0.75)

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

QUESTION 7.04 (2.25)

Answer the following questions regarding Reactor Coolant Pump (RCP) operations:

a. WHY are the RCP seal leakoff isolation valves closed when RCS pressure is below 115 psia? (0.75)

b. If the plant is at 90% power, HOW long can a RCP operate with a seal failure before power must be below P-8? (0.50)

c. HOW long can a RCP be allowed to operate with a seal failure before its seal leakoff isolation valve must be closed? Assume the plant is at 50% power. (0.50)

d. If the plant is operating at 35% power and RCP 1A has a seal failure, WHY is steam generator (S/G) level increased to 75% prior to stopping the RCPs? (0.50)

QUESTION 7.05 (3.00)

Answer the following questions concerning procedure AOP-3566, "Immediate Boration"

a. WHAT are the FIVE (5) entry conditions for AOP-3566? (2.00)

b. WHY is Immediate Boration charging flow required to be < 233 gpm? (0.50)

c. WHY is Immediate Boration charging flow required to be > 33 gpm? (0.50)

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

QUESTION 7.06 (3.00)

Answer the following questions concerning "Steam Generator Tube Rupture" procedure EOP 35 E-3.

a. WHAT are the TWO (2) criteria for determining whether the RCPs should be stopped? (1.00)

b. WHAT are the FOUR (4) criteria for identifying a ruptured steam generator? (1.00)

c. WHAT TWO (2) parameters are used to determine adverse containment conditions? Include setpoints. (1.00)

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

QUESTION 7.07 (3.50)

The following questions concern "Reactor Startup" procedure OP 3202.

a. WHAT TWO (2) operator actions are required if criticality is NOT achieved when control rods reach the MAXIMUM position on the ECP? (1.00)

b. WHAT operator action is required when diluting the RCS boron concentration by more than 50 ppm? (0.50)

c. WHY is boron concentration adjustment NOT allowed while withdrawing a control rod bank? (0.50)

d. HOW is proper alignment and bank overlap determined during rod withdrawal for criticality? Include in your answer WHAT is done and HOW often (or at what points) it is done. (1.00)

e. While recording critical data at 10E-8 amps, you determine that loop D Tavg has been 535 F. for the last 15 minutes. WHAT action are you required to take? Include any applicable time limitations. Assume no failed instruments. (0.50)

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

QUESTION 7.08 (2.50)

Answer the following questions regarding "Loss of Instrument Air" procedure AOP 3562.

a. If instrument air pressure was decreasing, WHAT THREE (3) air compressors could be started in an attempt to restore pressure? (1.00)

b. Under WHAT TWO (2) conditions should the reactor be tripped? (1.00)

c. HOW would Tavg be controlled after a reactor trip due to a loss of instrument air? (0.50)

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

QUESTION 7.09 (2.50)

The following questions deal with the Emergency Operating Procedures' usage rules:

a. While implementing the actions required by accident recovery procedure ES-1.2, "Post LOCA Cooldown and Depressurization," the STA reports the following critical safety function status tree conditions. Place the below conditions in the order they should be addressed. (1.00)

Containment Integrity	- ORANGE
Inventory	- YELLOW
Core Cooling	- ORANGE
Heat Sink	- YELLOW

b. HOW is the operator made aware of tasks that must be fully completed before proceeding to another instruction? (0.50)

c. ARE the CAUTION statements from E-1 still applicable if a transition to FR-H.1 is performed? (0.50)

d. WHAT procedure's Foldout page is applicable for ES-0.4? (0.50)

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

QUESTION 7.10 (2.00)

Answer the following questions regarding EOP 35 ES-1.1, "SI Termination," and EOP 35 ES-1.2, "Post LOCA Cooldown and Depressurization."

a. If SI is terminated while in ES-1.1, WHAT TWO (2) criteria are checked to verify that ECCS flow is not required? (1.00)

b. HOW and WHY can RCS depressurization affect Pressurizer level while in ES-1.2? (1.00)

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

QUESTION 8.01 (2.00)

Answer the following questions using procedure ACP-QA-2.06B, "Station Bypass/Jumper Control." (Attachment 5)

a. If a Technical Specification change is required and an unreviewed safety question is found to exist, WHAT TWO (2) organizations must approve the change before the installation of a jumper can be authorized? (0.50)

b. WHEN is the Shift Supervisor allowed to grant exception to performing a second verification of a jumper installation? (0.50)

c. WHO are TWO (2) people (by job title) that must complete and sign the Assessment Section of the jumper-lifted lead-bypass sheet? (0.50)

d. Under WHAT conditions can a jumper be installed WITHOUT using procedure ACP-QA-2.06B? (0.50)

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

QUESTION 8.02 (1.75)

a. WHO are THREE people (by job title) that can give authorization to take the reactor critical? Assume the reactor had been previously shutdown using procedure OP-3207, "Reactor Shutdown." (0.75)

b. WHAT TWO (2) actions are required of a licensed operator per ACP-6.01, "Control Room Procedure," and 10 CFR Part 55 to ensure that when a trainee manipulates the controls of the reactor, the actions are performed correctly? (0.50)

c. WHAT is the minimum fire brigade composition as required per ACP-OA-2.05? (0.50)

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

QUESTION 8.03 (2.00)

For each of the following safety tags, state the limitations which are placed on equipment when the tag is used.

a. Red
b. Yellow
c. Green Striped
d. Blue

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

QUESTION 8.04 (2.00)

Per Section 3/4.5.4 of Technical Specifications (Attachment 6), RWST boron concentration must be verified through a surveillance test at least once per 7 days. Given the following test dates, have any Technical Specification surveillance requirements been violated AND is the plant in a Limiting Condition for Operation (LCO)? Justify your answer for EACH part. Assume the plant has been at 100% power since 11/21/87.
NOTE: November has 30 days

12/14/87 at 1000
12/5/87 at 1300
11/28/87 at 2200
11/21/87 at 1600

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

QUESTION 8.05 (2.00)

For EACH of the following conditions described below, utilize the Code of Federal Regulations provided to you to determine whether the NRC should be notified within ONE hour or FOUR hours AND indicate WHY by specifying the appropriate section numbers/letters. example: xx.xx (i)(i)(a)

a. A controlled liquid effluent release was determined to have occured at 5 times the Maximum Permissible Concentration (MPC).

b. An operator made the decision to take actions that departed from facilit Technical Specifications in an emergency to protect the public health an safety.

c. During a refueling outage, several pipe snubbers that were attached to the RCS cold legs were found to be inoperable.

d. While performing a surveillance test at 100% power, a Safety Injection signal was mistakenly generated and an estimated 2000 gallons of RWST water was injected into the core.

QUESTION 8.06 (1.00)

a. WHO are THREE (3) people (by job title) which the shift supervisor is DIRECTLY responsible for notifying of an emergency during normal working hours? (0.75)

b. WHICH NRC classification level requires full activation of the SEO? (0.25)

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

QUESTION B.07 (2.00)

Classify the following emergency events under the NRC incident classification scheme using EPIP-4701-3, "Emergency Action Levels." (Attachment 7) Justify your answer for EACH part.

a. Thuderstorms have caused a loss of offsite power with "A" diesel generator out-of-service. Diesel generator "B" starts, but does not load due to a fire in the undervoltage circuitry.

b. Steam generator 1A is completely depressurized with the air ejector radiation monitor in the alarmed condition.

c. SI has been actuated and RCS pressure is 1300 psig and decreasing.

d. A plant worker has notified the control room that there is a fire in the turbine building.

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

QUESTION 8.08 (3.00)

Using Attachment 8, Technical Specification section 3.8.1.1, determine WHEN (date/time) the plant would be required to be in MODE 3 and WHAT LCO applies at EACH step of the sequence of events shown below. Assume the plant is at 100% power. Consider the information from all previous steps in the sequence of events to be applicable for each new step.

a. Normal Station Service Transformer A (NSSA) is taken out-of-service for maintenance on 12/15/87 at 1100.

b. Emergency Generator A fails its surveillance test on 12/15/87 at 1200.

c. NSSA is returned to service on 12/15/87 at 2000.
Emergency Generator B fails its surveillance test on 12/15/87 at 2000.

d. Emergency Generator A is returned to service on 12/16/87 at 0200.

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

QUESTION 8.09 (3.00)

Answer the following questions relative to Technical Specification Bases.

a. What are FOUR (4) reasons WHY the reactor must be made critical with an RCS Tavg of atleast 551 degrees F.?

b. WHY does the turbine have overspeed protection? List TWO (2) reasons.

c. WHY is there a limit on the specific activity of the reactor coolant? List TWO (2) reasons.

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

QUESTION 8.10 (2.00)

Per Technical Specifications, WHAT FOUR (4) conditions must be met so that continuous monitoring of hot channel factors is not required during normal operations?

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

QUESTION 8.11 (2.75)

In accordance with the requirements of Technical Specification 6.12.2,

a. Fully explain HOW access into a room, where general area radiation levels exceed 1000 MR/h, is CONTROLLED. (1.25)

b. HOW would access be controlled inside containment, where an enclosure CAN NOT be constructed around an individual high radiation area and general area radiation levels exceed 1000 MR/h? (1.50)

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

QUESTION 8.12 (1.50)

Using Attachment 9, Technical Specification section 3/4.11, answer the following questions concerning radioactive effluent releases.

a. State the Technical Specification definition of REAL MEMBER OF THE PUBLIC. (0.75)

b. Before starting a radioactive liquid effluent release, Health Physics notifies you that the total whole body dose to a REAL MEMBER OF THE PUBLIC would be increased to 20 mrem for the calender year after the release. Could this release be performed without exceeding Technical Specification limits? Jusitify WHY or WHY NOT. (0.75)

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

ANSWER 5.01 (2.50)

a. reduction in boron concentration [0.50]
fission product buildup [0.50]
b. ejected RCCA [0.25] at BOL [0.25] Minimum [0.25]
main steam line break [0.25] at EOL [0.25] Maximum [0.25]

REFERENCE
Millstone 3 Reactor Theory EO 84.
Millstone 3 Lesson Plan "Reactivity Coefficients and Defects" pages 20-22
K/A 192004 K1.06 3.1
192004K106 ...(KA'S)

ANSWER 5.02 (2.00)

a. Less Negative [0.25] because the fuel temperature coefficient (FTC) becomes less negative as fuel temperature (Rx power) increases [0.50]
b. ~~No Change [0.25] because plutonium buildup [0.25] which makes the FTC become more negative [0.25] is offset [0.25] by the decreasingly less negative fuel temperature change per percent power change [0.25]~~

REFERENCE
Millstone 3 Reactor Theory EO 85
Millstone 3 Lesson Plan "Reactivity Coefficients and Defects" page 9
K/A 192004 K1.07 2.9
192004K107 ...(KA'S)

b. less negative [0.25] because the buildup of PU-240 and PU-242 [0.25] cause FTC to become more negative [0.25] but the fuel temperature change per percent power change becomes smaller [0.25] overall, the two ~~both~~ compete so that the coefficient becomes less negative [0.25] over core life.

ANSWER 5.03 (2.40)

a. Not Changing
b. More Negative
c. Decreasing
d. Decreasing
e. Not Changing [8 X 0.30]
f. Less Negative
g. Increasing
h. Increasing

REFERENCE
Millstone 3 Reactor Theory EO 94
Millstone 3 Lesson Plan "Xenon and Samarium" pages 21-24
K/A 001000 K5.38 4.1
001000K538 ...(KA'S)

Ey

ANSWER 5.04 (3.75) SEE Pages 45A-45C

ITEM	VALUE	ALLOWANCE (+/-)
1.2	+1475	50
3.1	~~3000~~ 0	50
3.2	2850	50
4.3	85	
5.1	1100	40
5.2	200	40
6.3	9.4	0.1
6.4	-2820	
7.1	-3110	
7.2	331	
7.7	869	
8.2	~~2000~~ -200	
8.3	~~C 57~~ D 180	10
8.4	~~200~~ 2000	
8.5	~~B 180~~ C 57	10

EJ

[15 X 0.25]

REFERENCE
Millstone 3 Reactor Theory EO 105.b.
Note to Facility: EO refers to OP-3304 rather than the correct OP-3209A
Millstone 3 OP-3209A
K/A 192008 K1.07 3.6
192008K107 ...(KA'S)

90C	1/27/88	3-88-189
FORM APPROVED BY UNIT 3 SUPERINTENDENT	EFFECTIVE DATE	PORC MTG. NO.

CALCULATED BY	DATE	APPROVED BY	DATE

ESTIMATED CRITICAL CONDITION - FIXED ROD POSITION
REFERENCE CRITICAL DATA

DATE ~~12/15/87~~ 12/17/87 TIME 0100

RCS TAVG 587 °F

RCS PRESSURE 2250 PSIA

POWER 100 %

RCS BORON 900 ppm BURNUP 4015 MWD/MTU

CONTROL BANK D 180 STEPS CONTROL BANK C 228 STEPS

OTHER 100% power maintained for last 10 days

XENON ______ pcm (-)

SAMARIUM ______ pcm (-)

LAST SHUTDOWN TIME 1550 DATE 11/17/87

ESTIMATED STATUS AT CRITICALITY

DATE 12/15/87 TIME 2100

TAVG 557 °F RCS PRESS 2250 psia

BORON (PRESENT) 1200 ppm BURNUP (PRESENT) 4015 MWD/MTU

DESIRED CRITICAL POSITION

BANK C AT 150 STEPS

1. POWER DEFECT (OPS Form 3209-3)
(STEP 5.3)
1.1 Reference Power 100 %
1.2 pcm at Reference Power + 1475 ±25 pcm ①

2. MODERATOR DEFECT. (OPS Form 3209-4)
(STEP 5.4.1)
2.1 Tavg (at reference conditions) 587 °F
2.2 Tref (at reference conditions) 587 °F
2.3 MTC (at reference conditions) N/A pcm/°F
2.4 Moderator Defect (at reference conditions)
(2.1 - 2.2) X MTC = N/A pcm

(STEP 5.4.2)

2.5 Estimated Tavg 557 °F

2.6 MTC (at estimated conditions) N/A pcm/°F

2.7 Moderator Defect at estimated conditions
(2.5 - 557) x MTC = N/A pcm

---

3. XENON DEFECT (Computer or OPS Form 3209-5/6) circle one.
(STEP 5.5)

3.1 Estimated Xenon ~~3800±50~~ 0 pcm (-) 2

3.2 Reference Xenon 2850±50 pcm (-) 3

3.3 Xenon Defect (3.1 - 3.2) +2850 ~~-950~~ pcm Eg

---

4. SAMARIUM DEFECT (Computer or OPS Form ~~3209-7~~ 3209-14) circle one.
(STEP 5.6)

4.1 Estimated Samarium 685 pcm (-)

4.2 Reference Samarium 600 pcm (-)

4.3 Samarium Defect (4.1 - 4.2) - ~~+~~ 85 pcm 4 Eg

(STEP 5.7)

5. INTEGRATED ROD WORTH (OPS Form 3209-8/~~9/10~~) Use 3209-8 ONLY

5.1 Estimated Rod Worth 1100 ±40 pcm (-) 5

5.2 Reference Rod Worth 200 ±40 pcm (-) 6

5.3 Rod Worth Defect (5.1 - 5.2) -900 pcm

---

6. BORON DEFECT
(STEP 5.8)

6.1 Present Boron Concentration 1200 ppm

6.2 Reference Boron Concentration 900 ppm

6.3 Boron Worth (OPS Form 3209-1) 9.4 ± 0.1 pcm/ppm (-) 7

6.4 Boron Defect (6.1 - 6.2) x 6.3 -2820 pcm 8

---

7. CALCULATIONS

(STEP 5.9)

7.1 Sum Defects (1.2 + 2.4 + 2.7 + 3.3 + 4.3 + 5.3 + 6.4) ~~-340~~ +520 pcm 9

7.2 Boron Equivalent of Defects (7.1 + 6.3) -55 ~~331~~ ppm 10

7.3 Nominal PPM at Reference BU N/A ppm

7.4 Nominal PPM at Present BU N/A ppm

7.5 Burnup Change (7.3 - 7.4) N/A ppm (+)

7.6 Boron Change to Go Critical (7.2 + 7.5) ~~331~~ -55 ppm

7.7 Critical Boron Concentration (6.1 - 7.6) ~~869~~ 1255 ppm 11

Ey

---

8. LIMITS ON CONTROL ROD POSITION.

(STEP 5.10)

8.1 Rod Worth at ECP

Bank C at 150 steps 1100 pcm (-)

8.2 Rod Worth at Minimum Insertion

(8.1 + 900 pcm) -200 ~~2000~~ pcm 12

8.3 Rod Position at Minimum Insertion

Bank D ~~C~~ at 180 ~~57~~±10 steps 13

8.4 Rod Worth at Maximum Insertion

(8.1 - 900 pcm) 2000 ~~200~~ pcm (-) 14

8.5 Rod Position at Maximum Insertion (Cannot be below 0% power rod insertion limit)

Bank C ~~D~~ at 57 ~~180~~±10 steps 15

Ey

ACTUAL CRITICAL DATA (STEP 5.22)

DATE N/A TIME N/A

Rod Position Control Bank D at N/A Steps

Control Bank C at N/A Steps

Other: N/A

Tavg Loop 1 T411A N/A °F Loop 2 T421A N/A °F

Loop 3 T431A N/A °F Loop 4 T441A N/A °F

ANSWER 5.05 (3.00)

a.
 1. A OR F
 2. G OR
 3. F
 4. A OR F

Ey

b.
 1. C
 2. ~~H~~ G
 3. ~~B~~ C [12 X 0.25]
 4. B

c.
 1. H OR O
 2. D OR H
 3. H
 4. A

REFERENCE
Millstone 3 EO ACA-01-C-006
Millstone 3 Text "Transient Analysis" pages 37-39, Table 7
K/A 192008 K1.18 3.5
K/A 192008 K1.20 3.9
K/A 192008 K1.21 3.8
K/A 192008 K1.24 3.6
192008K118 192008K120 192008K121 192008K124 ...(KA'S)

ANSWER 5.06 (2.50)

See Attached Core Heat Balance **Pages 46A & 46B**

REFERENCE
Millstone 3 Heat Transfer, Fluid Flow, and Thermodynamics (HT,FF,& T) EO 54
Millstone 3 SP-31002 ENG Form 31002-5
K/A 193007 K1.08 3.4
193007K108 ...(KA'S)

CORE HEAT BALANCE

Average over Measurement Interval

A.	Pzr Pressure	2250	psia
B.	RCS Loop 3 Tc	557	°F
C.	Letdown Flow (CHS-F132)	75	gpm
D.	Charging Flow (CHS-F121)	87	gpm
E.	Charging Pressure (CHS-P120)	2500	psia
F.	Charging Temperature (CHS-T126)	510	°F
G.	VCT Temperature (CHS-T116)	100	°F

		S/G 1	S/G 2	S/G 3	S/G 4
H.	Steam Pressure (PSIA)	1100		SAME	
I.	Blowdown Flow (gpm)	50		AS	
J.	Feed Temperature (°F)	420		S/G	
K.	Feed Pressure (PSIA)	1400		1	

M.	Letdown Enthalpy (From A & B)	[1]	559.0	BTU/lbm
N.	Charging Enthalpy (From E & F)	[1]	~~512.5~~ 500.2	BTU/lbm
P.	Charging Specific Volume (From E & ~~K~~ G)	[½]	0.016	ft³/lbm
Q.	Charging Flow Correction Factor			

Sy

1 Ch 1

$$.12716 \times \frac{2.0 \times 10^{-5} \times (\text{CHS-T116}) + .9983}{\sqrt{V} \times 1.0011} = \text{N/A}$$

1 Ch 2

Where V is the specific volume of the Charging Fluid (CHS-T116 and CHS-P120).

R.	Corrected Charging Flow (DxQ)		N/A	gpm
S.	Charging Flow (60xR)/(7.48xP)		N/A	lbm/hr
T.	CVCS Heat Loss Sx(M-N)/3412141		4.69	MWT
U.	RCP Seal Flow (Total)-12 gpm		N/A	gpm
V.	Seal Enthalpy (From E & G)	[½]	74.5	BTU/lbm
W.	Seal Specific Volume (From E & G)		N/A	ft³/lbm
X.	Corrected Seal Flow (60xU)/(7.48xW)		N/A	lbm/hr
Y.	Seal Heat Loss (X)x(M-V)/3412141		4.74	MWT

		S/G 1	S/G 2	S/G 3	S/G 4
Z.	Feed Enthalpy BTU/lbm (From J & K)	398.7	[1]		
AA.	SAT STM Enthalpy BTU/lbm (From H)	1189.1	[½]		
BB.	SAT Water Enthalpy BTU/lbm (From H)	557.5	[½]		
CC.	S/G Enthalpy (AA)	1189.1	---		
DD.	S/G Δ Enthalpy (CC-Z)	790.4	[¼]	SAME	
EE.	Actual Feed Flow	1.9 E+6	---	AS	
FF.	S/G Power [(EE)(DD)] BTU/hr	1.5 E+9	[¼]	S/G	
GG.	Sat Water Spec. Vol. (From H)	0.02159	[½]	1	
HH.	Blowdown Flow (60xI)/(7.48 x GG)	N/A			
JJ.	Blowdown Loss [~~CC~~ HH (DD-BB)]	-0-			
KK.	Total S/G Power (FF - JJ) BTU/hr	1.5 E+9	---		
LL.	Total NSSS Power Σ (KK) BTU/hr	6.0 E+9	[¼]		

MM. LL/3412141 = 1758.4 MWT [¼]

NN. Net RCP Heat Input = 16 MWT

PP. TOTAL CORE POWER (MM-NN+T+Y) = 1751.9 MWT [¼]

CORE POWER in % (PP)(100)/3411 = 51.36 % [¼]

[1] = 0.3 X 3 = 0.9

[½] = 0.2 X 5 = 1.0

[¼] = 0.1 X 6 = 0.6

2.5

ANSWER 5.07 (2.20)

a. TRUE
b. TRUE [4 X 0.55]
c. FALSE
d. TRUE

REFERENCE
Millstone 3 HT,FF,& T EOs 24,25,27,29
Millstone 3 Lesson Plan "Fluid Movement" pages 14,15,24-26
K/A 191004 K1.05 2.4
K/A 191004 K1.07 2.9
K/A 191004 K1.12 2.7
K/A 191004 K1.15 2.8
191004K105 191004K107 191004K112 191004K115 ...(KA'S)

ANSWER 5.08 (2.40)

a. Increases [0.30] as PRZR temperature rises, so does saturation pressure [0.50]
b. Increases [0.30] RCS core delta T will decrease [0.30] reducing T(hot) [0.20] (because T(cold) is constant)
c. Decreases [0.30] because more power is being produced in the top half of the core [0.30] causing (hot channel factor bounds OR DNB) limits in this area to be approached [0.20]

REFERENCE
Millstone 3 HT,FF,& T EOs 55,58
Millstone 3 Lesson Plan "Boiling Processes" pages 24,25
K/A 193008 K1.05 3.6
193008K105 ...(KA'S)

ANSWER 5.09 (2.25)

a. Increases [0.25] moderator density decreases [0.25] increasing the probability for absorbtion by a control rod (because of increased diffusion length) [0.25]
b. Increases [0.25] boron concentration is reduced [0.25] reducing the competition between boron and RCCAs [0.25]
c. Decreases [0.25] there is less neutron flux for interaction as the rods move to the top of the core [0.50]

OR For cycle #1 [0.25] flux shift [0.25] causes rod worth to decrease [0.5].

cy

REFERENCE
Millstone 3 Reactor Theory EO 76
Millstone 3 Lesson Plan "Neutron Poisons" - Justification for Learning - Criteria 3, pages 11,12
K/A 192005 K1.05 3.1
K/A 192005 K1.06 2.9
K/A 192005 K1.07 2.8
192005K105 192005K106 192005K107 ...(KA'S)

ANSWER 5.10 (2.00)

- core exit TCs - stable or decreasing
- RCS hot leg temperatures - stable or decreasing [4 X 0.50]
- RCS cold leg temperatures - at saturation for S/G pressure
- RCS subcooling based on core exit TCs - greater than 30 degrees F
- S/G pressures - stable or decreasing

REFERENCE
Millstone 3 HT,FF,& T EO 67
Millstone 3 EOP 35 E-3 Table 2 page 1 of 1
K/A 193008 K1.22 4.2
193008K122 ...(KA'S)

ANSWER 6.01 (2.00)

a. 1 charging pump [0.25]
   1 safety injection pump [0.25]
   1 residual heat removal pump [0.25]
b. close the reactor trip breakers [0.55]
c. RCS overpressurization [~~0.35] at reduced RCS temperatures [0.35]~~
   [0.70] [2]

REFERENCE
Millstone 3 EO ECC-02-C-010; RPS-02-C-018
Millstone 3 Text "RPSAS" page 65
Millstone 3 Technical Specifications pages B 3/4 3-3,4-15,5-1
K/A 006000 K6.02 3.9
K/A 006000 K6.03 3.9
K/A 006020 K4.06 4.2
006000K602 006000K603 006020K406 ...(KA'S)

ANSWER 6.02 (2.40)

a. 4.
b. 6.
c. 2. [6 X 0.40]
d. 1.
e. 8.
f. 7.

REFERENCE
Millstone 3 EO ECC-02-C-006
Millstone 3 Text "ECCS" pages 86,87,120
K/A 013000 A3.02 4.2
013000A302 ...(KA'S)

ANSWER 6.03 (1.70)

a. a neutron and a boron interact to yield an ionized (+) lithium nucleus and an ionized (+) alpha particle [0.30] these ions create additional ion pairs [0.20] which migrate to the detector's charged electrodes [0.20]
b. 1X10E-11 amp signal is used as a reference for gamma compensation [0.50]
c. SR detectors cannot be energized [0.50]

REFERENCE
Millstone 3 EOs NIS-02-C-004,005,006
Millstone 3 Text "Excore N.I." pages 6,16,20,26
K/A 015000 K4.01 3.3
K/A 015000 K5.01 3.2
K/A 015000 K6.02 2.9
015000K401 015000K501 015000K602 ...(KA'S)

ANSWER 6.04 (2.40)

a. prevent flashing [0.30] of water after orifices [0.30]
b. prevent damage to demineralizers [0.60]
c. protect the RCP seals [0.60]
d. hydrogen [0.30]
oxygen control [0.30]

REFERENCE
Millstone 3 EOs CHS-02-C-000,002,005
Millstone 3 Text "CVCS" pages 5,16
Millstone 3 OP3304A pages 52,62
K/A 004000 K4.03 3.6
K/A 004000 K5.04 3.2
K/A 004010 K3.01 3.9
K/A 004010 K4.01 3.1
004000K403 004000K504 004010K301 004010K401 ...(KA'S)

ANSWER 6.05 (2.50)

a. limits pressure rise inside containment [0.50]
   reduces pressure to < atmospheric [0.25] within 60 minutes [0.25]
b. minumize corrosion in containment (by increasing pH) [0.50]
c. 11 minutes +/- 10 seconds [0.50]
d. ensure adequate available NPSH for pump operation [0.50]

REFERENCE
Millstone 3 EOs CDA-02-C-001,004,007
Millstone 3 Text "CTMT Depress." pages 1,2,25; "ECCS" page 82
K/A 026000 K4.04 4.1
K/A 026020 K4.03 4.3
026000K404 026020K403 ...(KA'S)

ANSWER 6.06 (3.00)

a. because one RCS loop can be isolated [0.50]
b. demineralized water storage tank [0.30]
   condensate storage tank [0.30] [0.10] for order
   service water (Long Island Sound) [0.30]
c. SIS
   LOP [4 X 0.25]
   CDA
   low-low S/G level
d. reduce the possibility of S/G overfill [0.50]

REFERENCE
Millstone 3 EOs FWA-02-C-000,001,004
Millstone 3 Text "AFW" pages 9,12,17
K/A 061000 K1.01 4.2
K/A 061000 K1.07 3.8
K/A 061000 K4.02 4.6
061000K101 061000K107 061000K402 ...(KA'S)

ANSWER 6.07 (2.00)

a. significant hydrogen production occurs (resulting in an increased risk of explosion) [0.50]
b. Kirk Key interlock [0.50]
c. air start solenoid valve [0.50]
d. no [0.50]

REFERENCE
Millstone 3 EOs 125-02-C-000,003
Millstone 3 Text "125VDC System" pages 3,4; "Diesel Generator and Support Systems" page 5
Millstone 3 AOP 3563 Attachments A,B,C,D page 1
K/A 063000 K1.03 3.5
K/A 063000 K3.01 4.1
K/A 063000 K3.02 3.7
K/A 063000 K5.02 2.6
06300CK103 063000K301 063000K302 063000K502 ...(KA'S)

ANSWER 6.08 (4.00)

a. reactivity excursions [0.50]
bypassed when P-10 satisfied [0.25]
b. DNB [0.50]
blocked below P-7 [0.25]
c. multiple [0.25] drop rod [0.25]
d. heat sink [0.50]
~~blocked when loop isolation valves closed [0.25]~~ can not be blocked [0.25]
e. minimize RCS thermal transients [0.50]
blocked below P-9 [0.25]
f. excessive heat flux [0.25] leading to DNB [0.25]

REFERENCE
Millstone 3 EOs RPS-02-C-017,036
Millstone 3 Text "RPSAS" pages 42-57
K/A 012000 K4.02 4.3
012000K402 ...(KA'S)

ANSWER 6.09 (3.00)

a. Tavg is used with Tref to develop rod speed and direction input for rod control [0.50] the difference between the two temperatures generates a programmed response as follows:

-1 F. to +1 F. --- no rod motion [0.25]
> +/- 1 F. --- varies from 8 to 72 steps/minute [0.25]

b. Tavg is used to generate a reference level for Pressurizer level [0.50] program level varies from 25% at 557 F. [0.25] to 61.5% at 587 F. [0.25]

c. 2/4 [0.10] Low Tavg channels at 564 F. [0.20] coincident with a reactor trip (P-4) signal [0.20] causes feedwater isolation [0.50]

REFERENCE
Millstone 3 EOs TIS-02-C-003; ROD-02-C-007,025; PPL-02-C-002; FWS-02-C-002
Millstone 3 Texts "Rod Control" page 23; "Pzr. Press. & Level" pages 24,25; "Feedwater" page 12
K/A 001000 A1.01 3.8
K/A 011000 A1.04 3.1
K/A 059000 A3.06 3.2
001000A101 011000A104 059000A306 ...(KA'S)

ANSWER 6.10 (2.00)

Ez

Train B block valve (MV 8000B) will ~~open [0.40]~~ receive an open signal [0.40]
PORV (PCV-456) will not open [0.40]
Train A block valve (MV 8000A) will ~~open [0.40]~~ receive an open signal [0.40]
PORV (PCV-455A) will open [0.40] because train "A" of COP system uses autioneered LOW temperature for a pressure setpoint [0.40]

REFERENCE
Millstone 3 EOs PPL-02-C-003,004
Millstone 3 Text "Pzr. Press. & Level Control" pages 13-15
K/A 010000 K4.03 4.1
010000K403 ...(KA'S)

ANSWER 7.01 (2.50)

a. RHR piping protection (prevent overpressurization) [0.50]
b. minimize thermal shock to the RHR heat exchangers [0.50]
c. prevent possible RCS overpressurization [0.50]
d. STOP the RHR pump
   OPEN the RHR/RWST suction valve [3 X 0.33]
   START the RHR pump

REFERENCE
Millstone 3 EOs RHR-02-C-007,009,010,011
Millstone 3 OP 3209 page 29,32; OP 3310A page 6; Text "RHR" page 3
K/A 005000 K1.11 3.6
K/A 005000 K4.01 3.2
K/A 005000 K5.05 3.1
K/A 005000 G0.10 3.5
005000G010 005000K111 005000K401 005000K505 ...(KA'S)

ANSWER 7.02 (2.00)

a. to allow time for the (hydraulic transient) pressures associated with a pump start to subside throughtout the system [0.50]
b. three (3) [0.50]
c. 24 F [0.50]
   environmental protection concerns [0.50]

REFERENCE
Millstone 3 EOs CWS-02-C-001,003,006,007
Millstone 3 OP 3325A page 8,10
K/A 191004 K1.16 2.9
K/A 075000 K1.02 3.1
K/A 075000 G0.10 2.3
075000G010 075000K102 191004K116 ...(KA'S)

ANSWER 7.03 (1.75)

a. crane or hoist hook [0.50]
b. emergency handwheel [0.50]
c. isolation [0.40] and repair [0.35]

REFERENCE
Millstone 3 EOs FHS-02-C-002,019
Millstone 3 OP 3303C pages 7,8
K/A 000036 EA1.04 3.7

K/A 000036 EK3.03 4.1
000036A104 000036K303 ...(KA'S)

ANSWER 7.04 (2.25)

a. prevent contamination [0.35] from the seal leakoff line from being forced back into the RCP seal chamber [0.40]
b. 30 minutes [0.50]
c. 5 minutes [0.50]
d. prevent a reactor trip on low-low level due to S/G shrink [0.50]

REFERENCE
Millstone 3 EOs RCP-02-C-010; A54-01-C-003,005
Millstone 3 AOP-3554 page 3; OP-3301D page 5
K/A 003000 K6.02 3.1
K/A 000015 EK3.03 4.0
K/A 000015 EK3.07 4.2
000015K303 000015K307 003000K602 ...(KA'S)

ANSWER 7.05 (3.00)

a. - control bank height below the low-low limit
- failure of ~~one~~ two or more control rods to fully insert following a reactor trip or shutdown
- uncontrolled cooldown of the RCS following a reactor trip or shutdown
- uncontrolled or unexplained reactivity addition (indicated by abnormal control bank insertion, increasing Tavg, or increasing nuclear power)
- failure of the reactor makeup control system

Ey

[5 X 0.40]
b. maintain NPSH for charging pump ~~limit cold water addition (CAE)~~ [0.50]
c. provide a minimum rate of negative reactivity insertion ~~(CAE)~~ [0.50]

REFERENCE
Millstone 3 EOs A66-02-C-001,002,004
Millstone 3 AOP-3566 pages 2,3
K/A 000024 EK3.01 4.4
K/A 000024 EK3.02 4.4
000024K301 000024K302 ...(KA'S)

ANSWERS -- MILLSTONE 3 -87/12/15-YACHIMIAK, E.

ANSWER 7.06 (3.00)

a. at least one charging or SI pump running [0.50]
   RCS pressure < 1435 psia (1700 psia for adverse containment) [0.50]
b. - unexpected increase in S/G level
   - high S/G sample radiation
   - high S/G steamline radiation [4 X 0.25]
   - high S/G blowdown line radiation
c. containment Temperature ~~pressure~~ [0.25] > 180°F ~~15.0 psia~~ [0.25]
   radiation [0.25] > ~~1000 R/hr~~ 1x10^5 R [0.25]

Ey

REFERENCE
Millstone 3 EOs E30-01-C-001,009; EOU-01-C-007
Millstone 3 EOP 35 E-3 pages 3,4,13
Millstone 3 exam 85/05/14 question 7.05
K/A 000038 EK3.06 4.5
000038K306 ...(KA'S)

ANSWER 7.07 (3.50)

a. - terminate the startup by fully inserting all control banks [0.50]
   - recalculate the ECP [0.50]
b. manually energize the PZR heaters (to induce spray flow) [0.50]
c. positive reactivity must not be changed by more than one controlled method at a time [0.50]
d. rod motion is stopped [0.25] at every 114 steps [0.25] and bank demand position [0.25] is compared to digital rod position indication [0.25]
e. be in Hot Standby [0.25] within the next 15 minutes [0.25]

REFERENCE
Millstone 3 EOs G02-01-C-000,007,011,016,019
Millstone 3 OP 3202 pages 6-10
K/A 001000 G0.10 3.5
K/A 001010 A3.01 3.9
K/A 001010 A3.02 3.7
K/A 001010 A3.03 4.0
001000G010 001010A301 001010A302 001010A303 ...(KA'S)

ANSWER 7.08 (2.50)

a. standby instrument air
   shutdown instrument air [3 X 0.33]
   service air
b. air pressure decreasing rapidly [0.50]
   loss of turbine/feedwater control [0.50]
c. use of S/G atmospheric steam bypass valves [0.50]

REFERENCE
Millstone 3 EOs A62-01-C-000,001
Millstone 3 AOP 3562 page 3
K/A 000065 EK3.08 3.9
K/A 000065 EA2.06 4.2
000065A206 000065K308 ...(KA'S)

ANSWER 7.09 (2.50)

a. 1) Core Cooling
   2) Containment Integrity [0.25 X 4]
   3) Heat Sink
   4) Inventory
b. the step containing the task or an associated NOTE or CAUTION will explicitly state the requirement [0.50]
c. yes [0.50]
d. E-0 [0.50]

REFERENCE
Millstone 3 EOs EOU-01-C-000,001,002
Milstone 3 EOPs E-3 page 11, E-0 foldout page
Westinghouse Owners Group ERG Users Guide pages 5,6,10
K/A 194001 A1.02 3.9
194001A102 ...(KA'S)

ANSWER 7.10 (2.00)

a. RCS subcooling [0.50]
   PZR level [0.50]
b. PZR level rapidly increases [0.50] due to voiding in the reactor vessel head region [0.50]

REFERENCE
Millstone 3 "EOP 35 ES-1.1, ES-1.2" EO 7.
Millstone 3 ES-1.1 page 7; ES-1.2 page 12
K/A 000040 EK3.04 4.7
K/A 000040 EA2.05 4.5
000040A205 000040K304 ...(KA 5)

ANSWER 8.01 (2.00)

a. NRB [0.25] and NRC [0.25]
b. If the second verification would result in significant radiation exposure [0.50]
c. SS [0.25] Duty officer [0.25]
d. If identified and controlled in another approved procedure [0.50]
OR if equipment is deemed inoperable
EY

REFERENCE
Millstone 3 Terminal Objective (TO) BYJ-02-C-000
Millstone 3 ACP-QA-2.06B pages 3,9,10,12

ANSWER 8.02 (1.75)

a. Station Superintendent [0.25]
Unit Superintendent [0.25]
Operation's Supervisor [0.25]
b. instruct [0.25] and directly observe the individual [0.25]
c. one fire brigade leader [0.25]
four fire brigade members [0.25]

REFERENCE
Millstone 3 Enabling Objectives (EOs) CRP-02-C-012; CRP-02-C-024;
FPP-02-C-002
Millstone 3 ACP-QA-2.05 page 10, ACP-QA-6.01 page 6
10 CFR Part 55 Section 55.9(b)

ANSWER 8.03 (2.00)

a. not to be operated under any circumstance [0.50]
b. contains precautions or information which should be understood prior to operating [0.50]
c. prevents reenergizing equipment if it trips [0.50]
d. only to be energized by order of the individual to whom the tag was issued [0.50]

REFERENCE
Millstone 3 EOs TAG-02-C-010,021,023,025
Millstone 3 ACP-QA-2.06A pages 3,4

ANSWER 8.04 (2.00)

- YES [0.20] because the interval between 12/5 and 12/14 [0.40] exceeds the maximum allowable extension of 25% of the surveillance interval [0.40]
- YES [0.20] because a failure to perform the surveillance test within the specified time interval [0.40] constitutes a failure to meet the OPERABILITY requirements for the LCO [0.40]

REFERENCE
Millstone 3 EO SRV-02-C-001
Millstone 3 ACP-QA-9.02 page 4
Technical Specification (T.S.) 4.0.3

ANSWER 8.05 (2.00)

a. 4 hour 50.72 (b)(2)(iv)(B) [0.50]
b. 1 hour 50.72 (b)(1)(i)(B) [0.50]
c. 4 hour 50.72 (b)(2)(i) [0.50]
d. ~~1 hour 50.72 (b)(1)(iv)~~ [0.50]
4 hour 50.72 (b)(2)(ii) EY

REFERENCE
ACP-QA-10.01
10 CFR 50.72

ANSWER 8.06 (1.00)

a. - security shift supervisor [0.25]
- duty officer [0.25] → OR any superintendent
- operations supervisor [0.25]

b. alert (or higher) [0.25]

EY

REFERENCE
Millstone 3 Emergency Plan Training EOs 12,16
EPIP-4010A page 3
EPIP-4112 page 3

ANSWER 8.07 (2.00)

a. SITE AREA -- for > 15 minutes [0.50]
b. ALERT -- steamline break with primary to secondary leakage [0.50]
c. SITE AREA -- LOCA [0.50] OR ALERT w/appropriate justification
d. UNUSUAL EVENT -- if fire last for < 15 minutes [0.50]

REFERENCE
Millstone 3 Emergency Plan Training EO 6
EPIP-4701-3

grade per assumptions made full credit for higher classification with correct basis

EY

ANSWER 8.08 (3.00)

a. 12/18/87 at 1700 [0.50] LCO 3.8.1.1.a. [0.25]
b. 12/16/87 at 0600 [0.50] LCO 3.8.1.1.b. [0.25]
c. 12/16/87 at 0400 [0.50] LCO 3.8.1.1.~~d~~e. [0.25]
d. 12/~~18~~ 19/87 at ~~1700~~ 0200 [0.50] LCO 3.8.1.1.~~d~~e. [0.25]

EY

REFERENCE
Millstone 3 TO TSG-02-C-000
TS 3.8.1.1

ANSWER 8.09 (3.00)

a. to ensure: - MTC within analyzed range
- trip instrumentation within operating range
- P-12 above its setpoint [4 X 0.25]
- the pressurizer is in an OPERABLE status
- the reactor vessel is abone its minimum RT(NDT) temperature
b. protects safety related components, equipment, and structures [0.50] against ~~excessive overspeed which can generate~~ missiles [0.50]
c. ensures that SITE BOUNDARY doses will not exceed 10 CFR Part 100 guideline values [0.50] following a SGTR accident [0.50]

EY

REFERENCE
Millstone 3 TO TSG-02-C-000
TS Bases 3/4.1.1.4, 3/4.3.4, 3/4.4.8

ANSWER 8.10 (2.00)

- control rods in a single group move together with no individual rod insertion differing by more than +/- 12 steps, indicated, from the group demand position [0.50]
- control rod groups are sequenced with overlapping groups [0.50]
- the control rod insertion limits are maintained [0.50]
- the axial power distribution (expressed in terms of AFD) is maintained within limits [0.50]

REFERENCE
Millstone 3 EO TSG-02-C-005
Millstone 3 Technical Specifications page B 3/4 2-5
K/A 193009 K1.07 3.3
193009K107 ...(KA'S)

ANSWER 8.11 (2.75)

a. doors are locked [0.50] and the key maintained under administrative control [0.50] of the SCO [0.25] and/or Health Physics Supervision [0.25]

FACILITY NOTE: T.S section 6.12.2 states that the shift Foreman maintains key control. ACP-6.01 does not define who this person is.

b. barricade the area [0.50]
conspicuously post the radiation levels [0.50]
a flashing light shall be activated as a warning device [0.50]

REFERENCE
Millstone 3 EO TSG-02-C-016
T.S. 6.12.2 page 6-24
K/A 194001 K1.03 3.4
194001K103 ...(KA'S)

ANSWER 8.12 (1.50)

a. an individual who is exposed [0.25] to existing dose pathways [0.25] at one particular location [0.25]
b. Yes [0.25] because T.S. 3.11.3 allows 25 mrem to the whole body [0.50]

REFERENCE
Millstone 3 EOs TSG-02-C-001; LWS-02-C-002; GWS-02-C-000
Millstone 3 T.S. section 1.0 page 1-4, section 3/4-11 page 11-6
K/A 068000 G0.05 2.9

K/A 071000 GO.05 3.1
068000G005 071000G005 ...(KA'S)

Master Key

Attachment 1

$f = ma$

$v = s/t$

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

$w = mg$

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

$E = mc^2$

$KE = 1/2\, mv^2$

$a = (V_f - V_o)/t$

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

$PE = mgh$

$V_f = V_o + at$

$w = \theta/t$

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

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

$W = v\,\Delta P$

$A = \frac{\pi D^2}{4}$

$\Delta E = 931\,\Delta m$

$\dot{m} = V_{av} A \rho$

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

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

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

$Pwr = W_f \Delta h$

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

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

$TVL = 1.3/\mu$

$HVL = -0.693/\mu$

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

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

$SUR = 26.06/T$

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

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

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

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

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

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

$T = (\beta - \rho)/(\bar{\lambda}\rho)$

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

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

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

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

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

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

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

$I_1 d_1 = I_2 d_2$

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

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

$\Sigma = \sigma N$

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

$R/hr = 6\ CE/d^2$ (feet)

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.
1 ft. H_2O = 0.4335 lbf/in.

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)
1 BTU = 778 ft-lbf

ATTACHMENT 2

E. J. Mroczka — Form Approved by Station Superintendent

2-1-85 — Effective Date

STATION PROCEDURE COVER SHEET

A. IDENTIFICATION

Number OP 3209A Rev. 0

Title REACTIVITY CALCULATIONS - ESTIMATED CRITICAL CONDITIONS

Prepared By DAVID MC DANIEL

B. REVIEW

I have reviewed the above procedure and have found it to be satisfactory.

TITLE	SIGNATURE	DATE
DEPARTMENT HEAD	F. F. Burton	11/20/85
Asst Rx Eng	Keith Jensen	10/1/85

C. UNREVIEWED SAFETY QUESTION EVALUATION DOCUMENTATION REQUIRED:

(Significant change in procedure method or scope as described in FSAR) YES [ ] NO [X]
(If yes, document in PORC/SORC meeting minutes)

ENVIRONMENTAL IMPACT

(Adverse environmental impact) YES [ ] NO [X]
(If yes, document in PORC/SORC meeting minutes)

D. INTEGRATED SAFETY REVIEW REQUIRED

(Affects response of Safety Systems, performance of safety-related control systems or performance of control systems which may indirectly affect safety system response.) YES [ ] NO [X]
(If yes, document in PORC/SORC meeting minutes.)

E. PROCEDURE REQUIRES PORC/~~SORC~~ REVIEW YES [X] NO [ ]

F. PORC/~~SORC~~ APPROVAL

PORC/~~SORC~~ Meeting Number 3-85-189

G. APPROVAL AND IMPLEMENTATION

The attached procedure is hereby approved, and effective on the date below:

JOCurrent — Station/Service/Unit Superintendent

11/27/85 — Effective Date

SF 301
Rev. 7
Page 1 of 1

REACTIVITY CALCULATIONS - ESTIMATED CRITICAL CONDITIONS

Page No.	Eff. Rev.
1 - 9	0

1. OBJECTIVE

1.1 This procedure provides a method and forms for determining Estimated Critical Conditions (ECC). An ECC may be determined by performing this procedure, or running SP-3R11 on the Modcomp Process Computer, when it is deemed valid by Reactor Engineering.

2. PREREQUISITES

2.1 The most recent steady-state critical position data is available for calculating an ECC (this data will be supplied by Reactor Engineering).

3. INITIAL CONDITIONS

3.1 None

4. PRECAUTIONS

4.1 The Rod Insertion Limit (RIL) shall never be exceeded when the reactor is critical (except during Low Power Physics Testing) (LPPT). Whenever the reactor is in Hot or Cold Shutdown the RCS must be borated such that the reactor would not be critical with rods below the RIL.

4.2 If criticality is not achieved within ± .9% Δk/k (900 pcm) of the calculated ECC, insert the Control Banks in sequence to zero steps and recalculate the ECC. If the newly calculated ECC does not differ from the original ECC (as modified for any conditions which have changed) notify Reactor Engineering. If Reactor Engineering can not find any mistakes place the plant in Hot Standby. An evaluation will be conducted by PORC. Upon approval, start up will commence using Reactor Engineering's recommendations.

4.3 This procedure shall be performed no more than four hours prior to going critical per Technical Specification 4.1.1.1.1c.

4.4 The reactor shall not be brought critical with rods below the Rod Insertion Limit per Technical Specification 3.1.3.6 unless allowed per Special Test Exception 3.10.1.

NOTE: Computer Program SP 3R11 "Estimated Critical Position" may be used to determine an ECC in place of this procedure.

4.5 when selecting a Critical Rod position the Estimated Critical Rod height should be chosen such that the control rods have at least 400pcm left prior to exceeding the upper limit (Control Bank D at 228 steps or the Rod withdrawal limit if applicable), and the estimated rod position should be at least 400pcm above the Rod Insertion Limit. CH 2

5. PROCEDURE

NOTE 1: The burnup (in megawatt days per metric tonne ie. MWD/MTU) change from the reference condition to the present conditions should not exceed 200 MWD/MTU unless designated by Reactor Engineering on the ECC Data Sheet.

NOTE 2: If it is desired to select a critical rod position and vary boron concentration perform Steps 5.1 through 5.10 and complete OPS Form 3209A-1.

If it is desired to select a critical boron concentration and vary rod position perform Steps 5. through 5.20 and complete OPS Form 3209A-~~1~~2. CH 1

5.1 Enter reference critical position data. This should be the most recent steady state data available (supplied by Reactor Engineering) on OPS Form 3209-15.

CAUTION: Ensure proper use of signs (+ or -) throughout this calculation.

5.2 Enter Estimated Conditions at Time of Criticality. Data for Date, Time, Temperature, and Pressure, are estimated. Rod Position is the Desired Position. Boron and Burnup are entered at the existing conditions.

5.3 Item 1. Determine the power defect at the Reference Power. The number on the curve is negative, but a positive number should be entered on OPS Form 3209A-1. (Use OPS Form 3209-3).

5.4 Moderator Defects - Item 2.

NOTE: The power defect curve assumes the reactor is operating with Tavg equal to Tref ± 2°F. If Tavg is not equal to Tref ± 2°F it will be necessary to perform Step 5.4.1.

5.4.1 Determine the difference between T average at the reference condition and Tref at the reference condition. Multiply this difference by the Moderator Temperature Coefficient (MTC) (OPS Form 3209-4/5) at the reference condition to determine the moderator defect. (This number will be negative if Tavg > Tref and MTC is negative.)

NOTE: If the estimated critical temperature is other than 557°F ±2°F, it will be necessary to perform Step 5.4.2.

5.4.2 Determine the difference between Tavg at the estimated condition and 557°F. Multiply by the MTC (OPS Form 3209-4/5) to determine Moderator Defect at the Estimated Condition. (The value is negative if Tavg > 557°F and MTC negative.)

NOTE: If the computer is not available, use OPS Form 3209-5 and OPS Form 3209-6 to obtain Xenon worth if the reactor was shutdown from an equilibrium Xenon state. If the reactor was not at or near an equilibrium Xenon state prior to shutdown, Reactor Engineering will determine Xenon worth.

5.5 Item 3. Determine the Xenon worth at the estimated time of criticality (use SP 3R7/3R9 or OPS Form 3209-5/6. Subtract the reference Xenon worth from the estimated Xenon worth to obtain the Xenon defect. (The value is negative if estimated Xenon is greater than reference Xenon.)

5.6 Item 4. Repeat Step 5.5 for determining Samarium worth. (Use SP 3R7/3R9 or OPS Form 3209-7).

5.7 Item 5. Determine the integrated rod worths at the estimated and reference conditions. Subtract the reference rod worth from the estimated rod worth to obtain the rod worth defect. (Use OPS Form 3209-8, 9, or 10.) The value is Negative if the estimated position is below the reference position.

5.8 Item 6. Determine the Differential Boron Worth at the present boron concentration. Multiply by the change in boron concentration from the present condition to the reference condition to obtain the boron defect. The value is Negative if the present boron concentration is greater than the reference boron concentration. (Use OPS Form 3209-11.)

5.9 Item 7. Add all the reactivity defects. Divide by the differential boron worth to obtain the boron equivalent. Use OPS Form 3209-13 to compensate for burnup if the difference between the reference burnup and present burnup is greater than 200 MWD/MTU. Add this to the boron equivalent of the summation of the reactivity defect to obtain the boron change required to go critical at the desired rod position. If the value is positive, dilution is required (i.e. addition of + reactivity). If the value is negative, boration is required (i.e. negative reactivity should be added).

5.10 Item 8. Determine the ± 900 pcm limits around the ECC by using the rod worth at the ECC. Add and subtract 900 pcm from the ECC and obtain the equivalent rod position using OPS Form 3209-8, 9, or 10. Do not allow the maximum rod insertion to exceed the 0% power Rod Insertion Limit (RIL) or the minimum rod insertion to exceed any rod withdrawal limits imposed unless physics tests are in progress.

NOTE: If Steps 5.1 through 5.10 were performed to calculate an ECC by selecting control rod position and varying boron skip to Step 5.21.

If it is desired to select a boron concentration and vary control rod position perform Steps 5.11 through 5.20 and complete OPS Form 3209A-2. Normally Steps 5.11 through 5.20 should be done if it is desired to go critical on the existing RCS boron concentration.

5.11 Enter reference critical position data. This should be the most recent steady state data available (supplied by Reactor Engineering) on OPS Form 3209-15.

CAUTION: Ensure proper use of signs (+ or -) throughout this calculation.

5.12 Enter Estimated Conditions at Time of Criticality. Data for Date, Time, Temperature, and Pressure, are estimated. The Burnup is entered at the existing conditions. The boron concentration should be the desired critical boron concentration (normally the existing boron concentration).

5.13 Item 1. Determine the power defect at the Reference Power. The number on the curve is negative but a positive number should be entered on OPS Form 3209A-2 (Use OPS Form 3209-3).

5.14 Moderator Defects - Item 2.

NOTE: The power defect curve assumes the reactor is operating with Tavg equal to Tref ± 2°F. If Tavg is not equal to Tref ± 2°F it will be necessary to perform Step 5.14.1.

5.14.1 Determine the difference between T average at the reference condition and Tref at the reference condition. Multiply this difference by the Moderator Temperature Coefficient (MTC) (OPS Form 3209-4/5) at the reference condition to determine the moderator defect. (This number will be negative if Tavg > Tref and MTC is negative.)

NOTE: If the estimated critical temperature is other than 557°F it will be necessary to perform Step 5.4.2.

5.14.2 Determine the difference between Tavg at the estimated condition and 557°F. Multiply by the MTC (OPS Form 3209-4/5) to determine Moderator Defect at the Estimated Condition. (The Value is Negative if Tavg > 557°F and MTC negative.)

NOTE: If the computer is not available, use OPS Form 3209-5 and OPS Form 3209-6 to obtain Xenon worth if the reactor was shutdown from an equilibrium Xenon state. If the reactor was not at or near an equilibrium Xenon state prior to shutdown, Reactor Engineering will determine Xenon worth.

5.15 Item 3. Determine the Xenon worth at the estimated time of criticality (use SP 3R7/3R9 or OPS Form 3209-5/6. Subtract the reference Xenon worth from the estimated Xenon worth to obtain the Xenon defect. (The value is Negative if estimated Xenon is greater than reference Xenon.)

5.16 Item 4. Repeat Step 5.15 for determining Samarium worth. (Use SP 3R7/3R9 or OPS Form 3209-7).

5.17 Item 5. Determine the Differential Boron Worth at the present boron concentration. Multiply by the change in boron concentration from the present condition to the reference condition to obtain the boron defect. The value is Negative if the present boron concentration is greater than the reference boron concentration. (Use OPS Form 3209-11.)

5.18 Item 6. Add all the reactivity defects. Correct for burnup if the difference between the reference burnup and present burnup is greater than 200 MWD/MTU.

5.19 Item 7. Multiply the total Reactivity Defect by -1 and add it to the control rods worth inserted at the reference condition. If the resulting value is positive dilute the RCS or wait for Xenon to decay.

5.20 Item 8. If the value determined in Step 5.19 is negative determine critical rod height from OPS Form 3209-8. Determine the ± 900 pcm limits around the ECC by using the rod worth at the ECC. Add and subtract 900 pcm from the ECC and obtain the equivalent rod position using OPS Form 3209-8, 9, or 10. Do not allow the maximum rod insertion to exceed the 0% power Rod Insertion Limit (RIL) or the minimum rod insertion to exceed any rod withdrawal limits imposed unless physics tests are in progress.

5.21 Give OPS Form 3209A-1/OPS Form 3209A-2 completed through item 8 to another licensed operator for review and approval.

5.22 After going critical, stabilize power in the zero power range (approximately 10^{-8} amp on the intermediate range) and record actual critical data.

5.23 Forward completed OPS Form 3209A-1/3209A-2 to Reactor Engineering for review.

6. CHECKOFF LISTS

6.1 OPS Form 3209A-1, Estimated Critical Condition, Fixed Rods.
6.2 OPS Form 3209A-2, Estimated Critical Condition, Fixed Boron.
6.3 OPS Form 3209-3, Power Defect
6.4 OPS Form 3209-4, Moderator Temperature Coefficient
6.5 OPS Form 3209-5, Xenon Worth At Steady State
6.6 OPS Form 3209-6, Xenon Worth After Trip
6.7 OPS Form 3209-7, Samarium Worth After Trip
6.8 OPS Form 3209-8, Integral Rod Worth In Overlap HFP, EQ Xenon, BOL
6.9 OPS Form 3209-9, Integral Rod Worth Bank D only HFP, EQ Xenon, BOL
6.10 OPS Form 3209-10, Integral Rod Worth HZP, Xenon Free, BOL
6.11 OPS Form 3209-11, Boron Worth
6.12 OPS Form 3209-12, Boron Required For Shutdown
6.13 OPS Form 3209-13, Boron Rundown
6.14 OPS Form 3209-14, Core Data
6.15 OPS Form 3209-15, ECC Reference Data

DM:jlm

90C 11/27/85 3-85-189

FORM APPROVED BY UNIT 3 SUPERINTENDENT | EFFECTIVE DATE | PORC MTG. NO.

CALCULATED BY ____ DATE ____ APPROVED BY ____ DATE ____

ESTIMATED CRITICAL CONDITION - FIXED ROD POSITION
REFERENCE CRITICAL DATA

DATE ~~12/15/87~~ 12/17/87 TIME 0100

RCS TAVG 587 °F

RCS PRESSURE 2250 PSIA

POWER 100 %

RCS BORON 900 ppm BURNUP 4015 MWD/MTU

CONTROL BANK D 180 STEPS CONTROL BANK C 228 STEPS

OTHER 100% power maintained for last 10 days

XENON ____ pcm (-)

SAMARIUM ____ pcm (-)

LAST SHUTDOWN TIME 1550 DATE 11/17/87

ESTIMATED STATUS AT CRITICALITY

DATE 12/15/87 TIME 2100

TAVG 557 °F RCS PRESS 2250 psia

BORON (PRESENT) 1200 ppm BURNUP (PRESENT) 4015 MWD/MTU

DESIRED CRITICAL POSITION

BANK C AT 150 STEPS

1. POWER DEFECT (OPS Form 3209-3)
   (STEP 5.3)
   1.1 Reference Power ____ %
   1.2 pcm at Reference Power ____ pcm

2. MODERATOR DEFECT. (OPS Form 3209-4)
   (STEP 5.4.1)
   2.1 Tavg (at reference conditions) 587 °F
   2.2 Tref (at reference conditions) 587 °F
   2.3 MTC (at reference conditions) N/A pcm/°F
   2.4 Moderator Defect (at reference conditions)
   (2.1 - 2.2) X MTC = N/A pcm

(STEP 5.4.2)

2.5 Estimated Tavg ___557___ °F

2.6 MTC (at estimated conditions) ___N/A___ pcm/°F

2.7 Moderator Defect at estimated conditions

(2.5 - 557) x MTC = ___N/A___ pcm

---

3. XENON DEFECT (Computer or OPS Form 3209-5/6) circle one.

(STEP 5.5)

3.1 Estimated Xenon ______ pcm (-)

3.2 Reference Xenon ______ pcm (-)

3.3 Xenon Defect (3.1 - 3.2) ______ pcm

---

4. SAMARIUM DEFECT (Computer or OPS Form ~~3209-7~~) circle one.
3209-14

(STEP 5.6)

4.1 Estimated Samarium ___685___ pcm (-)

4.2 Reference Samarium ______ pcm (-)

4.3 Samarium Defect (4.1 - 4.2) ______ pcm

(STEP 5.7)

5. INTEGRATED ROD WORTH (OPS Form 3209-8/~~9/10~~) Use 3209-8 ONLY

5.1 Estimated Rod Worth ______ pcm (-)

5.2 Reference Rod Worth ______ pcm (-)

5.3 Rod Worth Defect (5.1 - 5.2) ______ pcm

---

6. BORON DEFECT

(STEP 5.8)

6.1 Present Boron Concentration ______ ppm

6.2 Reference Boron Concentration ______ ppm

6.3 Boron Worth (OPS Form 3209-1) ______ pcm/ppm (-)

6.4 Boron Defect (6.1 - 6.2) x 6.3 ______ pcm

---

7. CALCULATIONS

(STEP 5.9)

7.1 Sum Defects (1.2 + 2.4 + 2.7 + 3.3 + 4.3 + 5.3 + 6.4) _____ pcm

7.2 Boron Equivalent of Defects (7.1 ÷ 6.3) ____________ ppm

7.3 Nominal PPM at Reference BU ____N/A____ ppm

7.4 Nominal PPM at Present BU ____N/A____ ppm

7.5 Burnup Change (7.3 - 7.4) ____N/A____ ppm (+)

7.6 Boron Change to Go Critical (7.2 + 7.5) ____________ ppm

7.7 Critical Boron Concentration (6.1 - 7.6) ____________ ppm

---

8. LIMITS ON CONTROL ROD POSITION.

(STEP 5.10)

8.1 Rod Worth at ECP

Bank _____ at _____ steps ________ pcm (-)

8.2 Rod Worth at Minimum Insertion

(8.1 + 900 pcm) ________ pcm

8.3 Rod Position at Minimum Insertion

Bank _____ at _____ steps

8.4 Rod Worth at Maximum Insertion

(8.1 - 900 pcm) ________ pcm (-)

8.5 Rod Position at Maximum Insertion (Cannot be below 0% power rod insertion limit)

Bank _____ at _____ steps

ACTUAL CRITICAL DATA (STEP 5.22)

DATE ___N/A___ TIME ___N/A___

Rod Position Control Bank D at ___N/A___ Steps

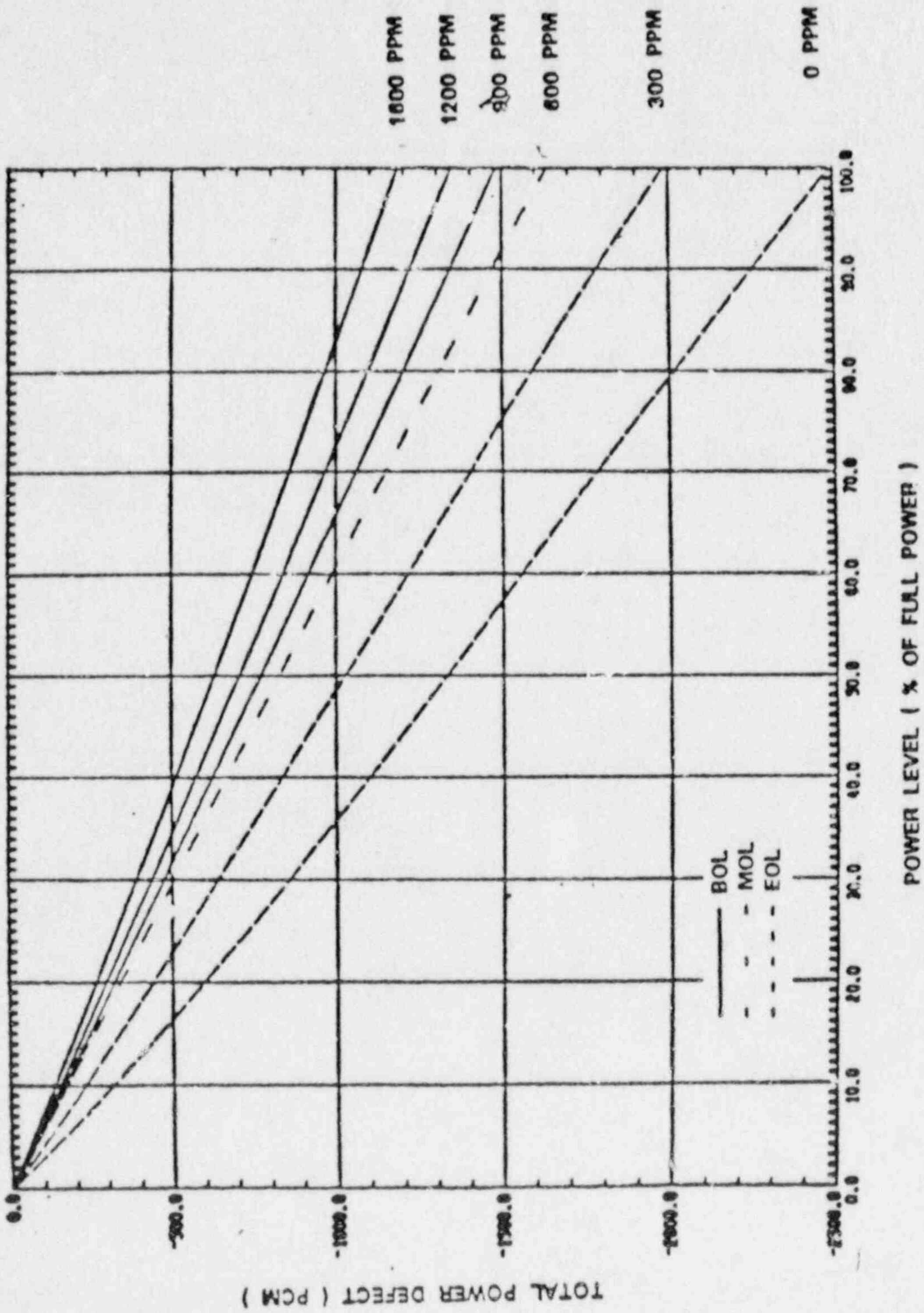
Control Bank C at ___N/A___ Steps

Other: ___N/A___

Tavg Loop 1 T411A _N/A_ °F Loop 2 T421A _N/A_ °F

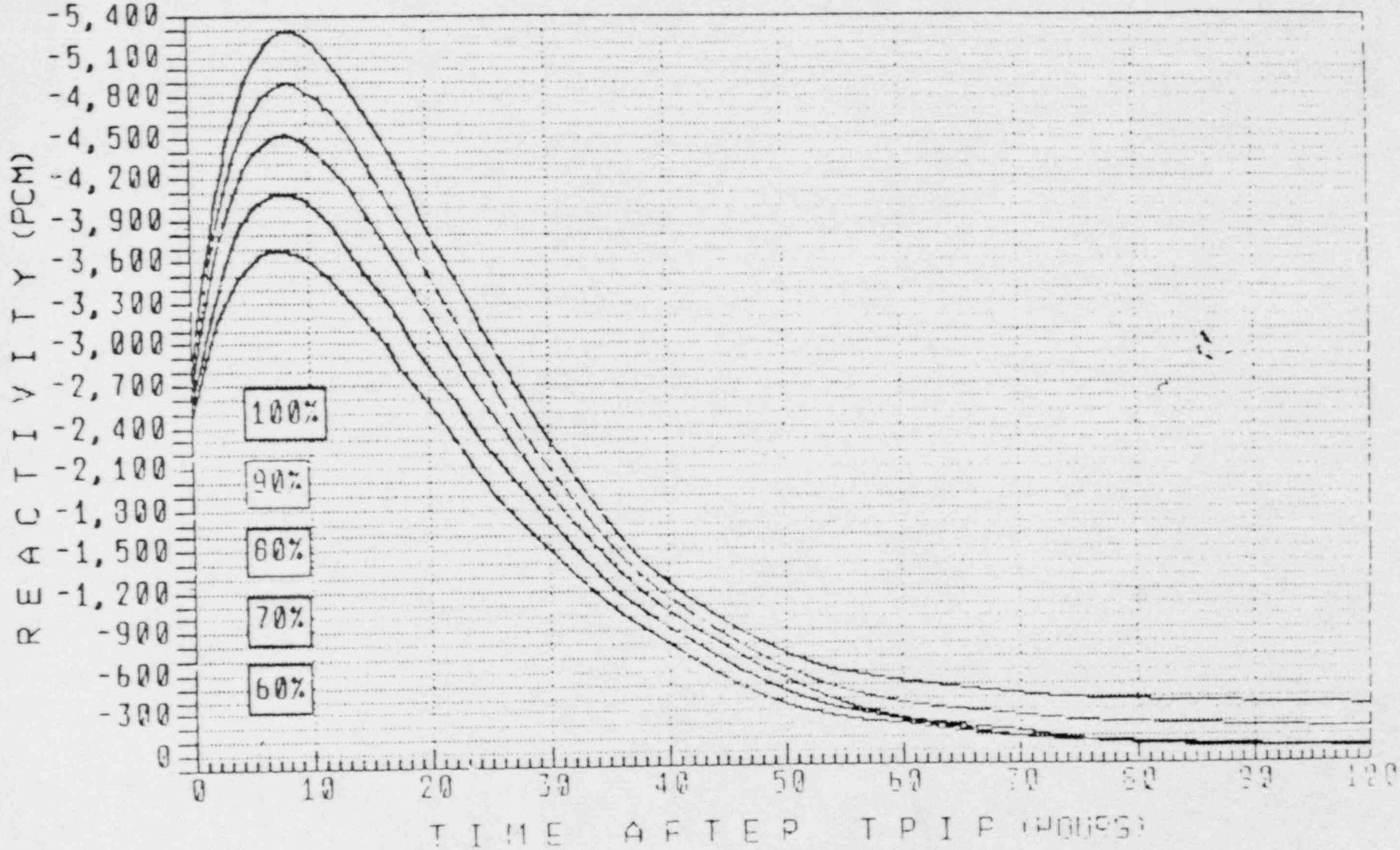
Loop 3 T431A _N/A_ °F Loop 4 T441A _N/A_ °F

[signature]	11/27/85	3-85-243
FORM APPROVED BY UNIT 3 SUPERINTENDENT	EFFECTIVE DATE	PORC MTG. NO.



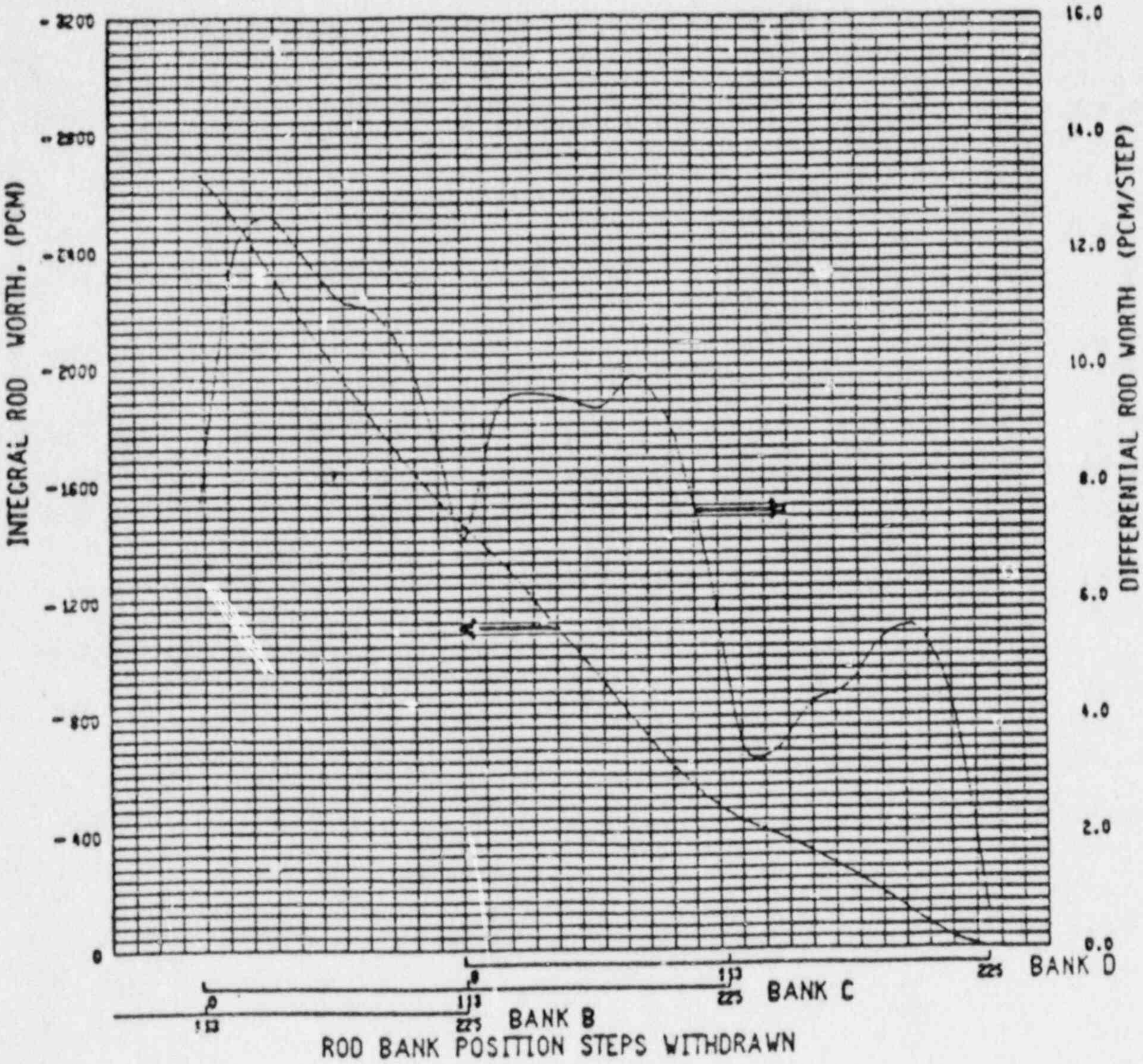
POWER DEFECT VS. POWER LEVEL AT BOL, MOL, AND EOL, CYCLE 1

Reactivity Insertion Due To Xenon
BOL 60% to 100%
Millstone Unit 3

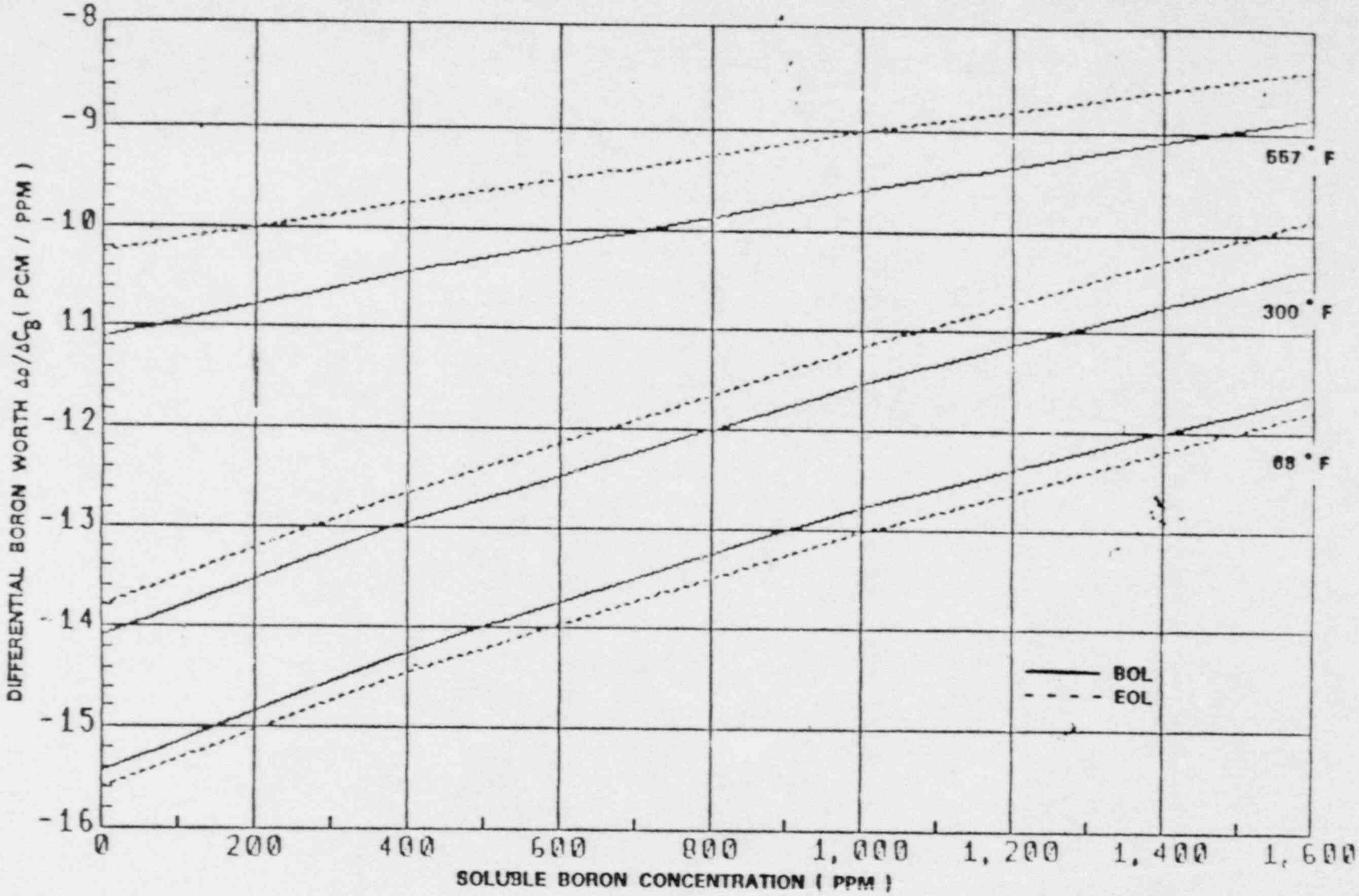


3-10-87 | 3-87-32

FORM APPROVED BY UNIT 3 SUPERINTENDENT | EFFECTIVE DATE | PORC MTG. NO.



DIFFERENTIAL AND INTEGRAL ROD WORTH VS. STEPS WITHDRAWN
BANKS D, C, B MOVING WITH 113 STEP OVERLAP
EOL, HFP, EQUILIBRIUM XENON CONDITIONS



DIFFE.'ENTIAL BORON WORTH VS. BORON CONCENTRATION AT BOL AND EOL CYCLE 1

FORM APPROVED BY UNIT 3 SUPERINTENDENT

EFFECTIVE DATE 11/27/85

PORC MTG. NO. 3-85-263

FORM APPROVED BY UNIT 3 SUPERINTENDENT — 11/27/85 EFFECTIVE DATE — 3-85-243 PORC MTG NO.

MISCELLANEOUS CORE DATA

Total Shutdown and Control Bank Worth Hot Zero Power Xenon Free	9010 pcm
Worst Case Stuck Rod at BOL is F-2	1110 pcm
Equilibrium Samarium after 25 days of operation at a steady state power level	-600 pcm
$\bar{\beta}_{eff}$ at BOL	.0069
Refueling Boron Concentration to maintain $k_{eff} < .95$ with all rods inserted.	1472 ppm
Refueling Boron Concentration to maintain $k_{eff} < .95$ with all rods out	2000 ppm

ATTACHMENT 3

CORE HEAT BALANCE

Average over Measurement Interval

A.	Pzr Pressure	2250	psia
B.	RCS Loop 3 Tc	557	°F
C.	Letdown Flow (CHS-F132)	75	gpm
D.	Charging Flow (CHS-F121)	87	gpm
E.	Charging Pressure (CHS-P120)	2500	psia
F.	Charging Temperature (CHS-T126)	510	°F
G.	VCT Temperature (CHS-T116)	100	°F

		S/G 1	S/G 2	S/G 3	S/G 4
H.	Steam Pressure (PSIA)	1100		SAME	
I.	Blowdown Flow (gpm)	50		AS	
J.	Feed Temperature (°F)	420		S/G	
K.	Feed Pressure (PSIA)	1400		1	

M.	Letdown Enthalpy (From A & B)		BTU/lbm
N.	Charging Enthalpy (From E & F)		BTU/lbm
P.	Charging Specific Volume (From E & ~~F~~ G)		ft³/lbm
Q.	Charging Flow Correction Factor		

| ch
| 1

$$\frac{.12716 \times 2.0 \times 10^{-5} \times (\text{CHS-T116}) + .9983}{\sqrt{V} \times 1.0011} = \text{N/A}$$

| ch
| 2

Where V is the specific volume of the Charging Fluid (CHS-T116 and CHS-P120).

R.	Corrected Charging Flow (DxQ)	N/A	gpm
S.	Charging Flow (60xR)/(7.48xP)	N/A	lbm/hr
T.	CVCS Heat Loss Sx(M-N)/3412141	4.69	MWT
U.	RCP Seal Flow (Total)-12 gpm	N/A	gpm
V.	Seal Enthalpy (From E & G)		BTU/lbm
W.	Seal Specific Volume (From E & G)	N/A	ft³/lbm
X.	Corrected Seal Flow (60xU)/(7.48xW)	N/A	lbm/hr
Y.	Seal Heat Loss (X)x(M-V)/3412141	4.74	MWT

		S/G 1	S/G 2	S/G 3	S/G 4
Z.	Feed Enthalpy BTU/lbm (From J & K)				
AA.	SAT STM Enthalpy BTU/lbm (From H)				
BB.	SAT Water Enthalpy BTU/lbm (From H)				
CC.	S/G Enthalpy (AA)				
DD.	S/G Δ Enthalpy (CC-Z)			SAME	
EE.	Actual Feed Flow	1.9 E+6		AS	
FF.	S/G Power [(EE)(DD)] BTU/hr			S/G	
GG.	Sat Water Spec. Vol. (From H)			1	
HH.	Blowdown Flow (60xI)/(7.48 x GG)	N/A			
JJ.	Blowdown Loss [~~CC~~ HH (DD-BB)]	-0-			
KK.	Total S/G Power (FF - JJ) BTU/hr				
LL.	Total NSSS Power Σ (KK) BTU/hr				

MM. LL/3412141 = ______________ MWT

NN. Net RCP Heat Input = 16 MWT

PP. TOTAL CORE POWER (MM-NN+T+Y) = ______________ MWT

CORE POWER in % (PP)(100)/3411 = ___________ %

ATTACHMENT 4

Table 3
Properties of superheated steam and compressed water (temperature and pressure)

Abs press. lb/sq in. (sat. temp)		Temperature, F 100	200	300	400	500	600	700	800	900	1000	1100	1200	1300	1400	1500
	v	0.0161	392.5	452.3	511.9	571.5	631.1	690.7								
1	h	68.00	1150.2	1195.7	1241.8	1288.6	1336.1	1384.5								
(101.74)	s	0.1295	2.0509	2.1152	2.1722	2.2237	2.2708	2.3144								
	v	0.0161	78.14	90.24	102.24	114.21	126.15	138.08	150.01	161.94	173.86	185.78	197.70	209.62	221.53	233.45
5	h	68.01	1148.6	1194.8	1241.3	1288.2	1335.9	1384.3	1433.6	1483.7	1534.7	1586.7	1639.6	1693.3	1748.0	1803.5
(162.24)	s	0.1295	1.8716	1.9369	1.9943	2.0460	2.0932	2.1369	2.1776	2.2159	2.2521	2.2866	2.3194	2.3'09	2.3811	2.4101
	v	0.0161	38.84	44.98	51.03	57.04	63.03	69.00	74.98	80.94	86.91	92.87	98.84	104.80	110.76	116.72
10	h	68.02	1146.6	1193.7	1240.6	1287.8	1335.5	1384.0	1433.4	1483.5	1534.6	1586.6	1639.5	1693.3	1747.9	1803.4
(193.21)	s	0.1295	1.7928	1.8593	1.9173	1.9692	2.0166	2.0603	2.1011	2.1394	2.1757	2.2101	2.2430	2.2744	2.3046	2.3337
	v	0.0161	0.0166	29.899	33.963	37.985	41.986	45.978	49.964	53.946	57.926	61.905	65.882	69.858	73.833	77.807
15	h	68.04	168.09	1192.5	1239.9	1287.3	1335.2	1383.8	1433.2	1483.4	1534.5	1586.5	1639.4	1693.2	1747.8	1803.4
(213.03)	s	0.1295	0.2940	1.8134	1.8720	1.9242	1.9717	2.0155	2.0563	2.0946	2.1309	2.1653	2.1982	2.2297	2.2599	2.2890
	v	0.0161	0.0166	22.356	25.428	28.457	31.466	34.465	37.458	40.447	43.435	46.420	49.405	52.388	55.370	58.352
20	h	68.05	168.11	1191.4	1239.2	1286.9	1334.9	1383.5	1432.9	1483.2	1534.3	1586.3	1639.3	1693.1	1747.8	1803.3
(227.96)	s	0.1295	0.2940	1.7805	1.8397	1.8921	1.9397	1.9836	2.0244	2.0628	2.0991	2.1336	2.1665	2.1979	2.2282	2.2572
	v	0.0161	0.0166	11.036	12.624	14.165	15.685	17.195	18.699	20.199	21.697	23.194	24.689	26.183	27.676	29.168
40	h	68.10	168.15	1186.6	1236.4	1285.0	1333.6	1382.5	1432.1	1482.5	1533.7	1585.8	1638.8	1992.7	1747.5	1803.0
(267.25)	s	0.1295	0.2940	1.6992	1.7608	1.8143	1.8624	1.9065	1.9476	1.9860	2.0224	2.0569	2.0899	2.1224	2.1516	2.1807
	v	0.0161	0.0166	7.257	8.354	9.400	10.425	11.438	12.446	13.450	14.452	15.452	16.450	17.448	18.445	19.441
60	h	68.15	168.20	1181.6	1233.5	1283.2	1332.3	1381.5	1431.3	1481.8	1533.2	1585.3	1638.4	1692.4	1747.1	1802.8
(292.71)	s	0.1295	0.2939	1.6492	1.7134	1.7681	1.8168	1.8612	1.9024	1.9410	1.9774	2.0120	2.0450	2.0765	2.1068	2.1359
	v	0.0161	0.0166	0.0175	6.218	7.018	7.794	8.560	9.319	10.075	10.829	11.581	12.331	13.081	13.829	14.577
80	h	68.21	168.24	269.74	1230.5	1281.3	1330.9	1380.5	1430.5	1481.1	1532.6	1584.9	1638.0	1692.0	1746.8	1802.5
(312.04)	s	0.1295	0.2939	0.4371	1.6790	1.7349	1.7842	1.8289	1.8702	1.9089	1.9454	1.9800	2.0131	2.0446	2.0750	2.1041
	v	0.0161	0.0166	0.0175	4.935	5.588	6.216	6.833	7.443	8.050	8.655	9.258	9.860	10.460	11.060	11.659
100	h	68.26	168.29	269.77	1227.4	1279.3	1329.6	1379.5	1429.7	1480.4	1532.0	1584.4	1637.6	1691.6	1746.5	1802.2
(327.82)	s	0.1295	0.2939	0.4371	1.6516	1.7088	1.7586	1.8036	1.8451	1.8839	1.9205	1.9552	1.9883	2.0199	2.0502	2.0794
	v	0.0161	0.0166	0.0175	4.0786	4.6341	5.1637	5.6831	6.1928	6.7006	7.2060	7.7096	8.2119	8.7130	9.2134	9.7130
120	h	68.31	168.33	269.81	1224.1	1277.4	1328.1	1378.4	1428.8	1479.8	1531.4	1583.9	1637.1	1691.3	1746.2	1802.0
(341.27)	s	0.1295	0.2939	0.4371	1.6286	1.6872	1.7376	1.7829	1.8246	1.8635	1.9001	1.9349	1.9680	1.9996	2.0300	2.0592
	v	0.0161	0.0166	0.0175	3.4661	3.9526	4.4119	4.8585	5.2995	5.7364	6.1709	6.6036	7.0349	7.4652	7.8946	8.3233
140	h	68.37	168.38	269.85	1220.8	1275.3	1326.8	1377.4	1428.0	1479.1	1530.8	1583.4	1636.7	1690.9	1745.9	1801.7
(353.04)	s	0.1295	0.2939	0.4370	1.6085	1.6686	1.7196	1.7652	1.8071	1.8461	1.8828	1.9176	1.9508	1.9825	2.0129	2.0421
	v	0.0161	0.0166	0.0175	3.0060	3.4413	3.8480	4.2420	4.6295	5.0132	5.3945	5.7741	6.1522	6.5293	6.9055	7.2811
160	h	68.42	168.42	269.89	1217.4	1273.3	1325.4	1376.4	1427.2	1478.4	1530.3	1582.9	1636.3	1690.5	1745.6	1801.4
(363.55)	s	0.1294	0.2938	0.4370	1.5906	1.6522	1.7039	1.7499	1.7919	1.8310	1.8678	1.9027	1.9359	1.9676	1.9980	2.0273
	v	0.0161	0.0166	0.0174	2.6474	3.0433	3.4093	3.7621	4.1084	4.4505	4.7907	5.1289	5.4657	5.8014	6.1363	6.4704
180	h	68.47	168.47	269.92	1213.8	1271.2	1324.0	1375.3	1426.3	1477.7	1529.7	1582.4	1635.9	1690.2	1745.3	1801.2
(373.08)	s	0.1294	0.2938	0.4370	1.5743	1.6376	1.6900	1.7362	1.7784	1.8176	1.8545	1.8894	1.9227	1.9545	1.9849	2.0142
	v	0.0161	0.0166	0.0174	2.3598	2.7247	3.0583	3.3783	3.6915	4.0008	4.3077	4.6128	4.9165	5.2191	5.5209	5.8219
200	h	68.52	168.51	269.96	1210.1	1269.0	1322.6	1374.3	1425.5	1477.0	1529.1	1581.9	1635.4	1689.8	1745.0	1800.9
(381.80)	s	0.1294	0.2938	0.4369	1.5593	1.6242	1.6776	1.7239	1.7663	1.8057	1.8426	1.8776	1.9109	1.9427	1.9732	2.0025
	v	0.0161	0.0166	0.0174	0.0186	2.1504	2.4662	2.6872	2.9410	3.1909	3.4382	3.6837	3.9278	4.1709	4.4131	4.6546
250	h	68.66	168.63	270.05	375.10	1263.5	1319.0	1371.6	1423.4	1475.3	1527.6	1580.6	1634.4	1688.9	1744.2	1800.2
(400.97)	s	0.1294	0.2937	0.4368	0.5667	1.5951	1.6502	1.6976	1.7405	1.7801	1.8173	1.8524	1.8858	1.9177	1.9482	1.9776
	v	0.0161	0.0166	0.0174	0.0186	1.7665	2.0044	2.2263	2.4407	2.6509	2.8585	3.0643	3.2688	3.4721	3.6746	3.8764
300	h	68.79	168.74	270.14	375.15	1257.7	1315.2	1368.9	1421.3	1473.6	1526.2	1579.4	1633.3	1688.0	1743.4	1799.6
(417.35)	s	0.1294	0.2937	0.4307	0.5665	1.5703	1.6274	1.6758	1.7192	1.7591	1.7964	1.8317	1.8652	1.8972	1.9278	1.9572
	v	0.0161	0.0166	0.0174	0.0186	1.4913	1.7028	1.8970	2.0832	2.2652	2.4445	2.6219	2.7980	2.9730	3.1471	3.3205
350	h	68.92	168.85	270.24	375.21	1251.5	1311.4	1366.2	1419.2	1471.8	1524.7	1578.2	1632.3	1687.1	1742.6	1798.9
(431.73)	s	0.1293	0.2936	0.4367	0.5664	1.5483	1.6077	1.6571	1.7009	1.7411	1.7787	1.8141	1.8477	1.8798	1.9105	1.9400
	v	0.0161	0.0166	0.0174	0.0162	1.2841	1.4763	1.6499	1.8151	1.9759	2.1339	2.2901	2.4450	2.5987	2.7515	2.9037
400	h	69.05	168.97	270.33	375.27	1245.1	1307.4	1363.4	1417.0	1470.1	1523.3	1576.9	1631.2	1686.2	1741.9	1798.2
(444.60)	s	0.1293	0.2935	0.4366	0.5663	1.5282	1.5901	1.6406	1.6850	1.7255	1.7532	1.7988	1.8325	1.8647	1.8955	1.9250
	v	0.0161	0.0166	0.0174	0.0186	0.9919	1.1584	1.3037	1.4397	1.5708	1.6992	1.8256	1.9507	2.0746	2.1977	2.3200
500	h	69.32	169.19	270.51	375.38	1231.2	1299.1	1357.7	1412.7	1466.6	1520.3	1574.4	1629.1	1684.4	1740.3	1796.9
(467.01)	s	0.1292	0.2934	0.4364	0.5660	1.4921	1.5505	1.6123	1.6578	1.6990	1.7371	1.7730	1.8069	1.8393	1.8702	1.8998

Table 3
Properties of superheated steam and compressed water (temperature and pressure)

Abs press. lb/sq in. (sat. temp)		Temperature, F 100	200	300	400	500	600	700	800	900	1000	1100	1200	1300	1400	1500
600 (486.20)	v	0.0161	0.0166	0.0174	0.0186	0.7944	0.9456	1.0726	1.1892	1.3008	1.4093	1.5160	1.6211	1.7252	1.8284	1.9309
	h	69.58	169.42	270.70	375.49	1215.9	1290.3	1351.8	1408.3	1463.0	1517.4	1571.9	1627.0	1682.6	1738.8	1795.6
	s	0.1292	0.2933	0.4362	0.5657	1.4590	1.5329	1.5844	1.6351	1.6769	1.7155	1.7517	1.7859	1.8184	1.8494	1.8792
700 (503.08)	v	0.0161	0.0166	0.0174	0.0186	0.0204	0.7928	0.9072	1.0102	1.1078	1.2023	1.2948	1.3858	1.4757	1.5647	1.6530
	h	69.84	169.65	270.89	375.61	487.93	1281.0	1345.6	1403.7	1459.4	1514.4	1569.4	1624.8	1680.7	1737.2	1794.3
	s	0.1291	0.2932	0.4360	0.5655	0.6889	1.5090	1.5673	1.6154	1.6580	1.6970	1.7335	1.7679	1.8006	1.8318	1.8617
800 (518.21)	v	0.0161	0.0166	0.0174	0.0186	0.0204	0.6774	0.7828	0.8759	0.9631	1.0470	1.1289	1.2093	1.2885	1.3669	1.4446
	h	70.11	169.88	271.07	375.73	487.88	1271.1	1339.2	1399.1	1455.8	1511.4	1566.9	1622.7	1678.9	1735.0	1792.9
	s	0.1290	0.2930	0.4358	0.5652	0.6885	1.4869	1.5484	1.5980	1.6413	1.6807	1.7175	1.7522	1.7851	1.8164	1.8464
900 (531.95)	v	0.0161	0.0166	0.0174	0.0186	0.0204	0.5869	0.6858	0.7713	0.8504	0.9262	0.9998	1.0720	1.1430	1.2131	1.2825
	h	70.37	170.10	271.26	375.84	487.83	1260.6	1332.7	1394.4	1452.2	1508.5	1564.4	1620.6	1677.1	1734.1	1791.6
	s	0.1290	0.2929	0.4357	0.5649	0.6881	1.4659	1.5311	1.5822	1.6263	1.6662	1.7033	1.7382	1.7713	1.8028	1.8329
1000 (544.58)	v	0.0161	0.0166	0.0174	0.0186	0.0204	0.5137	0.6080	0.6875	0.7603	0.8295	0.8966	0.9622	1.0266	1.0901	1.1529
	h	70.63	170.33	271.44	375.96	487.79	1249.3	1325.9	1389.6	1448.5	1504.4	1561.9	1618.4	1675.3	1732.5	1790.3
	s	0.1289	0.2928	0.4355	0.5647	0.6876	1.4457	1.5149	1.5677	1.6126	1.6530	1.6905	1.7256	1.7589	1.7905	1.8207
1100 (556.28)	v	0.0161	0.0166	0.0174	0.0185	0.0203	0.4531	0.5440	0.6188	0.6865	0.7505	0.8121	0.8723	0.9313	0.9894	1.0468
	h	70.90	170.56	271.63	376.08	487.75	1237.3	1318.8	1384.7	1444.7	1502.4	1559.4	1616.3	1673.5	1731.0	1789.0
	s	0.1289	0.2927	0.4353	0.5644	0.6872	1.4259	1.4996	1.5542	1.6000	1.6410	1.6787	1.7141	1.7475	1.7793	1.8097
1200 (567.19)	v	0.0161	0.0166	0.0174	0.0185	0.0203	0.4016	0.4905	0.5615	0.6250	0.6845	0.7418	0.7974	0.8519	0.9055	0.9584
	h	71.16	170.78	271.82	376.20	487.72	1224.2	1311.5	1379.7	1440.9	1499.4	1556.9	1614.2	1671.6	1729.4	1787.6
	s	0.1288	0.2926	0.4351	0.5642	0.6868	1.4061	1.4851	1.5415	1.5883	1.6298	1.6679	1.7035	1.7371	1.7691	1.7996
1400 (587.07)	v	0.0161	0.0166	0.0174	0.0185	0.0203	0.3176	0.4059	0.4712	0.5282	0.5809	0.6311	0.6798	0.7272	0.7737	0.8195
	h	71.68	171.24	272.19	376.44	487.65	1194.1	1296.1	1369.3	1433.2	1493.2	1551.8	1609.9	1668.0	1726.3	1785.0
	s	0.1287	0.2923	0.4348	0.5636	0.6859	1.3652	1.4575	1.5182	1.5670	1.6096	1.6484	1.6845	1.7185	1.7508	1.7815
1600 (604.87)	v	0.0161	0.0166	0.0173	0.0185	0.0202	0.0236	0.3415	0.4032	0.4555	0.5031	0.5482	0.5915	0.6336	0.6748	0.7153
	h	72.21	171.69	272.57	376.69	487.60	616.77	1279.4	1358.5	1425.2	1486.9	1546.6	1605.6	1664.3	1723.2	1782.3
	s	0.1286	0.2921	0.4344	0.5631	0.6851	0.8129	1.4312	1.4968	1.5478	1.5916	1.6312	1.6678	1.7022	1.7344	1.7657
1800 (621.02)	v	0.0160	0.0165	0.0173	0.0185	0.0202	0.0235	0.2906	0.3500	0.3988	0.4426	0.4836	0.5229	0.5609	0.5980	0.6343
	h	72.73	172.15	272.95	376.93	487.56	615.58	1261.1	1347.2	1417.1	1480.6	1541.1	1601.2	1660.7	1720.1	1779.7
	s	0.1284	0.2918	0.4341	0.5626	0.6843	0.8109	1.4054	1.4768	1.5302	1.5753	1.6156	1.6528	1.6876	1.7204	1.7516
2000 (635.80)	v	0.0160	0.0165	0.0173	0.0184	0.0201	0.0233	0.2488	0.3072	0.3534	0.3942	0.4320	0.4680	0.5027	0.5365	0.5695
	h	73.26	172.60	273.32	377.19	487.53	614.48	1240.9	1353.4	1408.7	1474.1	1536.2	1596.9	1657.0	1717.0	1777.1
	s	0.1283	0.2916	0.4337	0.5621	0.6834	0.8091	1.3794	1.4578	1.5128	1.5603	1.6014	1.6391	1.6743	1.7075	1.7339
2500 (668.11)	v	0.0160	0.0165	0.0173	0.0184	0.0200	0.0230	0.1681	0.2293	0.2712	0.3068	0.3390	0.3692	0.3980	0.4259	0.4529
	h	74.57	173.74	274.27	377.82	487.50	612.08	1176.7	1303.4	1386.7	1457.5	1522.9	1585.9	1647.8	1709.2	1770.4
	s	0.1280	0.2910	0.4329	0.5609	0.6815	0.8048	1.3076	1.4129	1.4766	1.5269	1.5703	1.6094	1.6456	1.6796	1.7116
3000 (695.33)	v	0.0160	0.0165	0.0172	0.0183	0.0200	0.0228	0.0982	0.1759	0.2161	0.2484	0.2770	0.3033	0.3282	0.3522	0.3753
	h	75.88	174.88	275.22	378.47	487.52	610.08	1060.5	1267.0	1363.2	1440.2	1509.4	1574.8	1638.5	1701.4	1761.8
	s	0.1277	0.2904	0.4320	0.5597	0.6796	0.8009	1.1966	1.3692	1.4429	1.4976	1.5434	1.5841	1.6214	1.6561	1.6888
3200 (705.08)	v	0.0160	0.0165	0.0172	0.0183	0.0199	0.0227	0.0335	0.1588	0.1987	0.2301	0.2576	0.2827	0.3065	0.3291	0.3510
	h	76.4	175.3	275.6	378.7	487.5	609.4	800.8	1250.9	1353.4	1433.1	1503.8	1570.3	1634.8	1698.3	1761.2
	s	0.1276	0.2902	0.4317	0.5592	0.6788	0.7994	0.9708	1.3515	1.4300	1.4866	1.5335	1.5749	1.6126	1.6477	1.6806
3500	v	0.0160	0.0164	0.0172	0.0183	0.0199	0.0225	0.0307	0.1364	0.1764	0.2066	0.2326	0.2563	0.2784	0.2995	0.3198
	h	77.2	176.0	276.2	379.1	487.6	608.4	779.4	1224.6	1338.2	1422.2	1495.5	1563.3	1629.2	1693.6	1757.2
	s	0.1274	0.2899	0.4312	0.5585	0.6777	0.7973	0.9508	1.3242	1.4112	1.4709	1.5194	1.5618	1.6002	1.6358	1.6691
4000	v	0.0159	0.0164	0.0172	0.0182	0.0198	0.0223	0.0287	0.1052	0.1463	0.1752	0.1994	0.2210	0.2411	0.2601	0.2783
	h	78.5	177.2	277.1	379.8	487.7	606.9	763.0	1174.3	1311.6	1403.6	1481.3	1552.2	1619.8	1685.7	1750.6
	s	0.1271	0.2893	0.4304	0.5573	0.6760	0.7940	0.9343	1.2754	1.3807	1.4461	1.4976	1.5417	1.5812	1.6177	1.6516
5000	v	0.0159	0.0164	0.0171	0.0181	0.0196	0.0219	0.0268	0.0591	0.1038	0.1312	0.1529	0.1718	0.1890	0.2050	0.2203
	h	81.1	179.5	279.1	381.2	488.1	604.6	746.0	1042.9	1252.9	1364.6	1452.1	1529.1	1600.9	1670.0	1737.4
	s	0.1265	0.2881	0.4287	0.5550	0.6726	0.7880	0.9153	1.1593	1.3207	1.4001	1.4582	1.5061	1.5481	1.5863	1.6216
6000	v	0.0159	0.0163	0.0170	0.0180	0.0195	0.0216	0.0256	0.0397	0.0757	0.1020	0.1221	0.1391	0.1544	0.1684	0.1817
	h	83.7	181.7	281.0	382.7	488.6	602.9	736.1	945.1	1183.8	1323.6	1422.3	1505.9	1582.0	1654.2	1724.2
	s	0.1258	0.2870	0.4271	0.5528	0.6693	0.7826	0.9026	1.0176	1.2615	1.3574	1.4229	1.4748	1.5194	1.5593	1.5962
7000	v	0.0158	0.0163	0.0170	0.0180	0.0193	0.0213	0.0248	0.0334	0.0573	0.0816	0.1004	0.1160	0.1298	0.1424	0.1542
	h	86.2	184.4	283.0	384.2	489.3	601.7	729.3	901.8	1124.9	1281.7	1392.2	1482.6	1563.1	1638.6	1711.1
	s	0.1252	0.2859	0.4256	0.5507	0.6663	0.7777	0.8926	1.0350	1.2055	1.3171	1.3904	1.4466	1.4938	1.5355	1.5735

ATTACHMENT 5

NUCLEAR ENGINEERING AND OPERATIONS PROCEDURE

NEO 8.01

JUMPER, LIFTED LEAD AND BYPASS CONTROL

APPROVED J. F. Gulu
Senior Vice President,
Nuclear Engineering and Operations

REVISION 1.A

DATE August 15, 1986

CONCURRENCE [signature] for
Manager, Quality Assurance

NUCLEAR ENGINEERING AND OPERATIONS PROCEDURE

NEO 8.01

JUMPER, LIFTED LEAD AND BYPASS CONTROL

1.0 PURPOSE

The purpose of this procedure is to define, control and specify the review requirements for jumpers, lifted leads, and bypasses in a manner that ensures conformity with design intent and operability requirements and preserves plant and personnel safety. Changes that are not defined in this procedure are addressed in Reference 3.2.

Jumpers, lifted leads, and bypasses that are positively identified and controlled in other station-approved procedures that meet the requirements of Section 6.1 are excluded from the requirements of this procedure. However, this procedure may be, in part or in entirety, implemented whenever additional control is desired even though implementation is not specifically required.

2.0 APPLICABILITY

This procedure applies to the Nuclear Engineering and Operations (NEO) Group, including the Northeast Nuclear Energy Company (NNECO) and the Connecticut Yankee Atomic Power Company (CYAPCO).

3.0 REFERENCES

3.1 Northeast Utilities Quality Assurance Topical Report, Section 3 - Design Control; Section 11 - Test Control; and Section 14 - Inspection, Test, and Operating Status.

3.2 ANSI N18.7-1976 - Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants.

3.3 NEO 3.03 - Preparation, Review, and Disposition of Plant Design Change Request (PDCRs).

3.4 NEO 3.12 - Safety Evaluations.

4.0 DEFINITIONS

4.1 Blind or Blank Flange

A plate-like device in a mechanical system to stop flow.

4.2 Disabled Annunciator Alarm

A change that disables the visual and/or audible alarm function of an annunciator.

4.3 Electrical Jumper

A temporary electrical connection that bypasses a component within an electrical circuit, changing the circuit design or configuration. Jumpers may be made permanent through the PDCR process (Reference 3.2) in which case the jumper becomes permanent and is no longer referred to as a jumper.

4.4 Mechanical Jumper

A temporary connection such as a spool piece, hose, tubing, or piping that joins two systems together or bypasses a component within a system, thus changing the system's design or configuration. (This does not include hoses connected from system drains to floor drains or providing air to pneumatic tools or breathing apparatus.)

4.5 Lifted Lead

A break in continuity of a circuit such that one connection of a wire previously connected to two terminals is removed.

4.6 Shielding

Material used to attenuate radiation.

4.7 Other Bypass and Jumper Devices

4.7.1 In addition to the items defined in 4.1 to 4.6 the following are also considered bypasses or jumper devices.

4.7.1.1 Gagged relief or safety valves

4.7.1.2 Installed/removed filters or strainers (other than for routine maintenance)

4.7.1.3 Plugged floor drains

4.7.1.4 Pulled circuit cards

4.7.1.5 Gagged or blocked process control valves

4.7.1.6 Block walls

4.8 Jumper, Lifted Lead and Bypass Control Log

A log maintained by operations personnel in the control room. This log shall consist of all active Jumper, Lifted Lead and Bypass Sheets (Attachment 8.A) and the Jumper, Lifted Lead and Bypass Index sheets (Attachment 8.B). Each Jumper, Lifted Lead or Bypass request shall have a unique number assigned.

4.9 Bypass - Lifted Lead and Jumper Control Tag

A uniquely numbered tag (Attachment 8.C) used to identify each Jumper, Lifted Lead or Bypass in accordance with this procedure.

4.10 Safety Evaluation

A safety evaluation is a written review of the change completely in accordance with Reference 3.4. The review is the responsibility of the Unit Engineering Staff (it may be performed by NUSCO Engineering) and independently reviewed by the Unit Engineering Supervisor or his designee.

4.11 Technical Evaluation

A technical evaluation is a written in-depth evaluation of items checked either "YES" or "DON'T KNOW" on Attachment 8.A, Section A - Technical Assessment. The review is the responsibility of the Unit Engineering Staff (it may be performed by NUSCO Engineering) and independently reviewed by the Unit Engineering Supervisor or his designee.

4.12 Operable

The definition of operable is the current definition found in the associated unit's technical specification.

4.13 Safety-Related Systems

Category I, radioactive waste and fire protection systems.

4.14 Safety Assessment

A safety assessment is a review of the proposed Jumper, Lifted Lead or Bypass to determine whether a safety evaluation is required.

4.15 Technical Assessment

A technical assessment is a review of the proposed Jumper, Lifted Lead or Bypass to determine whether a technical evaluation is required.

4.16 Duty Officer Qualified Person

A station employee who periodically stands Unit Duty Officer watches. He need not be on duty to execute his responsibilities under this procedure.

5.0 RESPONSIBILITIES

5.1 Vice President, Nuclear Operations

Responsible for maintaining this procedure current.

5.2 Plant Operations Review Committee (PORC)

Responsible for reviewing all Jumpers, Lifted Leads and Bypasses made in accordance with this procedure within 14 days following installation. If the Jumper, Lifted Lead or Bypass requires a safety evaluation, the PORC is responsible for reviewing the safety evaluation and determining that the change is safe and does not constitute an unreviewed safety question before the Jumper, Lifted Lead or Bypass is installed on operable equipment.

5.3 Operations Supervisor

Responsible for ensuring overall compliance with this procedure.

5.4 Operations Shift Supervisor

Responsible for overall control, i.e., administration and authorization, of Jumpers, Lifted Leads or Bypasses during the assigned shift. Responsible for approving the installation and restoration of Jumpers, Lifted Leads and Bypasses. Responsible for completing the technical and safety assessment sections of Attachment 8.A.

5.5 Plant Personnel

Responsible for ensuring that operations personnel are properly notified prior to placement or removal of any Jumper, Lifted Lead or Bypass. Responsible for installing, removing, and proper placement of Jumpers, Lifted Leads and Bypasses in accordance with this procedure. Responsible for ensuring that devices are compatible with their intended functions; e.g., current capacity of wire, termination, insulation, pressure and temperature ratings, etc. Additionally, responsible for ensuring that Jumpers, Lifted Leads and Bypasses meet the requirements of the technical evaluation and applicable special instructions noted on Attachment 8.A.

5.6 Engineering Supervisor

Responsible for performing and documenting technical evaluations and safety evaluations before a Jumper, Lifted Lead or Bypass is installed on operable equipment when required by this procedure.

5.7 Duty Officer Qualified Person

Responsible for completing the technical and safety assessment sections of Attachment 8.A.

5.8 Department Heads

Responsible for reviewing installations to determine if they are still needed or if they should be made permanent. Responsible for initiating design changes for those Jumpers, Lifted Leads or Bypasses that are to be made permanent.

6.0 INSTRUCTIONS

6.1 General

6.1.1 This procedure shall be considered for safety-related and nonsafety-related systems. If the equipment is declared inoperable, then this procedure does not need to be implemented. This procedure, the unit tag-out procedure, or station-approved procedure (including a work order) can be used as desired to control Jumpers, Lifted Leads and Bypasses in equipment that is declared inoperable.

6.1.2 Jumpers (both electrical and mechanical) shall be compatible for the use intended; e.g., size, terminal, type, insulation, pressure rating, material, piping construction, etc.

6.1.3 When they are installed on operable equipment, Jumpers, Lifted Leads and Bypasses that are positively identified and controlled in other station-approved procedures are excluded from the requirements of this procedure. In order to qualify for this exclusion, the other procedure must specify the following:

6.1.3.1 Either Shift Supervisor or Senior Control Room Operator notification and sign-off when Jumpers, Lifted Leads and Bypasses are installed and removed.

6.1.3.2 Documentation of the placement or removal of Jumpers, Lifted Leads and Bypasses.

6.1.3.3 Independent verification when Jumpers, Lifted Leads and Bypasses are installed or removed on safety-related systems.

6.1.4 Multiple Jumper, Lifted Lead or Bypass Control tags may be issued under a single "Jumper, Lifted Lead and Bypass Control sheet." The devices may be both

electrical and mechanical devices as long as they are issued for a common purpose.

6.2 Initiation

6.2.1 The requestor shall fill out the description section of the Jumper - Lifted Lead - Bypass Control Sheet (Attachment 8.A).

6.2.2 The following information shall be recorded on Attachment 8.A.

6.2.2.1 Date.

6.2.2.2 Unique index number.

6.2.2.3 The equipment and functions that are affected.

6.2.2.4 The reason (purpose) for the request.

6.2.2.5 The applicable drawings or design documents.

6.2.2.6 The tag number, location, and type(s) shall be indicated.

6.2.2.7 Description necessary to locate the device including terminations.

6.2.2.8 Any special actions, instructions, or requirements.

6.2.2.9 The requestor shall sign and date the description section.

6.3 Assessment

6.3.1 The Assessment Section Items 1 to 3 of Attachment 8.A shall be completed by both a Shift Supervisor and a Duty Officer qualified member of the affected unit. They shall add any special-action instructions/ requirements (description section) as necessary. They shall ensure that the installation does not adversely affect the intended safety, reliability, or efficiency of the plant.

6.3.1.1 They shall determine if the equipment is to be declared operable with the Jumper, Lifted Lead or Bypass installed. If the equipment is declared inoperable, Steps 6.3.1.2, 6.3.1.3, 6.3.3, 6.3.4, and 6.3.5 need not be completed. Jumpers, Lifted Leads or

Bypasses installed on inoperable equipment shall be reviewed to ensure that they are either removed or controlled in accordance with this procedure by being assessed prior to returning the equipment to operability.

6.3.1.2 They shall determine (technical assessment) if a more detailed written technical review is needed by answering the questions in Section A (Attachment 8.A). If any of the answers in this section are "YES" or "DON'T KNOW," then Attachment 8.A shall be forwarded to the associated unit's Engineering Department which is responsible for completing a detailed technical evaluation. The detailed technical evaluation will address items marked "YES" and "DON'T KNOW" from the technical assessment.

6.3.1.3 They shall determine (safety assessment) if a detailed safety evaluation is needed by answering the questions in Section B (Attachment 8.A). If any of the answers in this section are marked "YES" or "DON'T KNOW," then Attachment 8.A shall be forwarded to the associated unit's Engineering Department which is responsible for completing a safety evaluation.

6.3.1.4 All special actions, such as temporary operating instructions, etc., shall be noted on Attachment 8.A (description section). The results of the assessments, if applicable, will be appended to Attachment 8.A.

6.3.1.5 The Shift Supervisor and Duty Officer Qualified persons who complete the assessment section must document their approval with their signatures. (Documentation of phone conversations is acceptable if the Duty Officer Qualified Person is not onsite. If performed by a phone conversation, then note the date, time, and when called on Attachment 8.A or Attachment to 8.A.

6.3.2 If all of the questions in Sections A and B are marked "NO," or are not required to be performed or are addressed with completed assessments and approvals, the Shift Supervisor may authorize approval for installation.

6.3.3 The associated unit's Engineering Department shall be responsible for completing a detailed technical evaluation applicable to the questions in Section A that are marked "YES" or "DON'T KNOW." The Engineering Department shall be responsible for completing a detailed safety evaluation if any of the questions in Section B are marked "YES" or "DON'T KNOW."

6.3.4 The PORC must review and recommend approval of all safety evaluations required by Attachment 8.A, Section B before installation on operable equipment. Copies of the approved technical evaluations and/or safety evaluations shall be attached to Attachment 8.A to be forwarded to the Shift Supervisor for installation approval. Copies of the safety evaluation shall also be attached to the PORC meeting minutes and forwarded to the Nuclear Review Board (NRB) for review.

6.3.5 If a technical specification change is required or an unreviewed safety question is found to exist, then NRB and NRC approval is required before installation can be authorized.

6.4 Installation

6.4.1 Shift Supervisor approval is required for installation. Responsibilities are as follows:

6.4.1.1 Assigning a unique number from the Jumper, Lifted Leads and Bypass Control Index (Attachment 8.B) to the Jumper, Lifted Leads and Bypass Control Sheet (Attachment 8.A).

6.4.1.2 Entering into the index the number, type, affected equipment/function, restoration required by (date/mode), installation date, and the department requesting the installation.

6.4.1.3 Reviewing Attachment 8.A to ensure that it is complete and that all required reviews and approvals have been obtained.

6.4.1.4 Reviewing the Jumper, Lifted Leads or Bypass to ensure it is compatible with existing plant conditions.

6.4.1.5 Implementing any special actions or technical specification requirements.

6.4.1.6 Issuing caution tags (as appropriate) and noting the issuance in the comments section.

6.4.1.7 Filling out or have filled out a Jumper, Lifted Leads - Bypass Control Tag for each Lifted Lead or device. The tag shall identify the termination points.

Jumpers that are of such length that both ends are not visible when installed shall have a Tag attached at each end.

6.4.1.8 Signing and dating Attachment 8.A to identify that the Shift Supervisor has authorized approval for installation.

6.4.1.9 Ensuring that a verification of the installation is completed.

6.4.1.10 Placing the original copy of Attachment 8.A in the Jumper, Lifted Leads and Bypass Control Log (including supporting documentation). A copy can be put in the log until appropriate PORC approval is obtained.

6.4.2 The plant personnel performing the installation (installer) shall implement the following:

6.4.2.1 Install each jumper, lifted lead, bypass or device in accordance with safe station work practices.

6.4.2.2 Install at least one Jumper, Lifted Lead - Bypass Control tag for each Lifted Lead or device.

6.4.2.3 Insulate lifted leads from other circuits and from ground.

6.4.2.4 Ensure jumpers and other devices (both electrical and mechanical) are compatible for the intended use.

6.4.2.5 Identify the proper circuits or other components before disconnecting any leads or installing any devices.

6.4.2.6 Sign and date Attachment 8.A after installation is complete.

6.4.3 The plant personnel performing the independent verification (verifier) shall, by means of a visual verification or functional check, ensure proper installation. If verification is done by performing a functional check, the method of performing the

check shall be described in the comments section of Attachment 8.A.

An exception to performing a second verification may be granted by the shift supervisor if the second verification would result in a significant radiation exposure. However, every reasonable effort shall be made to confirm operability; e.g., functional test, remote viewing, etc. The basis for this exception shall be noted in the comments section of Attachment 8.A.

6.4.3.1 The person performing the independent verification then signs and dates Attachment 8.A after verification is complete.

6.4.4 For those Jumpers, Lifted Leads and Bypasses which did not require prior PORC approval, the Operations Supervisor shall have Attachment 8.A (or a copy of Attachment 8.A) forwarded to the PORC once the installation is complete. PORC must then perform a review within 14 days following installation

6.5 <u>Restoration</u>

6.5.1 The shift supervisor is responsible for the following:

6.5.1.1 Authorizing restoration by signing and dating Attachment 8.A after ensuring that the restoration is permissible.

6.5.1.2 Entering the restoration date into the Index after the restoration is verified.

6.5.1.3 Removing any caution tags and deleting any special instructions that were issued as a result of that installation and briefing operators on the shift.

6.5.1.4 Ensuring that an appropriate verification of the restoration is completed and performing any functional check.

6.5.2 The plant personnel performing the restoration (restorer) shall ensure proper restoration and that all tags are removed. After restoration is complete, the restorer shall sign and date Attachment 8.A.

6.5.3 The plant personnel performing the independent verification (verifier) shall, by means of a visual verification or functional check, ensure proper

restoration. If the verification is done by performing a functional check, the method shall be noted in the comments section of Attachment 8.A.

An exception to performing a second verification may be granted by the shift supervisor if the second verification would result in a significant radiation exposure. However, every reasonable effort shall be made to confirm restoration to full operability; e.g., function test, remote viewing, etc. The basis for this exception shall be noted in the comments section of Attachment 8.A.

6.5.3.1 The person(s) performing the independent verification then signs and dates Attachment 8.A after verification is complete.

6.6 Reviews and Audits

6.6.1 The Jumper, Lifted Lead and Bypass Control Log shall be reviewed during each shift and prior to a mode change to ensure technical specification requirements are being met.

6.6.2 The Operations Department shall perform a monthly review of the log for administrative errors. This review also will identify those Jumpers, Lifted Leads or Bypasses either no longer needed or in effect for more than six months.

6.6.2.1 A copy of administrative errors that have been discovered as a result of the review shall be forwarded to the Operations Supervisor for corrective action.

6.6.2.2 Those installations that are determined to be no longer needed shall be restored per Section 6.5.

6.6.2.3 A copy of those installations that have been installed for longer than six months shall be forwarded, after each time the audit is performed, by the Operations Supervisor to the responsible Department Head to determine if they are still needed.

6.6.2.4 Completed Index Sheets (Attachment 8.B) that reflect restoration shall be forwarded to the Operations Supervisor for review, disposition and forwarding to nuclear records.

6.6.2.5 The Operations Department review of jumper, lifted leads, and bypass control sheets shall be documented by noting it in the audit section of Attachment 8.A.

6.6.2.6 The Operations Supervisor will coordinate with other departments as part of this audit, to include a check for correct placement, condition, and tagging of Jumpers, Lifted Leads and Bypasses.

6.6.2.7 The Operations Supervisor shall review results of the audit with PORC.

6.6.3 The responsible Department Head shall review installations that are installed longer than six months to see if they are still needed or if they should be made permanent.

6.6.3.1 Those that are determined to be no longer needed shall be restored per Section 6.5.

6.6.3.2 For those that are to be made permanent, submit design changes per Reference 3.2.

7.0 FIGURES

Figure No.	Figure Title
7.1	Jumper, Lifted Leads and Bypass Control Flowchart

8.0 ATTACHMENTS

Attachment No.	Attachment Title
8.A	Jumper - Lifted Lead - Bypass Control Sheet
8.B	Jumper - Lifted Lead - Bypass Control Log
8.C	Jumper - Lifted Lead - Bypass Control Tag
8.D	Major Changes from Previous Revision (Rev. 1 Updated to Rev. 1.A).

JAN 31 1986

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A contained borated water volume between 1,166,000 and 1,207,000 gallons,

b. A boron concentration between 2000 and 2200 ppm of boron,

c. A minimum solution temperature of 40°F, and

d. A maximum solution temperature of 50°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the contained borated water volume in the tank, and

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature.

APPROVAL: [illegible signature]

DATE: 11-15-85

SORC MTG. No. 85-41

UNIT 3
EMERGENCY ACTION LEVELS

ATTACHMENT 7

CLASSIFICATION	UNUSUAL EVENT		ALERT		SITE AREA EMERGENCY		GENERAL EMERGENCY	
DEFINITION	Events in progress or have occurred which indicate a potential degradation of the level of safety of the plant.		Events in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant.		Events in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public.		Events in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity.	
POSTURE CODE	DELTA-ONE/DELTA-TWO		CHARLIE-ONE		CHARLIE-TWO		BRAVO/ALPHA	
KEY CONDITIONS	SYMPTOM	EMERGENCY ACTION LEVEL	SYMPTOM	EMERGENCY ACTION LEVEL	SYMPTOM	EMERGENCY ACTION LEVEL	SYMPTOM	EMERGENCY ACTION LEVEL
BARRIER FAILURE OR IMMINENT BARRIER FAILURE	A. ECCS initiated with discharge to vessel. (DELTA-ONE)	A. SIAS annunciation AND valve status open OR flow to vessel.					A. Loss of 2 of 3 barriers with a high potential for loss of the 3rd barrier. (BRAVO)	A. 1. See Table 1 A. 2. Also see radioactive release key conditions.
	B. Failure of a reactor coolant systems safety or relief valve to close. (DELTA-ONE)	B. High temperature alarms (greater than 135°F) on discharge piping from relief/safety valves OR pressurizer relief tank level temperature, AND/OR pressure high alarms OR acoustic monitor alarms.						
	C. Sudden fuel damage indication. (DELTA-ONE)	C. Letdown system radiation monitor failed fuel alarm AND chemistry sampling verifies increase in total NG and I concentrations in the RCS by 0.4 milli Ci/cc.	C. Potential for loss of fuel clad OR severe loss of fuel cladding.	C. See Table 1, Part A.	C. Degraded core with possible loss of coolable geometry.	C. See Table 1 part A, Loss of Fuel Clad Barrier AND Part B, Potential Loss of RCS.	C. Any potential core melt situation. (BRAVO)	C. See Table 1, Part A and Part B, Loss of Fuel Clad and Loss of RCS with no ability to restore core cooling.

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EMERGENCY ACTION LEVELS

CLASSIFICATION	UNUSUAL EVENT		ALERT		SITE AREA EMERGENCY		GENERAL EMERGENCY	
DEFINITION	Events in progress or have occurred which indicate a potential degradation of the level of safety of the plant.		Events in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant.		Events in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public.		Events in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity.	
POSTURE CODE	DELTA-ONE/DELTA-TWO		CHARLIE-ONE		CHARLIE-TWO		BRAVO/ALPHA	
KEY CONDITIONS	SYMPTOM	EMERGENCY ACTION LEVEL	SYMPTOM	EMERGENCY ACTION LEVEL	SYMPTOM	EMERGENCY ACTION LEVEL	SYMPTOM	EMERGENCY ACTION LEVEL
BARRIER FAILURE OR IMMINENT BARRIER FAILURE	D. Abnormal coolant temperature and/or pressure. (DELTA-ONE)	D. High Tavg alarm OR pressurizer pressure greater than 2300 psia for greater than 2 hours OR pressurizer pressure greater than 2350 psia OR pressurizer pressure less than 2225 psia for greater than 2 hours.						
	E. Suddenly exceeding either primary to secondary leak rate Technical Specification or primary system leak rate Technical Specification. (DELTA-ONE)	E. Primary to secondary leak in any steam generator greater than 0.5 gpm OR primary unidentified leakage greater than 1 gpm, OR identified greater than 10 gpm.	E. Primary system leakrate greater than the capacity of one charging pump.	E. Containment area radiation monitor alarm AND pressurizer level program calls for additional charging pump to start.	E. Loss of Coolant Accident. (LOCA)	E. See Table 1, Part B, Loss of RCS.	E. LOCA and failure to isolate ctmt. OR potential to rupture ctmt. (BRAVO)	E. See Table 1, Part A, B, and C.
			F. Rapid failure of Steam Generator tubes.	F. Leakage in excess of normal makeup capacity AND reactor trip on low pressure or reactor pressure decreasing uncontrollably AND air ejector/steam generator blowdown, radiation monitor alarm AND no significant increase in ctmt. pressure, sump level or ctmt. area radiation monitor alarm.	F. Rapid failure of steam generator tubes with total loss of offsite power.	F.1 Leakage in excess of normal makeup capacity AND undervoltage alarms on bus 34C and 34D AND Reactor trip on low pressure AND Air ejector or steam generator blowdown high radiation alarm AND No significant increase in ctmt. pressure, sump levelor ctmt. area radiation monitor alarm.		

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CLASSIFICATION	UNUSUAL EVENT		ALERT		SITE AREA EMERGENCY		GENERAL EMERGENCY	
DEFINITION	Events in progress or have occurred which indicate a potential degradation of the level of safety of the plant.		Events in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant.		Events in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public.		Events in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity.	
POSTURE CODE	DELTA-ONE/DELTA-TWO		CHARLIE-ONE		CHARLIE-TWO		BRAVO/ALPHA	
KEY CONDITIONS	SYMPTOM	EMERGENCY ACTION LEVEL	SYMPTOM	EMERGENCY ACTION LEVEL	SYMPTOM	EMERGENCY ACTION LEVEL	SYMPTOM	EMERGENCY ACTION LEVEL
BARRIER FAILURE OR IMMINENT BARRIER FAILURE						F.2 Leakage in excess of normal makeup capacity AND undervoltage alarms on bus 34C and 34D AND Reactor pressure decreasing uncontrollably AND Air ejector or steam generator blowdown radiation monitor high alarm AND No significant increase in ctmt. pressure, sump level or ctmt. area radiation monitor alarm.		
	G. Loss of containment integrity requiring shutdown by Technical Specification. (DELTA-ONE)	G. See Table 1 Part C Containment Barrier.						
			H. Fuel handling accident with release of radioactivity to containment or fuel handling building.	H. (Containment area radiation monitor alarm OR containment air monitor alarm OR Spent fuel pool area radiation monitor alarm OR spent fuel pool area radiation monitor alarm) AND Shift Supervisor's opinion.	H. Major damage to spent fuel in containment or fuel handling building.	H. Water loss from spent fuel pool to below fuel level OR containment or spent fuel pool area radiation monitor greater than 1,000 mr/hr AND Shift Supervisor's opinion.		

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CLASSIFICATION	UNUSUAL EVENT		ALERT		SITE AREA EMERGENCY		GENERAL EMERGENCY	
DEFINITION	Events in progress or have occurred which indicate a potential degradation of the level of safety of the plant.		Events in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant.		Events in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public.		Events in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity.	
POSTURE CODE	DELTA-ONE/DELTA-TWO		CHARLIE-ONE		CHARLIE-TWO		BRAVO/ALPHA	
KEY CONDITIONS	SYMPTOM	EMERGENCY ACTION LEVEL	SYMPTOM	EMERGENCY ACTION LEVEL	SYMPTOM	EMERGENCY ACTION LEVEL	SYMPTOM	EMERGENCY ACTION LEVEL
RADIOACTIVE RELEASES	A. Unplanned radioactive release. (DELTA-TWO)	A. Any accidental, unplanned, OR uncontrolled radioactive release. (Normal or expected releases from maintenance or other operational activities are not included). See EPIP Form 4701-6.	A. Radiological effluents greater than 10 times Technical Specification instantaneous limits for more than 15 minutes.	A. Gaseous or liquid effluent radiation monitor greater than 10 times the Technical Specification instantaneous limits for more than 15 minutes.	A. Effluent monitor readings corresponding to greater than 50 mR/hr whole body dose rate at the maximum off-site location OR offsite W.B. dose between 0.05 to 1.0 rem OR offsite thyroid dose between 0.25 to 5 rem.	A. MP1 stack high range monitor between 0.3 and 6 uCi/cc OR MP3 ventilation vent monitor between 0.03 and 0.6 uCi/cc OR MP3 main steam line monitor between 10 and 200 uCi/cc OR EMT's detect levels anywhere offsite of: dose rates greater then 50 mr/hr OR I-131 concentrations greater than 7.0xE-8 uCi/cc.	A. Effluent monitor readings or offsite measurements or estimates corresponding to 1 to 5 R/hr whole body dose rate or 5 to 25 R/hr thyroid dose rate at the maximum off-site location. (BRAVO)	A.1. If EOF is activated, use actual meteorology and actual release or actual field team data (i.e. EMT's detect levels offsite of: dose rates between 1 R/hr and 5 R/hr OR I-131 concentrations between 1.5 E-6 and 7.5 E-6 uCi/cc). If not, Use A.2. If fast moving event, use A.2. A.2. MP1 stack high range monitor between 6 and 30 uCi/cc OR MP3 ventilation vent monitor between 0.6 and 3 uCi/cc OR MP3 main steam line monitor between 200 and 1000 uCi/cc.
							B. Effluent monitor readings or offsite measurements or estimates corresponding to greater than 5 R/hr whole body dose rate at the site boundary or greater than 25 R/hr thyroid dose rate at the maximum off-site location. (ALPHA)	B.1. If EOF is activated use actual meteorology and actual release or actual field team data (i.e. EMT's detect levels offsite of: dose rates greater than 5R/hr OR I-131 concentrations greater then 7.0xE-6 uCi/cc). If not, use B.2. If fast moving event, use B.2.

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CLASSIFICATION	UNUSUAL EVENT		ALERT		SITE AREA EMERGENCY		GENERAL EMERGENCY	
DEFINITION	Events in progress or have occurred which indicate a potential degradation of the level of safety of the plant.		Events in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant.		Events in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public.		Events in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity.	
POSTURE CODE	DELTA-ONE/DELTA-TWO		CHARLIE-ONE		CHARLIE-TWO		BRAVO/ALPHA	
KEY CONDITIONS	SYMPTOM	EMERGENCY ACTION LEVEL	SYMPTOM	EMERGENCY ACTION LEVEL	SYMPTOM	EMERGENCY ACTION LEVEL	SYMPTOM	EMERGENCY ACTION LEVEL
RADIOACTIVE RELEASES							(ALPHA)	B.2. MP1 stack high range monitor greater than 30 uCi/cc OR MP 3 ventilation vent monitor greater than 3 uCi/cc OR MP3 main steam line monitor greater than 1000 uCi/cc.
STEAM LINE BREAK	A. Steamline break. (DELTA-ONE)	A. Break inside Containment: Steam generator pressure low alarm AND containment high-pressure/temperature alarm AND main steam isolation signal. A.1. Break outside of containment: Steam generator low pressure low alarm AND main steam isolation signal.	A. Steamline break with significant primary to secondary leakage.	A. Same as Unusual Event AND 1. Containment area radiation monitor alarms OR 2. Air ejector radiation monitor alarm OR 3. Blowdown radiation monitor alarm.	A. Steamline break with significant primary to secondary leakage and indication of fuel damage.	A. Same as Alert AND 1. Containment area Radiation Monitor alarms OR 2. Air ejector radiation monitor alarm OR 3. Blowdown radiation monitor alarm AND Chemistry sampling indicates dose equivalent I-131 greater than 300 uCi/ml OR greater than 0.4 milli Ci/cc total NG and I, in the Reactor Coolant System.		

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EMERGENCY ACTION LEVELS

CLASSIFICATION	UNUSUAL EVENT		ALERT		SITE AREA EMERGENCY		GENERAL EMERGENCY	
DEFINITION	Events in progress or have occurred which indicate a potential degradation of the level of safety of the plant.		Events in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant.		Events in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public.		Events in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity.	
POSTURE CODE	DELTA-ONE/DELTA-TWO		CHARLIE-ONE		CHARLIE-TWO		BRAVO/ALPHA	
KEY CONDITIONS	SYMPTOM	EMERGENCY ACTION LEVEL	SYMPTOM	EMERGENCY ACTION LEVEL	SYMPTOM	EMERGENCY ACTION LEVEL	SYMPTOM	EMERGENCY ACTION LEVEL
LOSS OF POWER	A. Loss of all tie lines between the switchyard and the 345-kV grid, or total loss of on-site AC power capability. (DELTA-ONE)	A. All off-site line isolation breakers open OR both diesel generators are declared inoperable AND the normal station service transformer is unavailable.	A. Loss of off-site power and loss of all on-site AC power for less than 15 minutes.	A. Undervoltage alarms on Bus 34C and 34D AND diesel generator fail to start and load AND condition exists for less than 15 minutes.	A. Loss of off-site power and loss of on-site AC power for more than 15 minutes.	A. Undervoltage alarms on Bus 34C and 34D AND diesels fail to start and load, AND condition exists for MORE than 15 minutes.		
			B. Loss of all on-site vital DC power for less than 15 minutes.	B. Battery trouble alarms for batteries 1, 2, 3 & 4 for less than 15 minutes.	B. Loss of all on-site DC power for more than 15 minutes.	B. Battery trouble alarms for batteries 1, 2, 3 & 4 for more than 15 minutes.		
EQUIPMENT FAILURE	A. Loss of engineered safety feature or fire protection system function requiring shutdown by Technical Specification. (DELTA-ONE)	A. As determined by Technical Specification Limiting Conditions for Operation (LCO's).	A. Loss of functions needed for plant cold shutdown.	A. Operation beyond the action statements in Technical Specifications for systems required for cold shutdown.				
			B. Failure of reactor protection system to initiate and complete a reactor trip which brings the reactor subcritical.	B. Core mimic display indicates a significant number of Control Rods remaining out of core, coincident with a reactor trip alarm. Temperature (TAVG) OR primary system pressure remaining high or rising.				

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UNIT 3
EMERGENCY ACTION LEVELS

CLASSIFI-CATION	UNUSUAL EVENT		ALERT		SITE AREA EMERGENCY		GENERAL EMERGENCY	
DEFINITION	Events in progress or have occurred which indicate a potential degradation of the level of safety of the plant.		Events in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant.		Events in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public.		Events in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity.	
POSTURE CODE	DELTA-ONE/DELTA-TWO		CHARLIE-ONE		CHARLIE-TWO		BRAVO/ALPHA	
KEY CONDITIONS	SYMPTOM	EMERGENCY ACTION LEVEL	SYMPTOM	EMERGENCY ACTION LEVEL	SYMPTOM	EMERGENCY ACTION LEVEL	SYMPTOM	EMERGENCY ACTION LEVEL
EQUIPMENT FAILURE			C. Loss of all alarms or annunciators for greater than 15 minutes.	C. Direct observation.	C. Loss of all alarms (annunciators, for greater than 15 minutes AND Shift Supervisors opinion that a transient has occurred or is in progress.	C. Shift Supervisor opinion.		
					D. Loss of safety systems such that a high potential for affecting reactor safety or causing significant releases exists.	D. Direct observation of the loss of safety systems OR Shift Supervisors opinion.		

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UNIT 3
EMERGENCY ACTION LEVELS

CLASSIFICATION	UNUSUAL EVENT		ALERT		SITE AREA EMERGENCY		GENERAL EMERGENCY	
DEFINITION	Events in progress or have occurred which indicate a potential degradation of the level of safety of the plant.		Events in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant.		Events in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public.		Events in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity.	
POSTURE CODE	DELTA-ONE/DELTA-TWO		CHARLIE-ONE		CHARLIE-TWO		BRAVO/ALPHA	
KEY CONDITIONS	SYMPTOM	EMERGENCY ACTION LEVEL	SYMPTOM	EMERGENCY ACTION LEVEL	SYMPTOM	EMERGENCY ACTION LEVEL	SYMPTOM	EMERGENCY ACTION LEVEL
EQUIPMENT FAILURE	E. A major loss of emergency assessment capability, off-site response capability or communications capability. (DELTA-ONE)	E. Indications or alarms on process or effluent monitoring system parameters not functional in the control room to an extent requiring plant shutdown OR complete loss of radiopager system OR loss of radiological assessment capability, (e.g. loss of all meteorological instrumentation and its backup capability).						
SECURITY THREAT	A. Security threat. (DELTA-ONE)	A. Notification by Security of a security threat.	A. Ongoing security compromise.	A. Notification by Security of an ongoing security compromise.	A. Imminent loss of physical control of Station due to security incident.	A. Notification by Security of a physical attack on the plant involving imminent occupancy of the control room, auxiliary shut down panels or other vital areas as defined in the station security plan.	A. Loss of physical control of the Station due to a security incident. (BRAVO)	A. Notification by Security of a physical attach on the plant that has resulted in unauthorized personnel occupying control room or any other vital area described in the station security plan.
FIRE	A. Fire lasting more than 10 minutes within the unit. (DELTA-ONE)	A. Fire detection panel alarm OR fire pump running alarms lead to investigation which determines an actual fire is in progress.	A. Fire potentially affecting safety systems.	A. Fire alarms OR fire pump start alarms OR fire system flow alarms OR Shift Supervisors opinion.	A. Fire compromising the functions of safety systems.	A. Indication of actual fire (fire detection panel alarm), fire pump running alarm AND/OR Shift Supervisors opinion.		

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UNIT 3
EMERGENCY ACTION LEVELS

CLASSIFICATION	UNUSUAL EVENT		ALERT		SITE AREA EMERGENCY		GENERAL EMERGENCY	
DEFINITION	Events in progress or have occurred which indicate a potential degradation of the level of safety of the plant.		Events in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant.		Events in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public.		Events in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity.	
POSTURE CODE	DELTA-ONE/DELTA-TWO		CHARLIE-ONE		CHARLIE-TWO		BRAVO/ALPHA	
KEY CONDITIONS	SYMPTOM	EMERGENCY ACTION LEVEL	SYMPTOM	EMERGENCY ACTION LEVEL	SYMPTOM	EMERGENCY ACTION LEVEL	SYMPTOM	EMERGENCY ACTION LEVEL
NATURAL PHENOMENON	A. Significant or natural phenomenon being experienced or anticipated. Includes: a) Any earthquake b) 50-year flood or tsunami c) Any tornado d) Any hurricane e) Low water at intake structure. (DELTA-ONE)	A. Direct observation OR notification by external agencies OR seismic monitors indicates seismic activity OR sustained wind speed greater than 75 mph (measured at the 142-ft. elevation) OR flood or low water alarm.	A. Severe natural phenomena being experienced or projected near design basis levels.	A. Direct observation OR notification by external agencies OR earthquake greater than 0.02 g (OBE) OR tornado striking facility OR sustained hurricane winds near 85 mph OR flood, low water, or surge near design basis levels.	A. Severe natural phenomena being experienced or projected to be greater than design basis levels AND with plant not in cold shutdown.	A Plant not in cold shutdown AND direct observation OR notification by external agencies OR earthquake greater than 0.17g (SSE) OR tornado striking facility OR sustained hurricane winds greater than 85 mph OR flood, low water, or surge greater than design levels or with failure of vital equipment at lower levels.		
OTHER HAZARDS	A. Other hazards being experienced which could endanger the facility (e.g., on-site airplane crash, train derailment, explosion, on-site toxic or flammable gas release). (DELTA-ONE)	A. Direct observation of a hazard capable of endangering the facility or Shift Supervisors opinion.						
			B. Evacuation of control room anticipated or required with control of shutdown systems from local stations.	B. Direct observation of an event requiring evacuation of the control room.	B. Evacuation of control room AND control of shutdown systems not established from local stations in 15 minutes.	B. Evacuation of control room AND control of shut-down systems not established from local stations in 15 minutes.		

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UNIT 3
EMERGENCY ACTION LEVELS

CLASSIFICATION	UNUSUAL EVENT		ALERT		SITE AREA EMERGENCY		GENERAL EMERGENCY	
DEFINITION	Events in progress or have occurred which indicate a potential degradation of the level of safety of the plant.		Events in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant.		Events in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the pub. c.		Events in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity.	
POSTURE CODE	DELTA-ONE/DELTA-TWO		CHARLIE-ONE		CHARLIE-TWO		BRAVO/ALPHA	
KEY CONDITIONS	SYMPTOM	EMERGENCY ACTION LEVEL	SYMPTOM	EMERGENCY ACTION LEVEL	SYMPTOM	EMERGENCY ACTION LEVEL	SYMPTOM	EMERGENCY ACTION LEVEL
OTHER HAZARDS	C. Any event that results in the nuclear power plant not being in a controlled or expected condition while operating or shutdown (any inadvertent criticality), including unanalyzed conditions that significantly compromise plant safety, conditions outside the plant's design basis, or conditions not covered by operating and emergency procedures. (DELTA-ONE)	C. Direct observation OR Shift Supervisors opinion.						
	D. Transportation of contaminated injured person from site to hospital. (DELTA-ONE)	D. Direct observation.						

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TABLE 1

BARRIER FAILURE OR IMMINENT BARRIER FAILURE EALS (6)

#	Parameter Instrumentation/Information	High Potential for Loss of Barrier (1)	Loss of Barrier (1)	Applicable Mode (Note 2)
A.	Fuel Clad Barrier			
1.	Primary coolant concentration	1. Greater than 4 milli Ci/cc of total MC and I at time of shutdown OR greater than 300 micro Ci/gm I-131 DEQ.	1. Greater than 40 milli Ci/cc of total MC and I at time of shutdown.	All
2.	Five core exit thermocouples	2. 700°F and increasing.	2. Above 1200°F	All but 5 & 6
3.	Reactor water level	3. RVMLS less than 19%	3. N.A.	All but 5 & 6
4.	Containment fuel manipulator crane radmonitor	4. N.A.	4. Greater than 2×10^4 mr/hr from shine dose.	All but 5 & 6
B.	Reactor Coolant System Pressure Barrier			
1.	RCS Leak Rate	1. RCS leak rate greater than capacity of one charging pump.	1. In excess of normal makeup capability.	All but 5 & 6
2.	RCS Subcooling	2. 0°F	2. 100°F superheat	All but 5 & 6
3.	Containment Hi Rad Monitor (7)	3. N.A.	3. a. With normal coolant activity about 5 to 50 R/hr. b. With clad barrier loss coolant activity as follows: (Also see note 3)	All but 5 & 6

Time after Shutdown	Reading
0 to 0.5 hr	g.t. 5×10^3 R/hr
0.5 to 4 hr	g.t. 2×10^3 R/hr
4 to 48 hr	g.t. 1×10^3 R/hr
48 to 72 hr	g.t. 5×10^2 R/hr

#	Parameter Instrumentation/Information	High Potential for Loss of Barrier (1)	Loss of Barrier (1)	Applicable Mode (Note 2)
C.	Containment Barrier			
1.	Containment Pressure	1. 60 psia and increasing.	1. Rapid decrease following initial increase above 10 psia with no sprays.	All but 5 & 6
2.	Containment Isolation Valves status after CIAS	2. N.A.	2. Valves not closed.	All but 5 & 6
3.	Main Steam Relief Line Radiation Monitor	3. N.A.	3. Offscale and steam flow indication. (See Note 6).	All but 5 & 6
4.	Also see Radiation Release under Site Area Emergency and General Emergency. (See Note 5)	4. N.A.	4. N.A.	All

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TABLE 1 NOTES

1. Each parameter listed below represents, by itself, a high potential for loss or loss of the barrier indicated, as appropriate.

2. Mode Definition:
   1 = Power Operation
   2 = Startup
   3 = Hot Standby
   4 = Hot Shutdown
   5 = Cold Shutdown
   6 = Refueling

3. This indicates loss of 2 barriers, the RCS and the fuel clad.

4. This indicates loss of 2 barriers, the RCS and the containment.

5. If Radiation Release levels exist that exceed the Site Area Emergency EAL, then 2 barriers are lost, (i.e., the RCS and Containment barriers). However, if Radiation Release levels exceed the General Emergency EAL, then all 3 barriers are lost.

6. Classification by Barrier Key Conditions are as follows:

7. ALERT

   a. Potential loss of fuel clad or RCS.
   b. Loss of RCS OR fuel clad.

   SITE AREA

   a. Potential loss of 2 barriers.
   b. Loss of any 1 barrier and potential loss of another.
   c. Loss of any 2 barriers.

   GENERAL

   a. Potential loss of 3 barriers.
   b. Loss of 1 barrier and potential loss of other 2 barriers.
   c. Loss of 2 barriers and a potential loss of another.
   d. Loss of all 3 barriers.

All
[illegible]
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3/4.8 ELECTRICAL POWER SYSTEMS

JAN 31 1986

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E Distribution System, and

b. Two separate and independent diesel generators, each with:

   1) A separate day tank containing a minimum volume of 205 gallons of fuel,

   2) A separate Fuel Storage System containing a minimum volume of 32,760 gallons of fuel,

   3) A separate fuel transfer pump,

   4) Lubricating oil storage containing a minimum total volume of 280 gallons of lubricating oil, and

   5) Capability to transfer lubricating oil from storage to the diesel generator unit.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With either an offsite circuit or diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Specifications 4.8.1.1.1a. and 4.8.1.1.2a.5) within 1 hour and at least once per 8 hours thereafter; restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Specifications 4.8.1.1.1a. and 4.8.1.1.2a.5) within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours from the time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one diesel generator inoperable in addition to ACTION a. or b. above, verify that:

   1. All required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and

ACTION (Continued)

2. When in MODE 1, 2, or 3, the steam-driven auxiliary feedwater pump is OPERABLE.

If these conditions are not satisfied within 2 hours be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by performing the requirements of Specification 4.8.1.1.2a.5) within 1 hour and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing the requirements of Specification 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two diesel generators to OPERABLE status within 72 hours from time of initial loss or be in least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the Onsite Class 1E Distribution System shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring (manually and automatically) unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released from the site (see Figure 5.1-3) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microCurie/ml total activity.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released from the site exceeding the above limits, restore the concentration to within the above limits within 15 minutes.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program specified in Section I of the REMODCM.

4.11.1.1.2 The results of radioactive analyses shall be used in accordance with the methods of Section II of the REMODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

ATTACHMENT 3

REACTOR OPERATOR EXAM

1. PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW.

1.01b (5) Answer key is incorrect. Should be "remains the same". OPS Form 3209B-1, steps 3.4 and 3.5 negate changes in reactivity effects.

1.05 c RCCA worth decreases with core age, as described in attached Neutron Poisons Lesson Plan (RTI-01-C), pages 21 and 22.

1.06 b Answer should be "Less Negative", because the smaller change in fuel temperature per percent power change at EOL outweighs the more negative FTC. [Reference: Reactivity Coefficients and Defects Lesson Plan, pages 13, 14 previously provided]

1.07 c Agreed that "cold leg temperature decrease" would be an acceptable alternative for the second half of the answer key.

1.09 (2) Question does not elicit numerical values for Xenon concentration. Operators are not required to memorize the level of specificity provided in the answer key. Based upon the question, any response indicating higher concentration at 100% should be acceptable.

2. PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS

2.01 (3) Agreed that either TRUE or FALSE can be acceptable answer [Reference: P&ID EM 122B]

2.02 b. Agreed that Containment Instrument Air compressors could be part of an acceptable answer [Reference: P&ID EM 122B]

2.07 Question does not elicit the response provided in the answer key. Implication is to give three reasons for the designed sequence delay, for which there is not three discreet reasons. Any three true statements concerning elapsed time should be considered acceptable.

2.08 a. Original RO question was deleted during administration of exam and replaced with SRO exam question 6.01 with increased point value. Facility questions the validity of this action; design basis information may not be appropriate at the RO level.

2.08 c. Agreed that RCS overpressure is an acceptable answer. [Reference: T/Spec 3/4 1-10, 1-9, B 3/4 1-3]

2.09 c Agreed that acceptable answer is diesel has lost control power. [Reference: ESK 8KC] Question badly worded. To elicit correct response, SRO exam question 6.07 a is better worded.

3. INSTRUMENTS AND CONTROLS

3.03 Agreed that all thermocouple temperatures read out on control room CRT, not just the highest one. [Reference: SPDS Text, Figure 5.1]

3.05 c Agreed answer could assume steam dumps in either steam pressure or T-AVE mode, if explained properly.

3.07 c Agreed excessive KW/FT is complete answer. [Reference: Reactor Trip List]

3.08 c Agreed that "except 480 volt load center and motor control center supply breakers and the running charging pumps," should not be required as part of the answer as this statement does not constitute an automatic action due to LOP.

3.09 b Agreed that any answer consistent with concept stated in answer key is acceptable. i.e "live zero" current is also an acceptable answer. [Reference: Excore NI Text p. 20]

3.11 Agreed that block valves would probably be open in given plant conditions. Therefore, any answer which indicated this or explained the fact that the block valves receive an open signal when the COPS switch is placed in the ARM position, is an acceptable response to that aspect of the question. [Reference: PZR Pressure and Level Control, p. 12]

4. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL

4.05 a. Agreed that correct answer does not need to include "do not use cheater bars" as this is a prohibition vice a requirement as regards manual seating of motor operated valves. [Reference: OPS-3.07 p.6]

4.09 a. Agreed answer should include "failure of two or more control rods" vice one. [Reference: AOP 3566 Rev. 1]

4.09 b Agreed answer is "to maintain sufficient NPSH to boric acid transfer pumps." [Reference: Immediate Boration Lesson Plan - p. 4]

SENIOR REACTOR OPERATOR EXAM

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

5.02 Answer should be "Less Negative", because the smaller change in fuel temperature per percent power change at EOL outweighs the more negative FTC. [Reference: Reactivity Coefficients and Defects Lesson Plan, pages 13, 14 previously provided]

5.04 Correction to reference data (Ops Form 3209A-1) was made during administration of exam, which makes answer key invalid. Based upon the correction, student would also need Samarium curves, which were not provided.

5.05 Trends provided in the question are misleading and confusing when used as an evaluation device (rather than instructional aid as intended). Examiner should assess human factors of using word processing equipment to draw graphs.

Part a.2 should be answered by power going up and stabilizing at a higher value. This was not one of the choices.

Part b.3 answer key is incorrect. Pzr level will increase on heatup, then trend to a lower than initial value on program. Answer should be c.

Part b.2 elicits a trend that is not one of the choices. Recommend this be deleted.

5.07 d Millstone 3 circ water pumps are propeller-type pumps. As such, the question does not provide enough information to allow a meaningful evaluation.

5.09 b RCCA worth decreases with core age, as described in attached Neutron Poisons Lesson Plan (RTI-01-C), pages 21 and 22.

6. PLANT SYSTEMS DESIGN, CONTROL AND INSTRUMENTATION

6.01 c - same as 2.08c

6.03 b - same as 3.09b

6.08 d Loop isolation valves no longer interlocked with steam generator low low level trip.
[Reference: 108 D. 684 Sh. 7 Rev. 6]

6.10 - same as 3.11

7. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL

7.01 d Facility expressed concern that we do not expect operators to memorize cautions or notes contained within normal operating procedures.
[Reference: OP 3310A p. 12]

7.03 c Same concern as 7.01 d. Many reasons could have been given for emergency retrieval of transfer car. Primary reasons involve drive failure and desire to move transfer car.
[Reference: OP 3303C p. 7]

7.05 a same as 4.09 a

7.05 b same as 4.09 b

7.06 Agreed correct answer is 180°F and 10^5R
[Reference: E-O Rev. 1 p. 4]

7.07 c Information given in question is not clear, and as such may not elicit expected response. Clarification was not provided to all students. Correct answers based upon interpretation of question may vary.

8. ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS

8.01 Agreed that "if equipment is deemed inoperable" is acceptable answer.
[Reference ACP-QA-2.06B p.5]

8.05 d Agreed that 4 hours is correct answer based on 50.72 (b) (2) (ii) vice 1 hour based on 50.72 (b) (1) (iv) as signal initiating safety injection was "mistakenly generated" and hence is not a "valid signal."
[Reference 10CFR50.72 pp. 509-510]

8.06 Facility argued that this question was ambiguous and that any person in chain of command should be appropriate answer since definition of what constitutes "emergency" as framed in question is nebulous. In any event, agreed that any superintendent listed in EPIP 4112 can be substituted for duty officer in answer.
[Reference: EPIP 4112 p. 5]

8.07 a/d Agreed to grade per valid assumptions made, as regards to length of time casualty existed, since length of time casualty existed determines emergency classification.

8.07 c Agreed to grade per valid assumptions made by examinee as to the cause of SI and decreasing RCS pressure as a variety of accidents could have resulted in these plant conditions, i.e, SGTR, steam break, as well as a LOCA could have resulted in similar plant indications.

8.08 b Agreed to accept 12/15/87 2000 as alternative answer. If examinee with proper explanation assumes 3.8.1.1.(c) applies, then hot standby is required within 8 hrs vice 18 hrs as required by 3.8.1.1.(b).
[Reference LCO 3/4 8.1]

8.08 c Agreed that 3.8.1.1.(e) applies at this point vice 3.8.1.1(d).
[Reference LOCO 3/4 8.1]

8.08d Agreed that 12/19/87 at 0200 is an acceptable alternative answer. If examinee assumes "initial loss" indicates time starts from initial loss of both diesel generators vice loss of one generator, this will add 9 hours to answer.
[Reference LCO 3/4 8.1]

8.~~10~~ 09 b/c Facility argued that question asked for two reasons when proper answer really was one reason with two parts. Hence, due to the nature of the question, extraneous information may have been elicited by examinees in attempting to answer question fully.