A

NRC FORM 157 (10-75)

U.S. NUCLEAR REGULATORY COMMISSION

EXAMINATION REPORT

LICENSE APPLIED FOR: RO ☐ SRO ☒

CURRENTLY A LICENSED OPERATOR HERE? YES ☒ NO ☐

APPLICANT'S NAME	REACTOR	LOCATION
	Three Mile Island #II	Middletown Pa.

WRITTEN EXAMINATION

OPERATOR (SECTION 55.21)	ADMINISTERED BY:	DATE	WAIVED ☒
	GRADED BY:	GRADE %	EVALUATION PASSED ☐
	CATEGORY GRADES A ___ % B ___ % C ___ % D ___ % E ___ % F ___ % G ___ %		FAILED ☐
SENIOR OPERATOR (SECTION 55.22)	ADMINISTERED BY:	DATE	WAIVED ☒
	GRADED BY:	GRADE %	EVALUATION PASSED ☐
	CATEGORY GRADES H ___ % I ___ % J ___ % K ___ % L ___ %		FAILED ☐

OPERATING TEST

(SECTION 55.23)	ADMINISTERED BY: R. J. Bursey	DATE June 6, 1978	WAIVED ☐ PASSED ☒ FAILED ☐ HOT ☒ COLD ☐

COMMENTS

Partial exam based on exam administered 9/21/77
spent 2.5 hrs in Control Room with applicant

RECOMMENDATION	APPROVE FOR SENIOR LICENSE ☒ OPERATOR LICENSE ☐	SIGNATURE [signature] 6/12/78
	DO NOT APPROVE FOR SENIOR LICENSE ☐ OPERATOR LICENSE ☐	EXAMINER [signature] DATE 6/8/78
ISSUE LICENSE	SENIOR LICENSE ☐ OPERATOR LICENSE ☐	SIGNATURE [signature] Chief, Operator Licensing Branch 6/13/78

NOTE: The operator and oral examination check sheet must be completed for each applicant. If applicant has failed the operating and oral test at the Senior Level, the check sheet should be completed to indicate applicant's evaluation at the operator level.

NRC FORM 157 (10-75)

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CURRENTLY A LICENSED OPERATOR HERE? YES ☐ NO ☑

EXAMINATION REPORT

APPLICANT'S NAME	REACTOR	LOCATION
	Three Mile Island #II	Middletown Pa.

WRITTEN EXAMINATION

OPERATOR (SECTION 55.21)	ADMINISTERED BY: [signature]	DATE 9/17/77	WAIVED ☐
	GRADED BY: [signature]	GRADE 87 %	EVALUATION PASSED ☑ FAILED ☐
	CATEGORY GRADES A 86% B 83% C 81% D 89% E 91% F 89% G 92%		
SENIOR OPERATOR (SECTION 55.22)	ADMINISTERED BY [signature]	DATE 9/18/77	WAIVED ☐
	GRADED BY: [signature]	GRADE 84 %	EVALUATION PASSED ☑ FAILED ☐
	CATEGORY GRADES H 84% I 77% J 89% K 90% L 81%		

OPERATING TEST

(SECTION 55.23)	ADMINISTERED BY: MG Woram	DATE 9/21/77	WAIVED ☐ PASSED ☐ FAILED ☒ HOT ☑ COLD ☐

COMMENTS

Applicant on site for only 6 months. Not familar enough with control room. Progress very slow and answers had to be lead out of him. Not a bad job outside of control room. Unable to apply theory to practical situations

RECOMMENDATION	APPROVE FOR: SENIOR LICENSE ☐ OPERATOR LICENSE ☒	SIGNATURE [signature]	9/28/77 DATE
	DO NOT APPROVE FOR: SENIOR LICENSE ☑ OPERATOR LICENSE ☐	EXAMINER [signature]	10/5/77 DATE
ISSUE LICENSE	[redacted] OPERATOR LICENSE ☑	SIGNATURE [signature] Chief, Operator Licensing Branch	10/17

NOTE: The operator and oral examination check sheet must be completed for each applicant. If applicant has failed the operating and oral test at the Senior Level, the check sheet should be completed to indicate applicant's evaluation at the operator level.

NRC FORM 157 (10-75)

OPERATING AND ORAL EXAMINATION SUMMARY REPORT

If an SRO applicant is unsatisfactory at the SRO Level, complete the RO column to evaluate applicant's adequacy at the RO Level. **For each unsatisfactory ("U"), list the page number(s) of the operating oral examination notes on which the unsatisfactory responses are explained.*	EVALUATION		
	SRO	RO	*PAGE NUMBER FOR "U"
1. OPERATING DEMONSTRATION			
1.1 Pre-startup and Instrument Checks	S		
1.2 Console Operation			
a. Manipulations	–		
b. Understanding	–		
2. FACILITY EQUIPMENT			
a. Major	S		
b. Auxiliary	S		
c. Engineered Safeguards Systems	S		
3. INSTRUMENTATION			
a. Nuclear	S		
b. Process	S		
4. REACTOR PROTECTION	S		
5. PROCEDURES			
a. Normal	S		
b. Abnormal	S		
c. Emergency	S		
6. REACTIVITY EFFECTS (Except Console Operation)	S		
7. ADMINISTRATIVE REQUIREMENTS	S		
8. EMERGENCY PLAN	–		
9. RADIATION PROTECTION	M		
10. RESPONSIBILITIES			

COMMENTS

OPERATING AND ORAL EXAMINATION SUMMARY REPORT

*If an SRO applicant is unsatisfactory at the SRO Level, complete the RO column to evaluate applicant's adequacy at the RO Level. *For each unsatisfactory ("U"), list the page number(s) of the operating oral examination notes on which the unsatisfactory responses are explained.	EVALUATION		
	SRO	RO	*PAGE NUMBER FOR "U"
1. OPERATING DEMONSTRATION			
1.1 Pre-startup and Instrument Checks	S	S	
1.2 Console Operation			
a. Manipulations	–		
b. Understanding	S	S	
2. FACILITY EQUIPMENT			
a. Major	S	S	
b. Auxiliary	S	S	
c. Engineered Safeguards Systems	S	S	
3. INSTRUMENTATION			
a. Nuclear	U	S	5
b. Process	U	S	4
4. REACTOR PROTECTION	S	S	
5. PROCEDURES			
a. Normal	S	S	
b. Abnormal	S	S	
c. Emergency	S	S	
6. REACTIVITY EFFECTS (Except Console Operation)	U	S	8
7. ADMINISTRATIVE REQUIREMENTS	S	S	
8. EMERGENCY PLAN	S	S	
9. RADIATION PROTECTION	U	S	7, 8
10. RESPONSIBILITIES	S	S	

COMMENTS

50-208

NRC FORM 157
(10-75)

U. S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR LICENSE EXAMINATION

Facility Three Mile Island - Unit 2

Reactor Type PWR - B&W

Date Administered 8/17/77

Examiner J. J. Buzy

Applicant

1730

Instructions to Applicant:

Use separate paper for the answers; staple question sheet on top of the answer sheets. Points for each question are indicated in parenthesis after the question. An overall score of 70% or greater is passing.

Category Value	% of Total	Applicant's Score	% of Cat. Value	
14	13.5	12.1	86	A. Principles of Reactor Operation
14	13.5	11.6	83	B. Features of Facility Design
17	16	13.7	81	C. General Operating Characteristics
16	15	14.2	89	D. Instruments and Controls
16	15	14.5	91	E. Safety and Emergency Systems
16	15	14.3	89	F. Standard and Emergency Operating Procedures
13	12	12.0	92	G. Radiation Control and Safety
106		92.4		

Final Grade 87 %

76-219

A. PRINCIPLES OF REACTOR OPERATION (14)

A. 1. a) What is meant by the term "delayed neutron fraction"? (1.0)

b) How and why does the delayed neutron fraction change during core lifetime? (1.5)

A. 2. The TMI procedure for approach to criticality requires the operating staff to plot a one over M (1/M) during rod withdrawal.

a) What data is required for the 1/M plot? (0.5)

b) How is the data utilized? Note: a sketch may be used as part of your answer. (1.5)

A. 3. Assume your reactor is on a stable .8 DPM startup rate as power level reaches 2×10^{-11} on the intermediate range monitor. (Time is "0").

a) Plot the trace on Figure A1 until the power level reaches 10^{-8}. (1.5)

b) After a two-minute "hold" to take data at 10^{-8} the operator withdraws the control rods to obtain a 1 DPM startup rate. Plot this trace for 5 minutes; include the period of rod motion. Explain any changes during the five minutes. (1.5)

c) Assume the reactor was at 10^{-6} at 3 minutes on figure A1 and a reactor trip occurred. Plot and discuss the resulting trace for the remaining period of time on figure A1. (1.0)

A. 4. Xenon conditions include periods when xenon is at equilibrium and at peak conditions.

a) What is meant by equilibrium xenon? (1.0)

b) How does peak xenon develop? (1.5)

c) Assume a plant startup and power escallation is in progress at the time of peak xenon. How does this differ from a power escallation after a one-week shutdown? Note: a sketch may be used as part of your answer. It is not necessary to include values for xenon. (1.5)

A. 5. Water is used as a coolant in your reactor. List three other functions of water in the reactor. (1.5)

B. FEATURES OF FACILITY DESIGN (14)

B. 1. The nuclear services closed cooling water (NSCCW) system provides cooling water for the nuclear components during normal operation and also provides cooling for components during a loss of coolant accident.

a) List those components which use the NSCCW during normal operation. List the components (i) Inside and (ii) Outside the reactor building. (2.5)

b) What components are supplied during a LOCA? Note: if components appear in part A, mark them with an asterisk*. (1.0)

B. 2. Condensate and feedwater are heated in a number of heat exchanges before water enters the steam generators. Sketch a one-line diagram of the combined systems from the condenser until it leaves the last heat exchanger. Include all pumps and source of heat for each heat exchanger for one string. (2.5)

B. 3. The fire protection system at your facility consists of a number of pumps located in unit one and two. The fire main supplies a number of hydrants, hosereel stations, a wet pipe sprinkler and a deluge system.

a) List all pumps which supply the fire main; include capacity and when each pump will start. (2.0)

b) Explain the difference between the wet pipe sprinkler and the deluge water systems. (1.0)

B. 4. One of the components associated with the reactor coolant system is the R. C. drain tank.

a) List the equipment which discharges to the drain tanks. (1.0)

b) What limits apply to the tank? Include the reasons for these limits. (1.5)

c) If the drain tank level is too high, where may the operators discharge the excess liquid? (0.5)

B. 5. One of the Limits and Preventions require that systems containing liquids should be free of air before they are operated to prevent water hammer.

a) What is water hammer and why should plant operators be concerned about water hammer? (1.0)

b) Does water hammer occur only with systems containing liquid and air? Explain your answer. (1.0)

C. GENERAL OPERATING CHARACTERISTICS (17)

C. 1. The once-through steam generators (OTSG) at your facility will produce superheated steam during power operation.

a) With the aid of a simple sketch, explain how superheat is accomplished in the OTSG. (1.5)

b) How does superheat vary from 15 to 100% power? A sketch may be used as part of your answer. (1.0)

C. 2. The following are two of the limitations and precautions in the Unit Startup procedure. Provide the value and reason for each of the limitations.

a) Do not attempt to start a reactor coolant pump when power is greater than _____. (2.0)

b) During startup and shutdown when reactor power is less than _____ percent, the OTSG water level shall be maintained between _____ and _____. (2.0)

C. 3. Unless specified by the reactor engineer, group 8 (APSR) will be positioned about 32% withdrawn prior to an approach to criticality.

a) With the aid of a sketch, indicate how the APSRs worth varies between 0 - 100% withdrawn. (Include approximate reactivity values) (1.5)

b) Explain when it is necessary to move the APSRs (i) in and (ii) out. (2.0)

C. 4. Two of the limits and precautions for the control rod drives concern the gas concentration in the RCS and the cooling water to the mechanism.

a) What are the sources of gas in the R.C.S.? (1.0)

b) Why is gas concentration a factor for the CRDMs? (1.0)

c) Is it necessary to maintain a minimum concentration of gas the RCS during power operation? Explain your answer. (1.0)

d) The temperature of cooling water to the CRDM may vary from 85 - 105°F. Is there a preferred temperature range? Explain. (1.0)

C. 5. Assume rated power conditions and a trip occurs to a reactor coolant pump.

a) Provide the initial steady state values for parameters listed in figure C5. (1.0)

b) Show the response to these parameters until conditions stabilize. Assume the reactor does not trip. (2.0)

D. INSTRUMENTS AND CONTROL (16)

D. 1. Refer to the transient in question C.5. Discuss the action of the control system(s) for each of the following parameters:

a) Load MWe (1.5)

b) Pressurizer level (1.5)

c) Feedwater flow A & B (2.0)

D. 2. The intermediate channel utilizes a compensated ion chamber (CIC).

a) With the aid of a simple sketch, explain the operating principle of a CIC. (1.0)

b) Is compensation required at 100%, 15%, or lower power levels? Explain your answer. (1.0)

CAN A CHANGE IN COMPENSATION RESULT IN EXCEEDING

c) ~~Can compensation change~~ the "overlap" requirements between the other channels? Explain. (1.0)

D. 3. The pressurizer electromatic relief value (RC-2) may be operated in the manual or automatic mode. Discuss how the valve may be operated in each mode. A simple sketch may be used as part of your answer. (2.0)

D. 4. Unit 2 of TMI utilizes a boron feed and bleed PERMISSIVE controller.

a) What are the input signals to the controller? (1.0)

b) What are the output signals of the controller? (1.5)

c) When is the controller utilized? (0.5)

D. 5. Two of the alarms on the Diamond Control panel are the (i) motor rotation fault and the (ii) asymmetric monitor.

a) Discuss the purpose of each alarm. (1.5)

b) What immediate action is required by the operating staff for each alarm? (1.5)

E. SAFETY AND EMERGENCY SYSTEMS (16)

E. 1. The power distribution system at TMI-2 contains two DC buses, 2-1DC and 2-2DC.

a) Use a one-line diagram to show what components supply these busses and their source(s) of power. Include any cross ties between these busses and supplies. (2.0)

b) If a station blackout occurs, what loads are supplied by the D.C. bus until the emergency diesel generators begin to supply power? (1.5)

E. 2. List the conditions that will cause a reactor trip with four R.C. pumps in service. Include present set points. (2.5)

E. 3. Assume a loss of coolant accident (LOCA) occurs. The LOCA increases reactor building pressure at 2PSI/minute and doubles each minute (2, 4, 8, etc.). Utilize this rate for the following:

a) What automatic action(s) occur during the first four minutes? (1.0)

b) What instruments will the operator observe to assure initial and continuing operation of equipment? (2.5)

c) What is the postulated peak reactor building pressure in the design basis LOCA? (0.5)

E. 4. Three Mile Island Unit 2 is provided with a feedwater latching system.

a) List the instrumentation associated with this system and explain the logic that will actuate the system. (1.5)

b) List the automatic action that will occur when this system is actuated. Assume rated power operation. (2.0)

E. 5. Assume a primary to secondary leak develops in one of the OTSG. The leak begins at one GPM and doubles every 5 minutes (1, 2, 4, 8, etc.)

a) List the systems that will alert the operator. Attempt to arrange them in order of increasing leak rate. (1.0)

b) When the operators have identified the leaking OTSG, should they trip the reactor or begin an orderly shutdown? Explain the reason for your answer. (1.5)

F. STANDARD AND EMERGENCY OPERATING PROCEDURES (16)

F. 1. Assume your unit is at the power level cutoff point and waiting to meet the requirements before increasing power. All systems are in automatic. A malfunction occurs in the circulating water system and vacuum begins to decrease at 1" Hg/min.

a) What is the power level cutoff point and when may power be increased above this level? (1.5)

b) How will the ICS respond as vacuum decreases? (1.0)

c) What immediate action is required by the operator (assume vacuum continues to decrease)? (1.0)

d) List the heat sink(s) available if corrective action can not restore vacuum. (1.0)

F. 2. One of the Limits and Precautions of the Decay Heat/Injection Pumps states "When operating in the ES mode with suction on the building sump, pump discharge must be throtled to _____ GPM to limit runout. Do not throttle below _____GPM as this may result in cavitation and inadequate cooling."

a) What are the above flow rates? (0.5)

b) What is pump runout and what are the consequences if the pump continues to operate in this condition? (1.5)

c) What is cavitation and how would the operator observe this condition on the D.H./Injection pumps? (1.5)

F. 3. During reactor startup, limits are imposed on the amount of reactivity which may be added above and below the estimated critical position.

a) What are these limits and how does the operator convert them to rod position? Assume an ECP of 50% on Group 6/7. (1.5)

b) What action(s) are required if these limits are exceeded? (1.5)

F. 4. In the event of a reactor trip, list the manual actions required by the TMI-2 emergency procedure. (3.0)

F. 5. In the event of an unanticipated criticality:

a) What symptons would alert the operating staff? (1.0)

b) What immediate action is required by the operator? (1.0)

TC 355

G. RADIATION CONTROL AND SAFETY (13)

G. 1. Assume you have been working in a contaminated area. After you remove your protective clothing you monitor yourself before leaving the area.

a) Should an ionization or GM device be used? Explain. (1.0)

b) What type of radiation will the instrument detect? (0.7)

c) What is the criteria for determining if a person is contaminated? (0.8)

G. 2. After working in area for two hours, you discover your dosimeter is off scale and you leave the area. A survey indicates a hot spot on a drain line reads 1200 mr/hr gamma at two feet. Your distance from the line was about five feet.

a) Estimate the dose which you may have received. (1.5)

b) If your previous exposure to the quarter was 900 mr, what exposure limits may have been exceeded? (1.0)

c) With the aid of a simple sketch, explain the operating principle of a pocket dosimeter. (1.0)

G. 3. Three of the classified areas in the Radiation Protection Manual at TMI are (i) controlled area (ii) radiation area and (iii) high radiation area.

a) Define each area; include any limits. (1.5)

b) May the NRC examiner enter each area? Explain your answer. (1.0)

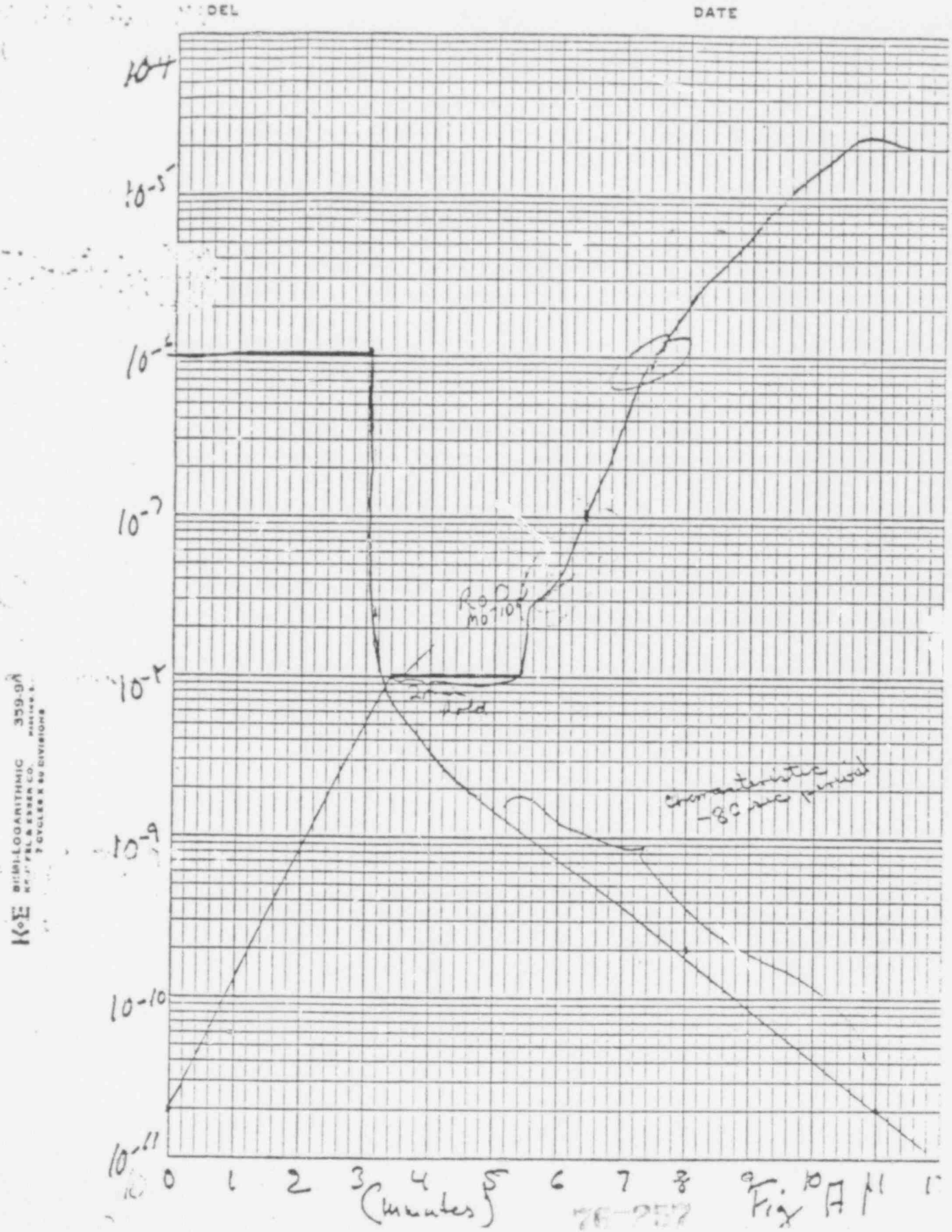
G. 4. In the event of a radioactive spill, one of the basic rules is to administer aid to anyone who is injured. What other basic rules apply? (2.0)

G. 5. During a refueling outage you are in the reactor building on the main fuel handling bridge. A spent fuel element drops into the core.

a) What instrumentation will alert the personnel in the reactor building of any release of fission products? (1.0)

b) What automatic action(s) may occur if radiation levels exceed preset limits? (1.0)

c) What actions are required by the control room staff? (0.5)



DEL
DATE
10-4
10-5
10-6
10-7
10-8
10-9
10-10
10-11
0
1
2
3
4
5
6
7
8
9
10
11
(minutes)
Rod
MOTION
characteristic
-80 sec period
K+E SEMI-LOGARITHMIC 359-96
KEUFFEL & ESSER CO.
7 CYCLES X 90 DIVISIONS

76-257

Fig A1

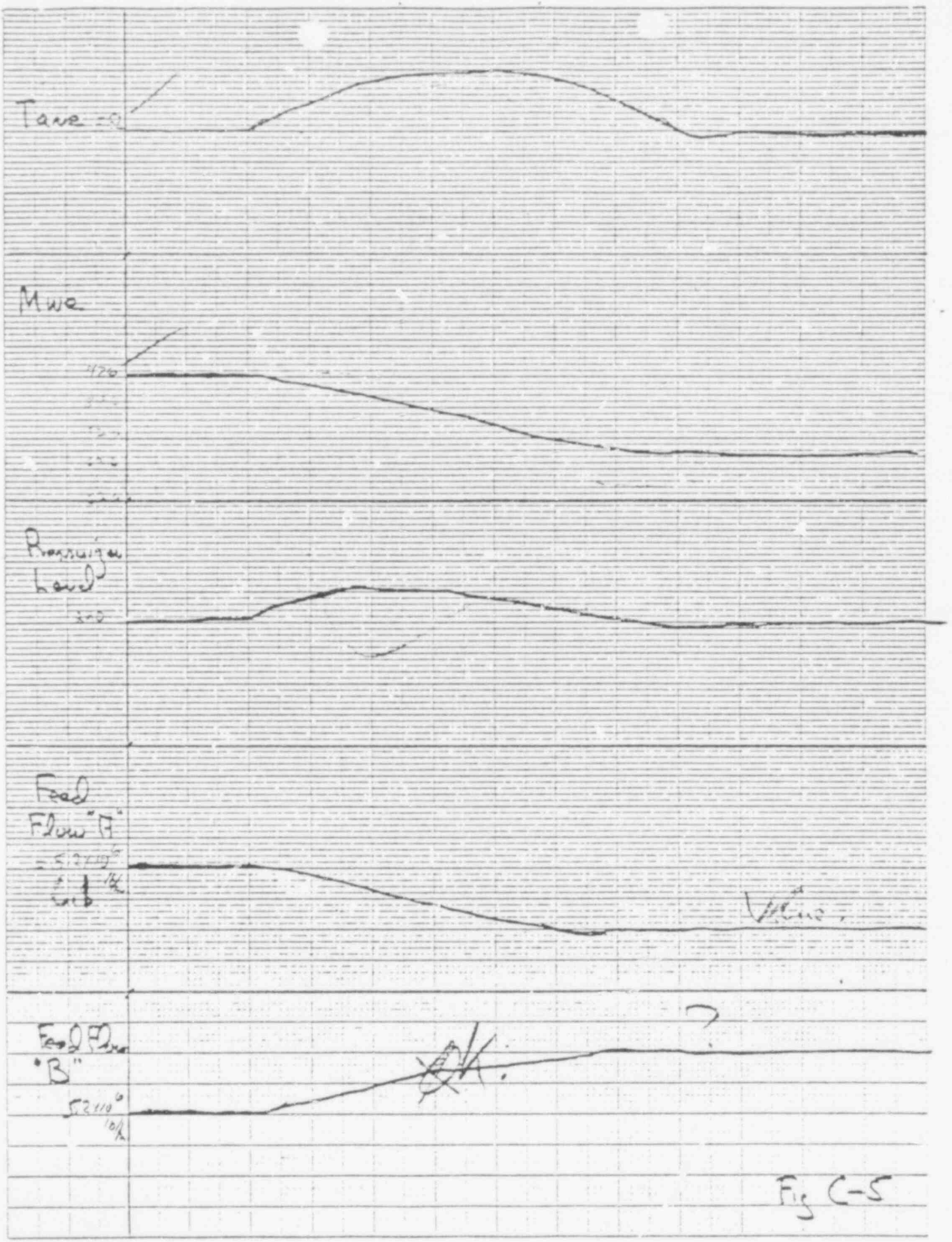


Fig C-5

FIGURE 2B: CONTROL ROD GROUPS 5,6 & 7 Revision 0
REACTIVITY WORTHS @ 532°F, 2155 psig, 0% F.P. 06/14/77

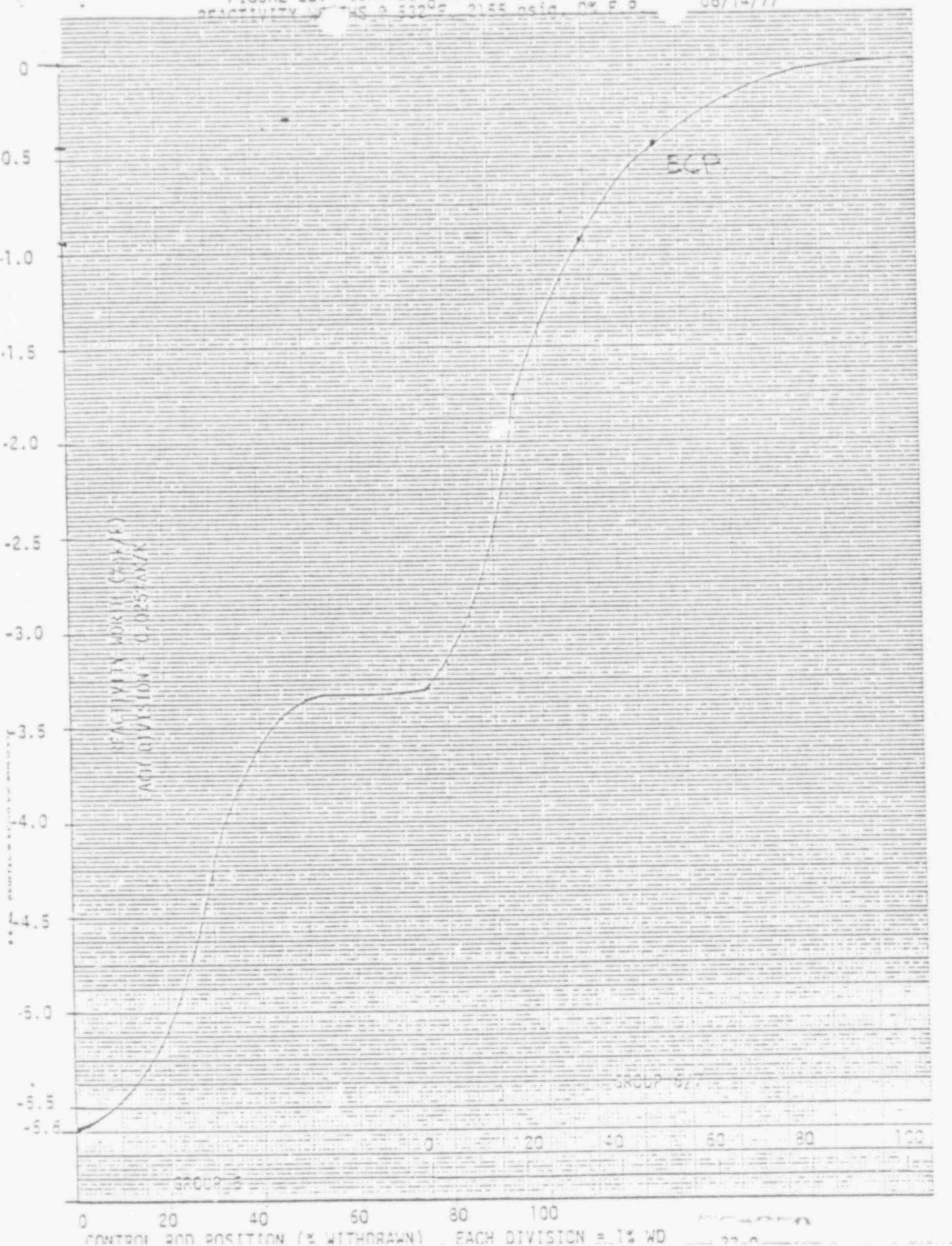


Table 1. MISCELLANEOUS — 10 CFR 20 Appendix B

Material	Half-Life	Energy MEV per disintegration	Table I Col I Air μc/ml	Table I Col II Water μc/ml	Table II Col I Air μc/ml	Table II Col II Water μc/ml
A-41	1.83 h	1.3	2×10^{-6}		4×10^{-8}	
Co-60	5.3 y	2.5	3×10^{-7}	1×10^{-3}	1×10^{-8}	3×10^{-5}
I-131	8.1 d	0.36	9×10^{-9}	6×10^{-5}	1×10^{-10}	3×10^{-7}
Kr-85	10.8 y	0.04	1×10^{-5}		3×10^{-7}	
Ni-65	2.5 h	0.59	9×10^{-7}	4×10^{-3}	3×10^{-8}	1×10^{-4}
Pu-239	2.4×10^{4} y	0.008	2×10^{-12}	1×10^{-4}	6×10^{-14}	5×10^{-6}
Sr-90	28 y		1×10^{-9}	1×10^{-5}	3×10^{-11}	3×10^{-6}
Xe-135	9.2 h	0.25	4×10^{-6}		1×10^{-7}	
Any single radionuclide with $T_1 > 2$ hr which does not decay by α or spontaneous fission			3×10^{-9}	9×10^{-5}	1×10^{-10}	3×10^{-6}

Neutron Energy (MEV)	Neutrons per cm^2 equivalent to 1 rem	Average flux to deliver 100 mrem in 40 hours
Thermal	970×10^{6}	670
0.02	400×10^{6}	280
0.5	43×10^{6}	30
10	24×10^{6}	17

LINEAR ABSORPTION COEFFICIENTS (cm^{-1})

Energy (MEV)	Water	Concrete	Iron	Lead
0.5	0.090	0.21	0.63	1.7
1.0	0.067	0.15	0.44	0.77
1.5	0.057	0.13	0.40	0.57
2.0	0.048	0.11	0.33	0.51
2.5	0.042	0.097	0.31	0.49
3.0	0.038	0.088	0.30	0.47

FORMULAS AND CONVERSIONS

$$r/hr = \frac{6CE}{d^2} \qquad I_x = I_o e^{-\mu x} \qquad I_x = \frac{I_o}{d^2} \qquad N = N_o e^{-\lambda t}$$

$$T_{\frac{1}{2}} = \frac{0.693}{\lambda} \qquad p = \frac{\Sigma \phi V}{3\times10^{10}} \qquad \rho = \frac{1}{T\kappa} + \Sigma \frac{\beta_i}{1-\lambda T} \qquad T = \frac{\beta - \rho}{\lambda \rho} \qquad T = \frac{l}{\rho - \beta}$$

$$T = \frac{l^*}{\rho} + \frac{\beta - \rho}{\lambda \rho}$$

1 gal = 3.78 liters
1 in = 2.54 cm
1 yr = 3.15×10^{7} sec.

1 kg = 2.2 lbs
1 gm/cm^3 = 62.4 lbs/ft^3
1 ft^3 = 7.48 gal

$N_A = 6.02\times10^{23}$
$\pi = 3.14$
$e = 2.72$

76-269

POOR ORIGINAL

A1 a) The term delayed neutron fraction is used to denote the fractional number of neutrons produced that are delayed. Immediately upon the fission event most neutrons are produced. These neutrons are known as prompt neutrons. There are a fraction of neutrons that are produced some time after the fission event from the decay of fission products. These fission products are known as delayed neutron precursors and the neutrons they produce are delayed neutrons.

A1 b) The delayed neutron fraction decreases over core life. This is due to the fact that over core life U_{235} is burned up and Pu_{239} is built-up. Pu_{239} is in the decay chain for U_{238} absorption of a neutron. $U_{238} + {}_0n^1 \rightarrow U^{239} \longrightarrow Pu^{239}$. Pu_{239} is a fissionable isotope and contributes significantly to neutron production over core life. At the beginning of life β is due only to U_{235} and U_{238}. At the end of life it is due to U_{235}, U_{238} and Pu_{239}. Typical values are as follows.

$$\beta_{BOL} = (.0064)(.93) + (.0157)(.07) = (.00595) + (.0012)$$
$$\beta_{BOL} \cong .0071$$

At the end of life the contribution of Pu^{239} has to be taken into effect along with the burn up of U^{235}. So that at EOL β_{eff} is ~~equal~~ to

$$\beta_{EOL} = (.0064)(.50) + (.0151)(.07) + (.0021)(.43).$$

$$\beta_{EOL} = .0032 + .0012 + .0010$$

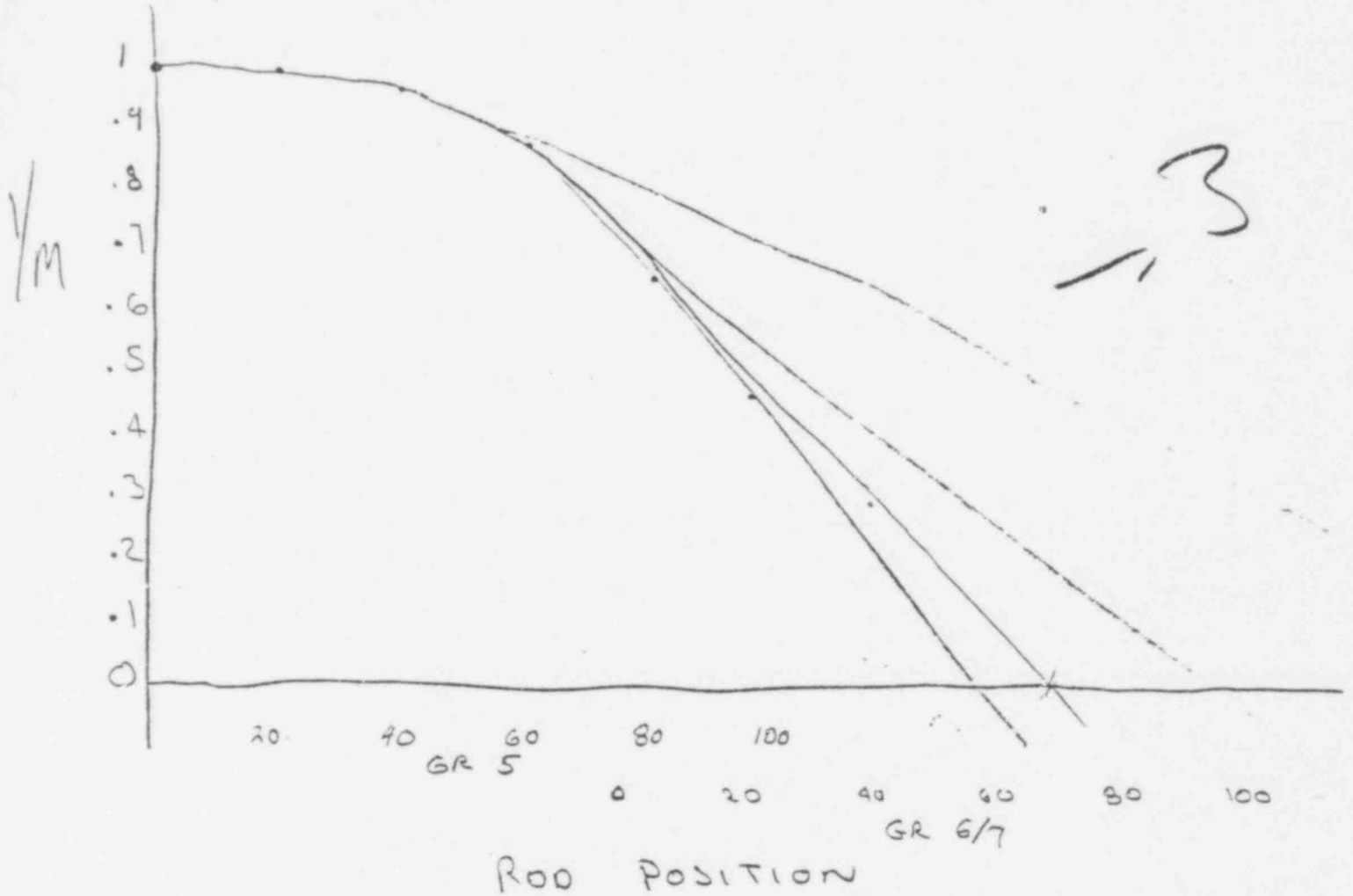
$$\beta_{EOL} \cong .0053$$

-3

A2 a) The data required to plot a 1/M graph is rod position and count rate. (Sea rod with initial C.R.)

A2 b) The initial countrate is known and is used with the corresponding rod position and M is determined. As you approach critical M goes to infinity. It is hard to determine when you approach ∞, so a 1/M plot is used. As you approach critical and M → ∞ 1/M approaches zero. After each rod withdrawal counts are allowed to stablize and the count rate corresponding to this rod height is recorded and devided into the initial count rate. As counts increase with each addition of positive reactive resulting from rod withdrawal 1/M approaches 0 indicating that the reactor is approaching critical. A typical curve appears on the following page. If you extrapoliate the last three points on the curve the critical rod position can be estimated. TMI uses the 1/M to predict the approach to critical and to check this prediction against estimated critical rod height.

76 263


Y/M
1
.9
.8
.7
.6
.5
.4
.3
.2
.1
0
20
40
60
80
100
GR 5
0
20
40
60
80
100
GR 6/7
ROD POSITION

1.3

76 264

A3 a) at 2×10^{-11}

level at 1×10^{-8}

SUR = .8 DPD

$P = P_0 10^{SUR\,T}$

$\frac{1 \times 10^{-8}}{2 \times 10^{-11}} = 10^{SUR\,t}$

$\log \frac{1 \times 10^{-8}}{2 \times 10^{-11}} = SUR^t \log 10$

$\log 5 \times 10^2 = .8t\,(1)$

$t = \frac{2.7}{.8}$

$t = 3.4$ min

A3 b) 1 DPM

$P_0 = 10^{-8}$

$P = P_0 10^{SUR\,t}$

$P = 1 \times 10^{-8}\, 10^{1\,DPM \cdot 5\,min}$

$P = 1 \times 10^{-8} \cdot 1 \times 10^{5}$

$P = 1 \times 10^{-3}$

76-265

2

shall on P. Jump

3

The amount of reactivity added to initiate a 1DPM startup rate is sufficient to increase reactor power to 1×10^{-3} amps neglecting any reactivity feedback due to moderator temperature or doppler coefficient. During rod motion the startup rate will increase sharply to some value greater than 1DPM. When rod motion stops the SUR will level off on a stable 1DPM increase. This

increase in power will continue until negative reactivity from inserting rods or negative reactivity due to the doppler and moderator coefficients begins to turn power. The point of adding heat is reached at approximately 5×10^{-7} amps. From this point doppler and moderator coffic will begin to add negative reactivity. Doppler due to fuel temperature increasing which results in the resonance absorption peaks of U_{238} and Pu_{240} broadening and absorbing more neutrons. The increasing temperature of the moderator causes the density of the moderator to decrease. This decreasing density allows the neutron to travel farther prior to reaching thermal energies. The longer the time the neutron spends in the fast and epithermal regions the more chance it has to be absorbed or leak out of the core. The combined effect of the moderator and doppler negative reactivity feed back will turn reactor power and cause it level off at some value. The value is dependent on many things (core age, soluble poison concentration, the value of doppler & moderator coefficients). For this large increase in positive reactivity (enough to give a 10PM start up rate) power will level off in the power range. Power will start to turn shortly after reaching the point of adding heat

and the start-up rate will begin to decrease. Power will continue to increase at a slower rate until negative reactivity feedback from doppler and moderator have add sufficient reactivity to over come the rod withdrawal and then power will level off.

A3 c. If a Rx trip accured at 1×10^{-6} amps. You would immediately see a prompt drop in power due to the fact that prompt neutron production would immediately ~~stop~~. The power would ~~then~~ level off on a ~80 sec period due to the longest lived delayed neutron precursor Br^{87}. Power would continue to decrease on the -1/3 DPM SUR into the source range.

A4 a) Equilibrium Xe is the core condition when Xe production (from the decay of I^{135} and from fission yield) equals Xe burn-up and Xe decay. Xe is produced in a small quantity from the fission process. Most of the Xe in the core results from the decay of iodine^{135} per the following equation

$$Te^{135} \xrightarrow[\tau = 2\,min]{\beta^-} I^{135} \xrightarrow[6.7\,hr]{\beta^-} Xe^{135} \xrightarrow[9.2\,hr]{\beta^-} Cs^{135}.$$

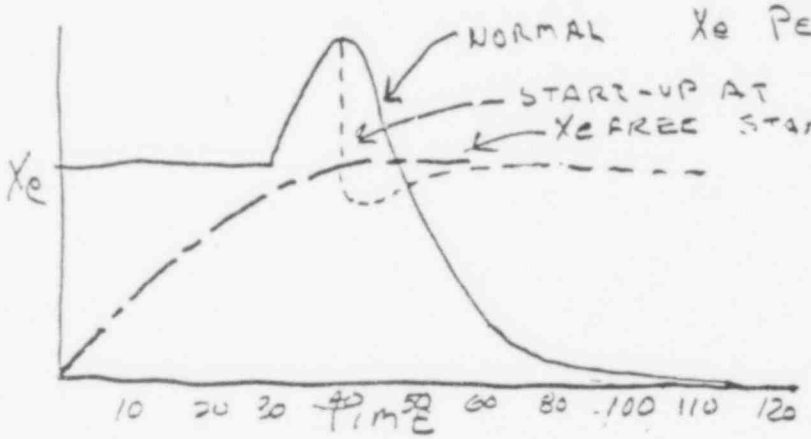
Xe is removed from the core by burn-up $Xe^{135} + \sigma N' \rightarrow Xe^{136}$ stable and by natural decay to Cs^{135}.

A4 b) Peak Xe develops after reactor shutdown due to the fact that Xe burn-up is stopped and the fact that the I^{135} present in the core at t=0 decays to Xe faster than X^{135} decays to Cs. Xe production from fission also stops but more Xe is burned-up (decay of I^{135}) than produced from fission. The net effect is that Xe builds up due to the I^{135} present in the core at the time of shutdown. 3

76-288

A4 c) If the reactor is started up after shutdown while Xe is peaking then, more positive reactivity has to be added initially to over come the Xe peak. After the peak

When Xe begins to decay the extra reactivity that was added to over come Xe has to be removed at the rate corresponding to Xe decay.
Starting up after a week shutdown would be starting up with a Xe free core. Reactivity would have to be added to increase power; But the Xe buildup to equilibrium would be slow and could be compensated for ~~with~~ no trouble.



this is part of off-peak Con

A5 In addition to being used as a reactor coolant, water is used as a moderator to thermalize fast neutrons, a shield and a neutron reflector.

B1 a) i) Inside the compartment

1. reactor coolant pump, motor COOLERS

ii) outside the reactor compartment

*1. instrument air compressors

*2. RB pumps & motors

*3. MU pumps and motors

4. RC evaporator

5. seal return coolers

*6. build spray pump motors

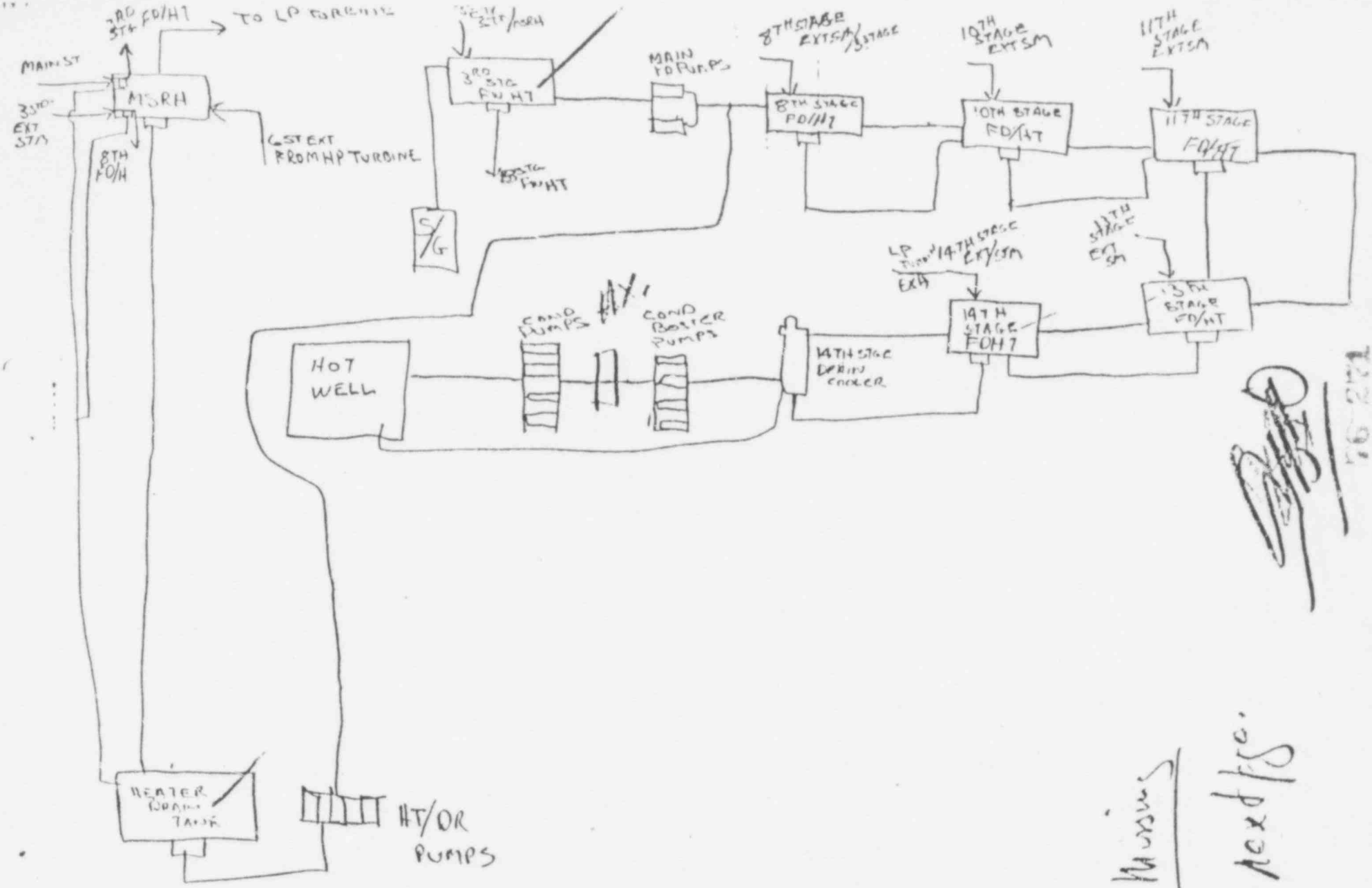
a) 3

-.8

b) (2)

-.4

76-270


3RD STG FD/HT
TO LP TURBINE
MAIN ST
MSRH
3RD STG EXT STM
8TH FD/H
6ST EXT FROM HP TURBINE
3RD STG FW HT
MAIN FD PUMPS
8TH STAGE FD/HT
10TH STAGE FD/HT
11TH STAGE FD/HT
S/G
HOT WELL
COND PUMPS
COND BOOSTER PUMPS
14TH STAGE DRAIN COOLER
14TH STAGE FDHT
15TH STAGE FD/HT
HEATER DRAIN TANK
HT/DR PUMPS

B2

B3

B4

B3 a)

1. Unit 1 motor driven pump ~ 2500 ~ 600 GPM
2. Unit 1 diesel driven pump "
3. Unit 1 RWP house diesel "
4. Unit 2 diesel pump "

the unit 1 motor driven pump starts at 90 psi header pres
unit 1 diesel pump starts at 85 psi " "
unit 2. diesel pump starts at 80 psi " "
unit 1 RWP house diesel is backup pump
there is also a small motor driven pump in unit 1 which only supplies daily make-up,

b) The wet pipe sprinkler sys has water up to the sprinker at all times. Actuation only opens the sprinkler and allows the water to flow The deluge system is filled with nitrogen which holds a valve shut actuation allows the A_2 pressure to be bleed off and the valve to open which then admits water into the system

[illegible] 5

B-4 a) RCP seal bleed off leakage from code reliefs, electro matic relief discharge & leakage system leakage from unit

b) The tank is equiped with a relief valve which lifts at 150PSI & a blowout disk set at 200 PSI to prevent overpressurizati. The tank is maintained with approximately 76 ft level to allow for steam if relief lift to be condensed by the water. The tank is also equipped with a leakage cooling system which recirc contents through a cooler to prevent the tank from over heating. Base drip

c) The excess liquid can be discharged to the liquid rad waste system where it can be directed to the MWHT, ~~RCBT~~, RC EVAP, MU System on unit 1.

76-273

U. S. NUCLEAR REGULATORY COMMISSION

SENIOR REACTOR OPERATOR LICENSE EXAMINATION

Facility: Three Mile Island - Unit 2
Reactor Type: PWR - B&W
Date Administered: 8/18/77
Examiner: J. J. Buzy
Applicant: 1532

INSTRUCTIONS TO APPLICANT:

Use separate paper for the answers. Staple question sheet on top of the answer sheets. Points for each question are indicated in parenthesis after the question.

Category Value	% of Total	Applicant's Score	% of Cat. Value	Category
14	20	11.7	84	H. Reactor Theory
14	20	10.8	77	I. Radioactive Materials Handling, Disposal and Hazards
14	20	12.4	89	J. Specific Operating Characteristics
14	20	12.6	90	K. Fuel Handling & Core Parameters
14	20	11.4	81	L. Administrative Procedures, Conditions and Limitations
70		58.9		Totals

Final Grade: 84 %

76-274

H. REACTOR THEORY (14)

H. 1. Two minutes after a reactor trip from full power, a large number of neutrons are still present in the core. List these neutrons and give their origin. (1.5)

H. 2. Assume your reactor is on a 1 DPM startup rate at 5×10^{-9}, group 6/7 is at 15% withdrawn.

a) What should be the rod position when the operator levels off at 10^{-8}? Show all work, include any assumptions. (2.0)

b) With maximum rod speed, how long should the operator move the rods to level off at 10^{-8}? Explain how you arrived at your answer. (1.5)

c) If power were allowed to increase, how far do you expect the power level to rise? Include any assumptions. (1.0)

d) Will the same reactivity addition produce the same S.U.R. during core lifetime? Explain. (1.5)

H. 3. Refer to figure 2B. Control rod groups 5-7 reactivity worth at 532°F.

a) Sketch the approximate reactivity worth of group 5-7 when the temperature is 582°F. (0.5)

b) Explain why the curve will change. (2.0)

H. 4. Refer to question K3. Assume the individual responsible for taking the count rate reads and records the data immediately after the fuel element is inserted into the core. Would the resulting plot be more or less conservative? Explain your answer. (2.0)

H. 5. One of the limiting conditions for operation at your station is the value of the moderator temperature coefficient. Is this value of concern early or at the end of core lifetime? Explain the reason(s) for your answer. (2.0)

I. RADIOACTIVE MATERIALS HANDLING, DISPOSAL AND HAZARDS (14.0)

I. 1. Periodically, during power operation, it is necessary to purge the reactor building. Describe the procedures and limitations that you, as shift supervisor, are responsible for during the purge. Include the following:

a) Administrative requirements, including individuals who approve the release. (1.0)

b) What monitors are required to be in service? (1.0)

c) Why is it necessary for the fire protection system to be operational during the release? (0.7)

d) Maximum reactor building pressure prior to release. (0.3)

I. 2. A smear is taken on a calibrated pressure gauge which was used in the waste disposal area.

Smear Area - 100 CM^2 Background - 16 CPM
5 minute count - 900 Scaler efficiency - 15%

a) Determine if the gauge can be returned to the instrument shop. (1.5)

b) Discuss how you would take a smear(s) for this gauge. (1.0)

I. 3. Assume a reactor coolant leak develops in the reactor building.

a) What are limits on reactor coolant system leakage as a function of location or identity of the leakage? (1.5)

b) Assume the leakage collects in the reactor building sump. Describe in sequence what happens to the water from the sump to its ultimate disposal. (1.5)

c) Prior to entering the reactor building to search for the leak, what installed monitoring system(s) can be used by the inspection team to evaluate the levels of radiation? (1.5)

I. 4. An operator has dropped a set of keys on top of a spent fuel element in the storage pool. He wants to fabricate a tool made of ten foot long, one inch diameter sections of electrical conduit joined by couplings. In addition, he has had a hook welded to a pipe cap to join with the end of the conduit sections which will be used to pick up the key ring. Should you allow him to assemble the tool and use it? Explain why or why not. (2.0)

Section I continued next page

I. 5. The maximum activity contained in one gas decay tank is limited to _____.

a) What is the limit and who is responsible to assure this limit is not exceeded? (1.0)

b) What is the reason for this limit? (1.0)

76 277

J. SPECIFIC OPERATING CHARACTERISTICS (14)

J. 1. Lithium hydroxide and hydrazine are two chemicals that are added to the reactor coolant system.

a) Discuss when it may be necessary to add these chemicals. (1.5)

b) What precautions must be observed when handling these chemicals? (1.0)

J. 2. During the unit heatup, one of the steps requires a hard bubble degasification when the coolant temperature reaches 350°F.

a) Outline how this step is performed. (1.0)

b) What would be the consequences if this step was not performed? Assume power operations with steady state and transient conditions. (1.5)

J. 3. About two years ago, TMI Unit One went critical during a plant heatup, while the RC temperature was 505°F.

a) What procedural steps were modified to prevent a repeat incident? (1.5)

b) What is the L.C.O. for minimum R.C. temperature for criticality and what is the basis for this L.C.O.? (1.5)

J. 4. When operating from 50 to 100% rated power with 4 RC pumps running, the staff is required to maintain cold leg differential temperatures to less than 5°F.

a) How does the operating staff comply with this limitation? Include controls the staff must manipulate. (1.5)

b) Why is this condition required? (1.0)

J. 5. Assume a reactor startup is in progress and within 5% of rod position for criticality. At the same time, an I.C.S. representative is "fine tuning" the control system. Suddenly, all turbine bypass valves go full valves open. Assume (i) one main feed pump in service and (ii) R.C. temperature decreases at 30°F/min.

a) What protective device(s) should respond to trip the reactor if no action is taken? (0.3)

b) What immediate action is required by the operating staff? (0.7)

Section J continued on next page

J. 5. c) Assume no action by the operating staff. What systems will respond if the turbine by-pass valves cannot be shut. Attempt to show if the plant systems will or will not terminate this transient without operator action. (2.5)

K. FUEL HANDLING & CORE PARAMETERS (14)

K. 1. Listed below are several fuel handling devices and maneuvers. List and explain the purpose of the interlock(s) during each maneuver.

a) Control rod handling mechanism while inserting and withdrawing control rods. (1.0)

b) Rotation of the fuel transfer baskets and the main fuel or spent fuel storage handling bridge. (1.0)

c) Fuel grapple mechanism and hoists when inserting or removing fuel in the core. (1.0)

K. 2. Assume a malfunction on one of the safety rods in group 2 results in a dropped rod. The reactor is at rated power and the reactor demand station is in manual.

a) What immediate action is required by the operator? (1.0)

b) Why is the above action necessary? (1.0)

c) Assume trouble shooting cannot begin for several hours, would the axial power imbalance and/or the quadrant power tilt become the limiting condition for operation? Explain your answer. (1.5)

K. 3. Initial fuel loading at most PWRs develops as a "slab". The source is located in the center and the fuel is loaded outward (see figure K-3). Temporary chambers (A&B) are located in the vessel and subcritical multiplication plots are made for each section of the slab loading.

a) Sketch the expected plot of subcritical multiplication for each chamber. (1.5)

b) Explain any difference in the plots. (2.0)

K. 4. During the initial power escallation and immediately following a control rod interchange, the rate of power increase is limited to 3%/hr. between 75 to 90% power; also, a 5 hour hold is required at 90% power.

a) What are the normal maneuvering rates between 75 to 90% power? Provide transient and controlled conditions. (0.5)

b) Why is it necessary to perform a control rod interchange and why are power increases reduced to 3%/hr? (2.0)

K. 5. Attached is figure 1A, Core Excess Reactivity, for your facility. Explain the shape of the curve during the first 100 EFPD. (1.5)

L. ADMINISTRATIVE PROCEDURES, CONDITIONS AND LIMITATIONS (14)

L. 1. If any safety limit is exceeded, the shift supervisor is required to notify the station/unit superintendent and have the reactor in hot standby conditions within one hour.

a) What parameters and limits are considered in the safety limits for your station? (1.5)

b) What instrumentation in the reactor protecting system should prevent reaching a safety limit? (1.5)

c) What are the plant conditions for hot standby? (0.5)

L. 2. Title 10 CFR Part 55, Operators Licenses, defines an operator as any individual who manipulates a control of a facility.

a) Define control as it applies to this definition and give two examples. (1.0)

b) Is it necessary to be a licensed operator to manipulate a control? Explain. (0.5)

c) Assume an operator was injured during a motorcycle accident. He is disabled for 5 months. When can he return to normal duties? (1.0)

L. 3. When a shift is relieved, a number of items must be completed.

a) What shift records are maintained? (1.5)

b) What conditions are required in order for an entire shift to be relieved? (1.0)

c) List the shift organization required by the technical specifications during (i) fuel loading and (ii) plant startup. (1.5)

L. 4. Assume you are the shift supervisor in charge of your shift. For each condition below provide (i) immediate and subsequent action and (ii) the reason for your action, including any administrative limits.

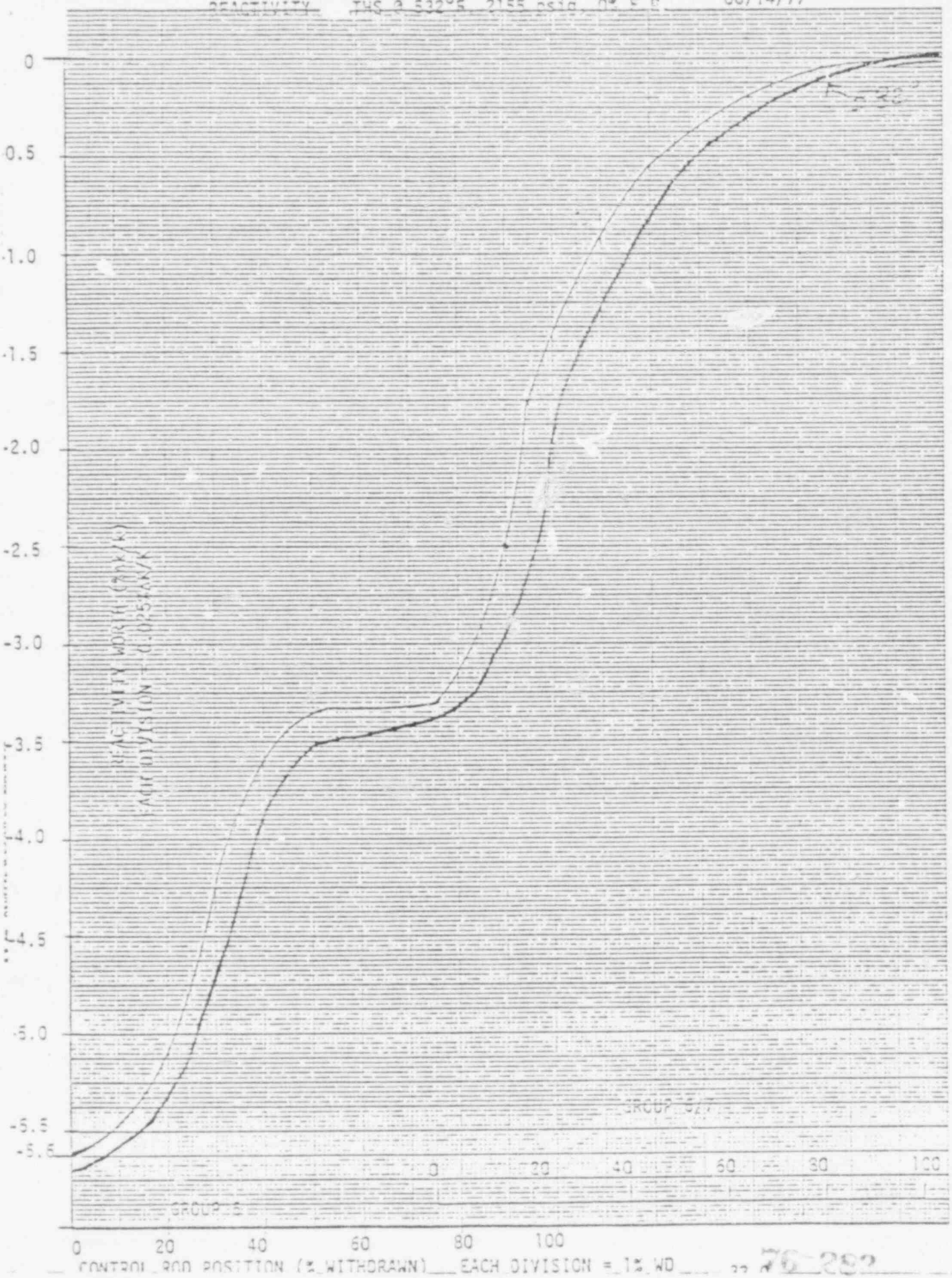
a) The unit is shutdown for OTSG tube inspection. You are informed that the sodium hydroxide tank was inadvertantly lined up to the decay heat pumps. About half of the tank volume is left. (1.5)

b) The plant is at rated power during the winter. The auxiliary operator informs you that the vent(s) on the BWST tank seem to have frozen due to condensation and ice formation. (1.0)

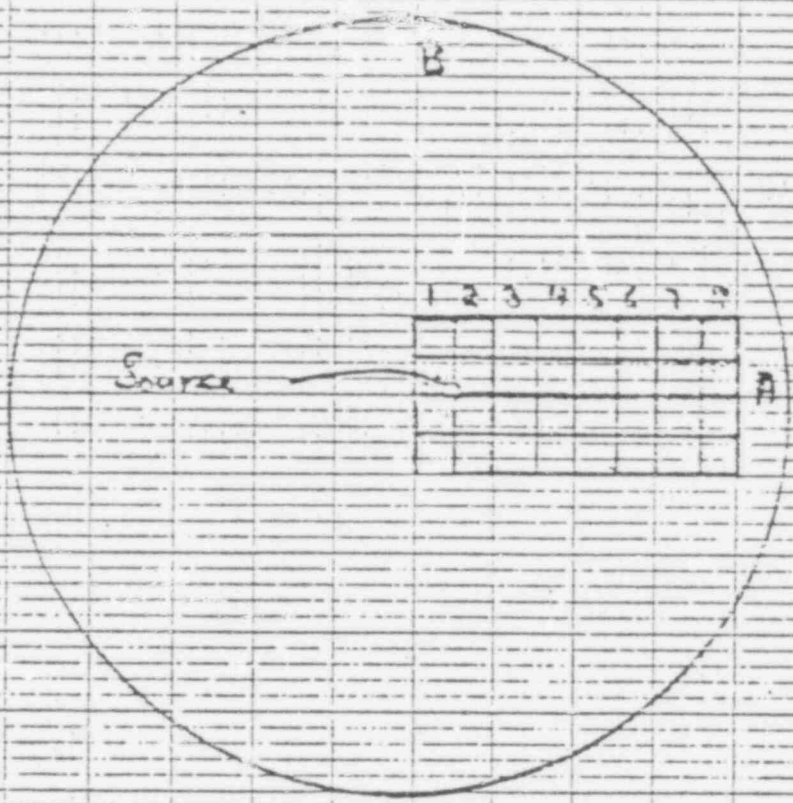
Section L continued on next page

L. 4. c) During a test of the containment isolation valves, one of the valves begins to leak at the packing gland. The test engineer suggests he can "back seat" the valve by holding in the open relay with an insulated screwdriver at the motor control center. (1.5)

FIGURE 2R- CONTROL ROD GROUPS 5,6 & 7 Revision 0
REACTIVITY THS 0.532°F 2155 psig 0% F.P. 08/14/77



76-282



KEUFFEL & ESSER CO. 10 X 10 TO ¼ INCH 40 1020

K3

76-251

FIGURE 1 : CORE EXCESS REACTIVITY VS EFPD

2103-1.9
Revision 0
06/13/97

Table 1. MISCELLANEOUS — 10 CFR 20 Appendix B

Material	Half-Life	Energy MEV per disintegration	Table I Col I Air μc/ml	Table I Col II Water μc/ml	Table II Col I Air μc/ml	Table II Col II Water μc/ml
A-41	1.83 h	1.3	2×10^{-6}		4×10^{-8}	
Co-60	5.3 y	2.5	3×10^{-7}	1×10^{-3}	1×10^{-8}	5×10^{-5}
I-131	8.1 d	0.36	9×10^{-9}	6×10^{-5}	1×10^{-10}	3×10^{-7}
Kr-85	10.8 y	0.04	1×10^{-5}		3×10^{-7}	
Ni-65	2.5 h	0.59	9×10^{-7}	4×10^{-3}	3×10^{-8}	1×10^{-4}
Pu-239	2.4×10^{4} y	0.008	2×10^{-12}	1×10^{-4}	6×10^{-14}	5×10^{-6}
Sr-90	28 y		1×10^{-9}	1×10^{-5}	3×10^{-11}	3×10^{-6}
Xe-135	9.2 h	0.25	4×10^{-6}		1×10^{-7}	
Any single radionuclide with $T_{\frac{1}{2}} > 2$ hr which does not decay by α or spontaneous fission			3×10^{-9}	9×10^{-5}	1×10^{-10}	3×10^{-6}

Neutron Energy (MEV)	Neutrons per cm^2 equivalent to 1 rem	Average flux to deliver 100 mrem in 40 hours
Thermal	970×10^{6}	670
0.02	400×10^{6}	280
0.5	43×10^{6}	30
10	24×10^{6}	17

LINEAR ABSORPTION COEFFICIENTS (cm^{-1})

Energy (MEV)	Water	Concrete	Iron	Lead
0.5	0.090	0.21	0.63	1.7
1.0	0.067	0.15	0.44	0.77
1.5	0.057	0.13	0.40	0.57
2.0	0.048	0.11	0.33	0.51
2.5	0.042	0.097	0.31	0.49
3.0	0.038	0.088	0.30	0.47

FORMULAS AND CONVERSIONS

$$\text{r/hr} = \frac{6CE}{d^2} \qquad I_x = I_o e^{-\mu x} \qquad I_x = \frac{I_o}{d^2} \qquad N = N_o e^{-\lambda t}$$

$$T_{\frac{1}{2}} = \frac{0.693}{\lambda} \qquad p = \frac{\Sigma \phi V}{3\times10^{10}} \qquad \rho = \frac{1}{T K} + \sum \frac{\beta_i}{1-\lambda T} \qquad T = \frac{\beta - \rho}{\lambda \rho}$$

$$T = \frac{\ell^*}{\rho} + \frac{\beta - \rho}{\lambda \rho} \qquad T = \frac{\ell}{\rho - \beta}$$

1 gal = 3.78 liters
1 in = 2.54 cm
1 yr = 3.15×10^{7} sec.

1 kg = 2.21 lbs
1 gm/cm^3 = 62.4 lbs/ft^3
1 ft^3 = 7.48 gal

$N_A = 6.03\times10^{23}$
$\pi = 3.14$
$e = 2.72$

76-258

H-1 After a reactor trip the neutron level decreases sharply due to the decrease in prompt neutron generated in the sustained chain reaction. There are a number of neutrons still being generated from the decay of the fission products. The primary source of these neutron are the delayed neutron precursors which generate neutrons in their decay scheme. There are a small number of neutrons generated from spontaneous fissions and from the installed neutron source. + photo units 5

H-2. a) $T = \frac{26}{SUR} = \frac{26}{10PM}$ assume $\bar{\beta}_{eff} = .0069$

$T = 26$ sec. $\lambda = .08$

$T = \frac{\beta - \rho}{\lambda \rho}$ rod worth = 2.5×10^{-4} $\Delta k/k/\%$

$\rho = \frac{\beta - \rho}{\lambda T}$

$\lambda T \rho + \rho = \beta$

$\rho(\lambda T + 1) = \beta$

$\rho = \frac{\beta}{\lambda T + 1}$

$\rho = \frac{.0067}{.0826 + 1} = \frac{6.9 \times 10^{-3}}{2.08 + 1} = \frac{6.9 \times 10^{-3}}{3.08}$ 76-257

$\rho = 2.24 \times 10^{-2}$ / 1% reactivity associated with 10PM starting rate

This much negative reactivity is required to be inserted to level power

reactivity − reactivity due to rod insertion = 0

2.24×10^{-3}, $\Delta K/K$ − $\underline{.025\% \ \Delta K/K/\%}$ (% inserted) = 0

$$\% \text{ inserted} = \frac{2.24 \times 10^{-3}}{.25 \times 10^{-3}} = 8.96 \cong \underline{9\% \text{ insertion}} \text{ requ}$$

rod position at 10^{-8} amps = 15% − 9% or

$\boxed{6\% \text{ on group } 6/7}$ no mark use rod number. −5

b) total rod travel = 144 in = 100% which means 1.44 in/% of rod travel

max rod speed = $30 \frac{in}{min}$ or $\frac{30 \frac{in}{min}}{1.44 \frac{in}{\%}} = 20.8 \frac{\%}{min}$

knowing rod withdrawal speed and % withdrawal needed to overcome ex ℓ can calculate W.T.

$$\text{withdrawal time} = \frac{9\%}{20.8 \frac{\%}{min}} = .433 \text{ min} : \underline{60 \text{ sec}}$$

No [illegible] ~3 min

W.T. = 25.98 sec = 26 sec.

$\boxed{W.T = 26 \text{ sec}}$

76-288

c) If power were allowed to increase it would be leveled off due to negative reactivity feedback due to doppler and moderator coefficients. assume at 532°F at startup and reactor begins to heat up to 582°F. Assume $\alpha_m = -1 \times 10^{-4} \ \Delta K/K/°F$

assume no effect due to doppler than α_M would have to add $-2.24\times10^{-3}\Delta K/K$ or $\frac{-2.24\times10^{-3}}{-1\times10^{-4}} = 22.4$ rise in temperature. Knowing that at 5% Rx power $T_{AV} \cong 534$ at 10% $\cong 565$ and at 15% it equals 582 that $532 + 22.4 = 554.4^\circ$ so without doppler power should level at approximately 8%. Concerning doppler power will level at some value less than 8% around 5%

αD_1 is no. for contribution ~ 18%

Power coeff ~.01%/% Pwr

d) As over core life β_{eff} decreases due to the burn-up of U_{235} and the buildup of Pu_{239}. This decrease in β_{eff} makes the reactor more responsive. So that at the end of core life the same reactivity addition will cause a higher S.U.R.

76-289

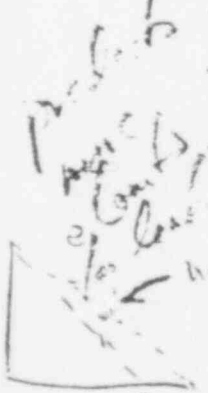
H3 a) see figure 2B

b) As moderator temperature increases the density of the moderator decrease. This decrease in moderator density results in the neutrons spending more time in the fast and epithermal energy levels prior to being thermalized by the moderator.

The more time the neutron spends in the fast and epithermal region the more the chance the it will be absorbed or leaks out of the core and not reach thermal energies. Also with an increase in temperature and decrease in density some of the water is displaced from the core. When this water is displaced so is some of the soulble poison. The displacement of soulble poison decreases the competition for neutron absorption. This decrease in competition for neutron absorption & the fact that the decrease in density allows the neutron to spend more time slowing down makes the control rod more likely to absorbe a neutron, thus increasing its worth.

76-290

H-4 The resulting plot would be less accurate if the reading were taking immediately after the fuel element is inserted into the core. Reading the indicator immediately would not allow for the full effects of sub-critical multiplication to take effect. Sub critical multiplication is neutron population due to a neutron source and fuel in a sub-critical reactor. If the operator waited for counts to stabilize after the fuel element were inserted



he would have a true indication of the neutron population in the core. If he does not let the count rate stabilize (reflect the effect of subcritical multiplication) than he does not have a true indication of neutron population and his plot will be less conservative.

76-291

H-5 Tech specs requires that the moderator temperature coefficient be less positive than $.5 \times 10^{-4}$ whenever reactor power is less than 95%, more negative than 0.0×10^{-4} when ever power is greater than 95% and less negative than 3×10^{-4}. This value is of concern early in core life because the moderator coefficient is slightly positive. This due to the high soluble poison concentration which adds a positive component to the moderator coefficient. As the core ages and Boron concentration is reduced this positive component of α_m decreases and α_m becomes more negative over core life. While the α_m is positive (should be negative by the completion of physic testing, before increasing to rated power) it provides a positive reactivity feedback. This is not desirable. A negative

feedback will add in controling the reactor and lessen the effects of a certain accidents while a positive α_m will not aid in turning reactor power on an increasing temperature. The value of α_m at end of core life is a concern because the more negative it is the more responsive the reactor is to changes in temperature and also vary because the is a maximum limit placed on this value by tech specs.

76-292

I

I-1 The shift foreman is responsible for getting the release forms approved by the shift supervision and the HP foreman. The compartment is sampled and the hydrogen and activity level determined. The radiation monitor upstream and downstream of the filters have to be operable and so does the stack monitor. When the reacto-building pressure is less than 1/5 PSI. The purge valves are opend and the exhaust fans started. The radiation monitors are monitored during the purge. Shift foreman is responsible to maintaining the length of time the purge valve are open while the reactor is operating. This is limited to 90 hr/year. The fire protection system must be in operation because the hydrogen in the Rx build presents a fire and explosion hazard.

I-2 a) smear area = 100 cm²

5 minute count = 900

Background = 16 CPM

ε = 15%

CPM = 900/5 = 180 CPM − 16 CPM background

CPM = 164

$\frac{C}{D} = E$ C = CPM, E = efficiency, D = DPM

$D = \frac{C}{E} = \frac{164}{.15}$

D = 1093, DPM

The gage in above the limit for uncontrolled release. It must be decontaminated before it can return to the instrument shop.

b) You take a swipe with a piece of filter cloth. taking care to cover approximately 100 cm². You then count the dose rate from the cloth.

76-294

I-3 a) 1. zero pressure boundary leakage

2. 28 gpm identified leakage - leakage into closed system, or leakage that is identified and does not interfere with leak rate detection and is not a pressure boundary leak. And S/G tube leakage

3. 1 gpm unidentified leakage

4. 1 gpm S/G tube leakage

5. 8 gpm ~~controlled~~ leakage (reactor coolant pump seal leakoff)

b) The water from the Rx building is pumped by the Rx Bld sump pumps to the MWHT from the MWHT the liquid radwaste system can send it to the unit 1 MWHT or to the unit 1 ~~main~~ evaporator.

c) HP-R-227 reactor building gas & particulate monitor HP-R-225 & 226 on the R.C. ventilation return B.G. unit.

I-4 No, you should not let the man use the tool to retrive his keys. The man could possibly damage the top of the fuel assy with the tool or damage the fuel cladding on the rods. He could drop the tool into the pool and then there would be 2 objects to retrive. The tool could come apart while he was using it and a part fall into pool. Also the keys are almost certainly contaminated and have a high radiation dose associated with them. Even if the worker could get them out they would present a radiation hazard if he/she would refer this problem to the SS and H.P sup. so a detailed instruction could be written for retrieving and disposal of the keys. — 1.0

Steering from tube

I-5

1) 2000 curies max. The H.P. supervisor is responsible to insure that the limit is not exceeded. — 1
+ From vault

b) Based on releasing this amount and not exceeding year limit or limits to off site personal.

J

J-1

a) Lithium hydroxide is used to control Ph and hydrazine is used to control oxygen content of the reactor coolant at temperatures less than 400°. Lithium hydroxide is added to the coolant system initially to establish a Ph between 4 and 8. It is added during operation if Ph is out of spec and the lithiated resin bed can not maintain proper band. hydrazine is add when reactor coolant temp is less than 400° and oxygen concentration is greater than .1 ppm.

b) Both chemicals can cause damage to the eyes and skin irritations. You should wave a face shield, gloves and and apron when handling the chemicals. [illegible]

76-297

J-2 a) The pressurizer steam space is vented by lining up the sample valve to the sample sink and degasing is accomplished. ~~[illegible]~~. MUT [illegible]

b) If the pressurizer is not degassed and gases are allowed to accumulate in the P3r than P3r operation at power is effected. The

normal PZR steam bubble is reduced in size by the accumulated gases. Therefore, on a power transient a pressure insurge causes the spray valve to stay on longer to limit the pressure rise. The spray flow does not cause the gas bubble to condense like the steam bubble does.

J-3 a) The operator has to plot a 1/M plot to determine when he approaches the critical position. Also the procedure requires that TAVE for the lowest loop be greater than 525 and this value verified within 15 min of attaining critical.

b) LCO for criticality is 525°F. The basis for this temperature was 1) PZR bubble, 2) lowest temperature for which moderator analized for accidents 3) temperature instruments in operating range and vessel. NOTE 298

J-4 a) The cold leg differential temperature is maintained within 5°F normally by the ICS ΔTC ckt. If it required to be done manually. Then the operator must take manual control of the feedwater demand stations and manually ratio feedflow to each generator to

maintain the limit.

b) The 5°ΔTC limit is placed on the reactor to limit flux tilt that would ~~result~~ from a large imbalance of cold water on one side of the core.

J-5 a) A 30°F/min cooldown rate would be sufficient to produce a low pressure trip at 1300#. The cooldown would also be sufficient to add enough reactivity to achieve criticality. Which if achieved would tend to increase pressure. But with the excessive cooldown rate the L.P. would trip the Rx.

b) If the turbine bypass valves are in auto take manual control and try to shut. If they can't be shut, insert rods and insure 1% SDM. Terminate feeding.

76-259

c) Assume no operator action!
Diamond is in manual at 5% below ECP
Feed water stations are in manual with 1 main pump operating - valves are throttled to maintain S/G nozzles flooded - somewhere less than 5% total feedwater flow.

turbine bypass valves full open — pass ≅ 15% power

This cooldown should be sufficient to cause a L.P. trip. Even if sufficient reactivity is added to produce criticality, before the point of adding heat is reach the reactor should trip on L.P.

If the reactor doesn't trip on low pressure and reaches criticality and the point of adding heat. Reactor power will continue to increase until tripped on high flux. I don't believe this will happen, α_{FT} and α_m should add enough negative reactivity feedback to prevent reactor power from increasing to this point. With the turbine bypass valves capacity of 15% and with feedwater throttled to some where less than 5% total feedwater flow. I believe the power will stablize at some value less than 5% after the S/G steam down.

a) Unbalanced Stm and Feed

b) Feedwater Infilling system

— 8

76 300

K

K-1 a) The control rod handling mechanism when the mechanism is clamped to the control rod prevents letting go of the rod unless there is no load on the mechanism. This prevents dropping a rod.

b) The main and spent fuel bridges can not be moved above the fuel transfer baskets unless they are moved into the corridor and the mast is up and fuel transfer basket is in the up position. This is to prevent damage to the basket from hitting it with the mast.

c) The fuel grapple mechanism will not close unless it is on a fuel assy. Once it closes it will not release until the low load limit is reached. The hoist will cutout on a high load limit. The grapple interlock prevents dropping an assy. And the hoist high load limit prevents applying to much force to the assy if it were hung-up.

76-301

K-2 a) The operator is to reduce power to less than 60% and verify that quadrant power tilt and imbalance are within limits.

b) The reduction in power is necessary so that power tilt does not exceed the limits. Exceeding the limits could result in reaching DNBR and exceeding core thermal limits which would result in fuel damage or failure. [illegible]

c) Tech specs require that if you exceed the imbalance limit you reduce power to until you are below the limit in 15 and reduce the high flux flux set point 2% for every 1% of imbalance and that you lower the power level cutoff 1% for each 1% of imbalance within 4 hr.
If you exceed tilt limit you should reduce power to within limits and reduce high flux set point 1% per 1% of tilt.
Assuming you violated both limits, than imbalance becomes L.C.O.

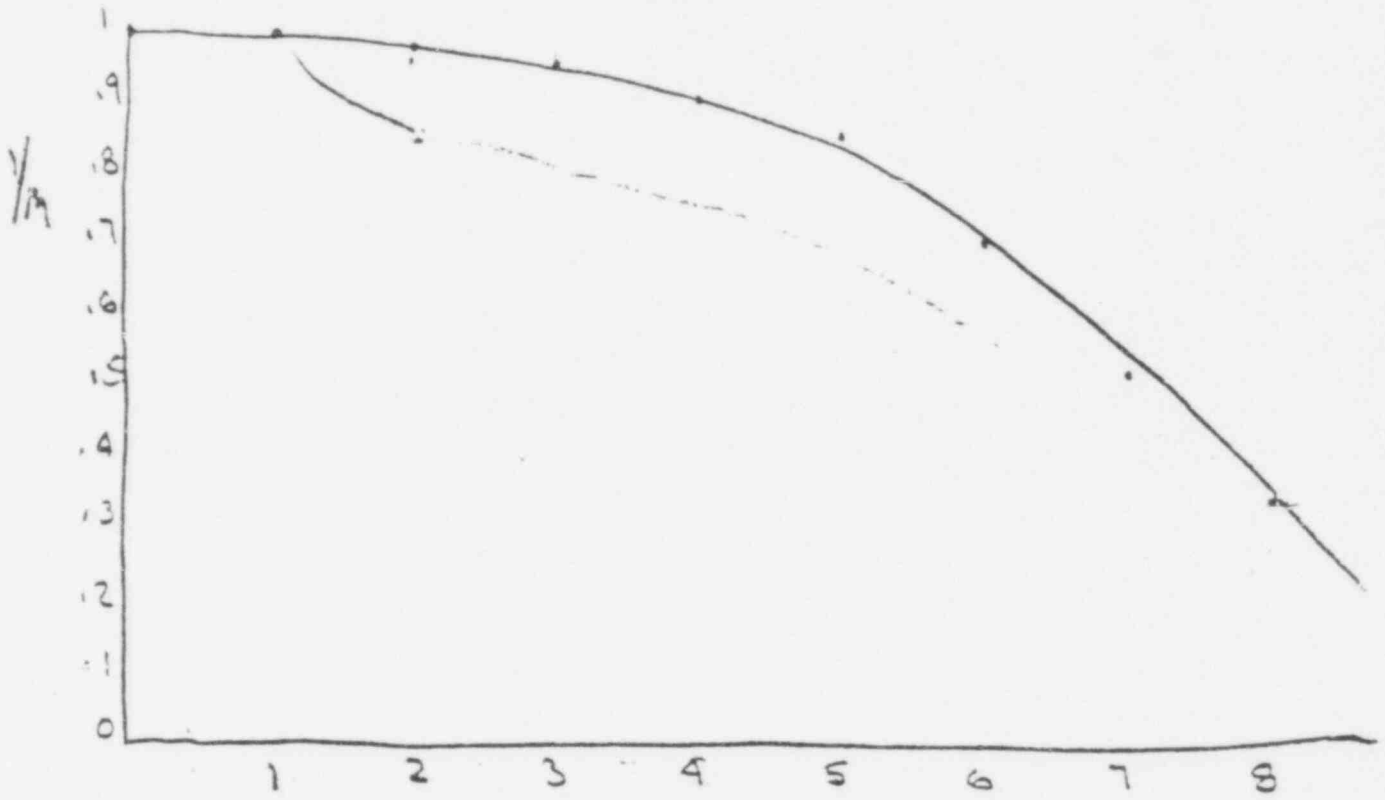
Verify begin to increase 2-3 hrs.

3

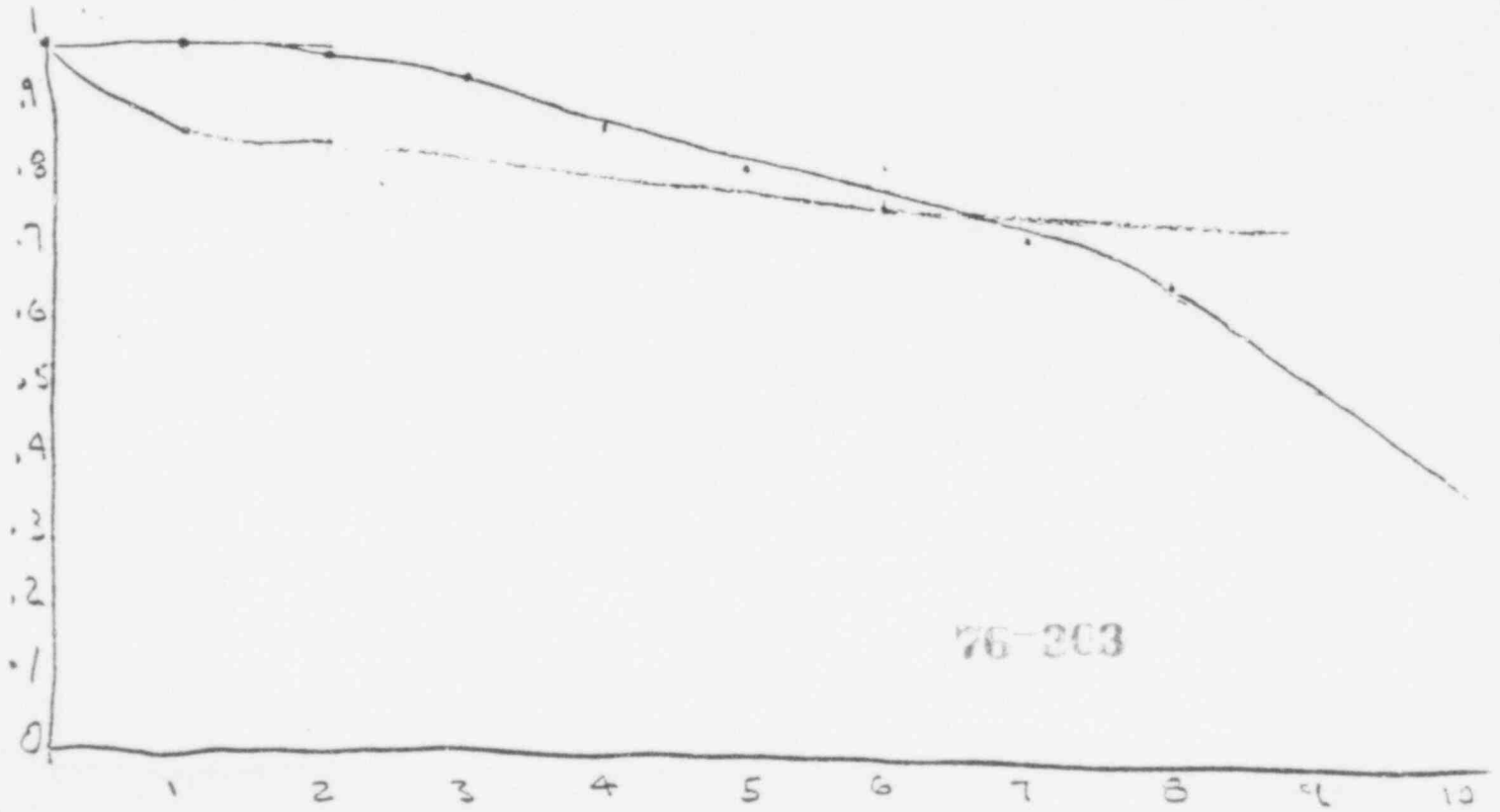
76 302

K=3 FOR A CHAMBER

a)



FOR B CHAMBER



76-203

b) Due to geometry the B chamber is not going to the full effects of the fuel load. Initially with the first 2 or 3 loading sequences the B chamber will see most of the effects of subcritical multiplication and its indication will be closely resemble the indication on A. However as fuel is loaded away from the source and towards A the level that B sees will begin to taper off while A will steadily increase. A is the best indication because it sees the full effect of sub-critical multiplication by the source because the fuel is being loaded between the source and the detector.

K-4

-2

a) Normal maneuvering rate between 75% and 90% is 20% a minute, max is 30% ~~Normal~~ 30%/hr

b) A control rod interchange is performed to limit flux peaks and to insure even burn-up of control rods thus prolonging control rod life. Power increases are reduced to 3%/hr to ~~keep~~ sweep out water logged fuel pins and to limit the effects of fuel densification, pellet-clad interactions and clad creep.

K-5

The reason that the excessive reactivity is not linear over the entire range, specifically during the first 100 days of full power operation, is because of the effect of the lump burnable poison and the buildup of fission product poison Xe and Sm. At the beginning of core life excessive reactivity is high due to the new fuel. Lump burnable poison is also present to help depress the ex reactivity of the new fuel. As the reactor begins operation and fuel begins to burn-up Xe and Sm begin to buildup at the same time the lump burnable poison burns-up. By the end of 100 EFPD all the burnable poison is used up and Xe and Sm have reached their equilibrium values and ex reactivity drops linearly with fuel burn-up.

76-216

L-1 a) THI has core safety limits on DNBR and linear heat flux and reactor coolant pressure. Limiting safety system setting are chosen so that the reactor coolant system does not exceed 2750psia and the core limit for minimum DNBR of 1.3 is not reached. To insure that these limits are not exceeded the following parameters are monitored reactor coolant pressure, temperature and flow; core power, tilt and imbalance.

76-207

b) The pressure signal is produced by a pressure transmitter the trip signal opens a set of contacts in the associated RPS channel. The temperature signal is generated by and RTD - the trip signal open a set of contacts in the asso. RPS channel. The flow signal comes from a gentaley flow tube. Also from a pump monitor which measures MW drawn to run pump & compairs it to a set pt. The flow signal from the flow nozzles is used in a function generator to develop the tilt and inbalance envelope. The signal from the pump monitor is compared with ϕ flux to generate a trips on loss of pumps. The flux signal comes from the power range nuclear instruments. It is used for high flux, compared with pump monitor signal for ϕ/pump trip and sent to the function generator

to develope the inbalance envelope along with flow. Each of the incoming signals has a bistable with it. When the bistable changes state due to a change in the input signal a set of contacts associated with the bistable open and trip the appropriate RPS channel.

c) Not Shy? -.5

L-2 a) control is defined as any operation or manipulation that would effect the reactivity of the reactor
i) withdrawing control rods
ii) throttling feed water flow boration etc

b) It is not necessary to be a licensed operator to manipulate a control. An individual who is training to be an operator can manipulate a control if he is directed by a licensed operator. or supervisor who holds

c) As the operator can not return to his normal duties until he satisfactorily demonstrated to the NRC that he is a qualified reactor operator. The NRC may choose to re-examine the individual or they may accept a satisfactory report from TMI that the individual has been upgraded.

+ Mech

76-209

3

1-3 a) CRO log, shift foreman log, shift & daily logs; Turbine building & out building logs; switch & tagging log; jumper & lifted lead log. Reactor Computer Printout

b) The plant is in a safe and stable condition. All on comming crew members are familiar with plant conditions and their individual responsibilities. (Applicable logs have been reviewed by the on comming shift. + Qualified crew
The on comming shift is aware of any problems and planned plant status activities

c) i) during fuel loading 1 SRO who's sole responsibility is fuel load, 1 CR in the control room at all times and 1 SRO on site.

ii) A minimum of 2 CRO's in control room one of which is an SRO and an SRO on site. Aux operators

76-210

L-4

a) Immediate action

1. isolate NaOH tank from DHR system.
2. sample coolant to check chemistry
3. notify unit superintendent & HP/chemistry supervisor
4. line up DHR system for purifying coolant and make chem add to return Ph to normal.

Sequent action wait for evaluation from chemistry and engineering departments.

Insure Ph in proper band. Outside of RCS

Reason. NaOH is not normally injected into the reactor coolant system. Its addition will have an effect on Ph which will accelerate corrosion. So returning Ph to normal would minimize the effects of this accident.

76-211

b) Immediate action

1. Remove the line from the BWST by slipping it off or small blow-torch.
2. Insure BWST heat tracing operable.

Sequent action

1. insulate the vent line or install additional heat tracing.

Reason. Vent vent the BWST for HP & LP injection in the event of LOCA.

If the RWST is not operable you have to restore it operable in 1 hr on HSB in 6 and CSD in 30.

C Immediate action

Do let the engineer backseat the valve by holding open the relay in the MCC.

Try to stop leak by tighting the packing gland.

If this work, secure from the test and repack the valve. The purpose of the is find leaks and repair them.

over-tange: valve [illegible]

76 312

B5 a) Water hammer is the term used to discribe a slug of water traveling at high velocity hitting a component (valve, heatexchanger, Bend in piping, ect,). Water hammer can occure when a pump is started and the system is not completed filled. The air in the system allows the water-slug to travel at higher than normal velocities this water when it strikes a component deposites it kenitic energy to the component which must be absorbed by the component. It can result in excessive vibration of system piping, broken components (valve stem and damage to the seat) piping erosion or broken or damaged pipe hangers. Water hammer can be elimated or reduced by insuring the system is properly filled and vented. It is a concern to the operator because of the damage that can results from it.

76-213

b) So water hammer can also exist when steam is admitted to a system which is partially filled with water. The steam will push the water slug and accelerate it to high velocities. The results are the same as if the system contained air

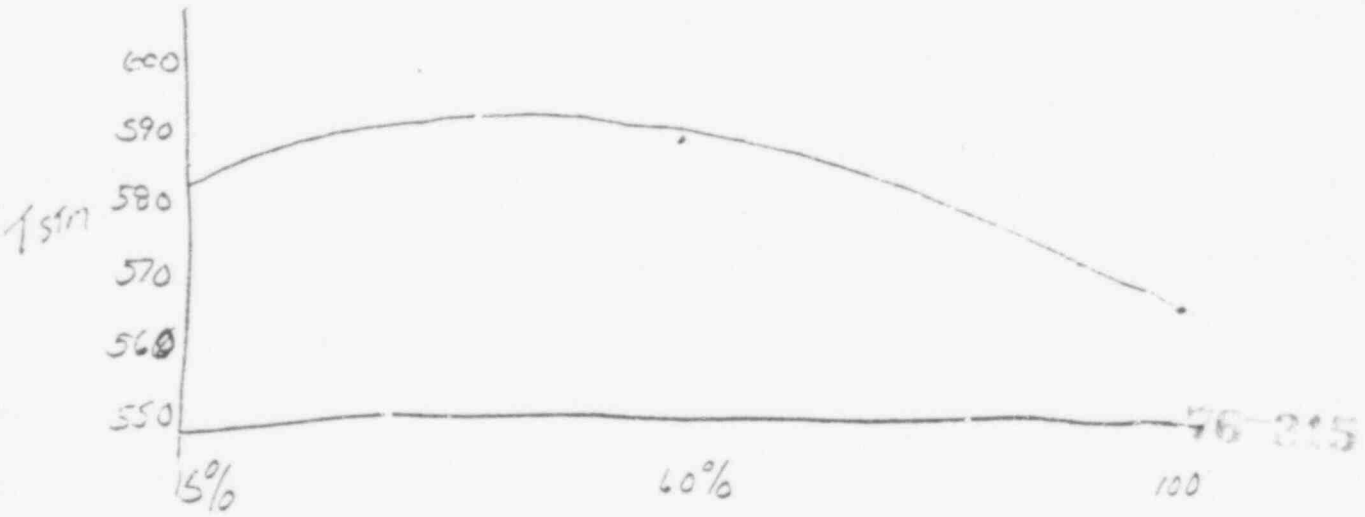
This type of water hammer can be reduced if the system is drained down good after steam has be removed to allow condensing steam to be drained out. It also helps to admit steam slowly and heat the piping up slowly while venting and blowing down the drains.

76-211

C1 a) Superheat is accomplished in the OTSG in the following manner. When feedwater enters the the generator it is mixed with aspiration steam which adds to the feed water preheating. By the time the feedwater comes down the down comer and into the tube bundle area it is already at saturation temps for the S/G press. The reactor coolant from the hot leg enters the S/G at the top and flows to the bottom. This results in cross flow between the reactor coolant and the feedwater. As soon as the feedwater reaches the bottom of the tube bundle it begins to boil (because it is already at T_{SAT}) the lower portion of the S/G is used for nucleate boiling. In this region the majority $\cong$ 90% of the total heat transfer across the S/G takes place. The feedwater which is now steam then enters

the film boiling region of the OTSG. Finally the feedwater which is now all steam enters the superheat area (top portion) of the OTSG, here it comes into contact with the hottest part of the tube bundle. The hot reactor coolant enters the top of the S/G allows the steam to be elevated and dried resulting in steam leaving the S/G with a high degree of superheat.

b) The degree of superheat drops slightly from 15% to 100% power because OTSG level is increased from 15% to 100%. Increasing the level increases the area of the S/G used for nucleate boiling which effectively reduces the area available for superheat. Superheat varies from 15 to 100% per sketch.

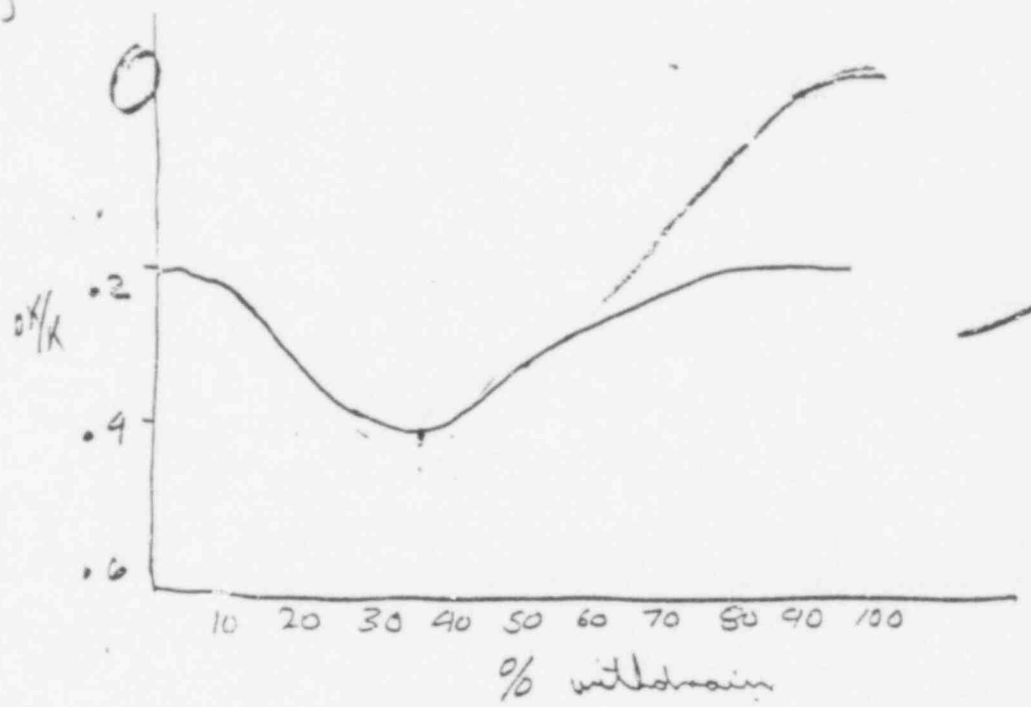


The S/G design value is to give 35° superheat at 100% power. Past experience has shown that superheat decreases only slightly at high power levels

C-2 a) 30%. A idle reactor coolant pump is not started at power levels greater than 30% to prevent a cold water accident and to allow ICS sufficient time to limit the power excursion.

b) 5%, 297 and 299 in. S/G water level is maintained between 297 in and 299 in in the startup range when reactor power is less than 5% to maintain the feed nozzles flooded to prevent thermal gradients across the nozzles. The 5% value is chosen because above this level feedwater demand is sufficient to maintain the nozzles full of water. Below the power level feedwater demand is insufficient to maintain the nozzle completely full of water.

C-3 a)



4

B) ASPR are used to control reactor power imbalance between top and bottom of core. If the ASPR are properly position and you have a positive imbalance you would drive rods in. If you had a negative imbalance you could withdraw APSR out

1.0

C-4 a) 1. hydrogen introduced to scavenge O_2
2. fission product gases.
3. gases from make up water
4. H_2 & O_2 dissociating from ON flux.

b) Gas concentration in the reactor coolant is a factor because excessive gases could come out of solution and accumulate in the upper housing of the CRD's. This gas accumulation would prevent the hydraulic damping that the reactor coolant has when it enters the upper housing when the rods are dropped.

c) A minimum gas concentration of 15 to 40 cc/kg hydrogen is maintained in the coolant to scavenge O_2. Oxygen enters the coolant from make up water and from dissociating from water under neutron flux. The hydrogen excess is maintained so the O_2 will combine with it and return to water. Maintaining a very low O_2 content reduces corrosion. 76-219

d) The preferred temperature range is 115 to 120°F the lower value is selected to prevent

condensation from forming on the CRD's. the upper value is slected to maintain the CRD's staton temperature at a safe operating temperature.

C-5 b) ASSUME 2A REACTOR COOLANT PUMP TRIPS

a) -, 2 b) -, 3

D-1 a) On loss of a reactor coolant pump the ICS recieves a signal to run back to less than 75% power. This run back signal is sent to all the ICS substations, And the plant begins a runback to less than 75% reactor power at 20%/min. ICS runs reactor power, feedwater and unit load demand back to less than 75%. how is MWe control.

note?

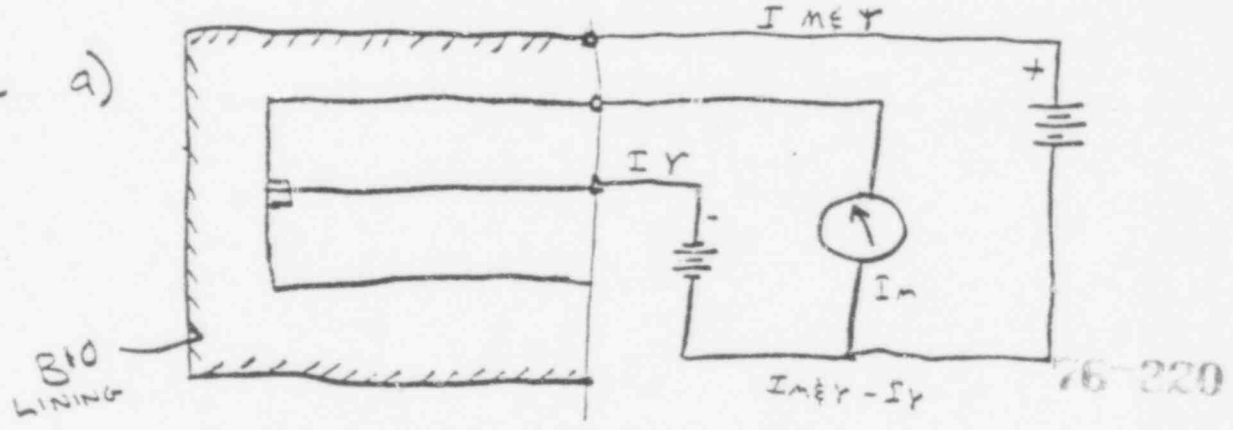
76-219

b) Pressurizer level will begin to increase as Tav increases. Letdown flow will increase to maintain Pzr level and will limit the rise in the Pzr. As Tav returns to normal after system has stablized Pzr will return to normal

level

c). The feedwater flow ckt will see a decrease in reactor coolant flow in the A loop it will start to ratio feedwater between the generators. Decreasing the feedflow to A generator and increasing the flow to B. At the same time it will recieve a signal that the plant is running back to 75% power. This signal to decrease feedflow will be ratioed by the feedwater ratio ckt to both generators. The ΔTC ckt of the feedwater demand ckt will adjust feedwater flow so that the ratio will have a minimum ΔTC.

D-2 a)



The compensated ion chamber consists of two chamber one sentive to m&γ the other sentive to only γ. The outer changer is coated with B^{10} and is sentivity to both neutrons and γ rays. The neutrons react with the B^{10} and produce an α particle. The α particle

causes ionization of the gas in the detector the resulting ionized particles collect on the sides of the chamber and result in current. A γ ray entering this chamber will also result in ionization of the gas and a current on the chamber walls. The inner chamber is not coated with B^{10} so it is not sensitive to neutrons. However, a γ entering this chamber will result in a current flow. The electrical signal from the dector is added so that the resulting meter indication is due to neutrons alone. The inner chamber is operated in the recombination region so that by adjusting the compensating voltage the signal due to γ rays does not appear on the meter.

76-321

b) Compensation is required at very low power levels. Neutron flux by the time the reactor enters the power range is so much greater than γ flux that the γ flux can be neglected. Also the γ flux is proportional to power in the power range. Because of the above reasons the Power range instruments are not compensated. The intermediate range instrument is compensated so that the operator will have a true indication of neutron flux from the source range to the power range. The source range instruments do not utilize a compensated

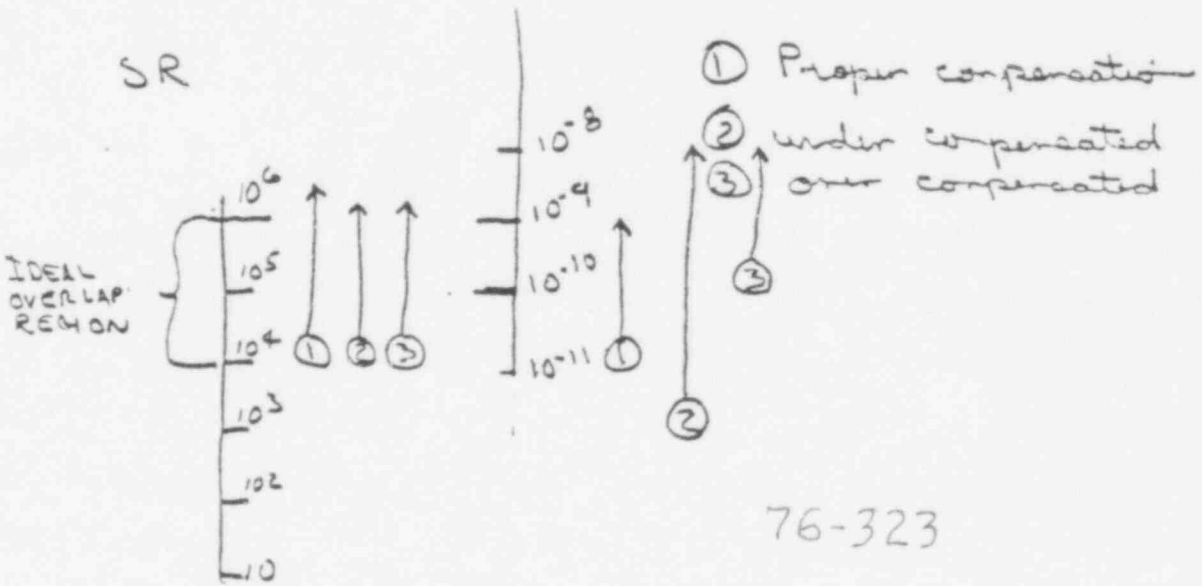
ion chamber but rather a BF_3 proportional counter. The proportional counter eliminates the the signal due to γ's by using a pulse height discriminator in its ckt. The discriminator only allows pulses of a certain value or higher to give indication. In this way the γ pluses can be removed because they are not as intense as the pluses generated by a neutron reaction with the BF^3 gas.

MISSED D2C SEE NEXT PAGE

D-3 The PZR electro-matic relief valve has a 3 position switch, OFF - MANUAL - AUTO. In off the valve is shut, in manual the valve will be open, in the auto position the valve is controlled automatically by a pressure signal. The valve will open when pressure reaches 2255 PSI and shut when it reaches 2205 PSI. Unless the NDTT switch is in the on position then the electro-matic relief will open at 500 PSI if the reactor coolant temperature is less than 275°F if the temperature is greater than 275 than the lift set point returns to 2255 PSI.

D2C

If you assume that the overlap requirement means a minimum of one decade between 10^5 and 10^6 in the source range and 10^{-11} and 10^{-9} in the intermediate range than changing compensation to either over or under compensation than you decrease the overlap between source and intermediate. If the CIC is grossly over compensated than the doctor will not see none of the neutrons produced. If it is ~~under~~ compensated than it will see some of γ's as neutrons. The sketch below show this graphically



When properly compensated 10^4 to 10^6 in source range would correspond to 10^{-11} to 10^{-9} in the IR

When under compensated 10^4 to 10^6 in SR would correspond to some value greater than 10^{-11} to some value greater than

10^{-9}. Thus reducing the area that the SR and IR travel together.
If ver compensated 10^{4} is source range would not correspond to anything in IR because meter would not be on scale yet. Thus overlap would be reduced.

D-4

Rx Power -.5

a) The feed and bleed permissive signal comes off of rod position. The distance from the nominal rod height the feed and bleed permissive comes in at ± 5 in from nominal rod height.

b) The signal goes to the feed and bleed permissive annunciation and the batch controller. Link -1

c) The controller is utilized when you are in the soluble poison control mode with the rods in the transient band and you wish to make power changes at greater than .5% min. The controller allows the operator to simultaneously feed and bleed. > 15% Power -3

76-325

D-5 a) i) The motor fault alarm will occure under the following conditions.

1) out motion with no command
2) out motion with an in command
3) in motion with no command.
4) lost of programmer ABT power.

ii) The asymitrie monitor will give the operator an alarm if a rod is out of position from the group average by more than 9 in.

The purposes of these alarms is warn the operator of specfic faults in the rod control system. The motor fault alarm indicates that there is rod motion with no command or that the rods are not moving in the diretion they are intended to. It also alerts the operator of a problem with the programmer motors. The asymetric rod alarm alerts the operate. to the fact the a rod is misaligned from its group. It could be stuck or dropped or misaligned.

b) i) If rod is withdrawing with no control, transfer to manual. 2
select effected rod or group.
drive rod in, if still withdrawing
trip the reactor and follow E.P. for reactor trip.

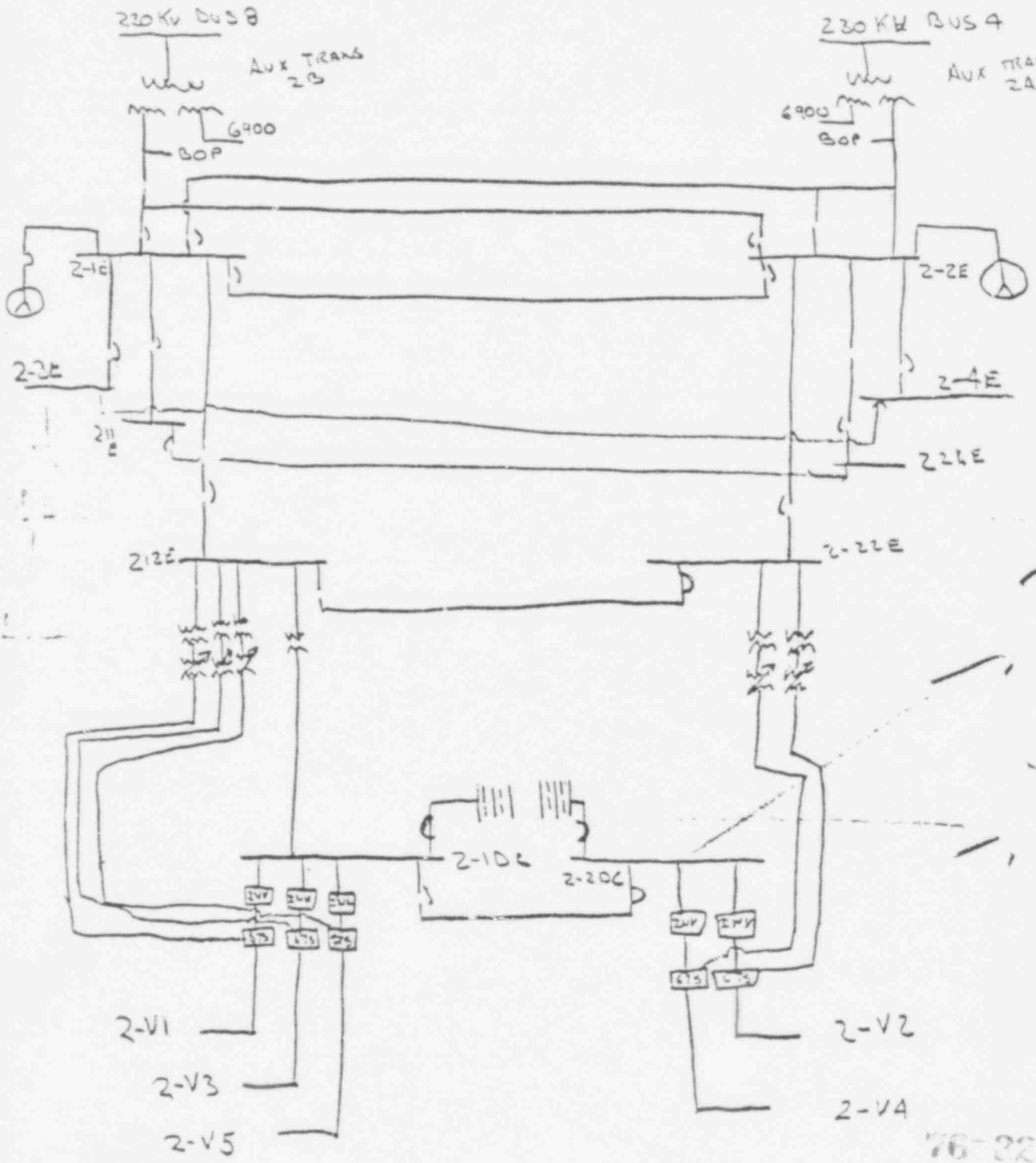
ii) If assymetric rod occures plant will run back under following condition, if greater than .60% power and ICS in auto

1. a loss of group 1→4 out limit
2. a group 1→4 in limit
3. a group 5 in limit, greater than 60% power & 9" asymm
4. a gr 6/7 in limit, w/ or 5 greater than 80% & a 9" asymmod.

If the plant runs back, operator action is to verify automatic actions have actuated, to not run appropriate ICS station back in manual to level corresponding to 60% value. And verify tilt with in limits.

If plant is less than 60% or you don't meet the condition to get an assymetric run back. Transfer effected rod to aux power supply and realign with group.

E-1 a)



76-328

b) 1. T-G DC LUBE OIL BEARING PUMP
2 T-G DC TURNING GEAR LUBE OIL PUMP
3. MAIN FO PUMP LUBE OIL PUMP
4 RCP DC LIFT PUMPS
5. EHC DC POWER
6. RPS
7. SPDS

E-2 1 high flux 105.5%
2 high temp 619°F
3 high pressure 2355 PSI
4 high Rx BLO Pressure 4 psi
5 low pressure 1800 Psi
6 variable low pressure 10.97 (TH) - 4650
7 ∅/flow/imbalance 105 (flow) - reduction due to imbalance
8 ∅/pump trip greater than 56.4% power loss of 2 RCP's or loss of both pumps in 1 loop.
9 manual trip.

1 min	2 min	3 min	4 min	5 min
2 psi/min	4 PSI	8	16 PSI	32 PSI

E-3 a) 4 PSI Rx signal 1. trips reactor
2. initates HP and LP injection
3. initiates Rx building Isolation
4. Starts Bld. spray pumps and lines system for cooling with exception of BS-V-1 valves.

76-229

b) MU pump running lights, DHR pump running lights. Valve position indicating lights. Decreasing BWST level, increasing Rx bldg sump level. Decrease in Pzr level drop (POSSIBLE), POSSIBLE stopping or slowing down of Reactor coolant pressure drop. Rx building isolation & cooling status lights. Initial indication is Rx building & HP & LP actuation lights on control panel. Continuing indication is status lights, level indications, pressure indication,

and amp meters for pumps.

c) Reactor containment is design for 60# building pressure, postulated peak after LOCA is 56.9#.

E-4

A) The feedwater latch system consists of eight pressure switches, 4 per generator with 2 pressure switches on each of the main steam lines. Each generator operates on a 'out of 2 X-2 logic. Meaning one of the two pressure switches on each header for the associated S/G has to trip before the feedwater latch signal is produced.

b) Shuts the associated MS-4 and MS-7 valves shuts the associated main feed water reg. valve, the associated main block valve, the associated associated start-up feed water reg and block valves.

E-5 a) 1 S/G pressure increasing
2. feed water flow to affected S/G decreasing due to increase pressure. following
3. VA-R-748 alarming 3
4. Primary pressure decreasing.
5 PZR heaters on to maintain pressure
6. PZR make-up valve opening to maintain level
7. Make up tank level dropping.
8. Primary pressure continuing to decrease with all PZR heaters on.

b) After the OTSG that is leaking is determined the operators should conduct a normal shutdown. During the shutdown extreme care should be taken not to lift the S/G safety valves and release radioactive to the atmosphere. The shutdown should continue until the reactor has been cooled to below the set temperature of the lowest set safety valve. To prevent lifting the relief and releasing radioactivity.

F-1 a) Power level cutoff is 92%. Power may be increased above this level if Xe is within 10% of the equilibrium value for rated power and is approaching stability or if the reactor has been operated in the 87 to 92% range for 2 hr in the soluble poison control mode, excluding Xe free start-ups.

b) ICS will see the loss of vacuum, which is a decrease in plant efficiency, as generated MW dropping off. The ICS will respond by increasing reactor power & feedwater demand to try and maintain unit load demand signal.

c) If vacuum is decreasing and ICS is attempting to maintain unit load demand, the operator should take manual control of the R/FW station and reduce reactor power so that the power level cutoff value is not exceed. He should continue to decrease power until the vacuum leak is stop or can be maintained at a lower power level.

76-232

d) If condenser vacuum can not be restored

then the reactor has to be shutdown. The available heat sink is the atmosphere thru the atmospheric dumps. Feedwater sources are the hot well both condensate storage tanks, demin water and as a last resort river water via nuc-river system.

Safety Vlv, Sta Ethan Pump,

3300 3000

F-2 a) 3000 GPM, 400 GPM

b) Pump runout is the condition where the pump is pumping more water than it was designed to pump. It is doing more work than the manufacture intended it to do. The consequences of this is that the pump or pump motor will overheat and continue operation will burn the pump and/or motor up.

c) Cavitation is caused when the suction pressure of the pump is at saturation pressure for the fluid. The fluid around the impeller goes to gas which causes the pump impeller to erode. When cavitation occurs you can usually hear it and the discharge pressure of the pump will drop sharply.

F-3

a) the limit is ± .5% Δk/k from ECP

ECP = 50% on GR 6/7

using the rod worth curve in fig. 2-B the operator finds 50% or ECP on rod index comes up at this point to the curve. When the ECP is on Rod worth curve the operator goes to vertical axis, and moves both up and down on vertical axis .5% Δk/k from this point he comes back to the curve and then drops to horizonal axis to find rod height at these points.

for 50% on GR 6/7

+ .5% Δk/k ≥ 100% on GR 6/7

- .5% Δk/k ≅ 39% on GR 6/7

max limit 100% on GR 6/7

So band is ECP 50% on GR 6/7

min limit 39% on GR 6/7

76-224

b) If the reactor goes critical before it reaches this band. The operator has to insert rods, to insure 1% SDM and investigate the reason for the ECP being off.

If he has not reached critical by the time he reaches the upper band he inserts rods to

the -.5% position, which insures 1% SDM and investigate reasons for ECP not being correct

F-4 Manual actions for reactor trip

1. manually trip the reactor
2. verify rod bottom lights
3. verify turbine trips & generator & field brks open — if not trip turbine oil.
4. Close MU-V-376
5. start 2 make-up pump if necessary to maintain PZR level 100" Control Vl.
6. Verify ICS is runing feedwater back to low level limits — if not take manual control and do so.
7. verify turbine bypass valves maintaining header pressure at 1010.
8. If PZR level drops to 80" verify heaters cutoff
9. If PZR level drops to 20" open DH-V5B and start 3 make up pump and open MU-V-16C,D
10. Verify S.R instrumentation is energized
11. If reactor power is not less than 10% in 1 min stop emergency boration.

-.2

76-335

F-5

a) 1. Increase in SR counts, IR channels and power range indications.
2. SUR with no rod motion and counts increasing.

b) 1. Insert rods.
2. monitor ER level and insure 1% SDM.
3 trip if necessary.
4 insure no dilution - taking place

G-1 a) A G-M device should be used. The GM devices operate in the GM region where incident particles cause a reaction that is monitorable. The GM tube is very sentinits to low levels of radiation.

76-236

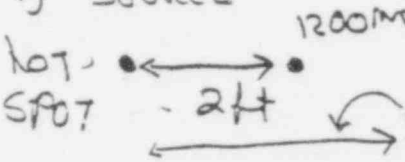
b) The instrument will detect β^- and γ radiation. Most GM tubes are equipped with a β^- window. You take a reading with the window closed you measure γ radiation, with the window open both β^- and γ radiation. The instrument has a correction factor which can be used to give β^- radiation alone.

G-1 c) Frisker are normally set to alarm if the dose is greater than 100 counts above background. If the frisker alarms you should notify H.P. dept. And they will assist you in the proper method for removing the contamination.

1000 DPM/100 cm² B-γ

100 " α

G-2 a) SOURCE



since radiation is due to a hot spot will treat it as a point source

$D_1 R_1^2 = D_2 R^2$

$1200 \text{ mr/hr} (2)^2 = D_2 (5)^2$

$D_2 = \dfrac{1200 \frac{\text{mr}}{\text{hr}} \cdot 4 \text{ ft}^2}{25 \text{ ft}^2}$

$D_2 = 192$ mr/hr at 5 ft.

Dose rate = $192 \text{ mr/hr} \cdot 2 \text{ hr}$

D.R. = 384 mr.

76-237

b) Previous exposure 900 mr
+ 384 mr
1284 mr = 1.284 Rem

You have exceeded the quarterly limit of 1.25 R/QT specified in 10 CFR 20. You have also exceed. TMI admin limits of 300 mr/wk and 1000 mr/QT. All of the above limits can be exceed if you have proper authorization.
The 10 CRF 20 limit of 1.25 R/QT can be increased to 3 R/QT providing that you don't exceed 5(N-18) and you have complete NRC form 4.
The TMI 300 mr/wk limit can be exceeded with verbal OK from H.P. supervison.
The TMI 1000 mr/wk limit can be exceeded with written permission of the H.P. supervi

76-239

c) Two small metal beams (strips) are electrically charged. The charge holds the beams apart. As the dosimeter is exposed to radiation the incident particles reduced the charge and the beams come closer together. This motion is transmitted to a scale which corresponds to the amount

of radiation the dosimeter has been exposed to

G-3 a) a controlled area is defined as an area where radiation could exceed 5 mr/hr or 100 mr/40 work wk potential exceeds clean areas

a radiation area is defined as an area where radiation dose levels are 5 mr/hr or 100 mr/40 work wk

a high radiation area is defined as an area where you could be exposed to 100 mr/hr dose rate

76-329

b) A NRC examiner may enter any of the areas as long as he does not exceed his 10 CFR 20 limits and is authorized to under a RWP for the radiation and high radiation areas and meets the requirements of the RWP for these

areas. And has his dosimeter record upto date

G-4 Secure the spill. Shut valve ~~if~~ open, if bottle turned over upright it, if a leak isolate it act.
Isolate the area of the spill by roping off, barricading or tell every one to stay out
Warn other personnel that there is a spill - notify the health ~~physic~~ department and the control room and SS/SF.
Minimize your own exposure ~~by~~ staying as far away as possible without leaving immediate area and using installed components for shielding.

G-5 a) reactor building area monitor, done monitor
B.C.

b) purge valves shut, ventilation ~~sys~~ lined up to recirc through filters

76-340

c) sound the reactor building evacuation building alarm, establish reactor compartment integrity, notify SS/SF unit supervisor. Insure personnel is evacuated from the Rx bldg.

monitor stack release and ditimmic release rate. SS will prepare a entry team with H.P. to take survey.