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

EXAMINATION REPORT NO. 50-412/88-06 (OL)

FACILITY DOCKET NO. 50-412

FACILITY LICENSE NO. NPF-73

LICENSEE: Duquesne Light Company
Post Office Box 4
Shippingport, Pennsylvania 15077

FACILITY: Beaver Valley Unit 2

EXAMINATION DATES: February 23-24, 1988

CHIEF EXAMINER: for Edward Yachimiak, Operations Engineer, DRS — 30 Mar 88 Date

APPROVED BY: for Peter W. Eselgroth, Chief, PWR Section, Operations Branch, DRS — 30 Mar 88 Date

SUMMARY: One Senior Reactor Operator (SRO) candidate was administered written and operating examinations. Both parts of the examination were completed successfully and a license was issued.

REPORT DETAILS

TYPE OF EXAMINATION: Replacement

EXAMINATION RESULTS: One (1) SRO candidate passed both the written and operating portions of the examination.

CHIEF EXAMINER AT SITE: E. Yachimiak, NRC

OTHER EXAMINERS: R. Temps, NRC

Personnel Present at the Exit Meeting

NRC Personnel

R. M. Gallo, Chief, Operations Branch
R. Temps, Operations Engineer
E. Yachimiak, Operation Engineer

Facility Personnel

A. J. Morabito, Manager, Nuclear Training
T. W. Burns, Director, Operations Training
T. D. Noonan, Plant Manager
T. E. Kuhor, Nuclear Operations Instructor

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

U. S. NUCLEAR REGULATORY COMMISSION *Attachment 1*
SENIOR REACTOR OPERATOR LICENSE EXAMINATION

FACILITY: BEAVER VALLEY 2

REACTOR TYPE: PWR-WEC3

DATE ADMINSTERED: 88/02/23

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
24.00 * ~~25.00~~	25.00	________	________	5.	THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS
24.50 * ~~25.00~~	25.00	________	________	6.	PLANT SYSTEMS DESIGN, CONTROL, AND INSTRUMENTATION
23.50 * ~~25.00~~	25.00	________	________	7.	PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
24.50 * ~~25.00~~	25.00	________	________	8.	ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS
96.5 * ~~100.0~~		________ Final Grade	________%		Totals

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

____
Candidate's Signature

* Category and total point values were changed due to question deletion and/or point redistribution due to answer key changes.

LJ

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

uring the administration of this examination the following rules apply:

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

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.

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

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

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

Use only the paper provided for answers.

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

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.

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

. Skip at least three lines between each answer.

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

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

. 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.

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

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

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

. 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.

3. 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 (3.00)

a. Explain, in terms of neutron flux, WHY a dropped rod could be worth approximately 200 pcm whereas a stuck rod could be worth 1000 pcm, even though the same rod could be considered in both cases. (2.00)

b. WHAT are TWO (2) reasons for having control rod bank overlap? (1.00)

QUESTION 5.02 (3.00)

a. If critical data is recorded with Control Bank D at 120 steps, Tave at 547 F., and boron concentration at 1000 ppm, WHAT would be the expected SUR if Bank D is raised to 152 steps? Assume Bank D's differential rod worth is 6 PCM/step and lambda is 0.1/sec. Show all work and state all assumptions. (2.00)

b. A control rod falls into the core when reactor power is at 50% at BOL. HOW (Higher, Lower, No Change) would the resultant steady-state Tave been different if this event had occured at EOL? Justify your answer. Assume a reactor trip DOES NOT occur, rods are in MANUAL, and all other systems are in automatic. NO calculations are necessary. (1.00)

QUESTION 5.03 (3.00)

For EACH of the primary paramters listed below, state HOW (Increases, Decreases, No Change) and explain WHY an INCREASE in that parameter affects the DNBR. Assume the other parameters remain constant.

a. Reactor Power
b. Tave
c. Core Flow
d. Pressurizer Pressure

QUESTION 5.04 (4.00)

Given Attachment 1, "Estimated Critical Position Calculation,"

a. Complete ALL blank spaces on the form. (3.00)

b. Calculate WHAT rate (gpm) of boron addition would be needed to change the Estimated Critical Rod Position from Bank D at 75 steps to Bank D at 228 steps, assuming the current time is 0800 on 2/23/88. (1.00)

QUESTION 5.05 ~~(2.00)~~ (1.00)

Answer the following statements concerning Heat Exchanger Operation by responding TRUE or FALSE.

~~a. Once turbulent flow in a heat exchanger has been established, UA becomes approximately a fixed value.~~ delete EJ

~~b. If the ΔT across a heat exchanger is not constant then ΔT_m, the median (average) temperature, is used to accurately calculate the heat transfer rate.~~ delete EJ

c. The heat removal rate for a heat exchanger will Increase if either of the fluid flowrates through the heat exchanger is Increased.

d. The U-tubes of the steam generators can experience thermal shock if the feedwater flowrate is increased rapidly.

QUESTION 5.06 (2.50)

WHAT are FIVE (5) indications that natural circulation has been established after a loss of offsite power occurs.

QUESTION 5.07 (1.00)

During normal plant operations, WHEN does the reactor vessel experience the highest stresses AND WHAT TWO (2) primary parameters can be controlled to limit these stresses?

QUESTION 5.08 (2.00)

a. Do xenon oscillations converge (dampen) more rapidly at BOL or EOL? Justify your answer in terms of reactivity effects.

b. Would the magnitude and frequency of xenon oscillations be Less at 50% power or 100% power? Justify your answer.

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

QUESTION 5.09 (2.00)

For EACH of the following statements below, state HOW (Increase, Decrease, No Change) ACTUAL Shut Down Margin (SDM) would be affected.

a. The plant is in Mode 5 when a charging pump is mistakenly started resulting in the injection of 200 gallons of boric acid into the RCS.

b. The plant is in Mode 3 when all the shutdown bank rods are withdrawn out of the core.

c. The plant status changes from Mode 5 to Mode 4.

d. A control rod drops into the core with the plant in Mode 1 at 50% power. The reactor does not trip.

QUESTION 5.10 (1.00)

Explain HOW the Moderator Temperture Coefficient (MTC) can act to increase reactor power when turbine steam demand Increases. Assume the plant is initially at 75% power with rod control in MANUAL and all other systems in automatic.

QUESTION 5.11 (1.50)

Answer the following statements concerning Pump Operation by responding TRUE or FALSE.

a. If flow through a pump Increases or the temperature of the fluid Increases, the Required Net Positive Suction Head (NPSH) will Increase.

b. When a pump is operated at Run-out conditions, cavitation typically WILL occur.

c. Run-out is limited by motor horsepower.

(***** END OF CATEGORY 5 *****)

QUESTION 6.01 ~~(3.00)~~ (2.50)

The plant is operating at 50% power when a control system hot leg RTD fails high. Does this failure INCREASE, DECREASE, or NOT AFFECT the following: Consider each item independently. Assume no operator action and that all control systems are in automatic.

a. affected channel overpower delta T trip setpoint
b. steam bypass cooldown valves (first bank)
c. charging flow (initially)
d. control rod bank position
e. rod insertion limit setpoint
~~f. affected channel actual overtemperature delta T indication~~

delete Cy

QUESTION 6.02 (3.00)

a. WHAT is the purpose of the high positive AND negative rate reactor protection trips, respectively?

b. WHAT TWO (2) reactor protection trips are automatically reinstated below P-10?

c. In WHAT TWO (2) ways are the SRNI's affected when the logic for P-10 is satisfied?

QUESTION 6.03 (2.50)

Using Attachment 2, OP. Manual Fig. No. 13-2, "Quench Spray System," identify the following components on the attachment as specified in each part below.

a. Highlight the "A" quench spray pump recirculation flowpath back to the RWST. (0.50)

b. Circle the THREE (3) building/area boundaries that the "B" containment quench spray header passes through. (0.75)

c. Circle WHERE the flowrate for the "A" chemical injection pump is measured. (0.50)

d. Circle the THREE (3) valves that realign when the RWST level reaches the level setpoint for 2QSS-LSKK100B-1. (0.75)

QUESTION 6.04 (2.00)

a. HOW (Increase, Decrease, No Change) will an INCREASE in the reference junction temperature effect indicated thermocouple temperature?

b. HOW (High, Low, As Is) will an RTD temperature indication fail if a short circuit occurs across the RTD?

c. WHAT is the major disadvantage of using a Thermowell RTD for RCS wide range temperature measurement?

d. Given the graph shown in Attachment 3, identify the curve which represents the calibration curve for a HOT calibrated instrument.

QUESTION 6.05 (2.00)

For EACH of the following radiation monitors, state the automatic actions which occur, if any, when the monitors alarm HIGH.

a. 2SWS-RQI101 - Component Cooling Service Water

b. 2HVR*RQI104A - Containment Purge

c. 2RMC*RQ201 - Control Room Area

d. 2GWS-RQI102 - Air Ejector Delay Bldg Exhaust

QUESTION 6.06 (3.00)

Answer the following questions concerning the Auxiliary Feedwater System.

a. WHAT are FOUR (4) conditions/signals (including applicable logic) that can cause a Motor Driven Auxiliary Feedwater Pump (MDAFWP) to automatically start? Assume the following conditions have been met:

(1) CS in AUTO
(2) NMEPT of bus undervoltage
(3) Normal power supply breaker E/ is closed (2.00)

b. WHAT TWO (2) plant conditions/signals (including applicable logic) will cause the Turbine Driven Auxiliary Feedwater Pump (TDAFWP) to automatically start? (1.00)

QUESTION 6.07 (3.00)

A significant leak occurs in the reference leg capillary of 2RCS-LT459, pressurizer (PRZR) level transmitter. Assume the plant is at 100% power and all control systems are in automatic with 2RCS-LT459 selected as the controlling channel.

a. State TWO (2) automatic actions which would occur because of this failure. (1.00)

b. State FOUR (4) control room indications that are available to alert the operator of this failure. (2.00)

QUESTION 6.08 (2.00)

WHAT do the following Safety Injection System interlocks prevent?

a. Low Head Safety Injection (LHSI) pump minimum flow recirculation isolation valve [2SIS*MOV8890A] opens when LHSI pump [2SIS*P21A] discharge flow is low.

b. Safety injection accumulator discharge stop valve [2SIS*MOV865A] opens when its control switch is in AUTO and 2 out of 3 pressurizer pressure channels are greater than 2,000 psig.

c. If a valve receives an SI, CIA, or CIB signal, the motor thermal overload interlocks are bypassed.

d. LHSI discharge valves [2SIS*MOV863A,863B] to charging pum suction header opens in AUTO only if:

1. LHSI pump discharge header valve [2SIS*MOV8811A] is fully open, and
2. High Head SI alt. mini flow isolation valves [2CHS*MOV380A/B,383A/B] are shut, and
3. a recirculation mode initiation signal is present.

QUESTION 6.09 (2.00)

WHAT are FOUR (4) sources of Hydrogen in the containment building?

QUESTION 6.10 (2.50)

The plant is stable in Mode 5 with the "A" Residual Heat Removal System (RHS) in service.

a. At WHAT pressure (psig) will the RHS isolate from the RCS? (0.50)

b. WHAT is the design capacity of the RHS suction line relief valve [2RHS*RV721A]? Include ALL applicable information. (0.80)

c. Loss of primary component cooling water can affect WHAT TWO (2) RHS components, when operating? (0.70)

d. Failure of RHS Hx bypass flow control valve, [2RHS*FCV605A] to the closed position will result in a (Increase, Decrease, No Change) to RCS temperature? (0.50)

(***** END OF CATEGORY 6 *****)

QUESTION 7.01 (1.50)

For the following questions assume B.V.P.S. - O.M. 51, Station Shutdown Procedure, is in use.

a. When using condenser steam dumps, WHAT operator action(s) must be taken to cooldown the RCS below the Lo-Lo Tavg setpoint? (0.50)

b. When the Residual Heat Removal System (RHS) is in operation, at least one reactor coolant pump must remain in service until RCS temperature is less than 200 degrees F. WHY? (0.50)

c. If minimum RCS flow requirements CANNOT be met while in Mode 4, the operator's immediate response is to refer to WHAT procedure? (0.50)

QUESTION 7.02 (2.50)

Answer the following statements concerning Refueling by responding TRUE or FALSE.

a. Inching can ONLY be accomplished when the key is set in the "RUN" position and the inching permissive light is illuminated.

b. The emergency pullout cable is used to pull the transfer car back into containment after the conveyor motor is disengaged from the transfer system.

c. If the gripper is engaged while holding a RCCA and the Dillon Load Cell reads greater than 1200 lbs., actuation of the gripper interlock bypass switch will allow the gripper to be disengaged.

d. Before fuel handling operations in the fuel building can commence, the fuel building vent system shall be in service and discharging through at least one train of SLCRS HEPA filters and charcoal adsorbers.

e. If the operator notices a large unexplained change in load on the Dillon readout, he should immediately reverse direction.

(***** CATEGORY 7 CONTINUED ON NEXT PAGE *****)

QUESTION 7.03 (2.00)

Answer the following questions concerning B.V.P.S. procedure AOP-2.1.3, "Continuous Insertion of RCCA Control Bank."

a. WHAT anticipated operational transient could cause a continuous bank insertion of the controlling bank? (0.50)

b. If a malfunction causes a RCCA control bank to insert past the Low-Low insertion limit, WHAT immediate operator action is required? (0.50)

c. If rod control is transferred to Manual and a continuous insertion condition is still present, WHAT TWO (2) operator actions should be performed? (1.00)

QUESTION 7.04 (2.50)

Answer the following questions concerning B.V.P.S. - O.M. AOP-2.6.3, "Loss Reactor Coolant Flow."

a. WHAT THREE (3) symptoms/indications would an operator visually identify in the control room to verify that Annunciator A2-5E, "Reactor Coolant Loop Flow Low," was in the alarmed condition? (1.50)

b. If a partial loss of reactor coolant flow is indicated, between WHAT TWO (2) RPS protective interlocks (including setpoints) is it possible for a reactor trip to occur? (1.00)

QUESTION 7.05 (3.00)

Answer the following questions concerning B.V.P.S. - EOP FR-S.1, "Response to Nuclear Power Generation/ATWS."

a. WHAT are the TWO (2) indications that a reactor trip has NOT occurred? (1.00)

b. WHAT are the THREE (3) operator actions that can be taken to shut down the reactor if a reactor trip CANNOT be verified? (1.50)

c. WHY is a turbine trip required during an ATWS event? (0.50)

UESTION 7.06 (1.50)

a. WHAT procedure (by name) would you consult if annunciator A1-1E, "Containment air part al pressure high-low," alarmed?

b. WHAT could cause containment pressure to slowly increase with little or no humidity increase, and a possible decrease in temperature?

c. If the plant is in Mode 2, and containment pressure, temperature, and humidity ALL begin to increase rapidly, WHAT action should the operator take?

ESTION 7.07 (3.00) ~~(3.50)~~

A condition arises that requires entry into containment at 40% power. The operator entering containment needs to work in a gamma radiation field of 150 mrem/Hr for aprroxiamtely 2.0 hours. The below candidates are presented to you:

Candidate	1	2	3	4
Sex	male	male	female	male
Age	27	38	24	20
Qtr/exposure	-	1000 mrem	500 mrem	1000 mrem
Life exposure	1000 mrem	54730 mrem	5200 mrem	9500 mrem
Remarks	quarterly history unavailable	Form NRC-4 unavailable	3 months pregnant	-

Each candidate is technically competant and physically capable of performing the task. All candidates have a completed Form NRC-4 and have a documented current calender quarter exposure history, with the exceptions for those candidates stated above. Emergency limits do NOT apply. For EACH person, indicate if you would ACCEPT or REJECT that person to perform the task based on EXPOSURE REQUIREMENTS ONLY. Justify EACH answer AND include ALL applicable limits.

ESTION 7.08 (2.00)

Answer the following question concerning B.V.P.S. - O.M. AOP-2.38.1, "Loss of 120 VAC Vital Bus."

WHAT are FOUR (4) automatic actions that an operator can visually verify in the control room if power to 120 VAC Vital Bus 1 is lost? ONLY consider safety system actuations.

(***** CATEGORY 7 CONTINUED ON NEXT PAGE *****)

QUESTION 7.09 (1.00)

WHAT are the normal expected values for Source Range (SR) AND Intermediate Range (IR) Nuclear Instrumentation an operator would expect to see when verifying that the SR has reenergized after a reactor trip from power?

QUESTION 7.10 (3.00)

Answer the following questions concerning B.V.P.S. - O.M. 2.24.2, "Steam Generator Feedwater System."

a. WHAT action must an operator take in order to prevent a reactor trip if a Steam Generator (SG) Feed Pump Auto-Stop annunciator alarms with the plant at 75% power? (0.50)

b. WHAT are FIVE (5) indications/conditions that an operator would verify if a Hi-Hi SG level-trip occured with the plant at 40% power? (2.00)

QUESTION 7.11 ~~(2.50)~~ (1.50)

Answer the following questions concerning Liquid Waste System Operation.

~~a. WHICH TWO (2) flowrates (numerical values NOT required) are used in calculating the Unit 2 Cooling Tower Blowdown Flow when the Unit 2 blowdown flow instrument [2CWS-FR101] is out of service, and a liquid waste discharge is to be made by way of the Unit 1/2 cooling tower blowdown line? (1.00)~~ delete CJ

b. Before sampling the contents of the "A" waste drain tank, WHAT action must be taken by the operator? Include any applicable precautionary setpoints or time related values. (1.00)

c. WHAT action should an operator take if local-liquid waste process effluent [2SGC-RQI100] high alarm actuates AND is verified to be in the alarmed condition? (0.50)

(***** END OF CATEGORY 7 *****)

QUESTION 8.01 (3.00)

Using Attachment 4, classify the following events in accordance with BV-2 EPP/I-1, Recognition and Classification of Emergency Conditions, AND justify your answer and any assumptions. Consider each case separately.

a. B.V.P.S. EOP E-1, "Loss of Reactor or Secondary Coolant," is in use. Pressurizer level is off-scale low and RCS pressure is 1500 psig and decreasing. The reactor was manually tripped because pressurizer level could not be maintained.

b. A turbine trip from 75% power occured and the reactor did not automatically trip (ATWS). The reactor remained critical until an operator manually inserted control rods.

c. A truck carrying Ammonia gas is involved in a collision at the the plant main entrance. Gas is leaking from the truck.

d. An earthquake is registered on-site with the plant in Mode 1. The severe ground motion results in the generation of a missile in the turbine building from the detachment of a LP turbine blade.

QUESTION 8.02 ~~(3.00)~~ (2.50)

Using B.V.P.S. - Unit 2 Technical Specifications, list ALL applicable action statements, by number, for EACH of the following equipment failures. Consider EACH failure independently.

a. The fuel oil transfer pump for Diesel Generator 21 has been found to be inoperable. A reactor startup is in progress with reactor power at 1% and increasing.

b. RHS Heat Exchanger outlet thermocouples, TE606A and B, have been found to be inoperable.

c. Control room bottled air system pressure is found to be at 1500 psig.

QUESTION 8.03 (2.00)

Diesel Generator (DG) 21's operability load test is scheduled for today. The last THREE (3) tests were completed 35, 69 and 102 days ago respectively. The plant is at 100% power. Using B.V.P.S. - Unit 2 Technical Specifications, are DG 21's operability requirements being met? Explain WHY and/or WHY NOT.

(***** CATEGORY 8 CONTINUED ON NEXT PAGE *****)

QUESTION 8.04 (2.00)

Answer the following statements concerning Clearances by responding TRUE or FALSE.

a. The NSS, NSOF, and the STA (or NCO) all must sign the "Authorization for Removal From Service" lines of the Emergency Safeguards Equipment Clearance Checklist.

b. Only the NSS needs to sign the Equipment/Radiation Clearance Log for the clearance to become effective.

c. A Master Clearance can be used to cover maintenance that requires equipment to be operated in order to perform the necessary work.

d. A Caution Tag may be removed by Test Group Personnel without obtaining the NSS/NSOF's permission.

QUESTION 8.05 (2.00)

In accordance with B.V.P.S. Site Administrative Procedure (SAP) 3B, "Reporting Requirements," utilize the Code of Federal Regulations provided to you to determine whether the NRC should be notified within ONE (1) hour or FOUR (4) hours AND indicate WHY by specifying the appropriate section numbers/letters. example: 10 CFR xx.xx (1)(i)(a)

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

b. An Unusual Event is declared in accordance with the Emergency Plan.

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

d. While the plant was in Mode 3, a Safety Injection signal was generated and an estimated 2000 gallons of RWST water was injected into the core.

QUESTION B.06 (2.00)

In accordance with B.V.P.S. OM 2.48.2 Procedure C, "Adherence and Familiarization to Operating Procedures,"

a. WHEN can an operator take action that departs from a license condition or Technical Specifications? (0.75)

b. WHOM, as a minimum, must approve the above actions to be taken? (0.50)

c. Do non-licensed personnel ever have the authority to take independent action(s) that they deem necessary to place the plant in a safe condition? Justify your answer. (0.75)

QUESTION B.07 (2.00)

Match EACH of the following statements (a-d) with the most appropriate report listed (1-4).

a. The oncoming NSS signs this report signifying that he is assuming responsibility for the station.

b. This report contains, in chronological order, the times when the Emergency Plan is implemented and/or radioactive effluents are released.

c. This report is signed by the oncoming Nuclear Operator signifying that he is assuming responsibility for his area of duties.

d. This report is reviewed to ensure familiarity with plant operations during times of watch relief or vacation.

Report Number

1. Shift Operating Report
2. Nuclear Shift Operting Foreman's Report
3. Nuclear Control Operator Report
4. Nuclear Operator's Report

QUESTION 8.08 (2.50)

Utilizing B.V.P.S. - Unit 2 Technical Specifications, state the Containment Isolation Valve surveillance requirements (including ALL applicable time contraints) for EACH of the highlighted portions of the systems indicated below. Assume the Plant is in Mode 3.

a. Chemical and Volume Control System (Attachment 5) (1.10)

b. **Recirculation Spray System (Attachment 6) (0.70)**

c. Containment Area Ventilation System (Attachment 7) (0.70)

QUESTION 8.09 (2.25)

Use B.V.P.S. - Unit 2 Technical Specification Table 3.3-6 and determine WHAT SEVEN (7) Area or Process radiation monitoring instruments must be functional following a LOCA.

QUESTION 8.10 (3.00)

For EACH of the following statements, determine whether the Reactor Coolant System (RCS) chemistry has been maintained within the requirements specified by Technical Specifications. Justify your answer.

a. The plant has been at 100% power for 21 days. Chemistry notifies you that RCS fluoride concentration has increased to 0.20 ppm.

b. The plant is in Mode 5 and is being prepared for a startup. Chemistry notifies you that RCS chloride concentration is 2.00 ppm.

c. The plant is stable in Mode 3 when Chemistry notifies you that dissolved oxygen concentration is 1.50 ppm.

d. The plant was in Mode 1 when Chemistry notified you at 1000 on 2/23/88 that chloride concentration was 0.25 ppm. Attempts to reduce the chloride concentration were unsuccessful. The plant was shutdown at 1800 on 2/24/88.

(***** CATEGORY 8 CONTINUED ON NEXT PAGE *****)

QUESTION 8.11 (1.25)

a. WHAT is the FULL Technical Specification Basis for the RCS operational leakage limit stated in 3.4.6.2c.? (0.80)

b. WHAT Technical Specification (state by number) addresses the surveillance program established to prevent the leakage limits in 3.4.6.2c. from ~~being~~ becoming an operational concern.

Cy

(***** END OF CATEGORY 8 *****)
(********** END OF EXAMINATION **********)

ANSWER 5.01 (3.00)

a. Rod worth is a function of the ratio of local flux to average flux (squared). [0.50] If a rod is dropped with all other rods withdrawn, the dropped rod depresses the local flux relative to the rest of the core so that its worth is small (~ 200 pcm). [0.75] When a rod is stuck with all other rods inserted, the tip of the stuck rod is exposed to a much higher local flux than the rest of the core causing its worth to increase (~ 1000 pcm). [0.75]

b. - to maintain a more uniform differential rod worth
- minimize the possibility of creating a positive delta I [2 X 0.50]
- to ensure that any control rod motion will have some effect on total core reactivity

REFERENCE

B.V.P.S. LP-RT-8 Enabling Objectives 2,8,11
B.V.P.S. Reactor Theory Text Chapter 8 pages 20-24,27
K/A 001000 K5.02 3.4 -
K/A 001000 A2.03 4.2
001000A203 001000K502 ..(KA's)

ANSWER 5.02 (3.00)

a. rho = (152 steps - 120 steps) x 6 pcm/step [0.60]
= 192 pcm [0.20]

SUR = 26 x [rho x lambda / (Beff - rho)] [0.60]
= 26 x [192 pcm x 0.1/sec / (660 pcm - 192)] [0.40]
= 1.07 DPM [0.20]

b. Tave would be higher [0.30] since MTC [0.30] is larger [0.20] at EOL [0.20]

REFERENCE

B.V.P.S. LP-RT-5 Enabling Objectives 6,7
B.V.P.S. LP-RT-5 page 8
K/A 192003 K1.06 3.3
K/A 000003 EK1.16 3.2
000003K116 192003K106 ..(KA's)

ANSWER 5.03 (3.00)

a. Decreases [0.35] because raising power increases the heat flux on the fuel rod, reducing the DNBR [0.40]
b. Decreases [0.35] because the subcooling margin decreases [0.40]
c. Increases [0.35] because more heat can be absorbed by the water [0.40]
d. Increases [0.35] because the subcooling margin increases [0.40]

REFERENCE

B.V.P.S. LP-TMO-7 Enabling Objectives 11,12
B.V.P.S. LP-TMO-7 pages 21,21,23,26
K/A 193008 K1.05 3.6
193008K105 ..(KA's)

ANSWER 5.04 (4.00)

a. See attached ECP calculation.
b. Rod worth of Bank D at 75 steps = -910 pcm [0.25]
boron change required = -910 (pcm) / -9.8 (pcm/ppm) = 93 ppm [0.25]

Using nomograph CB-31:
PPM boron in coolant = 993 ppm
PPM boron addition = 93 ppm
boric acid volume = ~700 gallons [0.25]
required rate = 700/2/60 = 5.8 gpm [0.25]
OR Using nomograph CB-32:
PPM boron in coolant = 993 ppm
boron addition rate = 93/2 = 46.5 ppm/hr [0.25]
boric acid flow = ~5.5 gpm [0.25]

REFERENCE

B.V.P.S. LP-RT-9 Enabling Objective 6
B.V.P.S. - OM 1.50.4
K/A 192008 K1.07 3.6
K/A 194001 A1.08 3.1
194001A108 192008K107 ..(KA's)

F. ESTIMATED CRITICAL POSITION CALCULATION

FORM ECP-1 (Page 1 of 7)

NOTE: Reference Guide in Chapter 49, Section 4, Procedure M.

Tavg Assumed to Equal 547F ± 1F at Startup

A. CRITICAL DATA

PRIOR TO SHUTDOWN	EXPECTED CRITICAL
Date 2 / 22 / 88 Time 0400	Date 2 / 23 / 88 Time 1000
Boron Conc 900 ppm Power 100 %	Boron Conc. ______ ppm
Xenon equil. %	Xenon ______ %
Samarium equil. %	Samarium ______ %
Control Rod Position:	Control Rod Position:
A 228 C 228 B 228 D 228	A 228 C 223 B 228 D 75

B. REACTIVITY BALANCE

	I	II	III
Reactivity Defects	Prior to Shutdown	expected at criticality	Difference I-II
1. Power (Fig. 50-7)	- 1400 [0.25] pcm	- 0 pcm	(±) -1400 pcm [0.20]
2. Control Rods (Fig. 50-8) or Boron (Fig. 50-10)	- 0 pcm	- 910 pcm [0.25]	(±) +910 pcm [0.20]
3. Xenon	- 2850 [0.25] pcm	- 2300 [0.25] pcm	(±) -550 [0.20] pcm
4. Samarium	- 610 [0.25] pcm	- 740 [0.25] pcm	(±) +130 [0.20] pcm
5. Reactivity Change (sum of 1-4) =			(±) -910 pcm

F. ESTIMATED CRITICAL POSITION CALCULATION (continued)

FORM ECP-1 (Page 2 of 7)

NOTE: If Reactivity Change is greater than ±500 pcm. perform I/M plot, Table 50-1.

C. CRITICAL BORON CONCENTRATION (Use if critical boron concentration is desired.)

I	II	III	IV	V
Reactivity Change (B-5)	Boron Worth (Fig. 50-10)	Boron Change I divided by II	Boron Conc at shutdown	Boron Conc for startup III + IV
(±) -910 pcm	-9.8 pcm/ppm [0.20]	(±) +93 ppm [0.25]	900 ppm	993 ppm [0.20]

D. CRITICAL ROD POSITION (Use if critical rod position is desired)

I	II	III	IV
Reactivity Change (B-5)	Reactivity due to Rod Prior to Shutdown (Fig. 50-8)	Reactivity for Criticality (I+II)	Critical Rod Position (Fig. 50-8)
(±) pcm N/A	- pcm N/A	(±) pcm N/A	steps N/A

ANSWER 5.05 ~~(2.00)~~ (1.00) Cγ

delete
~~a. TRUE~~
~~b. FALSE~~ [~~4~~ 2 X 0.50]
c. TRUE
d. TRUE

REFERENCE

B.V.P.S. LP-TMO-3 Enabling Objectives 4,7
B.V.P.S. LP-TMO-3 pages 8,12
K/A 191006 K1.03 2.3
K/A 191006 K1.04 2.7
K/A 191006 K1.07 2.6
191006K107 191006K104 191006K103 ..(KA's)

ANSWER 5.06 (2.50)

1) core exit TCs - stable or decreasing [5 X
2) RCS hot leg temperatures - stable or decreasing
3) RCS cold leg temperatures - at saturation for existing S/G pre
4) RCS subcooling (based on core exit thermocouples) - greater th
subcooling per attachment (7)
5) S/G pressures - stable or decreasing
6) delta T of less than or equal to 60 °F

REFERENCE

B.V.P.S. LP-TMO-7 Enabling Objective 16
B.V.P.S. EOP ES-0.1, "Reactor Trip Response," Attachment 5
K/A 193008 K1.22 4.2
193008K122 ..(KA's)

ANSWER 5.07 (1.00)

during plant cooldown [0.50]
temperature OR cooldown rate [0.25]
pressure [0.25]

REFERENCE

B.V.P.S. - Unit 2 Technical Specifications page B 3/4 4-7
K/A 193010 K1.07 4.1
193010K107 ..(KA's)

(***** CATEGORY 5 CONTINUED ON NEXT PAGE *****)

REFERENCE

B.V.P.S. LP-RT-6 Enabling Objectives 2,9
B.V.P.S. LP-RT-6 page 6
K/A 192004 K1.13 2.9
192004K113 ..(KA's)

ANSWER 5.11 (1.50)

a. TRUE
b. FALSE [3 X 0.50]
c. TRUE

REFERENCE

B.V.P.S. LP-TMD-4 Enabling Objective B
B.V.P.S. LP-TMD-4 pages 6,7
K/A 191004 K1.11 2.4
K/A 191004 K1.12 2.7 -
K/A 191004 K1.15 2.8
191004K115 191004K112 191004K111 ..(KA's)

(***** END OF CATEGORY 5 *****)

ANSWER 6.01 ~~(3.00)~~ (2.50)

a. NOT AFFECT
b. NOT AFFECT
c. INCREASE
d. DECREASE [0.50 X 6]
e. INCREASE
~~f. NOT AFFECT~~ deleted EG

REFERENCE

B.V.P.S. 2LP-SQS-1.1 Enabling Objective 6
B.V.P.S. 2LP-SQS-1.3 Enabling Objective 10,12
B.V.P.S. 2LP-SQS-7.1 Enabling Objective 7
B.V.P.S. 2LP-SQS-21.1 Enabling Objective 4
B.V.P.S. - O.M. 2.01.1 pages 12,20; 2.7.1 page 35;
2.21.1 page 22;2.6.1 page 64
B.V.P.S. - Unit 2 Technical Specifications table 2.2-1
K/A 001050 K5.01 3.6
K/A 004010 A1.01 3.6 -
K/A 041020 A3.02 3.4
041020A302 004010A101 001050K501 ..(KA's)

ANSWER 6.02 (3.00)

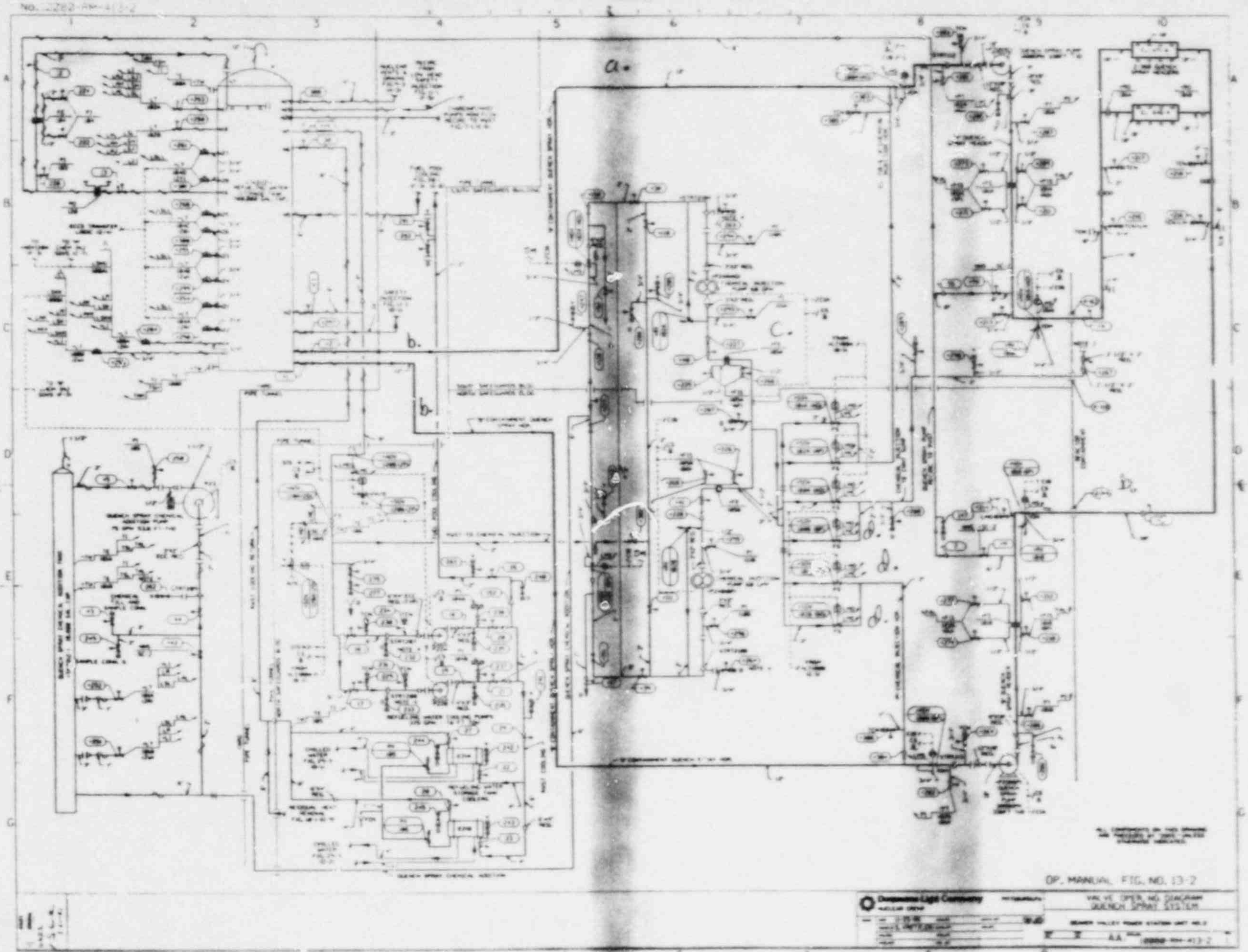
a. the high positive trip protects against a rod ejection accident [0.50] while the high negative trip protects against a rod drop accident [0.50]
b. PR low level reactor trip [0.50] and IR High level reactor trip [0.50]
c. the SR reactor trip signal is automatically blocked [0.50] and the SR high voltage is prevented from energizing [0.50]

REFERENCE

B.V.P.S. 2LP-SQS 2.2 Enabling Objective 4
B.V.P.S. 2LP-SQS 2.2 page 20
K/A 015000 K1.01 4.2
K/A 015000 K4.07 3.8
015000K407 015000K101 ..(KA's)

ANSWER 6.03 (2.50)

a. [0.50]
b. [0.75] Use attached drawing
c. [0.50] as answer key
d. [0.75]


OP. MANUAL FIG. NO. 13-2
Duquesne Light Company
VALVE OPER. NG DIAGRAM
QUENCH SPRAY SYSTEM

REFERENCE

B.V.P.S. 2LP-SQS-13.1 Enabling Objectives 2,4,5
B.V.P.S. OP. Manual Fig. No. 13-2
K/A 194001 A1.07 3.2
194001A107 ..(KA's)

ANSWER 6.04 (2.00)

a. Decrease [0.50]
b. Low [0.50]
c. Thermowell RTDs have a relatively long response time [0.50]
d. A [0.50]

REFERENCE

B.V.P.S. LP-TMO-7 Enabling Objective 5
B.V.P.S. LP-TMO-7 page 11
K/A 191002 K1.13 2.8
K/A 191002 K1.14 2.9 -
191002K114 191002K113 ..(KA's)

ANSWER 6.05 (2.00)

Eg
25A (CNMT PURGE EXHAUST AND AIR SUPPLY ISOLATION DAMPERS)

a. none
b. closes 2HVR*MOD23A and 2HVR*MOD~~23B (applicable valve names acceptable)~~
c. actuates control room pressurization
d. none [4 X 0.50]

REFERENCE

B.V.P.S. 2LP-SQS-43.1 Enabling Objective 4
B.V.P.S. 2LP-SQS-43.1 pages 16,21,24,39
K/A 072000 G0.04 3.7
072000G004 ..(KA's)

ANSWER 6.06 (3.00)

a. 2 S/Gs [0.25] at low-low level [0.25]
SI signal present [0.50]
TDAFWP running [0.25] and pressure low [0.15] after T/D [0.10]
both MFWPs not running [0.25] and either of the MFWPs control switches in Afterstart [0.25]
b. 2/3 [0.25] RCP bus undervoltage [0.25]
2/3 [0.10] detectors in 1/3 S/Gs [0.15] at the low-low level [0.25]

REFERENCE

B.V.P.S. 2LP-SQS-24.1 Enabling Objective 16
B.V.P.S. 2LP-SQS-24.1 pages 16,18
K/A 061000 K4.02 4.6
K/A 061000 K4.06 4.2
061000K406 061000K402 ..(KA's)

ANSWER 6.07 (3.00)

auto actions: backup heaters turn ON [0.50]
flow control valve 2CHS-FCV122 goes to its minimum open position [0.50]

indications:
- level deviation alarm
- one (1) channel of RPS trip status lights for high PRZR level light
- computer alarm
- high level trip alarm annunciator [4 X 0.50]
- high level indication on level meter
- high level indication on level recorder
- flow control valve 2CHS-FCV122 indication at minimum open position

REFERENCE

B.V.P.S. 2LP-SQS-6.2 Enabling Objectives 8,17
B.V.P.S. - OM 2.6.1 page 63 & figures 6-38,6-39
K/A 011000 K1.01 3.9
K/A 011000 K1.04 3.9
K/A 011000 K3.01 3.4
K/A 011000 K5.13 3.4
K/A 011000 A2.10 3.6
011000A210 011000K513 011000K301 011000K104 011000K101
..(KA's)

ANSWER 6.08 (2.00)

a. prevents pump cavitation from occuring [0.50]
b. prevents accumulators from being inoperable [0.50]
c. prevents the valve's motor operator from tripping on thermal overload so that the valve will reach its designated safe position [0.50]
d. prevents pumping contaminated sump water into the RWST [0.50]

***** CATEGORY 6 CONTINUED ON NEXT PAGE *****

REFERENCE

B.V.P.S. 2LP-SQS-9.1 Enabling Objective 6
B.V.P.S. - OM 2.11.1 pages 14,18,19,23
K/A 006000 K4.06 4.2
K/A 006000 K4.09 4.1
K/A 006000 K4.19 3.4
006000K419 006000K409 006000K406 ..(KA's)

ANSWER 6.09 (2.00)

1. metal-water reaction between the zirconium fuel cladding and the reactor coolant
2. pressurizer gas space and RCS water
3. radiolytic decomposition of water collected on the containment floor with a corresponding generation of oxygen
4. radiolytic decompostion of water in the reactor core [4 X 0.50]
5. corrosion of metals by solutions used for emergency cooling or containment spray

REFERENCE

B.V.P.S. 2LP-SQS-46.1 Enabling Objective 11
B.V.P.S. - OM 2.46.1 page 1
K/A 028000 K5.03 3.6
028000K503 ..(KA's)

ANSWER 6.10 (2.00)

a. > 700 psig [0.50]
b. TWO (2) [0.25] charging pumps [0.25]
at the relief valve set pressure [0.30]
c. RHS heat exchanger [0.35]
RHS pump seal cooler [0.35]
d. Decrease [0.50]

REFERENCE

Enabling Objectives UNAVAILABLE
B.V.P.S. - OM 2.10.1 pages 1,2,5,6,20,21
K/A 000025 K1.01
K/A 000025 K3.02
K/A 000025 A1.01
000025A101 000025K302 000025K101 ..(KA's)

(***** END OF CATEGORY 6 *****)

ANSWER 7.01 (1.50)

a. place steam bypass interlock selection switch to the DEFEAT TAVG position [0.50]
b. prevent reactor vessel void formation (maintain RCS subcooling) [0.50]
c. B.V.P.S. - E.O.P. ES-0.2, "Natural Circulation Cooldown" [0.50]

REFERENCE

B.V.P.S. 2LP-SQS-21.1 Enabling Objectives 4;
2LP-SQS-50.51.52.1 Enabling Objectives 2,3
B.V.P.S. - O.M. 2.51.4 pages C9,D2,D4; 2.51.2 page 3; 2.53C.4 page 3
K/A 005000 G0.10 3.5
K/A 005000 G0.15 3.9
K/A 041020 A4.08 3.1
041020A408 005000G015 005000G010 ..(KA's)

ANSWER 7.02 (2.50)

a. FALSE
b. TRUE
c. FALSE [5 X 0.50]
d. TRUE
e. FALSE

REFERENCE

B.V.P.S. 2LP-FHP-1.0 Enabling Objectives 8,9,12
B.V.P.S. 2LP-FHP-1.0 pages 4,28,31,44,47
K/A 034000 A1.01 3.2
K/A 034000 A3.01 3.1
K/A 034000 G0.07 3.7
034000G007 034000A301 034000A101 ..(KA's)

ANSWER 7.03 (2.00)

a. turbine runback (OTdt or OPdt) OR load rejection [0.50]
b. emergency boration OR boration at concentration and flowrate atleast that as stated in Technical Specifications [0.50]
c. trip the reactor [0.50] and go to E-0 [0.50]

(***** CATEGORY 7 CONTINUED ON NEXT PAGE *****)

REFERENCE

B.V.P.S. - O.M. 53C AOP-2.1.3 page 1
B.V.P.S. - O.M. 1 page AAM1
K/A 001000 A1.04 3.9
K/A 001000 A3.02 3.6
001000A302 001000A104 ..(KA's)

ANSWER 7.04 (2.50)

RCP amps low Ey

a. RCP A,B, or C bright lights illuminated
   Low RCS flow indication in Loop A,B, or C [3 X 0.50]
   high delta T
b. P-8 [0.25] 30% [0.25]
   P-7 [0.25] 10% [0.25]

REFERENCE

B.V.P.S. 2LP-SQS-53C.1-Enabling Objectives 1,4
B.V.P.S. - O.M. AOP-2.6.3 page 1
K/A 000015 G0.11 3.6
000015G011 ..(KA's)

ANSWER 7.05 (3.00)

a. rod bottom lights NOT lit [0.50]
   neutron flux NOT decreasing [0.50]
b. manually trip the reactor
   manually insert control rods [3 X 0.50]
   initiate emergency boration
c. to prevent excessive cooldown of the RCS [0.50]

REFERENCE

B.V.P.S. 2LP-SQS-53A.1 Enabling Objectives 2,3
B.V.P.S. - EOP FR-S.1 page 1
K/A 000029 EK3.06 4.3
K/A 000029 G0.11 4.6
000029G011 000029K306 ..(KA's)

(***** CATEGORY 7 CONTINUED ON NEXT PAGE *****)

ANSWER 7.06 (1.50)

a. Loss of Containment Vacuum (AOP-2.12.1) [0.50]
b. a breach of (leakage into) containment [0.50]
c. manually trip the reactor [0.50]

REFERENCE

B.V.P.S. 2LP-SQS-53C.1 Enabling Objectives 1,3
B.V.P.S. - O.M. AOP-2.12.1 page 1
K/A 000029 EA2.01 4.3
K/A 000029 G0.11 4.2
000069G011 000069A201 ..(KA's)

ANSWER 7.07 ~~(3.50)~~ (3.00)

Re is allowed 1250 mrem/QTR whole body dose

#1 - ~~REJECT~~ ACCEPT [0.25] since ~~he has no quarterly history available and would exceed the 200 mrem/qtr whole body limit~~ [0.50]
#2 - REJECT [0.25] since he does not have a Form NRC-4 available and would exceed the 1250 mrem/qtr whole body limit [0.50]
#3 - REJECT [0.25] since she will exceed the allowable exposure limit during the term of her pregnancy [0.50] has already exceeded
#4 - ~~ACCEPT~~ [0.25] since he ~~will not exceed the quarterly limit [0.50] or the~~ whole body limit of 10000 mrem lifetime exposure [0.50]
Ris

REFERENCE

Enabling Objectives UNAVAILABLE
B.V.P.S. - R.C.M. pages 5,6,7
K/A 194001 K1.03 3.4
194001K103 ..(KA's)

ANSWER 7.08 (2.00)

1) atmospheric steam dump valves fail closed, if open
2) letdown will isolate [4 x 0.50]
3) PRZR heaters will deenergize
4) standby service water pump (2SWE-P21A) auto starts, if not already running
5) component cooling water to containment instrument air compressor closes
6) primary component cooling water supply and return isolation valves (2CCP*MOV175-1,176-1,177-1,178-1) close

REFERENCE

B.V.P.S. 2LP-SQS-53C.1 Enabling Objective 5
B.V.P.S. - O.M. AOP-2.38.1 pages 1,2
K/A 000057 EA2.19 4.3
000057A219 ..(KA's)

ANSWER 7.09 (1.00)

SR: < 1E+5 ~~1E+5 (+/- 2.5E+4)~~ counts/second [0.50]
IR: 1E-10 (+/- 0.5E-10) amps ([0.50]

REFERENCE

B.V.P.S. 2LP-SQS-2.2 Enabling Objective 4
B.V.P.S. - O.M. 2.2.4 pages B4,C1
K/A 000032 EA2.04 3.5
000032A204 ..(KA's)

ANSWER 7.10 (3.00)

Reduce power to within the capacity of 1 MFWP
a. place the SG Startup Feedwater Pump in service ~~[0.50]~~ [1 x 0.50]
b. - turbine trip
- main feedwater pump (MFWP) tripped
- MFWP discharge valves closed
- MFW Reg valves closed
- SG Bypass flow control valves closed
- MFW isolation trip valves closed

[5 X 0.50] OR Feedwater isolation

REFERENCE

B.V.P.S. 2LP-SQS-24.1 Enabling Objectives 7,9A(14)
B.V.P.S. - O.M. 2.24.2 pages AAE1
K/A 000054 G0.09 3.1
K/A 000054 G0.10 3.2
000054G010 000054G009 ..(KA's)

(1.50)
ANSWER 7.11 ~~(2.50)~~

Ej

~~a. Unit 1 and 2 (cooling tower) flow (from [FR-SW 101]) [0.50]~~ delete
~~Unit 1 cooling tower blowdown flow [0.50]~~
b. recirculate the tank [0.50] for a minimum of TWO (2) tank volumes OR 8.5 hours [0.50]
c. verify closed ([2SGC-HSV-100]) liquid waste EFF high rad isolation valve [0.50]

REFERENCE

B.V.P.S. 2LP-SQS-17.1 Enabling Objectives 2d,9,5e
B.V.P.S. - O.M. 2.17.2 page 1, 2.43.4 page AEE1
K/A 000059 EA2.02 3.9
K/A 000059 EA2.05 3.9
000059A205 000059A202 ..(KA's)

(***** END OF CATEGORY 7 *****)

ANSWER 8.01 (3.00)

a. SITE AREA [0.40]
TAB 5 -- RCS/Containment leak exceeds make-up capacity [0.35]
b. ALERT [0.40] TAB 14 -- Reactor not subcritical after valid scram signal(s) [0.35]
c. Unusual Event [0.40]
TAB 18 -- Toxic gas nearby release potentially harmful [0.35]
d. ALERT [0.40]
TAB 28 -- Turbine rupture causing casing penetration [0.35]

grader: award 1/2 credit if event is classified more conservatively
award full credit if clasification is also properly justified

REFERENCE

Enabling Objectives UNAVAILABLE
B.V.P.S. - Unit 2 Implementing Procedures BV-2 EPP/I-1 Table 1
K/A 194001 A1.16 4.4 -
194001A116 ..(KA's)

ANSWER 8.02 ~~(3.00)~~ 2.50

VOIDE

a. ~~3.8.1.1 (A.C. Sources) [0.50] AND 3.0.4. (cannot continue startup since you cannot change modes by relying on action statements)~~ [0.50]
b. 3.3.3.5. (remote shutdown monitoring) [1.00]
c. 3.7.7.1.b (control room habitability: 4.7.7.2.a specifies pressure requirement of 1825 psig) [1.00]

REFERENCE

Enabling Objectives UNAVAILABLE
B.V.P.S. - Unit 2 Technical Specifications
B.V.P.S. - O.M. 2 page 2.10.1
K/A 062000 G0.05 3.8
K/A 016000 G0.05 3.5
016000G005 062000G005 ..(KA's)

ANSWER 8.03 (2.00)

No [0.50] each test is within 25% of the required time interval [0.75] but the THREE (3) consecutive combined test intervals exceed 3.25 of the required interval [0.75]

(***** CATEGORY 8 CONTINUED ON NEXT PAGE *****)

REFERENCE

Enabling Objectives UNAVAILABLE
B.V.P.S. Technical Specifications 4.0.2
K/A 064000 G0.05 3.8
064050G008 ..(KA's)

ANSWER 8.04 (2.00)

a. TRUE
b. FALSE
c. FALSE [4 X 0.50]
d. FALSE

REFERENCE

Enabling Objective UNAVAILABLE
DLC SAP Chapter 41 pages 17,47,50; Chapter 42 page 6
K/A 194001 K1.02 4.1 -
194001K102 ..(KA's)

ANSWER 8.05 (2.00)

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

REFERENCE

Enabling Objective UNAVAILABLE
B.V.P.S SAP 3B page 4, Appendix E
10 CFR 50.72
K/A 194001 A1.06 3.4
194001A106 ..(KA's)

ANSWER 8.06 (2.00)

a. during an emergency [0.25] when this action is immediately needed to protect the public health and safety [0.50]
b. a licensed senior operator [0.50]
c. yes [0.25] but only in the event of an emergency or casualty not covered by an approved procedure [0.50]

REFERENCE

Enabling Objectives UNAVAILABLE
B.V.P.S. - OM 2.48.2 page 8
K/A 194001 A1.02 3.9
194001A102 ..(KA's)

ANSWER 8.07 (2.00)

a. 1.
b. 3. [4 X 0.50]
c. 4.
d. 2.

REFERENCE

Enabling Onjectives UNAVAILABLE
B.V.P.S. - OM 2.48.5 Procedure A, "Logs and Reports," pages 3,5,6
K/A 194001 A1.06 3.4
194001A106 ..(KA's)

ANSWER 8.08 (2.50)

a. cycle valve 2CHS*AOV204 [0.20] in less than 60 seconds [0.20] and valves 2CHS*AOV200,A,B,C [0.20] in 10 seconds [0.20] through one complete cycle of full travel [0.20]
b. cycle valves 1 SS*MOV155A,156A [0.20] to the open position [0.20] in less than 60 seconds [0.20]
c. cycle valves 2HVR*MOD23A,B [0.20] through one complete cycle of full travel [0.20] in 10 seconds [0.20]

ALL the above valves must be cycled atleast once per 92 days [0.30]

REFERENCE

Enabling Objective UNAVAILABLE
B.V.P.S. - Unit 2 Technical Specifications Section 3/4.6.3 Table 3.6-1
K/A 103000 K4.06 3.7
K/A 103000 G0.05 4.1
103000G005 103000K406 ..(KA's)

ANSWER 8.09 (2.25) Ey

2RMR-RQ~~205~~ 207, 206 (Containment Area) [0.50]
2HVS-RQ109C (Mid Range Noble Gas) [0.50]
2HVS-RQ109D (High Range Noble Gas) [0.50]
2MSS-RQ101A,B,&C (Main Steam Discharge) [0.75]

REFERENCE

B.V.P.S. 2LP-SQS-43.1 Enabling Objective 4
B.V.P.S. - Unit 2 Technical Specifications Table 3.3-6 Action 36
K/A 016000 G0.04 3.4
016000G004 ..(KA's)

ANSWER 8.10 (3.00) Ey

a. ~~Yes~~ NO [0.25] because RCS fluoride concentration is ~~less~~ Greater than the ~~transient~~ steady state limit [0.50]
b. Yes [0.25] because the Technical Specification is not applicable in Mode 5 [0.50]
c. No [0.25] because the concentration exceeds the transient limit [0.50]
d. No [0.25] because the LCO action statement was not met [0.50]

REFERENCE

Enabling Objective UNAVAILABLE
B.V.P.S. - Unit 2 Technical Specifications Section 3/4.4.7
K/A 194001 A1.14 2.9
194001A114 ..(KA's)

ANSWER 8.11 (1.25)

a. ensure that the dosage contribution [0.20] from the tube leakage will be limited to a small fraction of the 10 CFR Part 100 limits [0.20] in the event of either a steam generator tube rupture [0.20] or a steam line break [0.20]
b. 3/4.4.5 [0.45]

REFERENCE

Enabling Objective UNAVAILABLE
B.V.P.S. - Unit 2 Technical Specifications Section 3/4.4.6.2, 3/4.4.5
K/A 000037 G0.04 3.9
000037G004 ..(KA's)

(***** END OF CATEGORY B *****)
(********** END OF EXAMINATION **********)

Master Key

$f = ma$

$v = s/t$

Cycle efficiency = (Net work out)/(Energy in)

$w = mg$

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

$E = mc^2$

$KE = 1/2\ mv^2$

$a = (V_f - V_o)/t$

$A = \lambda N$

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

$PE = mgh$

$V_f = V_o + at$

$w = e/t$

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

$W = v\ \Delta P$

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

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

$\Delta E = 931\ \Delta m$

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

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

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

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

$Pwr = W_f \Delta h$

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

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

$TVL = 1.3/\mu$

$HVL = -0.693/\mu$

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

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

$SUR = 26.06/T$

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

$\Sigma = \sigma N$

$I_1 d_1 = I_2 d_2$

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

$R/hr = (0.5\ CE)/d^2$(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 $lbf/in.^2$

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

F. ESTIMATED CRITICAL POSITION CALCULATION

FORM ECP-1 (Page 1 of 7)

Unit 2 ECP form used during Exam.

NOTE: Reference Guide in Chapter 49, Section 4, Procedure M.

Tavg Assumed to Equal 547F ± 1F at Startup

Ey

A. CRITICAL DATA

PRIOR TO SHUTDOWN	EXPECTED CRITICAL
Date 2 / 22 / 88 Time 0400	Date 2 / 23 / 88 Time 1000
Boron Conc 900 ppm Power 100 %	Boron Conc. ______ ppm
Xenon equil. %	Xenon ______ %
Samarium equil. %	Samarium ______ %
Control Rod Position:	Control Rod Position:
A 228 C 228	A 228 C 223
B 228 D 228	B 228 D 75

B. REACTIVITY BALANCE

Reactivity Defects	I Prior to Shutdown	II expected at criticality	III Difference I-II
1. Power (Fig. 50-7)	- pcm	- pcm	(±) pcm
2. Control Rods (Fig. 50-8) or Boron (Fig. 50-10)	- pcm	- pcm	(±) pcm
3. Xenon	- pcm	- pcm	(±) pcm
4. Samarium	- pcm	- pcm	(±) pcm

5. Reactivity Change (sum of 1-4) = (±) pcm

F. ESTIMATED CRITICAL POSITION CALCULATION (continued)

FORM ECP-1 (Page 2 of 7)

NOTE: If Reactivity Change is greater than ±500 pcm, perform 1/M plot, Table 50-1.

C. CRITICAL BORON CONCENTRATION (Use if critical boron concentration is desired.)

I	II	III	IV	V
Reactivity Change (B-5)	Boron Worth (Fig. 50-10)	Boron Change I divided by II	Boron Conc at shutdown	Boron Conc for startup III + IV
(±) pcm	pcm/ppm	(±) ppm	ppm	ppm

D. CRITICAL ROD POSITION (Use if critical rod position is desired)

I	II	III	IV
Reactivity Change (B-5)	Reactivity due to Rod Prior to Shutdown (Fig. 50-8)	Reactivity for Criticality (I+II)	Critical Rod Position (Fig. 50-8)
(±) pcm N/A	- pcm N/A	(±) pcm N/A	steps N/A

F. ESTIMATED CRITICAL POSITION CALCULATION (continued)

E. ROD LIMITS

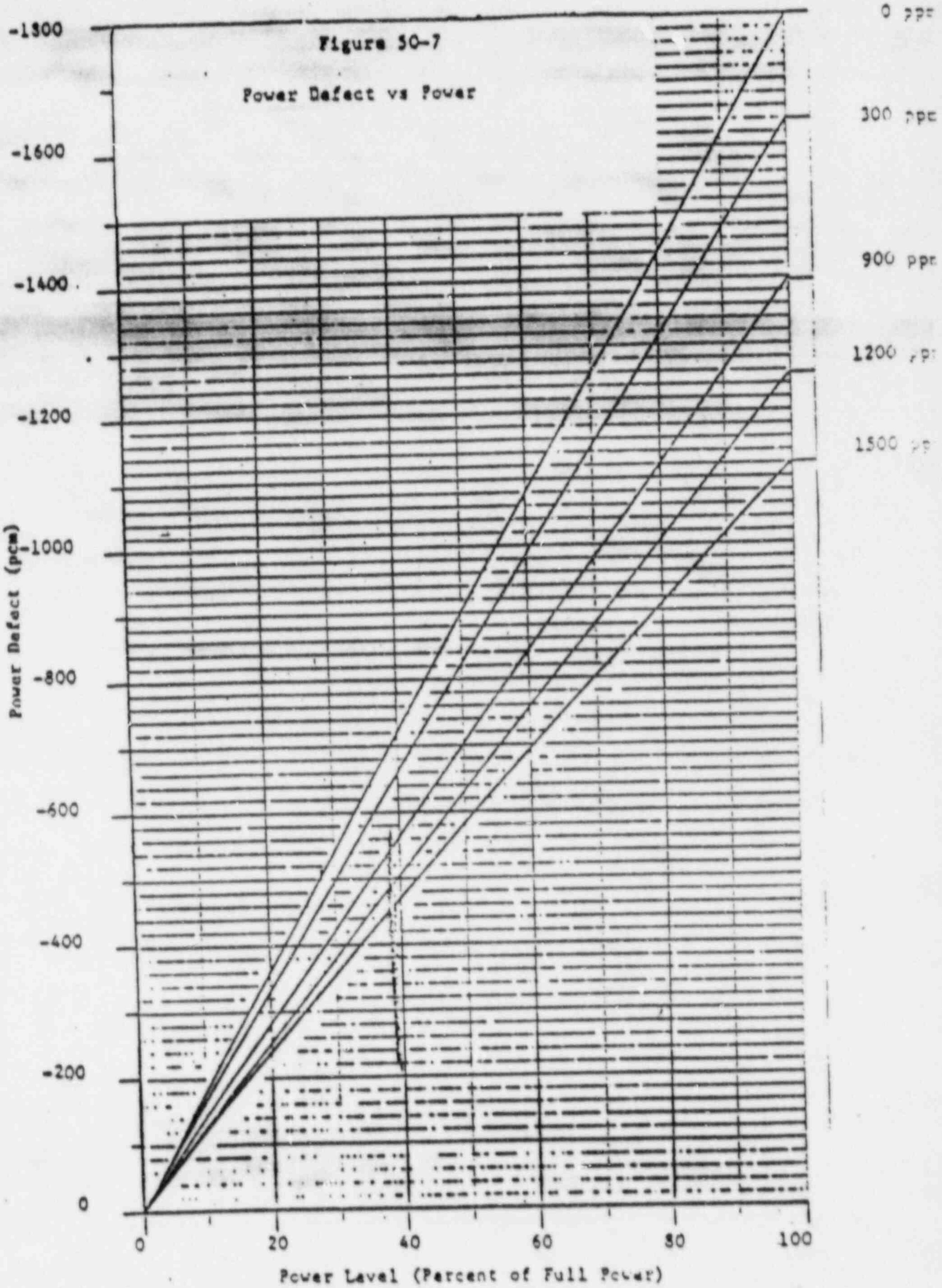
FORM ECP-1 (Page 3 of 7)

I	II	III	IV	V
Expected Rod Defect at Crit (B-2 or D-III)	I + 500pcm (use 0 if positive)	I-500pcm	Rod Position for II from Fig. 50-8	Rod Position for III from Fig. 50-8
- pcm .N/A	- pcm N/A	- pcm N/A	Bank N/A at N/A	Bank N/A at N/A

MAXIMUM ROD HEIGHT FOR CRITICALITY (Item E-IV) = Bank N/A at N/A Steps

NOTE: If criticality is not achieved by Maximum Rod Height, insert rods to (Item E-V) bank N/A at N/A Steps and recalculate ECP.

MINIMUM ROD HEIGHT FOR CRITICALITY = Bank C at 115 Steps (T.S. 3.1.3.5)



Power Defect vs Power

Figure 50-7 (9-13)

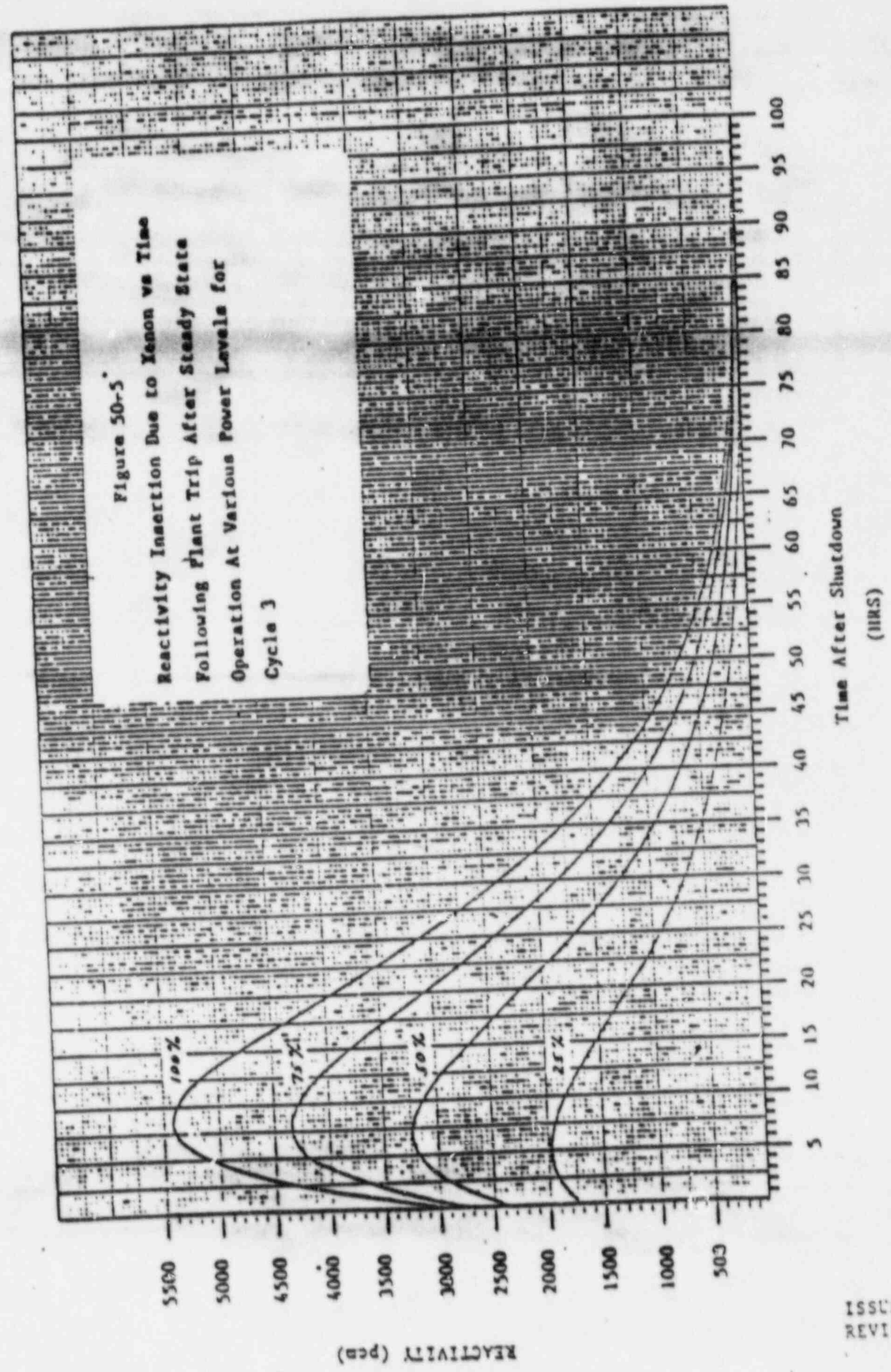


Figure 50-5
Reactivity Insertion Due to Xenon vs Time Following Plant Trip After Steady State Operation At Various Power Levels for Cycle 3

ISSUE 2
REVISION 1

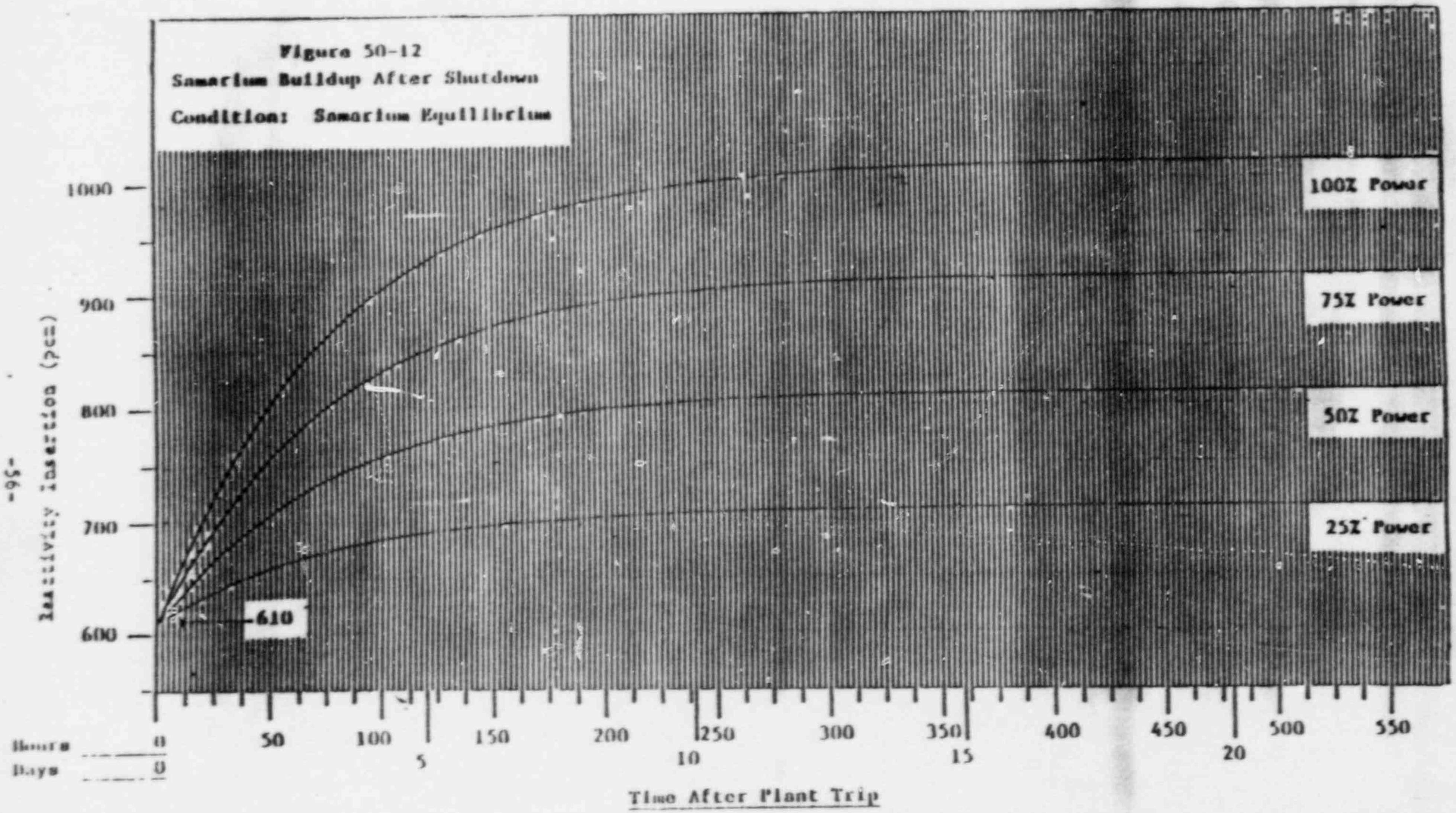
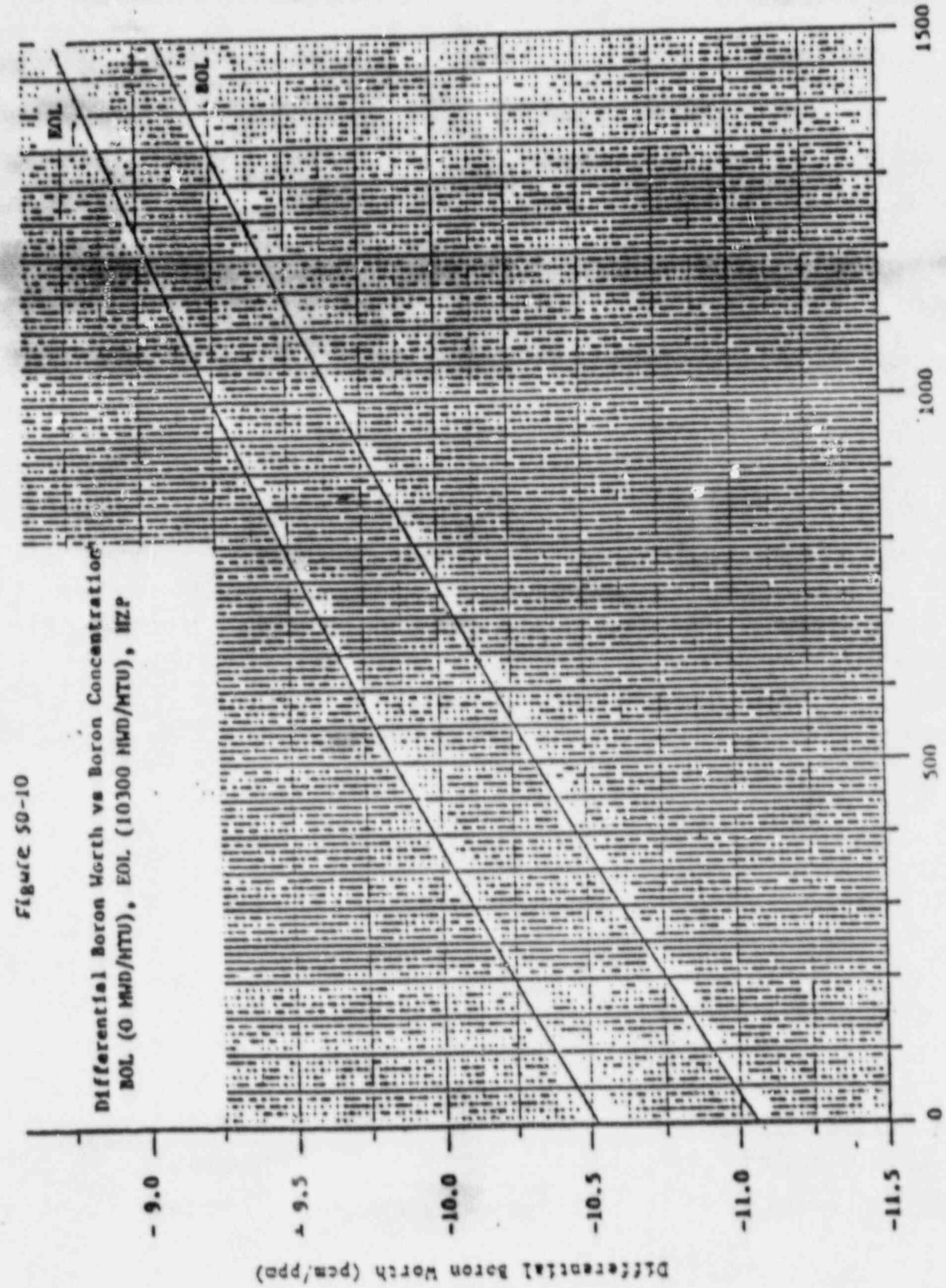


Figure 50-12 (9-18)

Samarium Buildup Aft[illegible] [illegible]utdown

from Samarium Equilibrium



Differential Boron Worth vs Boron Concentration
BOL (0 MWD/MTU), EOL (10300 MWD/MTU), HZP
Figure 50-10 (9-16)

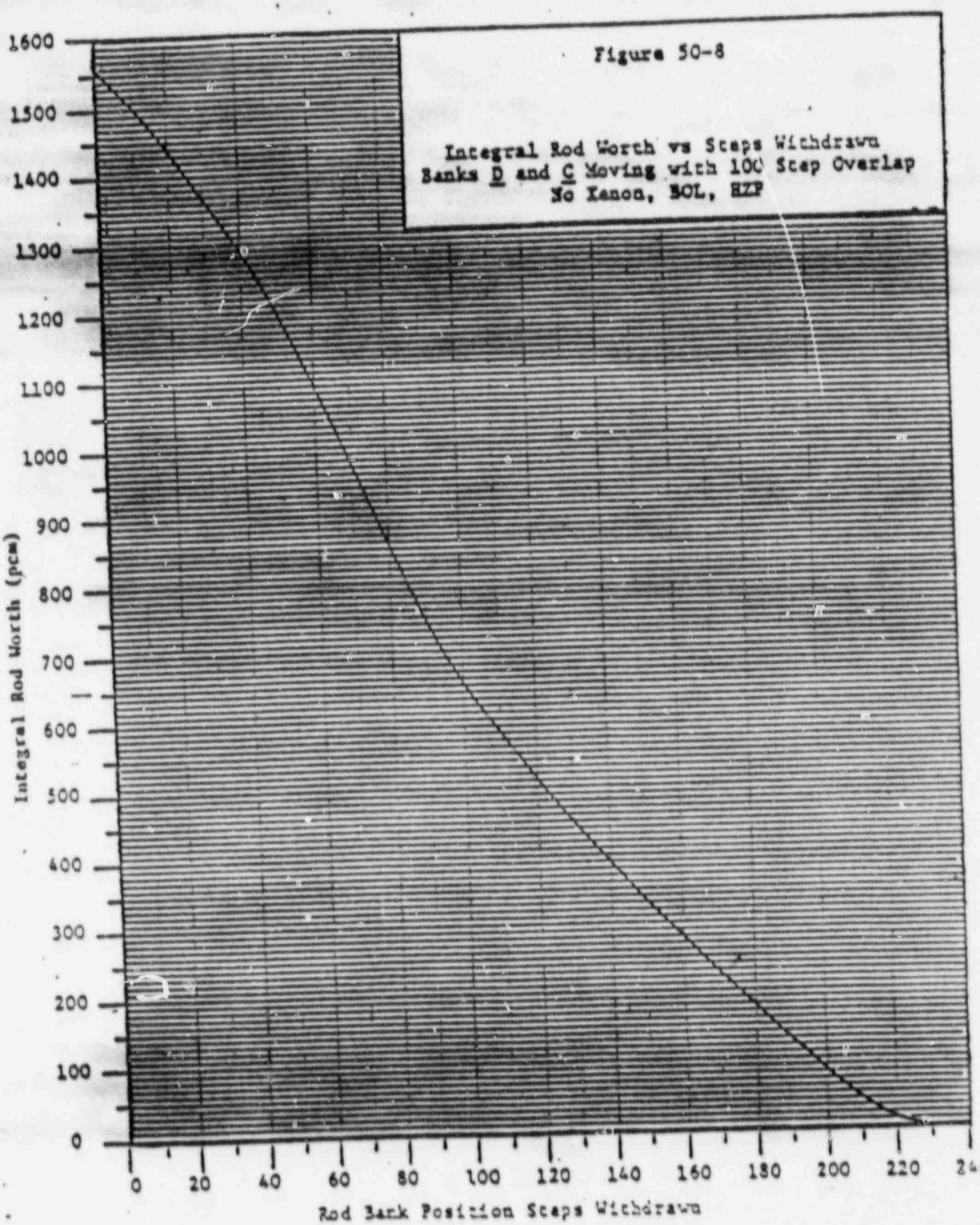
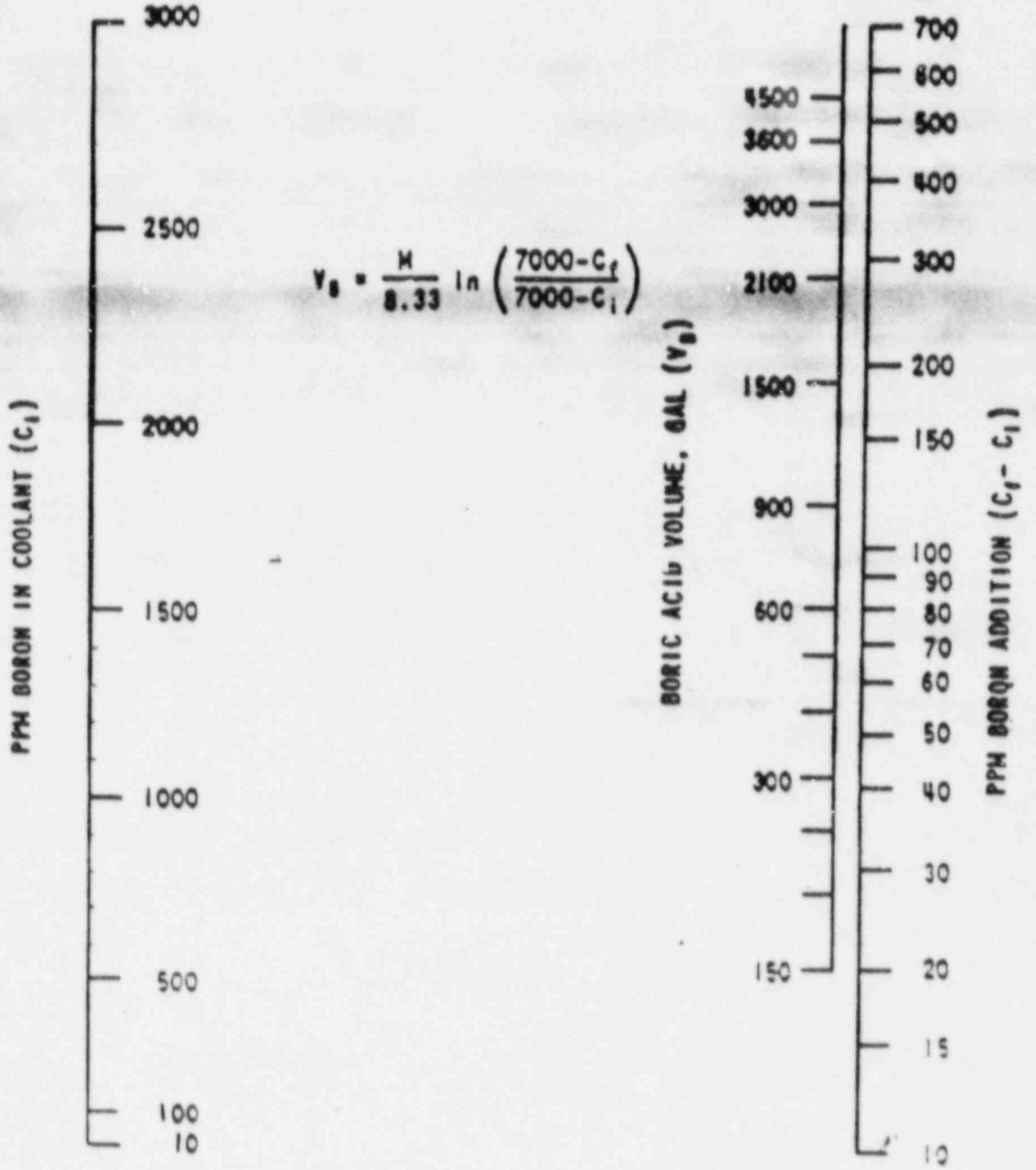


Figure 50-8 (9-14)

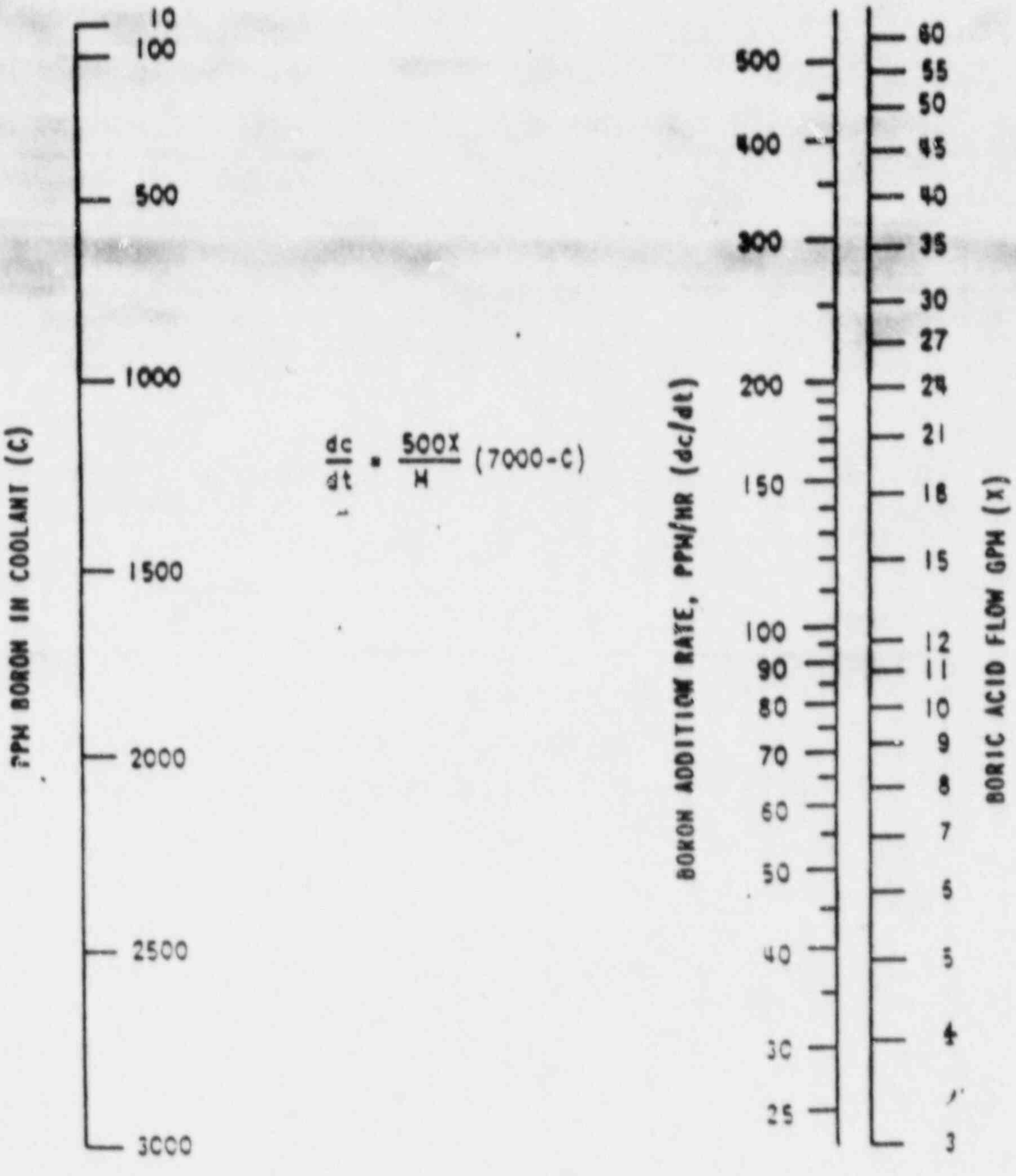
Integral Rod Worth vs. Steps Withdrawn



Boron Addition - Refer to Table 1 for Correction Factors

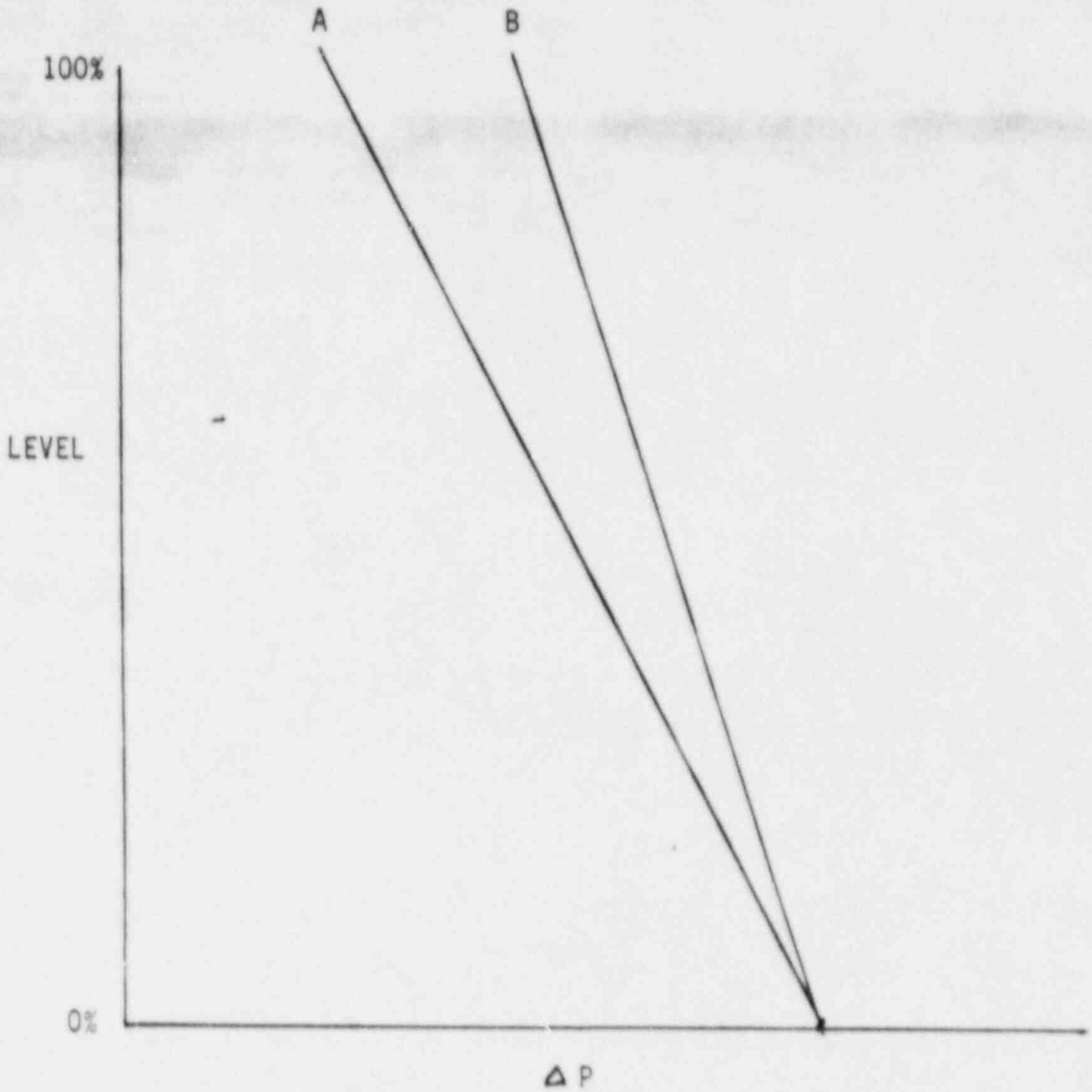
CB-32

FOR INFORMATION ONLY



Boron Addition Rate - Refer to Table 1 for Correction Factors

ATTACHMENT 3



ACTION LEVEL CRITERIA FOR CLASSIFICATION OF EMERGENCY CONDITIONS

INITIATING CONDITION	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
	Off-normal Events Which Could Indicate a Potential Degradation of the Level of Safety of the Plant.	Events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant.	Events Which Involve Actual or Likely Major Failures of Plant Functions Needed for Protection of the Public.	Events Which Involve Actual or Imminent Substantial Core Degradation or melting with Potential for Loss of Containment Integrity.
Radioactive Effluent Applicable to Any Release Point(s) and Resulting from Any Initiating Event TAB 1	Unplanned airborne release gives offsite dose rate greater than 0.5 mRem/hr. -or- Unplanned liquid release in excess of MPC limits.	Unplanned airborne release gives offsite dose rate greater than 2.0 mRem/hr. -or- Unplanned liquid release results in downstream community water radioactivity greater than 12 times EPA standards.	Release Corresponds to >20 mrem/hr. at Site Boundary -or- Offsite Dose Due to Event is Projected to Exceed 170 mrem to Whole Body or Child Thyroid.	Radiological effluent release results in offsite dose projected to exceed 1 rem to the Whole Body or 5 rem to the Child Thyroid. and/or Radiological effluent corresponds to greater than 125 mRem/hr. whole body dose rate or 600 mRem/hr. child thyroid at the site boundary.
Release or Loss of Control of Radioactive Material Within the Plant. TAB 2	→	Fuel Handling Accident Resulting in Release of Radioactivity to Occupied Areas Such That the Direct Radiation Levels in the Areas Increase by a Factor of > 1000 -or- Other Verified, Uncontrolled Events Which Result in an Unexpected Increase of In-Plant Direct Radiation Levels by a Factor of > 1000.	Major Damage to Spent Fuel Due to Fuel Handling Accident -or- Uncontrolled Decrease in Fuel Pool Water to Below Level of Fuel.	

ACTION LEVEL CRITERIA FOR CLASSIFICATION OF EMERGENCY CONDITIONS

INITIATING CONDITION	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
Reactor Coolant System (RCS) Temperature Low TAB 3	Below Tech Spec Limiting Conditions for Operation (LCO)			Loss of 2 of 3 Fission Product Barriers With a Potential Loss of Third Barrier. Applicable to Any Initiating Event that May Lead to this Condition -or- Any Initiating Events, from Whatever Source that Makes Release of Large Amounts of Radioactivity in a Short Time Probable, For Example: 1. LOCA With Failure of ECCS. 2. LOCA With Initially Successful ECCS. Subsequent Failure of Heat Removal Systems with Likely Failure of Containment. 3. Loss of All Onsite and Offsite Power Concurrent With Total Loss of Emergency Feedwater. 4. Loss of Feedwater and Condensate Followed by Failure of Emergency Feedwater System. 5. Reactor Protection System Fails to Initiate or Complete a Required Scram, Followed by Loss of Core Cooling and Make-up System -or- Loss of Plant Control Occu
RCS Pressure High TAB 4	Exceeds LCO Limit			
RCS/Containment Leak TAB 5	Exceeds LCO	Exceeds 50 gpm	Exceeds Make-up Capacity	
RCS/Secondary Leak TAB 6	Exceeds LCO	>200 gpm -or- >10 gpm w/ MSL Break -or- MSL Break w/ MSIV Failure	>50 gpm w/ MSL Break w/Indication of Fuel Failure -or- >1000 gpm	
Main Steam Line Break or Rapid Depressurization of Secondary Side TAB 7	→			
Fuel Cladding Degradation TAB 8	RCS Activity Exceeds LCO -or- Reactor Coolant Monitor Alarm, or or analyses 1 uCi/gm, Steady State	RCS ^{131}I Activity >300 uCi/gm	Degraded Core-Possible Loss of Coolable Geometry.	
RCS Safety or Relief Valve Failure TAB 9	Leak Exceeds LCO or Valve Inoperable			
RCS Temperature High TAB 3	Exceeds LCO			
RCS Pressure Low TAB 4	Below LCO			

ACTION LEVEL CRITERIA FOR CLASSIFICATION OF EMERGENCY CONDITIONS

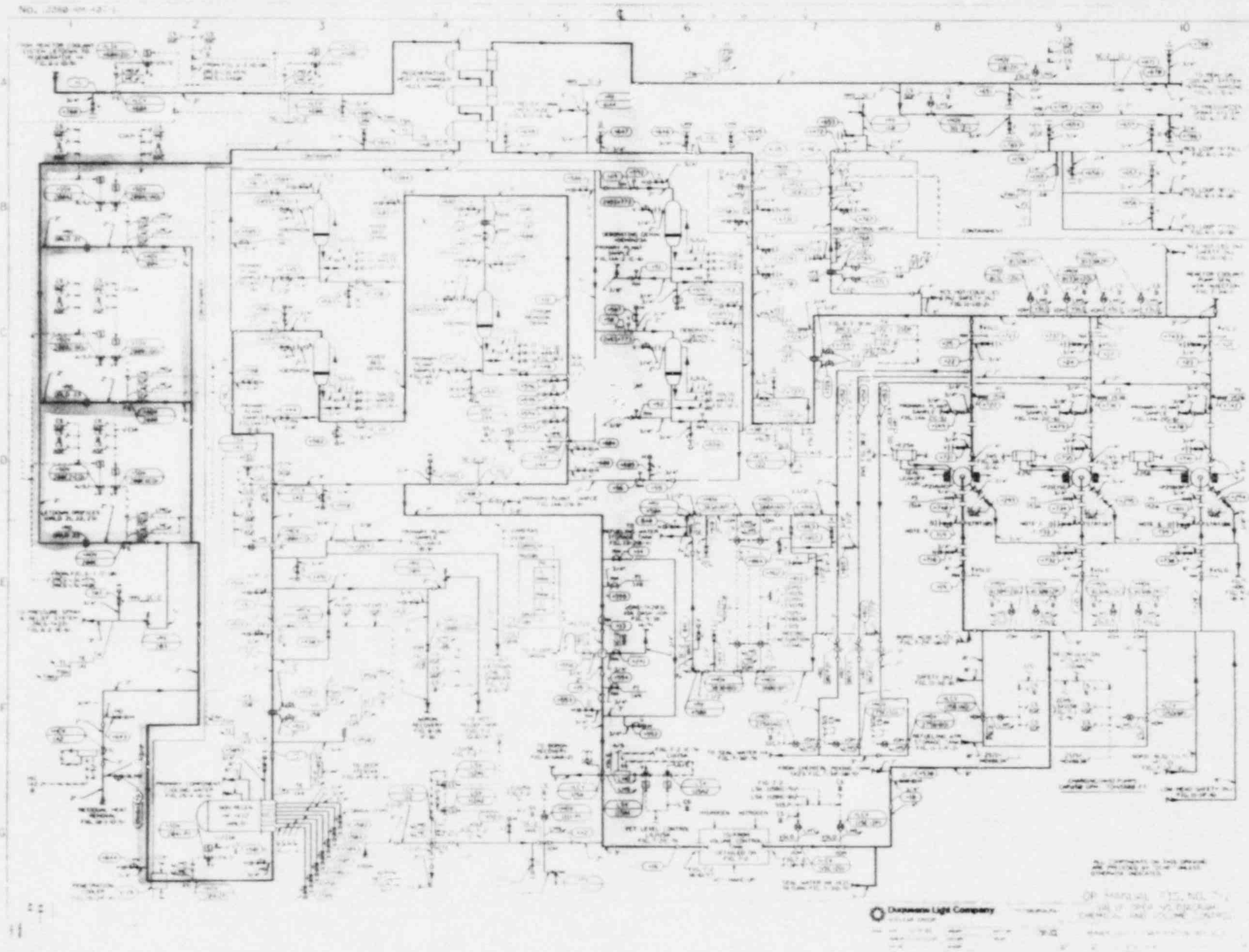
INITIATING CONDITION	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
Initiation of ECCS TAB 10	Valid Safety Circuit Trip or Necessary Manual Initiation.			Loss of 2 of 3 Fission Product Barriers With a Potential Loss of Third Barrier. Applicable to Any Initiating Event that May Lead to this Condition. -or- Any Initiating Events, from Whatever Source that Makes Release of Large Amounts of Radioactivity in a Short Time Probable, For Example: 1. LOCA With Failure of ECCS. 2. LOCA With Initially Successful ECCS. Subsequent Failure of Heat Removal Systems with Likely Failure of Containment. 3. Loss of All Onsite and Offsite Power Concurrent With Total Loss of Emergency Feedwater. 4. Loss of Feedwater and Condensate Followed by Failure of Emergency Feedwater System. 5. Reactor Protection System Fails to Initiate or Complete a Required Scram, Followed by Loss of Core Cooling and Make-up Systems -or- Loss of Plant Control Occurs.
RCS Pump Failure TAB 11	→	RCS pump Seizure leading to fuel failure		
Loss of Containment Integrity TAB 12	Requiring Shutdown by LCO		Containment Pressure >5 & <45 psig	
Loss of Engineered Safety or Fire Protection Features TAB 13	Requiring Shutdown by LCO			
Failure of Reactor Protection System to Initiate or Complete a Scram TAB 14	→	Reactor Not Subcritical after Valid Scram Signal(s).		
Loss of Plant Control/Safety Systems TAB 15	→	Loss of Capability to Achieve Cold Shutdown	Loss of Capability to Achieve Hot Shutdown	
Loss of Indicators, Annunciators or Alarms TAB 16	Loss on Process or Effluent Parameters, Requiring Shutdown by LCO	Loss of All Alarms (Annunciators) Sustained for 5 mins.	Loss of All Alarms 15 min with Plant Not in Cold S/D -or- Plant Transient Occurs While All Alarms are Lost.	
Control Room Evacuation TAB 17	→	Required or Anticipated. Control of Shutdown Systems Established at Remote Shutdown Panel.	Required. Shutdown System Control at Remote Shutdown Panel Not Established Within 15 min.	

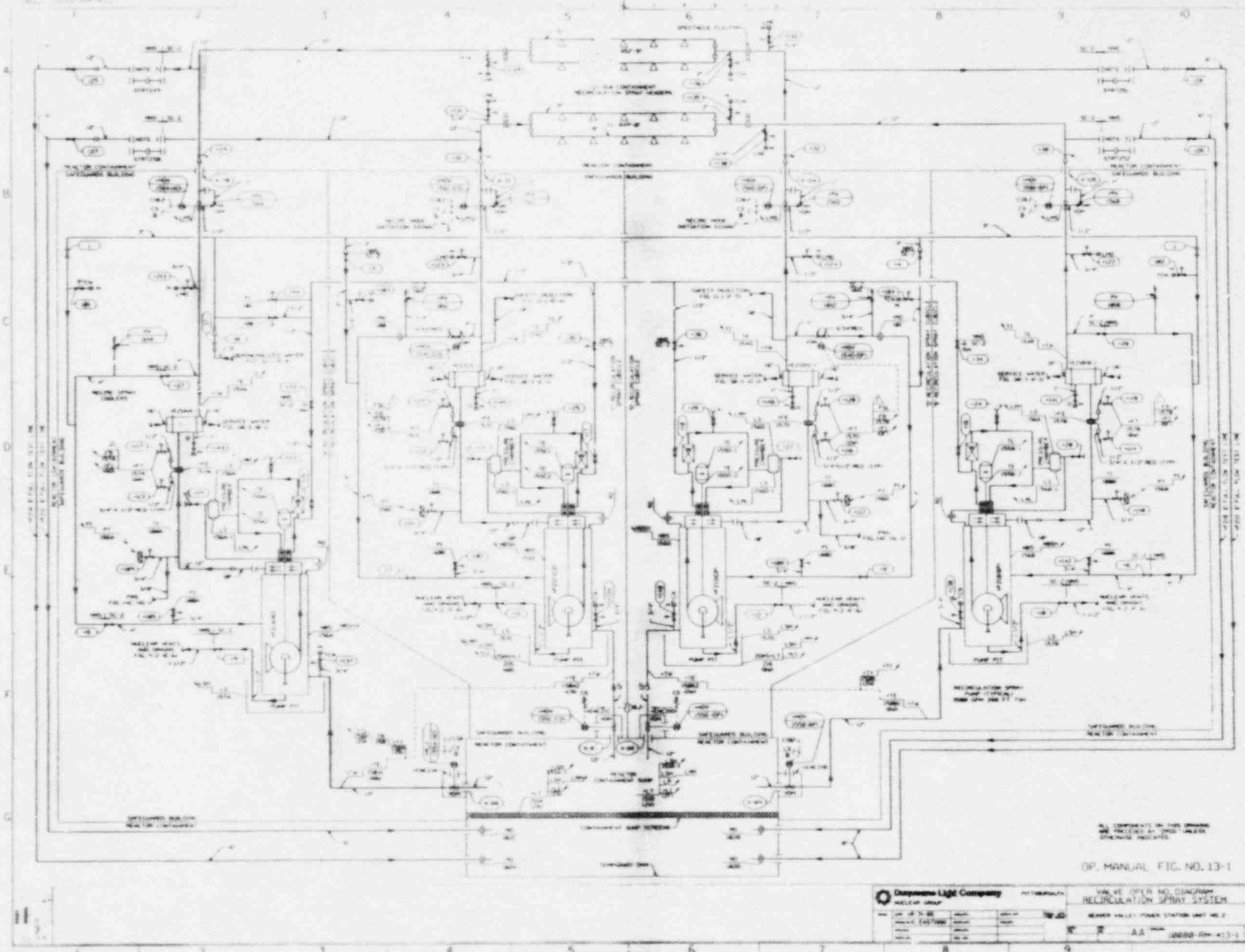
ACTION LEVEL CRITERIA FOR CLASSIFICATION OF EMERGENCY CONDITIONS

INITIATING CONDITION	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
Toxic or Flammable Gases TAB 18	Near-by or On-Site Release Potentially Harmful.	Enters Facility. Potential Habitability Problems.	Enters Vital Areas and Restricts Necessary Access.	Loss of 2 of 3 Fission Product Barriers With a Potential Loss of Third Barrier. Applicable to Any Initiating Event that May Lead to This Condition. -or- Any Initiating Events, from Whatever Source that Makes Release of Large Amounts of Radioactivity in a Short Time Probable, For Example: 1. LOCA With Failure of ECCS. 2. LOCA With Initially Successful ECCS. Subsequent Failure of Heat Removal Systems With Likely Failure of Containment. 3. Loss of All Onsite and Offsite Power Concurrent With Total Loss of Emergency Feedwater. 4. Loss of Feedwater and Condensate Followed by Failure of Emergency Feedwater System. 5. Reactor Protection System Fails to Initiate or Complete a Required Scram, Followed by Loss of Core Cooling and Make-up Systems -or- Loss of Plant Control Occurs.
Security Compromise TAB 19	In Accordance with Security Plans		Imminent Loss of Physical Control of Plant.	
Loss of On-Site AC Power TAB 20	Loss of Capability	Temporary Loss of Both.	Loss of Both Exceeds 15 mins.	
Loss of All Off-Site Power TAB 20	Upon Occurence			
Loss of All On-Site DC Power TAB 21	→	Upon Occurrence	Loss of Vital DC Power For More than 15 mins.	
Tornado or Other High Winds TAB 22	Warning. Probable Effect on Station.	Strikes Vital Plant Structures.	Winds in Excess of Design Levels	
Flood or Low Water TAB 23	Flood < 705 feet MSL, Requiring S/D. Low Water < LCO.	Flood > 705 feet MSL	Flood > 735 feet MSL -or- Damage to Vital Equipment.	
Earthquake TAB 24	Detected on Site Seismic Instrumentation.	Greater than OBE Occurs	Greater than SSE Occurs	
Fire TAB 25	Fire within protected area lasting more than 10 minutes.	Potentially Affecting Safety Systems.	Affecting Safety Systems Required for Shutdown.	

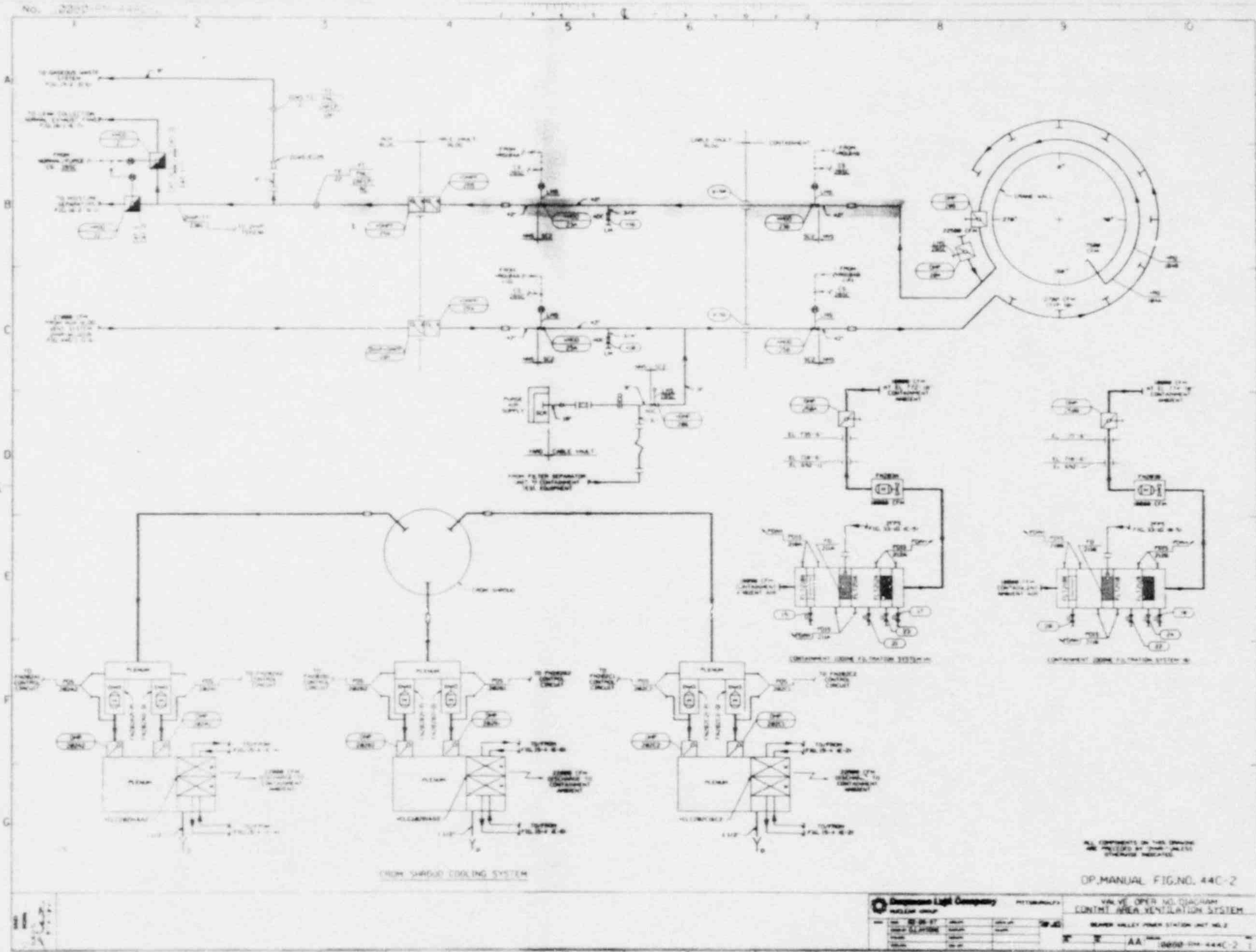
ACTION LEVEL CRITERIA FOR CLASSIFICATION OF EMERGENCY CONDITIONS

INITIATING CONDITION	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
Explosion TAB 26	Near or On-site Explosion Potential Significant Damage	Known Damage to Facility, Affecting Operation.	Severe Damage to Safe Shutdown Equipment.	Loss of 2 of 3 Fission Product Barriers With a Potential Loss of Third Barrier. Applicable to Any Initiating Event that May Lead to this Condition. -or- Any Initiating Events, from Whatever Source that Makes Release of Large Amounts of Radioactivity in a Short Time Probable, For Example: 1. LOCA With Failure of ECCS. 2. LOCA With Initially Successful ECCS. Subsequent Failure of Heat Removal Systems with Likely Failure of Containment. 3. Loss of All Onsite and Offsite Power Concurrent With Total Loss of Emergency Feedwater. 4. Loss of Feedwater and Condensate Followed by Failure of Emergency Feedwater System. 5. Reactor Protection System Fails to Initiate or Complete a Required Scram, Followed by Loss of Core Cooling and Make-up Systems -or- Loss of Plant Control Occur
Aircraft TAB 27	Unusual Activity Over Facility -or- Aircraft Crashes Onsite	Aircraft or Missile From Whatever Source Strikes and Significantly Degrades a Station Safety Structure.	Crash Affects Vital Structures by Impact or Fire.	
Train TAB 28	Derailment in Onsite Areas			
Watercraft TAB 28	Strikes Intake Structure, Resulting in Flow Reduction			
Contaminated Injury TAB 28	Transportation of Injured and Contaminated Individual(s) to Offsite Hospital.			
Oil Pipeline Rupture TAB 28	Rupture of Pipeline Onsite w/ or w/o Fire			
Turbine Rupture TAB 28	Turbine rotating component failure causing rapid plant shutdown.	Turbine failure causing casing penetration		
S/G Tube Failure with loss of offsite power. TAB 29		Failure of one S/G Tube with loss of offsite power.	Rapid failure of S/G Tubes (> 200 gpm) with loss of offsite power.	


Duquesne Light Company



OP. MANUAL FIG. NO. 13-1



FROM SHROUD COOLING SYSTEM
PLENUM
OP.MANUAL FIG.NO. 44C-2
Duquesne Light Company
VALVE OPER NO. DIAGRAM
CONTMT AREA VENTILATION SYSTEM

Attachment 2

Duquesne Light

Nuclear Group
P.O. Box 4
Shippingport, PA 15077-0004

Telephone (412) 393-6000

February 29, 1988
ND2VPN: 5350

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

Reference: Beaver Valley Power Station, Unit #2
Docket 50-412, License NPF 73
License Examination Report

Dear Mr. Gallo:

Please find enclosed comments generated by our Training Section associated with the written examination administered February 23, 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

JDS/cal

Enclosure

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

QUESTION 5.03 (3.00)

For EACH of the primary parameters listed below, state HOW (Increases, Decreases, No Change) and explain WHY an INCREASE in that parameter affects the DNBR. Assume the other parameters remain constant.

a. Reactor Power
b. Tave
c. Core flow
d. Pressurizer Pressure

ANSWER 5.03 (3.00)

a. Decreases (0.35) because raising power increases the heat flux on the fuel rod, reducing the DNBR (0.40)

b. Decreases (0.35) because the subcooling margin decreases (0.40)

c. Increases (0.35) because more heat can be absorbed by the water (0.40)

d. Increases (0.35) because the subcooling margin increases (0.40)

REFERENCE

B.V.P.S. LP-TMO-7 Enabling Objectives 11,12
B.V.P.S. LP-TMO-7 pages 21, 22, 23, 26
K/A 193008 K1.05 3.6
193008K105 ..(KA's)

COMMENT:

5.03.c

Better heat transfer should also be an acceptable reason for DNBR increasing with increasing core flow, as can be seen by equation from Attachment 5.03.c with $\dot{m}$ increasing.

5.03E

Facility

$f = ma$

$v = s/t$

Cycle efficiency = (Net work out)/(Energy in)

$w = mg$

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

$E = mc^2$

$KE = 1/2\ mv^2$

$a = (V_f - V_o)/t$

$A = \lambda N$

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

$PE = mgh$

$V_f = V_o + at$

$w = \theta/t$

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

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

$W = v\ \Delta P$

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

$\Delta E = 931\ \Delta m$

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

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

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

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

$Pwr = W_f \Delta h$

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

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

$TVL = 1.3/\mu$

$HVL = -0.693/\mu$

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

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

$SUR = 26.06/T$

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

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

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

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

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

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

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

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

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

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

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

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

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

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

$I_1 d_1 = I_2 d_2$

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

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

$\Sigma = \sigma N$

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

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

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.

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

QUESTION 5.05 (2.00)

Answer the following statements concerning Heat Exchanger Operation by responding TRUE or FALSE.

a. Once turbulent flow in a heat exchanger has been established, UA becomes approximately a fixed value.

b. If the ΔT across a heat exchanger is not constant then ΔT_m, the median (average) temperature, is used to accurately calculate the heat transfer rate.

c. The heat removal rate for a heat exchanger will increase if either of the fluid flowrates through the heat exchanger is Increased.

d. The U-tubes of the steam generators can experience thermal shock if the feedwater flowrate is increased rapidly.

ANSWER 5.05 (2.00)

a. TRUE
b. FALSE (4 x 0.50)
c. TRUE
d. TRUE

REFERENCE

B.V.P.S. LP-TMO-3 Enabling Objectives 4,7
B.V.P.S. LP-TMO-3 pages 8, 12
K/A 191006 K1.03 2.3
K/A 191006 K1.04 2.7
K/A 191006 K1.07 2.6
191006K107 191006K104 191006K103 ..(KA's)

COMMENT:

Part a. asks if UA will vary or not for a heat exchanger with turbulent flow. Operators monitor flow, pressure, and delta T for heat exchangers. UA is not something that can be monitored, nor is whether a heat exchanger has laminar or turbulent flow. This question goes beyond the knowledge required of an operator. K/A 191006 K1.03 requires knowledge of "Basic heat transfer in a heat exchanger". We ask that this question be withdrawn. The statement is also incorrect since fouling would cause UA to vary once turbulent flow is established.

Part b. tests the knowledge of the proper name for the symbol ΔT_m. This is a minor point. The accepted method of calculating heat transfer across a heat exchanger such as a steam generator is to use the average temperature (i.e. $Q = UA (T_{avg} - T_{stm})$, not the log mean temperature. This is accurate enough for our purposes. This knowledge is not a good measure of an operator's ability to safely operate the plant and we ask that the question be withdrawn.

QUESTION 5.06 (2.50)

WHAT are FIVE (5) indications that natural circulation has been established after a loss of offsite power occurs.

ANSWER 5.06 (2.50)

1) core exit TCs - stable or decreasing (5 x 0.50)

2) RCS hot leg temperatures - stable or decreasing

3) RCS cold leg temperatures - at saturation for existing S/G pressure

4) RCS subcooling based on core exit thermocouples) - greater than subcooling per attachment (7)

5) S/G pressures - stable or decreasing

REFERENCE

B.V.P.S. LP-TMO-7 Enabling Objectives 16
B.V.P.S. EOP ES-0.1, "Reactor Trip Response," Attachment 5
K/A 193008 K1.22 4.2
193008K122 ..(KA's)

COMMENT:

Less than or equal to 60 F temperature difference hot leg to cold leg should also be an acceptable indication of natural circulation in the answer key, as can be seen in attachment 5.06 EOP background document on natural circulation.

5.06

include natural circulation verification. The steps that verify natural circulation flow are included in the EOPs after SI flow is terminated. If the SI system is in operation, natural circulation flow is not verified since with SI on there are more important steps to be taken and the SI flow may affect the indications used to confirm natural circulation.

If natural circulation flow based on the symptoms listed in the attachment is not verified, then the EOPs direct the operator to increase steam dump flow to try to establish verifiable natural circulation flow

The following symptoms are used in the Natural Circulation attachment to verify natural circulation flow:

A. RCS subcooling based on core exit TCs should be greater than instrument inaccuracies.

B. The core exit TCs, RCS hot leg temperatures and SG pressures should be decreasing slowly with time, as core decay heat falls off.

C. With SG pressures held relatively constant, the RCS cold leg temperatures should remain relatively constant at or slightly above the saturation temperature for the SG pressures being maintained.

In addition to the symptoms used in the Natural Circulation attachment, the following symptoms can be used for confirmation of natural circulation flow:

A. The hot-to-cold leg temperature difference should be approximately equal to the full-power forced convection temperature difference.

B. The core exit average temperature (core exit TCs averaged reading) should be higher than the average cold leg temperature. This averaged reading should also decrease as core decay heat falls off, in step with core exit TC, hot leg temperature, and SG pressure readings in all active loops.

To facilitate the verification of transient equilibrium attainment in the natural circulation process, the operators should start to record these parameters at regular intervals beginning as soon as instructed in the EOPs. The continuous recording will provide trending information on the parameters of importance in order to eliminate the effects of pointwise variations in the readings and minimize the chance of misinterpretation of any one set of readings. Variations in discrete readings, and between the same parameters in different loops, can result from several causes:

- Asymmetry in the heat transfer and heat transport processes between loops.
- Instrument inaccuracies.
- Difference in instrument sensing element placement between loops.
- Variations in feed flowrates to steam generators.

QUESTION 5.08 (2.00)

a. Do xenon oscillations converge (dampen) more rapidly at BOL or EOL? Justify your answer in terms of reactivity effects.

b. Would the magnitude and frequency of xenon oscillations be less at 50% power or 100% power? Justify your answer.

ANSWER 5.08 (2.00)

a. EOL (0.25) the negative power coefficient of reactivity tends to dampen the oscillations (0.50). This coefficient is more negative at EOL (0.25).

b. 50% power (0.25) the lower neutron flux at 50% power does not produce xenon as fast as at 100% power (0.50) since the rate at which xenon is produced is slower, the magnitude and frequency of the oscillations will be less (0.25)

REFERENCE

B.V.P.S. LP-RT-7 Enabling Objectives 5.6
B.V.P.S. Reactor Theory Text Chapter 6 page 51; Chapter 7 page 17
K/A 001050 A2.06 4.0
K/A 192006 K1.06 3.4
192006K106 001050A206 ..(KA's)

COMMENT:

When xenon oscillations occur at Beaver Valley, the oscillations are plotted for a period of time, usually 24 hours before an attempt is made to dampen them. This is done to determine the frequency and magnitude of the oscillation. See attached procedure 2.49.4.G. As is apparent from the procedure, operators do not have to estimate magnitude or frequency on their own at different plant conditions. Since these oscillations are slow, and since they are plotted to determine their magnitude and frequency, the knowledge asked for in this question is not a good measure of whether or not an operator has enough knowledge of xenon oscillations to control them. We ask that this question be deleted from your exam bank and replaced with a more operationally oriented question.

G. DAMPENING AXIAL XENON OSCILLATIONS

PURPOSE

The purpose of this procedure is to provide a means of dampening axial xenon oscillations and thus help stabilize the reactor core.

AL CONDITIONS

1. Not Applicable

STRUCTIONS

CAUTION: IF IT APPEARS THAT THE TECHNICAL SPECIFICATION LIMIT ON EITHER ROD INSERTION LIMITS OR AXIAL FLUX DIFFERENCE WILL BE VIOLATED, THEN IT WILL BE NECESSARY TO REDUCE POWER PRIOR TO PERFORMING THIS PROCEDURE.

NOTE: Utilize the delta flux ($\Delta\phi$) channel that is the most restrictive channel with respect to target flux when plotting xenon oscillations. When any channel is within 1.5% of target limit, al four channels must be continuously monitored.

1. Plot delta flux ($\Delta\phi$) vs. Time for approximately 24 hours or for a time period sufficient to determine the frequency of the axial oscillation and the midpoint about which the $\Delta\phi$ oscillates. (Best results are achieved when rods are > 215 steps on D).

2. Using the curve, predict when the peak at the top of core will occur. See OM Figure 49-5 for an example.

3. At approximately 1-1/2 hours before the peak at the top of the core, record $\Delta\phi$ and commence inserting control rods. When a $\Delta\phi$ corresponding to the midpoint of the oscillation is achieved, then maintain constant control rod position.

4. Find the difference between the $\Delta\phi$ recorded in step 3 and the midpoint determined in step 1. This difference is called E.

5. Delta Flux will now drift towards the bottom of the core. When it reaches a value of E below the midpoint, withdraw control rods to achieve the midpoint $\Delta\phi$ again. The oscillations should now be dampened.

6. Continue to plot $\Delta\phi$ for several hours. If the oscillations resumes as determined by a repeat of step 1 above, repeat steps 2 through 5 until the oscillation has been reduced to acceptable limits.

G. DAMPENING AXIAL XENON OSCILLATIONS (Continued)

REFERENCES

NOTE: All references used prior to March 10, 1986 are located in Section 5.

1. BVPS-2 OMCN 2-87-608 (Rev 1)

QUESTION 5.09 (2.00)

For EACH of the following statements below, state HOW (Increase, Decrease, No Change) ACTUAL Shut Down Margin (SDM) would be affected.

a. The plant is in Mode 5 when a charging pump is mistakenly started resulting in the injection of 200 gallons of boric acid into the RCS.

b. The plant is in Mode 3 when all the shutdown bank rods are withdrawn out of the core.

c. The plant status changes from Mode 5 to Mode 4.

d. A control rod drops into the core with the plant in Mode 1 at 50% power. The reactor does not trip.

ANSWER 5.09 (2.00)

a. Increase

b. Decrease (4 x 0.05)

c. No Change

d. No Change

REFERENCE

B.V.P.S. LP-RT-9 Enabling Objectives 2,4
B.V.P.S. LP-RT-9 pages 4-7
K/A 192002 K1.14 3.9
192002K114 ..(KA's)

COMMENT:

5.09.b

The answer for 5.09.b could be <u>No Change</u> if the BVPS Technical Specification (T.S.) definition of Shutdown Margin is applied. Refer to Attachment 5.09.b & c for T.S. definition. Please consider possible interpretations of the question during grading.

5.09.c

The answer key should be INCREASES since the mode change from 5 to 4 would result in a higher moderator temperature. With a negative moderator temperature coefficient, this would result in the addition of negative reactivity. Refer to Attachment 5.09.b & c for T.S. definition of Shutdown Margin.

5.09.b&C

CONTAINMENT INTEGRITY (Continued)

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.1.

1.8.2 All equipment hatches are closed and sealed.

1.8.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3, and

1.8.4 The containment leakage rates are within the limits of Specification 3.6.1.2.

1.8.5 The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

CHANNEL CALIBRATION

1.9 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.10 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.11 A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

CORE ALTERATION

1.12 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe conservative position.

SHUTDOWN MARGIN

1.13 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

QUESTION 6.01 (3.00)

The plant is operating at 50% power when a control system hot leg RTD fails high. Does this failure INCREASE, DECREASE, or NOT AFFECT the following: Consider each item independently. Assume no operator action and that all control systems are in automatic.

a. affected channel overpower delta T trip setpoint
b. steam bypass cooldown valves (first bank)
c. charging flow (initially)
d. control rod bank position
e. rod insertion limit setpoint
f. affected channel actual overtemperature delta T indication.

ANSWER 6.01 (3.00)

a. NOT AFFECT
b. NOT AFFECT
c. INCREASE (0.50 x 6)
d. DECREASE
e. INCREASE
f. NOT AFFECT

REFERENCE

B.V.P.S. 2LP-SQS-1.1 Enabling Objective 6
B.V.P.S. 2LP-SQS-1.3 Enabling Objective 10,12
B.V.P.S. 2LP-SQS-7.1 Enabling Objective 7
B.V.P.S. 2LP-SQS-21.1 Enabling Objective 4
B.V.P.S. - O.M. 2.01.1 pages 12,20; 2.7.1 page 35
2.21.1 page 22; 2.6.1 page 64
B.V.P.S. - Unit 2 Technical Specifications Table 2.2-1
K/A 001050 K5.01 3.6
K/A 004010 A1.01 3.6
K/A 041020 A3.02 3.4
041020A302 004010A101 001050K501 ..(KA's)

COMMENT:

6.01.f

The question asked for affected channel (control) actual overtemperature delta T indication for a failed high hot leg RTD. At BVPS, there are no indicators for actual overtemperature delta T. There are delta T indications for Control delta T, Protection delta T, and overtemperature delta T setpoint. The candidate should be given full credit for the following answers depending on which indicator he believes the question to be asking him about:

1. Control delta T - increases
2. Protection delta T - increases
3. Overtemperature delta T setpoint - decreases

See attachment 6.01.f, pages 1-5 for Vertical Board indicators.

SPECIFIC INSTRUMENTATION AND CONTROL

2RCS*TX410A

Type: Westinghouse 7300 NLD isolator
Function: Provide signal to RCP 21A thermal overload trip circuit

2RCS*TY410A

Type: Westinghouse 7300 signal comparator
Function: Provides signal to computer [T2882D] AUCT RCS TRAIN B TEMP RCT410 and Annunciator Window No. AE-3A AUCT RCS TEMPERATURE LOW

2RCS*TE410F

Sensing Point: Loop A cold leg RTD
Type: RDF CORP PLATINUM RTD
Range: 0-700F
Function: Provides a signal to 2RCS*TI410F

2RCS-TI410F

Type: Westinghouse VX252
Range: 0-700F
Function: Gives Loop A cold leg temperature indication at the Alternate Shutdown Panel (ASP).

2RCS*TE411B

Sensing Point: Loop A hot leg manifold
Type: RdF. Platinum RTD Model 21204
Range: 32-700F
Function: Provide signal to [2RCS-TU411B and TU411C]

2RCS-TU411C

Type: Westinghouse 7300 NSA summing amplifier
Function: Receives signal from [2RCS*TE411B and TE411C] and develops a Loop A average temperature signal which is passed to [2RCS-TU408] via the TAVG DEFEAT switch located at Vertical Board - Section B and also provides Loop A average temperature signal to the following:

SPECIFIC INSTRUMENTATION AND CONTROL

2RCS-TX408B

Type: Westinghouse 7300 NLP isolator
Function: Provide auctioneered high Tavg signal to [2RCS-TX408C]

2RCS-TX408C

Type: Westinghouse 7300 NLP isolator
Function: Provide signal to [2RCS-JU408J] Group 4 Rod Control

2RCS*TE411C

Sensing Point: Loop A cold leg manifold
Type: RdF Platinum RTD Model 21204
Range: 510-650F
Function: Provide signal to [2RCS-TC411C, and TU411B and TZ411C]

2RCS-TZ411C

Type: Westinghouse 7300 NCI computer input
Function: Provide signal to computer [T0402A], RCLA COLD TEMP

[redacted]

Type: Westinghouse 7300 NSA summing amplifier
Function: [redacted]

2RCS-TZ411B

Type: Westinghouse 7300 NCI computer input
Function: Provide signal to computer [T0404A], RCLA CONTROL DT

[redacted]

Type: Westinghouse VX-252
Range: 0-150%
[redacted]

SPECIFIC INSTRUMENTATION AND CONTROL

2RCS*TE411D

Sensing Point: Loop A hot leg manifold
Type: RdF Platinum RTD Model 21204
Range: 530-650F
Function: Installed spare

2RCS*TE412B

Sensing Point: Loop A hot leg manifold
Type: RdF Platinum RTD Model 21204
Range: 530-650F
Function: Provide signal to [2RCS-TT412H]

2RCS-TT412H

Type: Westinghouse 7300 NRA RTD amplifier
Function: Provide signal to [2RCS-TU412J and TU412K]

2RCS-TU412J

Type: Westinghouse 7300 NSA summing amplifier
Function: Provide signal to the following:

2RCS-TX412S

Type: Westinghouse 7300 NLP isolator
Function: Supply signal to [2RCS-TSH412B and TSH412C] and to [2RCS-TX412A]

2RCS-TX412A

Type: Westinghouse 7300 NLP isolator
Function: Provide signal to the following:

2RCS-TZ412A

Type: Westinghouse 7300 NCI computer input
Function: Provide signal to computer [T0403A], RCLA PROTECTION DT

SPECIFIC INSTRUMENTATION AND CONTROL

[redacted]

Type: Westinghouse VX-252
Range: 0-150%
Function: Indicate Loop A Delta T (I) at Vertical Board - Section B

2RCS-TR412

Type: Westinghouse Hagan Optimac Model 100, 3 Pen
Range: 0-150%
Function: Record Loop A Delta T (Pen 1), Loop Overtemperature Delta T Set Point (Pen 2), and Loop Overpower Delta T Set Point (Pen 3) at Vertical Board - Section B. Loop selected by REC LOOP SELECTOR switch at Benchboard - Section B, positions are Loop A - Loop B - loop C

2RCS*TE412C

Sensing Point: Loop A cold leg manifold
Type: RdF Platinum RTD Model 21204
Range: 510-630F
Function: Installed spare

2RCS*TE412D

Sensing Point: Loop A cold leg manifold
Type: RdF Platinum RTD Model 21204
Range: 510-630F
Function: Provide signal to [2RCS-TT412]

2RSC-TT412

Type: Westinghouse 7300 NRA RTD amplifier
Function: Provide signal to [2RCS-TU412J and TU412K]

2RCS-TU412K

Type: Westinghouse 7300 NAS summing amplifier
Function: Provide signal to the following:

2RCS-TX412L

Type: Westinghouse 7300 NLP isolator
Function: Supply signal to [2RCS-TU412G]

6

SPECIFIC INSTRUMENTATION AND CONTROL

2RCS-TSH412C

Type: Westinghouse 7300 NAL signal comparator
Function: Receive signal from [2RCS-TU412F and TX412S] and provide signal to [OVER-POWER DT REACTOR TRIP]

2RCS-TX412C

Type: Westinghouse 730 NLP isolator
Function: Provide signal to [2RCS-TR412] via REG LOOP SELECTOR switch and to the following devices:

2RCS-TI412C

Type: Westinghouse VX-252
Range: 0-150F
Function: Indicate Loop A Overtemp Delta T SP(1) at Benchboard - Section B

2RCS-TZ412C

Type: Westinghouse 7300 NCI computer input
Function: Provide signal to computer [T0410A], RCLA OVERTEMP DT 1 SP

2RCS*TE413

Sensing Point: Loop A hot leg
Type: Platinum RTD RDF Corp. P/N 21205
Range: 0-700F
Function: Provide signal to [2RCS-TX423A], to [2RCS*PS403A] for [2RCS*PCV456] interlock, and to the following:

2RCS-TI413A

Type: Westinghouse VX-252
Range: 0-700F
Function: Indicate Loop A hot leg temp at Emergency Shutdown Panel (SDP)

2RCS*TI413

Type: Westinghouse VX-252
Range: 0-700F
Function: Indicate Loop A hot leg temperature (PAM 1) at Vertical Board - Section A

QUESTION 6.03 (2.50)

Using Attachment 2, Op. Manual Fig. No. 13-2, "Quench Spray System," identify the following components on the attachment as specified in each part below.

a. Highlight the "A" quench spray pump recirculation flowpath back to the RWST. (0.50)

b. Circle the THREE (3) building/area boundaries that the "B" containment quench spray header passes through. (0.75)

c. Circle WHERE the flowrate for the "A" chemical injection pump is measured. (0.50)

d) Circle the THREE (3) valves that realign when the RWST level reaches the level setpoint for 2QSS-LSKK100B-1. (0.75)

ANSWER 6.03 (2.50)

a.	(0.50)		
b.	(0.75)	Use attached drawing	(0.50 x 6)
c.	(0.50)	as answer key	
d.	(0.75)		

REFERENCE

B.V.P.S. 2LP-SQS-13.1 Enabling Objective 2.4.5
B.V.P.S. Op. Manual Fig. No. 13-2
K/A 194001 A1.07 3.2
194001A107 ..(KA's)

COMMENT:

6.03.d

Question should have included written description of name for 2QSS*LSKK100B1 (RWST Level Extreme Low-Low). (See attachment for 6.03.d.)

6.03.d

SPECIFIC INSTRUMENTATION AND CONTROL

2QSS-LZ100A

Type: Westinghouse 7300 Series Signal Comparator
Function: Provides signal to computer point [L0500A], RWST Level

2QSS*LI100A

Sensing Point: Refueling water storage tank
Type: Westinghouse level indicator, model VX-252
Range: 0 to 730 inches
Function: Indicate REFUEL WTR STOR TK LEVEL on Vertical Board - Section C

2QSS*LT100B

Sensing Point: Refueling water storage tank
Type: Rosemount differential pressure transmitter, model 1153DB5PA
Range: -3 to 727 inches of water
Function: Provide level signals to the following equipment:

2QSS-LSK100B

Type: Westinghouse 7300 NAL2 dual signal comparator
Function: Provide level signals to computer point [L0517D], REFUEL WTR STG TK L LOW-LOW that corresponds to RSS pumps to cold leg recirculation switchover and to Annunciator Window No. A6-1D, REFUEL WATER STORAGE TANK LEVEL OFF NORMAL

2QSS-LSKK100B

Type: Westinghouse 7300 NAL2 dual signal comparator
Function: Provide signal to computer point [L0518D], REFUEL WTR STG TK L EXTREME LOW-LOW and Annunciator Window No. A6-1D, REFUEL WATER STORAGE TANK LEVEL OFF NORMAL

2QSS*LSKK100B1

Type: Westinghouse 7300 NAL1 Single Comparator
Function: Provides a level signal to [2QSS*LYKK100B1] for the following components: 2QSS*SOV100B, 2QSS*SOV101B, and 2QSS*SOV102B to initiate chemical injection valve switchover

QUESTION 6.04 (2.00)

a. HOW (Increase, Decrease, No Change) will an INCREASE in the reference junction temperature effect indicated thermocouple temperature?

b. HOW (High, Low, As Is) will an RTD temperature indication fail if a short circuit occurs across the RTD?

c. WHAT is the major disadvantage of using a Thermowell RTD for RCS wide range temperature measurement?

d. Given the graph shown in Attachment 3, identify the curve which represents the calibration curve for a HOT calibrated instrument.

ANSWER 6.04 (2.00)

a. Decrease (0.50)

b. Low (0.50)

c. Thermowell RTDs have a relatively long response time (0.50)

d. A (0.50)

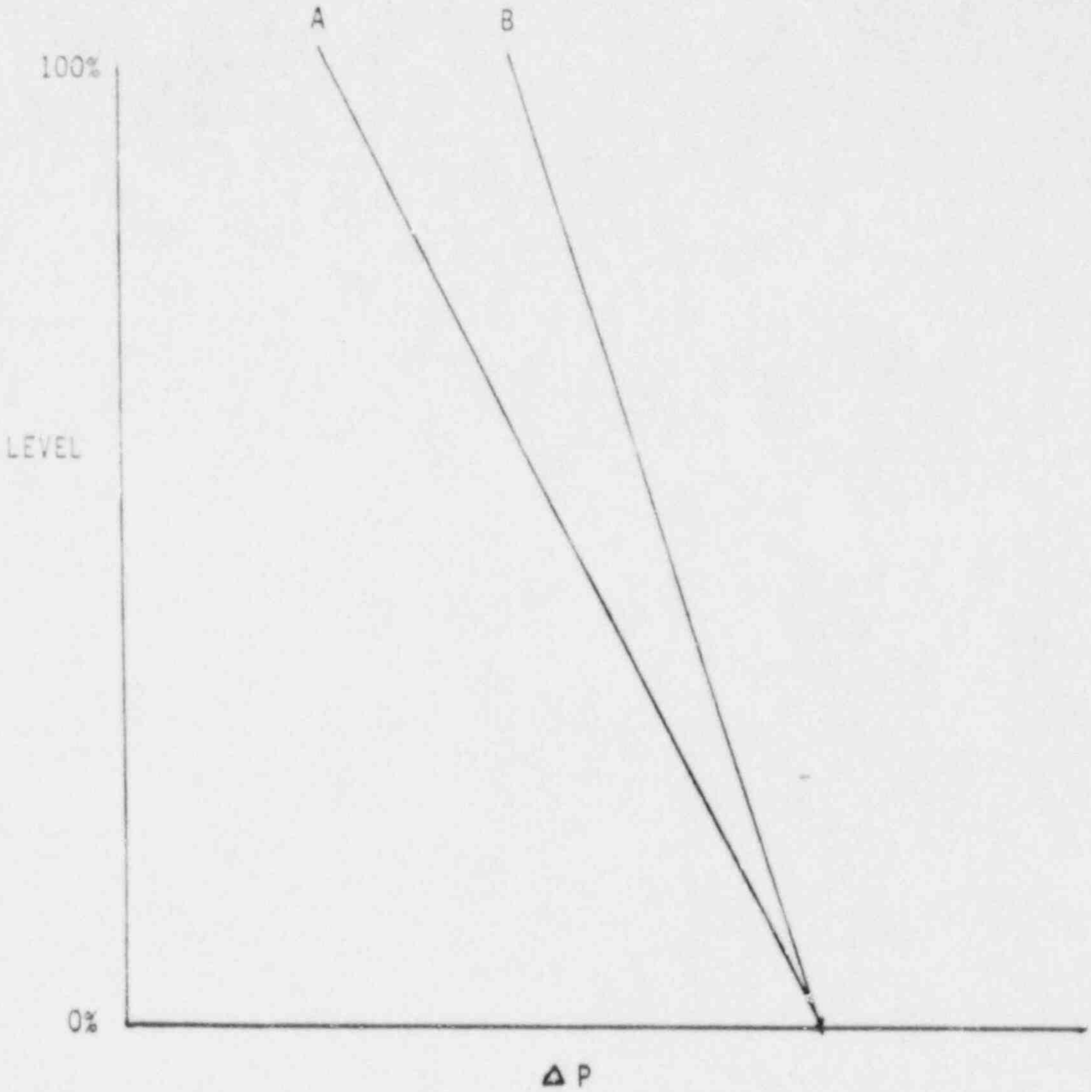
REFERENCE

B.V.P.S. LP-TMO-7 Enabling Objective 5
B.V.P.S. LP-TMO-7 page 11
K/A 191002 K1.13 2.8
K/A 191002 K1.14 2.9
191002K114 191002K113 ..(KA's)

COMMENT:

The answer to part d. is incorrect. For a given delta P, a cold calibrated channel should indicate a lower level than a hot calibrated channel due to the density difference. Therefore, the correct answer should be that curve B is the hot calibrated instrument.

ATTACHMENT 3



QUESTION 6.05 (2.00)

For EACH of the following radiation monitors, state the automatic actions which occur, if any, when the monitors alarm HIGH.

a. 2SWS-RQI101 - Component Cooling Service Water

b. 2HVR*RQI104A - Containment Purge

c. 2RMC*RQ201 - Control Room Area

d. 2GWS-RQI102 - Air Ejector Delay Bldg. Exhaust

ANSWER 6.05 (2.00)

a. none (4 x 0.50)

b. closes 2HVR*MOD23A and 2HVR*MOD23B (applicable valve names acceptable)

c. actuates control room pressurization

d. none

REFERENCE

B.V.P.S. 2LP-SQS-43.1 Enabling Objective 4
B.V.P.S. 2LP-SQS-43.1 pages 16,21,24,39
K/A 072000 G0.04 3.7
072000G004 ...(KA's)

COMMENT:

6.05.b

The answer key incorrectly identifies 2HVR*MOD23B as an auto action for 2HVR*RQ104A. The correct damper is 2HVR*MOD25A as seen in attachment 6.05.b page 1. The referenced lesson plan had a typographical error (see attachment 6.05.b, page 2). This has been entered into the Training Department's Action List and shall be corrected in the near future. Additionally, the applicable damper names have been provided for use as identified on the answer key. (See attachment 6.05.b, page 3.)

SPECIFIC INSTRUMENTATION AND CONTROL

[2HVL-DAU112]

Type: G.A. Technologies, RM-80
Range: Particulate: 10(-10) to 10(-5) µCi/CC
Gaseous: 10(-7) to 10(-2) µCi/CC
Function: Inputs to DRMS and to Ann. Windows A4-5A RADIATION MONITORING SYSTEM TROUBLE, A4-5C RADIATION MONITORING LEVEL HIGH and provides local alarm and indication.

[2HVL-VP112]

Type: KVRZ Model 455
Range: 0-20,000 CFM
Function: Senses Flow rate in the Condenstae Polishing Stack SKID and inputs to [2HVL-DV122]

[2HVR*RQ104A]

Sensing Point: Containment purge exhaust
Type: Beta Scintillator, inline duct mtd gas
Function: Inputs to [2HVR*DAU104A]

[2HVR*DAU104A]

Type: G.A. Technologies, RM-80
Range: 10(-6) to 10(-1) µCi/CC
Function: Inputs to ADDS (Annunciation and Digital Display System) Windows A4-5A RADIATION MONITORING SYSTEM TROUBLE, A4-5C RADIATION MONITORING LEVEL HIGH, and provides local alarm and indication as well as closes damper 2HVR*MOD23A and [2HVR*MOD25A] on high radiation and to a strip chart recorder and an RM-23 in the control room

[2HVR*RQ104B]

Sensing Point: Containment purge exhaust
Type: Beta Scintillator, inline duct mtd gas
Function: Inputs to [2HVR*DAU104B]

.05.B. page 2

b) Monitor Items - C/S (Checksource) actuation used in conjunction with the channel.

b. Power Supplies - 120 VAC is supplied to the monitor where it converts it to four 24 VDC outputs, three 5 VDC outputs, two 8 VDC outputs and a 5 VAC output. A +24 VDC output is supplied to two High Voltage Power Supplies (adjustable from 500 to 1250 VDC or 700 to 2000 VDC).

NOTE: RD-25A uses 250 to 1250 VDC.

c. Model 3101 Mark Numbers (RD-25A)

1) 2HVR*RQI104A - Containment Purge

TP-13

- Location - 780' Containment
- Function - Monitors containment activity during initial purge and refueling operations
- Power Supply - 120 VAC [PNL-AC2-E7] Bkr E7-10
- Auto Actions - Closes [2HVR*MOD23A] and [redacted] on high rad.

2HVR*DAU104A

- Location - 730' Service Building

2) 2HVR*RQI104B - Containment Purge

- Location - 780' Containment
- Function - Same as 2HVR*RQ104A
- Power Supply - 120 VAC [PNL-AC2-E8] Bkr E8-9

Auto Actions [redacted] and [2HVR*MOD25B]

2HVR*DAU104B

- Location - 730' Service Building

6.05.la'page 34,

ACX. LOCAL-CONTAINMENT PURGE [2HVR*RQ104A(B)] HIGH ALARM LEVEL

Ann. Window No. N/A

Setpoint
Later

Device
[2HVR*DAU104A(B)]

PROBABLE CAUSE

Radioactive gases or particulates in Containment.

CORRECTIVE ACTIONS

1. Verify the alarmed condition at the operators console.

2. Close or verify close the following motor operated dampers at the Building Service Panel.

   a. [2HVR*MOD23A], CNMT Purge Exhaust Isolation Damper

   b. [2HVR*MOD23B] CNMT Purge Exhaust Isolation Damper.

   c. [2HVR*MOD25A] CNMT Purge Air Supply Isol Damper.

   d. [2HVR*MOD25B] CNMT Purge Air Supply Isol Damper.

3. Evacuate the Containment.

4. Instruct all personnel in the Containment to report to RadCon for dose assessment and possible decontamination.

5. Notify RADCON, have them conduct surveys, if possible to locate the source of the activity.

6. Take remedial actions as necessar" to reduce the activity.

REFERENCES

NOTE: All references used prior to 4-13-87 are located in Section 5.

1. BVPS-2 OMCN 2-87-23 (Rev 1)

QUESTION 6.10 (2.50)

The plant is stable in Mode 5 with the "A" Residual Heat Removal System (RHS) in service.

a. At WHAT pressure (psig) will the RHS isolate from the RCS? (0.50)

b. WHAT is the design capacity of the RHS suction line relief valve [2RHS*RV721A]? Include ALL applicable information. (0.80)

c. Loss of primary component cooling water can affect WHAT TWO (2) RHS components, when operating? (0.70)

d. Failure of RHS Hx flow control valve, [2RHS*FCV605A] to the closed position will result in a (Increase, Decrease, No Change) to RCS temperature? (0.50)

ANSWER 6.10 (2.50)

a. >700 psig (0.50)

b. TWO (2) (0.25) charging pumps (0.25)
at the relief valve set pressure (0.30)

c. RHS heat exchanger (0.35)
RHS pump seal cooler (0.35)

d. Decrease (0.50)

REFERENCE

Enabling Objectives UNAVAILABLE
B.V.P.S. - O.M. 2.10.1 pages 1,2,5,6,20,21
K/A 000025 K1.01
K/A 000025 K3.02
K/A 000025 A1.01
000025A101 00025K302 000025K101 ..(KA's)

COMMENT:

6.10.d

The correct nomenclature for [2RHS*FCV605A] is provided in attachment 6.10.d. The nomenclature used in the question is incorrect.

Valving Completed ______________
SS or SOF

NO.	SIZE IN.	S&W TYPE	QA	NOMENCLATURE	FIG.NO./GRID	LOCATION	NSA
*365	3/4	VSS150-A-2	I	B Train Suction Hdr Test Conn.	10-10-2	Reactor Cntmnt.- EL.692'11"	S
*366	3/4	T78	I	Refueling Cavity Hdr Drain	20-1/E-8	Reactor Cntmnt. EL.709'6"	S
*367	3/4	VOS060-W-4	I	RWST Hdr TMD Connection	13-2/D-8	Safeguards Bldg EL.729'6"	S
*605A	8	FCV	I	RHS Train A Hx Bypass Flow Control	10-1/B-5	Reactor Cntmnt.- EL.707'6"	I
*605B	8	FCV	I	RHS Train B Hx Bypass Flow Control	10-1/F-5	Reactor Cntmnt.- EL.707'6"	T
625	1	VSS150-A-2	I	A Loop Supply to RHS Vent	10-1/D-2	Reactor Cntmnt EL.718'8"	S
*701A	12	MOV	I	RHS Train A Supply Isolation	10-1/C-1	Reactor Cntmnt.- EL.704'0"	S
*701B	12	MOV	I	RHS Train B Supply Isolation	10-1/D-1	Reactor Cntmnt.- EL.692'11"	S
*702A	12	MOV	I	RHS Train A Supply Isolation	10-1/C-1	Reactor Cntmnt.- EL.704'0"	S
*702B	12	MOV	I	RHS Train B Supply Isolation	10-1/D-1	Reactor Cntmnt.- EL.692'11"	S
*720A	10	MOV	I	RHS Train Return to B Loop Isolation	10-1/C-8	Reactor Cntmnt.- EL.692'11"	S
*720B	10	MOV	I	RHS Train Return to C Loop Isolation	10-1/F-8	Reactor Cntmnt.- EL.692'11"	S
*721A	3	RV	I	RHS Train A Supply Relief	10-1/C-1	Reactor Cntmnt.- EL.712'9"	-
*721B	3	RV	I	RHS Train B Supply Relief	10-1/E-1	Reactor Cntmnt.- EL.712'9"	-
*750A	2	MOV	I	RHS Train A Letdown Isolation	10-1/D-5	Reactor Cntmnt.- EL.718'8"	S

SECTION SEVEN GENERAL COMMENT:

Beaver Valley Power Station's administrative procedures on adherence to operating procedures state, "When extensive operations, infrequent operations or any operations requiring documentation are to be performed, the operating procedure must be present and followed." Because of this, there is no need to, and operators are not trained to, memorize procedures with the exception of the immediate action steps of the EOP's.

Section 7 of this examination contains seven questions (over 25% of the section) that require the candidate to repeat from memory information contained in Normal, Abnormal or Alarm Response Procedures that are required to be "present and followed" when these operations are performed. These questions are not a measure of an operators ability to do his job and should not be used to evaluate whether or not he should be granted a license.

QUESTION 7.01 (1.50)

For the following questions, assume B.V.P.S. - O.M. 51, Station Shutdown Procedure, is in use.

a. When using condenser steam dumps, WHAT operator action(s) must be taken to cooldown the RCS below the Lo-Lo Tavg setpoint? (0.50)

b. When the Residual Heat Removal System (RHS) is in operation, at least one rector coolant pump must remain in service until RCS temperature is less than 200 degrees F. WHY? (0.50)

c. If minimum RCS flow requirements CANNOT be met while in Mode 4, the operator's immediate response is to refer to WHAT procedure? (0.50)

ANSWER 7.01 (1.50)

a. place steam bypass interlock selection switch to the DEFEAT TAVG position (0.50)

b. prevent reactor vessel void formation (maintain RCS subcooling) (0.50)

c. B.V.P.S. - E.O.P. ES-0.2, "Natural Circulation Cooldown" (0.50)

REFERENCE

B.V.P.S. 2LP-SQS-21.1 Enabling Objectives 4:
2LP-SQS-50.51.52.1 Enabling Objectives 2,3
B.V.P.S. - O.M. 2.51.4 pages C9,D2,D4; 2.51.2 page 3; 2.53C.4 page 3
K/A 005000 G0.10 3.5
K/A 005000 G0.15 3.9
K/A 041020 A4.08 3.1
041020A408 005000G015 005000G010 ..(KA's)

COMMENT:

The answer to part c. is contained in cautions in the Normal Operating Procedures for Station Shutdown. It is not an immediate action in an emergency operating procedure and, therefore, is not required to be committed to memory by Beaver Valley's administrative procedures (Operating Manual 1/2.48.2.C.3). We ask that the question be withdrawn.

QUESTION 7.03 (2.00)

Answer the following questions concerning B.V.P.S. procedure AOP-2.1.3, "Continuous Insertion of RCCA Control Bank."

a. WHAT anticipated operational transient could cause a continuous bank insertion of the controlling bank? (0.50)

b. If a malfunction causes a RCCA control bank to insert past the Low-Low insertion limit, WHAT immediate operator action is required? (0.50)

c. If rod control is transferred to Manual and a continuous insertion condition is still present, WHAT TWO (2) operator actions should be performed? (1.00)

ANSWER 7.03 (2.00)

a. turbine runback (OTdt or OPdt) OR load rejection (0.50)

b. emergency boration OR boration at concentration and flowrate at least that as stated in Technical Specifications (0.50)

c. trip the reactor (0.50) and go to E-0 (0.50)

REFERENCE

B.V.P.S. - O.M. 53C AOP-2.1.3 page 1
B.V.P.S. - O.M. 1 page AAM1
K/A 001000 A1.04 3.9
K/A 001000 A3.02 3.6
001000A302 001000A104 ..(KA's)

COMMENT:

The answers to parts b. and c. are contained in an Alarm Response Procedure and an Abnormal Operating Procedure respectively. Neither of these procedures are required to be memorized. We ask that these questions be withdrawn.

QUESTION 7.06 (1.50)

a. WHAT procedure (by name) would you consult if annunciator A1-1E, "Containment Air Partial Pressure High-Low." alarmed?

b. WHAT could cause containment pressure to slowly increase with little or no humidity increase, and a possible decrease in temperature?

c. If the plant is in Mode 2, and containment pressure, temperature, and humidity ALL begin to increase rapidly, WHAT action should the operator take?

ANSWER 7.06 (1.50)

a. Loss of Containment Vacuum (AOP-2.12.1) (0.50)

b. a breach of (leakage into) containment (0.50)

c. manually trip the reactor (0.50)

REFERENCE

B.V.P.S. 2LP-SQS-53C.1 Enabling Objectives 1, 3
B.V.P.S. - O.M. AOP-2.12.1 page 1
K/A 000029 EA2.01 4.3
K/A 000029 G0.11 4.2
000069G011 000069A201 ..(KA's)

COMMENT:

When an annunciator alarms, as given in part a., the Alarm Response Procedure should be consulted. The Alarm Response Procedure will give direction for responding to the situation including references to other procedures. In this particular case, the Alarm Response Procedure does not reference the AOP for loss of containment vacuum. Since this annunciator is a symptom in the AOP, the AOP could also be consulted for guidance. We ask that either the Alarm Response Procedure or the AOP be an acceptable answer for full credit. We have submitted paperwork to the procedures group to change the Alarm Response Procedure so that it references the AOP.

The answer to part c. is contained in an AOP. It is not an immediate action and is not required to be memorized. We ask that this question be withdrawn.

QUESTION 7.07 (3.50)

A condition arises that requires entry into containment at 40% power. The operator entering containment needs to work in a gamma radiation field of 150 mrem/hr for approximately 2.0 hours. The below candidates are presented to you:

Candidate	1	2	3	4
Sex	male	male	female	male
Age	27	38	24	20
Qtr/exposure	--	1000 mrem	500 mrem	1000 mrem
Life exposure	1000 mrem	54730 mrem	5200 mrem	9500 mrem
Remarks	quarterly history unavailable	Form NRC-4 unavailable	3 months pregnant	

Each candidate is technically competent and physically capable of performing the task. All candidates have a completed Form NRC-4 and have a documented current calendar quarter exposure history, with the exceptions for those candidates stated above. Emergency limits do NOT apply. For EACH person, indicate if you would ACCEPT or REJECT the person to perform the task based on EXPOSURE REQUIREMENTS ONLY. Justify EACH answer AND include ALL applicable limits.

ANSWER 7.07 (3.50)

#1 - REJECT (0.25) since he has no quarterly history available and would exceed the 200 mrem/qtr whole body limit (0.50)

#2 - REJECT (0.25) since he does not have a Form NRC-4 available and would exceed the 1250 mrem/qtr whole body limit (0.50)

#3 - REJECT (0.25) since she will exceed the allowable exposure limit during the term of her pregnancy (0.50)

#4 - ACCEPT (0.25) since he will not exceed the quarterly limit (0.50) or the whole body limit of 10000 mrem lifetime exposure (0.50)

REFERENCE

Enabling Objectives UNAVAILABLE
B.V.P.S. - R.C.M. pages 5, 6, 7
K/A 194001 K1.03 3.4
194001K103 ..(KA's)

COMMENT:

Candidate #1 would be <u>accepted</u> based on 10 CFR 20 limits of 1250 mrem/qtr. This is the reference the candidate was give to use. Candidate #3 cannot be evaluated with the given information. Exposure limits for pregnant women are in Reg. Guide 8.13 which was not available. We ask that the key be changed to accept candidate #1 and that candidate #3 be deleted from the question.

QUESTION 7.08 (2.00)

Answer the following question concerning B.V.P.S - O.M. AOP-2.38.1, "Loss of 120 VAC Vital Bus."

WHAT are FOUR (4) automatic actions that an operator can visually verify in the control room if power to 120 VAC Vital Bus 1 is lost? ONLY consider safety system actuations.

ANSWER 7.08 (2.00)

1) atmospheric steam dump valves fail closed, if open

2) letdown will isolate (4 x 0.50)

3) PRZR heaters will deenergize

4) standby service water pumps (2SWE-P21A) auto starts, if not already running

5) component cooling water to containment instrument air compressor closes

6) primary component cooling water supply and return isolation valves (2CCP*MOV175-1,176-1,177-1,178-1) close

REFERENCE

B.V.P.S. - 2LP-SQS-53C.1 Enabling Objective 5
B.V.P.S. - O.M. AOP-2.38.1 pages 1,2
K/A 000057 EA2.19 4.3
000057A219 ..(KA's)

COMMENT:

The reference from the K/A catalog states "Ability to determine or interpret the plant automatic actions that will occur on the loss of a vital AC electrical instrument bus". The way to determine the automatic actions that will occur is to consult the AOP for loss of that vital bus. There are 17 different automatic actions that could occur depending on which vital bus is lost. It is not necessary to rely on an operator's memory to verify the correct auto actions for the correct vital bus failure. This is why procedures are required to be present and followed for infrequent operations such as responding to loss of vital bus. We ask that the question be withdrawn.

QUESTION 7.09 (1.00)

WHAT are the normal expected values for Source Range (SR) AND Intermediate Range (IR) Nuclear Instrumentation an operator would expect to see when verifying that the SR has reenergized after a reactor trip from power?

ANSWER 7.09 (1.00)

SR: 1E+5 (+/- 2.5E+4) counts/second (0.50)

IR: 1E-10 (+/- 0.5E-10) amps (0.50)

REFERENCE

B.V.P.S. - 2LP-SQS-2.2 Enabling Objective 4
B.V.P.S. - O.M. 2.2.4 pages B4, C1
K/A 000032 EA2.04 3.5
000032A204 ..(KA's)

COMMENT:

1 x 10^5 cps is the Source Range Hi-Flux Trip setpoint. The actual reading would be less than 10^5 cps. We ask that the answer key be changed accordingly.

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SET POINTS

Reactor Protection System
(Steamline Isolation)

High-2 containment pressure	3 PSIG

Reactor Protection System (Reactor Trip)

Source Range high level	10 E+5 CPS
Intermediate Range high level	Current equivalent to 25% of full power
Power Range, high range, high level	109% of full power
Power Range, low range, high level	25% of full power
Power Range, high neutron flux rate (positive)	+ 5% of rated thermal power with a timer ≥2 seconds constant
Power Range, high neutron flux rate (negative)	- 5% of rated thermal power with a timer ≥2 seconds constant
High pressurizer pressure	2385 PSIG
High pressurizer water level	92% of span
Low pressurizer pressure	1945 PSIG
Lead time constant	10 seconds
Lag time constant	1 second
Loss of Primary Coolant Flow	
Low flow	90%
Low frequency	≥ 57.5 Hz
Low voltage	2750 VAC
Undervoltage time delay	0.5 second
Reactor coolant pump trip	2/3 pump breakers auto trip
Low-Low steam generator water level	15.5% of span
Coincident low level and steam/feedwater flow mismatch	Steamflow 40% greater than feed flow with 25% of SG water level

QUESTION 7.10 (3.00)

Answer the following questions concerning B.V.P.S. - O.M. 2.24.2, "Steam Generator Feedwater System."

a. **WHAT action must an operator take in order to prevent a reactor trip if a Steam Generator (SG) Feed Pump Auto Stop annunciator alarms with the plant at 75% power?** (0.50)

b. WHAT are FIVE (5) indications/conditions that an operator would verify if a Hi-Hi SG level trip occurred with the plant at 40% power? (2.00)

ANSWER 7.10 (3.00)

a. place the SG Startup Feedwater Pump in service (0.50)

b.
- turbine trip
- main feedwater pump (MFWP) tripped
- MFWP discharge valves closed (5 x 0.50)
- MFW reg valves closed
- SG bypass flow control valves closed
- MFW isolation trip valves closed

REFERENCE

B.V.P.S. - 2LP-SQS-24.1 Enabling Objectives 7, 9A (14)
B.V.P.S. - O.M. 2.24.2 page AAE1
K/A 000054G0.09 3.1
K/A 000054 G0.10 3.2
000054G010 000054G009 ..(KA's)

COMMENT:

The answer to part a. is the second corrective action of an Alarm Response Procedure and is not required to be memorized. We ask that this question be withdrawn.

QUESTION 7.11 (2.50)

Answer the following questions concerning Liquid Waste System Operation.

a. WHICH TWO (2) flowrates (numerical values NOT required) are used in calculating the Unit 2 Cooling Tower Blowdown Flow when the Unit 2 blowdown flow instrument [2CWS-FR101] is out of service, and a liquid waste discharge is to be made by way of the Unit 1/2 cooling tower blowdown line? (1.00)

b. Before sampling the contents of the "A" waste drain tank, WHAT action must be taken by the operator? Include any applicable precautionary setpoints or time related values. (1.00)

c. WHAT action should an operator take if local-liquid waste process effluent [2SGC-RQI100] high alarm actuates AND is verified to be in the alarmed condition? (0.50)

ANSWER 7.11 (2.50)

a. Unit 1 and 2 (cooling tower) flow (from [FR-CW-101]) (0.50)
Unit 1 cooling tower blowdown flow (0.50)

b. recirculate the tank (0.50) for a minimum of TWO (2) tank volumes OR 8.5 hours (0.50)

c. verify closed ([2SGC-HSV-100]) liquid waste EFF high rad isolation valve (0.50)

REFERENCE

B.V.P.S. - 2LP-SQS-17.1 Enabling Objectives 2d, 9, 5e
B.V.P.S. - O.M. 2.17.2 page 1, 2.43.4 PAGE AEE1
K/A 000059 EA2.02 3.9
K/A 000059 EA2.05 3.9
000059A205 00059A202 ..(KA's)

COMMENT:

The answer to part a. is contained in a Normal Operating Procedure and is not required to be done from memory. We ask that the question be withdrawn.

QUESTION 8.01 (3.00)

Using Attachment 4, classify the following events in accordance with BV-2 EPP/I-1, Recognition and Classification of Emergency Conditions, AND justify your answer and any assumptions. Consider each case separately.

a. B.V.P.S. EOP E-1, "Loss of Reactor or Secondary Coolant," is in use. Pressurizer level is off-scale low and RCS pressure is 1500 psig and decreasing. The reactor was manually tripped because pressurizer level could not be maintained.

b. A turbine trip from 75% power occurred and the reactor did not automatically trip (ATWS). The reactor remained critical until an operator manually inserted control rods.

c. A truck carrying Ammonia gas is involved in a collision at the plant main entrance. Gas is leaking from the truck.

d. An earthquake is registered on-site with the plant in Mode 1. The severe ground motion results in the generation of a missile in the turbine building from the detachment of a LP turbine blade.

ANSWER 8.01 (3.00)

a. SITE AREA (0.40)
TAB 5 -- RCS/Containment leak exceeds ;ke-up capacity (0.35)

b. ALERT (0.40) TAB 14 -- Reactor not subcritical after valid scram signal(s) (0.35)

c. UNUSUAL EVENT (0.40)
TAB 18 -- Toxic gas nearby release potentially harmful (0.35)

d. ALERT (0.40)
TAB 28 -- Turbine rupture causing casing penetration (0.35)

REFERENCE

Enabling Objectives UNAVAILABLE
B.V.P.S. - Unit 2 Implementing Procedures BV-2 EPP/I-1 Table 1
K/A 194001 A1.16 4.4
194001A116 ..(KA's)

COMMENT:

It was required of the candidate to utilize a copy of the BVPS-2 EAL Tab Matrix for the purpose of classifying an event. This comment is written to inform you that the matrix is not to be used for classification purpose but only as a guide to choose the most appropriate Tab. For future testing, please include a copy of both the EAL Tab Matrix (Attachment 1) and the EAL Tabs (Attachment 2) if it is required for a candidate to classify an EPP event. (Refer to enclosed references.)

Question 8.01

2.2 Emergency Action Levels (EAL's) are specified in the TABS (Attachment 2) to this procedure. EAL TAB reference guides are contained in Attachment 1 to this procedure and in the BVPS Emergency Plan, Table 4.1.

NOTE.

The EAL's in the TABS to this procedure have precedence over other EAL's matrix which should be used for a quick reference to the TABS and not for classification purposes.

2.3 EAL's will be triggered by the results of offsite dose projections and/or assessment of plant status by the onshift operating staff or the Emergency Organization, if activated. In many cases, the proper classification will be immediately apparent. In other cases, more extensive assessment is necessary to determine the applicable emergency classification.

2.4 Attachment 1 provides EAL TAB references and should only be used as a guide which demonstrates how an initiating condition leads directly to the appropriate emergency classification, based on the magnitude of the event. Plant operating personnel should use Attachment 1 as a guide only.

2.5 Attachment 2 (TAB's) provides the data sheets to be used in classifying emergency conditions. Each data sheet identifies the initiating condition (ie., RCS Pressure Hi/Low) and the respective Emergency Action Levels for each classification. Identified with each EAL is a representative listing of the various instruments and other indicators which may be symptoms of an emergency, which should be used to assess and classify the condition. Symptoms <u>should</u> <u>not</u> be confused with the actual EAL criteria.

2.6 The EAL's have been developed to provide adequate response to postulated emergency conditions that could exist at Beaver Valley. Emergency conditions could arise, however, that are not covered by a specific EAL. In these cases emergency conditions should be classified by the general definitions of the emergency classes which are:

a. Unusual Event - Off-normal events which could indicate a potential degradation of the level of safety of the plant.

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b. Alert - Events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant.

c. Site Area Emergency - Events which involve actual or likely major failures of plant functions needed for protection of the public.

d. General Emergency - Events which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity.

E. Procedure

1.0 Verify

1.1 Upon receipt of an initial indication (alarm, surveillance report, observation, etc.) that an emergency condition may exist, an assessment shall be initiated to verify the validity of the indication and whether the EAL criteria have been met. This may be performed by comparison to redundant instrument channels, comparison to other related plant parameters, physical observations and field measurements.

1.2 If this assessment cannot be completed within 15 minutes of the initial indication, it shall be considered that the emergency condition indicated does exist and appropriate emergency actions shall be initiated in accordance with the applicable emergency implementing instructions or procedures.

2.0 Classification

2.1 Using Attachment 1 for guidance, locate the appropriate Tab in Attachment 2. Tabs are arranged by initiating event(s); either a single plant parameter (ie., RCS Pressure Hi/Low) or parameters (ie., Fuel Clad Degradation).

2.2 Determine the appropriate emergency classification by comparing the verified plant parameters to the EAL's for each emergency condition.

REFERENCE ONLY
NOT TO BE USED FOR
CLASSIFICATION PURPOSES

ACTION LEVEL CRITERIA FOR CLASSIFICATION OF EMERGENCY CONDITIONS

INITIATING CONDITION	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
	Off-normal Events Which Could Indicate a Potential Degradation of the Level of Safety of the Plant.	Events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant.	Events Which Involve Actual or Likely Major Failures of Plant Functions Needed for Protection of the Public.	Events Which Involve Actual or Imminent Substantial Core Degradation or melting with Potential for Loss of Containment Integrity.
Radioactive Effluent Applicable to Any Release Point(s) and Resulting from Any Initiating Event TAB 1	Unplanned airborne release gives offsite dose rate greater than 0.5 mRem/hr. -or- Unplanned liquid release in excess of MPC limits.	Unplanned airborne release gives offsite dose rate greater than 2.0 mRem/hr. -or- Unplanned liquid release results in downstream community water radioactivity greater than 12 times EPA standards.	Release Corresponds to 20 mrem/hr. at Site Boundary -or- Offsite Dose Due to Event Is Projected to Exceed 170 mrem to Whole Body or Child Thyroid.	Radiological effluent release results in offsite dose projected to exceed 1 rem to the Whole Body or 5 rem to the Child Thyroid. and/or Radiological effluent corresponds to greater than 125 mRem/hr. whole body dose rate or 600 mRem/hr. child thyroid at the site boundary.
Release or Loss of Control of Radioactive Material Within the Plant. TAB 2		Fuel Handling Accident Resulting in Release of Radioactivity to Occupied Areas Such That the Direct Radiation Levels in the Areas Increase by a Factor of 1000 -or- Other Verified, Uncontrolled Events Which Result in an Unexpected Increase of In-Plant Direct Radiation Levels by a Factor of 1000.	Major Damage to Spent Fuel Due to Fuel Handling Accident -or- Uncontrolled Decrease in Fuel Pool Water to Below Level of Fuel.	

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REFERENCE ONLY

ACTION LEVEL CRITERIA FOR CLASSIFICATION OF EMERGENCY CONDITIONS

...IATING CONDITION	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
Initiation of ECCS TAB 11	Valid Safety Circuit Trip or Necessary Manual Initiation.			Loss of 2 of 3 Fission Product Barriers With a Potential Loss of Third Barrier. Applicable to Any Initiating Event that May Lead to this Condition. -or- Any Initiating Events, from Whatever Source that Makes Release of Large Amounts of Radioactivity in a Short Time Probable, For Example: 1. LOCA With Failure of ECCS. 2. LOCA With Initially Successful ECCS. Subsequent Failure of Heat Removal Systems with Likely Failure of Containment. 3. Loss of All Onsite and Offsite Power Concurrent With Total Loss of Emergency Feedwater. 4. Loss of Feedwater and Condensate Followed by Failure of Emergency Feedwater System. 5. Reactor Protection System Fails to Initiate or Complete a Required Scram, followed by Loss of Core Cooling and Make-up Systems -or- Loss of Plant Control Occurs
RCS Pump Failure TAB 12	→	RCS pump Seizure (locked rotor) leading to fuel failure		
Loss of Containment Integrity TAB 13	Requiring Shutdown by LCO		Containment Pressure 5 & 45 psig	
Loss of Engineered Safety or Fire Protection Features TAB 14	Requiring Shutdown by LCO			
Failure of Reactor Protection System to Initiate or Complete a Scram TAB 15	→	Reactor Not Subcritical after Valid Scram Signal(s).		
Loss of Plant Control/Safety Systems TAB 16	→	Loss of Capability to Achieve Cold Shutdown	Loss of Capability to Achieve Hot Shutdown	
Loss of Indicators, Annunciators or Alarms TAB 17	Loss on Process or Effluent Parameters, Requiring Shutdown by LCO	Loss of All Alarms (Annunciators) Sustained for 5 mins.	Loss of All Alarms 15 min with Plant Not in Cold S/D -or- Plant Transient Occurs While All Alarms are Lost.	
Control Room Evacuation TAB 18	→	Required or Anticipated. Control of Shutdown Systems Established at Remote Shutdown Panel.	Required. Shutdown System Control at Remote Shutdown Panel Not Established Within 15 min.	

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TABLE 4.1 - ACTION LEVEL CRITERIA FOR CLASSIFICATION OF EMERGENCY CONDITIONS

INITIATING CONDITION	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
Explosion TAB 27	Near or On-site Explosion Potential Significant Damage	Known Damage to Facility, Affecting Operation.	Severe Damage to Safe Shutdown Equipment.	Loss of 2 of 3 Fission Product Barriers With a Potential Loss of Third Barrier. Applicable to Any Initiating Event that May Lead to this Condition. -or- Any Initiating Events, from Whatever Source that Makes Release of Large Amounts of Radioactivity in a Short Time Probable, For Example: 1. LOCA With Failure of ECCS. 2. LOCA With Initially Successful ECCS. Subsequent Failure of Heat Removal Systems with Likely Failure of Containment. 3. Loss of All Onsite and Offsite Power Concurrent With Total Loss of Emergency Feedwater. 4. Loss of Feedwater and Condensate Followed by Failure of Emergency Feedwater System. 5. Reactor Protection System Fails to Initiate or Complete a Required Scram, Followed by Loss of Core Cooling and Make-up Systems -or- Loss of Plant Control Occurs.
Aircraft TAB 28	Unusual Activity Over Facility -or- Aircraft Crashes Onsite	Aircraft or Missile From Whatever Source Strikes and Significantly Degrades a Station Safety Structure.	Crash Affects Vital Structures by Impact or Fire.	
Train TAB 29	Derailment in Onsite Areas			
Watercraft TAB 29	Strikes Intake Structure, Resulting in Flow Reduction			
Contaminated Injury TAB 29	Transportation of Injured and Contaminated Individual(s) to Offsite Hospital.			
Accident at SSDP TAB 29	Requires Protective Actions at BVPS			
Oil Pipeline Rupture TAB 29	Rupture of Pipeline Onsite w/ or w/o Fire			
Turbine Rupture TAB 29	Turbine rotating component failure causing rapid plant shutdown.	Turbine failure causing casing penetration		

REFERENCE ONLY
NOT TO BE USED FOR
CLASSIFICATION PURPOSES

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QUESTION 8.02 (3.00)

Using B.V.P.S. - Unit 2 Technical Specification, list ALL applicable action statements, by number, for EACH of the following equipment failures. Consider EACH failure independently.

a. The fuel oil transfer pump for Diesel generator 21 has been found to be inoperable. A reactor startup is in progress with reactor power at 1% and increasing.

b. RHS Heat Exchanger outlet thermocouples, TE606A and B, have been found to be inoperable.

c. Control room bottled air system pressure is found to be at 1500 psig.

ANSWER 8.02 (3.00)

a. 3.8.1.1 (A.C. Sources) (0.50) AND 3.04 (cannot continue startup since you cannot change modes by relying on action statements (0.50)

b. 3.3.3.5. (remote shutdown monitoring) (1.00)

c. 3.7.7.1.b (control room habitability; 4.7.7.2.a specifies pressure requirement of 1825 psig) (1.00)

REFERENCE

Enabling Objectives UNAVAILABLE
B.V.P.S. - Unit 2 Technical Specifications
B.V.P.S. - O.M. 2 page 2.10.1
K/A 06200 G0.05 3.8
K/A 016000 G0.05 3.5
016000G005 062000G005 ..(KA's)

COMMENT:

8.02.a

The question asks to list ALL applicable action statements for the equipment failures listed. Part a. of the question states that a fuel oil transfer pump for Diesel Generator 21 had been found inoperable. The answer stated that the Diesel Generator was inoperable by T.S. 3.8.1.1 due to part b.3. This is incorrect. The Diesel Generator is still operable, since it has 2 fuel oil transfer pumps and would still meet the requirements of the Technical Specification LCO. Therefore, the correct answer is - NONE - No Technical Specifications applicable. (See attached references.)

Question 8.02.a

MAJOR COMPONENTS

Bore and stroke (inches)	15.75 x 18.11
Total displacement In. (3)	42,324
Brake horsepower	5,899
Operating speed, RPM	514
Alarm protection energized, RPM	360
Compression ratio	13:1
Lube oil system capacity, Gal.	1,400
Cooling system capacity, Gal. (closed loop)	600
Fuel oil day tank, Gal.	1,100
Air starting system supply pressure, PSIG minimum	300

The two emergency units are located in the Diesel Generator Building and are physically and electrically isolated from each other.

Each unit is capable of carrying the required emergency load on its respective bus during step loading and steady state following a loss of preferred AC power to the 4 KV emergency buses.

Diesel engine supporting systems consist of a fuel oil system, a starting system, a cooling system, a lubricating oil system, a turbocharger, and engine protective devices.

Each emergency diesel generator is equipped with an overspeed governor which shuts off the injection of fuel to the cylinders when the engine exceeds a speed of 565 to 576 rpm.

Fuel Oil System

The fuel oil system for each emergency diesel generator consists of a 58,000 gallon emergency diesel generator fuel oil storage tank [2EGF*TK21A, 21B] [redacted] emergency diesel generator fuel oil transfer pumps, a 1,100 gallon emergency diesel generator fuel oil day tank, an engine driven fuel pump, and an electric driven fuel priming pump.

Question 8.02.a

3/4.8 ELECTRICAL POWER SYSTEMS

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. Separate day tank containing a minimum of 350 gallons of fuel,

2. A separate fuel storage system containing a minimum of 53,225 gallons of fuel,

3. **A separate fuel transfer pump,**

4. Lubricating oil storage containing a minimum total volume of 504 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 Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.5 within one 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 COLD SHUTDOWN within the next 36 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 Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.5 within one 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 COLD SHUTDOWN within the next 36 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 COLD SHUTDOWN within the next 36 hours.

Question 8.02.a

ELECTRICAL POWER SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

c. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by performing Surveillance Requirements 4.8.1.1.2.a.5 within one 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 4 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 COLD SHUTDOWN within the next 36 hours.

d. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within one 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 COLD SHUTDOWN within the next 36 hours. Restore at least two diesel generators to OPERABLE status within 72 hours from time of initial loss or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determine OPERABLE at least once per 7 days by verifying correct breaker alignment, indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months by transferring (manually and automatically) unit power supply from the unit circuit to the system circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the day tank,

2. Verifying the fuel level in the fuel storage tank,

Question 8.02.a

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that a sample of diesel fuel from the fuel storage tank is within the acceptable limits specified in Table 1 of ASTM D975 when checked for viscosity, water and sediment,

4. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank,

5. Verifying the diesel starts from ambient condition,

6. Verifying the generator is synchronized, loaded to $\geq$ 4,238 kw, and operates for at least 60 minutes, and

7. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

8. Verifying the lubricating oil inventory in storage.

b. At least once per 18 months during shutdown by:

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service,

2. Verifying the generator capability to reject a load of $\geq$ 825 kw without tripping,

3. Simulating a loss of offsite power in conjunction with a safety injection signal, and:

    a) Verifying de-energization of the emergency busses and load shedding from the emergency busses.

    b) Verifying the diesel starts from ambient condition on the auto-start signal, energizes the emergency busses with permanently connected loads, energizes the auto-connected emergency loads through the load sequencer and operates for $\geq$ 5 minutes while its generator is loaded with the emergency loads.

4. Verifying that on a loss of power to the emergency busses, all diesel generator trips, except engine overspeed, generator differential current, and generator overexcitation are automatically disabled.

5. Verifying the diesel generator operates for at least 60 minutes while loaded to $\geq$ 4,238 kw.

SURVEILLANCE REQUIREMENTS (Continued)

6. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000 hour rating of 4,535 kw.

7. Verifying that the automatic load sequence timer is OPERABLE with each load sequence time within ±10% of its required value.

c. Check for and remove accumulated water:

1. From the day tank, at least once per 31 days and after each operation of the diesel where the period of operation was greater than 1 hour, and

2. From the fuel oil storage tank, at least once per 92 days.

d. At least once per 92 days and from new fuel oil prior to its addition to the storage tanks by verifying that a sample obtained in accordance with ASTM D270-1975 meets the following minimum requirements and is tested within the specified time limits:

1. As soon as a sample is taken (or prior to adding new fuel to the storage tank) verify in accordance with the tests specified in ASTM D975-1977 that the sample has:

a) A water and sediment content of less than or equal to 0.05 volume percent;

b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes;

c) An API Gravity of within 0.3 degrees of 60°F, or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate or an absolute specific gravity at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89, or an API Gravity of greater than or equal to 27 degrees but less than or equal to 39 degrees; and

2. Within one week after obtaining the sample, verify an impurity level of less than 2 milligrams of insolubles per 100 milliliter is met when tested in accordance with ASTM D2274-1970. The analysis on the sample may be performed after the addition of new fuel oil.

3. Within two weeks of obtaining the sample, verify that the other properties specified in Table 1 of ASTM D975-1977 and Regulatory Guide 1.137 Position 2.a are met (when tested in accordance with ASTM D975-1977). An analysis for sulfur shall be performed in accordance with ASTM D129, ASTM D1552-1979 or ASTM D2622-1982.

Question 8.02.a

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting** both diesel generators simultaneously, during shutdown, and verifying that both diesel generators accelerate to at least 514 rpm in less than or equal to 10 seconds.

f. At least once per 10 years by:

1) Draining each main fuel oil storage tank, removing the accumulated sediment, and cleaning the tank using a sodium hypochlorite solution or other appropriate cleaning solution, and

2) Performing a pressure test, of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code, at a test pressure equal to 110% of the system design pressure.

**This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading recommendations.

QUESTION 8.09 (2.25)

Use B.V.P.S. - Unit 2 Technical Specification Table 3.3-6 and determine WHAT SEVEN (7) Area or Process radiation monitoring instruments must be functional following a LOCA.

ANSWER 8.09 (2.25)

2RMR-RQ205,206 (Containment Area (0.50)
2HVS-RQ109C (Mid Range Noble Gas) (0.50)
2HVS-RQ109D (High Range Noble Gas) (0.50)
2MSS-RQ101A,B,&C (Main Steam Discharge (0.75)

REFERENCE

B.V.P.S. 2LP-SQS-43.1 Enabling Objective 4
B.V.P.S. - Unit 2 Technical Specifications Table 3.3-6 Action 36
K/A 016000 G0.04 3.4
016000G004 ..(KA's)

COMMENT:

8.09

The question asks to determine what SEVEN (7) Area or Process radiation monitoring instruments must be functional following a LOCA (using Table 3.3-6). However, Table 3.3-6 does not address a condition of operability, for the radiation monitors listed, following a LOCA. They must only be OPERABLE per their applicable modes as outlined in the table. However, it may be interpreted, that with the conditions stated (i.e., use of Table 3.3-6 and following a LOCA), the plant could possibly be in Modes 1-4. At that time, all 11 of the following radiation monitors must be operable:

- Containment Area (2RMR-RQ206,207)
- Control Room Area (2RMC-RQ201,202)
- RCS Leakage Detection Gas & Particulate (2RMR-RQ303A,B)
- Mid Range Noble Gas (2HVS-RQ109C)
- High Range Noble Gas (2HVS-RQ109D)
- Main Steam Discharge (2MSS-RQ101A,B&C)

Therefore, the correct answer should be <u>any</u> 7 of the 11 radiation monitors listed above. (See attached references.)

Question 8.09

INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-3.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	SETPOINT#	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Fuel Storage Pool Area (2RMF-RQ202)	1	*	≤ 75.8 mR/hr	10^{-1} to 10^{4} mR/hr	19
b. Containment [illegible] ([illegible]-RQ206 & 207)	2	1, 2, 3 & 4	≤ 3.29×10^{3} R/hr	1 to 10^{7} R/hr	36
c. Control Room [illegible] ([illegible]-RQ201 & 202)	[illegible]	1, 2, 3, 4, 5## & 6##	≤ 0.476 mR/hr	10^{-2} to 10^{3} mR/hr	46, 47
2. PROCESS MONITORS					
a. Containment					
i. Gaseous Activity (Xe-133) [illegible] ([illegible]-RQ303B)	1	1, 2, 3 & 4	N/A	10^{-6} to 10^{-1} µCi/cc	20
ii. Particulate Activity (I-131) [illegible]	1	1, 2, 3 & 4	N/A	10^{-10} to 10^{-5} µCi/cc	20
b. Fuel Building Vent					
i. Gaseous Activity (Xe-133) (2RMF-RQ301B)	1	**	≤ 7.82×10^{-6} µCi/cc	10^{-6} to 10^{-1} µCi/cc	21

*With fuel in the storage pool or building
**With irradiated fuel in the storage pool
#Above background
##During movement of irradiated fuel

Question 8.09

TABLE 3.3-6 (Continued)

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	SETPOINT#	MEASUREMENT RANGE	ACTION
2. PROCESS MONITORS (Continued)					
ii. Particulate (I-131) (2RMF-RQ301A)	1	**	$<6.70\times10^{-9}$ µCi/cc	10^{-10} to 10^{-5} µCi/cc	21
c. Noble Gas and Effluent Monitors					
i. Supplementary Leak Collection and Release System					
1) [redacted]		1, 2, 3 & 4	N.A.	10^{-4} to 10^{2} µCi/cc	36
2) [redacted]		1, 2, 3 & 4	N.A.	10^{-1} to 10^{5} µCi/cc	36
ii. Containment Purge Exhaust (Xe-133)(2HVR-RQ104A & B)	1	6	< 3 x background	10^{-6} to 10^{-1} µCi/cc	22
iii. [redacted]	1/SG	1, 2, 3 & 4	$< 3.9 \times 10^{-2}$ µCi/cc	10^{-2} to 10^{3} µCi/cc	36

#Above background

Question 8.09

TABLE 3.3-6 (Continued)

ACTION STATEMENTS

ACTION 19 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 20 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 21 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the applicable ACTION requirements of Specifications 3.9.12 and 3.9.13.

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

ACTION 36 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 43 - With the number of OPERABLE channels less than required by the Minimum channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

2) Return the channel to OPERABLE status within 30 days or explain in the next Semi-Annual Effluent Release Report why the inoperability was not covered in a timely manner.

ACTION 46 - With the number of OPERABLE channels one less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable channel to OPERABLE status within 7 days or close the control room series normal air intake and exhaust isolation dampers.

ACTION 47 - With no OPERABLE channels either restore one inoperable channel to OPERABLE status within 1 hour or close the control room series normal air intake and exhaust isolation dampers.

QUESTION 8.11 (1.25)

a. WHAT is the FULL Technical Specification Basis for the RCS operational leakage limit stated in 3.4.6.2c? (0.80)

b. WHAT Technical Specification (state by number) addresses the surveillance program established to prevent the leakage limits in 3.4.6.2c. from becoming an operational concern?

ANSWER 8.11 (1.25)

a. Ensure that the dosage contribution (0.20) from the tube leakage will be limited to a small fraction of the 10 CFR Part 100 limits (0.20) in the event of either a steam generator tube rupture (0.20) or a steam line break (0.20)

b. 3/4.4.5 (0.45)

REFERENCE

Enabling Objective UNAVAILABLE
B.V.P.S. - Unit 2 Technical Specifications Section 3/4.4.6.2, 3/4.4.5
K/A 000037 G0.04 3.9
000037G004 ..(KA's)

COMMENT:

8.11.a

The question asks to state the FULL Technical Specification Basis for the RCS operational leakage limit stated in 3.4.6.2c. The answer given is stated as it appears in the Bases Section and assigns point values for certain portions of the statement. By using the word FULL in the question, it implies that only this answer is acceptable. It is requested that other answers be accepted which convey the intent of the basis even though it may not be written exactly as it appears in the BVPS-2 Technical Specifications. (See attached references.)

Question B.11.a

REACTOR COOLANT SYSTEM

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 reactor-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 2235 ± 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.

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 2235 ± 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

Question 8.11.a

REACTOR COOLANT SYSTEM

BASES

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems."

3/4.4.6.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 28 GPM with the modulating valve in the supply line fully open at RCS pressures in excess of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the accident analyses.

[redacted]

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Should PRESSURE BOUNDARY LEAKAGE occur through a component which can be isolated from the balance of the Reactor Coolant System, plant operation may continue provided the leaking component is promptly isolated from the Reactor Coolant System since isolation removes the source of potential failure.

3/4.4.6.3 PRESSURE ISOLATION VALVE LEAKAGE

The leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series valve failure. It is apparent that when pressure isolation is provided by two in-series valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are

ATTACHMENT 3

NRC Response to Facility Examination Review Comments

SECTION 5

5.03c.: Comment accepted.

5.05a.: Comment accepted. The question was deleted from the examination, reducing the value of Section 5 by 0.50 points.

5.05b.: Comment accepted. The question was deleted from the examination, reducing the value for Section 5 by 0.50 points.

5.06: Comment accepted.

5.08: Comment not accepted. Both of the enabling objectives require the candidate to fully understand xenon oscillations. LP-RT-7 Enabling Objectives (EO) 5 states. "Describe Xenon oscillations," and EO 6 states, "Discuss Xenon oscillation dampening at both BOL and EOL." The question was not intended to measure the candidate's ability to control xenon oscillations, which is not addressed by the enabling objectives, but simply to measure his understanding of WHEN and HOW xenon oscillations could affect plant operations.

5.09b.: Comment noted.

5.09c.: Comment accepted.

SECTION 6

6.01f.: Comment accepted. The question was deleted because the question did not accurately describe the indications available to an operator at BVPS, thus resulting in the possibility for more than one correct answer. The value of Section 6 was reduced by 0.50 points.

6.03: Comment noted.

6.04: Comment not accepted. The calibration curves show that if the delta P correction for a hot calibrated and cold calibrated instrument are the same at 0% level, then the hot calibrated instrument, because of its lower fluid density, needs more delta P correction than the cold calibrated instrument to balance the reference leg pressure at 100%.

6.05b.: Comment accepted. In addition, the applicable valves names were added to the answer key.

6.10: Comment noted.

RESPONSE TO SECTION SEVEN GENERAL COMMENT

NRC NUREG-1021, "Operator Licensing Examiner Standards," Chapter ES-202 and ES-402 state that each candidate must demonstrate complete knowledge and understanding of symptoms, automatic actions and immediate action steps specified in abnormal and emergency procedures. For a candidate to demonstrate a complete knowledge and understanding of abnormal and emergency procedures, he must be able to recognize an abnormal condition, and be able to take knowledge based actions to mitigate the consequence of the abnormal condition. This is the position stated by Beaver Valley Power Station in its letter from J. D. Sieber to S. J. Collins, of the NRC, dated July 28, 1987. Section 7 of this examination contains questions which require the candidate to demonstrate his knowledge of abnormal procedures by examining his knowledge and understanding of inter-system relationships and time dependent, knowledge based operator actions.

SECTION 7

7.01: Comment not accepted. This question calls for the candidate to recognize an abnormal condition ("minimum RCS flow requirements CANNOT be met while in Mode 4") and to demonstrate the necessary knowledge to proceed with the mitigation of the consequences of the abnormal condition (knowing the correct procedure to implement, Emergency Operating Procedure (EOP) ES-0.2, "Natural Circulation Cooldown"). The facility's Lesson Plan (LP) for the EOPs does not address the requirement for an operator to know the conditions for entry into the EOPs. Without a facility LP enabling objective, the K/A catalog rating is used to determine the safety significance of the question. The K/A catalog importance rating for this question is 3.9 (on a scale of 1.0 to 5.0).

7.03: Comment not accepted. The comment presented by the facility for parts b. and c. of this question does not address the issue concerning memorization of knowledge based operator actions. A senior operator who does not possess the prerequisite knowledge to direct an operator to Emergency Borate, does not possess the required level of knowledge necessary to ensure the plant will remain within its design basis envelope. Additionally, an operator who could not recognize the need to manually trip the reactor, under the conditions of the question, would not possess the level of knowledge required to ensure safe operation of the plant.

7.06a.: Comment noted.

7.06c.: Comment not accepted. The knowledge level required to ensure safe operation of the plant should encompass the immediate actions an operator would take to place the plant into a safe condition (ie. manually trip the reactor).

7.07: Comment noted. The answer key has been modified as follows:

#1 - Accept [0.25] since he is allowed 1250 mrem/qtr [0.50]
#2 - ... AS PER ANSWER KEY ... [0.75]
#3 - ... AS PER ANSWER KEY ... [0.75]
#4 - Reject [0.25] since he has already exceeded his whole body limit of 10000 mrem lifetime exposure [exposure [0.50]

This revision resulted in a reduction in the value of Section 7 by 0.50 points.

7.08: Comment not accepted. The question addresses system design interactions and relationships. An operator should not need to consult the AOP if his level of knowledge met the requirements of the facility training program, as discussed in the lesson plan referenced in the answer key (Attachment 1) regarding each of the systems affected by a loss of 120 VAC Vital Bus 1.

7.09: Comment accepted.

7.10a.: Comment not accepted, per the AOP, the loss of a steam generator feed pump requires that prompt, time dependent operator action to be taken. If an operator possessed an acceptable level of knowledge concerning SG water level control and feedwater system operation, he would be able to correctly identify the need to place the SG startup feedwater pump into service. However, after further review of the question and its answer, the following alternate correct response was identified:

"reduce power to within the capacity of 1 MFWP"

This response was added to the answer key.

7.11a. Comment accepted. The question was deleted reducing the value of Section 7 by 0.50 points.

SECTION 8

8.01: Comment noted.

8.02a.: Comment accepted. The answer key was modified and the value of Section 8 reduced by 0.50 points.

8.09: Comment not accepted. Only the seven (7) monitors listed in the answer key have the required range to allow for their operation in a post-LOCA environment.

8.11: Comment noted.

ATTACHMENT 3

NRC Response to Facility Examination Review Comments

SECTION 5

5.03c.: Comment accepted.

5.05a.: Comment accepted. The question was deleted from the examination, reducing the value of Section 5 by 0.50 points.

5.05b.: Comment accepted. The question was deleted from the examination, reducing the value for Section 5 by 0.50 points.

5.06: Comment accepted.

5.08: Comment not accepted. Both of the enabling objectives require the candidate to fully understand xenon oscillations. LP-RT-7 Enabling Objectives (EO) 5 states. "Describe Xenon oscillations," and EO 6 states, "Discuss Xenon oscillation dampening at both BOL and EOL." The question was not intended to measure the candidate's ability to control xenon oscillations, which is not addressed by the enabling objectives, but simply to measure his understanding of WHEN and HOW xenon oscillations could affect plant operations.

5.09b.: Comment noted.

5.09c.: Comment accepted.

SECTION 6

6.01f.: Comment accepted. The question was deleted because the question did not accurately describe the indications available to an operator at BVPS, thus resulting in the possibility for more than one correct answer. The value of Section 6 was reduced by 0.50 points.

6.03: Comment noted.

6.04: Comment not accepted. The calibration curves show that if the delta P correction for a hot calibrated and cold calibrated instrument are the same at 0% level, then the hot calibrated instrument, because of its lower fluid density, needs more delta P correction than the cold calibrated instrument to balance the reference leg pressure at 100%.

6.05b.: Comment accepted. In addition, the applicable valves names were added to the answer key.

6.10: Comment noted.

RESPONSE TO SECTION SEVEN GENERAL COMMENT

NRC NUREG-1021, "Operator Licensing Examiner Standards," Chapter ES-202 and ES-402 state that each candidate must demonstrate complete knowledge and understanding of symptoms, automatic actions and immediate action steps specified in abnormal and emergency procedures. For a candidate to demonstrate a complete knowledge and understanding of abnormal and emergency procedures, he must be able to recognize an abnormal condition, and be able to take knowledge based actions to mitigate the consequence of the abnormal condition. This is the position stated by Beaver Valley Power Station in its letter from J. D. Sieber to S. J. Collins, of the NRC, dated July 28, 1987. Section 7 of this examination contains questions which require the candidate to demonstrate his knowledge of abnormal procedures by examining his knowledge and understanding of inter-system relationships and time dependent, knowledge based operator actions.

SECTION 7

7.01: Comment not accepted. This question calls for the candidate to recognize an abnormal condition ("minimum RCS flow requirements CANNOT be met while in Mode 4") and to demonstrate the necessary knowledge to proceed with the mitigation of the consequences of the abnormal condition (knowing the correct procedure to implement, Emergency Operating Procedure (EOP) ES-0.2, "Natural Circulation Cooldown"). The facility's Lesson Plan (LP) for the EOPs does not address the requirement for an operator to know the conditions for entry into the EOPs. Without a facility LP enabling objective, the K/A catalog rating is used to determine the importance of the question. The K/A catalog importance rating for this question is 3.9 (on a scale of 1.0 to 5.0).

7.03: Comment not accepted. The comment presented by the facility for parts b. and c. of this question does not address the issue concerning memorization of knowledge based operator actions. A senior operator should possess the required level of knowledge necessary to ensure the plant will remain within its design basis envelope by directing emergency boration. Additionally, an operator should recognize the need to manually trip the reactor, under the conditions of the question.

7.06a.: Comment noted.

7.06c.: Comment not accepted. The knowledge level required to ensure safe operation of the plant should encompass the immediate actions an operator would take to place the plant into a safe condition (ie. manually trip the reactor).

7.07: Comment noted. The answer key has been modified as follows:

#1 - Accept [0.25] since he is allowed 1250 mrem/qtr [0.50]
#2 - ... AS PER ANSWER KEY ... [0.75]
#3 - ... AS PER ANSWER KEY ... [0.75]
#4 - Reject [0.25] since he has already exceeded his whole body limit of 10000 mrem lifetime exposure [exposure [0.50]

This revision resulted in a reduction in the value of Section 7 by 0.50 points.

7.08: Comment not accepted. The question addresses system design interactions and relationships. An operator should not need to consult the AOP if his level of knowledge met the requirements of the facility training program, as discussed in the lesson plan referenced in the answer key (Attachment 1) regarding each of the systems affected by a loss of 120 VAC Vital Bus 1.

7.09: Comment accepted.

7.10a.: Comment not accepted. Per the AOP, the loss of a steam generator feed pump requires that prompt, time dependent operator action to be taken. Concerning SG water level control and feedwater system operation, an operator should be able to correctly identify the need to place the SG startup feedwater pump into service. However, after further review of the question and its answer, the following alternate correct response was identified:

"reduce power to within the capacity of 1 MFWP"

This response was added to the answer key.

7.11a. Comment accepted. The question was deleted reducing the value of Section 7 by 0.50 points.

SECTION 8

8.01: Comment noted.

8.02a.: Comment accepted. The answer key was modified and the value of Section 8 reduced by 0.50 points.

8.09: Comment not accepted. Only the seven (7) monitors listed in the answer key have the required range to allow for their operation in a post-LOCA environment.

8.11: Comment noted.