ATTACHMENT 3

Duquesne Light

Nuclear Group
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December 9, 1987
ND3VPN:5249

Mr. Robert M. Gallo
Operations Branch
Division of Reactor Safety
U. S. Nuclear Regulatory Commission
Region 1
631 Park Avenue
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 December 2, 1987 at our facility.

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

Very truly yours

A.J. Morabito

for J. D. Sieber
Vice President Nuclear

JDS/TWB:cal

Enclosure

cc: Central File (2)

FACILITY REVIEW OF WRITTEN EXAMINATION

QUESTION 1.02 (2.00)

Beaver Valley Unit 2 has a total reactor coolant flow rate of 9.566 E7 lbm/hr at a hot leg temperature of 606 degrees F and a cold leg temperature of 546 degrees F. Steam generator pressure is 770 psig. Assume the feedwater entering the steam generators is at saturation. Calculate the total feedwater flow rate in lbm/hr. Neglect all other heat inputs and losses. Assume the specific heat capacity of the reactor coolant is 1.0. State all other assumptions. SHOW ALL WORK.

ANSWER 1.02 (2.00)

Qrx = m1c[T(hot)] - T(cold)] where m1 = mass flow rate of RCS
(0.25 for formula)

Qsg = m2[h(stm) - h(feed)] where m2 = mass flow rate of feedwater
(0.25 for formula)

(Qrx = Qsg, therefore,) m1c[T(hot) - T(cold)] = m2[h(stm) - h(feed)]
(0.25 for relationship)

therefore, $m2 = \frac{m1c[T(hot) - T(cold)]}{h(stm) - h(feed)}$

from steam tables, h(stm) = 1200 BTU/lbm (0.25)
h(feed) = 507 BTU/lbm (0.25)

therefore, $m2 = \frac{9.566\ E7\ (1)\ (606 - 546)}{1200 - 507} \frac{lbm}{hr}$

therefore, m2 = 8.28 E6 lbm/hr (0.75)

REFERENCE
BV EXAM BANK, 1-32
BV LP-TMO-5, LO. 3
BV LP-TMO-3, LO. 7

3.3	3.1	4.0	KA VALUE(S)
002000K501	002000K511	193003K125	...(KA'S)

RO EXAM REVIEW (12-2-87)

1.02 Point values of 0.25 are established in the answer key for stating the formulae for Qrx and Qsg. The relationship between these two formulae can be established without needing to state each individually. Therefore, it is requested that grader discretion be used and that an overall understanding of the answer development be used as a guide instead of a particular response (i.e., the two formulae).

FACILITY REVIEW OF WRITTEN EXAMINATION

QUESTION 1.04 (2.50)

Explain HOW and WHY reactor power AND Tave respond during and for ONE (1) hour following ONE (1) minute of Emergency Boration at the following power levels until equilibrium conditions are attained. No other operator actions are taken. Assume rod control is in manual and no manual or automatic protection signals are generated.

a. 100% equilibrium rated power (1.50)

b. Critical at 1 E-8 amps following a refueling outage (1.00)

ANSWER 1.04 (2.50)

a. Power decreases initially (0.10) due to negative reactivity added by boration (0.20), but will subsequently increase (to match secondary power) (0.20) due to (positive reactivity added from) decreasing Tave (0.20).

Tave decreases initially (0.20) due to primary to secondary power mismatch (0.20) and continues to decrease until after boration is stopped (0.20) and will stabilize when reactor power equals secondary power (0.20).

b. Power decreases initially (0.10) due to (negative reactivity added by) boration (0.20), and continues to decrease (0.10) until it stabilizes at a level caused by equilibrium subcritical multiplication (0.20).

Tave does not change (0.20) because it is independent of power (at power levels < POAH). (0.20)

REFERENCE
BV EXAM BANK, 5-28
BV LP-RT-7, LO. 3,4
3.8 KA VALUE(S)
192008K120 ...(KA'S)

RO EXAM REVIEW (12-2-87)

1.04.b. The answer reflects that the examinee must state that Tavg does not change due to its power independence. Other answers which indicate Tavg independent of power, (i.e., steam dump setting), should also be accepted. (Refer to attached copy of BV Exam Bank, 5-28).

Question 1.04.b

5. Theory of Nuclear Power Plant Operation, Fluids, and Thermodynamics

5-28 a. Explain the response of reactor power and Tave during and after 2 minutes of Emergency Boration at 100% power. Assume rod control is in manual.

b. Explain the response of reactor power and Tave after 2 minutes of Emergency Boration at 10E-8 amps and no load Tave.

ANSWER:

5-28 a. Power decreases initially due to the boron addition. The primary to secondary mismatch causes Tave to decrease. The decrease in Tave inserts positive reactivity and restores reactor power to the same as initial power level.

b. Tave does not change due to the boration. ~~(Tave is determined by the amount of pump heat and the steam dump setting.)~~ After the initial transient, power decreases at a negative 1/3 DPM rate.

FACILITY REVIEW OF WRITTEN EXAMINATION

QUESTION 1.06 (1.50)

Since the DNBR is not a directly observable parameter, name SIX dissimilar parameters the operator monitors and/or controls to ensure the DNBR limit is not violated.

ANSWER 1.06 (1.50)

(any 6 at 0.25 points each)

1. RCS pressure
2. RCS temperature
3. RCS flow
4. Rx power
5. AFD
6. QPTR
7. Rod Position (sequencing, overlap, alignment)
   (CONSIDER OTHERS ON CASE-BY-CASE BASIS)

REFERENCE

BEAVER VALLEY THERMODYNAMICS, CH. 7, P. 17,18
BEAVER VALLEY TECH. SPECS., BASES, P. 2-2 THROUGH 2-5
BV LP-TMO-7, LO. 11

KA VALUE(S)	...(KA'S)
2.9	001000G006
3.4	193008K105

RO EXAM REVIEW (12-2-87)

1.06 Rod Position (sequencing, overlap, alignment) is given as an acceptable answer. It is requested that each of the items in parenthesis can be considered as an individual answer by it[self]. Therefore, the single answer of "rod position" would be rep[laced by] the following three individual responses:

- proper rod sequencing
- proper bank overlap
- rod alignment

FACILITY REVIEW OF WRITTEN EXAMINATION

QUESTION 2.06 (1.00)

Given the following methods/paths of secondary heat removal:

1. One residual heat release valve, HCV-104

2. One SG atmospheric steam dump valve, PCV-101A

3. One condenser steam dump valve, TCV-106H

a. Which ONE method/path has the greatest capacity for heat removal? (0.50)

b. Which ONE method/path can be controlled from the Alternate Shutdown Panel? (0.50)

ANSWER 2.06 (1.00)

a. 1. (0.50)

b. 2. (0.50)

REFERENCE
BVPS 2LP-SQS-21.1, PP. 5-8, 17
BVPS 2LP-SQS-21.1, ELO 4,15

3.3	3.1	3.6	2.9	KA VALUE(S)
039000K102	039000K106	041020A202	041020G009	(KA'S)

RO EXAM REVIEW (12-2-87)

2.06.a The question asks for the one method/path of greatest heat removal capacity. The choices are:

1. Residual Heat Release Valve
2. SG Atmospheric Dump Valve
3. Condenser Steam Dump Valve

The answer found in the answer key is: 1. Residual Heat Release Valve. This is the incorrect response. The correct answer is: 3. Condenser Steam Dump Valve. Included is a copy of the Interoffice Correspondence, dated 8/5/86, which lists the capacities of the Residual Heat Release Valve at various pressures. The capacity of the Condenser Steam Dump Valve, as per O.M. 21, has a maximum value of 890,000 lbm/hr. Therefore, in comparison, the Condenser Steam Dump Valve has a larger capacity for heat removal. (Refer to attached copy of OM 2.21.1, p.10, 11; interoffice correspondence dated 8/5/86)

Question 2.06.a

MAJOR COMPONENTS

Condenser Steam Bypass Valves [2MSS-TCV106A, 106B, 106C, 106D, 106E, 106F, 106G, 106H, 106J, 106K, 106L, 106M, 106N, 106P, 106Q], [2MSS-PCV106A, 106B, 106C]

Eighteen 8-inch steam dump valves are located on the south side of the main condenser neck, nine on each half. The steam dump valves are designed to pass up to 90% of full load steam flow. Main steam pressure is always on the steam dump valve inlets. To prevent moisture collection in the valve inlets and steam dump lines, the inlets of each steam dump are piped to a drain line.

Fifteen of the valves are designated temperature control valves [2MSS-TCV106A through 106Q]. These valves are blocked or locked out by Lo-Lo Tavg. The remaining three valves are designated cooldown valves [2MSS-PCV106A,B,C]. The Lo-Lo Tavg interlock may be bypassed for these three valves to allow cooldown of the reactor coolant system.

The eighteen steam dump valves are divided into four banks for control. The first and third banks consist of five valves each. Banks two and four have four valves each.

The steam dump valves are Copes-Vulcan eight inch, D100-160-3 reverse acting tandem trim valves equipped with direct acting top works and a direct acting Moore positioner. Increasing air pressure pushes the valve stem down to open the valve. Decreasing air pressure allows the spring to lift the valve stem, closing the valve. The positioner uses a booster relay for faster valve response. Diaphragm operating air is routed through four solenoid valves. The first two solenoid valves control the Lo-Lo Tavg interlock, Train A and B. The third solenoid valve acts as the arming solenoid for condenser available, Rx trip, load rejection, and steam pressure signals. The fourth solenoid is the trip open or modulate solenoid.

~~Each steam dump valve is mechanically restricted for maximum stroke to prevent a single valve flow of greater than 890,000 pph.~~ This limits the consequences of a stuck open valve.

Each steam dump valve discharge is piped into the condenser neck to discharge at an elevation just above the sixth point heater bottom. The steam is prevented from impinging the sixth point heaters by a deflector which directs the steam out, up, and down from the heater. The steam dump discharges are staggered the length and width of the condenser to distribute the heat load.

MAJOR COMPONENTS

Design Data

[2MSS-TCV106A] (Typical for all)

Type	8" D100-160-3
[redacted]	[redacted]
Pressure (Inlet/Outlet), PSIG	1085/-15
Temperature, F	556
Action, air-to-open	
Fail position (air/electricity)	Closed/Closed

The steam dump system has two automatic modes of operation, steam pressure mode and Tavg mode. The operational mode is operator selected by the Steam Dump Control Mode Selector Switch on the benchboard. In steam pressure mode, only the first two banks of valves are operational and they modulate to maintain the steam pressure setpoint set by the operator, using the benchboard mounted steam pressure controller. In Tavg mode, two steam dump controllers are available. The reactor trip controller operates the steam dump valves to restore no load Tavg following a reactor trip. Only the first two banks of valves are operational after a reactor trip. The load rejection controller operates all four banks of valves for large load rejections and the first two banks for small load rejections, to restore Tavg to program value.

All 18 valves trip closed if Tavg reaches Lo-Lo Tavg. If it is desired to cooldown the reactor plant, the Lo-Lo Tavg interlock may be manually defeated for the three cooldown valves only. Since the Lo-Lo Tavg interlock is dual train, two Steam Dump Control Interlock Selector Switches are provided, one for each train. The Steam Dump Control Interlock Selector Switches are also used for manually blocking the steam dump control system.

All 18 steam dump valves are blocked when the condenser is not available. To be available, the condenser must have sufficient vacuum and at least one cooling tower pump running.

Load rejection is sensed by turbine first stage pressure. First stage pressure transmiter [2MSS*PT447 sends a signal to bistables [PC447A] and [PC447B]. These bistables trip on rate of change of first stage pressure. [PC447A] trips on a rapid reduction in first stage pressure equivalent to a loss of load between 15% and 50%. [PC447B] trips on a 50% load rejection. The bistables are designed to latch ON since the rate of change signal will disappear as soon as first stage pressure reaches its new value. [PC447A] unblocks or arms the first and second bank of valves. [PC447B] unblocks or arms the third and fourth bank of valves. Both arming signals are negated if the condenser is unavailable or if Tavg reaches Lo-Lo setpoint. [PC447A and B] are reset by momentarily placing the Control Mode Selector switch to RESET. The switch spring returns to TAVG.

SENDER — RETAIN YELLOW COPY.
FORWARD WHITE AND PINK COPIES.

REPLIER — RETURN WHITE COPY.
RETAIN PINK COPY FOR FILE.

INTEROFFICE CORRESPONDENCE

TO: MARK GIGLIO	LOCATION 8	SUBJECT / REFERENCE / J.O. NO. 12241 2BVS-209A TELECOPY FROM BILL PFROMMER TO MARK GIGLIO 4/18/8[illegible]
FROM: ED HOYLE	LOCATION 8	

MESSAGE: — PLEASE REFER TO THE SUBJECT TELECOPY:

1) VALVE DRAWING HAS BEEN CORRECTED, SUBMITTED & APPROVED.

2) DESIGN PARAMETERS WILL BE SHOWN IN SPEC AS FOLLOWS:

2SVS*RCV161A,B,C

FLOW CONDITION	Cv	FLOW PPH	PRESS IN
MIN	164	26,200 ✓	100 psia ✓
NOR.	100	155,000 ✓	975 ✓
MAX.	212	260,832	800
**	164	236,000	908

2SVS*HCV104

FLOW CONDITION	Cv	FLOW PPH	PRESS IN
MIN	360	52,400	100
NOR.	95	155,000	1020
MAX.	* 360	467,872	921
✓ **	360	487,000	920

* E. HOYLE CALC. BASED ON REDUCED STROKE
** E. HOYLE CALC. BASED ON P_1 AT HEADER OF 1055 psia. IN BOTH CASES, COPES USED P_1 = 1040 psia.

DATE SIGNATURE TELEPHONE

REPLY:

3) NORMAL HEADER PRESSURE IS 1055 psia, HOWEVER, THIS VALUE SHOULD NOT APPEAR ON THE DATA SHEET INASMUCH AS THE ABOVE "PRESS IN" ARE USED IN THE SIZING OF THE VALVES. THE HEADER PRESSURE DETERMINES THE INLET PRESS AT THE VALVE, BUT, TO DO THIS, THE LINE CONFIGURATION IS REQUIRED.

4) NOTE 2 WILL BE CLARIFIED BY ADDING Cv VALUES.

5) SEE 2

6) SWEC CALC K-3 DATED 12/20/84.

7) SHOULD BE NOTE 3 & IS A W REQUIREMENT. 2 ABOVE MAX SHOULD SATISFY THIS REQUIREMENT.

8) SPEC 2BVS-209A WILL BE REVISED SEPT. '86.

CC JOE CRAMER 8

8-5-86 — DATE
[signature] — SIGNATURE
5376 — TELEPHONE

A 010 130

FACILITY REVIEW OF WRITTEN EXAMINATION

QUESTION 3.02 (1.50)

Indicate how the affected OT delta-T and OP delta-T SETPOINTS will INITIALLY change (INCREASES, DECREASES, or NO CHANGE) if the following events occur with reactor power at 50%. Consider OP delta-T SETPOINT and OT delta-T SETPOINT for each event separately.

a. Pressurizer pressure decreases from 2235 psig to 2150 psig. (0.50)

b. N-41 lower detector fails low. (0.50)

c. The narrow-range loop 3 T-hot RTD output fails low. (0.50)

ANSWER 3.02 (1.50)

a. OT delta-T setpoint DECREASES (0.25)
OP delta-T setpoint NO CHANGE (0.25)

b. OT delta-T setpoint DECREASES (0.25)
OP delta-T setpoint DECREASES (0.25)

c. OT delta-T setpoint INCREASES (0.25)
OP delta-T setpoint INCREASES (0.25)

REFERENCE
BVPS TECHNICAL SPECIFICATION, TABLE 2.2.1
BVPS 2LP-SQS-1.1, ELO 7

3.1	2.9	KA VALUE(S)
01200A205	012000K611	(KA'S)

RO EXAM REVIEW (12-2-87)

3.02.b The answer for OP delta-T is incorrect. The answer key states that the OP delta-T setpoint will decrease as a result of N-41 failing low. However, the delta flux input to the OP delta-T setpoint is set to zero (for all delta-I) and will not be affected by the N-41 lower detector failing low. Therefore, the correct response is no change. (Refer to attached copy of T.S. 2.2.1, p. 2-9, 10)

3.02.c The answer for OP delta-T is incorrect. The answer key states that the OP delta setpoint will increase as a result of the narrow range loop 3 T-hot RTD output failing low. However, the setpoint is unaffected by a decreasing average temperature and/or temperatures $\leq$ 576.2°F. As verified by the OP delta-T setpoint formula of Technical Specification 2.2.1 (p. 2-9,10), K5 is 0 for decreasing average temperature and K_6 is 0 for $T \leq T''$ (T'' = 576.2°F). Therefore, the correct answer is no change. (Refer to attached copy of T.S. 2.2.1, p. 2-9,10)

Question 3.02.b.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS
NOTATION (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \left\{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T''\right] - f_2(\Delta I)\right\}$$

Where:	ΔT	=	Measured ΔT by RTD Manifold Instrumentation;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$		=	Lead-lag compensator on measured ΔT;
τ_1, τ_2		=	Time constants utilized in lead-lag compensator for ΔT, $\tau_1 = 8$ s, $\tau_2 = 3$s;
$\frac{1}{1 + \tau_3 S}$		=	Lag compensator on measured ΔT;
τ_3		=	Time constant utilized in the lag compensator for ΔT, $\tau_3 = 0$ s;
ΔT_o		=	Indicated ΔT at RATED THERMAL POWER;
K_4		=	1.0781;
K_5		=	0.02/°F for increasing average temperature and 0 for decreasing average temperature;
$\frac{\tau_7 S}{1 + \tau_7 S}$		=	The function generated by the rate-lag compensator for T_{avg} dynamic compensation;
τ_7		=	Time constant utilized in rate-lag compensator for T_{avg}, $\tau_7 = 10$ s;

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS
NOTATION (Continued)

$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 s;
K_6	=	0.0012/°F for T > T" and K_6 = 0 for T ≤ T";
T	=	Average Temperature, °F;
T"	=	Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, ≤576.2°F);
S	=	Laplace transform operator, s^{-1}; and
$f_2(\Delta I)$	=	0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.6% of ΔT span.

NOTE 5: The sensor error for temperature is 1.72% and 0.73% of span for pressure.

NOTE 6: The sensor error for steam flow is 1.0%, for feedwater flow is 1.0%, and for steam pressure is 0.83% of span.

Question 3.02.c.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS
NOTATION (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau^1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T''\right] - f_2(\Delta i)\}$$

Where: ΔT = Measured ΔT by RTD Manifold Instrumentation;

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT;

τ_1, τ_2 = Time constants utilized in lead-lag compensator for ΔT, τ_1 = 8 s, τ_2 = 3s;

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT;

τ_3 = Time constant utilized in the lag compensator for ΔT, τ_3 = 0 s;

ΔT_o = Indicated ΔT at RATED THERMAL POWER;

K_4 = 1.0781;

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature;

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the rate-lag compensator for T_{avg} dynamic compensation;

τ_7 = Time constant utilized in rate-lag compensator for T_{avg}, τ_7 = 10 s;

Question 3.02.c.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS
NOTATION (Continued)

$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 s;
K_6	=	0.0012/°F for T > T" and K_6 = 0 for T < T";
T	=	Average Temperature, °F;
T"	=	Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, ≤576.2°F);
S	=	Laplace transform operator, s^{-1}; and
$f_2(\Delta I)$	=	0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.6% of ΔT span.

NOTE 5: The sensor error for temperature is 1.72% and 0.73% of span for pressure.

NOTE 6: The sensor error for steam flow is 1.0%, for feedwater flow is 1.0%, and for steam pressure is 0.83% of span.

FACILITY REVIEW OF WRITTEN EXAMINATION

QUESTION 3.04 (2.00)

The plant is operating at 80% turbine load with all control systems in automatic. A reactor protection system surveillance test of Train A is about to commence.

a. Which reactor trip bypass breaker will have to be racked in and closed to prevent a reactor trip during the test? (0.50)

b. List TWO (2) annunciators provided in the control room that will alert the operator when the reactor trip bypass breaker is closed. (1.00)

c. How will the Reactor Protection System respond if a technician attempts to rack in and close both reactor trip bypass breakers concurrently? (0.50)

ANSWER 3.04 (2.00)

a. B (0.50)

b. REACTOR PROTECTION SYSTEM TRAIN A(B) TROUBLE
SAFETY SYSTEM TRAIN A(B) INOPERABLE (any 2 of first 3 @ 0.50)
REACTOR TRIP BYPASS BKR A(B) RACKED IN/CLOSED
General Warning alarm (accept for half credit)

c. Both reactor trip bypass breakers will trip. (0.50)

REFERENCE
BVPS OM-2.1.1, P. 3
BVPS OM-2.1.4, P. AAF1, AAI1, ABY1
BVPS 2LP-SQS-1.?, ELO 12,14,16

4.0	3.2	2.8	KA VALUE(S)
012000A307	012000K406	012000K408	(KA'S)

RO EXAM REVIEW (12-2-87)

3.04.c The answer given, both reactor trip bypass breakers will trip, is only a portion of the correct response. The correct answer to the question is that an automatic reactor trip will result (i.e., two general warnings occurring simultaneously in Train A and Train B). Referring to OM 2.01.1, p. 43, 44, a list of inputs to the "Protection System Train A(B) Trouble" (general warning alarm), includes "7. either bypass breaker closed." However, it further states that "if trouble in both trains should develop simultaneously, the reactor will be tripped automatically by the alarm system (general warning circuitry). Therefore, if an attempt to rack in and close both reactor trip bypass breakers concurrently is made, an automatic reactor trip will result. (Refer to attached copy of OM 2.01.1, p. 43, 44)

rear panel of the Central Board Display. This test checks the ability of the circuits within the Control Board Display to signal the applicable warnings to the operator.

While performing this test the non-urgent alarm will be eliminated but an urgent alarm will be created (CR 19 flashes on each central control card). Also the GW LED flashes, step 0 LED (RB) is lit on every display card, and ROD DEVIATION LED (CR20) is lit on each Centrl Control Card.

Rod Error Test

This test checks the ability of the circuit within the display to signal the applicable warnings to the operator. Rod error codes (all 1's) are manually presented at the output of the display I/O Card by means of the ETA and ETB pushbutton on the rear panel of the Control Board Display.

While performing this test a non-urgent alarm will be created and RPI rod-at-bottom on RPIZ or more codes-at-bottom conditions. Also the step 0 LED (RB) is lit on every display card, the GW LED flashes on every display card and the DATA A or B FAILURE LED (CR17) flashes on each central control card.

Simulated Data Transmission Tests for Normal and Error Codes may be done while the system is not in normal operation. In theses test the detector/encoder card output are inhibited. The normal position or error codes are applied to each I/O Data Cabinet I/O Card (A021) by means of the switches on the associated test/monitor card (A101).

Alarm System

Two annunciators are provided in the control room labeled:

PROTECTION SYSTEM TRAIN A TROUBLE
PROTECTION SYSTEM TRAIN B TROUBLE

These annunciators are not operated through the multiplexing scheme but are signaled direct from the alarm system in the trains. **The annunciators are operated by the following failures or operations:**

1. Loss of either of two 48 volt DC power supplies

2. Loss of either of two 15 volt DC power supplies

3. Any printed circuit card not properly inserted

4. Input Error Inhibit switch in the INHIBIT position

INSTRUMENTATION AND CONTROL

5. Logic A, Permissives, or Memor switch not in OFF position.

6. Slave relay tester Mode Selector switch in the TEST position

7. Either bypass breaker closed

8. Multiplexing inhibited

9. Blown ground return fuse.

10. Mode selector switch not in the OPERATE position on the output relay test panel.

Loss of one of four 120 volt AC vital instrumentation busses is monitored by the lighting of multiple status lamps on the affected channel. If trouble in both trains should develop simultaneously, the reactor will be tripped automatically by the alarm system. Circuits for the alarm system are located on part of the semi-automatic tester card. An alarm relay, shown in Figure 1-40 is energized when none of conditions (1) through (10) exist. When a problem occurs in a train, the alarm relay in that train de-energizes, opening a contact into the undervoltage coil driver. However, the alarm relay of the other train must de-energize for the circuit to open and trip the reactor. Another contact on the alarm relay activates a Control Room annunciator.

Overvoltage protection is supplied on the 48 volt and 15 volt DC power supplies. The device is a silicon controlled rectifier that clamps the output by providing an artificial load. One key will be provided for all locks on train A doors. A different key will be provided to operate all train B locks.

Power Distribution

Train A and train B both receive power from the four 120 volt AC vital instrumentation busses. The channel I through IV busses enter their respective input cabinet compartments through fuses in the compartments. In the input compartments, the busses are used to operate relays driven by external contacts. Two of the four busses are run through line noise filters at the rear of the input compartment into the DC power supplies in the logic cabinet as shown in Figure 1-41. In train A, busses I and II feed the power supplies and bus I feeds the slave relays. In train B, busses III and IV feed the power supplies and bus IV feeds the slave relays. Separate feeds are brought into the output cabinet for the slave relays to avoid running unfiltered lines through the logic cabinet.

The two 48 volt DC and 15 volt DC power supplies in one train are auctioneered to form one 48 and one 15 volt DC bus. A zero volt bus or circuit common bus is formed by connecting the (-)48 and

FACILITY REVIEW OF WRITTEN EXAMINATION

QUESTION 3.06 (2.00)

The plant is operating at 70% turbine load with the first stage pressure transmitter used for T-ref generation (PT446) failed low. Rod control is in Manual and the Steam Bypass System is in the Steam Pressure mode.

a. Describe the INITIAL response of the control rods (direction and rod speed) if the Rod Control System is placed in Automatic. (1.00)

b. Describe the INITIAL response of the steam dump valves if the Steam Bypass System is placed in the T-ave mode. (1.00)

ANSWER 3.06 (2.00)

a. The control rods will insert (0.50) at 72 spm. (0.50)

b. All steam dump valves (0.50) will fully open. (0.50) CHECK AT FACILITY!

REFERENCE
BVPS LER 08/16/87-017
BVPS 2LP-SQS-1.3, ELO 11
BVPS 2LP-SQS-21L1, ELO 18

3.1	3.0	KA VALUE(S)
001000A102	016000A201	(KA'S)

RO EXAM BANK (12-2-87)

3.06.b The answer that all steam dump valves will fully open is incorrect. The question states that the transmitter used for T-ref generation (PT-446) failed low. This will generate a Tref-Tave mismatch for the load rejection controller of the steam dump system; however, the steam dumps will remain closed due to the absence of an arming signal provided by first stage pressure transmitter PT-447. Therefore, the correct response is that the steam dump system will not respond when placed in the Tave mode. (Refer to attached copy of OM 2.21.1, p. 11)

Question 3.06.b.

MAJOR COMPONENTS

Design Data

[2MSS-TCV106A] (Typical for all)

Type	8" D100-160-3
Flow (nor/max), PPH	636,000/890,000
Pressure (Inlet/Outlet), PSIG	1085/-15
Temperature, F	556
Action, air-to-open	
Fail position (air/electricity)	Closed/Closed

The steam dump system has two automatic modes of operation, steam pressure mode and Tavg mode. The operational mode is operator selected by the Steam Dump Control Mode Selector Switch on the benchboard. In steam pressure mode, only the first two banks of valves are operational and they modulate to maintain the steam pressure setpoint set by the operator, using the benchboard mounted steam pressure controller. In Tavg mode, two steam dump controllers are available. The reactor trip controller operates the steam dump valves to restore no load Tavg following a reactor trip. Only the first two banks of valves are operational after a reactor trip. The load rejection controller operates all four banks of valves for large load rejections and the first two banks for small load rejections, to restore Tavg to program value.

All 18 valves trip closed if Tavg reaches Lo-Lo Tavg. If it is desired to cooldown the reactor plant, the Lo-Lo Tavg interlock may be manually defeated for the three cooldown valves only. Since the Lo-Lo Tavg interlock is dual train, two Steam Dump Control Interlock Selector Switches are provided, one for each train. The Steam Dump Control Interlock Selector Switches are also used for manually blocking the steam dump control system.

All 18 steam dump valves are blocked when the condenser is not available. To be available, the condenser must have sufficient vacuum and at least one cooling tower pump running.

Load rejection is sensed by turbine first stage pressure. First stage pressure transmiter [2MSS*PT447 sends a signal to bistables [PC447A] and [PC447B]. These bistables trip on rate of change of first stage pressure. [PC447A] trips on a rapid reduction in first stage pressure equivalent to a loss of load between 15% and 50%. [PC447B] trips on a 50% load rejection. The bistables are designed to latch ON since the rate of change signal will disappear as soon as first stage pressure reaches its new value. [PC447A] unblocks or arms the first and second bank of valves. [PC447B] unblocks or arms the third and fourth bank of valves. Both arming signals are negated if the condenser is unavailable or if Tavg reaches Lo-Lo setpoint. [PC447A and B] are reset by momentarily placing the Control Mode Selector switch to RESET. The switch spring returns to TAVG.

FACILITY REVIEW OF WRITTEN EXAMINATION

QUESTION 3.07 (2.00)

The plant is operating at 45% turbine load with all control systems in automatic. A feedwater flow transmitter (FT496) used for control room feedwater flow indication and steam flow/feed flow mismatch for steam generator 21C is out of service and its associated protection bistables are tripped.

A malfunctioning emergency trip header drain valve causes a turbine trip. The reactor trips several seconds later and the main generator output breakers trip even later. Assume no operator action.

a. Why didn't the reactor trip directly as a result of the turbine trip? (0.50)

b. What is the most probable cause for the reactor trip? (0.50)

c. How soon after the turbine trips will the main generator output breakers trip? (0.50)

d. What is the basis for the time delay associated with tripping the main generator output breakers? (0.50)

ANSWER 3.07 (2.00)

a. Reactor power was below P-9 (49%) (0.50)

b. SG 21C low level coincident with SF/FF mismatch (0.50)

c. 30 seconds (after turbine trip) (0.50)

d. Prevent turbine overspeed (0.50)

REFERENCE
BVPS LER 08/25/87-019
BVPS OM-35.1, P. 37
BVPS 2LP-SQS-1.2, ELO 21
(No ELO for parts c and d)

4.3	3.7	3.0	2.5	KA VALUE(S)
015000K405	015000K407	062000G007	062000K402	(KA'S)

FACILITY REVIEW OF WRITTEN EXAMINATION

RO EXAM REVIEW (12-2-87)

3.07.d The answer "prevent turbine overspeed", is only one of three possible correct responses. Other acceptable answers are:

- allow for extended RCP flow
- prevent missile generation inside containment from RCP flywheel destruction

(Refer to attached copy of W. Bird letter dated 2/29/80)

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Question 3.07.d.

WNTC/Zion
523-4041
2/29/80
Turbine Trip/Generator Trip Time Delay

All SNUPPS & Zion Instructor's

There apparently have been some misconceptions floating around regarding the reason(s) for a time delay for the generator trip subsequent to most turbine trips. The following is a summary of a bulletin from Mark Merrian - Reactor Protection Analysis, entitled "Functional Requirements For Continuity of Electrical Power to Reactor Coolant Pumps." These requirements will be included in lesson materials for all programs. It is each instructors responsibility to ensure that he understands and passes along these requirements to the students.

Bases: A reactor trip results in a turbine trip which would result in a generator trip immediately if no time delay were incorporated. There are two requirements which are safety related which require a generator trip time delay.

1. If the reactor trips due to overpower, overtemperature, or low pressure condition, an immediate turbine trip - generator trip coincident with failure of automatic bus transfer of electrical buses could result in a loss of RCP flow. This loss of flow could make the accident consequences more severe than that reported in the Safety Analysis Report. However, if pumping power is lost with a time delayed generator trip, the loss of flow is not considered serious because the reactor has been shutdown for some time.

2. On a major [illegible] cold leg), as coolant rushes out the break, the RCP impeller, shaft, flywheel etc. can be oversped. This RCP overspeed can be minimized by generator trip time delay by locking RCP's at ~ 60HZ frequency until auto bus transfer. RCP overspeed could result in flywheel destruction forming missiles which could damage the containment liner or ECCS components within containment.

The generator is effectively motorized for the normal gen. trip time delay following most turbine trips. This feature minimizes the consequences of Reactor trips from overpower, overtemperature, and low pressure and minimizes RCP overspeed following major LOCA's. From the turbine designers viewpoint, the generator trip time delay also prevents turbine overspeed as a result of steam within the turbine shell expanding to the condenser. This is a vital concern but is Secondary to the reactor safety considerations mentioned earlier.

It may be noted that some turbine trips result in immediate generator trip. The probability of these events coupled with failure of electrical buses to auto-bus transfer is considered very unlikely.

William Bird

William Bird
Senior Instructor
SNUPPS Phase III

WB/ki

FACILITY REVIEW OF WRITTEN EXAMINATION

QUESTION 3.12 (2.00)

a. List the FIVE (5) interlock signals/conditions required for a cold leg isolation valve to open when its control switch is taken to OPEN. Assume electrical power is available. Include setpoints. (1.00)

b. On which Motor Control Centers are the motor breakers for the Loop B Hot Leg Isolation Valve and Loop C Cold Leg Isolation Valve located? (1.00)

ANSWER 3.12 (2.00)

a. 1. Hot leg temperature within 5/20 degrees F of auctioneered temperature of operating loops. 90.20)

CHECK TEMP AT FACILITY.

2. Cold leg temperature within 5/20 degrees F of auctioneered temperature of operating loops. (0.20)

3. Isolation valve vent relief line flow >/= 200 gpm for 90 minutes. (0.20)

4. Isolation bypass valve open for 90 minutes. (0.20)

5. Hot leg isolation valve open for 90 minutes. (0.20)

b. Hot leg valve - MCC-2-19-1 (0.50)
Cold leg valve - MCC-2-18 (0.50)

REFERENCE
BVPS OM-6.3, PP. 6,7
BVPS 2LP-SQS-6.2, ELO 8

3.2	3.1	KA VALUE(S)
002000K409	062000A204	...(KA'S)

RO EXAM REVIEW (12-2-87)

3.12.a The correct temperature to be used in the answer is 5°F.

(Refer to attached copy of OM 2.6.1, p. 48)

Question 3.12.a.

INSTRUMENTATION AND CONTROL

The flow through the relief line is low so that the temperature and boron concentration are brought to equilbrium with the remainder of the system at a relatively slow rate.

[redacted]

a. [redacted] leg temperature within 5F of auctioneered temperature of operating [redacted]

b. Cold leg temperature within 5F of auctioneered temperature of operating loops.

c. Isolation valve vent relief line flow satisfactory (200 gpm) for 90 minutes (Trains A and B)

d. Isolation bypass valve open for 90 minutes (Trains A and B)

e. Hot leg isolation valve open for 90 minutes (Trains A and B)

Cold Leg Isolation Valve [2RCS*MOV591] may be closed provided both of the following conditions exist:

a. Control switch in CLOSE

b. No motor thermal overload

Isolation Bypass Valves [2RCS*MOV585, 586, 587]

Refer to Figure 6-34

The isolation bypass valves are controlled from Benchboard - Section B. Switch positions are CLOSE-OPEN with red (open) and green (closed) indicating lights.

The operation of Isolation Bypass Valve [2RCS*MOV585] is described below, and which is also typical for [2RCS*MOV586 and 587].

Isolation Bypass Valve [2RCS*MOV585] will open provided both of the following conditions are satisfied:

a. Control switch in OPEN

b. No motor thermal electrical protection trip

Isolation Bypass Valve [2RCS*MOV585] will close provided both the following conditions are satisfied:

a. Control switch in CLOSE

FACILITY REVIEW OF WRITTEN EXAMINATION

QUESTION 4.09 (2.25)

Refer to the attached figure, Axial Flux Difference Limits.

Given each of the following indications, state whether axial flux difference (AFD) is being maintained INSIDE or OUTSIDE the target band. BRIEFLY EXPLAIN your answer. (0.75 each)

	POWER LEVEL	AFD CHANNEL 1	AFD CHANNEL 2	AFD CHANNEL 3	AFD CHANNEL 4
a.	75%	-14	-12	-13	-17
b.	65%	-1	+2	-2	+1
c.	55%	-11	INOPERABLE	-15	-11

ANSWER 4.09 (2.25)

a. Inside the target band, (0.50) because less than 2 AFD channels are outside the target band. (0.25)

b. Outside the target band, (0.50) because 2 AFD channels are outside the target band. (0.25)

c. Inside the target band, (0.50) because less than 2 AFD channels are outside the target band. (0.25)

REFERENCE
BVPS TECHNICAL SPECIFICATIONS, 3/4.2.1
NO ELOs PROVIDED FOR TECH. SPECS.

3.7	3.4	3.5	KA VALUE(S)
001000G005	001000G011	014000A104	(KA'S)

RO EXAM REVIEW (12-2-87)

4.09.a.c The following should also be considered a correct response for full credit:

- Inside the target band, because none of the AFD values are outside the target band.

FACILITY REVIEW OF WRITTEN EXAMINATION

QUESTION 4.11 (2.75)

a. What is the normal limit for annual nonemergency Whole Body radiation exposure to an NCO (on shift) in accordance with the BVPS Radiation Control Manual? (0.50)

b. List TWO (2) individuals (by title) who are authorized to INCREASE the above (part a.) annual nonemergency Whole Body radiation exposure limit. (1.00)

c. A 20-year old radiation worker (BVPS employee) has received 1 rem during the current quarter and has a lifetime accumulated whole body dose of 8.6 Rem (including current quarter exposure). Assuming this worker has a Form NRC-4 on file, calculate how long (in hours) he could remain in a 250 mrem/hour gamma radiation field without exceeding any nonemergency Whole Body exposure limits of 10 CFR 20 or BVPS Radiation Control Manual. SHOW ALL CALCULATIONS AND ASSUMPTIONS. (1.25)

ANSWER 4.11 (2.75)

a. 5 rem (0.50)

b. 1. Senior VP, Nuclear Group

2. VP, Nuclear Group (any 2 @ 0.50 each)

3. Senior Manager, Nuclear Operations

c. LIMITED BY 5(N-18)=10 rem lifetime whole body limit

(0.50 for correct limit)

10 rem - 8.6 rem = 1400 mrem (0.25 for correct method)

$\frac{1400 \text{ mrem}}{250 \text{ mrem/hr}}$ = 5.6 hours (0.50 for correct answer)

REFERENCE
BVPS RADIATION CONTROL MANUAL, PP. 6, 7
NO ELOs PROVIDED FOR RAD CON
2.8 KA VALUE(S)
194001K103 ...(KA'S)

FACILITY REVIEW OF WRITTEN EXAMINATION

RO EXAM REVIEW (12-2-87)

4.11.b The question should be deleted. It is not required knowledge of an R.O. to be able to state those individuals who may authorize an increase above an annual Whole Body radiation exposure limit. This is both an administrative management concern and Radiation Control Department concern only. However, it must also be noted that the answer key had omitted an additional correct response to the question. According to EPP/IP 5.3.D.2.3, the Emergency Director may also authorize "exposures in excess of normal guide/limits" (i.e., non-emergency whole body limit).

(Refer to attached copy of EPP/IP 5.3, p. 2; RCM, Ch. 1, p. 7)

Question 4.11.b.

2.0 Precautions

2.1 The provisions of this procedure are applicable only in actual emergency conditions, and are applicable to BVPS personnel performing assigned emergency functions and emergency volunteers (eg.: fireman) if applicable.

2.2 The exposure of personnel during emergency operations shall be maintained as low as reasonably achievable, and should be maintained less than the administrative guides established in the BVPS Radiation Control Manual (RCM), and/or less than the Federal radiation exposure standards established in 10 CFR 20.

Administrative means used during normal operations to minimize personnel exposure (such as radiation work permits, radiation clearances, and ALARA measures) should remain in force to the extent consistent with timely implementation of emergency measures.

If necessary operations require personnel exposure in excess of the normal methods, or if normal access control and radiological work practices will result in unacceptable delays, the Emergency Director may, at his discretion, waive or modify the established exposure control criteria and methods in accordance with the provisions of this EPP/IP. In making such decisions, the Emergency Director should call upon the expertise of the radiation control staff onsite.

2.3 The Emergency Director has the authority to perform appropriate protective and corrective measures necessary to mitigate the consequences of an Accident and to place the facility in a safe condition. The Emergency Director may approve personnel exposures in excess of normal guides/limits, but less than the planned radiation exposure criteria established in this EPP/IP, provided the pre-conditions of such exposure are met.

2.4 Personnel shall not enter any area where dose rates are unknown or unmeasureable with instruments and dosimetry immediately available.

2.5 Appropriate dosimetry equipment, which is capable of measuring the anticipated maximum exposure and type of radiations, shall be worn.

2.6 Extremity dosimeters shall be worn if anticipated exposure is greater than about five (5) times that of the whole body.

CHAPTER 1 - STANDARDS AND REQUIREMENTS

Part of Body	Column 1 - Assumed exposure in rems for calendar quarters prior to Jan. 1, 1961	Column 2 - Assumed exposure in rems for calendar quarters beginning on or after Jan. 1, 1961
Whole body, gonads, active blood-forming organs, head and trunk, lens of eye	3 3/4	1 1/4

7) If calculation of the individual's accumulated occupational dose for all periods prior to January 1, 1961 yields a result higher than the applicable accumulated dose value for the individual as of that date, 3 rems per quarter, the excess may be disregarded.

Part of Body	Cumulative Dose (rems)	Quarterly Dose (rems)	Annual Dose (rems)
Whole Body	5 (N-18)	3	*12
Skin of Whole Body	----	7.5	30
Hands and forearms; feet and ankles	----	18.75	75

*The cumulative, whole body occupational exposure of any permanent Duquesne Light employee shall not exceed 5 rem in any calendar year. This exposure includes occupational exposure from sources not under control of Duquesne Light. On a special case basis, the Senior Vice President, Nuclear Group, Vice President, Nuclear, or Senior Manager, Nuclear Operations may authorize exposure extensions above this limit, up to, but not to exceed, applicable 10CFR20 limits, for nonemergency exposure.

The cumulative, whole body occupational exposure of any contractor, vendor, or visitor personnel shall not exceed 5 rem in any calendar year. This exposure includes occupational exposures received at all other facilities. On a special case basis, the Senior Vice President, Nuclear Group, or Vice President, Nuclear, may authorize exposure extensions above this guide up to, but not to exceed, applicable 10CFR20 limits, for nonemergency exposure. Also, the affected individual's employer must supply, in writing, authorization for the exposure. (Refer to SAP 23 for guidance and the required forms.)

Individual exposure guidelines are upper guidance levels and do not authorize unwarranted or unnecessary dose.

DATE__________ TIME__________

Plant must be at steady state for 30 minutes. To verify this trend all computer points being used at 5 minute intervals for 30 minutes and verify that they remain within a 2% band respectively.

Power Level (NIS Cabinet) N-41________% N-42________% N-43________% N-44________% U1150________%

1. Record the individual blowdown flow readings: 2BDG-FIT100A______%, 2BDG-FIT100B______%, 2BDG-FIT100C______%
Multiply each % flow (in decimal format, ie. 22% = 0.22) times 140 to obtain the GPM for each line:
A______GPM B______GPM C______GPM

Add all three flows to obtain total blowdown flow: ______GPM. If less than 100 GPM, assume blowdown to be zero by placing a zero in Steps 5, 9, and 11.

		LOOP A	LOOP B	LOOP C
2.	Feedwater Temperature (F)*	2FWS-TI154A or T0418A	2FWS-TI154B or T0438A	2FWS-TI154C or T0458A
3.	Steam Pressure (PSIA)**	2MSS-PI474 or U0414 (________ + 14.7 = ________ PSIA)	2MSS-PI484 or U0434 (________ + 14.7 = ________ PSIA)	2MSS-PI494 or U0454 (________ + 14.7 = ________ PSIA)
4.	M_{FW} ($x10^6$ PPH)	2FWS-FR478 or U0410	2FWS-FR488 or U0430	2FWS-FR498 or U0450
5.	M_{BD} ($x10^6$ PPH)	2BDG-FIT100A___% = 0.___ 0.___ x 0.07 = ___ x 10^6PPH	2BDG-FIT100B___% = 0.___ 0.___ x 0.07 = ___ x 10^6PPH	2BDG-FIT100C___% = 0.___ 0.___ x 0.07 = ___ x 10^6PPH
6.	$M_{ST} = M_{FW} - M_{BD}$ ($x10^6$PPH)			
7.	$h_{FW}\left(\frac{BTU}{lb}\right)$			
8.	$h_{ST}\left(\frac{BTU}{lb}\right)$			
9.	$h_{BD}\left(\frac{BTU}{lb}\right)$			
10.	$M_{ST}\,h_{ST}\left(x10^6\,\frac{BTU}{hr}\right)$			
11.	$M_{BD}\,h_{BD}\left(x10^6\,\frac{BTU}{hr}\right)$			
12.	$M_{FW}\,h_{FW}\left(x10^6\,\frac{BTU}{hr}\right)$			
13.	Loop A Power $\left(x10^6\,\frac{BTU}{hr}\right)$	(Step 10 + Step 11 - Step 12 = __________)	X	X
14.	Loop B Power $\left(x10^6\,\frac{BTU}{hr}\right)$	X	(Step 10 + Step 11 - Step 12 = __________)	X
15.	Loop C Power $\left(x10^6\,\frac{BTU}{hr}\right)$	X	X	(Step 10 + Step 11 - Step 12 = __________)
16.	RCS Output $\left(x10^6\,\frac{BTU}{hr}\right)$ = Sum of Steps 13, 14, and 15 = __________			
17.	Net Reactor Power = [RCS Output (Step 16) - RCP Output] = [________ - 27.3] x 10^6 $\frac{BTU}{hr}$ = ________ x 10^6 $\frac{BTU}{hr}$			
18.	MW_{th} = Net Reactor Power (Step 17) x $\frac{10^{-6}\,MWHR}{3.413\,BTU}$ = __________ MW_{TH}			
19.	% Reactor Power = $\frac{MW_{th}\ (Step\ 18)}{2652}$ x 100% = __________%			
20.	If Reactor Power is > 2652 MW_{TH} immediately reduce power to < 2652 MW_{TH}			

GAIN ADJUST

NOTE: Match Tavg & Tref prior to adjusting NI Detector currents on NI's.

Adjust GAIN if:

1. NI meter readings exceeds 100% or;
2. If meter readings differ by ± 1% from calculated power, when calculated power is < 99% or;
3. If meter readings differ by $^{+1\%}_{-0\%}$ from calculated power, when calculated power is ≥ 99%.

If any of the above conditions exist adjust GAIN on the front of the power range drawer B until the indicator reads the same as the calculated value. Record final readings if adjustment is required.

CHANNEL	N-41	N-42	N-43	N-44
METER	%	%	%	%
DETECTOR A CURRENT	ua	ua	ua	ua
DETECTOR B CURRENT	ua	ua	ua	ua

Calculations performed by:

__________ __________
DATE/TIME

Calculation reviewed and approved by:

__________ __________
DATE/TIME

Operating Supervisor Review:

__________ __________
DATE/TIME

DAILY HEAT BALANCE

eat Balance Calculation Instructions

1. Record the individual loop blowdown flow percentages. Calculate the GPM's for each loop using the decimal equivalent of the percentage. Calculate the total blowdown flow. If total blowdown flow is less than 100 GPM, assume blowdown to be zero by using zero in steps 5, 9, and 11.

2. Obtain feedwater temperature (F) from the appropriate instrument.

3. Obtain steam pressure from the appropriate instrument and convert to absolute pressure.

4. Obtain the mass flow rate of feedwater ($\dot{M}_{FW}$) from the appropriate instrument.

5. Obtain the flow rate of blowdown (unless total blowdown < 100 GPM) from the appropriate instrument and convert to 10^6 PPH ($\dot{M}_{BD}$).

6. Calculate mass flow rate of steam ($\dot{M}_{ST}$): $\dot{M}_{ST} = \dot{M}_{FW} - \dot{M}_{BD}$.

7. Obtain feedwater enthalpy (h_{FW}) from saturated steam table using hf at feedwater temperature.

8. Obtain steam enthalpy (h_{ST}) from saturated steam table using hg at steam pressure.

9. Obtain blowdown enthalpy (h_{BD}), unless total blowdown < 100 GPM, from saturated steam table using hf at steam pressure.

10. Multiply $\dot{M}_{ST}$ and h_{ST} for applicable loop.

11. Multiply $\dot{M}_{BD}$ and h_{BD} for applicable loop (unless total blowdown < 100 GPM).

12. Multiply $\dot{M}_{FW}$ and H_{FW} for applicable loop.

13. Calculate Loop A Power: Step 10 + Step 11 - Step 12.

14. Calculate Loop B Power: Step 10 + Step 11 - Step 12.

15. Calculate Loop C Power: Step 10 + Step 11 - Step 12.

16. Calculate RCS output.

17. Calculate Net Reactor Power.

18. Calculate MW_{th}.

19. Calculate % Reactor Power.

FACILITY REVIEW OF WRITTEN EXAMINATION

QUESTION 5.01 (2.50)

a. Calculate reactor power (MWt) using the following information. (1.50)

feedwater temperature	= 420 F.
feedwater flow	= 3.60 E+6 lbm/hr (per S/G)
S/G pressure	= 785 psig
condenser pressure	= 2.5 psia
drain pump flow	= 3.0 E+6 lbm/hr
1st stage pressure	= 735 psig

b. What are TWO (2) components which, if not accounted for, cause the calculated value for reactor power to be greater than actual reactor power? (1.00)

ANSWER 5.01 (2.50)

a. h(feedwater) = 396.9 BTU/lbm (0.50)
h(steam) = 1200 BTU/lbm (0.50)
reactor power = (10.8 E+6 lbm/hr)*(1200 - 396.9 BTU/lbm) *(0.293 W/BTU/hr)/(1.0 E+6 W/MW) = 2541 MW (0.50)

b. RCPs (0.50)
PZR heaters (0.50)

REFERENCE
BVPS LP-TMO-6 Enabling Objective 11
BVPS Thermo Text Chapter 6, p. 37; Question 48
BVPS LP-TMO-6 page 18
K/A 193007 K1.08 3.4
193007K108 ...(KA's)

SRO EXAM REVIEW (12-2-87)

5.01.b Another acceptable answer should be blowdown flow. The attached daily Heat Balance shows m_{steam} calculated by subtracting $m_{blowdown}$ from $m_{feedwater}$.

FACILITY REVIEW OF WRITTEN EXAMINATION

QUESTION 5.04 (3.75)

Complete Parts B, D, and E of the attached Estimated Critical Position Calculation (Attachment 1) given the following information:

- A reactor trip occurs at 2030 on 11/28/1987 from 100% power after the plant had been operating for 30 days with Bank D rods at 190 steps.
- Boron concentration was 930 ppm before the trip and has been increased to 1190 ppm and is now steady at this value.
- The estimated time for criticality is 1000 on 11/30/1987.

ANSWER 5.04 (3.75)

B.1.I	-1400 pcm	
B.2.I	-9068 pcm	
B.2.II	-11231 pcm	
B.2.III	+2163 pcm	
B.3.I	-3000 pcm	
B.3.II	-1500 pcm	
B.3.III	-1500 pcm	
B.4.I	-610 pcm	[15 x 0.25]
B.4.II	-760 pcm	
B.4.III	+150 pcm	
D.II	-80 pcm	
D.III	-667 pcm	
D.IV	Bank D Step 98	
E.IV	Bank D Step 180	
E.V	Bank D Step 50	

REFERENCE
BVPS LP-RT-9 Enabling Objective 6
BVPS - OM 1.50.4
K/A 192008 K1.07 3.6
192008K107 ...(KA's)

FACILITY REVIEW OF WRITTEN EXAMINATION

SRO EXAM REVIEW (12-2-87)

5.04 Make allowances for differences in graph interpretation.

Answer	B 2II	should be	-11186 pcm
Answer	D II	should be	-140 pcm
Answer	D III	should be	-754 pcm
Answer	D IV	should be	88 steps on D
Answer	E IV	should be	160 steps on D
Answer	E V	should be	42 steps on D

FACILITY REVIEW OF WRITTEN EXAMINATION

QUESTION 5.05 (2.00)

Using the attached Shutdown Margin Calculation (Attachment 2), complete the form and calculate SHUTDOWN MARGIN in percent delta K/K. Assume boron concentration is 900 ppm and the reactor has been at 100% power for 5 days.

ANSWER 5.05 (2.00)

1.b. +1225 pcm (0.40)
rod worths: SDA 2589
SDB 1287 (0.40) for right group of numbers
CBA 1167
CBB 1960 (0.40) for conversion from % delta rho to pcm
CBC 1324
CBD 1300 (0.40) for Bank D at 190 steps
shutdown margin = -4.369% delta k/k (0.40)

REFERENCE
BVPS LP-RT-9 Enabling Objective 4
BVPS - OM 2.55A.4
K/A 192002 K1.13 3.7
192002K113 ...(KA's)

SRO EXAM REVIEW (12-2-87)

5.05 Shutdown Margin Calculation should allow for a different assumed Rod Height (not given in question). BVPS runs with all rods out, not with Bank D at 190 steps as per answer key. Therefore, calculations done with all rods out should be considered acceptable.

FACILITY REVIEW OF WRITTEN EXAMINATION

QUESTION 5.09 (2.25)

HOW (Increase, Decrease, No Change) and WHY would each of the following parameters affect the margin to DNB?

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

ANSWER 5.09 (2.25)

a. Increases (0.25) as PRZR temperature rises, so does saturation pressure (0.50)
b. Increases (0.25) because core delta T will decrease (to keep power constant) (0.25) reducing T(hot) (0.25)
c. Decreases (0.25) because more power is being produced in the top half of the core (0.25) causing the (hot channel factor) bounds in this area to be approached (0.25)
   DNB   or thermal limits

EJ

REFERENCE
BVPS LP-TMO-7 Enabling Objective 12
BVPS Thermodynamic Text Chapter 7 page 17
K/A 193008 K1.05 3.6
193008K105 ...(KA's)

SRO EXAM REVIEW (12-2-87)

5.09.c Delete reference to hot channel factors. This is not needed to answer the question.

FACILITY REVIEW OF WRITTEN EXAMINATION

QUESTION 5.10 (2.05)

a. WHICH initial temperature, 300 F or 547 F, will give the largest change in the magnitude of MTC as boron concentration is lowered from 1000 to 500 ppm? Justify your answer. (1.35)

b. WHY does power defect become more negative as the core ages? (0.70)

ANSWER 5.10 (2.05)

a. 547 (0.35) because the change in density/degree F is much greater than at lower temperatures (0.50) so more boron atoms enter or leave the core causing a larger reactivity change for a given temperature change (0.50).

b. MTC, a component of the power defect, becomes more negative ~~(0.35) due to decreased boron concentration (0.35).~~ (0.70)

REFERENCE
BVPS LP-RT-6 Enabling Objectives 1, 2, 10
BVPS Reactor Theory Text Chapter 6, pp. 6,13,16
K/A 192004 K1.06 3.1

SRO EXAM REVIEW (12-2-87)

5.10.b Delete second part of answer because it is not asked for in the question (i.e. due to decreased boron concentration).

FACILITY REVIEW OF WRITTEN EXAMINATION

QUESTION 6.01a (2.50)

a. List THREE (3) conditions/signals that will cause a steam generator feedwater pump to automatically trip. DO NOT include manual or motor/bus electrical trips). (0.75)

b. The reactor operator tries to start a main feed pump by placing the control switch to the START position and immediately releasing it. The pump does not start. Briefly explain WHY the pump did not start and how the operator should have manipulated the control switch. (0.75)

c. WHAT is the maximum flowrate for a single feedwater pump operating alone? WHY does it have this limit? (1.00)

ANSWER 6.01a (2.50)

a. - feedwater isolation signal OR SI or Hi-Hi S/G level EY
   - sustained low suction pressure (0.25 x 3)
   - lube oil pressure extreme low

b. Switch must be held in the START position (0.35) until the feedwater recirculating water valve is fully open (0.40)

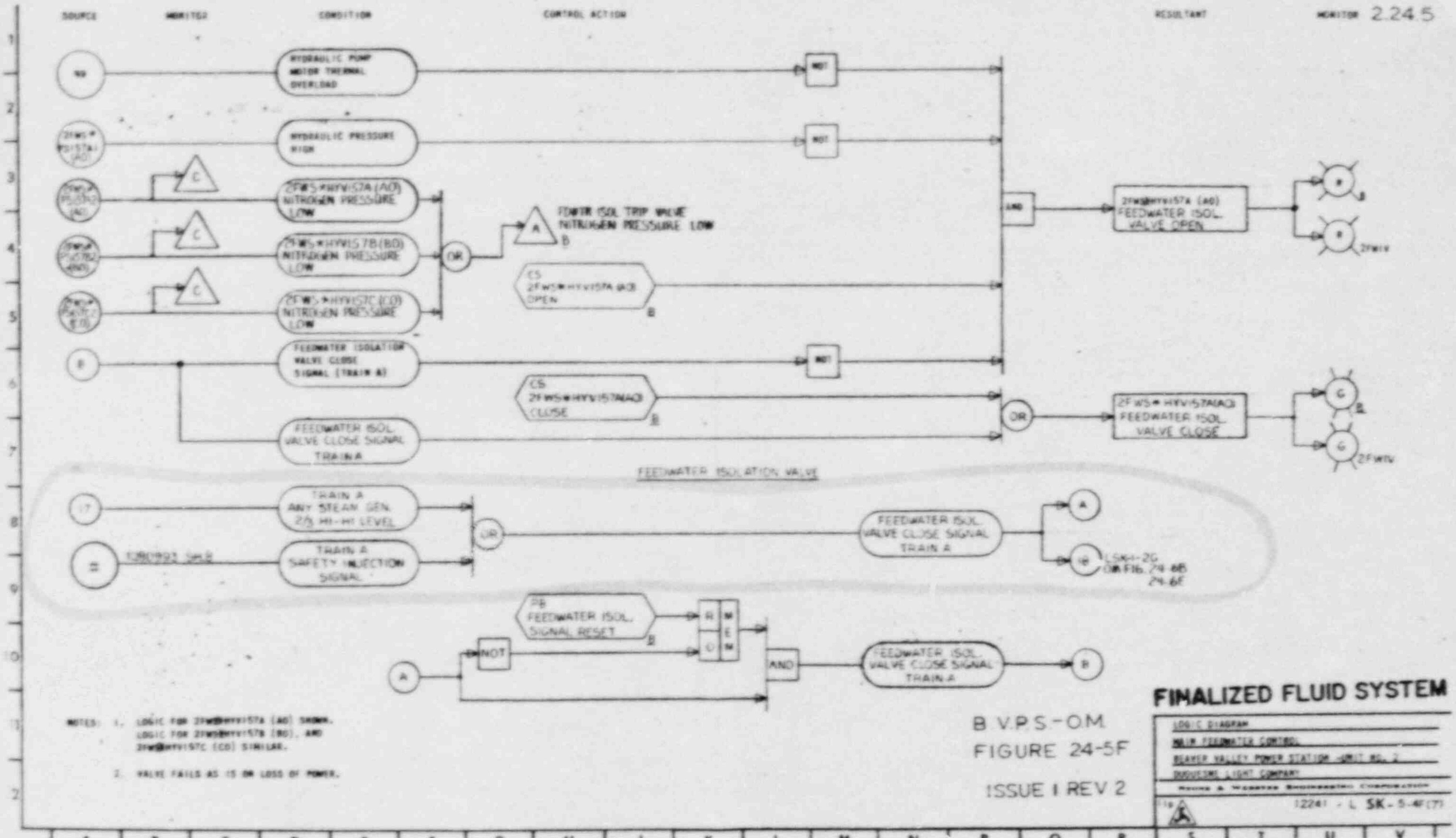
c. 16,000 gpm (8.0 mpph) (0.50)
   prevent runout conditions (0.50)

REFERENCE
Beaver Valley 2LP-SQS-24.1 Enabling Objectives 7, 16
BVPS - OM 2.24.1, pp. 20,21; 2.24.2, pp. 2,3
K/A 059000 K4.16 3.2
K/A 059000 G0.09 3.1
K/A 059000 G0.10 2.9
059000K416 059000G010 059000G009 ...(KA's)

SRO EXAM REVIEW (12-2-87)

6.01a Accept answer that states a feedwater isolation comes from a safety injection or a hi-hi steam generator level. See attached logic diagram.

QUESTION G.01A



FACILITY REVIEW OF WRITTEN EXAMINATION

QUESTION 6.02 (2.50)

a. During refueling operations radiation monitor 2HVR*RQ104A, Local-Containment Purge, alarms high. Before the alarm, WHERE is this effluent being released, and WHAT automatic actions occur to prevent this release from continuing? (1.00)

b. State the automatic actions which occur, if any, when 2ARC-RQ100, Local-Air Ejector Discharge, alarms high. (0.50)

c. WHY are there two (2) sample flow paths for the Wide Range Gas Monitor, 2HVS*RQI109B, and WHAT type of effluent does it monitor? (1.00)

ANSWER 6.02 (2.50)

a. auxiliary building vent (elevation 773) (0.50)
containment purge exhaust isolation dampers close (0.25)
containment purge air supply dampers close (0.25)
b. none (0.50)
c. Samples different radiation levels at different flow rates (0.6)
Monitors noble gases (0.40)

REFERENCE
Beaver Valley 2LP-SQS-43.1 Enabling Objectives 2.p., 4., 12.
BVPS - OM 2.43.4, pp. ACN, ACX; 2.43.5, Table 43-1, p. 3
2LP-SQS-43.1, p. 27 of 48
K/A 072000 K4.01 3.6
K/A 072000 A3.01 3.1
072000K401 072000A301 ...(KA's)

SRO EXAM REVIEW (12-2-87)

6.02.a During refueling the containment purge path must, by Tech. Specs., be discharging through the Main Filter Banks and out the Elevated Release on Containment, not out the auxiliary building vent as the answer key indicates.

QUESTION 6.02

REFUELING OPERATIONS

CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock is closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. Closed by an isolation valve, blind flange, or manual valve, or

2. Exhausting at less than or equal to 7500 cfm through OPERABLE Containment Purge and Exhaust Isolation Valves with isolation times as specified in Table 3.6-1 to OPERABLE HEPA filters and charcoal adsorbers of the Supplemental Leak Collection and Release System (SLCRS).

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

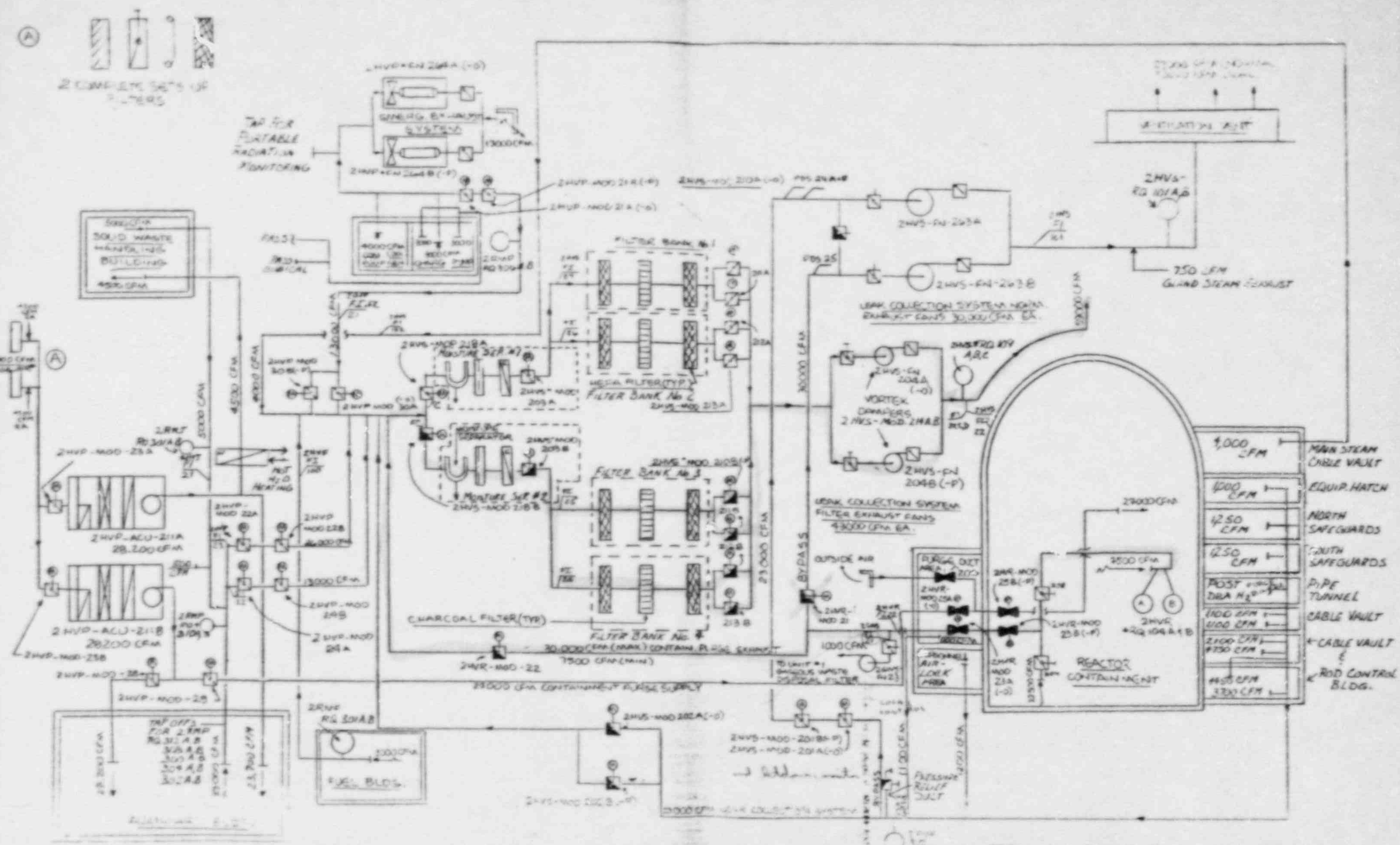
4.9.4.1 Each of the above required containment penetrations shall be determined to be in its above required condition within 150 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment.

4.9.4.2 The containment purge and exhaust system shall be demonstrated OPERABLE by:

a. Verifying the flow rate to the SLCRS at least once per 24 hours when the system is in operation.

b. Testing the Containment Purge and Exhaust Isolation Valves per the applicable portions of Specification 4.6.3.1.2, and

c. Testing the SLCRS per Specification 4.7.8.1 with the exception of item 4.7.8.1.c.2.



TP-2LP-SQS-16,44C,44D

FACILITY REVIEW OF WRITTEN EXAMINATION

QUESTION 6.03 (1.75)

a. WHAT TWO (2) chemicals are added to the RCS to control pH and oxygen when in MODE 5? Indicate which chemicals for which function. (0.50)

b. During which type of dilution operation (alternate dilute or dilute) will reactor coolant hydrogen concentration become depleted faster? Justify your answer. (0.75)

c. WHY must the VCT be maintained at a minimum pressure of 15 psig? (0.50)

ANSWER 6.03 (1.75)

a. Li7OH - pH (0.25); hydrazine - O2 (0.25)

b. Alternate dilute (0.25) because this flow path partially bypasses the VCT (0.25) preventing the hydrogen cover gas from being absorbed by the dilution flow (0.25)

c. Required backpressure for the reactor coolant pump seals (0

REFERENCE
Beaver Valley 2LP-SQS-7.1 Enabling Objectives 1, 10, 17
BVPS - OM 2.7.1, pp. 3, 4, 37; 2.7.2, pp. 1, 5
K/A 004000 K4.01 3.3
K/A 004000 K4.02 2.6
K/A 004000 K4.08 3.2
K/A 004000 G0.10 3.4
004000K408 004000K402 004000K401 004000G010 004000G009
...(KA's)

SRO EXAM REVIEW (12-2-87)

6.03.a Answer should be given full credit for LiOH vice Li7OH. Training material for BVPS uses LiOH.

Question 6.03A

1) containment depressurization system (A-4)

2) chemical and volume control system

III. Reactor plant system sampling requirements

A. Sampling frequencies are listed in BVPS chemistry manual, Chapter 3 "Sampling and Testing".

1. Sampling requirements are listed by system in body of chapter as well as appendices.

a. Give students handout for RCS (A-5 thru A-9)

2. Sampling requirements come from Westinghouse recommendations, surveillance requirements of Tech. Specs., and action statements of Tech. Specs. Note: Tech. Specs will be covered in LP-CHEM-19.

IV. Analysis of coolant chemistry

A. Predictable changes over fuel cycle.

1. Start of cycle.

a. Boron concentration ~1000ppm.

b. Li^+ concentration as necessary (> mid range).

2. During cycle

a. Boron "burned out" (converted to Li^+) or else diluted from system.

b. Boron concentration can be increased or decreased to load follow.

c. Li^+ concentration will increase as boron is "burned out". At the start of cycle when boron concentrations are high (and hence Li^+ production is high) Li^+ might have to be removed by cation exchanger.

1. $B^{10} + {}_0n^1 \rightarrow {}_3Li^7 + {}_2He^4$

2. ${}_3Li^7 + H_2O \rightarrow H + LiOH \rightleftarrows Li^+ + OH^-$

3. End of cycle

a. Boron <100ppm

b. Li^+ as maintained

FACILITY REVIEW OF WRITTEN EXAMINATION

QUESTION 6.04 (2.40)

a. List FOUR (4) locations where SR indication can be found. (1.40)

b. A reactor shutdown is in progress with the source range (SR) detectors reading about 10,000 cps and both intermediate range (IR) detectors reading 1 x 10E-11 amps. Ten minutes later the SR detectors read about 1,000 cps but the IR detectors still read 1 x 10E-11 amps. WHY does the IR detector output remain the same? (0.50)

c. The plant is operating at 100% power with NI-44 out-of-service. If an automatic reactor trip occurs and NI-43 is failed as is, WHAT effect, if any, will this have on the nuclear instrumentation system's ability to monitor neutron flux? (0.50)

ANSWER 6.04 (2.40)

a. NIS rack
benchboard - section B
emergency shutdown panel (SDP) (0.35 x 4)
alternate shutdown panel (ASP)

b. 1 x 10E-11 amp signal is used as a reference for gamma compensation (0.50)

c. Source range detectors cannot be energized (0.50)

REFERENCE
Beaver Valley 2LP-SQS-2.2 Enabling Objectives 4.g., 8.a., 16
BVPS - OM 2.2.1, pp. 11,16,27; 2.2, p. 1; 2.01.1, p. 11
K/A 015000 K5.01 3.2
K/A 015000 G0.06 2.8
K/A 015000 G0.07 4.0
K/A 015000 A3.03 3.9
015000K501 015000G006 015000A403 ...(K/A's)

SRO EXAM REVIEW (12-2-87)

6.04.a There are two additional answers not given in key.

- Remote recorder 2NME-NR45
- PSMS computer (see OM 2, Section 1, p. 16 and Section 2, pp. 4, 5)

QUESTION 604A

MAJOR COMPONENTS

SR Remote Count Rate Meter [2NMS-NI31B, 32B, 31BA, 32BA]

The SR remote meter indication is an analog signal proportional to the count-rate being received, and is obtained from the 0-1 milliampere isolation amplifier output.

The meters are mounted on the Benchboard - Section B (2NMS-NI31B, 32B)and the Emergency Shutdown Panel (2NMS-NI31BA, 32BA) and calibrated logarithmically from 1E+0 to 2E+6 counts per second. These meters give the same indication at the Benchboard and Emergency Shutdown Panel as is displayed by the local meter on the corresponding source-range drawer.

Remote Recorder (2NME-NR45)

This two-pen recorder located on the Vertical Board - Section B is capable of continuously recording any two NIS channels at a time. Each pen receives its signal through a multiposition switch (1N45 and 2N45) located on the Benchboard - Section B which can select any one of the eight nuclear channels. In the case of the source ranges, a 0-50 millivolt DC signal, proportional to the count-rate range of 1E+0 to 1E+6 counts per second, is supplied for recording during source range operation.

SR Startup Rate Circuitry (N37)

The startup-rate (SUR) drawer receives four input signals (0-10 VDC), one from each of the source and intermediate range channels. Four rate amplifier modules condition these signals and transmit their rate signals to its respective SUR meters on Benchboard - Section B and the Emergency Shutdown Panel. The indicators for source range channels NI-31 and NI-32 are [2NMS-NI31D, 32D] and [2NMS-NI31DA, 32DA] respectively. A test module is provided which may be used to inject a test signal into any one of the rate circuits. The test signal can be monitored on a test meter mounted on the front panel of the SUR drawer. Two power supplies are installed in such a manner as to ensure rate indication from at least one source and one intermediate range channel.

Intermediate Range Channels

Intermediate range (IR) output information is tabulated in Specific Instrumentation and Control. Each IR channel receives a DC signal from a compensated ion chamber and supplies a signal positive high VDC and a negative compensating voltage to its respective detector. The compensating voltage cancels that signal imposed on the detector resulting from gamma radiation. Both voltage supplies are adjustable through controls located inside the IR channel drawer. The detector signal is received by the intermediate range logarithmic amplifier. This modular unit, comprised of several operational amplifiers and

Question 6.04 A

SET POINTS

DELTA FLUX ALARM	60 penalty minutes in last 24 hours
NIS CHANNEL IN TEST	N/A
NIS INT RNG NEUTRON FLUX HIGH REACTOR TRIP	Current Equiv to 25% Power
NIS 2/4 PWR RNG LOW SETPOINT NEUTRON FLUX HIGH RX TRIP	25%
NIS 2/4 PWR RNG HIGH SETPOINT NEUTRON FLUX HIGH RX TRIP	109%
NIS SOURCE RNG NEUTRON FLUX HIGH REACTOR TRIP	10^5 counts/sec
NIS 2/4 PWR RNG NEUTRON FLUX RATE HIGH REACTOR TRIP	109%
P-10 PERMISSIVE	10% of full power
NOT P-7	<10% of full power
NIS SOURCE RANGE TRIP BLOCKED	N/A
NIS INTERMEDIATE RANGE TRIP BLOCKED	N/A
POWER RANGE LOW SETPOINT TRIP BLOCKED	N/A
NOT P-8	N/A
P-6 PERMISSIVE	N/A
NOT P-9	N/A

Computer Digital

DELTA FLUX OUTSIDE TARGET BAND	power above 90% and delta flux outside target band
DELTA FLUX IN ALARM	60 penalty minutes in last 24 hours
PWR RNG CHN 1 P9 PERM	≤49% of full power

Question 604A

SET POINTS

PWR RNG CHN 2 P9 PERM	≤49% of full power
PWR RNG CHN 3 P9 PERM	≤49% of full power
PWR RNG CHN 4 P9 PERM	≤49% of full power
PWR RNG CHN 1 P8 PERM	30% of full power
PWR RNG CHN 2 P8 PERM	30% of full power
PWR RNG CHN 3 P8 PERM	30% of full power
PWR RNG CHN 4 P8 PERM	30% of full power
INT RNG 1 HIGH FLUX	Current equiv to 25% power
INT RNG 2 HIGH FLUX	Current equiv to 25% power
INT RNG CHN 1 P6 PERM	1E-10 Amperes
INT RNG CHN 2 P6 PERM	1E-10 Amperes
SOURCE RNG CHN 1 FLUX	10^5 CPS
SOURCE RNG CHN 2 FLUX	10^5 CPS

Computer Analog

SOURCE RNG DET. 1 FLUX LOG	N/A
SOURCE RNG DET. 2 FLUX LOG	N/A
INT RNG DET. 1 FLUX LOG	N/A
INT RNG DET. 2 FLUX LOG	N/A
PWR RNG CHN 1 FLUX	109%
PWR RNG CHN 2 FLUX	109%
PWR RNG CHN 3 FLUX	109%
PWR RNG CHN 4 FLUX	109%

FACILITY REVIEW OF WRITTEN EXAMINATION

QUESTION 6.06 (3.00)

a. WHAT PCCW realignment occurs on a low-low PCCW surge tank level? (0.50)

b. Component cooling water pump 2CCP*P21C is racked in on bus 2AE and is in auto. WHAT THREE (3) conditions need to be satisfied before the pump will automatically start? Assume all associated equipment functions properly. (1.00)

c. WHAT are SIX primary component cooling water (PCCW) containment loads that are isolated by a containment isolation phase B signal? (1.50)

ANSWER 6.06 (3.00)

a. 2CCP*MOV175-1, 176-1, 177-1, 178-1 (non-nuclear safety portion of PCCW) PCCW system supply and return isolation valves close (0.50)

b.
- no containment isolation phase B signals present
- PCCW system header pressure low (3 x 0.33)
- diesel loading sequence signal present

c.
- reactor coolant pumps
- CRDM shroud cooling coils
- shield tank cooler
- excess letdown heat exchanger (0.25 x 6)
- residual heat removal heat exchanger
- residual heat removal pump seal coolers
- primary drains cooler

REFERENCE
Beaver Valley 2LP-SQS-15.1 Enabling Objectives 2, 4, 9
BVPS - OM 2.15.1, pp. 9, 17, 24, 25
K/A 008000 K1.02 3.4
K/A 008000 K3.01 3.5
K/A 008000 A3.04 3.7
008000K301 00800K102 008000A304 ...(KA's)

FACILITY REVIEW OF WRITTEN EXAMINATION

SRO EXAM REVIEW (12-2-87)

6.06.a Valve numbers should not be required for correct answer.

6.06.b The second and third answers in the key are incomplete. The correct answers are:

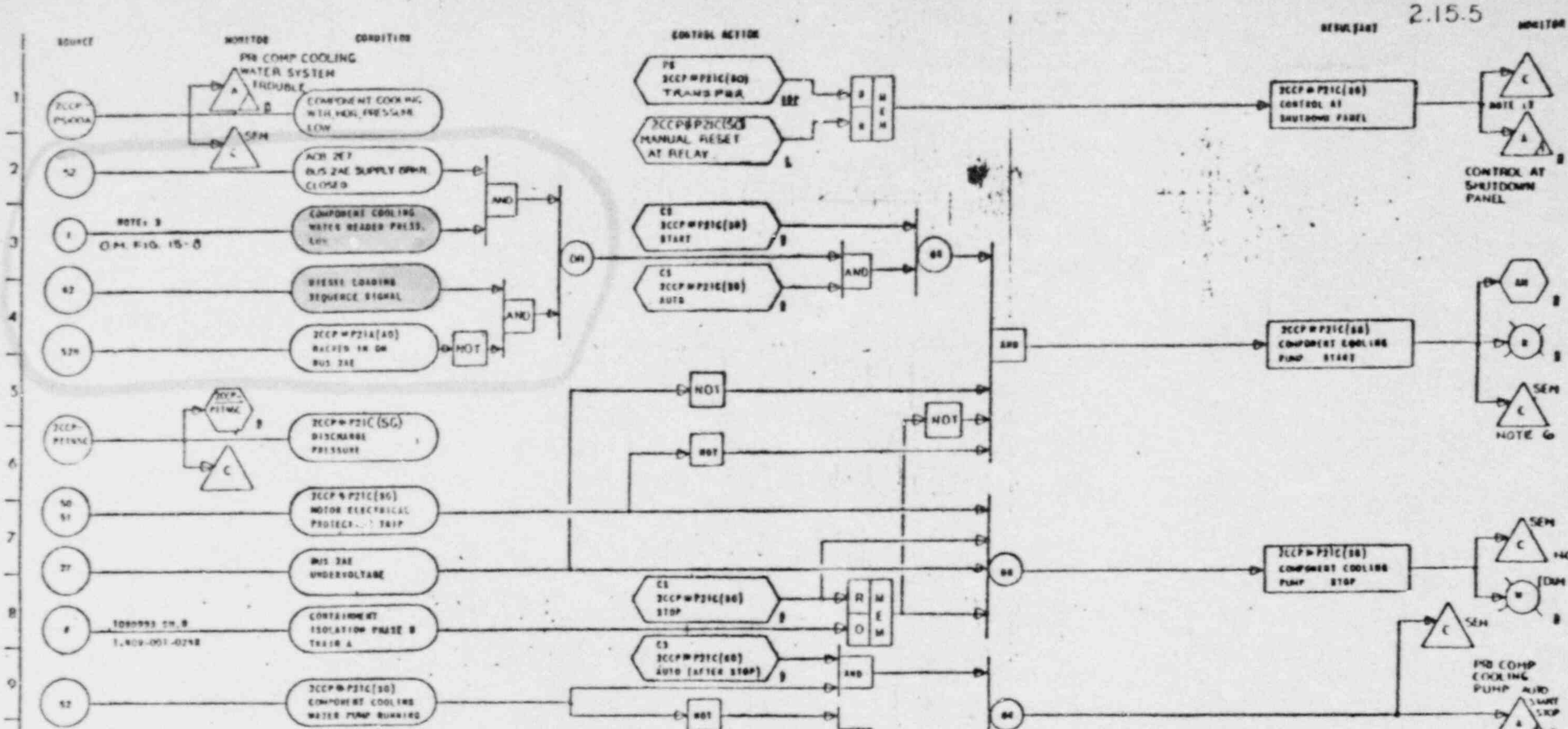
- diesel loading sequence and pump A "not racked in"
- System header pressure low and breaker 2E7 closed.

See attached diagram.

6.06.c Reactor Coolant Pumps is given as an acceptable answer. The RCP's actually have four separate cooling units: Thermal Barrier, Stator, Upper Lube Oil, and Lower Lube Oil. It is requested that each of these be considered a correct answer.

QUESTION 6.06 B

2.15.5



NOTES:

1. LOGIC FOR PRIMARY COMPONENT COOLING WATER PUMP 2CCP*P21C(SG) ON BUS 2AE SHOWN. LOGIC FOR PUMP 2CCP*P21C(SG) ON BUS 2DF SIMILAR.
2. ANNUNCIATOR AND COMPUTER INPUTS COMMON TO ALL SHUTDOWN PANEL TRANSFER SWITCHES.
3. PRESSURE SWITCH 2CCP-PS102 FOR PUMP 2CCP*P21C(SG) ON BUS 2DF.
4. CONTROL FROM BENCH BOARD SHOWN, CONTROL FROM SHUTDOWN PANEL SIMILAR.
5. CONTROL IN THE BENCH BOARD IS ONLY AVAILABLE WHEN THE CONTROL TRANSFER IS RESET.
6. ONE COMPUTER INPUT WILL PROVIDE BOTH ON OFF INDICATION.

B.V.P.S.-O.M.
FIGURE 15-9
ISSUE 1 REV 2

LOGIC DIAGRAM - COOLING WATER SYSTEM
PRIMARY COMPONENT COOLING WATER PUMPS
BEAVER VALLEY POWER STATION - UNIT NO. 2
DUQUESNE LIGHT COMPANY
STONE & WEBSTER ENGINEERING CORPORATION
12241 - LSK-12-7B(7)

FACILITY REVIEW OF WRITTEN EXAMINATION

QUESTION 6.09 (3.00)

For the following reactor protection trips, state its Technical Specification basis AND at what point, if any, the trip is blocked.

a. pressurizer high water level
b. low feedwater flow
c. low reactor coolant flow
d. overpower delta-T
e. pressurizer low pressure
f. power range high negative neutron flux rate
g. reactor trip on a turbine trip

ANSWER 6.09 (3.00)

a. RCS overpressurization (0.30)
   blocked below P-7 (0.20)
b. loss of heat sink (0.30)
c. low DNBR protectic (0.30)
   2 out of 3 loop trip blocked below P-7 (0.15)
   1 out of 3 loop trip blocked below P-8 (0.15)
d. fuel rod rating protection (0.30)
e. low DNBR protection (0.30)
   blocked below P-7 (0.20)
f. low DNBR protection (0.30)
g. provides additional protection beyond that required (0.30)
   blocked below P-9 (0.20)

REFERENCE
Beaver Valley 2LP-SQS-1.1 Enabling Objectives 5, 7
BVPS - OM 2.01.1, pp. 6 through 9
BVPS - Unit 2 Technical Specifications B2-3 through B2-6
K/A 012000 K4.02 3.9
012000K402 ...(KA's)

SRO EXAM REVIEW (12-2-87)

6.09.f An acceptable answer should be: a negative rate trip protects against two or more dropped rods. See attached Tech. Spec. Bases B2-3.

QUESTION 6.09F

specified in Table 2.2-1, in percent span, from the analysis assumptions. Use of Equation 2.2-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

The methodology to derive the trip setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the trip setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Power Range, Neutron Flux

The Power Range, Neutron Flux channel high setpoint provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The low setpoint provides redundant protection in the power range for a power excursion beginning from low power. The trip associated with the low setpoint may be manually bypassed when P-10 is active (two of the four power range channels indicate a power level of above approximately 10 percent of RATED THERMAL POWER) and is automatically reinstated when P-10 becomes inactive (three of the four channels indicate a power level below approximately 10 percent of RATED THERMAL POWER).

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

The Power Range Negative Rate trip provides protection to ensure that the minimum DNBR is maintained above 1.30 for control rod drop accidents. At high power a multiple rod drop accident could cause local flux peaking which, when in conjunction with nuclear power being maintained equivalent to turbine power by action of the automatic rod control system, could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate trip for those control rod drop accidents for which DNBRs will be greater than 1.30.

FACILITY REVIEW OF WRITTEN EXAMINATION

QUESTION 6.11 (1.909)

a. For a large break LOCA, WHAT are the minimum emergency core cooling system pumps required to cover exposed fuel and limit possible core damage? (0.70)

b. Following SI reset, WHAT operator action(s) must be performed in order to reinstate automatic re-initiation of SI? (0.50)

c. The accumulators are isolated during a normal plant cooldown. WHAT RCS system safety limit can be violated if this is not done? (0.70)

ANSWER 6.11 (1.90)

a. 1 HHSI pump (0.35)
   1 LHSI pump (0.35)
b. close the reactor trip breakers (0.50)
c. RCS overpressurization (0.35) at reduced RCS temperatures (0.35)

REFERENCE
Beaver Valley 2LP-SQS-11.1 Enabling Objectives 1, 4, 12
BVPS - OM 2.11., p. 2
BVPS - E.O.P. 2.53.1, ES-1.1, p. 2
Beaver Valley - Unit 2 Technical Specifications, p. B3/4 5-1
K/A 006000 K6.02 3.9
K/A 006000 K6.03 3.9
K/A 006020 K4.06 4.2
006020K406 006000K603 006000K602 ...(KA's)

SRO EXAM REVIEW (12-2-87)

6.11.c Delete last part of answer, question establishes the situation as already being at a reduced temperature.

NOTE: BVPS has two safety limits per Tech. Specs.; neither would pertain to this question. See Tech. Specs. 2.1 and 2.1.2.

FACILITY REVIEW OF WRITTEN EXAMINATION

QUESTION 7.01 (1.50)

For the following questions, assume BVPS - OM 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. With Residual Heat Removal System (RHS) in service, WHY should at least one reactor coolant pump remain in service until RCH temperature is less than 200 degrees F? (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. BVPS - E.O.P. ES-0.2, "Natural Circulation Cooldown" (0.50)

REFERENCE
BVPS 2LP-SQS-2 E.O. 4; 2LP-SQS-50.51.52.1 E.O. 2, 3
BVPS - OM 2.51.4, pp. C9, D2, D4; 2.51.2 p. 3; 2.53C.4, p. 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)

SRO EXAM REVIEW (12-2-87)

7.01.b This question should be deleted. It is not required knowledge. The question is from a note in Procedure D, Chapter 51. The note also contains the reason the note is put in procedure.

7.01.c It is given that the plant is in Mode 4, so RCP's and/or RHR may be in service. If you are using RHR and flow was lost, you would go to AOP 2.10.1, "Loss of RHR". Therefore, AOP 2.10.1 should be an additional acceptable answer.

QUESTION 7018

D. STATION SHUTDOWN - COOLDOWN FROM THE HOT SHUTDOWN (MODE 4) TO THE COLD SHUTDOWN (MODE 5) (Continued)

NOTE: At least one reactor coolant pump should be maintained in operation until reactor coolant system temperature is below 200F. This will provide the reactor vessel head and other stagnant areas with cooling when RHS is in service and maintain the entire reactor coolant system subcooled.

5. Adjust [2RHS*HCV758A(758B)], Benchboard-A, Residual Heat Removal Outlet Flow Control Valve, as necessary, to maintain reactor coolant system temperature at 250 ± 5F. ________/_____

6. When the steam generator pressures reach 25 psig (as verified by computer points P0401A, P0420A, P0440A), place the steam generators in wet layup in accordance with OM-2.21.4.C, "Wet Layup From Hot Shutdown." ________/_____

7. Establish steam generator chemistry as follows:

   a. If necessary, shutdown the steam generator start-up feed pump in accordance with OM-2.24.4.H, "Steam Generator Feedwater System Shutdown." ________/_____

   b. Shutdown the steam generator blowdown system as described in OM-2.25.4.C, "Steam Generator Blowdown System - System Shutdown". ________/_____

   CAUTION: DUE TO DEAD WEIGHT CONCERNS DO NOT EXCEED 98% LEVEL ON WIDE RANGE SG LEVEL INDICATORS.

   c. Coincident with the addition of wet layup chemicals to the steam generators, increase steam generator levels to at least 94% in the wide range in accordance with OM-2.24.4.I, "Feeding Steam Generators at Low Pressure And Little or No Steam Flow." This establishes a level above the primary separators so that the layup chemicals can mix by thermal circulation. ________/_____

FACILITY REVIEW OF WRITTEN EXAMINATION

QUESTION 7.05 (2.00)

a. WHAT would be TWO Reactor Coolant Pump indications that a loop isolation valve was drifting closed? (1.00)

b. WHAT TWO (2) physical actions should the reactor operator take if loop A isolation valve 2RCS*MOV591 is confirmed to be drifting closed? (1.00)

ANSWER 7.05 (2.00)

a. low flow (0.50
   low pump amps (0.50)

b. trip the reactor (0.50)
   trip RCP "A" (0.50)

REFERENCE
BVPS 2LP-SQS-53C.1 E.O.'s 4, 5
BVPS - OM AOP 2.6.3
K/A 000017 EA1.10 2.6
K/A 000017 EA1.12 3.1
K/A 000017 G0.11 3.6
000017G011 000017A112 000017A110 ...(KA's)

SRO EXAM REVIEW (12-2-87)

7.05.a At BVPS, RCP amps increase as flow starts to decrease in the RCS, so amps could initially increase as loop stop valve drifts shut. This should be an additional correct answer. See attached RCP performance curves.

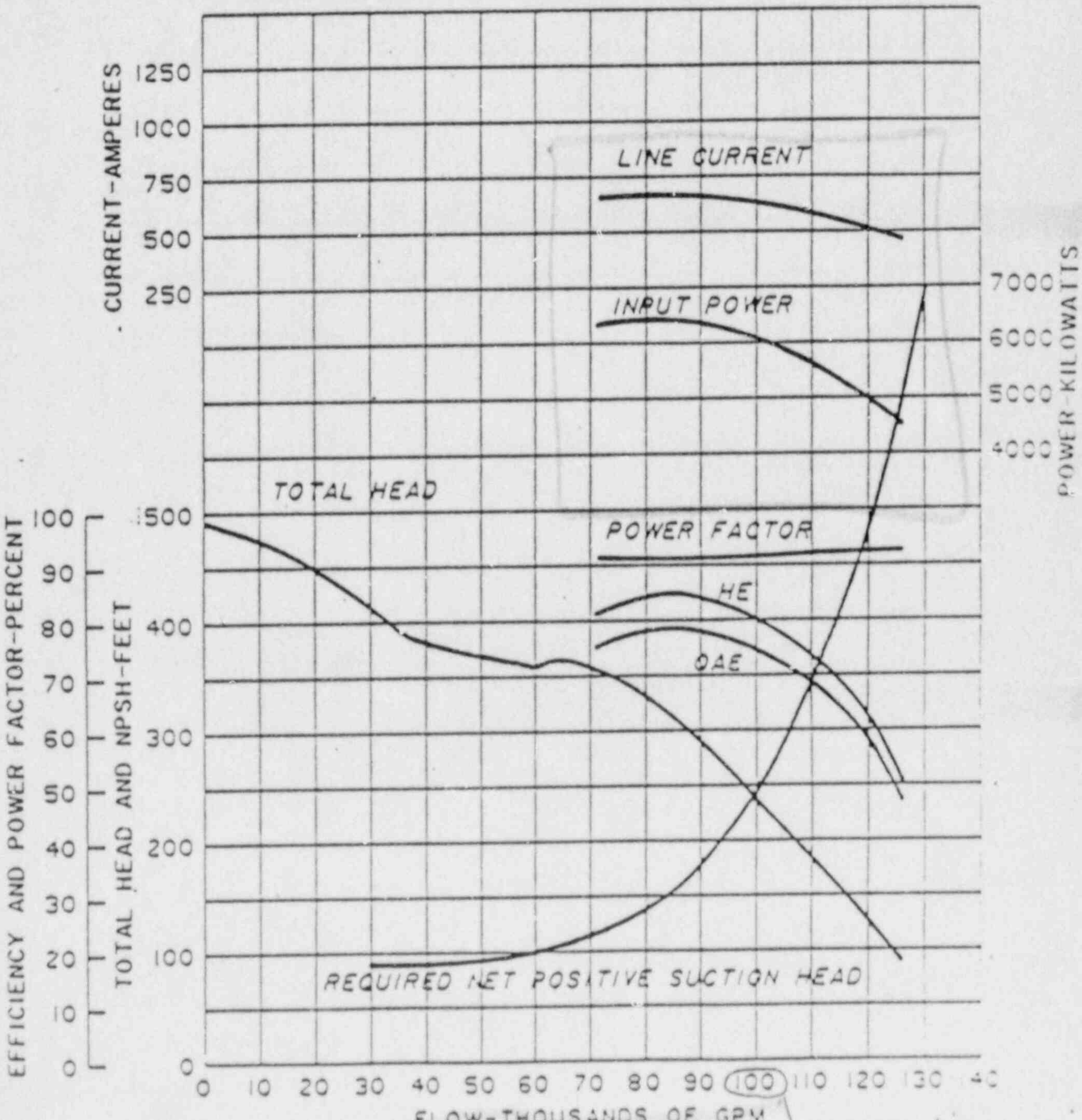
question 705A

DUQUESNE LIGHT COMPANY

BEAVER VALLEY POWER STATION - UNIT NO. 1

70.0°F., 500PSIA., 4160 VOLTS

DLW CURVE NO. 3U326-5C-577 (101584)



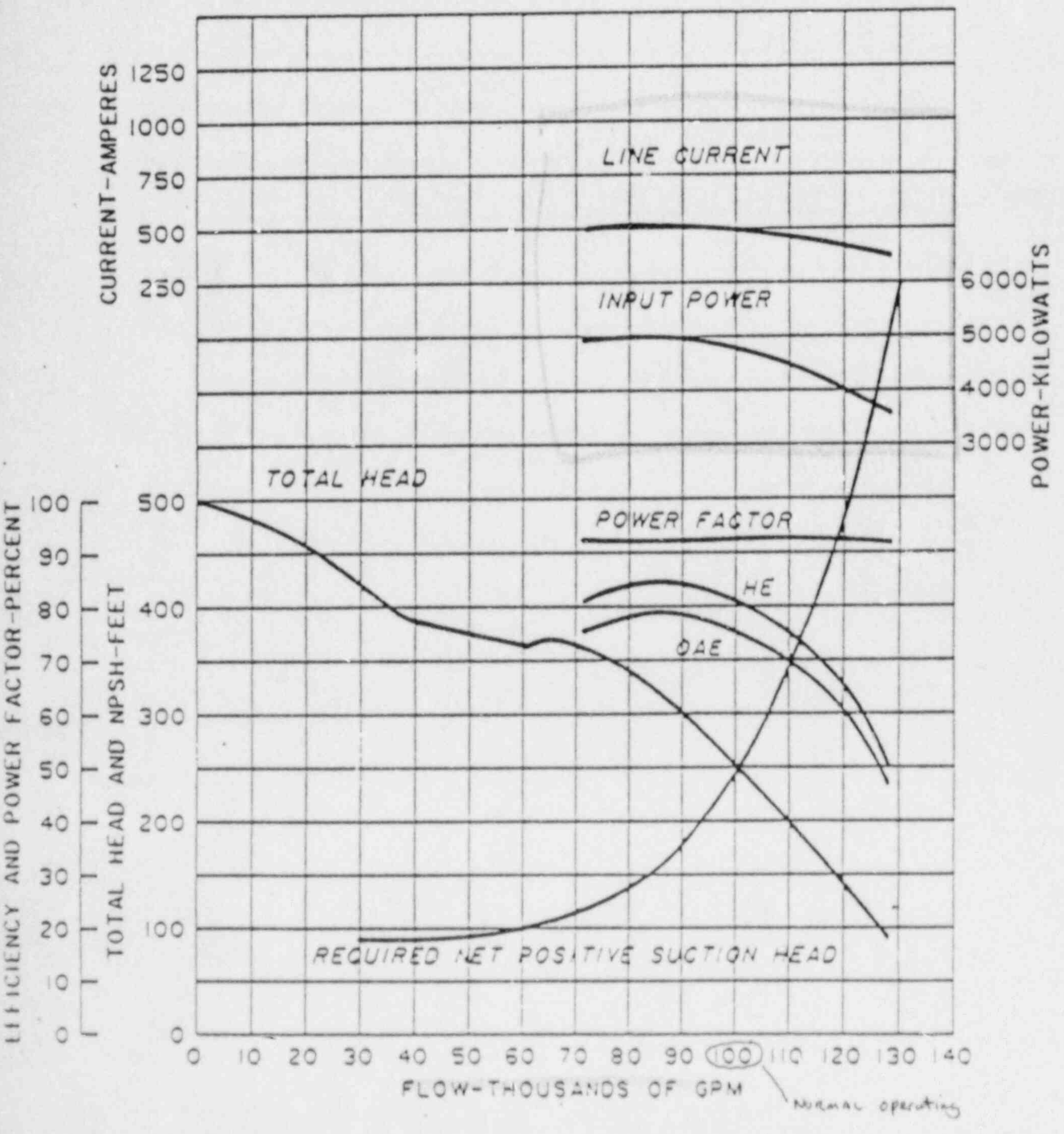
Cold Performance Curves (Average)

RCP PERFORMANCE CURVE (UNIT 1)

Question 7.05A

DUQUESNE LIGHT COMPANY
BEAVER VALLEY POWER STATION - UNIT NO. 1

542.3°F., 2250 PSIA., 4160 VOLTS
DLW CURVE NO. 3U326-5H-577 (101584)



Hot Performance Curves (Average)

RCP PERFORMANCE CURVE (UNIT 1)

FACILITY REVIEW OF WRITTEN EXAMINATION

QUESTION 7.07 (2.50)

During a plant startup with reactor power at 12% a loss of 125 VDC from switchboard 2-2 occurs.

a. WHAT FOUR (4) automatic actions will occur? (2.00)

b. What would be the most likely cause of a trip during this transient? (0.50)

ANSWER 7.07 (2.50)

a. - Feedwater flow control valves (0.40) and feedwater bypass flow control valves (0.40) fail closed (0.20)
 - Letdown isolation valve (2CHS*AOV204) (0.40) fails closed (0.20)
 - Steam driven auxiliary feedpump starts (0.40)

b. - Steam generator low level in coincidence with feedflow/steamflow mismatch (0.50)

REFERENCE
BVPS - OM AOP 2.39.5, p. 1
K/A 000058 EK3.02 4.2
K/A 000058 EA2.03 3.9
000058K302 000058A203 ...(KA's)

SRO EXAM REVIEW (12-2-87)

7.07.b The given answer is impossible since it requires a 40% of full flow steam-feedflow mismatch and the problem states that the plant is at 12% power. The likely cause of the trip would be on low-low steam generator level (15.5%).

FACILITY REVIEW OF WRITTEN EXAMINATION

QUESTION 7.10 (3.00)

For the following questions, assume that a main steam line break from steam generator 21B has occurred inside containment.

a. As directed in BVPS - EOP E-2, "Faulted Steam Generator Isolation, " WHY are the supply valves to the turbine-driven auxiliary feedwater (AFW) pump closed? (0.65)

b. Because of elevated secondary radiation levels, the operator transitions to E-3, "Steam Generator Tube Rupture." After asking for steam generator (SG) samples to be taken, the chemistry department informs the operator that SG 21C has high activity. If the turbine-driven AFW pump is the only available source of feed flow, which SG should be used to supply it with steam? Justify your answer. (1.35)

c. If pressurizer pressure control is lost and the operator transitions to ECA-3.3, "SGTR without Pressurizer Pressure Control" (Attachment 3), WHAT are TWO (2) reasons for NOT using auxiliary spray at step 3 when normal PRZR spray cannot be established? (1.00)

ANSWER 7.10 (3.00)

a. minimize RCS cooldown (0.65)
OR
to isolate steam from the S/Gs (0.65)

b. 21C (0.50)
faulted SG 21B should remain isolated unless needed for RCS cooldown. SG 21A is available, but 21C must be used to steam the AFW pump (0.85)

c. possible spray nozzle failure (0.50)
insufficient pressure differential (0.50)

REFERENCE
BVPS LP-LRT-VII-65 E.O.s B.3, B.4; LP-LRT-VII-67 E.O B.3
BVPS - EOP 2.53A.1 Procedure E-2, p. 2; Executive Volume 2.53B.4
Background Information for Procedure E-2, p. 14
BVPS - EOP Executive Volume 2.53B.4, p. 14
K/A 000038 EK3.06 4.5
K/A 000040 EK3.04 4.7
000040K304 000038K306 ...(KA's)

FACILITY REVIEW OF WRITTEN EXAMINATION

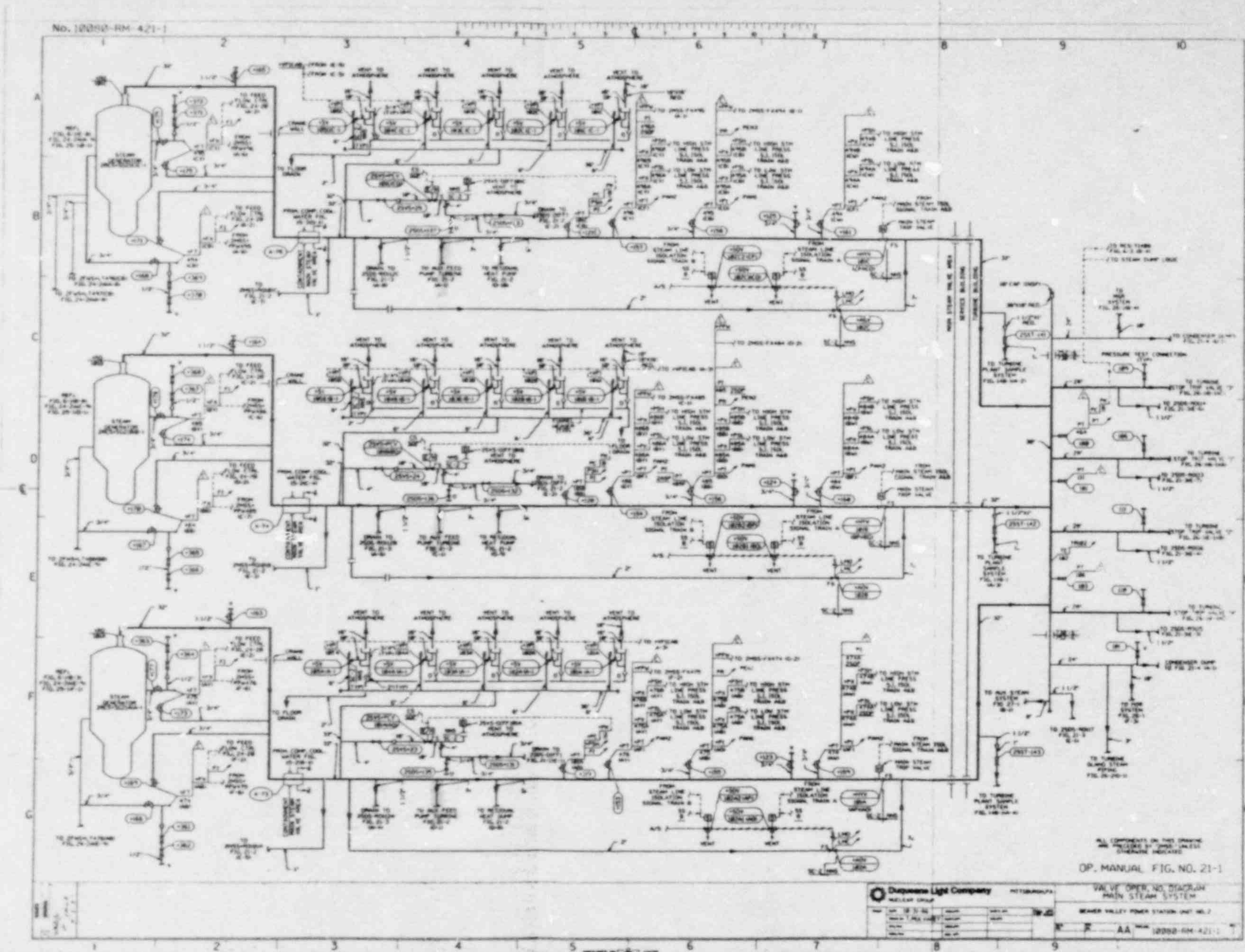
SRO EXAM REVIEW (12-2-87)

7.10.b The correct answer should be SG-A since it is neither faulted or ruptured (intact) and all three SG's can supply the turbine driven aux. feed pump. See attached.

7.10.c An alternate correct answer based on plant conditions should be:

- the plant is still in SI configuration and charging and letdown are isolated.

Question 7.10 B



Question 7.10B

To provide a more rapid means of depressurizing the RCS and ruptured steam generators, a post-SGTR cooldown method using steam dump, ES-3.3, was developed. This is the fastest of the three methods. However, the radiological consequences must be considered particularly if steam dump to the condenser is unavailable. The NRC has indicated that controlled radiological releases from the affected steam generators should not exceed 10 CFR 20 limits. In addition, if water exists in the steamline, steam release may cause water hammer effects resulting in damage to secondary side equipment. Consequently, this method should not be used if water may exist in the main steamlines.

Although the three post-SGTR cooldown procedures are presented as alternate methods, they are similar. Consequently, one could begin with the backfill method, continue with the blowdown method, and complete the recovery using steam dump, provided the limitations of each method are observed. Similarly, for multiple tube failures, one could execute combinations of the three methods at the same time. For example, one could depressurize one ruptured steam generator using blowdown and another using the backfill method.

These procedures provide the flexibility necessary to cool down and depressurize the plant to cold shutdown conditions for a wide variety of SGTR scenarios. The operating shift must evaluate the three methods to establish a preferred post-SGTR cooldown method and prioritize the alternate methods. The actual recovery method must be determined on an event specific basis with consideration of the available equipment and evolution of the event. For example, normal letdown may be needed to complete recovery using the backfill method, ES-3.1, while steam generator blowdown is needed to implement ES-3.2. The procedure ES-3.3 should not be used if water may exit in the main steamlines or steam releases may lead to unacceptable radiological releases.

2. Select Faulted or Ruptured Steam Generator For RCS Cooldown

In the unlikely event that no intact steam generator is available, one must select either a faulted steam generator, i.e., one with a secondary side break, or a ruptured steam generator for cooling the RCS to RHS operating conditions. This decision should be based upon consideration of the concerns created by each method and an evaluation of the parameters that effect them. A secondary side break leads to uncontrolled steaming of the affected steam generator and possible overcooling of the RCS. Continued feed flow to this steam generator will increase the amount of steam discharged and can increase the uncontrolled cooldown of the RCS. The potential consequences of continuing to feed a faulted steam generator depends on the size and location of the secondary side break. For breaks inside containment, feed flow to the affected steam generator will result in additional discharge to containment and potentially higher containment pressures and temperatures.

FACILITY REVIEW OF WRITTEN EXAMINATION

QUESTION 8.02 (2.00)

Answer the following questions 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 NSS/NSOF's permission.

ANSWER 8.02 (2.00)

a. TRUE
b. FALSE
c. FALSE
d. TRUE

REFERENCE
DLC SAP Chapter 41, pp. 17, 47, 50; Chapter 42, p. 6
K/A 194001 K1.02 4.1
194001K102 ...(KA's)

SRO EXAM REVIEW (12-2-87)

8.02.d The correct answer is FALSE. The test crew can only post and/or remove caution tags without NSS/NSOF permission if authorized by an approved test procedure. The question did not state that the test crew member was performing an approved procedure. See attached SAP 42, pp. 2, 3, and 6.

C. The NSOS is responsible for:

1. Scheduling the performance of the periodic audits in accordance with Section VI.D.

2. Maintaining documentation that periodic audits were conducted.

D. Test Group Personnel

1. Are authorized to Caution Tag equipment in a position required by an approved Test Procedure.

V. REFERENCES

A. Commitment References

None

B. Additional References

BVPS OM 1.48.6, Issue 2, Rev. 14.

VI. INSTRUCTIONS

A. Utilization of Caution Tags.

1. Caution Tags are not to be used for personnel safety protection in lieu of a Red Danger Tag for performing work on any plant component.

2. Caution Tags are not to be used for lifting leads or temporarily modifying electrical circuits in lieu of a Jumper and Lifted Leads Tag.

3. The yellow Caution Tag may be utilized to alert station personnel of an unusual operating condition.

4. An apparatus which has a Caution Tag posted shall not be operated until a supervisor in charge of the operation is familiar with the condition under which the tag was posted and authorizes such operation.

5. Yellow Caution stickers should be used on controls mounted on the control boards instead of Caution Tags (Ref. Attachment 1), when a tag has the potential to obscure indicating lights, nameplates, annunciators, or indications.

6. Any Caution Tag used to fulfill instructions dictated by approved procedure does not need documented in the Caution Tag Log provided the procedure(s) address

returning the system or component to its normal state, including placement and removal of the Caution Tag(s). The procedure number or test number shall be placed on the tag in place of the log tag number.

7. A Caution Tag used in conjunction with a clearance does not need documented in the Caution Tag Log, since the Caution Tag is documented on the Clearance's Valving/Switching Procedure Form. When using the Caution Tag with a clearance, the Caution Tag entries shall be filled out as described in this procedure, except the Clearance Permit number shall be indicated on the Caution Tag in lieu of a Caution Tag Log Index sequential number.

8. Caution Tags utilized for procedure or test compliance may be stored for reuse if the procedure is used frequently.

9. Caution Tags must be positioned such that they do not obscure indicating lights, meter faces, annunciators, or nameplates.

10. Any condition requiring a Caution Tag that becomes a permanent plant condition will be noted by permanent Caution Tag and removed from the Caution Tag Log. The list of permanent caution tags will be kept by the N.S.S.

11. Special Caution Tag/Sticker markings may be utilized as directed by the Unit NSOS. These special markings designate specific conditions of annunciator window or control switch status as shown on Attachment 4.

12. Circumstances which should require Caution Tagging:

   a. A Caution Tag may be posted on a component to instruct an operator to notify specific station personnel prior to the operation of the component.

   b. A Caution Tag may be used to identify a component that is in a temporary configuration or to inform personnel of the use of temporary equipment that is required to support the operation of the tagged component.

   c. A Caution Tag may be used to inform an operator of special additional manual actions that are required to operate the tagged component.

c. 480V and 4160V Breakers - install the tag in clear view on the outside of the breaker door

C. Removal of Caution Tags.

1. Except as specified in IV.D, VI.A.6 and VI.A.7, obtain NSS/NSOF's permission to remove the Caution Tag.

2. Remove the Caution Tag from the component.

3. Fill out the remaining portion of the applicable Units Caution Tag Log (Ref. Attachment 2) for the Caution Tag removed, including:

a. The reason the Caution Tag was removed.

b. The date the Caution Tag was removed.

c. The name of the person removing the Caution Tag.

4. Line out the Caution Tag Log Index for the Caution Tag removed, including:

a. The serial number.

b. The system number.

c The item tagged.

D. Periodic Audit of Caution Tags.

1. The Caution Tag Log Review, (OST 1/2.48.6) shall be performed to ensure that Operations conform to the administrative requirements for Caution Tags.

2. Caution Tagged components inside containment are exempt from the Caution Tag Log Review when containment vacuum is established.

3. The Caution Tag Log Review will be completed prior to leaving Mode 5 following an extended maintenance outage.

4. The review shall be conducted a minimum of 3 times per year.

FACILITY REVIEW OF WRITTEN EXAMINATION

QUESTION 8.07 (2.40)

For the following questions, utilize Attachment 5.

a. WHAT would be the effect on the Safety Injection System during a LOCA is CONTROLLED LEAKAGE had exceeded the Technical Specification limit prior to the event? (0.60)

b. Would Limiting Condition for Operation 3.4.6.2 be exceeded if containment sump discharges totaled 625 gallons during the last hour? Justify WHY or WHY NOT. (1.40)

c. Is reactor-to-secondary leakage included as part of IDENTIFIED LEAKAGE? (0.40)

ANSWER 8.07 (2.40)

a. Safety injection flow would be less than assumed by the FSAR accident analyses (0.60)

b. no (0.40)
600 gallons (10 GPM x 60 mins) of IDENTIFIED LEAKAGE (0.50)
plus 60 gallons of UNIDENTIFIED LEAKAGE (0.50) is the limit for 1 hour

c. yes (0.40)

REFERENCE
BVPS - Unit 2 T.S 3.4.6.2; Definition 1.14; B 3/4 4-4
K/A 006000 G0.05 4.2
K/A 006000 G0.06 4.0
006000G006 006000G005 ...(KA's)

SRO EXAM REVIEW (12-2-87)

8.07.b This question should make allowances for assumptions as to where the water was coming from since it wasn't specified in question. If the student assumed that the entire supply was either identified or unidentified, the answer would be yes. They also could have assumed that the water was not from the RCS but from other sources such as main feed, chilled water, CCP, fire header leakage, PG water, etc., in which case the answer would be no, since the T.S. only refers to RCS leakage.

ATTACHMENT 4

NRC RESPONSE TO FACILITY COMMENTS

The following statements address the NRC's disposition of comments on the written examination submitted by the facility (see Attachment 2) and changes made to the answer key during the grading process.

REACTOR OPERATOR EXAMINATION

1. PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW

1.02 Comment accepted. No answer key change required.

1.04.b Comment accepted. No answer key change required.

1.06 Comment accepted. No answer key change required.

2. PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS

2.06.a Comment accepted. Answer key changed to reflect additional information provided by the facility. Information should also be incorporated into training material.

3. INSTRUMENTS AND CONTROLS

3.02b & c Comment accepted. Answer key changed.

3.04.c Comment accepted. Answer key changed to reflect information provided by the facility. Information should also be incorporated into training material and procedures.

3.06.b Comment accepted. Answer key changed to reflect information provided by the facility.

3.07.d Comment accepted. Answer key changed to reflect information provided by the facility. Information should also be incorporated into training material and procedures.

3.12.a Comment accepted. Answer key changed to reflect information provided by the facility. Material originally provided was inconsistent and should be corrected.

4. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL

4.09.a & c Comment not accepted. In both cases one AFD value is outside the target band.

4.11.b Comments not accepted. Knowledge of facility ALARA program is covered by the KA Catalog, NUREG-1122, KA # 194001K104, with an importance to safety factor of 3.3. An essential element of the ALARA program should be who is authorized to establish and extend exposure limits.

The question neither stated nor implied that emergency exposure was involved.

SENIOR REACTOR OPERATOR EXAMINATION

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

5.01 Comment accepted. Blowdown flow was accepted as an alternate answer.

5.04 Comment accepted. Allowances were incorporated into the answer key.

5.09 Comment noted. The answer key was changed to eliminate "hot channel factor" as part of the required response. "Thermal limits" or "DNB bounds" are, however, required for a full credit answer.

5.10 Comment accepted.

6. PLANT SYSTEMS DESIGN, CONTROL AND INSTRUMENTATION

6.01.a Comment accepted.

6.02.a Comment accepted. The answer key was changed to "Containment purge (elevation 782)."

6.03.a Comment accepted.

6.04.a Comment accepted. The answer key was changed to accept alternate answers.

6.06 Comments noted. The answer key for 6.06a. was modified per the facility's request. The answer key for 6.06.c was modified to accept any of the RCP components as a correct answer.

6.09 Comment noted. The answer key was modified as follows, "low DNBR protection [0.15] for control rod drop accidents [0.15]."

6.11 Comment accepted.

7. <u>PROCEDURES - NORMAL, ABNORMAL, EMERGENCY, AND RADIOLOGICAL CONTROL</u>

7.01.b Comment not accepted. BVPS enabling objective 2LP-SQS-50.51.52.1 number 2 states the requirement to be able to explain the basis for "Cautions" and "Notes."

7.01.c Comment not accepted. The "Immediate" actions required by BVPS-OM 51 is to refer to BVPS-E.O.P. ES-0.2.

7.05.a Comment accepted.

7.07.b Comment accepted. The answer key was changed to, "low-low steam generator level."

7.10.b Comment accepted. The answer key was changed to, "S/G 21A [0.50] because it is neither faulted or ruptured [0.85]."

7.10.c Comment accepted. The following alternate answer was incorporated into the answer key, "normal charging flow is isolated."

8. <u>ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS</u>

8.02.d Comment accepted.

8.07.b Comment accepted. Grading was based on the candidates' assumptions. The answer key was modified to incorporate alternate correct responses.

Attachment 5

NRC COMMENTS ON FACILITY REFERENCE MATERIAL

Generic

Many lesson plans (LP's) focus only on differences between units. For future exams either the Unit 2 material should stand alone or Unit 1 references should be provided.

Figures referred to in LP's as TP's were not included.

Specific

1 RHS- LP was very sparse. Contained no Learning Objectives (LO's).

2. Reactor Control- LP provides only LO's. Operating Manual (OM) does not support LO's. Figures do not match OM references.

3. P-9 - LP-SQS-1.1 shows 70%, OM-2.1 shows 4%, alarm response OM-2.26 shows 70%.

4. LP-SQS-26.2 stated "Same as Unit 1", but Unit 1 information not provided.

5. LP-SQS-24.1 - LP for SG Feedwater does not discuss SG WLC system.

6. RCS Isolation Valve Interlock- OM-2.6.3, p 7,9 lists 20 F; OM-2.6.1,p 48 lists 5 F; LP-SQS-6.2.2 lists "predetermined value."

7. RIL Curve - Figure in Curve Book is different from Tech Specs.

8. RCP Start Interlock - OM-6.3 states that RCP start requires Tc isolation valve closed with loop by-pass isolation valve open; LP-SQS-6.2, p 6 states that pump start requires Tc and loop bypass valves both open.

9. Main Generator and Transformers- No LP provided.

10. OM-26 - References Figure 26-20 for Turbine Runback; actual reference should be 26-22.

11. OM-7.1 - Figure 7-7 shows that either 460A or 460B not open will auto close AOV200A,B,&C. LP-SQS-7.1 states that 460A and 460B must both be closed to auto close AOV200A,B&C.

12. OM-43.1 - References Figures 43-2 through 43-12 which were not provided.