ATTACHMENT 2

Duquesne Light

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August 23, 1988
ND1VPN:5574

Mr. Robert M. Gallo, Chief
Operations Branch
Division of Reactor Safety
U.S. Nuclear Regulatory Commission
Region 1
475 Allendale Road
King of Prussia, PA 19406

Reference: Beaver Valley Power Station, Unit #1
Docket 50-334, License DPR-66
License Examination Report

Dear Mr. Gallo:

Please find enclosed comments generated by our Training Section associated with the written examination administered August 17, 1988 at our facility.

If you have any questions concerning this report, please contact Mr. T. W. Burns at (412) 393-5751.

Very truly yours,

J. D. Sieber
Vice President, Nuclear Group

JDS/cjr

Enclosure

cc: T. W. Burns
Central File (2)

QUESTION 5.07 (3.20)

a. How is DNBR affected (increase, decrease, or no change) by the indicated changes of the following, and BRIEFLY explain WHY. Consider each case separately. Initial plant condition is operating at 95% power (1.20).

   1) Boron concentration in the RCS is decreased by 10 ppm.
   2) Pressurizer heaters are energized.
   3) Reactor power is increased to 100%.

b. Using the attached TS Figure 2.1-1, Reactor Core Safety Limit, explain (briefly) for each of the three sections (A, B, and C) what parameter is being limited (1.20).

c. If Section A of the safety limit curve is exceeded, what protection would be lost and WHY (0.80)?

ANSWER 5.07 (3.20)

a. 1) Decrease (0.2), would increase Tave which reduces the subcooling margin (0.2).

   2) Increase (0.2), would increase RCS pressure and increase subcooling margin (0.2).

   3) Decrease (0.2), would increase heat flux and reduce subcooling margin (0.2).

b. Section A. Limits ave. coolant temperature (Enthalpy) at core exit to less than saturation (0.4).

   Section B. Limits hot channel core exit quality to 15% or less (0.4).

   Section C. Limits DNBR to 1.3 or greater (for normal ops and mod. freq. incidents) (0.4).

c. All trips that rely on Delta Temperature (0.4). Because if Th becomes saturated there will be no change in hot leg temperature and will no longer be an indication of power (0.4).

REFERENCE

KA 00300 K5.01 3.9 002000 K5.09 4.2 002000 G0.05 4.1 193008 K1.05 3.6
BVPS THERMO TEXT Pgs. 17 through 20
BVPS LP-TMO-7 EO 12 and 14
002000G005 002000K509 003000K501 193008K105 ...(KA'S)

COMMENT

5.07.0.1 The answer key is correct for the initial effect on DNBR.
5.07.2 However, the final effect for Part 1 would be "no change" if auto Rod Control is assumed since Tavg will be returned to setpoint and for Part 2 would be "no change" since the pressurizer spray valves would open to return pressure to setpoint. Please accept either answer since the question did not specify whether the initial or final effect on DNBR was desired.

QUESTION 6.03 (3.00)

Listed below are valves that position in response to an ESF signal. Identify the exact signal (SIS, CIA, or CIB) that actuates the valve and whether the valve position, after ESF actuation, is open or closed.

a. CCR to RCP 1B (TV-1CC-103B)
b. RCS Letdown (TV-1CH-204)
c. Stm Gen 1C Blowdown (TV-1BD-100C)
d. ORS Spray Pump 2B Suction for Containment (MOV-1RS-155B)
e. RCP Seal Return (MOV-1CH-369)
f. Prim Grade Water to PRT (1RC-72)
g. PZR Vapor Space Sample (TV-1SS-112A1)
h. Coolant Charging System (MOV-1CH-289)
i. Main Condenser Air Eject Vent (TV-1SV-100A)
j. BI Tank to Cold Legs (MOV-1SI-867C)

ANSWER 6.03 (3.00)

a. S CIB
b. S CIA
c. S CIA
d. O CIB
e. S CIA
f. S CIA
g. S CIA
h. S SIS
i. S CIB
j. O SIS

(0.15 for position)
(0.15 for signal)

REFERENCE

BV-1, Chapter 47, Table 1
KA 103000 A3.01 4.2
BVPS LP-SQS-47.2 EO No. 4
103000A301 ...(KA'S)

COMMENT

6.03.e
6.03.f
The RCP seal return valves are MOV-1CH-378 and 381. MOV-1CH-369 does not exist. 1RC-72 is a check valve which does not receive any ESF signals. This is an error in the referenced table from Operating Manual Chapter 47. Please change the answer to Part f to read "This valve does not receive an ESF signal".

QUESTION 6.06 (2.20)

The reactor is operating at a steady state 25% power, all control systems are in automatic. Turbine load is increased to 100% and the controlling steam pressure detector for #1 S/G sticks at the 25% value. Explain how and why this will effect #1 steam generator level - assume no operator action. Continue your explanation to steady state conditions or plant trip. State any assumptions made.

ANSWER 6.06 (2.20)

m-stm = K(P-stm) (delta-P)

Since the steam pressure component stays constant (it should go down) while the delta-P increases, indicated steam flow will be higher than actual steam flow (0.6).

The summing network for flow will send a signal to the flow controller to open the feed regulating valve (0.5).

As level starts to increase, the level error will signal for the FRV to close (0.5).

Eventually, the flow error will be cancelled out by the level error, and the FRV will be positioned such that steam generator level will operate at a steady state level higher than desired program (0.6).

REFERENCE

BV-1, Chapter 24, Section 1, Pages 17-20
BVPS LP-SQS-24.1 EO No. 6
KA 059000 K1.04 3.4 059000 A3.02 3.1
059000A302 059000K104 ...(KA'S)

COMMENT

6.06 The answer key states that the flow error will be cancelled out by the level error and the steam generator will operate at a steady state level higher than the desired program. This is not true. The level controller is a proportional-integral controller and it will continue to vary its output and change the position of the feed regulating valve until the level error becomes zero. The flow error is strictly a proportional signal. Please change the answer to indicate that the steady state level will be at the programmed value. Refer to Figure 7.7-6 from the UFSAR.

REV. 0 (1/82)

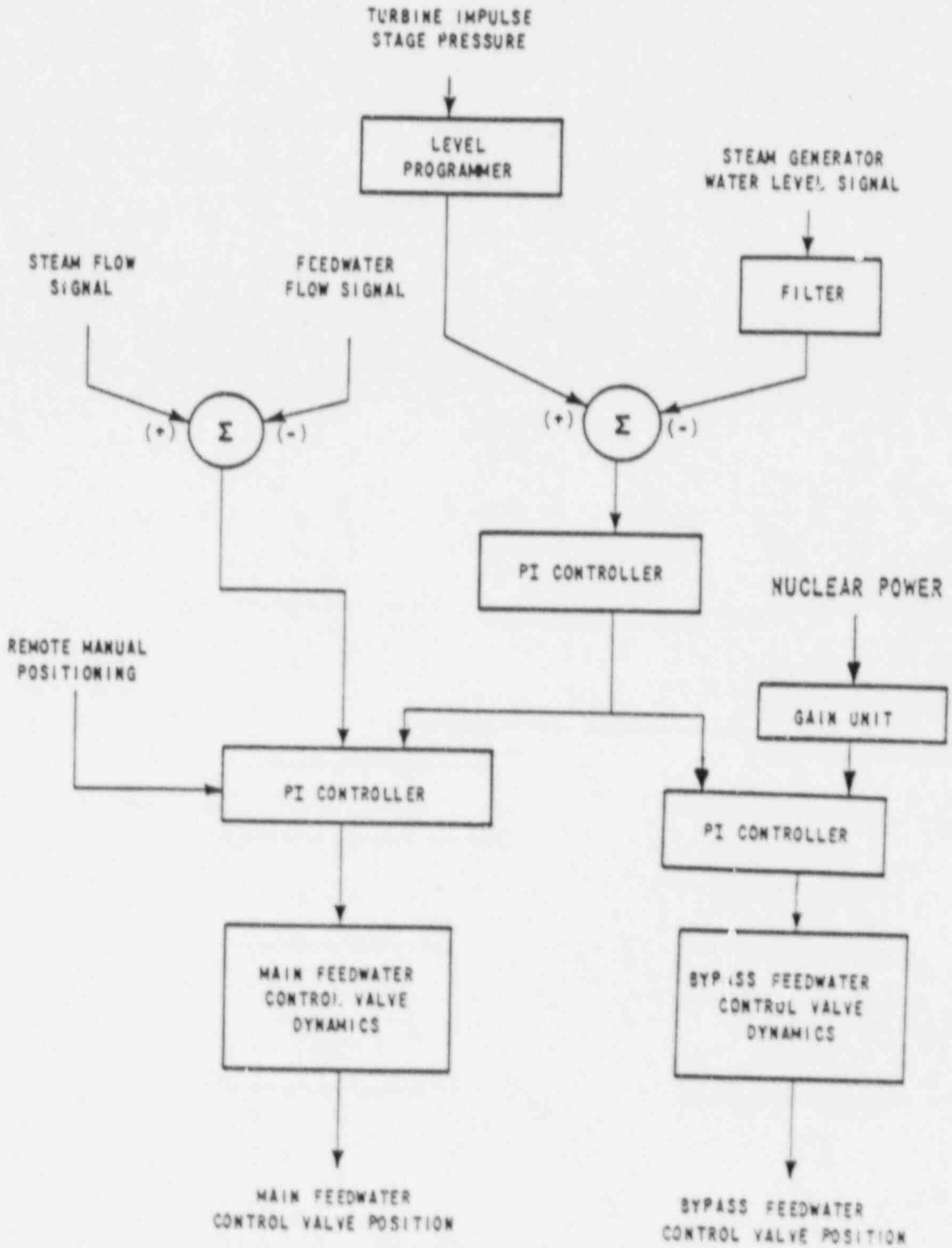


FIGURE 7.7-6
BLOCK DIAGRAM OF STEAM GENERATOR WATER LEVEL CONTROL SYSTEM
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

QUESTION 6.09 (3.00)

The following questions relate to the auxiliary feedwater system.

a. What are four of the six conditions that will start the electric-driven AFW pumps (FW-P-3A and 3B)? (1.00)

b. What four conditions will prevent auto start of the electric-driven AFW pumps (FW-P-3A and 3B)? (1.00)

c. Auxiliary feed throttle valves, MOV-1FW-151A,B,C,D,E,F receive AUTO open signals from what components? (0.50)

d. Briefly explain how, when using the dedicated AFW pump (FW-P-4), feed flow and steam generator levels (component and location(s)) are controlled. (0.50)

ANSWER 6.09 (3.00)

(Any four at 0.25 each)

a. 1) Control switch to start
   2) Two of three SG lo-lo level
   3) Both main feed pumps tripped
   4) Opposite AFW start signal not followed by discharge pressure
   5) Safety injection signal (train A for 3A and B for 3B)
   6) PNL-AMSAC

(Four at 0.25 each)

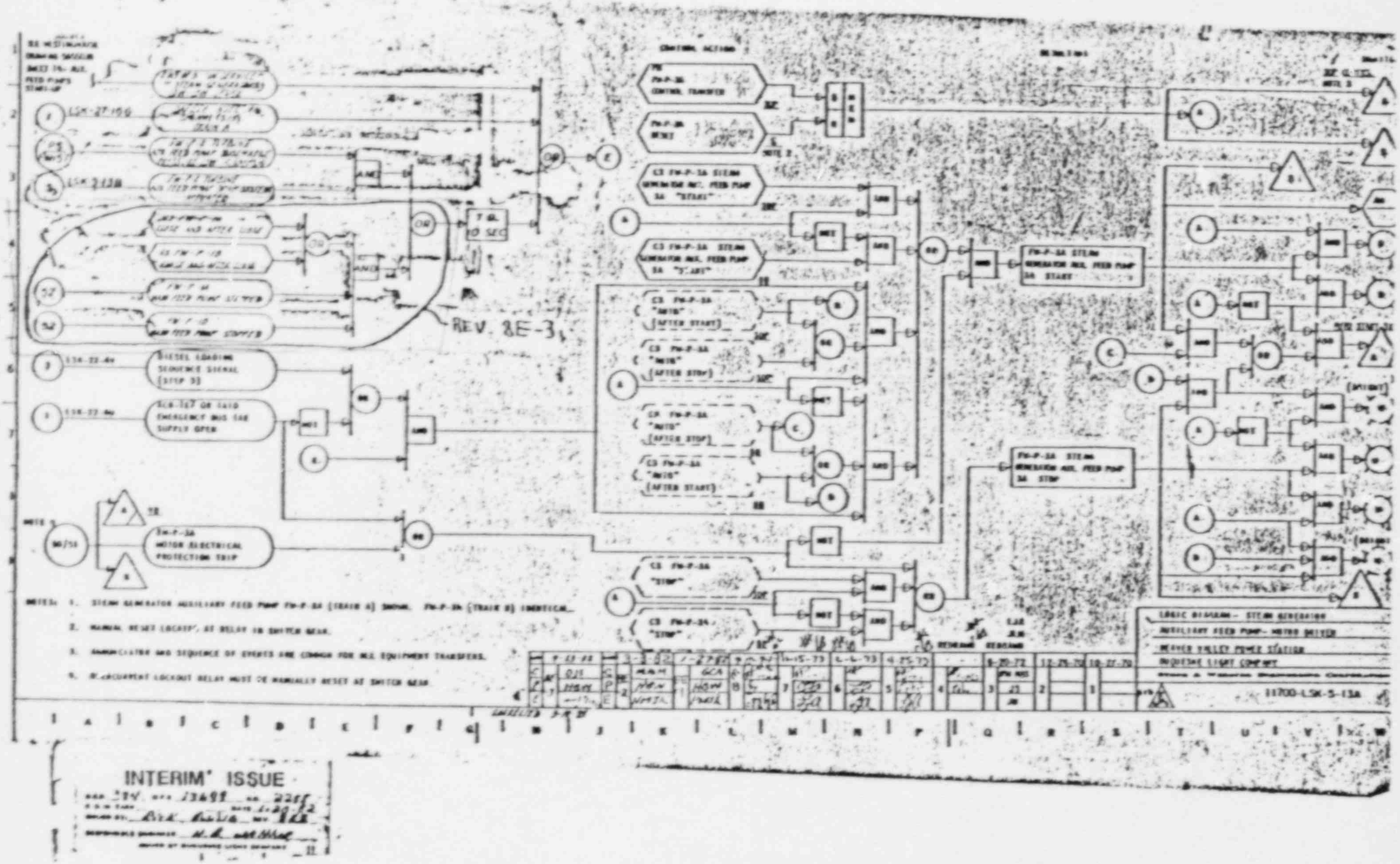
b. Control switch in pull-to-lock or stop
   Bus loss of voltage (1AE or 1DF)
   Diesel loading sequence (not timed to start yet)
   Motor electrical protection

c. Breaker closure of either MD AFW (0.25)
   Steam-driven AFW steam supply trip valves (TV-MS-105A and B) not fully open (0.25)

d. Feed flow is manually controlled at the feedwater bypass control manifold (0.25).
   Level information is from the Control Room or the backup indicating panel (0.25).

REFERENCE

BVPS OM 1.24.1 Pages 15 through 18 and 3
BVPS LP-SQS-24.1 EO 7
KA 061 000 K4.02 4.2 061 000 G0.09 3.9
061000G009 061000K402 ...(KA's)



CONTROL ACTION
REV. 8E-3
CS FW-P-3A STEAM GENERATOR AUX. FEED PUMP 3A "START"
FW-P-3A STEAM GENERATOR AUX. FEED PUMP 3A START
FW-P-3A STEAM GENERATOR AUX. FEED PUMP 3A STOP
DIESEL LOADING SEQUENCE SIGNAL (STEP 3)
FW-P-3A MOTOR ELECTRICAL PROTECTION TRIP
T.D. 10 SEC
NOTES: 1. STEAM GENERATOR AUXILIARY FEED PUMP FW-P-3A (TRAIN A) SHOWN. FW-P-3B (TRAIN B) IDENTICAL.
2. MANUAL RESET LOCATED AT RELAY IN SWITCH GEAR.
3. ANNUNCIATOR AND SEQUENCE OF EVENTS ARE COMMON FOR ALL EQUIPMENT TRANSFERS.
4. ALL EQUIPMENT LOCKOUT RELAY MUST BE MANUALLY RESET AT SWITCH GEAR.
LOGIC DIAGRAM - STEAM GENERATOR
AUXILIARY FEED PUMP - MOTOR DRIVEN
BEAVER VALLEY POWER STATION
DUQUESNE LIGHT COMPANY
11700-LSK-5-13A
INTERIM' ISSUE

INSTRUMENTATION AND CONTROLS (continued)

its relay rack (see power supply and control location list). There is also a TEST OPEN pushbutton for each trip valve located at the valve.

From the benchboard, [FW-P-2] is started by placing either or both trip valve control switches in open. [FW-P-2] is stopped by closing the open trip valve (or valves). In AUTO, [FW-P-2] is started by a Lo-Lo steam generator level (2/3) in any steam generator, a (2/3) reactor coolant pump bus undervoltage (buses 1A and 1B, 1A and 1C, or 1B and 1C) or PNL-AMSAC.

When [FW-P-2] is running due to an automatic start signal it cannot be stopped using the benchboard trip valve control switches. Steam supply to trip valves [TV-1MS-105A and B] can be isolated by steam supply isolation valve [MOV-1MS-105]. Valve [MOV-1MS-105] is controlled by a benchboard CLOSE-OPEN switch. [MOV-1MS-105] is normally open so [FW P-2] is available for automatic start.

Auxiliary Feed Pumps 3A and 3B [FW-P-3A, B]

Auxiliary feed pumps 3A and 3B are driven by 4160V electric motors. Pump [FW-P-3A] is powered from emergency bus 1AE and [FW-P-3B] is powered from emergency bus 1DF. See Figure 24-8 for [FW-P-3A] logic (FW-P-3B, identical). [FW-P-3A] starter (breaker 1AE-16) has breaker control switches on both the benchboard and the emergency shutdown panel. Control is transferred to the emergency shutdown panel by a transfer pushbutton located at that panel. Control is transferred back to the benchboard by resetting the transfer relay located at the breaker switchgear. Control logic is identical at both panels.

[FW-P-3A] is started by:

1. Placing the controlling control switch to START
2. Two out of three steam generators having Lo-Lo level
3. Both main feed pumps tripped
4. Safety injection signal (train A starts [FW-P-3A] and train B starts [FW-P-3B]
5. [FW-P-2] start signal (not followed by [FW-P-2] discharge pressure)
6. PNL-AMSAC

[FW-P-3A] is stopped or prevented from starting by:

1. Placing the controlling control switch in PULL-TO-LOCK or STOP
2. 1AE bus loss of voltage
3. Diesel loading sequencer (not timed to start yet)
4. Motor electrical protection

INSTRUMENTATION AND CONTROLS (continued)

[FCV-FW-102] will be closed by de-energizing solenoid operated valve [SOV-FW-102] provided any one of the following conditions is met:

a. Turbine driven auxiliary feed pump [FW-P-2] inlet steam pressure is below 355 psig.

b. Turbine driven auxiliary feedwater pump [FW-P-2] suction flow is greater than 505 gpm.

In addition, [FIS-FW-151A, B] and [FIS-FW-152] will cause annunciation in the control room if recirculation flow remains below the recirculation valve open setpoint for 10 seconds.

Auxiliary Feed Throttle Valves [MOV-1FW-151A, B, C, D, E, F]

Each steam generator auxiliary feed water supply line contains two parallel piped, redundant, motor operated throttle valves. Steam generator 1A auxiliary feedwater flow is controlled by [MOV-1FW-151E and F], steam generator 1B by [MOV-1FW-151C and D], steam generator 1C by [MOV-1FW-151A and B]. All six valves are normally open, however, they also receive a back-up auto open signal. The auto open signals are initiated by the motor-driven aux. feed pumps [FW-P-3A or 3B] breaker closure or the steam-driven pump [FW-P-2] steam supply trip valves [TV-MS-105A or B] not fully closed. Refer to Figure No. 24-9 for [MOV-FW-151E] logic (other 5 identical). Once initiated, the A.T.C. maintains the open signal for 30 seconds. This signal does not prevent valve operation by the throttle valve control switches, but if the control switch is released within the 30 second period the throttle valves will reopen.

See Figure 24-9 for [MOV-1FW-151E] logic (other five identical). [MOV-1FW-151E] has two three position (CLOSE-OPEN-spring return to NORMAL) control switches; one located on the benchboard, the other on the emergency shutdown panel. Control is transferred to the emergency shutdown panel by a transfer pushbutton located at that panel. Control is transferred back to the benchboard by resetting the transfer relay at its relay rack (see power supply and control location list). Control logic is identical at both panels.

[MOV-1FW-151E] travels in the open direction when the controlling switch is placed in the OPEN position. Valve travel stops when the control switch is released. [MOV-1FW-151E] travels closed when the controlling control switch is in the CLOSE position. Valve travel stops when the control switch is released or the full open or closed positions are reached regardless of switch position.

QUESTION 7.02 (3.50)

Answer the following questions concerning E-0, Reactor Trip or Safety Injection:

a. The Main Turbine has not tripped and you attempt a manual trip as required, with no response. What additional actions are you required to take in order to shutdown the turbine? (1.00)

b. List three plant conditions that require SI initiation? Include setpoints. (0.75)

c. What four parameters are checked to determine if SI flow should be terminated? (1.25)

d. Following an SI reset, what condition must be met before an automatic reinitiation of SI will occur? (0.5)

ANSWER 7.02 (3.50)

a. Close main steam trip valves (four at 0.25 pts. each)
Runback the turbine
Close main steam bypass valves
Close non-return valves

b. 1) Pressurizer pressure < 1845 psig (three at 0.25 pts. each)
2) Containment pressure > 1.5 psig
3) Steamline pressure < 510 psig

c. 1) RCS subcooling criteria met (0.25)
2) Feed flow to intact SG's (> 350 gpm) (0.25) or narrow range level in at least one intact SG (0.25)
3) RCS pressure (stable or increasing) (0.25)
4) PZR level (> 5%) (0.25)

d. Reactor trip breakers must be closed (0.5)

REFERENCE

BVPS EOP E-0 Pages 3, 5, 16, 19
KA 000 007 EK 3.01 4.6 000 038 EK3.09 4.5
BVPS LP-LRT-VII-61 EO B1
LP-LRT-VII-62 EO A1

COMMENT

7.02.b "wo other plant conditions require SI initiation in E-0. These are contained on the left-hand page. Please add "RCS subcooling less than attachment" and "Pressurizer level less than 5%" to the answer key as alternate acceptable answers.

QUESTION 7.07 (2.75)

The plant is operating at 100 percent power. Urgent repair of a motor operated valve must be performed inside containment to prevent a plant shutdown. The repair is expected to take one (1) hour. The maintenance man that is supposed to do the repair is 25 years old, his lifetime exposure on his NRC Form 4 through the last quarter is 33,000 mrem. He has received an additional 700 mrem this quarter. The area that he will be working in has a radiation field of 675 mrem/hr. gamma and 450 mrem/hr neutron.

a. Can the man selected complete the task without exceeding any NRC (1.75) exposure limits. Show all calculations and state any assumptions made.

b. Technical Specifications, Section 6.12, requires personnel that enter high radiation areas exercise one of three specified options. Briefly list TWO of those options. (1.00)

ANSWER 7.07 (2.75)

a. 5(N-18) = 35 rem lifetime exposure allowed (0.25)
Lifetime to date = 33000 mrem + 700 mrem = 33700 mrem (0.25)
Total lifetime available = 35000 mrem - 33700 mrem = 1300 mrem (0.25)
Total quarterly exposure available = 3000 mrem - 700 mrem = 2300 mrem (0.25)
Lifetime exposure is more limiting (0.25)
675 mrem/hr gamma + 450 mrem neutron = 1125 mrem/hr total (0.25)

Yes, man can perform task. (0.25)

(Any two at 0.5 pts. each)

b. 1. Carry a radiation monitoring instrument which continuously indicates dose rate in the area.

2. Carry/wear a radiation monitoring device which integrates the dose rate and alarms at a preset value.

3. Be accompanied by a qualified individual who is equipped with a radiation monitoring device.

REFERENCE

KA 194 001 K1.03 3.4
BVPS LP-RC-01 EO 19, 23, 27
194001K103 ...(KA'S)

COMMENT

7.07.a The statement "Lifetime exposure is more limiting" in the answer key is worth .25 points. The question does not ask for this information. Please delete this from the key.

QUESTION 8.01 (2.40)

The plant is operating at 75% power and the latest leak rate data shows:

- 13.2 GPM - Corrected RCS leakage rate
- 1.5 GPM - Leakage into the Pressurizer Relief Tank
- 1.2 GPM - Leakage into the Primary Drains Transfer Tank
- 3.4 GPM - Leakage through SI-23, RCS Loop 1A, cold leg isolation (previous leakage rate was 1.6 GPM)
- 0.8 GPM - Total primary to secondary leakage
- 4.2 GPM - Leakage past RCP seals

What RCS leakage limits, if any, have been exceeded? Refer to attached Technical Specifications. Show ALL work and STATE any assumptions.

ANSWER 8.01 (2.40)

RCS Pressure Isolation Valve Limits exceeded (0.70)
(3.4-1.6)/(5.0-1.6) = 1.8/3.4 = > 50% (0.50)

UNIDENTIFIED Leakage limits exceeded (0.70)
13.2 - (1.5 + 1.2+3.4+0.8+4.2) = 2.1 GPM (0.50)

REFERENCE

RS 3.4.6.2; TS 3.4.6.3
KA 002020 G0.05 4.1
BVPS LP-SQS-6.5 EO No. 7
002020G005 ... (KA'S)

COMMENT

8.01 If the assumption is made that all of the .8 gpm primary to secondary leakage is from one steam generator, then the 500 gallon per day tube leakage Technical Specification is also exceeded. Also, the total identified leakage is 11.1 gpm so this Technical Specification limit is also exceeded. Please add this to the answer key.

QUESTION 8.03 (2.00)

Using the EPP/Implementing procedures provided, classify the following events. Consider each separately. For each event STATE:

a. The classification
b. The justification for the classification
c. The Tab No. of the implementation procedure

1) During a routine surveillance of the No. 1 Emergency Diesel Generator on your shift, an electrical fault caused a fire in the output breaker. The breaker cannot be repaired for at least 24 hours. (1.00)

2) Chemistry has just reported the following Steam Generator microcurie/gram Dose Equivalent I-131 activity levels: 1A, 0.13 microcuries/gram; 1B, 0.09 microcuries/gram; 1C, 0.10 microcuries/gram. (1.00)

ANSWER 8.03 (2.00)

(Three at 0.33 each)

a. Site Area Emergency
Affects safety systems necessary for shutdown
Tab 26 EPP/I-1

(Three at 0.33 each)

b. Unusual Event
Steam Generator activity level exceeds 0.10 microcurie/gram equivalent I-131
Tab 6 EPP/I-1

REFERENCE

EPP/Implementing Procedure EPP/I-1
KA 194001 A1.16 4.4*
194001A116 ... (KA'S)

COMMENT

8.03.a Since no information was given in the question on whether problems existed with the electrical distribution systems other than the #1 Diesel Generator, and no information was given regarding the duration of the fire, or what mode the plant is in, not enough information is given to positively classify this event. Depending on assumptions, many different classifications are possible. A site area emergency and the resultant activation of utility, county, state, and federal response facilities is not necessary for one inoperable Diesel Generator. We ask that this question be deleted from the exam due to insufficient information provided in the question.

QUESTION 8.04 (1.80)

You are the on-duty Nuclear Shift Supervisor. The RCS is at normal Tave with a Reactor Startup about to commence when the maintenance foreman informs you that the 1A containment Hydrogen Recombiner failed its six (6) month surveillance test. He estimates that he may be able to repair the unit in twelve (12) hours. He also noted that the 1B Recombiner passed its surveillance with no problem. WHAT impact, if any, will this have on the plant startup and why.

ANSWER 8.04 (1.80)

The startup must be terminated (0.6) plant is about to enter mode 2 (0.6) and mode changes cannot be made with reliance on an action statement (0.6).

REFERENCE

BVPS LP-SQS-46.1 EO No. 7
KA 028000 G0.11 3.5
BV TS 3.6.4.2
028000G011 ... (KA'S)

COMMENT

8.04 This question cannot be answered without referencing the applicable Technical Specification to determine whether Technical Specification 3.04 is applicable or not. Since the Technical Specification was not provided we ask that the question be deleted from the exam.

QUESTION 8.05 (2.70)

Answer the following questions concerning "Operating Shift Complement and Functions" as discussed in BVPS OM Chapter 48, Procedure B.

a. The plant is operating at 100% power, you are the on-shift shift supervisor. Your NCO has become seriously ill and you have sent him to the hospital. Can the plant continue to operate? If so, for how long and under what conditions/guidelines. (1.20)

b. Who has the authority to restrict access to the Control Room? (0.50)

c. List four (4) of the seven (7) people that shall be granted access to the Control Room regardless of the situation. (1.00)

ANSWER 8.05 (2.70)

a. Yes (0.20). The plant can continue to operate for 2 hours (0.50) provided that immediate action is taken (callout) to return to the minimum shift complement (0.50).

b. The Nuclear Shift Supervisor (0.50)

c. Nuclear Operating Supervisors (Any 4 at 0.25 each)
Assistant Plant Manager
Plant Manager
Senior Manager, Nuclear Operations
Manager, Nuclear Safety
Vice President, Nuclear Operations
Senior Vice President, Nuclear Group

REFERENCE

BVPS LP-SQS-48-1 EO No's 1 & 5
KA 194001 A1.03 3.4
BVPS OM 1/2.48.1 Procedure B
028000G005 194001A103 ... (KA'S)

COMMENT

8.05 c Some of the job titles have been changed. Please change Senior Manager, Nuclear Operations to General Manager, Nuclear Operations, Senior Vice President, Nuclear Group to Executive Vice President, Power Generation, and Vice President, Nuclear Operations to Vice President, Nuclear Group in the answer key.

QUESTION 8.10 (2.75)

Answer the following concerning Safety Limits and Limiting Safety System Setpoints (LSSS).

a. WHAT are the TWO (2) Safety Limits and WHAT parameters must be monitored to determine if a Safety Limit has been exceeded. (1.25)

b. What is the BASES for the LSSS reactor trip setpoint values specified in Table 2.2-1 of the Technical Specifications? (0.50)

c. What notifications must be made if a Safety Limit is exceeded. Identify those notifications that are required to be made within one hour or less. (1.00)

ANSWER 8.10 (2.75)

a. RCS Pressure (2735 psig) (0.25)

Reactor Core (0.25)
Thermal Power (0.25)
Pressurized Pressure (0.25)
Auctioneered High Tave (0.25)

b. To ensure that the Reactor Core and the RCS do not exceed their Safety Limits. (0.50)

c. Reported to:
NRC (0.25) within one hour (0.25)
Senior Manager Nuclear Operations (within 24 hours) (0.25)
ORC (within 24 hours) (0.25)

REFERENCE

KA 028000 G0.05 3.6
BVPS question bank no. 8.1
BVPS TS Section 2.0, 2.2, and 6.7
028000G006 ... (KA'S)

COMMENT

8.10.c The question asks for notifications required within one hour or less if a safety limit is exceeded. The answer key lists two notifications required in 24 hours. Please delete these from the answer key.

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

FACILITY: BEAVER VALLEY 1&2

REACTOR TYPE: PWR-WEC3

DATE ADMINISTERED: 88/08/17

EXAMINER: BRIGGS, L.

CANDIDATE: MASTER

INSTRUCTIONS TO CANDIDATE:

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

CATEGORY VALUE	% OF TOTAL	CANDIDATE'S SCORE	% OF CATEGORY VALUE		CATEGORY
25.00	25.00	________	________	5.	THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS
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 (1.50)

a. The plant is currently in Mode 5 with one train of RHR in operation. Assume a nominal RHR flow of 4000 gpm and a reduction in temperature of 8 F across the RHR heat exchanger. The reactor engineer informs you that his calculated decay heat load is 0.3% of rated power. Assume that you have maximum component cooling water flow and minimum RHR bypass flow. With the above plant conditions is the one train of RHR removing all the decay heat being generated at th time? Show your calculations and state your assumptions. (1.50)

QUESTION 5.02 (3.00)

A Boron dilution TO 750 ppm increases the source range [illegible]ation to 132 CPS. During the dilution, Xenon concentration changed. [illegible]T was the Xenon's REACTIVITY contribution during this evolution?

INITIAL plant conditions:

Mode 3, BOL
Boron concentration, 900 ppm
All shutdown banks withdrawn
Actual reactivity present in the core is minus 4% delta K/K
Source range counts is 100
Differential Boron worth is minus 10 pcm/ppm

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

QUESTION 5.03 (3.50)

a. Explain both HOW AND WHY the following factors affect differential boron worth (more negative, less negative or no change).

1. Boron concentration increase (0.75)

2. Moderator temperature decrease (0.75)

3. Fission product buildup (0.75)

4. Core burnup from MOL to EOL with constant rod position (0.75)

b. Why does the critical boron concentration drop rapidly from 0 to 150 MWD/MTU of burnup as seen in Figure 1? (0.5)

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

QUESTION 5.04 (2.00)

When would a rod be worth more - if it were dropped while at power or if it were stuck out while all other rods were inserted? EXPLAIN.

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

QUESTION 5.05 (2.80)

a. Briefly explain WHY the hot calibrated pressurizer level instruments do not provide accurate pressurizer level indication when the plant is in mode 5 (cold shutdown). Include in your discussion the direction of error. (1.50)

b. Briefly explain what happens in the pressurizer to help maintain RCS pressure during an outsurge of water. (1.30)

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

QUESTION 5.06 (2.00)

The plant has just tripped from 100% power due to a loss of all RCPs, after an extended run (4 months) at full power. Natural circulation has been established with the following parameters:

Decay heat 2.5% of full power
Th is 590 F
Tc is 552 F

a. Calculate the percentage of full flow through the core. (0.50)

b. What are THREE conditions that will enhance natural circulation and explain WHAT each condition does to help natural circulation. (1.50)

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

QUESTION 3.07 (3.20)

a. How is DNBR affected (increase, decrease or no change) by the indicated changes of the following and Briefly explain WHY. Consider each case separately. Initial plant condition is operating at 95% power (1.20).

1. Boron concentration in the RCS is decreased by 10 ppm
2. Pressurizer heaters are energized
3. Reactor power is increased to 100%

b. Using the attached TS figure 2.1-1, Reactor Core Safety Limit explain (briefly) for each of the three sections (A,B & C) what parameter is being limited (1.20).

c. If section A of the safety limit curve is exceeded what protection would be lost and WHY (0.80).

QUESTION 5.08 (2.00)

For each of the following evolutions, state HOW (increase, decrease or no change) SHUTDOWN MARGIN (per TS definition) would be changed from the INITIAL plant condition given by going to the NEW plant condition. Assume that all systems are in automatic.

a. The plant is in MODE 5 when a charging pump is accidentally started resulting in the injection of 100 gallons of boric acid into the RCS.

b. The plant is in MODE 4 with all shutdown bank rods withdrawn to the full out position when plant temperature is increased by 50F.

c. The plant is at 500 F when all shutdown bank rods are withdrawn.

d. The plant is at 50 % power when a control rod drops into the core, the reactor DOES NOT trip.

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

QUESTION 5.29 (~~3.00~~)

a. When loading the core (fueling or refueling) neutron detector position is important. Is it conservative or non-conservative to load the core TOWARD the detector? Explain WHY. (2.00)

b. How does the rate of positive reactivity insertion rate affect the source range count level at criticality? Explain WHY. (1.00)

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

QUESTION 5.10 (2.00)

Answer the following questions TRUE or FALSE.

a. Delayed neutrons are more likely to escape resonance capture than prompt neutrons.

b. Effective delayed neutron fraction changes over core life due to the buildup of Plutonium and the depletion of U-235.

c. Effective delayed neutron fraction increases over core life.

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

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

QUESTION 6.01 (1.75)

Answer the following questions regarding the Residual Heat Removal System:

a. During a cooldown using RHR, how is cooldown rate controlled? (0.50)

b. How is minimum RHR flow ensured when adjusting cooldown rate? (0.50)

c. What pressure and temperature conditions must be satisfied to open the RHR inlet valves (MOV-1RH-700 & 701)? (0.50)

d. What condition will auto-close the RHR inlet valves? (0.25)

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

QUESTION 6.02 (2.00)

Answer the following questions concerning the Fire Protection Systems.

a. What device is used to prevent discharge of CO2 into an area where personnel are working? (0.50)

b. What is an alternate use for the 5 Ton CO2 (HP turbine enclosure) system besides fire protection? (0.50)

c. What mechanism causes automatic "Wet Pipe" (pressurized) sprinkler actuation to occur? (0.50)

d. How (what mechanism(s)) is the operator notified of an automatic actuation of an individual "wet pipe" sprinkler system? (1.00)

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

QUESTION 6.03 (3.00)

Listed below are valves that position in response to an ESF signal. Identify the exact signal (SIS, CIA, or CIB) that actuates the valve and whether the valve position, after ESF actuation, is open or closed.

a. CCR to RCP 1B (TV-1CC-103B)
b. RCS Letdown (TV-1CH-204)
c. Stm Gen 1C Blowdown (TV-1BD-100C)
d. ORS Spray Pump 2B Suction from containment (MOV-1RS-155B)
e. RCP Seal Return (MOV-1Ch-~~369~~), 370 378
f. Prim Grade Water to PRT (1RC-72)
g. PZR Vapor Space Sample (TV-1SS-112A1)
h. Coolant Charging System (MOV-1Ch-289)
i. Main Condenser Air Eject Vent (TV-1SV-100A)
j. BI Tank to Cold Legs (MOV-1SI-867C)

QUESTION 6.04 (2.00)

Answer the following questions regarding the Control Area Ventilation Systems.

a. High concentrations of WHAT gas will cause Control Room Pressurization Air Compressor (1VS-C-3) to trip. (0.50)

b. How MANY compressed air storage tanks, as a minimum, are needed to supply the Control Room with 100 percent (400 scfm) of its required air supply during a CIB isolation, and briefly explain WHY the 400 scfm flowrate is required. (1.00)

c. What component serves as emergency backup in the event of a loss of refrigeration capability of the Compressor-Condenser units, 1VS-E-4A or 4B? (0.50)

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

QUESTION 6.05 (2.60)

a. Pressurizer pressure transmitters (PT-455, 456 and 457) provide six (6) functions in addition to indication on BB-B1. List four (4) of those functions. (1.60)

b. Before entering solid plant operations the low pressure overpressure protection system is required to be enabled. What conditions will actuate the annunciator to alert the operator that the overpressure protection system has not been enabled. Include any applicable setpoints in your answer. (1.00)

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

QUESTION 6.06 (2.20)

The reactor is operating at a steady state 25% power, all control systems are in automatic. Turbine load is increased to 100% and the controlling steam pressure detector for #1 S/G sticks at the 25% value. Explain how and why this will effect #1 steam generator level. Assume no operator action. Continue your explanation to steady state conditions or plant trip. State any assumptions made.

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

QUESTION 6.07 (3.00)

Answer the following questions concerning the reactor protection (RPS) and engineered safeguards features system (ESFS).

a. Why are the feedwater lines isolated by a safety injection signal? (0.50)

b. What function does P-12 perform below its setpoint? (0.50)

c. What accident is each of the following trips designed to protect against and can the trip be bypassed/blocked? Specification of auto or manual is not required. (2.00)
   1. Power range high Positive Neutron flux rate trip
   2. Pressurizer high water level trip
   3. Pressurizer low pressure trip

QUESTION 6.08 (2.50)

Answer the following questions concerning the Reactor Coolant Pumps.

a. What is the purpose of the No. one (1) seal bypass valve and when (under what four (4) conditions) may it be opened? (1.50)

b. What type of seal is the No.1 seal. Why is a minimum differential pressure of 200 psid across the No. 1 seal required for RCP operation? (1.00)

QUESTION 6.09 (3.00)

The following questions relate to the auxiliary feedwater system.

a. What are four of the six conditions that will start the electric driven AFW pumps (FW-P-3A AND 3B) (1.00)

b. What four conditions will prevent auto start of the electric driven AFW pumps (FW-P-3A AND 3B) (1.00)

c. Auxiliary feed throttle valves, MOV-1FW-151A,B,C,D,E,F receive AUTO open signals from what components? (0.50)

d. Briefly explain how, when using the dedicated AFW pump (FW-P-4), feed flow and steam generator levels (component and location(s)) are controlled. (0.50)

QUESTION 6.10 (2.45)

a. List two (2) control rod interlocks that will inhibit automatic rod withdrawal only and explain the reason for each interlock. (1.20)

b. What are the five (5) automatic functions of an URGENT FAILURE in the rod control system. (1.25)

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

QUESTION 7.01 (2.50)

a. What are the two reasons for stopping all RCP's in the case of a small break LOCA? (1.00)

b. What is the definition of adverse containment conditions? (1.00)

c. What is the reason for specifying different process parameter values under adverse containment versus normal containment conditions? (0.50)

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

QUESTION 7.02 (3.30)

Answer the following concerning E-Ø, Reactor Trip or Safety Injection:

a. The Main Turbine has not tripped and you attempt a manual trip as required, with no response. What additional actions are you required to take in order to shutdown the turbine? (1.ØØ)

b. List three plant conditions that require SI initiation? Include setpoints. (Ø.75)

c. What four parameters are checked to determine if SI flow should be terminated? (1.25)

d. Following an SI reset, what condition must be met before an automatic reinitiation of SI will occur? (Ø.5)

QUESTION 7.03 (3.00)

Answer the following concerning the EOPs' and Status Trees rules of usage.

a. How does the operator know if the sequential performance of subtasks within a procedure is required? (0.50)

b. After entering E-0, WHEN does monitoring of the STATUS TREE's beg ? List two circumstances. (1.00)

c. How is a particular task that must be completed before proceeding to the next step identified? (0.50)

d. What action is required to be performed if while performing a Functional Recovery Procedure (FRP) that addresses an Orange terminus for Core Cooling an Orange terminus for Heat Sink is encountered. Briefly explain your answer. (1.00)

QUESTION 7.04 (2.60)

Answer the following questions concerning E-3, Steam Generator Tube Rupture.

a. A NOTE prior to step 41 alerts the operator to run RCPs in order of priority. What is the REASON for the preferred order of RCP operation. Be SPECIFIC. (0.80)

b. Step 41 tells you to start a RCP if none are running. It also tells the operator that, prior to starting the RCP, the pressurizer level is to be raised to greater than 65% (90% adverse) and subcooling is to be increased, based on core exit Tcs, to 25F plus subcooling listed on attachment 3. What are the BASES for these requirements and WHAT could occur if these actions are not taken prior to starting the RCP? (2.00)

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

QUESTION 7.05 (3.25)

Answer the following questions regarding FR-S1, Response to Nuclear Power Generation/ATWS.

a. What are THREE (3) indications that a trip HAS occurred? (0.75)

b. What is the basis for tripping the turbine. (0.50)

c. Why is tripping the turbine within 30 seconds important for an ATWS coincident with a loss of normal feedwater? (0.50)

d. What actions are required under immediate action step 6, Initiate Emergency Boration of RCS? (1.50)

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

QUESTION 7.06 (2.00)

In ECA-0.1, Loss of all AC Power Recovery Without SI Required, Step 16 requires the operator to verify natural circulation. What FIVE (5) parameters and their assciated indications/trends would be checked by the operator?

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

QUESTION 7.07 (2.75)

The plant is operating at 100 percent power. Urgent repair of a motor operated valve must be performed inside containment to prevent a plant shutdown. The repair is expected to take one (1) hour. The maintenance man that is supposed to do the repair is 25 years old, his lifetime exposure on his NRC Form 4 through the last quarter is 33,000 mrem. He has received an additional 700 mrem this quarter. The area that he will be working in has a radiation field of 675 mrem/hr. gamma and 450 mrem/hr. neutron.

a. Can the man selected complete the task without exceeding any NRC (1.75) exposure limits. Show all calculations and state any assumptions made.

b. Technical Specifications, section 6.12, requires personnel that enter high radiation areas exercise one of three specified options. Briefly list TWO of those options. (1.00)

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

QUESTION 7.08 (2.20)

Answer the following questions concerning Technical Specification Bases for Refueling Operations.

a. What is the reason that irradiated fuel cannot be moved in the reactor vessel unless the reactor has been shutdown for at least 150 hours. (0.60)

b. Why must at least one RHR loop be in operation when in mode 6? (1.00)

c. What is the reason for the minimum water level above the reactor vessel flange? (0.60)

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

QUESTION 7.09 (3.00)

Answer the following questions concerning FR-H.1, Response to Loss of Secondary Heat Sink.

a. What is the reason for the Caution preceding the first step which states that feed flow should not be established to any faulted steam generator if a non-faulted SG is available. (1.00)

b. Step 3 has the operator stop all RCPs if NO feed flow can be established. What is the reason for this? (1.00)

c. What is the reason for the Caution preceding step 19 which states if RWST level drops to less than 20 feet the SI system should be aligned for cold leg recirculation using ES-1.3. (1.00)

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

QUESTION 8.01 (2.40)

The plant is operating at 75% power and the latest leak rate data shows:

13.2 GPM - Corrected RCS leakage rate
1.5 GPM - Leakage into the Pressurizer Relief Tank
1.2 GPM - Leakage into the Primary Drains Transfer Tank
3.4 GPM - Leakage through SI-23, RCS Loop 1A, cold leg isolation
(Previous leakage rate was 1.6 GPM)
Ø.8 GPM - Total primary to secondary leakage
4.2 GPM - Leakage past RCP seals

What RCS leakage limits, if any, have been exceeded? Refer to attached Technical Specifications. Show ALL work and STATE any assumption.

(***** CATEGORY Ø8 CONTINUED ON NEXT PAGE *****)

QUESTION 8.02 (3.00)

BVPS - OM 1/2.48.3 discusses THREE (3) types of status boards/prints maintained in the control room.

a. LIST the 3 types of control room status boards/prints. (0.75)

b. BRIEFLY discuss the PRIMARY PURPOSE of each. (1.50)

c. LIST the person(s) responsible for updating (maintaining) each control room status board/print. (0.75)

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

QUESTION 8.03 (2.00)

Using the EPP/Implementing procedures provided, classify the following events. Consider each separately. For each event STATE:
1. The classification
2. The justification for the classification
3. The Tab No. of the Implementation procedure

a. During a routine surveillance of the No. 1 Emergency Diesel Generator on your shift, an electrical fault caused a fire in the output breaker. The breaker cannot be repaired for at least 24 hours. (1.00)

b. Chemistry has just reported the following Steam Generator (1.00) microcurie/gram Dose Equivalent I-131 activity levels: 1A, 0.13 microcuries/gram; 1B, 0.09 microcuries/gram; 1C, 0.10 microcuries/gram.

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

QUESTION 8.04 (1.80)

You are the on-duty Nuclear Shift Supervisor. The RCS is at normal Tave with a Reactor Startup about to commence when the maintenance foreman informs you that the 1A containment Hydrogen Recombiner failed its six (6) month surveillance test. He estimates that he may be able to repair the unit in 12 hours. He also noted that the 1B Recombiner passed its surveillance with no problem. WHAT impact, if any, will this have on the plant start-up and why.

QUESTION 8.05 (2.70)

Answer the following questions concerning "Operating Shift Complement and Functions" as discussed in BVPS OM CHAPTER 48, Procedure B.

a. The plant is operating at 100% power, you are the on-shift shift supervisor. Your NCO has become seriously ill and you have sent him to the hospital. Can the plant continue to operate? If so, for how long and under what conditions/guidelines. (1.20)

b. Who has the authority to restrict access to the control room? (0.50)

c. List four (4) of the seven (7) people that shall be granted access to the control room regardless of the situation. (1.00)

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

QUESTION 8.06 (2.40)

Answer the following questions concerning "Control Of Operating Procedures" as discussed in BVPS OM 1/2.48.2 D.

a. Under what condition can a On The Spot Operating Manual Change Notice (OMCN) be made to an EOP and WHO makes that decision. (0.80)

b. Who must approve a proposed OMCN that affects a procedure used on both units? (0.80)

c. How long does OSC and the Plant Manager have to review and approve on the spot changes made to operating procedures? (0.40)

d. How long does a OMCN remain in effect after approval by the Plant Manager? (0.40)

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

QUESTION 8.07 (2.75)

In accordance with DLC SAP Chapter 3B (provided) determine for each of the following events:
Type of notification (1 hr., 4 hr., etc.).
Individual responsible for making the initial notification.
Type of report required and time frame (eg. 30 day special report).

a. The radcon foreman has informed you that through a valving error the low level waste drain tank contents, which were supposed to be pumped to the high level waste drain tank, has been discharged directly to the cooling tower blowdown via the effluent filters. He also states that although the level of activity did not exceed two times the 10 CFR 20 limit he had informed the state department of environmental resources. (1.00)

b. You are the NSS and have just been informed that a construction group working just off site has damaged the ENS phone line. Repairs are expected to take eight (8) hours. You test the ENS line and it does not work. (0.75)

c. Safety Injection Tank SI-TK-1A has developed significant backleakage from the RCS. As the NSS you have decided to close its discharge valve and declare SI-TK-1A inoperable. (1.00)

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

QUESTION 8.08 (2.50)

It was discovered today, August 17, that a Diesel Generator MONTHLY surveillance that was due on Thursday, August 4, was actually performed on Friday, August 12. This surveillance over the last three months had been performed on July 5, June 6, and May 4, 1988.

Is this Diesel Generator considered operable? Explain your answer and the applicable criteria used to determine operability.

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

QUESTION 8.07 (2.70)

Answer the following questions concerning Conduct of Operations as discussed in BVPS OM 1/2.48.1, "Operations Shift Rules of Practice".

a. What initial operator actions should be taken if an instrument or control appears to be operating improperly. (0.90)

b. How can the operator check operability/inoperability of the instrument or control? (0.30)

c. If operability cannot be verified, the operator is authorize to take certain action to prevent/protect the station. What action is authorized and WHAT three (3) items, specifically are, to be prevented/protected. (1.50)

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

QUESTION 8.12 (2.75)

Answer the following concerning Safety Limits and Limiting Safety System Setpoints (LSSS)

a. WHAT are the TWO (2) Safety Limits and WHAT parameters must be monitored to determine if a Safety Limit has been exceeded. (1.25)

b. What is the BASES for the LSSS reactor trip setpoint values specified in Table 2.2-1 of the Technical Specifications? (0.50)

c. What notifications must be made if a Safety Limit is exceeded. Identify those notifications that are required to be made within one hour or less. (1.00)

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

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

Thermodynamics Formulas, Conversions, and Constants

$1\ ft^3 = 7.48$ gal.

1 gal. = 3.78 liters

1 lbm. = 454 grams

1 KW = 738 ft - lbf/sec

1 KW = 3413 BTU/hr

1 HP = 550 ft - lbf/sec

1 HP = 2545 BTU/hr

1 BTU = 778 ft - lbf

1 atm = 14.7 psia

14.7 psia = 29.92" of Hg

29.92" of Hg = 760 mm of Hg

$\dot{M} = \rho \vec{V} A = \rho \dot{V}$

$TDH = \oint g/g_c * \rho(z) * dz$

$h_p = \Delta P/\rho = \Delta P v$

$NPSH = (P_{suc} - P_{sat}) * v_{suc}$

$\eta_p = (h_p * \dot{M})/BHP = \Delta P * \dot{V}/BHP$

$$h_L = u_2 - u_1 = f * \frac{L}{D*2*g_c*\rho^2*A^2} * \dot{V}^2$$

$\dot{Q} = Mc_p*(\Delta T/\Delta t)$

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

$\dot{Q} = \dot{M}\Delta h$

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

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

$\dot{Q} = \sigma AeT^4$

$\dot{Q}_{net} = \sigma A(T_1^4 - T_2^4)F_a * F_e$

$$\dot{Q} = \frac{\Delta T}{\frac{L_1}{k_1 Am_1} + \frac{L_2}{k_2 Am_2} + \ldots \frac{L_n}{k_n Am_n}}$$

$k_{air} = 1.4$

$R_{air} = 53.3$ ft - lbf/lbm - °R

$\sigma = 0.172 * 10^{-8}$ BTU/hr - ft^2 - °R^4

mole = $6.023 * 10^{23}$ particles

$g_c = 32.2$ (lbm - ft)/(lbf - sec^2)

$r_m = (r_2 - r_1)/\ln(r_2/r_1)$

$\Delta T_m = (\Delta T_a - \Delta T_b)/\ln(\Delta T_a / \Delta T_b)$

$$U = \frac{1}{(\frac{1}{h_1} + \frac{L}{k} + \frac{1}{h_2})}$$

$$U = \frac{1}{\frac{1}{h_1 * \frac{r_1}{r_2}} + \frac{1}{k * \frac{r_m}{r_2}} + \frac{1}{h_2}}$$

$\phi = P_v/P_{sat}$

$CPR = P_c/P_{in} = (2/(k + 1))^{k/(k - 1)}$

$Q\% = (h - h_f) * 100\%/h_{fg}$

$\eta_{cycle} = w_{net}/q_{source}$

$A.E. = q_{source} - T_{sink}\Delta S_{source}$

$C_p = C_v + R/778$

$Pv = RT$

$P_1 v_1^k = P_2 v_2^k$

$$\frac{T_2}{T_1} = (\frac{P_2}{P_1})^{(k - 1)/k} = (\frac{v_1}{v_2})^{(k - 1)}$$

$\Delta u = c_v \Delta T$

$\Delta h = c_p \Delta T$

$$u_1 + P_1 v_1 + \vec{V}_1^2/2g_c + z_1 g/g_c + q_{12} = u_2 + P_2 v_2 + \vec{V}_2^2/2g_c + z_2 g/g_c + w_{21}$$

EQUATION SHEET

$f = ma$

$v = s/t$

Cycle efficiency $= \frac{\text{Net Work (out)}}{\text{Energy (in)}}$

$w = mg$

$s = v_o t + 1/2at^2$

$E = mC^2$

$a = (v_f - v_o)/t$

$KE = 1/2mv^2$

$v_f = v_o + at$

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

$PE = mgh$

$W = v\Delta P$

$\omega = \theta/t$

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

$\Delta E = 931\Delta m$

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

$\dot{Q} = \dot{m}\,\Delta h$

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

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

$\dot{Q} = UA(T_{ave} - T_{stm})$

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

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

$Pwr = W_f\,\dot{m}$

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

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

$TVL = 1.3/\mu$

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

$HVL = 0.693/\mu$

$SUR = 26.06/T$

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

$T = 1.44\ DT$

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

$$SUR = 26\,\frac{\lambda_{eff}\,\rho}{\bar{\beta} - \rho}$$

$CR_1(1 - K_{eff})_1 = CR_2(1 - K_{eff})_2$

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

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

$M = (1 - K_{eff})_o/(1 - K_{eff})_1$

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

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

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

$\ell^* = 1 \times 10^{-5}$ seconds

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

$\lambda_{eff} = 0.1$ seconds^{-1}

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

$I_1 d_1 = I_2 d_2$

$\Sigma = N\sigma$

$I_1 d^2{}_1 = I_2 d^2{}_2$

$$LMTD = \frac{\Delta T_2 - \Delta T_1}{\ln \Delta T_2/\Delta T_1}$$

<u>WATER PARAMETERS</u>

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

$R/hr = (0.5\ CE)/d^2$ (meters)
$R/hr = 6\ CE/d^2$ (feet)

<u>MISCELLANEOUS CONVERSIONS</u>

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 Btu = 778 ft-lbf
1 inch = 2.54 cm
°F = (9/5°C) + 32
°C = 5/9 (°F - 32

SS CONT COPY CB-13

INFORMATION COPY

UNIT
CYCLE 7

CRITICAL BORON CONCENTRATION • CORE BURNUP
ARO, HFP, EQ XENON

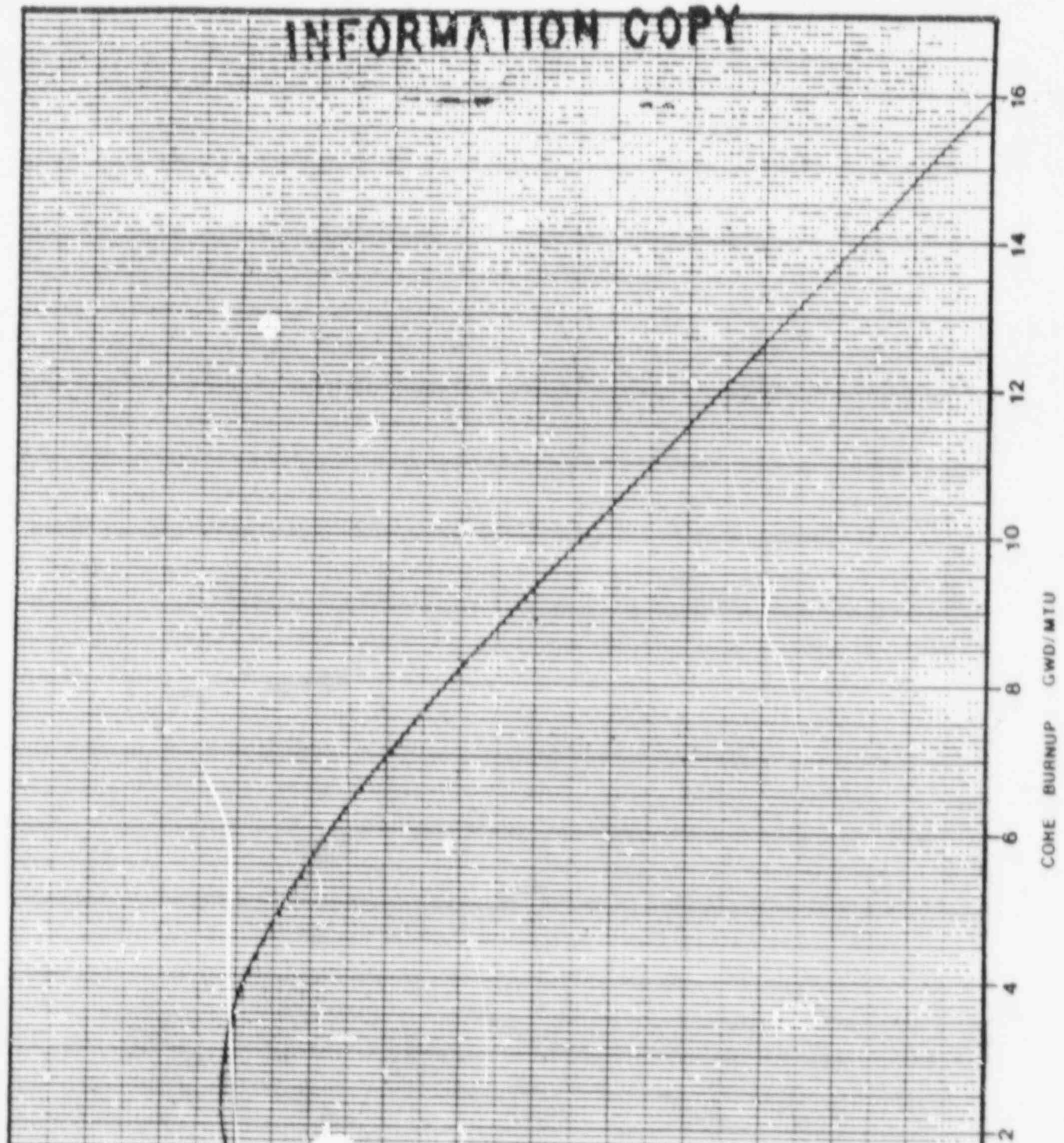


FIG. 1

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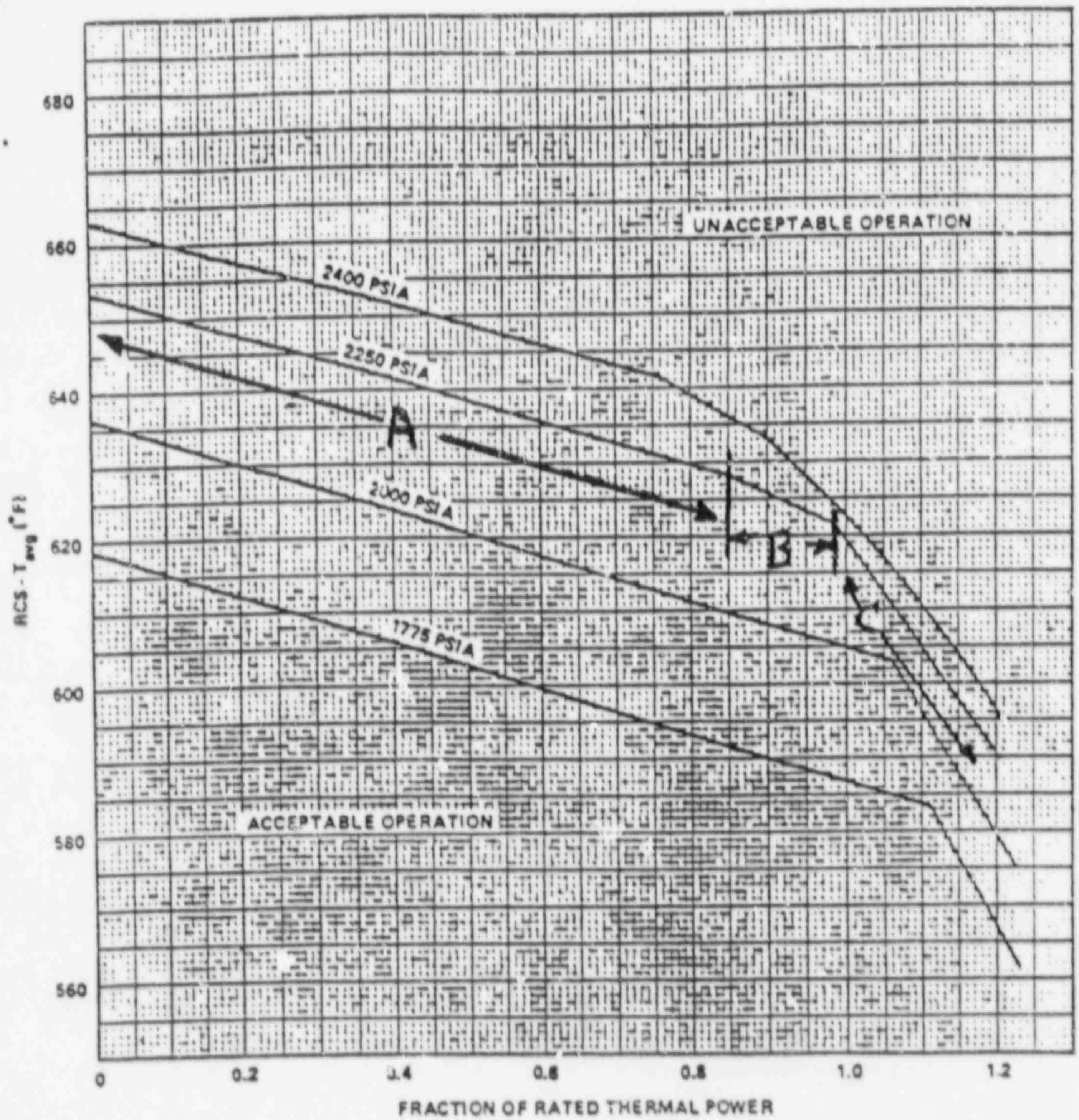

FIGURE 2.1-1 REACTOR CORE SAFETY LIMIT - THREE LOOPS IN OPERATION

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 1 GPM total primary-to-secondary leakage through all steam generators not isolated from the Reactor Coolant System and 500 gallons per day through any one steam generator not isolated from the Reactor Coolant System,

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 28 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2230 $\pm$20 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere particulate and gaseous radioactivity monitor at least once per 12 hours.

SURVEILLANCE REQUIREMENTS (Continued)

b. Monitoring the containment sump discharge at least once per 12 hours.

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2230 $\pm$ 20 psig at least once per 31 days with the modulating valve full open,

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation, and

e. Monitoring the reactor head flange leakoff temperature at least once per 24 hours.

REACTOR COOLANT SYSTEM

PRESSURE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.4.6.3 Reactor coolant system pressure isolation valves shall be operational.

APPLICABILITY Modes 1,2, 3 and 4.

Action:

1. All pressure isolation valves listed in Table 4.4-3 shall be functional as a pressure isolation device, except as specified in 2. Valve leakage shall not exceed the amounts indicated.

2. In the event that integrity of any pressure isolation valve specified in Table 4.4-3 cannot be demonstrated, reactor operation may continue, provided that at least two valves in each high pressure line having a non-functional valve are in, and remain in, the mode corresponding to the isolated condition.[a]

3. If Specification 1 and 2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

4. The provision of specification 4.0.4 is not applicable for entry into Mode 3 or 4.

[a] Motor operated valves shall be placed in the closed position and power supplies deenergized.

4.4.6.3.1 Periodic leakage testing [(a)] on each valve listed in Table 4.4-3 shall be accomplished prior to entering Mode 1 after every time the plant is placed in the cold shutdown condition for refueling, after each time the plant is placed in a cold shutdown condition for 72 hours if testing has not been accomplished in the preceding 9 months and prior to returning the valve to service after maintenance, repair or replacement work is performed.

4.4.6.3.2 Whenever integrity of a pressure isolation valve listed in Table 4.4-3 cannot be demonstrated the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of the other closed valve located in the high pressure piping shall be recorded daily.

(a) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

TABLE 4.4-3

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

System	Valve No.	Maximum[(a)(b)] Allowable Leakage
Loop 1, cold leg	SI-23	$\leq$ 5.0 GPM
	SI-12	$\leq$ 5.0 GPM
Loop 2, cold leg	SI-24	$\leq$ 5.0 GPM
	SI-11	$\leq$ 5.0 GPM
Loop 3, cold leg	SI-25	$\leq$ 5.0 GPM
	SI-10	$\leq$ 5.0 GPM

(a) 1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are considered unacceptable.

(b) Minimum test differential pressure shall not be less that 150 psid.

ANSWER 5.01 (1.50)

a. 4000 gpm x 60 min/hr x 1 cu. ft/7.48 gal x 1 lb./.0166 cu.ft
= 1930000 lbs/hr [0.3]
Q = mc(delta-T) = 1930000 lbs/hr x 1 BTU/lb-F x 8 F = 15440000 BTU/hr
[0.3]
Q/(3413000 BTU/hr/MW) = 4.52 MW [0.3]
4.52 MW/2652 MW = 0.17% power[0.3]
No, it is not removing all decay heat [0.3]

REFERENCE
BVPS Thermodynamics Manual Chapter 3
BVPS LP-TMU-3 EO No.7
BVPS System Description Chapter 10
KA 191006 K1.03 2.3 193007 K1.08 3.4
191006K103 193007K108 ...(KA'S)

ANSWER 5.02 (3.00)

Keff1 = 1/(1 - rho1)
Keff1 = 1/(1 - (-0.04)) = 0.9615 [0.5]

CR1 (1 - Keff1) = CR2 (1 - Keff2)
100 (1 - 0.9615) = 132 (1 - Keff2) Keff2 = 0.9709 [0.5]

rho2 = (0.9709 - 1)/0.9709 = -0.03 [0.5]

delta rho = rho2 - rho1 = -0.03 - (-0.04) = 0.01 = 1000 pcm [0.5]

Boron delta rho = -150 ppm x -10 pcm/ppm = 1500 pcm [0.5]

Xenon delta rho = 1000 pcm - 1500 pcm = -500 pcm [0.5]

REFERENCE
BVPS Reactor Theory LP EO 4.1 and 5.1
BVPS Rx THEORY TEXT Pgs 43 thru 45 Ch. 5
KA 001000 K5.28 3.8 192003 K1.02 2.3
001000K528 192003K102 ...(KA'S)

ANSWER 5.03 (3.50)

a. 1. Delta boron worth becomes less negative (0.25) due to increased competition for neutrons by more boron atoms (0.5).

2. Delta boron worth becomes more negative (0.25) because more neutrons are thermalized due to denser moderator and since boron is a 1/v absorber, the probability of absorption increases (0.5).

3. Delta boron worth becomes less negative (0.25) due to increased competition for neutrons by the poison atoms (0.5).

4. Delta boron worth becomes more negative (0.25) due to reduced boron concentration from MOL to EOL (0.5).

b. Negative reactivity caused by the buildup of Xe and Sm (0.5).

REFERENCE
BVPS Reactor Theory Manual Chapter 8, p 34, 45, 37
KA 001000 K5.20 3.2 001000 K5.28 3.8 001000 K5.30 3.1 192 007 K1.04 3.4
BVPS Rx THEORY LP EO Para. 8.1
001000K520 001000K528 001000K530 192007K104 ...(KA'S)

ANSWER 5.04 (2.00)

The stuck rod would be worth more (0.5). Reactivity worth is proportional to the relative flux squared (0.5). For a dropped rod, the flux is depressed adjacent to it (0.5) whereas if the same rod was stuck out, while the others were inserted, it would be exposed to a much higher flux than the flux in the rest of the core (0.5).

REFERENCE
BVPS Rx Theory Manual chapter 8 pgs. 14-16
KA 000003 EK1.03 3.8 000005 EK1.05 4.1 192005 K1.05 3.1
000003K103 000005K105 192005K105 ...(KA'S)

ANSWER 5.05 (2.80)

a. Pzr. fluid temp. is much lower in cold S/D which increases the density of the liquid [0.5]. Level is determined by a comparison of height times weight (density) of the pressurizer fluid to that of the reference leg [0.5] (which basically remains constant). Since the cold water is more dense the indicated level will be higher than actual level [0.5].

b. The steam bubble expands which causes the press. and temp. of the steam space to start to drop [0.5]. When this happens the fluid at the liquid steam interface is at a temperature above Tsat for the new pressure and it will flash to steam [0.5]. This tends to hold pressure up [0.3]

REFERENCE
BVPS THERMO TEXT Pgs.5,8 & 9
BVPS LP-TMO-7 No. 5 & 3
KA 011 000 K4.07 3.2 010 000 K5.01 4.0 193 001 K1.03 2.6
010000K501 011000K407 193001K103 ...(KA'S)

ANSWER 5.06 (2.00)

a. Q = v3/30
v = [3/) x 2.3]. 3 = 4.1% of full flow.

b. 1. Maintain heat sink [0.25]. To maintain thermal driving head [0.25].
2. Maintain coolant inventory (pressurizer level) [0.25]. Helps prevent formation of vapor pockets in the system [0.25].
3. Maintain subcooling (50F if possible) [0.25]. Prevents steam binding of coolant loops [0.25].

REFERENCE
KA 193 008 K1.23 4.1 000 015 EK1.01 4.6 000 055 EA2.02 4.6
BVPS THERMO TEXT Pgs 23 thru 25
BVPS LP-TMO-7 No. 16
000015K101 000055A202 193008K123 ...(KA'S)

ANSWER 5.07 (3.20)

a. 1. Decrease [0.2], would increase Tave which reduces the subcooling margin [0.2].
   2. Increase [0.2], would increase RCS press and increase subcooling margin [0.2]. O.N, NO CHANGE IF SPRAY IS BALANCING HEATERS e.g., PRESS. CONT.
   3. Decrease [0.2], would increase heat flux and reduce subcooling margin [0.2].

b. Section A. Limits ave. coolant temp. (Enthalpy) at core exit to less than saturation [0.4].

   Section B. Limits hot channel core exit quality to 15% or less [0.4].

   Section C. Limits DNBR to 1.3 or greater (for normal ops and mod. freq. incidents) [0.4]

c. All trips that rely on Delta Temperature [0.4]. Because if Th becomes saturated there will be no change in hot leg temperature and will no longer be an indication of power [0.4].

REFERENCE
KA 003000 K5.01 3.9 002000 K5.09 4.2 002000 G0.05 4.1 193008 K1.05 3.6
BVPS THERMO TEXT Pgs. 17 thru 20
BVPS LP-TMO-7 EO 12 and 14
002000G005 002000K509 003000K501 193008K105 ...(KA'S)

ANSWER 5.08 (2.00)

a. Increase (0.50 points each)
b. Increase
c. No change
d. No change

REFERENCE
BVPS Rx THEORY LP REACTIVITY BALANCES Para. 9.1
BVPS Rx THEORY TEXT Pgs 10 thru 27
KA 192 002 K1.14 3.9
192002K114 ...(KA'S)

ANSWER 5.09 ~~(3.00)~~ 2.50

a. Non-conservative [0.5]
As the core is loaded toward the detector the detector becomes dominated by core flux [0.5] and does not see changes due to subcritical multiplication until it overcomes the background core flux [0.5]. ~~Early extrapolation of the 1/M plot results in a late prediction of criticality [0.5].~~

b. The faster the reactivity insertion rate the lower the count rate at criticality [0.5] due to the reduced time for subcritical multiplication [0.5].

REFERENCE
BVPS Rx THEORY LP PARA. 5.1 AND PG.15
BVPS Rx THEORY TEXT PG 47 and 40
KA 192 008 K1.04 3.8 015 000 K5.06 3.7
015000K506 192008K104 ...(KA'S)

ANSWER 5.10 (2.00)

a. False

b. True

c. False

d. True

REFERENCE
BVPS Rx THEORY LP PARA. 5.1
BVPS Rx THEORY TEXT Pgs. 11,13 & 17
KA 192 003 K1.07 3.0
192003K107 ...(KA'S)

ANSWER 6.01 (1.75)

a. Adjusting RHR cooler flow rate valve (MOV-1RH-758) [0.50]

b. RHR bypass valve (MOV-1RH-605) will open or close to maintain flow when adjusting RHR flow control valve (1RH-758) (if in automatic). [0.50]

c. RCS pressure < 430 psig [0.25]
Pzr vapor space temp < 475F [0.25]

d. RCS pressure > 630 psig [0.25]

REFERENCE
BV-1, Chpt 10, Sec 1, pg 2 & Sec 2, pg 6
KA 005000 K4.07 3.5 005000 K4.03 3.2
BVPS LP-SQS-10.1 EO No.3 & 5
005000K403 005000K407 ...(KA'S)

ANSWER 6.02 (2.50)

a. Manual Lockout Switch [0.5]

b. A means to purge air or Hydrogen manually from the generator [0.5]

c. Fusible links in each sprinkler head will melt (above 165F) and cause sprinkler actuation. [0.5]

d. A pressure switch will initiate an area "Fire" annunciator in the control room [0.5] and a local water gong (horn for PAB El 752 and 768) [0.5]

REFERENCE
KA 086000 K4.04 3.4 086000 A3.02 3.3 086000 A4.03 3.4
BVPS LP-SQS-33.1 EO No.3
BVPS OM 1.33.1 Pgs. 3 THRU 5
086000A302 086000A403 086000K404 ...(KA'S)

ANSWER 6.03 (3.00)

a.	S	CIB	f.	~~S CIA~~ NON ESF VALVE		(0.15 for position)
b.	S	CIA	g.	S	CIA	(0.15 for signal)
c.	S	CIA	h.	S	SIS	
d	O	CIB	i.	S	CIB	
e.	S	CIA	j.	O	SIS	

REFERENCE
BV-1, ch. 47, Table 1
KA 103000 A3.01 4.2
BVPS LP-SOS-47.2 EO No.4
103000A301 ...(KA'S)

ANSWER 6.04 (2.00)

a. Carbon Monoxide [0.5]

b. Four (4) tanks [0.5]
To maintain a positive pressure in the control room [0.5]

c. River water cooling coils, 1VS-E-14A and 14B [0.5]

REFERENCE
KA 013000 K1.13 3.1
BVPS OM 1.44A.1 Pgs. 10 thru 15
BVPS LP-SOS-44A.1 EO No.1 & 2
013000K113 ...(KA'S)

ANSWER 6.05 (2.60)

(any 4 at 0.4 each)
a. Low press. Rx trip
High press. Rx trip
Safety Injection actuation
Accumulator Discharge Valve Block (2000 psig)
PORV actuation block below setpoint (2000 psig)
Low press. alarm ACCEPT OTΔT AND P11

b. RCS pressure [0.25] less than 250 psig [0.25] AND either keylock switch not in the automatic position [0.25] OR if either pressurizer relief line isolation valve (MOV-1RC-535 or 537) not fully open [0.25]

REFERENCE
BVPS OM 1.6.1 Pg. 59 & 54
BVPS LP-SQS-6.4 EO No.2 AND PG.12
KA 010 000 K4.03 4.1 010 000 G0.07 3.4
010000G007 010000K403 ...(KA'S)

ANSWER 6.06 (2.20)

m-stm = K(P-stm)(delta-P)

Since the steam pressure component stays constant (it should go down) while the delta-P increases, indicated steam flow will be higher than actual steam flow. (0.6)

The summing network for flow will send a signal to the flow controller to open the feed regulating valve. (0.5)

As level starts to increase, the level error will signal for the FRV to close (0.5)

Eventually, ~~the flow error will be cancelled out by the level error, and~~ the FRV will be positioned such that steam generator level will operate at a steady state level. ~~higher than desired program.~~ (0.6)
PROGRAMMED

REFERENCE
BV-1, Chpt 24, Section 1, pgs 17-20
BVPS LP-SQS-24.1 EO No.6
KA 059000 K1.04 3.4 059000 A3.02 3.1
059000A302 059000K104 ...(KA'S)

ANSWER 6.07 (3.00)

a. To prevent excessive cooldown of the reactor coolant system [0.25] and Reduce the consequences of a steam line break in containment [0.25]

b. Blocks the steam dumps (TCV 106 SERIES)and [0.25]
Allows manual bypass of the steam dump block to the three cooldown valves [0.25]

c. 1. Rod ejection accident of low worth rods from mid power [0.25]
cannot be bypassed/blocked [0.25]
2. Backup to high pressure trip [0.25] and to prevent discharge of water through pressurizer safety valves [0.25]
Trip can be blocked (auto <P7) [0.25]
3. Protect from low press. which could lead to DNBR < 1.3 [0.25] and Limit necessary range of protection from OT delta T [0.25]
Trip can be blocked (auto <P7) [0.25]

REFERENCE
BVPS OM 1.1.1 Pgs. 5 thru 9
BVPS OM 1.21.1 Pg 9
BVPS TS BASES 3/4.3.1 and 2
BVPS LP-SQS-1.1 EO 1 and 2
KA 012 000 K4.02 4.3 012 000 K4.06 3.5 006 000 G0.06 4.0*
006000G006 012000K402 012000K406 ...(KA'S)

ANSWER 6.08 (2.50)

a. To ensure adequate cooling and lubrication to the radial bearing [0.50]
(4 at 0.25 each)
No. 1 seal leakoff or RCP bearing temperatures approach alarm limits
RCS pressure is >100 but <1000 psig
No. 1 seal leakoff flow is less than 1 gpm
Seal injection flow is between 6 and 13 gpm

b. Controlled leakage film riding face seal [0.50]
200 psid is the minimum pressure that can float the seal [0.25] and ensure enough flow to cool and lubricate the seal [0.25]

REFERENCE
BVPS OM 1.6.4 Pg 7
BVPS EXAM BANK No 6-12
BVPS LP-SQS-6.3 Pgs. 5 thru 8 and EO 4 AND 5
KA 003 000 K6.02 3.1 003 000 A1.09 2.8
003000A109 003000K602 ...(KA'S)

ANSWER 6.09 (3.00)

(any 4 at 0.25 each)

a. 1. control switch to start
   2. Two of three SG lo-lo level
   3. Both main feed pumps tripped
   4. ~~Opposite~~ TURBINE DRIVEN AFW start signal not followed by discharge pressure
   5. Safety Inj. signal (train A for 3A and B for 3B)
   6. PNL-AMSAC

(~~4~~ ANY at 0.25 each)

b. Control switch in pull-to-lock or stop
   Bus loss of voltage (1AE or 1DF)
   Diesel loading sequence (not timed to start yet)
   Motor electrical protection
   BREAKER RACKED OUT

c. Breaker closure of either MD AFW [0.25]
   Steam driven AFW steam supply trip valves (TV-MS-105A or B) not fully ~~open~~ CLOSED [0.25]

d. Feed flow is manually controlled at the feedwater bypass control manifold [0.25]
   Level information is from the control room or the backup indicating panel [0.25]

REFERENCE
BVPS OM 1.24.1 Pgs. 15 thru 18 and 3
BVPS LP-SQS-24.1 EO 7
KA 061 000 K4.02 4.2 061 000 G0.09 3.9
061000G009 061000K402 ...(KA'S)

ANSWER 6.10 (2.45)

a. Low power (C-5) interlock prevents auto rod motion when impulse power is 15 percent or less [0.3] to prevent unstable operation [0.3].

High bank D rod withdrawal stop (C-11) [0.3] to prevent counter misalignment [0.3]

(5 at 0.2] each)

b. 1. Deenergizes lift coils
   2. Energizes both stationary and movable gripper coils at a reduced voltage
   3. Energizes red URGENT FAILURE light on the front panel of the power cabinet
   4. Stops all automatic rod motion
   5. Energizes ROD CONTROL URGENT FAILURE annunciator

REFERENCE
BVPS OM 1.1.1 Pgs. 18 and 48
BVPS OM 1.1.2 Pg. 3
BVPS LP-SQS-1.3 EO 8 & 10
KA 001 010 K4.10 3.4 001 050 A2.01 3.9
001010K410 001050A201 ...(KA'S)

ANSWER 7.01 (2.50)

a. 1. Prevent excessive inventory loss [0.5]
2. Preclude core uncovery from RCP's tripping at a later time [0.5]

b. Containment pressure > 5 psig [0.33] or containment radiation > 100000 R/hr (10E5) [0.33]
or integrated containment radiation > 1000000 R (10E6) [0.33]

c. Adverse containment value incorporates additional instrument error due to adverse environmental factors [0.5]

REFERENCE
BVPS Exec Vol E-0 step 21 pg. 32
BVPS EOP E-0 pg. 13 Step 21
BVPS EXEC VOL GEN. INFO. Pgs. 11 AND 12
KA 000 007 EK 3.23 4.3 000 011 EK 3.14 4.2
BVPS LP-LRT-VII-61 EO A1 and 65 Term. Obj. A.2
BVPS LP-LRT-VII-62 EO A1
000009K323 000011K314 ...(KA'S)

ANSWER 7.02 (3.50)

a. Close main steam trip valves (ANY four at 0.25 pts each)
runback the turbine
close main steam bypass valves
close non-return valves
LOCAL MANUAL TRIP OF TURBINE

b. (1) Pressurizer pressure < 1845 PSIG (ANY three at 0.25 each)
(2) Containment pressure > 1.5 PSIG.
(3) Steamline pressure < 510 PSIG
(4) RCS SUBCOOLING < ATTACHMENT (4)
(5) PRESSURIZER LEVEL < 5%

c. (1) RCS subcooling criteria met [0.25]
(2) Feed flow to intact SG's(> 350 GPM) [0.25] or
Narrow Range level in at least one intact SG [0.25]
(3) RCS pressure (stable or increasing) [0.25]
(4) PZR level (> 5%) [0.25]

d. Reactor trip breakers must be closed (0.5)

REFERENCE
BVPS EOP E-0 pgs. 3, 5, 16, 19
KA 000 007 EK 3.01 4.6 000 038 EK3.09 4.5
BVPS LP-LRT-VII-61 EO B1
LP-LRT-VII-62 EO A1

000007K101 000038K107 ...(KA'S)

ANSWER 7.03 (3.00)

a. Letters denote sequential importance. (bullets do not) [0.5]

b. As directed in E-0 [0.5]
When transferring out of E-0 [0.5]

c. The step or an associated note will explicitly state the requirement. [0.5]

d. No action is required [0.5] Core Cooling is a higher priority status tree than Heat Sink [0.5]

REFERENCE
EOP Ex Vol Chap. 53.B.2 pgs. 2 thru 8
KA 000 007 G0.12 3.9
BVPS LP-LRT-VII-61 TERMINAL OBJECTIVE
000007G012 ...(KA'S)

ANSWER 7.04 (2.80)

a. (Spray is preferred method of pressure control). Spray line connections are provided in only two loops [0.4], an RCP should be run in one of those two loops [0.4].

b. (Prior RCS depressurization without a RCP running could have formed a steam bubble in the Rx vessel head). Starting a RCP could cause a rapid condensation of the bubble [0.4] and draw liquid from the pressurizer and reduce RCS subcooling [0.4].

Also local flashing could occur if subcooling is not adequate [0.4].

These conditions would require SI reinitiation [0.4] and increase leakage into the ruptured Steam Generator [0.4].

REFERENCE
BVPS LP-IRT-VII-67 EO Nos. 3 AND 4
KA 000 038 EK3.06 4.5
BVPS EXEC. VOL. 1.53B.4 EOP E-3 Pgs. 138 & 139
000038K306 ...(KA'S)

ANSWER 7.05 (3.25)

(three at 0.25 each)

a. Reactor trip and bypass breakers indicate open
   Neutron flux indicates decreasing
   Rod bottom lights lit
   RPI at zero

b. Prevent uncontrolled cooldown of RCS [0.5]

c. Maintain Steam Generator inventory [0.5]

d. Start charging/HHSI pumps [0.25]

   Align boration path:
      Open (MOV-CH-350), Emer. Boration valve (BB-A) [0.25]
      Start in-ser BAT pump in FAST speed (BB-A) [0.25]

   Align charging flow:
      Open (FCV-CH-122), charging flow control valve [0.25]
      Verify letdown flow to BRS if VCT level increases to dive
      point [0.25]

   Check pressurizer pressure less than 2335 psig [0.25]

REFERENCE
KA 000 029 G0.11 4.6 000 029 EK3.06 4.3 000 029 EK3.12 4.7
BVPS EOP FR-S1 Pg.4
BVPS EOP EXEC. VOL. 1.53.4 Pg.64
BVPS LP-LRT-VII-61 EO B1 & B4
000029G011 000029K306 000029K312 ...(KA'S)

ANSWER 7.06 (2.00)

(five at 0.4 each)

1. RCS subcooling based on core exit TCs-greater than subcooling listed in Attachment 1
2. SG pressure-stable or decreasing
3. RCS Thot-stable of decreasing
4. Core exit TCs-stable or decreasing
5. RCS Tcold-at saturation temp. for SG pressure

REFERENCE
BVPS EOP ECA-0.1 Pg. 13
BVPS LP-LRT-VII-51 EO A.2.
KA 000 0550 EK1.02 4.4 000 056 EK1.01 4.2

ANSWERS -- BEAVER VALLEY 1&2 -88/08/17-BRIGGS, L.

000033K103 000056K101 ...(KA'S)

ANSWER 7.07 (2.~~75~~) .50

a. 5(N-18) = 35rem lifetime exposure allowed [0.25]
lifetime to date = 33000 mrem + 700 mrem = 33700 mrem [0.25]
total lifetime available = 35000 mrem - 33700 mrem = 1300 mrem [0.25]
total quarterly exposure available = 3000mrem - 700 mrem = 2300 mrem [0.25]
~~Lifetime exposure is more limiting [0.25]~~ 0
675mrem/hr gamma + 450mrem neutron = 1125mrem/hr total [0.25]

Yes, man can perform task.[0.25]

(any 2 at 0.5 each)

b. 1. Carry a radiation monitoring instrument which continuously indicates dose rate in the area.
2. Carry/wear a radiation monitoring device which integrates the dose rate and alarms at a preset value.
3. Be accompanied by a qualified individual who is equipped with a radiation monitoring device.

REFERENCE
KA 194 001 K1.03 3.4
BVPS LP-RC-01 EO 19, 23, 27
194001K103 ...(KA'S)

ANSWER 7.08 (2.20)

a. To allow short-lived fission products to decay [0.5]. The 150 hour decay time is used in the accident analysis [0.1].

b. 1. To ensure sufficient cooling capacity to remove decay heat and maintain the reactor less than 140F as required in the refueling mode [0.5].

2. To maintain sufficient circulation to minimize the effects of a boron dilution incident and prevent boron statification [0.5].

c. To ensure the depth is sufficient to remove 99% of the assumed 10% iodine gap activity released from a rupture of an irradiated fuel assembly [0.5]. The 23 feet of water is used in the accident analysis [0.1].

REFERENCE
KA 034 000 G0.06 3.4 034 000 K1.02 3.2
BVPS LP-FHP-1.0 EO No. 10

BVPS TS Sect. 3/4.9.? tbl. 10 and applicable Bases
034000G006 034000K102 ...(KA'S)

ANSWER 7.09 (3.00)

a. Reestablishment of feed flow may result in thermal and/or mechanical shock to the tubes that could result in leakage or tube rupture [0.33]. If this occurred in a faulted SG, leakage through the fault could not be controlled until the fault was corrected [0.33]. Feeding to a non faulted SG would allow control of the release if tube leakage occurs [0.33].

b. To reduce heat input to the RCS [0.33] which extends the effectiveness of the remaining water inventory in the SGs [0.33] and allows more time before the operator must initiate feed and bleed [0.33].

c. Feed and bleed depletes RWST inventory [0.33]. At 20 feet there should be sufficient water in the recirc. sump for SI suction [0.33] and the remaining RWST water is reserved for spray pump usage [033].

REFERENCE
BVPS EOP EXEC. VOL 1.53B.4 FR-H.1
BVPS EOP FR-H.1
BVPS LP-LRT-VII-65 EOs B.3 & 4
KA 000 038 EK3.06 4.3 000 038 EK3.08 4.2
000038K306 000038K308 ...(KA'S)

ANSWER 8.01 (2.40)

RCS Pressure Isolation Valve Limits exceeded [0.70]
(3.4-1.6)/(5.0-1.6) = 1.8/3.4 = > 50% [0.50]

UNIDENTIFIED Leakage limits exceeded [0.70] OR IDENTIFIED LEAKAGE EXCEEDED
(13.2 - (1.5+1.2+3.4+0.8+4.2)) = 2.1 GPM [0.50] OR 13.2 GPM - 2.1 GPM = 11.1 GPM
WILL ACCEPT > 500 GPD S.G. LEAKAGE LIMIT (IF ASSUMPTIONS STATED)

REFERENCE
TS 3.4.6.2; TS 3.4.6.3
KA 002020 G0.05 4.1
BVPS LP-SQS-6.5 EO No.7
002020G005 ...(KA'S)

ANSWER 8.02 (3.00)

(three at 0.25 each)
a. 1. Status System Prints
2. Systems Level Status Board
3. Station Equipment Status Boards

(three at 0.50 each)
b. 1. Plastic covered Valve Oper. No. Diagrams used to indicate system status (on sys. deemed necessary by NSOS).
2. Backlighted status board that is manually lit to indicate a system or train is out of service.
3. Plastic covered board used to list out-of-service Major safety and non safety related equipment.

(three at 0.25 each)
C. 1. Operations person performing the valving or switching operation.
2. Reactor Operator assigned to the control board.
3. The NSOF ensures board is updated by plant operators.

REFERENCE
BVPS LP-SQS-48-1 EO No.7
BVPS OM 1/2.48.3 Pgs.4 thru 6
KA 194001 A1.06 3.4
194001A106 ...(KA'S)

ANSWER 8.03 (2.00)

(three at 0.33 each)
a. ~~Site area emergency~~ B ALERT
FIRE WHICH POTENTIALLY Affects safety systems ~~necessary for shutdown~~ B
Tab 26 EPP/I-1

(three at 0.33 each)
b. Unusual Event
Steam Generator activity level exceeds 0.10 microcurie/gram equivalent I-131
Tab 6 EPP/I-1

REFERENCE
EPP/Implementing Procedure EPP/I-1
KA 194001 A1.16 4.4*
194001A116 ...(KA'S)

ANSWER 8.04 (1.80)

The start-up must be terminated [0.6] plant is about to enter mode 2 [0.6] and mode changes cannot be made with reliance on an action statement [0.6].

REFERENCE
BVPS LP-SQS-46.1 EO No.7
KA 028000 G0.11 3.5
BV TS 3.6.4.2
028000G011 ...(KA'S)

ANSWER 8.05 (2.70)

a. Yes [0.20]. The plant can continue to operate for 2 hours [0.50] provided that immediate action is taken (callout) to return to the minimum shift complement [0.50].

b. The Nuclear Shift Supervisor [0.50]

c. Nuclear Operating Supervisors (any 4 at 0.25 each)
Assistant Plant Manager
Plant Manager
GENERAL ~~Senior~~ Manager, Nuclear Operations UNIT
Manager, Nuclear Safety
Vice President, Nuclear ~~Operations~~ GROUP
EXECUTIVE ~~Senior~~ Vice President, ~~Nuclear Group~~ POWER GENERATION

REFERENCE
BVPS LP-SQS-48-1 EO No's 1 & 5
KA 194001 A1.03 3.4
BVPS OM 1/2.48.1 PROCEDURE B
0230000005 194001A103 ...(KA'S)

ANSWER 8.06 (2.40)

a. If re-validation of the EOP would not be required [0.4]
The Unit Nuclear Operating Supervisor [0.4]

b. One SRO from each unit [0.4] and one additional member from the plant management staff [0.4]

c. 14 days [0.4]

d. 90 days [0.4]

REFERENCE
KA 194001 A1.01 3.4
BVPS LP-SQS-48.1 EO No.9
BVPS OM 1/2.48.2 B
194001A101 ...(KA'S)

ANSWER 8.07 (2.75)

a. ~~4~~ 1 hour notification (via red phone) (No.~~30~~ 31) [0.25]
NSS/NSOF [0.25]
30 day [0.25] ~~special report~~ LER [0.25]

b. 1 hour report (No.117) [0.25]
NSS/NSOF [0.25]
No report required [0.25]

c. 1 hour report (No.127) [0.25]
NSS/NSOF [0.25]
30 day [0.25] LER [0.25]

REFERENCE
KA 194001 A1.05 3.8 194001 A1.08 3.1
BVPS LP-SOS-48.1 EO No.19
BVPS SAP 3B
194001A105 194001A108 ...(KA'S)

ANSWER 8.08 (2.50)

The Diesel is operable [0.5]

Surveillance interval cannot exceed 25% of the specified interval [0.5] or

3.25 times the specified interval for three (3) consecutive surveillance intervals [0.5]

The monthly was performed within the allowed 38.75 days (31 X 1.25 = 38.75)
July 5 to August 12 is 38 days [0.5]

May 4 to August 12 is 100 days which is within the allowed 100.75 days
(31 X 3.25 = 100.75) [0.5]

REFERENCE
KA 064000 G0.05 3.9 064000 G0.11 3.9
BVPS TS Para. 4.0.2
BVPS LP-SOS-36.2 EO No.5
064000G005 064000G011 ...(KA'S)

ANSWER 8.09 (2.70)

a. Believe the instrument [0.3] Respond conservatively [0.3] or allow control to take action if in the conservative and safe direction [0.3].

b. By comparison with two redundant instruments reading the same parameter that agree with each other [0.3].

c. Any action necessary [0.3], including tripping the reactor [0.3], to:
   1. prevent exceeding a TS safety limit [0.3]
   2. protect station personnel [0.3]
   3. Prevent damage to station equipment [0.3]

REFERENCE
BVPS LP-SQS-48-1 CO No.6
BVPS OM 1/2.48.1 PROCEDURE D, PARA. F
KA 194001 A1.13 4.1
194001A113 ...(KA'S)

ANSWER 8.10 (2.75)

a. RCS Pressure (2735 psig) [0.25]

Reactor Core [0.25]
Thermal Power [0.25]
Pressurizer Pressure [0.25]
Auctioneered High Tave [0.25]

b. To ensure that the Reactor Core and the RCS do not exceed their Safety Limits. [0.50]

c. Reported to:
NRC [0.25] within one Hour [0.25]
Senior Manager Nuclear Operations (within 24 hours) [0.25]
ORC (within 24 hours) [0.25]

REFERENCE
KA 028000 G0.05 3.6
BVPS Question bank No. 8.1
BVPS TS SECTION 2.0, 2.2 AND 6.7
028000G005 ...(KA'S)