11/16/81

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of METROPOLITAN EDISON COMPANY, et. al. (Three Mile Island Nuclear Generating Station Unit No. 1)	)))))))	Docket No. 50-289 (restart)

NRC STAFF TESTIMONY OF PAUL F. COLLINS
REGARDING THE ADEQUACY OF STAFF'S ADMINISTRATION
OF ITS EXAMINATIONS

Q. 1. Please state your name and position with the NRC.

A. My name is Paul F. Collins. I am Chief, Operator Licensing Branch, Office of Nuclear Reactor Regulation.

Q. 2. Have you prepared a statement of professional qualifications?

A. 2. Yes. A copy of this statement is attached to this testimony.

Q. 3. What is the purpose of your testimony?

A.3 The purpose of my testimony is to address and describe the actual administration of the October 1981 TMI-1 operator examinations, including proctoring, grading, safeguarding of the examination, licensee's review, and extent to which questions used in these examinations were the same as those used on previous examinations.

Q.4 What procedure was used for proctoring the October examinations?

A.4 We made arrangements with four professors from Pennsylvania State University, Capital Campus, Middletown, Pa., to proctor examinations. P. Collins met with them at the TMI training center on September 24, 1981 to discuss the proctoring and to permit them to see the examination room and to acquaint them with facility management. The items they proctored are indicated below.

	Dates
Professor William Aungst Pennsylvania State University Capital Campus Middletown, Pa 17047	October 22 & 29, 1981 - all day
Professor William Welsh Pennsylvania State University Capital Campus Middletown, Pa 17057	October 22 & 29, 1981 - half day (morning)

Professor Wesley Houser	October 21 & 28, 1981 - full day
Pennsylvania State University	October 22 & 29, 1981 - half day (afternoon)
Capital Campus	
Middletown, Pa 17057	
Professor Joseph Douglas	October 21 & 28, 1981 - full day
Pennsylvania State University	
Capital Campus	
Middletown, Pa 17057	

The proctors were present in the room when the examiner passed out the examination to the candidates and listened to the instructions given, including admonitions regarding cheating. There was at least one proctor present in the examination room at all times during administration of the examinations. They had no other concurrent duties to that of proctoring. They toured the room, assured that there was no communication between or among the candidates. They permitted only one candidate to be out of the room at a time and made a log of absences. At the conclusion of the examination, the proctors assured all paper was turned in.

Q.5 How is the grading being conducted?

A.5 The reactor operator examinations are being graded by the individuals who wrote the examinations at Battelle Pacific Northwest Laboratory.

The individuals indicated that one question or one section is being graded at a time for each candidate before proceeding onto the next question or section. This is in conformance with our normal grading practices.

One individual is grading categories A, B, and C, another is grading categories D, E, and F, and a third individual is grading categories G and H.

The senior operator examinations are being graded by NRC examiners in the Bethesda offices. The same question is being graded for all applicants prior to grading a subsequent question. One examiner is grading the "A" examination. Two examiners are grading the "B" examination, one will grade categories I, J, and K, while the other will grade categories L, M, and N.

All graders have been alerted to compare answers to determine if any cheating is apparent.

The examinations will be reviewed by OLB supervision, per Examiners' Standard ES-109, that includes a review for cheating.

Q.6 How were the examinations safeguarded?

A.6 The senior operator examinations were prepared in NRC offices in Bethesda, Md. Access to the offices is limited and controlled. The examinations were transported to the TMI training center in the possession of an NRC examiner. The examinations were passed out at the facility on the day they were administered.

The operator examinations were prepared in Richland, Washington at the Battelle Pacific Northwest Laboratories. Access to these offices is also controlled and limited. The exams were mailed to the NRC Bethesda offices, where they were reviewed, edited, and copies reproduced. The examinations were transported to the TMI training center in the possession of an NRC examiner. The examinations were passed out at the facility on the day they were administered.

A.7 Were the examinations reviewed by the Licensee?

A.7 Yes.

Q.8 How was the review conducted?

A.8 After the examinations were distributed to the candidates, the examiner met with licensee representatives outside the examination room and reviewed the questions and answers. At the conclusion of the review, the examiner returned to the examination room and explained changes, as necessary, to the candidates.

Q.9 Have you reviewed the October examinations to determine if questions were similar to those asked on other examinations?

A.9 Yes. We compared the TMI-1 October examinations with all the examinations administered since April 1981 at B&W-designed facilities. These facilities are: Arkansas Nuclear One, Unit 1; Oconee Nuclear Station, Unit 1, 2 & 3; Crystal River Nuclear Plant; Davis-Besse Nuclear Plant; and Rancho Seco Nuclear Plant.

Q.10 What were the results of that review?

A.10 Less than 4% of the questions on the two RO/SRO examinations given at TMI-1 in October of 1981 were similar to those given at the other B&W-designed facilities.

Q.11 Do you have the Master Copies and answer keys of the examinations administered at TMI-1 in April 1981?

A.11 Yes. We have copies of the "A" and "B" operator examination and senior operator examination.

Q.12 Do you have copies of the answer key as originally prepared?

A.12 No. The answer keys contain many hand written notations.

Q.13 Can you ascertain from the hand written notes who made them, when they were made, and why they were made?

A.13 We can distinguish who made the notations. However, we cannot determine when they were made, nor can we determine why they were made.

Q.14 Can you determine who prompted the notation?

A.14 No. The examination and answer key are typed at the same time. Changes to the key can occur and hand written notations be made because:

1. The complete answer was not typed.
2. The examination review process, prior to the administration of the examination, may reveal a change is necessary.
3. The licensee reviewers may recommend a change.
4. The grader may determine that the answer should be changed.

At this time it is impossible to determine why answers were changed or augmented.

Q.16 Are the Master and answer keys to the April examinations included?

A.16 Yes.

U.S. NUCLEAR REGULATORY COMMISSION

SENIOR REACTOR OPERATOR LICENSE EXAMINATION

Facility: TMI-1

Reactor Type: B&W

Date Administered: 4/22/81 - WED.

Examiner: Wilson/Boger/McMillen

Applicant: MASTER - A

INSTRUCTIONS TO APPLICANT:

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

Category Value	% of Total	Applicant's Score	% of Cat. Value	Category
15	17.4			I. Reactor Theory
12	14.0			J. Radioactive Materials Handling, Disposal and Hazards
13	15.1			K. Specific Operating Characteristics
13	15.1			L. Fuel Handling and Core Parameters
20	23.3			M. Administrative Procedures, Conditions and Limitations
13	15.1			N. Theory of Fluids and Thermodynamics
86	100.0			Totals

Final Grade __________ %

I. Nuclear Theory (15)

I.1 Following an increment of reactivity addition, while subcritical, there are two factors involved in the determination of how long it will take to reach an equilibrium count rate. What are these two factors? (2)

I.2 An increase in moderator temperature will influence the total non-leakage probability and on the response seen by the out-of-core detectors. What influence does the moderator temperature increase have on each of these parameters (i.e., increase or decrease) and on which will the effect be more significant? (2)

I.3 Tech Spec 3.1.3.4 states that the reactor must be maintained subcritical by at least 1% $\Delta k/k$ until a steam bubble is formed.

a. What is the basis of the shutdown margin Tech Spec? (1)

b. Assume the reactor is at hot shutdown, Tavg 532°F and OTSG's at low level limits. Would the consequences of a steam line break in this condition be worse at BOL or EOL? Why? (2)

c. Assume for any time in core life, the reactor is at hot shutdown, Tave 532°F, and flooded nozzles in the OTSG's (97-99% on operate range). Would the consequences be less or worse than if the OTSG's had been low level limited? Why? (2)

I.4 Each of the three following reactivity parameters changes over core life (according to OARP Nuclear Theory lesson plan): Equilibrium Xenon, moderator coefficient, power deficit. Below are the numerical values given in the lesson plan. For each reactivity parameter, briefly explain why it changes and how it would affect shutdown margin (larger or smaller) immediately following a trip from full power.

Parameter	BOL	EOL
a. Equilibrium Xenon	-2.64% $\Delta k/k$	-2.73% $\Delta k/k$
i. Reason for change:		
ii. Effect on SDM		(2)
b. Moderator Coefficient	$-.77 \times 10^{-4} \Delta k/k/°F$	$-2.63 \times 10^{-4} \Delta k/k/°F$
i. Reason for change		
ii. Effect on SDM		(2)
c. Power Deficit (0 to 100%)	-1.34% $\Delta k/k$	-2.06% $\Delta k/k$
i. Reason for change		
ii. Effect on SDM		(2)

J. RADIOACTIVE MATERIALS HANDLING, DISPOSAL AND HAZARDS (12)

J.1 Iodine 131 has a half life of 8 days and emits a 0.61 MEV Beta, a 0.36 MEV gamma 80% of the time and a 0.64 MEV gamma 10% of the time. For three curies of this material, (Show all calculations).

a. What would the activity be after it has decayed for 30 days? (2.0)

b. After this 30 day decay period, if all this activity were in one liquid radwaste tank could it be discharged directly to the environs? Briefly explain your answer. (1.0)

J.2 What two criteria must be satisfied in order for personnel to enter an area without an approved RWP, when the radiological conditions exceed the conditions which require an approved RWP. (2.0)

J.3 a. List two portable instruments at your facility that can be used to detect Beta-Gamma radiation. (1.0)

b. List one portable instrument at your facility that can be used to detect alpha radiation. (0.5)

c. List one portable instrument at your facility that can be used to detect neutrons. (0.5)

J.4 What are the 10 CFR Part 20 requirements for 24 hour notification of an overexposure of personnel to radiation? (1.5)

J.5 A unit used to describe the activity of a radioactive substance is a curie. If the biological hazard of a substance of X curies is to be assessed, what additional information must be known about the substance? (1.5)

J.6 a. What are the two facility limits on primary to secondary leakage? (1)

b. Explain the basis for each limit. (1)

K. SPECIFIC OPERATING CHARACTERISTICS (13)

K.1 Assume the unit is operating at full power with an imbalance of -6, when a spurious runback reduces power to 90% and shifts imbalance to -14.

a. Assuming no operator action, how will imbalance change over the next 4 hours? Explain. (1)

b. Consulting Figure K.1, outline two methods of returning the unit to an allowable imbalance. For each, describe how the method will cause imbalance to change. (2)

K.2 After a reactor trip from full power, situations exist that require taking the RCS solid.

a. How do you verify that the RCS is solid? (1)

b. Give five ways to reduce RCS pressure when solid. (1)

c. Outline a procedure to recover a pressurizer steam bubble. (1)

K.3 Explain how changing the turbine bypass valve reactor trip setpoint bias from 125 psi to 150 psi would affect facility operations. Figure K.3 should be used. (2)

K.4 While operating at full power, the unit experiences a trip. Two minutes later all four RCP's trip due to a loss of offsite power. Power is restored 30 seconds later. You arrive in the control room five minutes after the reactor trip verifying that the immediate actions of procedures 1202-4 and 1202-2 have been accomplished, you observe that hot leg temperature is rising at a rate of almost $5^{\circ}F$ per minute.

a. What is the most probable cause of this temperature increase? Explain. (1.5)

b. Should you restart one or more RCP's in response to this temperature increase? Explain why or why not. (1.5)

K.5 While operating at full power the unit experiences a transient which results in a reactor trip and ESAS initiation.

a. If the cause of the ESAS initiation is high reactor building pressure, should you trip the RCP's? Explain why or why not. (1)

b. If, due to unavoidable circumstances, the RCP's are not immediately tripped per EP 1202-6B, should you trip them several minutes after the reactor trip? Explain why or why not. (1)

FIGURE 4

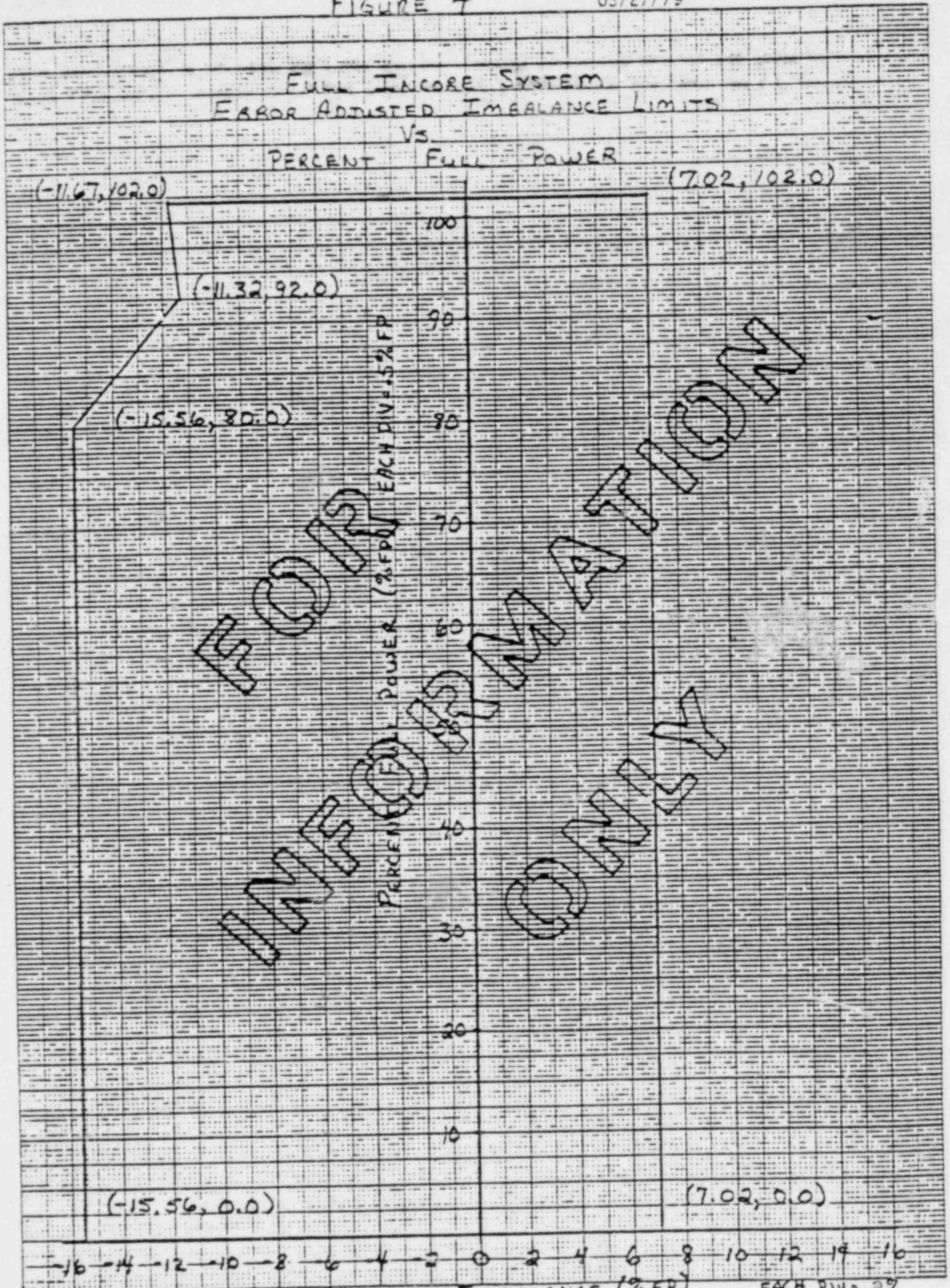


Figure K.1

Table 1 Saturated Steam Temperature Table—*Continued*

Temp Fahr t	Abs Press Lb per Sq In p	Specific Volume Sat Liquid v_f	Evap v_{fg}	Sat Vapor v_g	Enthalpy Sat Liquid h_f	Evap h_{fg}	Sat Vapor h_g	Entropy Sat Liquid s_f	Evap s_{fg}	Sat Vapor s_g	Temp Fahr t
460.0	466.87	0.01961	0.97463	0.99424	441.5	763.2	1204.8	0.6405	0.8299	1.4704	460.0
464.0	485.56	0.01969	0.93588	0.95557	446.1	758.6	1204.7	0.6454	0.8213	1.4667	464.0
468.0	504.83	0.01976	0.89885	0.91862	450.7	754.0	1204.6	0.6502	0.8127	1.4629	468.0
472.0	524.67	0.01984	0.86345	0.88329	455.2	749.3	1204.5	0.6551	0.8042	1.4592	472.0
476.0	545.11	0.01992	0.82958	0.84950	459.9	744.5	1204.3	0.6599	0.7956	1.4555	476.0
480.0	566.15	0.02000	0.79716	0.81717	464.5	739.6	1204.1	0.6646	0.7871	1.4518	480.0
484.0	587.81	0.02009	0.76613	0.78622	469.1	734.7	1203.8	0.6696	0.7785	1.4481	484.0
488.0	610.10	0.02017	0.73641	0.75658	473.8	729.7	1203.5	0.6745	0.7700	1.4444	488.0
492.0	633.03	0.02026	0.70794	0.72820	478.5	724.6	1203.1	0.6793	0.7614	1.4407	492.0
496.0	656.61	0.02034	0.68065	0.70100	483.2	719.5	1202.7	0.6842	0.7528	1.4370	496.0
500.0	680.86	0.02043	0.65448	0.67492	487.9	714.3	1202.2	0.6890	0.7443	1.4333	500.0
504.0	705.78	0.02053	0.62938	0.64991	492.7	709.0	1201.7	0.6939	0.7357	1.4296	504.0
508.0	731.40	0.02062	0.60530	0.62592	497.5	703.7	1201.1	0.6987	0.7271	1.4258	508.0
512.0	757.72	0.02072	0.58218	0.60289	502.3	698.2	1200.5	0.7036	0.7185	1.4221	512.0
516.0	784.76	0.02081	0.55997	0.58079	507.1	692.7	1199.8	0.7085	0.7099	1.4183	516.0
520.0	812.53	0.02091	0.53864	0.55956	512.0	687.0	1199.0	0.7133	0.7013	1.4146	520.0
524.0	841.04	0.02102	0.51814	0.53916	516.9	681.3	1198.2	0.7182	0.6926	1.4108	524.0
528.0	870.31	0.02112	0.49843	0.51955	521.8	675.5	1197.3	0.7231	0.6839	1.4070	528.0
532.0	900.34	0.02123	0.47947	0.50070	526.8	669.6	1196.4	0.7280	0.6752	1.4032	532.0
536.0	931.17	0.02134	0.46123	0.48257	531.7	663.6	1195.4	0.7329	0.6665	1.3993	536.0
540.0	962.79	0.02146	0.44367	0.46513	536.8	657.5	1194.3	0.7378	0.6577	1.3954	540.0
544.0	995.22	0.02157	0.42677	0.44834	541.8	651.3	1193.1	0.7427	0.6489	1.3915	544.0
548.0	1028.49	0.02169	0.41048	0.43217	546.9	645.0	1191.9	0.7476	0.6400	1.3876	548.0
552.0	1062.59	0.02182	0.39479	0.41660	552.0	638.5	1190.6	0.7525	0.6311	1.3837	552.0
556.0	1097.55	0.02194	0.37966	0.40160	557.2	632.0	1189.2	0.7575	0.6222	1.3797	556.0
560.0	1133.38	0.02207	0.36507	0.38714	562.4	625.3	1187.7	0.7625	0.6132	1.3757	560.0
564.0	1170.10	0.02221	0.35099	0.37320	567.6	618.5	1186.1	0.7674	0.6041	1.3716	564.0
568.0	1207.72	0.02235	0.33741	0.35975	572.9	611.5	1184.5	0.7725	0.5950	1.3675	568.0
572.0	1246.26	0.02249	0.32429	0.34678	578.3	604.5	1182.7	0.7775	0.5859	1.3634	572.0
576.0	1285.74	0.02264	0.31162	0.33426	583.7	597.2	1180.9	0.7825	0.5766	1.3592	576.0
580.0	1326.17	0.02279	0.29937	0.32216	589.1	589.9	1179.0	0.7876	0.5673	1.3550	580.0
584.0	1367.7	0.02295	0.28753	0.31048	594.6	582.4	1176.9	0.7927	0.5580	1.3507	584.0
588.0	1410.0	0.02311	0.27608	0.29919	600.1	574.7	1174.8	0.7978	0.5485	1.3464	588.0
592.0	1453.3	0.02328	0.26499	0.28827	605.7	566.8	1172.6	0.8030	0.5390	1.3420	592.0
596.0	1497.8	0.02345	0.25425	0.27770	611.4	558.8	1170.2	0.8082	0.5293	1.3375	596.0
600.0	1543.2	0.02364	0.24384	0.26747	617.1	550.6	1167.7	0.8134	0.5196	1.3330	600.0
604.0	1589.7	0.02382	0.23374	0.25757	622.9	542.2	1165.1	0.8187	0.5097	1.3284	604.0
608.0	1637.3	0.02402	0.22394	0.24796	628.8	533.6	1162.4	0.8240	0.4997	1.3238	608.0
612.0	1686.1	0.02422	0.21442	0.23865	634.8	524.7	1159.5	0.8294	0.4896	1.3190	612.0
616.0	1735.9	0.02444	0.20516	0.22960	640.8	515.6	1156.4	0.8348	0.4794	1.3141	616.0
620.0	1786.9	0.02466	0.19615	0.22081	646.9	506.3	1153.2	0.8403	0.4689	1.3092	620.0
624.0	1839.0	0.02489	0.18737	0.21226	653.1	496.6	1149.8	0.8458	0.4583	1.3041	624.0
628.0	1892.4	0.02514	0.17880	0.20394	659.5	486.7	1146.1	0.8514	0.4474	1.2988	628.0
632.0	1947.0	0.02539	0.17044	0.19583	665.9	476.4	1142.2	0.8571	0.4364	1.2934	632.0
636.0	2002.8	0.02566	0.16226	0.18792	672.4	465.7	1138.1	0.8628	0.4251	1.2879	636.0
640.0	2059.9	0.02595	0.15427	0.18021	679.1	454.6	1133.7	0.8686	0.4134	1.2821	640.0
644.0	2118.3	0.02625	0.14644	0.17269	685.9	443.1	1129.0	0.8746	0.4015	1.2761	644.0
648.0	2178.1	0.02657	0.13876	0.16534	692.9	431.1	1124.0	0.8806	0.3893	1.2699	648.0
652.0	2239.2	0.02691	0.13124	0.15816	700.0	418.7	1118.7	0.8868	0.3767	1.2634	652.0
656.0	2301.7	0.02728	0.12387	0.15115	707.4	405.7	1113.1	0.8931	0.3637	1.2567	656.0
660.0	2365.7	0.02768	0.11663	0.14431	714.9	392.1	1107.0	0.8995	0.3502	1.2498	660.0
664.0	2431.1	0.02811	0.10947	0.13757	722.9	377.7	1100.6	0.9064	0.3361	1.2425	664.0
668.0	2498.1	0.02858	0.10229	0.13087	731.5	362.1	1093.5	0.9137	0.3210	1.2347	668.0
672.0	2566.6	0.02911	0.09514	0.12424	740.2	345.7	1085.9	0.9212	0.3054	1.2266	672.0
676.0	2636.8	0.02970	0.08799	0.11769	749.2	328.5	1077.6	0.9287	0.2892	1.2179	676.0
680.0	2708.6	0.03037	0.08080	0.11117	758.5	310.1	1068.5	0.9365	0.2720	1.2086	680.0
684.0	2782.1	0.03114	0.07349	0.10463	768.2	290.2	1058.4	0.9447	0.2537	1.1984	684.0
688.0	2857.4	0.03204	0.06595	0.09799	778.8	268.2	1047.0	0.9535	0.2337	1.1872	688.0
692.0	2934.5	0.03313	0.05797	0.09110	790.5	243.1	1033.6	0.9634	0.2110	1.1744	692.0
696.0	3013.4	0.03455	0.04916	0.08371	804.4	212.8	1017.2	0.9749	0.1841	1.1591	696.0
700.0	3094.3	0.03662	0.03857	0.07519	822.4	172.7	995.2	0.9901	0.1490	1.1390	700.0
702.0	3135.5	0.03824	0.03173	0.06997	835.0	144.7	979.7	1.0006	0.1246	1.1252	702.0
704.0	3177.2	0.04108	0.02192	0.06300	854.2	102.0	956.2	1.0169	0.0876	1.1046	704.0
705.0	3198.3	0.04427	0.01304	0.05730	873.0	61.4	934.4	1.0329	0.0527	1.0856	705.0
705.47*	3208.2	0.05078	0.00000	0.05078	906.0	0.0	906.0	1.0612	0.0000	1.0612	705.47*

*Critical temperature

Figure K.3, p.1

Table 2 Saturated Steam Pressure Table

[illegible]	[illegible]	Sat Liquid	Evap	Sat Vapor	Sat Liquid	Evap	Sat Vapor	Sat Liquid	Evap	Sat Vapor	[illegible]
0.08865	32.018	0.01602	3302.4	3302.4	0.0003	1075.5	1075.5	0.0000	[illegible]	[illegible]	0.08865
0.25	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1087.4	[illegible]	[illegible]	[illegible]	0.25
0.50	79.586	0.01607	[illegible]	[illegible]	47.623	1045.6	1096.3	[illegible]	[illegible]	[illegible]	0.50
1.0	[illegible]	[illegible]	333.59	333.60	[illegible]	[illegible]	1105.8	[illegible]	[illegible]	[illegible]	1.0
5.0	162.24	0.01640	[illegible]	[illegible]	130.20	1000.9	[illegible]	0.2349	[illegible]	[illegible]	5.0
10.0	[illegible]	0.016592	38.404	38.420	[illegible]	[illegible]	[illegible]	0.2836	[illegible]	[illegible]	10.0
14.696	[illegible]	[illegible]	26.782	26.799	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	14.696
15.0	[illegible]	[illegible]	26.274	26.290	[illegible]	969.7	[illegible]	[illegible]	[illegible]	[illegible]	15.0
20.0	[illegible]	0.016834	20.070	20.087	[illegible]	960.1	1156.3	0.3358	[illegible]	[illegible]	20.0
30.0	250.34	0.017009	13.7266	13.7436	218.9	945.2	1164.1	0.3682	[illegible]	[illegible]	30.0
40.0	[illegible]	[illegible]	10.4794	10.4965	236.1	933.6	1169.8	[illegible]	1.2844	[illegible]	40.0
50.0	[illegible]	0.017274	8.4967	8.5140	250.2	923.9	[illegible]	[illegible]	[illegible]	1.6586	50.0
60.0	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	915.4	1177.6	[illegible]	[illegible]	1.6440	60.0
70.0	302.93	[illegible]	[illegible]	[illegible]	[illegible]	907.8	1180.6	[illegible]	[illegible]	[illegible]	70.0
80.0	312.04	0.017573	5.4536	[illegible]	[illegible]	900.9	[illegible]	0.4534	[illegible]	1.6208	80.0
90.0	320.28	0.017659	4.8779	4.8953	[illegible]	894.6	1185.3	0.4643	1.1470	[illegible]	90.0
100.0	327.82	0.017740	4.4133	4.4310	298.5	888.6	[illegible]	0.4743	1.1284	1.6027	100.0
110.0	334.79	0.01782	4.0306	4.0484	[illegible]	[illegible]	1188.9	0.4834	[illegible]	1.5950	110.0
120.0	[illegible]	0.01789	3.7097	3.7275	[illegible]	877.8	1190.4	0.4919	1.0960	[illegible]	120.0
130.0	[illegible]	0.01796	3.4364	3.4544	319.0	[illegible]	[illegible]	0.4998	[illegible]	[illegible]	130.0
140.0	353.04	0.01803	3.2010	[illegible]	325.0	868.0	1193.0	[illegible]	[illegible]	[illegible]	140.0
150.0	358.42	0.01809	2.9958	3.0139	330.6	863.4	[illegible]	[illegible]	1.0554	[illegible]	150.0
160.0	[illegible]	0.01815	[illegible]	2.8336	[illegible]	859.0	[illegible]	0.5206	1.0435	[illegible]	160.0
170.0	[illegible]	0.01821	2.6556	2.6738	[illegible]	854.8	1196.0	0.5269	[illegible]	[illegible]	170.0
180.0	373.08	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1196.9	0.5328	[illegible]	[illegible]	180.0
190.0	[illegible]	0.01833	[illegible]	2.4030	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1.5498	190.0
200.0	381.80	0.01839	2.2689	2.2873	355.5	842.8	1198.3	0.5438	[illegible]	1.5454	200.0
210.0	[illegible]	0.01844	[illegible]	[illegible]	359.9	[illegible]	1199.0	0.5490	[illegible]	[illegible]	210.0
220.0	389.86	0.01850	2.06779	2.08629	[illegible]	[illegible]	1199.6	[illegible]	0.9834	1.5374	220.0
230.0	[illegible]	0.01855	[illegible]	1.99846	[illegible]	[illegible]	[illegible]	0.5588	0.9748	1.5336	230.0
240.0	[illegible]	0.01860	1.89909	[illegible]	[illegible]	828.4	1200.6	0.5634	[illegible]	1.5299	240.0
250.0	[illegible]	0.01865	1.82452	[illegible]	[illegible]	825.0	[illegible]	0.5679	0.9585	[illegible]	250.0
260.0	[illegible]	[illegible]	1.75548	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	0.9508	1.5230	260.0
270.0	407.80	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	270.0
280.0	[illegible]	0.01880	[illegible]	1.65049	[illegible]	[illegible]	[illegible]	0.5805	[illegible]	1.5166	280.0
290.0	414.25	0.01885	1.57597	1.59482	390.6	[illegible]	1202.6	[illegible]	[illegible]	[illegible]	290.0
300.0	[illegible]	0.01889	1.52384	1.54274	[illegible]	808.9	1202.9	0.5882	[illegible]	1.5105	300.0
350.0	[illegible]	[illegible]	1.30642	1.32554	[illegible]	[illegible]	1204.0	0.6059	[illegible]	1.4968	350.0
400.0	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1204.6	[illegible]	[illegible]	1.4847	400.0
450.0	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1204.8	0.6360	[illegible]	[illegible]	450.0
500.0	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	0.6490	[illegible]	[illegible]	500.0
550.0	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	550.0
600.0	486.20	[illegible]	0.74962	0.76975	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	600.0
650.0	[illegible]	[illegible]	[illegible]	0.70843	481.9	[illegible]	[illegible]	0.6828	[illegible]	[illegible]	650.0
700.0	503.08	0.02050	0.63505	0.65556	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1.4304	700.0
750.0	510.84	[illegible]	0.58880	0.60949	[illegible]	699.8	[illegible]	[illegible]	[illegible]	[illegible]	750.0
800.0	[illegible]	[illegible]	0.54809	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	800.0
850.0	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	850.0
900.0	[illegible]	[illegible]	0.47968	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	900.0
950.0	538.39	[illegible]	0.45064	[illegible]	[illegible]	[illegible]	[illegible]	0.7358	[illegible]	[illegible]	950.0
1000.0	[illegible]	[illegible]	0.42436	0.44596	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1000.0
1050.0	[illegible]	[illegible]	0.40047	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1050.0
1100.0	[illegible]	[illegible]	0.37863	0.40058	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1100.0
1150.0	[illegible]	[illegible]	0.35859	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1150.0
1200.0	[illegible]	[illegible]	0.34013	[illegible]	[illegible]	[illegible]	1184.8	[illegible]	[illegible]	[illegible]	1200.0
1250.0	572.38	0.02250	0.32306	[illegible]	578.6	603.8	[illegible]	0.7780	[illegible]	[illegible]	1250.0
1300.0	[illegible]	0.02269	[illegible]	[illegible]	[illegible]	594.6	[illegible]	0.7843	[illegible]	[illegible]	1300.0
1350.0	[illegible]	0.02288	0.29250	[illegible]	[illegible]	585.4	1177.8	[illegible]	0.5620	[illegible]	1350.0
1400.0	[illegible]	[illegible]	[illegible]	[illegible]	598.8	[illegible]	[illegible]	0.7966	[illegible]	[illegible]	1400.0
1450.0	[illegible]	[illegible]	0.26584	[illegible]	[illegible]	[illegible]	1172.6	0.8026	[illegible]	[illegible]	1450.0
1500.0	[illegible]	0.02346	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	0.8085	[illegible]	[illegible]	1500.0
1550.0	600.59	0.02366	0.24235	[illegible]	616.0	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1550.0
1600.0	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	0.5076	[illegible]	1600.0
1650.0	[illegible]	[illegible]	[illegible]	[illegible]	630.4	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1650.0
1700.0	[illegible]	[illegible]	0.21178	[illegible]	[illegible]	[illegible]	[illegible]	0.8309	0.4867	[illegible]	1700.0
1750.0	[illegible]	0.02450	0.20263	[illegible]	[illegible]	[illegible]	[illegible]	0.8363	0.4765	[illegible]	1750.0
1800.0	[illegible]	[illegible]	0.19390	[illegible]	[illegible]	503.6	[illegible]	[illegible]	[illegible]	[illegible]	1800.0
1850.0	624.83	[illegible]	0.18558	[illegible]	[illegible]	494.6	[illegible]	0.8470	[illegible]	[illegible]	1850.0
1900.0	628.56	[illegible]	[illegible]	0.20278	660.4	[illegible]	[illegible]	[illegible]	0.4459	[illegible]	1900.0
1950.0	[illegible]	[illegible]	0.16999	[illegible]	[illegible]	[illegible]	[illegible]	0.8574	0.4358	[illegible]	1950.0
2000.0	635.80	0.02565	0.16266	[illegible]	[illegible]	[illegible]	[illegible]	0.8625	0.4256	[illegible]	2000.0
2100.0	642.76	0.02615	0.14885	[illegible]	683.8	[illegible]	1130.5	[illegible]	0.4053	[illegible]	2100.0
2200.0	649.45	0.02669	[illegible]	[illegible]	695.5	[illegible]	[illegible]	0.8828	0.3848	[illegible]	2200.0
2300.0	655.89	[illegible]	0.12406	[illegible]	[illegible]	406.0	[illegible]	0.8929	0.3640	[illegible]	2300.0
2400.0	[illegible]	0.02790	0.11287	0.14076	[illegible]	384.8	[illegible]	[illegible]	0.3430	[illegible]	2400.0
2500.0	668	0.02859	[illegible]	0.13068	[illegible]	361.6	[illegible]	[illegible]	[illegible]	[illegible]	2500.0
2600.0	[illegible]	0.02938	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	2600.0
2700.0	[illegible]	0.03029	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	0.9356	[illegible]	[illegible]	2700.0
2800.0	684.96	0.03134	[illegible]	0.10209	[illegible]	[illegible]	1055.8	0.9468	[illegible]	1.1958	2800.0
2900.0	[illegible]	[illegible]	0.06158	[illegible]	[illegible]	[illegible]	[illegible]	0.9588	[illegible]	[illegible]	2900.0
3000.0	695.33	0.03428	0.05073	0.08500	801.8	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	3000.0
3100.0	700.28	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	0.9914	0.1460	[illegible]	3100.0
3200.0	705.08	[illegible]	[illegible]	[illegible]	875.5	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	3200.0
3208.2*	705.47	0.05078	0.00000	0.05078	906.0	0.0	906.0	[illegible]	0.0000	[illegible]	3208.2*

*Critical pressure

Figure K.3, P. 2

Table 3 Superheated Steam

Abs Press Lb/Sq In (Sat Temp)		Sat Water	Sat Steam	200	250	300	350	400	450	500	600
				Temperature — Degrees Fahrenheit							
1 (101.74)	Sh			98.26	148.26	198.26	248.26	298.26	348.26	398.26	498.26
	v	0.01614	333.6	392.5	422.4	452.3	482.1	511.9	541.7	571.5	631.1
	h	69.70	1105.8	1150.2	1172.9	1195.7	1218.7	1241.8	1265.1	1288.6	1336.1
	s	0.1326	1.9781	2.0509	2.0841	2.1152	2.1445	2.1722	2.1985	2.2237	2.2708
5 (162.24)	Sh			37.76	87.76	137.76	187.76	237.76	287.76	337.76	437.76
	v	0.01641	73.53	78.14	84.21	90.24	96.25	102.24	108.23	114.21	126.15
	h	130.20	1131.1	1148.6	1171.7	1194.8	1218.0	1241.3	1264.7	1288.2	1335.9
	s	0.2349	1.8443	1.8716	1.9054	1.9369	1.9664	1.9943	2.0208	2.0460	2.0932
10 (193.21)	Sh			6.79	56.79	106.79	156.79	206.79	256.79	306.79	406.79
	v	0.01659	38.42	38.84	41.93	44.98	48.02	51.03	54.04	57.04	63.03
	h	161.26	1143.3	1146.6	1170.7	1193.7	1217.1	1240.6	1264.1	1287.8	1335.5
	s	0.2836	1.7879	1.7928	1.8273	1.8593	1.8892	1.9173	1.9439	1.9692	2.0166
14.696 (212.00)	Sh				38.00	88.00	138.00	188.00	238.00	288.00	388.00
	v	0.01672	26.80		28.42	30.52	32.60	34.67	36.72	38.77	42.86
	h	180.17	1150.5		1168.8	1192.8	1216.6	1239.9	1263.6	1287.4	1335.2
	s	0.3121	1.7568		1.7838	1.8158	1.8459	1.8743	1.9010	1.9265	1.9739
15 (213.03)	Sh				36.97	86.97	136.97	186.97	236.97	286.97	386.97
	v	0.01673	26.290		27.837	29.899	31.939	33.963	35.977	37.985	41.986
	h	181.21	1150.9		1168.7	1192.5	1216.2	1239.9	1263.6	1287.3	1335.2
	s	0.3137	1.7552		1.7804	1.8134	1.8437	1.8720	1.8988	1.9242	1.9717
20 (227.96)	Sh				22.04	72.04	122.04	172.04	222.04	272.04	372.04
	v	0.01683	20.087		20.788	22.356	23.900	25.428	26.946	28.457	31.466
	h	196.27	1156.3		1167.1	1191.4	1215.4	1239.2	1263.0	1286.9	1334.9
	s	0.3358	1.7320		1.7475	1.7805	1.8111	1.8397	1.8666	1.8921	1.9397
25 (240.07)	Sh				9.93	59.93	109.93	159.93	209.93	259.93	359.93
	v	0.01693	16.301		16.558	17.829	19.076	20.307	21.527	22.740	25.153
	h	208.52	1160.6		1165.6	1190.2	1214.5	1238.5	1262.5	1286.4	1334.6
	s	0.3535	1.7141		1.7212	1.7547	1.7856	1.8145	1.8415	1.8672	1.9149
30 (250.34)	Sh					49.66	99.66	149.66	199.66	249.66	349.66
	v	0.01701	13.744			14.810	15.859	16.892	17.914	18.929	20.945
	h	218.82	1164.1			1189.0	1213.6	1237.8	1261.9	1286.0	1334.2
	s	0.3682	1.6995			1.7334	1.7647	1.7937	1.8210	1.8467	1.8946
35 (259.29)	Sh					40.71	90.71	140.71	190.71	240.71	340.71
	v	0.01708	11.896			12.654	13.562	14.453	15.334	16.207	17.939
	h	228.03	1167.1			1187.8	1212.7	1237.1	1261.3	1285.5	1333.9
	s	0.3809	1.6872			1.7152	1.7468	1.7761	1.8035	1.8294	1.8774
40 (267.25)	Sh					32.75	82.75	132.75	182.75	232.75	332.75
	v	0.01715	10.497			11.036	11.838	12.624	13.398	14.165	15.685
	h	236.14	1169.8			1186.6	1211.7	1235.4	1260.8	1285.0	1333.6
	s	0.3921	1.6765			1.6992	1.7312	1.7608	1.7883	1.8143	1.8624
45 (274.44)	Sh					25.56	75.56	125.56	175.56	225.56	325.56
	v	0.01721	9.399			9.777	10.497	11.201	11.892	12.577	13.932
	h	243.49	1172.1			1185.4	1210.4	1235.7	1260.2	1284.6	1333.3
	s	0.4021	1.6671			1.6849	1.7173	1.7471	1.7748	1.8010	1.8492
50 (281.02)	Sh					18.98	68.98	118.98	168.98	218.98	318.98
	v	0.01727	8.514			8.769	9.424	10.062	10.688	11.306	12.529
	h	250.21	1174.1			1184.1	1209.9	1234.9	1259.6	1284.1	1332.9
	s	0.4112	1.6586			1.6720	1.7048	1.7349	1.7628	1.7890	1.8374
55 (287.07)	Sh					12.93	62.93	112.93	162.93	212.93	312.93
	v	0.01733	[illegible]			7.945	8.546	9.130	9.702	10.267	11.381
	h	256.43	[illegible]			1182.9	1208.9	1234.2	1259.1	1283.6	1332.6
	s	0.4196	[illegible]			1.6601	1.6933	1.7237	1.7518	1.7781	1.8266
60 (292.71)	Sh					7.29	57.29	107.29	157.29	207.29	307.29
	v	0.01738	7.174			7.257	7.815	8.354	8.881	9.400	10.425
	h	262.21	1177.6			1181.6	1208.0	1233.5	1258.5	1283.2	1332.3
	s	0.4273	1.6440			1.6492	1.6934	1.7134	1.7417	1.7681	1.8168
65 (297.98)	Sh					2.02	52.02	102.02	152.02	202.02	302.02
	v	0.01743	6.653			6.675	7.195	7.697	8.186	8.667	9.615
	h	267.63	1179.1			1180.3	1207.0	1232.7	1257.9	1282.7	1331.9
	s	0.4344	1.6375			1.6390	1.6731	1.7040	1.7324	1.7590	1.8077
70 (302.93)	Sh						47.07	97.07	147.07	197.07	297.07
	v	0.01748	6.205				6.664	7.133	7.590	8.039	8.922
	h	272.74	1180.6				1206.0	1232.0	1257.3	1282.2	1331.6
	s	0.4411	1.6316				1.6640	1.6951	1.7237	1.7504	1.7993
75 (307.61)	Sh						42.39	92.39	142.39	192.39	292.39
	v	0.01753	5.814				6.204	6.645	7.074	7.494	8.320
	h	277.56	1181.9				1205.0	1231.2	1256.7	1281.7	1331.3
	s	0.4474	1.6260				1.6554	1.6863	1.7156	1.7424	1.7915

Sh = superheat, F
v = specific volume, cu ft per lb

h = entr
s = entr

L. FUEL HANDLING AND CORE PARAMETERS (13)

L.1 The Safety Limits of the reactor core are based partially or two parameters that are not directly observable: DNBR and power peaking.

a. What four observable parameters are used to determine DNB. (2)

b. What limits are imposed to assure that power peaking limits are not exceeded? (1)

L.2 With respect to refueling shutdown containment integrity:

a. May the letdown valves, Mu-V2A, 2B, & 3, be opened even though they are not operable (automatic closure on ESAS signal is blocked)? Explain. (1.5)

b. Per the Technical Specifications, what two checks must be made on all manual containment isolation valves prior to criticality following refueling shutdown. (.5)

L.3 During power operations one safety rod drops into the core. The unit runs back to 55 percent power as expected. Efforts to withdraw the rod are unsuccessful.

a. Is this control rod considered inoperable? Give the basis for your answer. (1)

b. Explain whether or not continued operation at 55% power is allowed. Include any limitations that either allow or preclude operation. (1)

c. Would your answer change if the rod in question was located in regulating group 5? Explain your answer. (1)

L.4 With respect to the Spent Fuel Handling Bridge and Trolley:

a. Describe the methods of assuring that the trolley is properly positioned over a designated storage rack. (1)

b. Why must the control panel be energized when moving the bridge by hand? (.5)

c. What mechanical or electrical interlocks exist to prevent hitting the new fuel elevator when it is in the Up position? (.5)

L.5 a. What is the purpose of the power level cutoff hold point? (1)

b. Give an example of how violating this hold point could result in undesirable consequences. (2)

M. ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS (20)

M.1 When a heat balance calibration is performed to adjust the power range NI's to agree with the core thermal power, the Tech Specs (1.5.6) define a weighting factor, α, that is used in the core thermal power equation.

a. Define the weighting factor α (1)

b. Show the equation that is used when power is >15% but <50%. (1)

M.2 Administrative procedure 1001 addresses the situation of encountering a "Temporary,On-the-Spot" change to an operating procedure. Section 3.6.4.1 of AP 1001 discusses the use of such TCN's (Temporary Change Notice).

a. Give an example of when a TCN is not required. (1)

b. What must be done in place of using a TCN? (1.5)

M.3 The following LER's from TMI-1 were reported to the NRC during the two year period preceding the Unit 2 accident. As Shift Supervisor, for each of these incidents, describe:

i. Your immediate actions or those actions you would direct your operators to take.

ii. The Technical Specifications that are applicable or have been violated (do not include reporting requirements).

a. At 1452 hours on July 23, 1978 it was discovered that air valve on 'A' Emergency Diesel Generator had been tagged shut at 1115 hours during I&C System Maintenance. Apparently, Operations Personnel had not been notified. (2)

b. During a power level reduction to 65% for turbine stop value testing, the Quadrant Power Tilt exceeded 3.64% for a total of 50 minutes and reached a maximum value of 3.79%. (2)

c. In September 1978 during transmitter calibration an actual level of 12.42 ft. was discovered in the 'B' Core Flood Tank. (2)

d. During RCS heatup on March 27, 1979 following a refueling outage, MS-V-6, the steam regulating valve to the EFW pump was found closed. Apparently it had been closed for maintenance and although personnel safety tags had been removed the switching and tagging procedure did not require verification of safety related valve positions. (2)

M.4 With regard to "Controlled Key Locker Control", as specified in AP1011,

a. Who may authorize the change in position of locked valves? (.5)

b. Who is responsible for overall control of the controlled key locker? (.5)

c. Describe the administrative control over a key that remains captured in a lock for more than one shift. Include who has responsibility for the key on the next shift. (1)

M.5 Technical Specifications 3.5.4.2 states the minimum incore detectors for radial tilt shall be arranged in the following manner:

a. Two sets of four detectors shall lie in each core half. Each set of four shall lie in the same axial plane. The two sets in the same core half may lie in the same axial plane.

b. Detectors in the same plane shall have quarter core radial symmetry.

On the attached Figure M-5, sketch the minimum incore nuclear instrumentation that will satisfy this specification. (2)

M.6 Under what conditions are the following excluded from the control and documentation requirements of AP 1013 "Bypass of Safety Functions and Jumper Control"?

a. Electrical jumpers (EJ), Temporary Mechanical Modifications (TMM) and/or lifted leads. (1)

b. Spool pieces or blank flanges. (1)

M.7 a. When may a "DO NOT OPERATE" tag be used instead of a "CAUTION" tag? (.5)

b. When may "CAUTION" tags be used instead of an "Instrument Out of Service" sticker? (.5)

c. What operations personnel may decide whether a "CAUTION" tag or "DO NOT OPERATE" tag needs to be used? (.5)

N. THEORY OF FLUIDS AND THERMODYNAMICS (13)

N.1 What is the function and operating principle of a thermal sleeve? (1.5)

N.2 Reynolds number is used to determine the characteristics of fluid flow, i.e., whether the flow is turbulent or laminar.

a. Name three general parameters that are used to determine Reynolds number. (There are actually 5 parameters, the equation is not necessary). (1.5)

b. Generally, what number or range of Reynolds Numbers is used to distinguish laminar from turbulent flow? (0.5)

c. Turbulent flow has better heat removal characteristics. Why? (1.0)

N.3 Figure N-3 is a temperature - entropy diagram showing an idealized Carnot Cycle a-b-c-d. Your system however more approximates a Rankine cycle which include inefficiencies and superheat.

a. Sketch on Figure N-3 how your steam cycle differs from the Carnot cycle and more approximates the Rankine cycle. To begin your diagram, the line d-a' shows the condensation of the steam. As superheat, include only the superheated steam leaving the OTSG. (2.0)

b. Show on the diagram two additional realities of your steam cycle: (1.0)

i) A slight amount of condensate depression.
ii) Additional superheat from the Moisture-Separators-Reheaters.

N.4 Operating procedure OP 1102-16 "RCS Natural Circulation Cooling" addresses the expected plant response to initiation and cooldown by natural circulation.

a. One statement says "Prior to securing the reactor coolant pumps, establish conditions in the RCS favorable to the initiation of natural circulation." List two of these conditions and explain why they are favorable for the initiation of natural circulation. (2.0)

b. The reactor has been tripped, the 4 RCP's tripped, natural circulation established and header pressure biased controlled @ 1010 psig. Sketch and explain on Figure N.4 the expected behavior of Pressurizer level, T_H and T_C if the 125 psig header pressure bias is suddenly removed (for example by resetting the reactor trip). (1.5)

N.5 a. List the design flow capacity of the HPI pumps. (0.5)

b. If a LOCA occurred, what approximate flow indication would you expect to read on each of the flow indicators for the HPI lines if RCS pressure is about 1100 psi? Explain. (1.5)

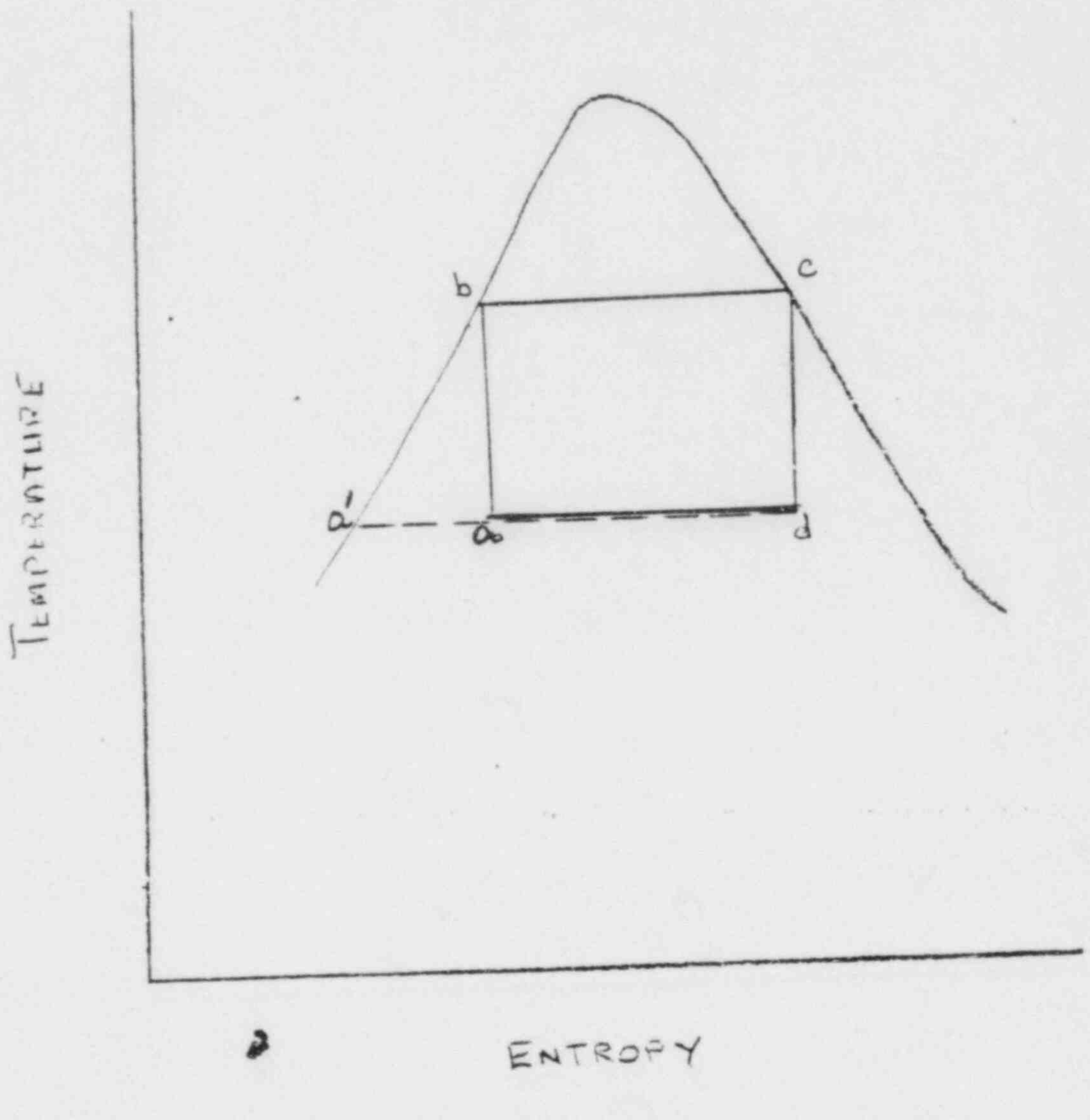


FIG N-3

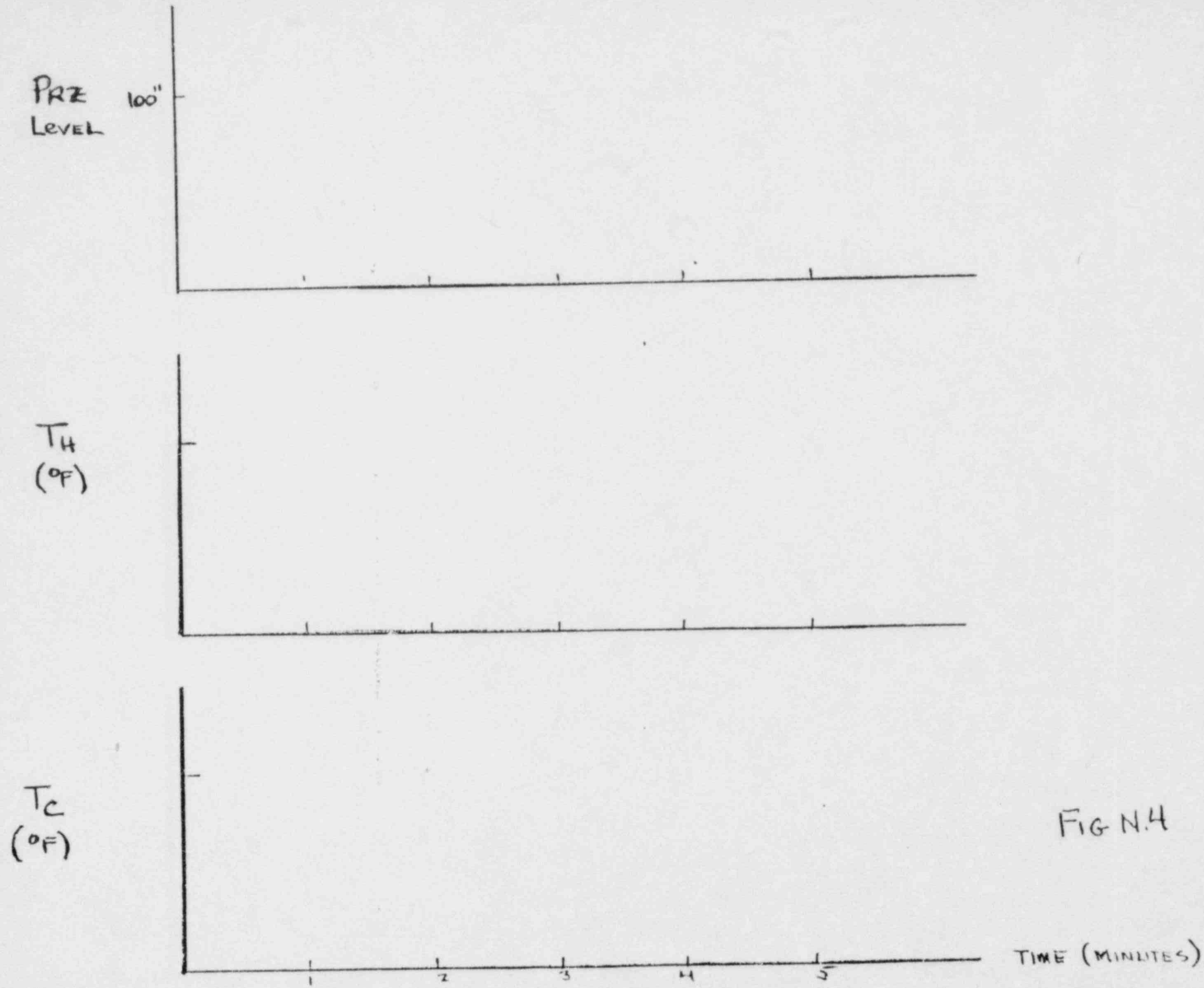


FIG N.4

Answers

I.1 OARP pg. 17

1. The magnitude of the change in keff.
2. Actual value of keff after the reactivity change is made.

I.2 OARP pg. 27

L_T (non-leakage) probabilities decrease slightly since the moderator temperature increase decreases the moderator and boron densities. More dramatic than the reactivity effect is the response seen by the out-of-core detectors.

Check basis

I.3 TS a. pg 3-7 No possibility of an accidental criticality as a result of decrease in RCS pressure.

b. EOL, MTC is most negative ∴ positive addition. Since boron concentration is very low the amount of reactivity added will be a maximum depending on the boil off in the OTSG's. If feedwater is immed. secured consequences will be minimal.

c. With that inventory in the OTSG's the cooldown will be greater and the +reactivity addition much greater.

I.4 a. i. Pg 82 Pu-239 (explan. is for Sm^{149} but true for Xe^{135}) - slightly higher fission yield curve.

ii. Effect on SDM - immed. following trip - none at equilibrium but soon there after starts building in to increase SDM up to peak Xe.

b. Pg 29

i. 1. Becomes more negative due to Boron conc. reduction, and

2. Buildup of Pu 240 & FP's that have high resonance peaks.

ii. Effect on SDM - reduces SDM since coeff becomes more negative - cooldown to 555^{o}F (?) (have seen 545^{o}F elsewhere) adds more +

c. Pg 25-31

i. Comprised of MTC + FTC, MTC explained above, FTC ↑ slightly due to Pu 240.

ii. SDM - Substantial increase in power deficit directly reduces SDM. For same total rod worth, difference in P.D. is 0.72% which is amount SDM is reduced.

J.1 a. $3e^{-\eta t} = 3e^{-.693(30)/8} = .223$ curies

b. Yes. Tech Specs allow up to 10 curies to be released. see appendix B tech spec.

J.2 Advanced HP training part 412 - 1C.

1. The person conducting the survey shall review the last survey performed in the area as guidelines for choosing the appropriate protective clothing and respiratory protection equipment, unless there is an indication that significant changes in the radiological conditions has occured in which case more restrictive protective clothing and respirators may be warranted.

2. For the purposes of exposure documentation, the individual performing the survey shall document his exposure received while performing the survey on the prepared RWP.

J.3 a. RO2 - E520 PIC 6 - telelector

b. Pac 4

c. PNR 4 or PAC 4

J.4 5 rems to whole body, 30 rems to skin, 75 rems to extremeties. See 50.72

J.5 isoform - ie solid, liquid or gas
distance from person
isotope of element
type and quanity of radiation emmitted
energy of radiation

J.6 a.1. TS 3.1.6 - primary leakage - 1 gpm total both OTSG
2. TS ~~?~~ 3.13 secondary activity - 1.0 μCi/gm I^{131}

b.1. Indicates a weakness in RCS boundary, small leaks - big leaks - also secondary activity buildup.

~~2. Steam line break 10 CFR 100 limits.~~

2. ~~Loss~~ of Load from 100% power - Stm Safeties Lifting for 3 min. pg 3-58

K.1 a. Imbalance will continue to shift more negative; Xenon will buildup in the top of the core and burn out in the bottom

b. 1. reduce power by boration (Rods out)

2. adjust APSR's and explain (core power - higher imbalance)

K.2 OARP Solid Ops. Module 4

a. Increase RCS inventory, a sharp increase in RCS pressure will be experienced (1103-5, B.6)

b. pg.3 1. reduce charging
2. increase letdown
3. open PORV
4. increase cooling
5. decrease seal injection

c. 1. energize heaters
2. increase letdown
3. verify steam bubble when pressurizer level comes back on scale - stable RCS pressure w/letdown greater than makeup, sat. pressure for sat. temperature.

K.3 (Make sure 1035 psig is below M/S Safety setpoint)

After a reactor trip, secondary pressure will be maintained at 1035 psig rather than 1010psig. This will result in a higher RCS temp. after a reactor trip and higher prz level. (OTSG pressure will govern RCS temp. at sat.) Less shrinkage in prz. per Figure K.3; before about 548°F; after about 550°F

OP 1102-16

K.4 a. RCS flow has been reduced due to the loss of the RCP's; less heat is removed by OTSG's, temperature will increase.

b. No; this is normal and expected during the transition to natural circulation which will take about 20 - 30 minutes.

K.5 a. No, trip of the RCP's is not required unless ESAS low RCS pressure setpoint is reached. In the case of high RB pressure alone you don't have an inventory problem which could lead to high void fractions

b. No, sufficient void fraction may exist such that core uncovery may occur.

?

L.1 a. TS pg 2-1

Neutron power
RCS flow
Temp.
Pressure

b. T.S. pg. 2-2
Limits have been placed on the basis of the reactor power imbalance produced by the power peaking

L.2 a. Yes, the TS only address penetrations providing direct access from the containment atmosphere to the outside atmosphere. TS 3.8.7

b. Verify that those that should be closed are 1) closed and 2) conspicuously marked TS 3.6.5

L.3 a. Yes, per TS 4.7.1.2, because it is greater than 9" unaligned from the rest of the group which must be fully w/d while critical (TS 3.1.3.5).

b. Allowed, per T.S. 3.5.2.2.e if P< 60% and quadrant tilts are acceptable.

c. Yes, operations are precluded per T.S. 3.5.2.2.f since it requires the other rods in the group to be positioned such that the inoperable rod is maintained within 9" of the group avg. Group 5 insertion to 9" is not allowed per T.S. rod insertion limits.

L.4 a. Target on south trolley rail - letter & black line,
P. 5 sight on trolley - metal plate w/ square cut out & cross - properly aligned when letter is visible & vertical portion of the cross matches the line.

p. 8 b. The selsyn units will lose their calibration.
p. 6 c. No mechanical, but bridge & trolley can't move unless the hoists are up.

L.5 a. PLCO hold point assures that acceptable peaking factors are maintained through xenon transients. PLCO hold assures a minimum xenon transient since Xe builds to equilibrium.

b. After load decrease, xenon could be beyond its peak in the upper half of the core. Increasing power by withdrawing rods will increase power in upper half of core & will burn out xenon in addition to its rapid decline due to decay. This will cause rapid power increase in the top of the core which could result in high local peak powers.

M.1 T.S. 1.5.6

a. = the weighted primary and secondary heat balance considering heat losses.

b. Core Thermal Power = $\alpha Q_{sec} + (1-\alpha) Q_{prim}$

$$\alpha = \frac{\text{Power} - 15\%}{85}$$

M.2 a. AP 1001; Temporary changes to procedure valve checklists do not require a TCN.

b. Deviations noted in ink on applicable checkoff sheets and must be reviewed and approved by 2 licensed personnel one of which is SRO (SS/Shift Foreman). Deviations logged in Shift Foreman's log.

M.3 a. i. Immediately test operability of 'B' diesel-generator; also clear tags of applicable, report to duty off/Ops Super.

LER 78-022

ii. T.S. 3.7.2.C. 'A' D-G removed from service without testing operability of redundant D-G.

b. i. LER 78-023; Nothing had to be done since Tech Specs require power to be reduced to below power level cutoff which it already was.

? ii. T.S. 3.5.2.4.a&d Xenon burnout and rod withdrawl associated with returning to power returned the QPT to within limits.
Return? not required

c. i. LER 78-027; LER says borated water was added to core flood tank to bring it back within T.S. limits. Is there a procedure for doing this? Can it be done within time limits? — No — must start load reduction

ii. T.S. 3.3.1.2.a. $1040 \pm 30\ ft^3$ = 12.55 ft. minimum

d.i. LER 78- ; Immediately open valve, clear tags, check logs, initiate ? TCN's to insure valves are opened. [illegible]

ii. T.S. 3.4.2 Turbine driven EFW pump must be operable. Can it be declared operable on auxiliary steam? No

M.4 a. AP1011 Shift Foreman

b. Shift Foreman (may delegate key issuance to CRO - SS may function as Shift Foreman)

c. pg. 3.8 Note to that effect will be made in controlled key log. Responsibility will be transferred to Shift Foreman relieving the Foreman who authorized the key issue.

M.5 On Figure.

M.6 a. AP 1013; Positively identified in approved procedures. Other procedures must meet five criteria (not necessary to specify each criteria)

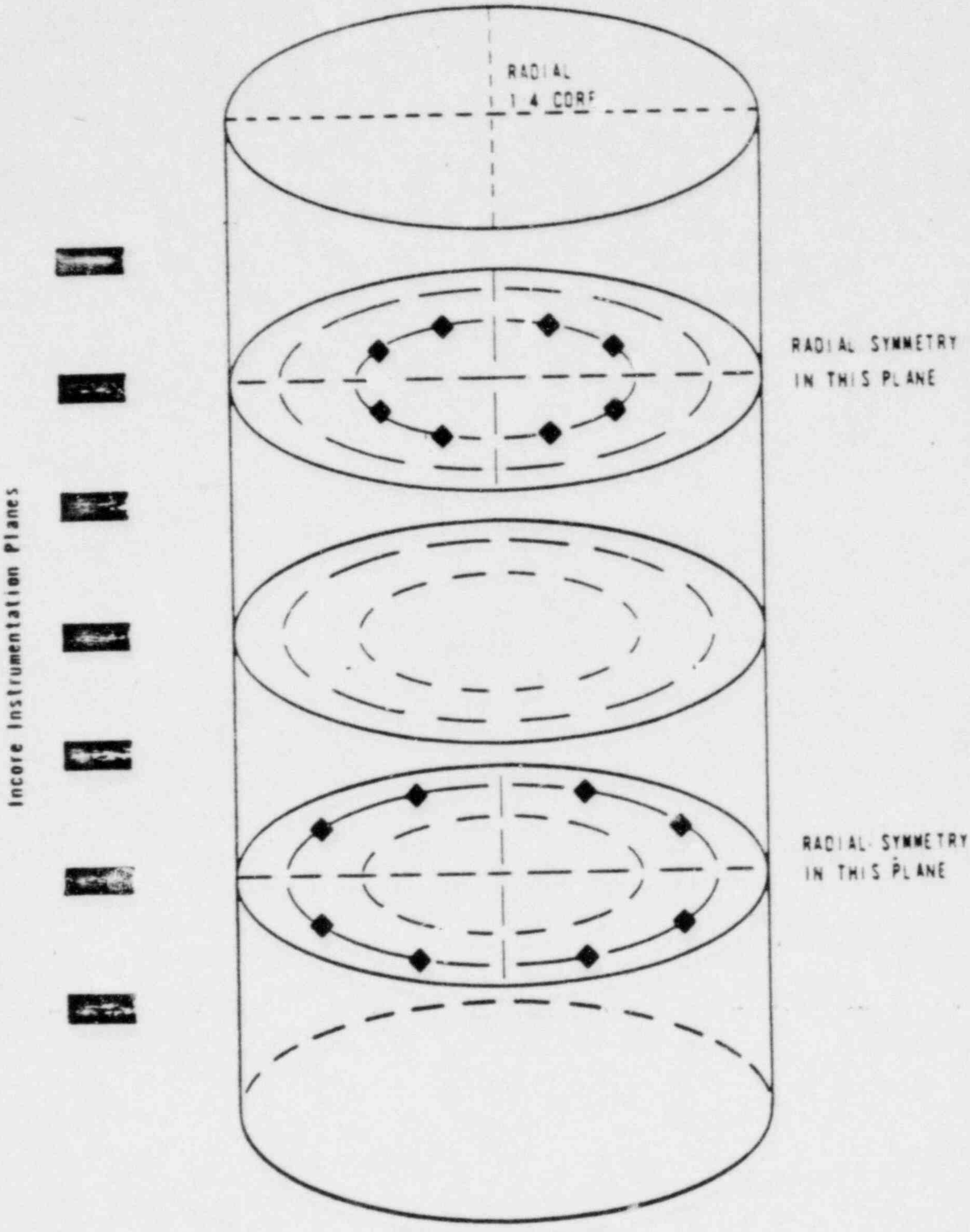
b. Those that are a design part of the system and are installed or removed by operating, maintatnance, or surveillance procedures.

M.7 a. AP1037 - When used in environments where CAUTION tags may easily deteriorate under extended use.

b. They cannot - Instrument Out-Of-Service stickers are covered by a separate procedure.

c. pg. 3.0; SS,SF, or CRO.

M-7



INCORE INSTRUMENTATION SPECIFICATION
RADIAL FLUX INDICATION
THREE MILE ISLAND NUCLEAR STATION UNIT 1

FIGURE 3.5-2

N.1 Functions to reduce thermal stresses at the connecting point of hot and cold piping. It does this by surrounding the cold pipe at the junction point with a sleeve. One end of the sleeve connects to the hot pipe while the other connects to the cold pipe. The sleeve acts as a transition piece to reduce the ΔT.

N.2 a. OARP pg. 63

1. average velocity; V_{av}
2. pipe diameter; D
3. kinematic viscosity (ν)
4. density (ρ)
5. dynamic viscosity (μ)

$$Re = \frac{V_{av} D}{\nu} \quad \text{or} \quad = \frac{V_{av} D \rho}{\mu}$$

b. OARP pg. 64

Laminar; Re < 2000
2000 < Re transition > 3500

Turbulent > 3500

c. OARP pg.65

Amount of mixing-- two different velocity profiles, turbulent flow promotes heat transfer to more mass of heat absorbing medium.

N.3 On Figures; ref: OARP pg 52 - 54

N.4 OP 1102-16 pg 8.0

a. 1. Adequate subcooling - without subcooling void formation may occur that would tend to block natural circulation particularly at higher elevations such as hot leg

2. OTSG level raised to 50% on operate range - provide large heat sink for RCS

b. See Figure

N.5 a. Module 6 GPU Letter dated 1/3/80

300 GPM @ 1800 psig

b. Table 4; about 200GPM per injection line; or

Pump head curve for 1100 psi x 3 pumps divided by 4 injection lines

Exhibit II-3

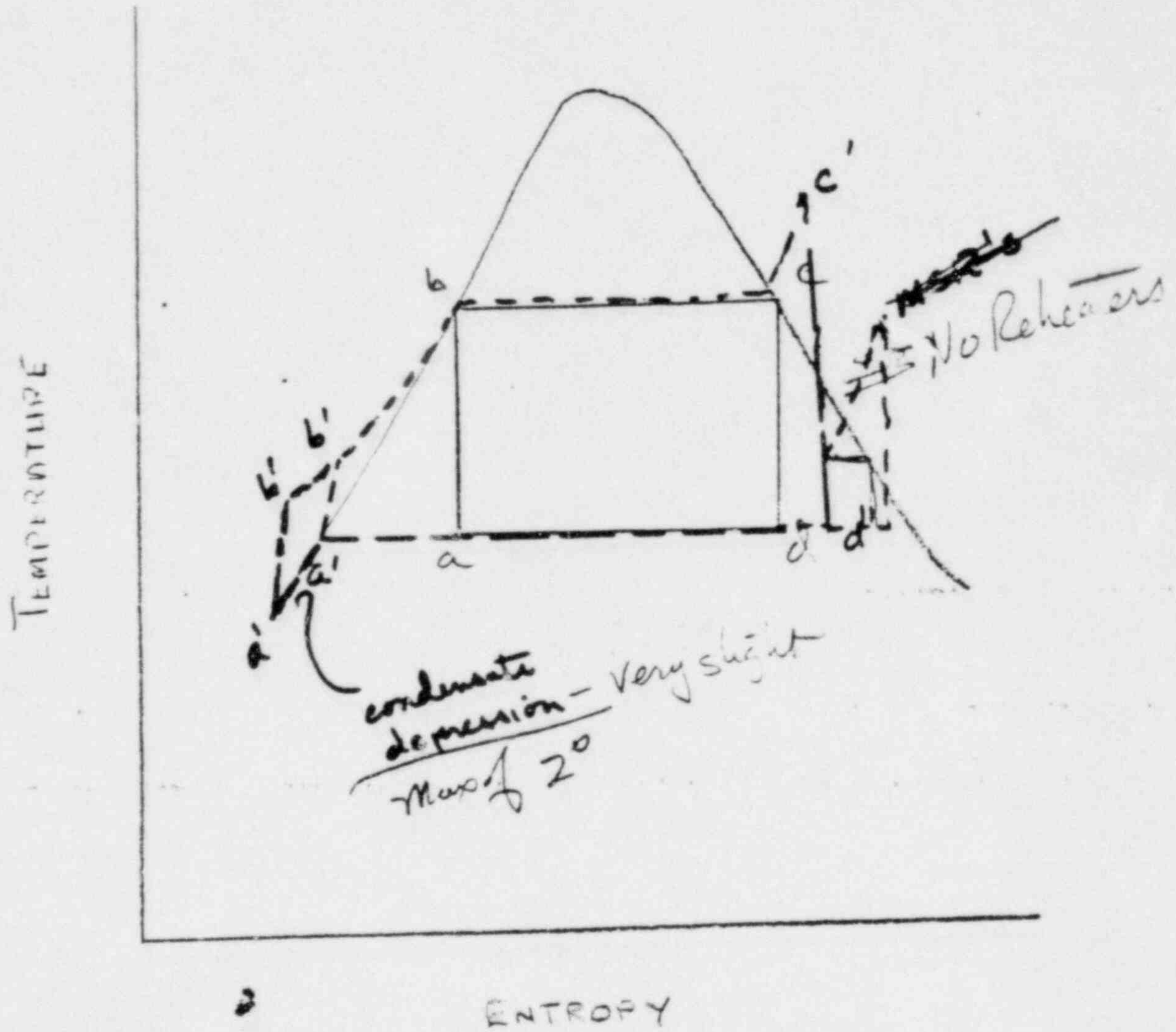
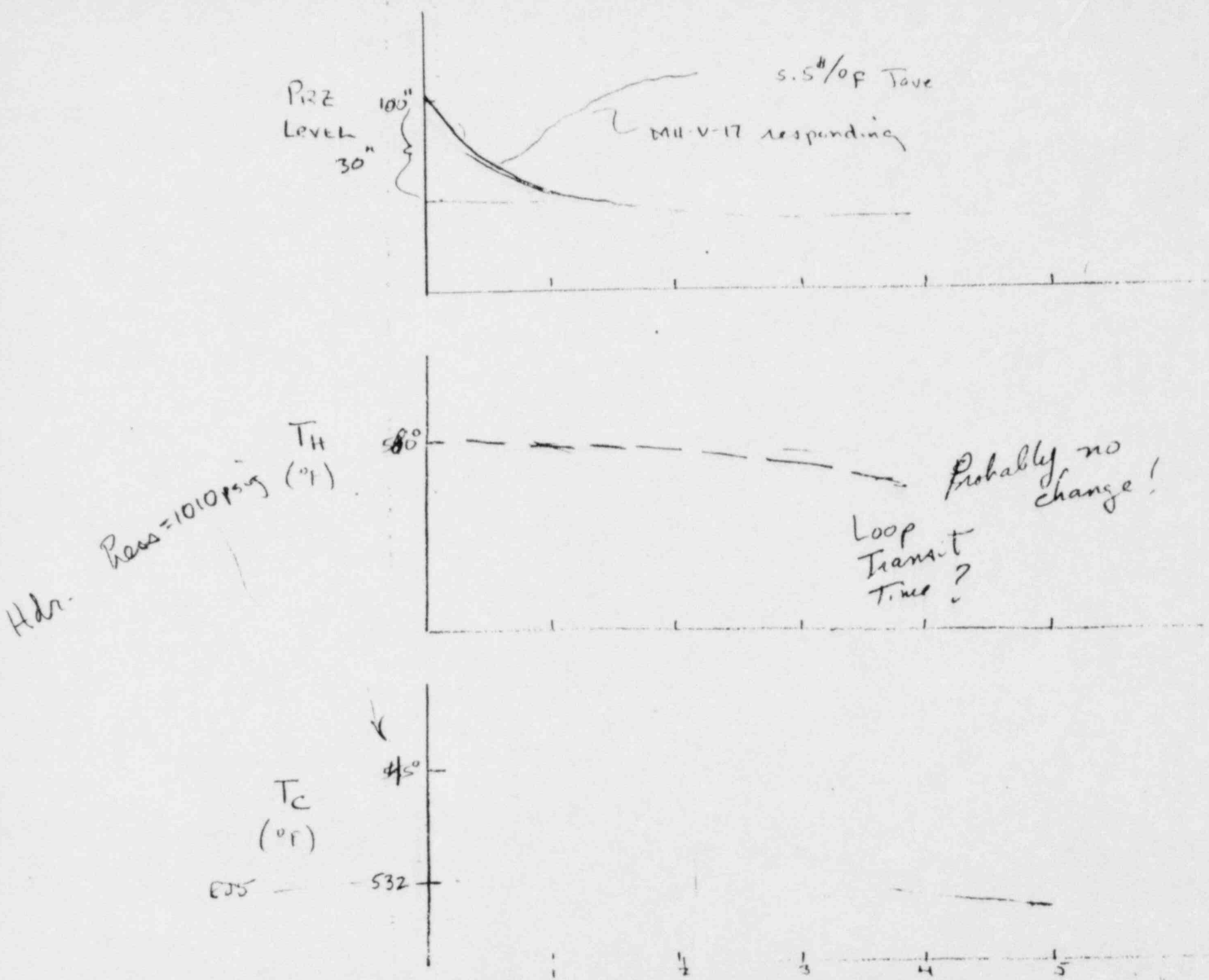


FIG II-3



Pzr Level
100"
30"
5.5"/°F Tave
MU-V-17 responding
T_H (°F)
580°
Probably no change!
Loop Transit Time?
Hdr. Press=1010 psig
T_C (°F)
545°
535
532
1
2
3
4
5

5.6 The Operations Department will insure that Reactor Building Access Hatch seals are leak-tested per SP 1303-11.18.

6.0 <u>EMERGENCY ENTRANCE TO REACTOR BUILDING</u>

6.1 Emergency entrance may be made without RWP, air and radiation survey, or atmospheric testing on direction of the Shift Supervisor.

6.2 A minimum of two persons, at least one of whom shall have received Advanced Radiation Protection Training, shall make such entries.

6.3 All entering personnel shall wear complete protective clothing, self-contained breathing apparatus, appropriate neutron and gamma dosimetry and carry neutron and gamma dose rate meters.

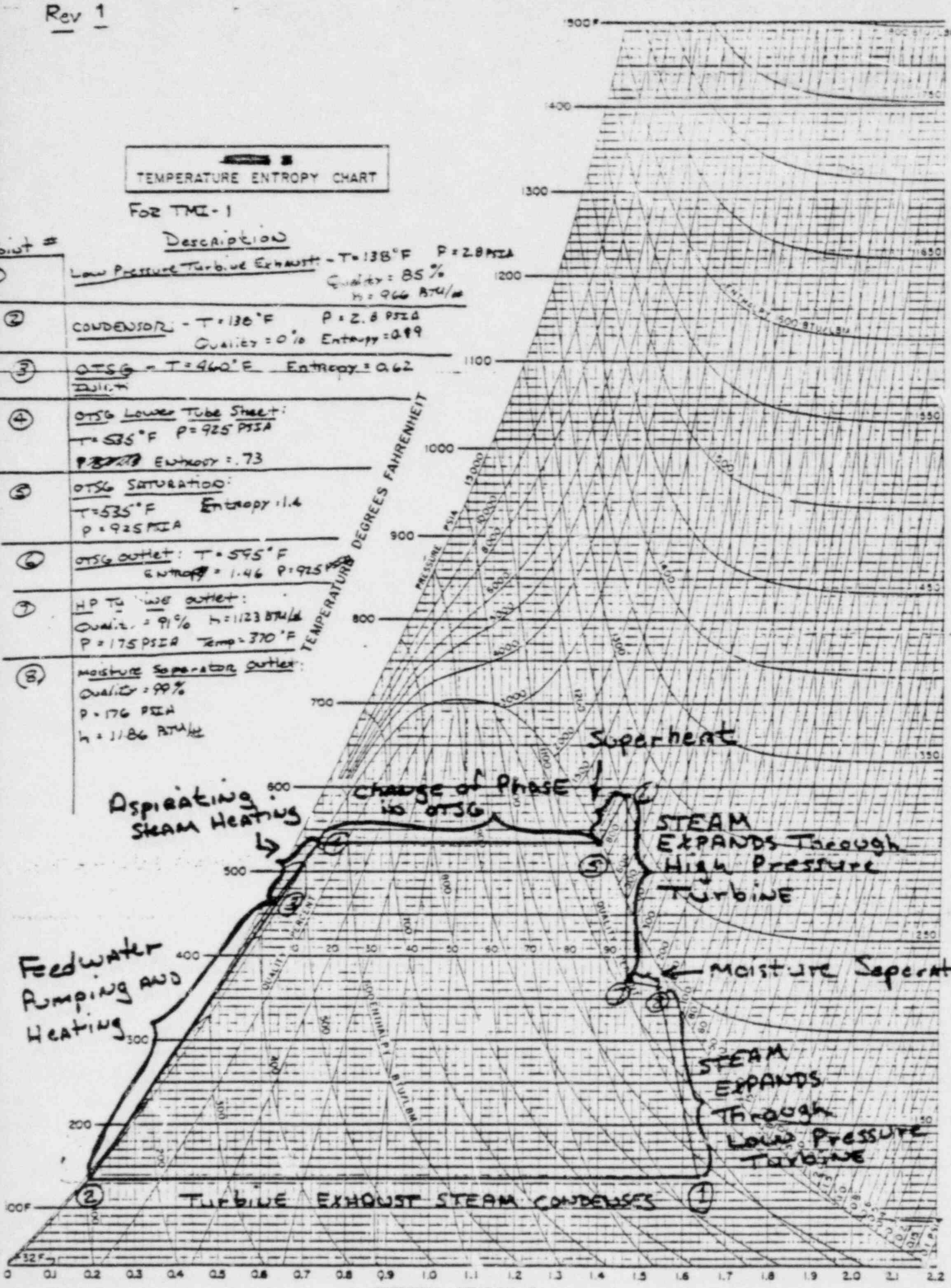
6.4 The Shift Supervisor shall submit a "Reactor Building Emergency Entrance Report" (Form 1630-2) to the Manager Unit 1 with a copy to the Manager, Radiological Controls after each such entry.

Rev 1

TEMPERATURE ENTROPY CHART

For TMI-1

Point #	Description
1	Low Pressure Turbine Exhaust: - T = 138°F P = 2.8 PSIA Quality = 85% h = 966 BTU/lbm
2	CONDENSOR: - T = 138°F P = 2.8 PSIA Quality = 0% Entropy = 0.19
3	OTSG - T = 460°F Entropy = 0.62 Inlet
4	OTSG Lower Tube Sheet: T = 535°F P = 925 PSIA Entropy = .73
5	OTSG SATURATION: T = 535°F Entropy = 1.4 P = 925 PSIA
6	OTSG outlet: T = 595°F Entropy = 1.46 P = 925 PSIA
7	HP Turbine outlet: Quality = 91% h = 1123 BTU/lbm P = 175 PSIA Temp = 370°F
8	moisture Seperator outlet: Quality = 99% P = 176 PSIA h = 1186 BTU/lbm



U.S. NUCLEAR REGULATORY COMMISSION

SENIOR REACTOR OPERATOR LICENSE EXAMINATION

Facility: TMI-1 –

Reactor Type: B&W

Date Administered: 4/24/81

Examiner: Wilson/Boger/McMillen

Applicant: MASTER- B

INSTRUCTIONS TO APPLICANT:

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

Category Value	% of Total	Applicant's Score	% of Cat. Value	Category
15	17.4	____	____	I. Reactor Theory
12	14.0	____	____	J. Radioactive Materials Handling, Disposal and Hazards
13	15.1	____	____	K. Specific Operating Characteristics
13	15.1	____	____	L. Fuel Handling and Core Parameters
20	23.3	____	____	M. Administrative Procedures, Conditions and Limitations
13	15.1	____	____	N. Theory of Fluids and Thermodynamics
86	100.0	____	____	Totals

Final Grade ________ %

I. Nuclear Theory (15)

I.1 The OARP Section on nuclear theory defines keff = $\frac{(n+1) \text{ gen}}{n \text{ gen}}$, or the ratio of neutrons in each succeeding generation. Assuming a stable count rate of 20 cps and a keff of 0.94 then control rods worth 3% $\Delta k/k$ are withdrawn from the core.

a. "By rule of thumb", what should happen to the count rate and keff several minutes following rod withdrawl? (1)

b. After this rod withdrawl in (a), keff is still less than one, therefore the number of neutrons in each succeeding generation should become less and less. What actually happens to count rate and why? (1)

I.2 As power level increases from 15% to 100%, reactivity is needed to overcome the negative effects of the Doppler defect. In other words, the reactivity due to Doppler defect increases as power increases. What happens to the Doppler Coefficient? Does it also increase, decrease or remain constant as power level rises? Explain. (2)

I.3 The OARP corrects the core beta value by a effectiveness factor which is approximately 98%. What is the reason for this effectiveness factor? (1)

I.4 Tech Spec 3.1.3.1 states that a Minimum Condition for Criticality is that the RCS Temperature must be above 525°F. What is the Tech Spec basis for this limit? (2)

I.5 Tech Spec 3.1.7 applies to the maximum positive moderator temperature coefficient of reactivity.

a. What are the two specifications that apply to the MTC with regard to power level? (1)

b. Although there is no specification regarding the maximum negative MTC what would be the consequences of having an MTC of $-1.0 \times 10^{-4} \Delta k/k$ and a main steam line break with a 200°F cooldown from 532°F. Would you expect a reactor restart to occur? Show all calculations and assumptions. (2)

I.6 OP 1103-15, Reactivity Balance Procedure gives guidance for you to periodically use the reactivity equation while the reactor is at power to check for any anomalies.

a. What three conditions constitute a reactivity anomally and/or discrepancy according to the procedure and Tech Specs? (1.5)

b. What factors are considered in the reactivity balance equation? (1.5)

c. As a prerequisite, reactor power must have been constant ±2% for the last 40 hours prior to performing the reactivity balance.. In the past 40 hours, power was decreased from 100% to perform turbine stop valve testing, then returned to 100%. [~~What factors in the reactivity balance equation would invalidate the calculation?~~] ~~Explain~~. Include assumptions. Discuss how each factor would affect the calculation (2)

J. RADIOACTIVE MATERIALS HANDLING, DISPOSAL AND HAZARDS (12)

J.1 Iodine 131 has a half life of 8 days and emits a 0.61 MEV Beta, a 0.36 MEV gamma 80% of the time and a 0.64 MEV gamma 10% of the time. For three curies of this material, (Show all calculations).

a. What would be the dose rate at a distance of 8 feet? (2.0)

b. How long could a person work in this radiation field before exceeding the weekly limit? (1.0)

J.2 State the radiological conditions concerning airborne activity and loose surface contamination which requires a properly completed and approved radiation work permit. (2.0)

J.3 List 4 "pre-operational" checks that should be completed before utilizing a portable radiation protection instrument. (2.0)

J.4 What are the 10 CFR Part 20 requirements for immediate notification of an overexposure of personnel to radiation. (1.5)

J.5 a. No eating or smoking is allowed in radiation areas. If the radiation level is low enough to allow entry into the area, why are the above prohibitions imposed? (.5)

b. Discuss the hazards of violating the above prohibitions. (1.0)

J.6 a. Under what conditions must the primary coolant system be sampled and analyzed for dose equivalent I-131? (.5)

b. What is the basis of this action? (1.5)

K. SPECIFIC OPERATING CHARACTERISTICS (13)

K.1 During the midnight shift, while at rated power near the end of core life, you are instructed to reduce load to 70% for 6 hours and then return to rated power to meet the morning demand.

a. Describe three factors to be considered if the evolution is performed solely with control rods. (1.5)

b. Describe two disadvantages of performing this evolution solely with soluable poison control. (1.5)

K.2 Operating limits are imposed on RCS heatup and cooldown rates and RCS temperature/pressure relationships.

a. Explain why heatup and cooldown rates must be limited. (1)

b. What is the purpose of the temperature/pressure limit? (1)

c. Explain why the temperature/pressure limit will shift with core age. (1)

K.3 Explain how changing the OTSG low level limit from 30 inches to 35 inches would affect facility operations. (2)

K.4 You are the responsible SRO on shift during a unit cooldown due to a primary to secondary leak that exceeded Technical Specification limits. Secondary activity analysis had previously identified that both OTSG's were leaking. The initial cooldown was via natural circulation due to a problem that precluded use of the RCP's. Both OTSG's have now been isolated since RCS temperature is 280°F and the DHR system is in operation. Assume the valve disc on DH-V2 separates from the valve stem and drops into the valve body.

a. List five control room indications that would result due to this situation. (1)

b. Outline two alternate methods of RCS heat removal after this incident. (1)

c. Compare the two methods in terms of:

i. heat removal capabilities
ii. offsite radiological consequences
iii. ability to control cooldown rate (1)

K.5 Reactor Coolant Pumps are required to be tripped if RCS pressure reaches the ESAS actuation setpoint of 1600 psig. List four circumstances which would allow RCP restart after manual tripping due to low RCS pressure conditions. (2)

L. FUEL HANDLING AND CORE PARAMETERS (13)

L.1 Figures L.1 and L.2 are taken from the Tech Spec Safety Limits bases.

a. Identify the portions of the curves of Figure L.1 that are based on DNBR limit and the portions that are based on Kw/ft. limits. (1.5)

b. Identify the curves of Figure L.2 that are based on DNBR limits and those that are based on Quality Limit. (1.5)

L.2 a. List the nuclear and radiation monitoring systems that must be operable during refueling. (1.5)

b. Does the required operable neutron flux monitor(s) have to be located in the control room? (.5)

L.3 While operating at full power with the Diamond Panel in manual, one safety rod drops into the core.

a. Assuming no operator action, describe and explain the response of reactor power. (1.5)

b. Explain how the radial position of this dropped rod will affect quadrant power tilt as seen by the excore nuclear instrumentation. (1.5)

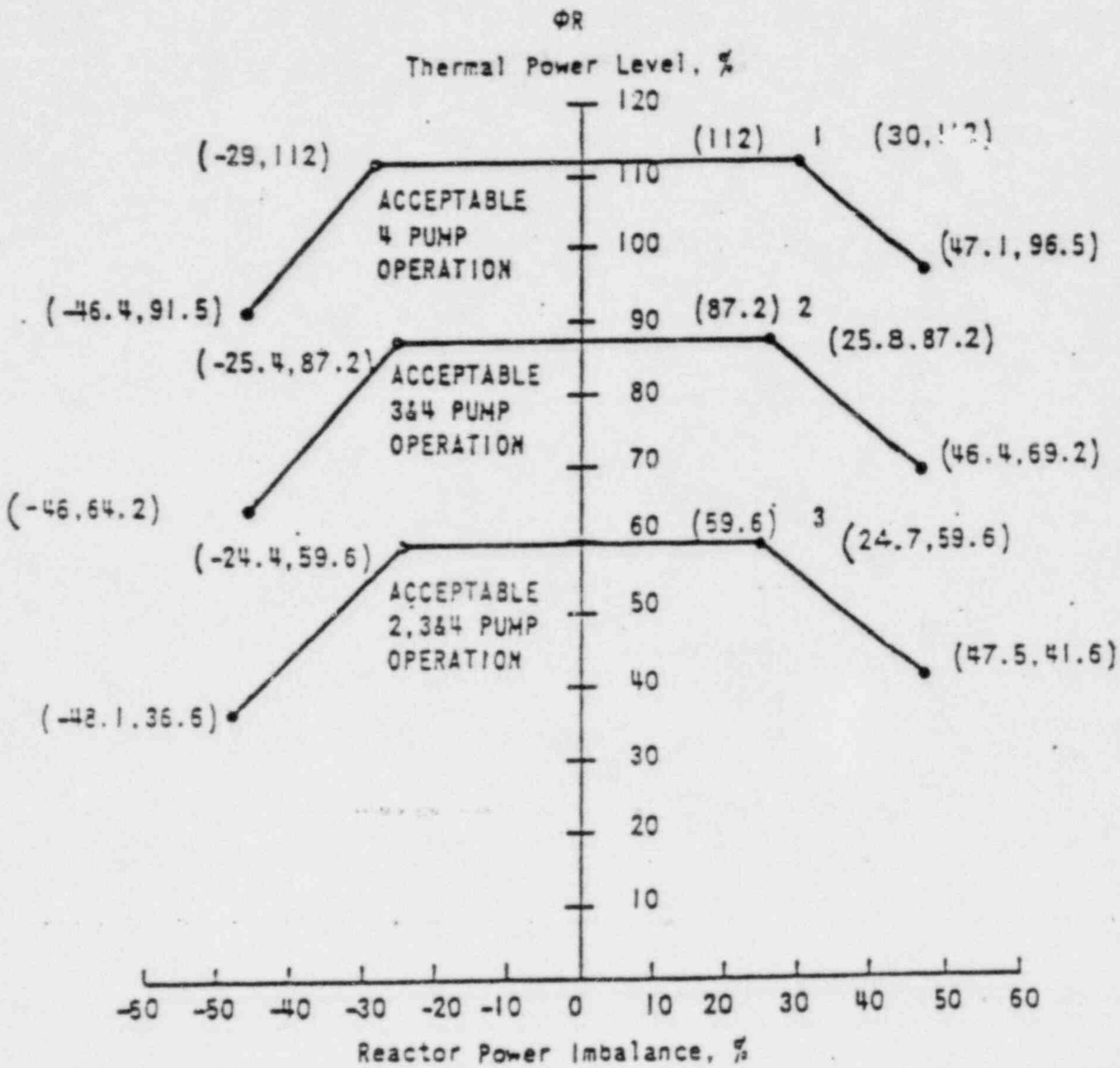
L.4 a. Discuss the hazards associated with APSR handling if the "Orifice Rod By-Pass Control" selector switch is positioned "Orifice Rod". (1)

b. Under what conditions may the control rod grapple electrical interlocks be by-passed? (.5)

c. What is the difference between a "control rod test" light indication and "control rod disengaged" indication. (.5)

L.5 a. Why is it necessary to have Mechnical Maneuvering Recommendations for power escalations? (2)

b. What limits will exist during the initial power escalation at cycle startup? (1)



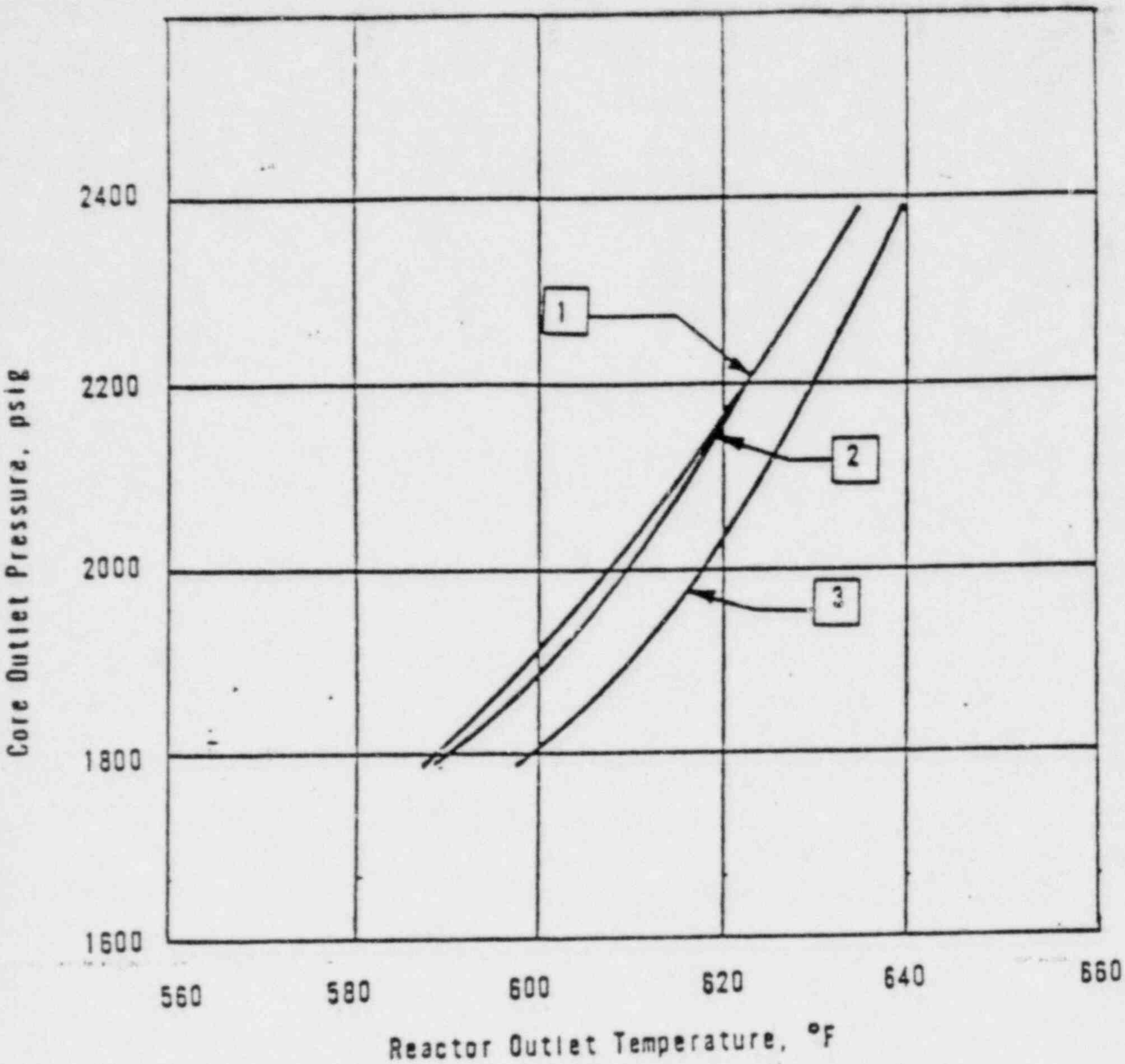
Curve	Reactor Coolant Flow (lb/hr)
1	139.8×10^6
2	104.5×10^6
3	68.8×10^6

TMI-1

CORE PROTECTION SAFETY LIMITS

[redacted]

Figure 2.1



CURVE	REACTOR COOLANT FLOW (LBS/HR)	POWER	PUMPS OPERATING (TYPE OF LIMIT)
1	139.8×10^6 (100%)*	112%	
2	104.5×10^6 (74.7%)	87.2%	
3	68.8×10^6 (49.2%)	59.6%	

*106.5% of Cycle 1 Design Flow

TMI-1

CORE PROTECTION SAFETY BASES

[redacted]

Figure L.2

M. ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS (20)

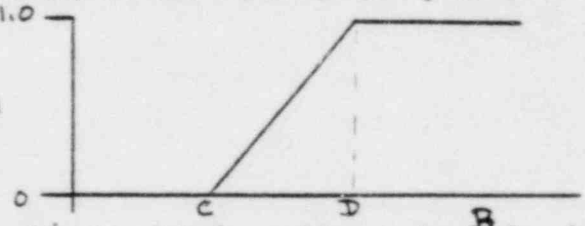
M.1 Through various observations it is determined that the pressurizer electromatic operated relief valve is leaking while the unit is at full power but the leakage is within acceptable limits.

a. What constitutes acceptable leakage limits for this situation? (1.0)

b. Is it permissible by Tech Specs to close the block valve to isolate the leak? Explain. (1.0)

M.2 According to OP 1104-14 (or the OARP) the purge of the Reactor Building may be limited by either of two considerations. What are they? (1.0)

M.3 When a heat balance calibration is performed to adjust the power range channel NI's to agree with the core thermal power, the Tech Specs (1.5.6) contain the following curve:



a. Provide the labels and/or numbers for A,B,C and D on the curve. (1.0)

b. Complete the following equation:
Core thermal Power = $(\_\_)\, Q_{sec} + (1 - \_\_)\, Q_{PRIM}$ (0.5)

c. When power is $>$15% and less than 100% what is the ______ (blank) in part (b) equal to? (0.5)

M.4 The following LER's from TMI-1 were reported to the NRC during the two year period preceding the Unit 2 accident. As Shift Supervisor, for each of these incidents, describe:

i) Your immediate actions or those actions you would direct your operators to take;

ii) The Technical Specifications that are applicable or have been violated (Do not include reporting requirements).

a. In May, 1978 during initial cycle 4 startup, it was discovered that the control rods had been in the "not allowed" region of the Tech Spec curve requirements for nine hours. The correct control rod index curve had not been incorporated into the control room copy of the operating procedures. (2.0)

b. In June, 1978 during full power operation, seven of nine control rods in Group 3 inadvertently drop into the core due to a shorted diode in DC power supply. (2.0)

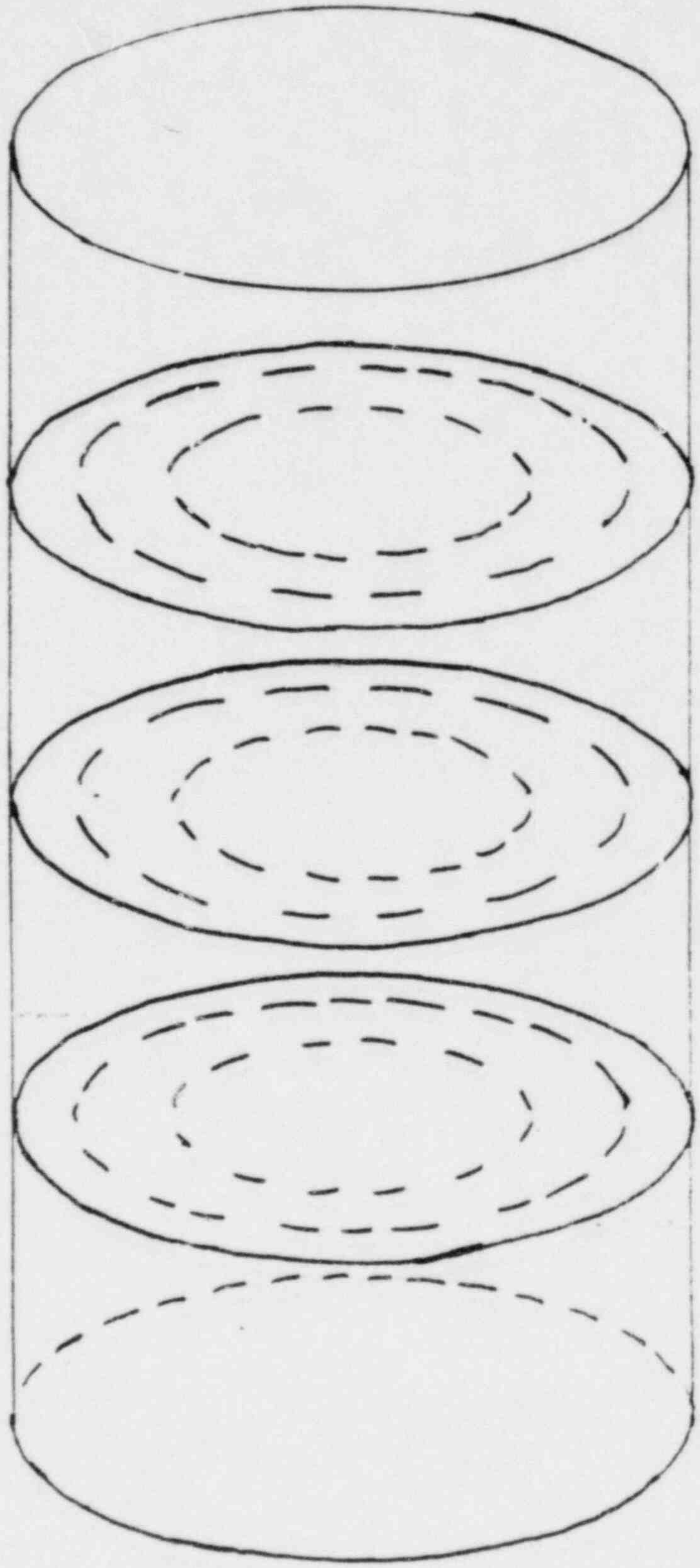


FIG M-7

M. continued

M.4 c. From 0440 hours on August 18, 1978 to 1400 hours on August 19, 1978, it was discovered that neither of the two inservice Nuclear River Water System pumps was selected to start on an E.S. Signal. (2.0)

d. In September 1978 during surveillance testing, a temperature sensor for the 'A' Diesel Generator Room was found to be inoperable. A work order was written but no other action was immediately taken. (2.0)

M.5 Administrative Procedure 1001 contains a section that addresses Temporary, On-the-Spot Changes (TCN's, 3.6.4). List ... cases or instances as given in the procedure that TCN's can be used for. (2.5)

M.6 AP 1002 Discusses the use of tags for isolation of equipment. As the responsible supervisor, what five stipulations must be met for the operations department to grant permission to operate or energize equipment that bears a Blue Tag. (2.5)

M.7 Technical Specification 3.5.4.1 states that minimum incore detectors for axial imbalance shall be arranged in the following manner.

a. Three detectors in each of three strings shall lie in the same axial plane with one plane in each axial core half.

b. The axial planes in each core half shall be symmetrical about the core mid-plane.

c. The detectors shall not have radial symmetry.

On the attached drawing, Figure M-7, sketch the minimum incore nuclear instrumentation that will satisfy this specification. (2.0)

N. THEORY OF FLUIDS AND THERMODYNAMICS (13)

N.1 Flow elements recently installed on the high pressure injection lines are, according to Lesson Plan RM-14, "cavitating venturis." From a fluid dynamic consideration what is the difference between a standard venturi and a cavitating venturi? Explain (1)

N.2 In the natural from full power circulation cooldown event at St. Lucie last June, a reactor trip was initiated followed several minutes later by a trip of all four RCP's. This was required due to a loss of Component Cooling Water to the RCP's. Although the TMI-1 design differs considerab[illegible] [illegible] is conceivable that tripping all four RCP's following a [illegible] .l reactor trip may be required under abnormal conditions.

a. Show on Figure N.2 the expected Rx trace of T_H and T_C for the above event. Begin at time of manual trip and assume the plant was maintained at Hot Shutdown for 2 hours. (2)

b. At 2 hours a decision is made to cooldown the plant at 60^{O}F per hour. Show and explain the trace of T_H and T_C for the next two hours. (1)

N.3 Figure N.3 is a Temperature entropy diagram showing an idealized Carnot Cycle a-b-c-d.

a. Identify each of the processes, that is, what is happening to the working fluid from a-b, from b-c, etc. This identification should include an example for each (e.g. isothermal expansion-firing of cylinder in an engine). (2)

b. Show on the diagram what happens if the temperature of the heat sink is increased (e.g. partial loss of vacuum in condenser. (1)

N.4 The analysis to support modifications to the HPI and MU&P System, in particular, the addition of V-217, state that the normal maximum flow through V-17 is about 75 gpm.

a. In actual practice what are the maximum flow rates at 1800 psig for <u>one</u> and <u>two</u> makeup pump operation? (1)

b. What additional flow capacity does the addition of MUV-217 give the operator? ~~(1)~~ (.5)

c. What was the basis of adding MUV-217? (1)

N.5 a. Using the partial pressure laws and assuming ideal gases, what does the addition of non-compressible gases such as hydrogen and nitrogen due to the partial pressure of steam in the pressurizer? (1)

b. Assuming the partial pressures of three gases (steam, hydrogen, and nitrogen) are reproportioned due to (a) above, how would total pressure respond to a sudden load reduction? (1)

N.6 In the long term following a LOCA when the BWST is emptied, it is necessary to shift pump suction to the RB sump. Assume initially when suction is shifted to the sump only one LPSI pump is running.

a. Figure N-6a represents a general head curve for a LPSI pump. Show and explain on the Figure how the Total Head and Flow Rate will change if ~~the second LPSI pump is started~~. The RCS Repressurizes (.75)

b. The RCS repressurizes so it is necessary to "piggyback" the pumps; that is discharge of one LPSI feeding suction of one HPSI pump. Show and explain on Figure N-6b the response of Head and Flow rate of the additional pump (.75)

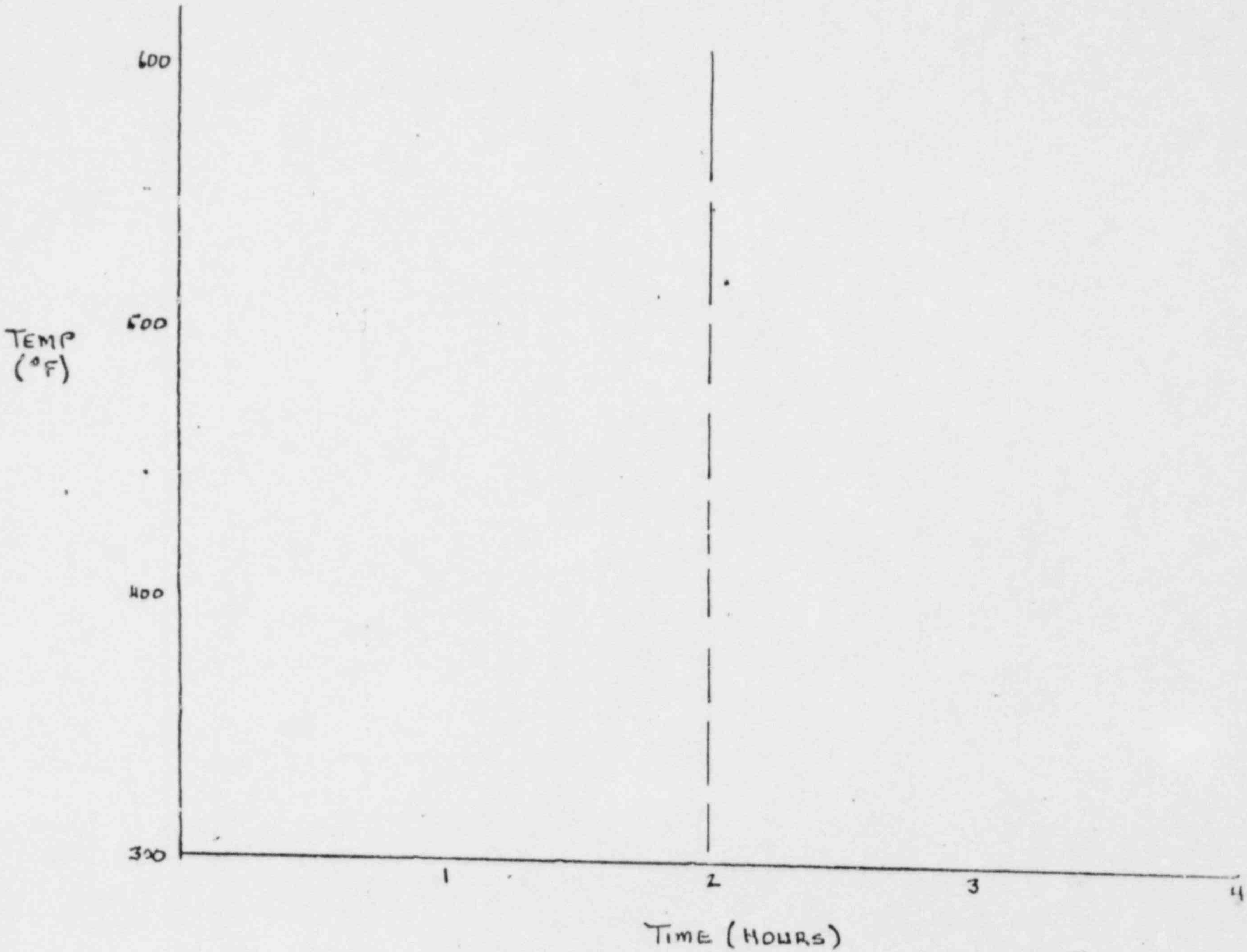


FIG N-2

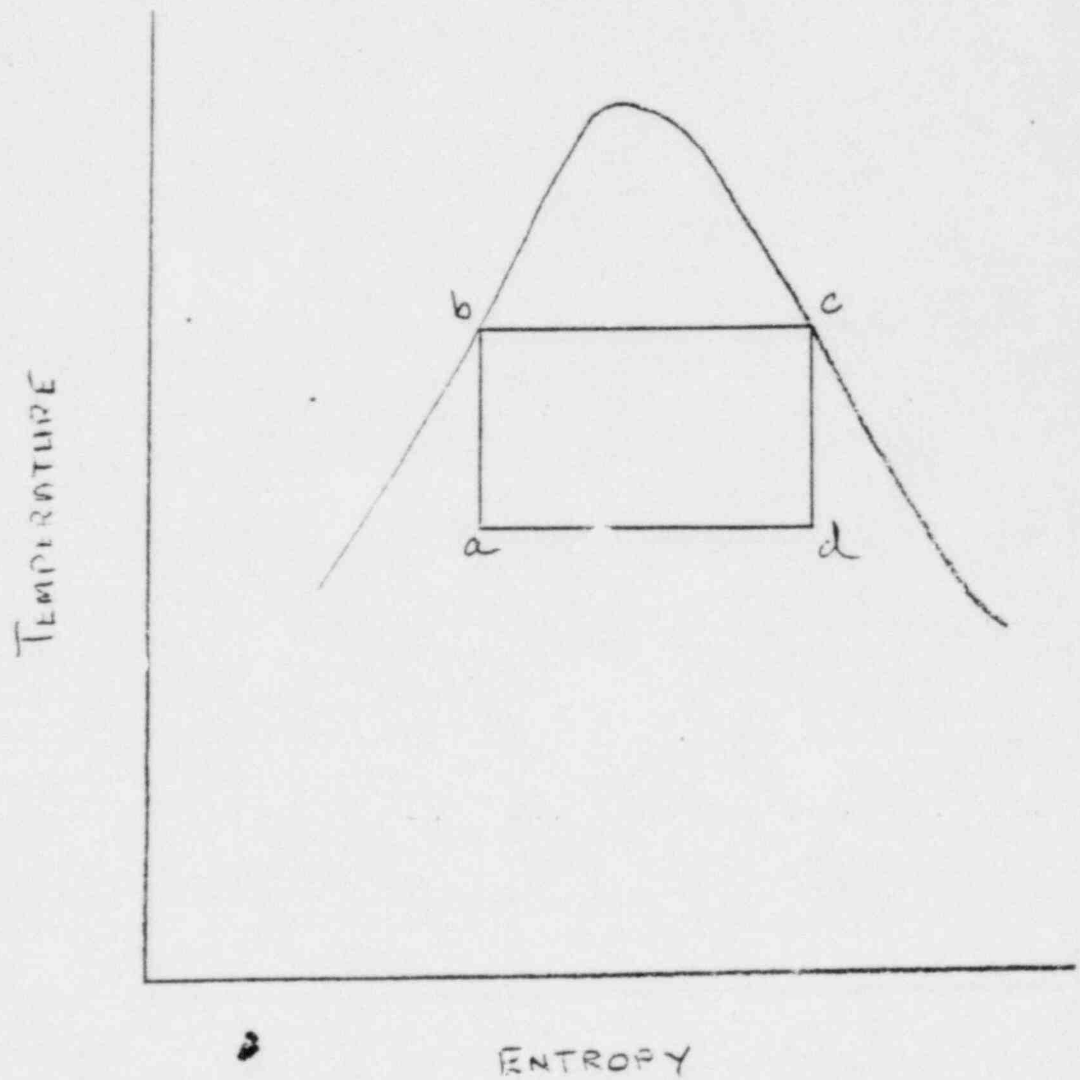
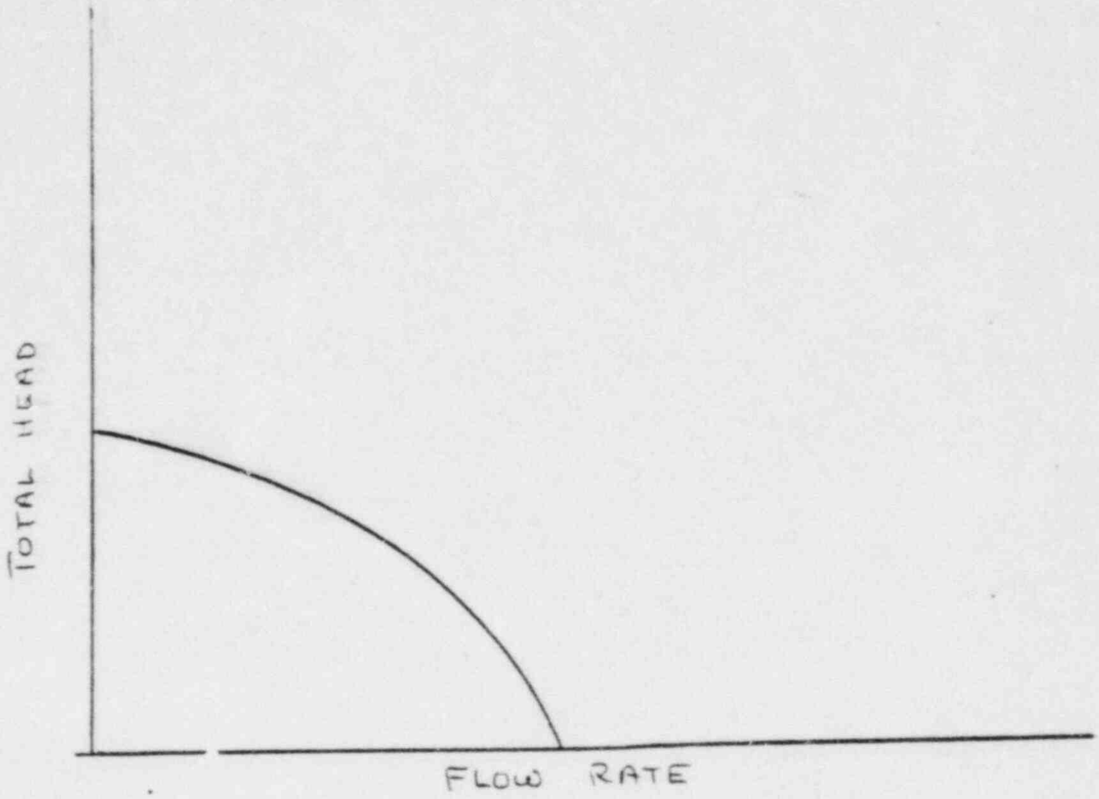


FIG IV-3

a.



b.

FIG N-6

I.1 a. Rx is 6% shutdown, 3% Δk added, 1/2 way to critical, count rate doubles and k = .97. Pg. 18 - Count rate doubles, keff 1/2 way to unity.

b. Pg 14 $N = S + SK + Sk^2 + \ldots = \frac{S}{I-K}$
As long as neutron source is present, the number of neutrons will be at a constant level while subcritical.

I.2 OARP Pg. 30
The coefficient decreases (becomes less negative) with increasing fuel temperature due to resonance peaks becoming overlapped. The total x-sectional area for neutron absorption continues to increase but at a decreasing rate. (Self-Shielding)

I.3 OARP pg. 38
delayed neutron born @ lower energy than prompt and is only 98% as effective as a prompt neutron in power production in the TMI core.

I.4 At BOL, MTC is expected to be slightly positive at operating temperatures with the operating configuration of control rods. Calculations show that >525°F, The +MTC is acceptable. At lower temps (<525) MTC will be less negative or more + than at operating ∴ therefore startup below 525 is prohibited (basis of non-positive MTC at power is Fw line break).

I.5 a. T.S. 3.1.7.1 (1) MTC shall not be + >95% power (2) MTC $\leq + 0.5 \times 10^{-4}$ $\Delta k/k/^{\circ}F$ <95% power.

b. $1.0 \times 10^{-4}\ \Delta k/k/^{\circ}F\ (200^{\circ}F) = 1 \times 10^{-2}$ = .01 or 1% reactivity. This is equal to the max Spec on SDM and therefore ρ(net) = 0. There is a possibility of restart.

I.6 a. OP 1103-15 pg 2
$\pm$ 0.5% Steady State Xe
$\mp$ 0.8% Transient Xe
≥ 1% Tech Spec 4.10

b. $\rho(\text{net}) = \rho(\text{fuel}) + \rho(\text{rods}) + \rho(\text{boron}) + \rho(\text{Temp}) + \rho(\text{power}) + \rho(\text{Xe}) + \rho(\text{Sm})$

c. (fuel) - in 40 hrs - decrease slightly but virtually negligible
(rods) - Assumption - power change made with boron - no change

(boron) - Xenon will be changing - in order to accommodate (xe), boron qill have to change
(temp) - No change if back to 100%

(power) - no change if 100%
(Xe) - If accounted for in boron change - no difference
(Sm) - very slight change but negligible

49 mr/hr

J.1 a. $6C\,E/d^2 = 6(.36 \times 8 + .64 \times .1)/\cancel{16} = 0.132 \times 3 =$ ~~0.396 R/hr~~

b. .75 hours (45 minutes) $\frac{3.5}{1.25}$ = 3 hr

J.2 Advanced HP training 412-1(a)

a. Airborne - excess of 3^{-10} or 3×10^{-11} u Ci/cc gross β with no α present (no Sr proven)

b. Loose surface - 1000 dpm/100 cm^2 β-γ or 100 dpm/100 cm^2 - α (< 1 mr/hr fixed

J.3 1. Calibration, 2. Battery check, 3. zero check, 4. source check, 5. physical condition.

J.4 -25 rems to whole body, 150 rems to skin 375 rems to extremeties.

J.5 a. Hazard associated with introducing removable contamination into the body through smoking & eating.

b. Ingesting radioactive materials dangerous.
   1. leads to concentrate in certain areas or organs.
   2. no shielding to minimize ionization damage.
   3. no way to put distance between you and material.

J.6 a. Thermal power change of more than 15% of rated thermal power within a 1 hour period (1102-4).

b. Power change could cause leaking fuel, need to sample RCS to verify below TS levels to assure 10 CFR 100 limits on S/G tube rupture are not exceeded.

not T.S. - procedural level

K.1 a. 1. Rod insertion limits
2. Imbalance
3. Power level cutoff (Xe Transient limits)
Water processing also a consideration @ EOL

b. 1) Waste generation
2. rate of increase limits

K.2 a. limit stresses on reactor vessel
b. NDTT
c. neutron embrittlement of reactor vessel materials

K.3 Raising the low level limit will improve the heat removal capabilities of the OTSG's. This will result in reaching the desired T_{avg} at a higher power level.

K.4 a. DHR flow ↓; Recorder
DHR low flow alarm
R emp. ↑
DHR pump amps abnormal
DHR pump discharge pressure ↓
DHR pump high vibration
(indication of valve closure is incorrect- limit switches for lights are on the stem which did not move)

b. Unisolate OTSG's and return to natural circulation cooldown, ~~and~~ or 2) use HPI through PORV

c
c. i. 1 has better heat removal
ii. 2 has less offsite consequences
iii. 1 has better control

K.5 OARP Quiz Q.12 Module 6

OP 1202-6B

Feedwater available to OTSG <u>and</u>
1. 50°F subcooling in RCS <u>or</u>
2. RCS pressure stable or increasing and > 1600psig <u>or</u>
3. RCS pressure > OTSG secondary pressure by 600 psi (pump bump) <u>or</u>
4. RCS pressure stable for <u>> 1 hour</u> and > 250 psi and OTSG secondary pressure > 100 psi

?

Also ICC procedure
Tube rupture

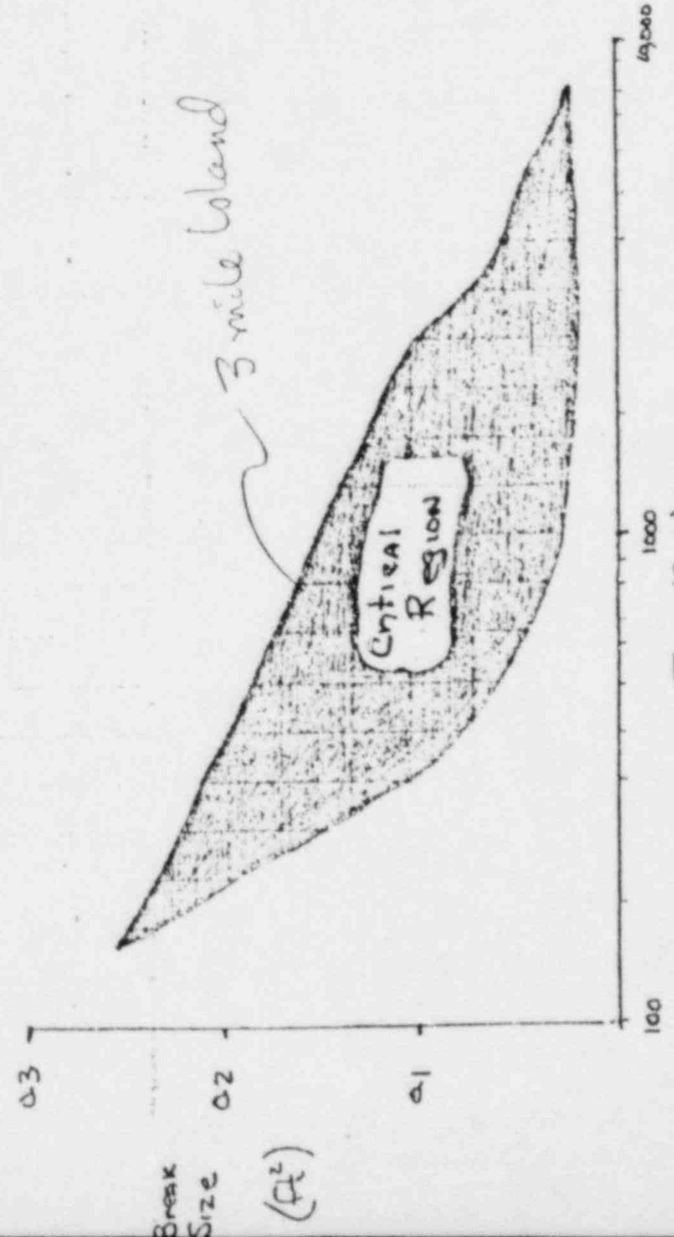
RCP TRIP FOLLOWING AUTO ES ACTUATION

1) must be accomplished within 1st Two Minutes.

RCP RESTART FOLLOWING AUTO ES ACTUATION

1) EF AVAILABLE, RCS >1600#, or >250# and 50° subcld
   Start one on operable OTSG, C/D.
2) EF AVAILABLE with RCS >600# Above OTSG Press
   10 sec Bump on Good OTSG, then every 15 mins
   do 10 sec Bump of others, one @ time
   let 5th one on.
3) EF AVAILABLE, OTSG Press <100# with RCS >250#
   Bump 10 Secs on Good OTSG
   30 Mins later start the alternate P on Good OTSG
1) Inadequate Core Cooling Guidelines
   > 1400° start 1/Loop
   > 1800° start all 4 if possible.
5) For LOCA Overcooling Event
   when 50° Subcld start 1/Loop

Small Break LOCA Size VS Time
in Critical Region for RCP Trip.



L.1 a.

b. Curve 1 - DNBR
Curve 2 - DNBR
Curve 3 - Quality

L.2 a. TS 3.18, RMG 6, 7, 9 RMA - 9 Ctmt Purge
2 neutron monitors operable while changing geometry, one while not

b. P. 16 OARP - No Also Tech Spec 3.8.2

ICS Responses differs from attached

L.3 a. Reactor power will initially dip due to the $-\rho$ of the rod, tave $\downarrow$ & fuel temp $\downarrow$, $+\rho$ due to -MTC and FTC, $-\rho$ from rod will compensated by the $+\rho$ from MTC and FTC, reactor power will return to the initial value.

b. The excore detectors see leakage flux - as the rod radial position approaches the core edge, the detectors will be more affected and the QPT will be more severe.

L.4 a. "Orifice Rod" position will instruct the telescopic cylinder to
pg. 17 retract for enough into the control rod tube to protect the orifice rods and permit bridge motion. Since the APSR's are longer than the orifice rods, an APSR could still be partially inserted in a fuel assembly when bridge motion was allowed.

pg. 20 b. Never - there are no by-passes.
pg. 18 c. In the "test" position the control component (orifice rod, BPRA) is still engaged by the grapple.

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pg. 20 b. Never - there are no by-passes.
pg. 18 c. In the "test" position the control component (orifice rod, BPRA) is still engaged by the grapple.

L.5 a. Primarily 2 reasons - both related to clad integrity
(1) Since UO_2 thermal expansion is greater and response time is shorter than for Zircoloy, need slow increase to minimize clad stresses due to creep. } clad touching also
(2) To allow pin hole leaks in the cladding to close. 2 water logged fuel

b. < 20% power rate is 10%/hr	Step # -1	OP 1102-4
20% > power to 40% rate is 30%/hr	-3	
> < 40% power rate is 3%/hr	-4	
power @ PLCO 5 hour hold	-4	

> 75% to 5%/hr
also hold 20% below PLC cutoff

15.1.3 CONTROL ROD MISOPERATION (STUCK-OUT, STUCK-IN, OR DROPPED CONTROL ROD)

15.1.3.1 Identification of Causes

In the event that a control rod cannot be moved, localized power peaking and shutdown margin must be considered. If a control rod is dropped into the core while operating, the resulting transient must be examined.

Adequate hot shutdown margin is assured by requiring a subcriticality of 1% $\Delta k/k$ with the control rod of greatest worth fully withdrawn from the core. The nuclear analysis reported in Section 4.3. demonstrates that this criterion can be satisfied. This criterion has been analyzed in terms of the minimum tripped rod worth available in the loss-of-coolant flow, rod ejection, startup, rod withdrawal, and steam-line-failure accidents. In all cases the available rod worth is sufficient to provide margins below any damage threshold.

A dropped control rod is defined as a deviation of a control rod from its group reference position by more than an indicated 9 inches. This definition then covers both the action of dropping a rod and sticking a rod while moving a group. The ICS is available to inhibit all rod-out motion and runback the steam generator load demand to 60% of rated load. The details of these actions are described in Section 7.7. Even though ICS action is available to prevent or mitigate this accident, the accident analysis was done without ICS action.

15.1.3.2 Analysis of Effects and Consequences

15.1.3.2.1 Safety Evaluation Criteria

The safety evaluation criteria for this accident are:

a. The core thermal power shall not exceed 112% of rated power.

b. The reactor coolant system pressure shall not exceed code pressure limits.

15.1.3.2.2 Methods of Analysis

A detailed B&W digital model[14] has been used to analyze the transient response to a dropped control rod. This program includes fuel pin, point kinetics, pressurizer, and loop models, including the steam generators.

The reactor is assumed to be operating at rated power when the control rod is dropped. To achieve the most adverse response, the most negative values of the moderator and Doppler coefficients were used along with the maximum calculated rod worth for rated power operation. The parameters used in this analysis are shown in Table 15.1.3-1.

15.1.3.2.3 Results of Analysis

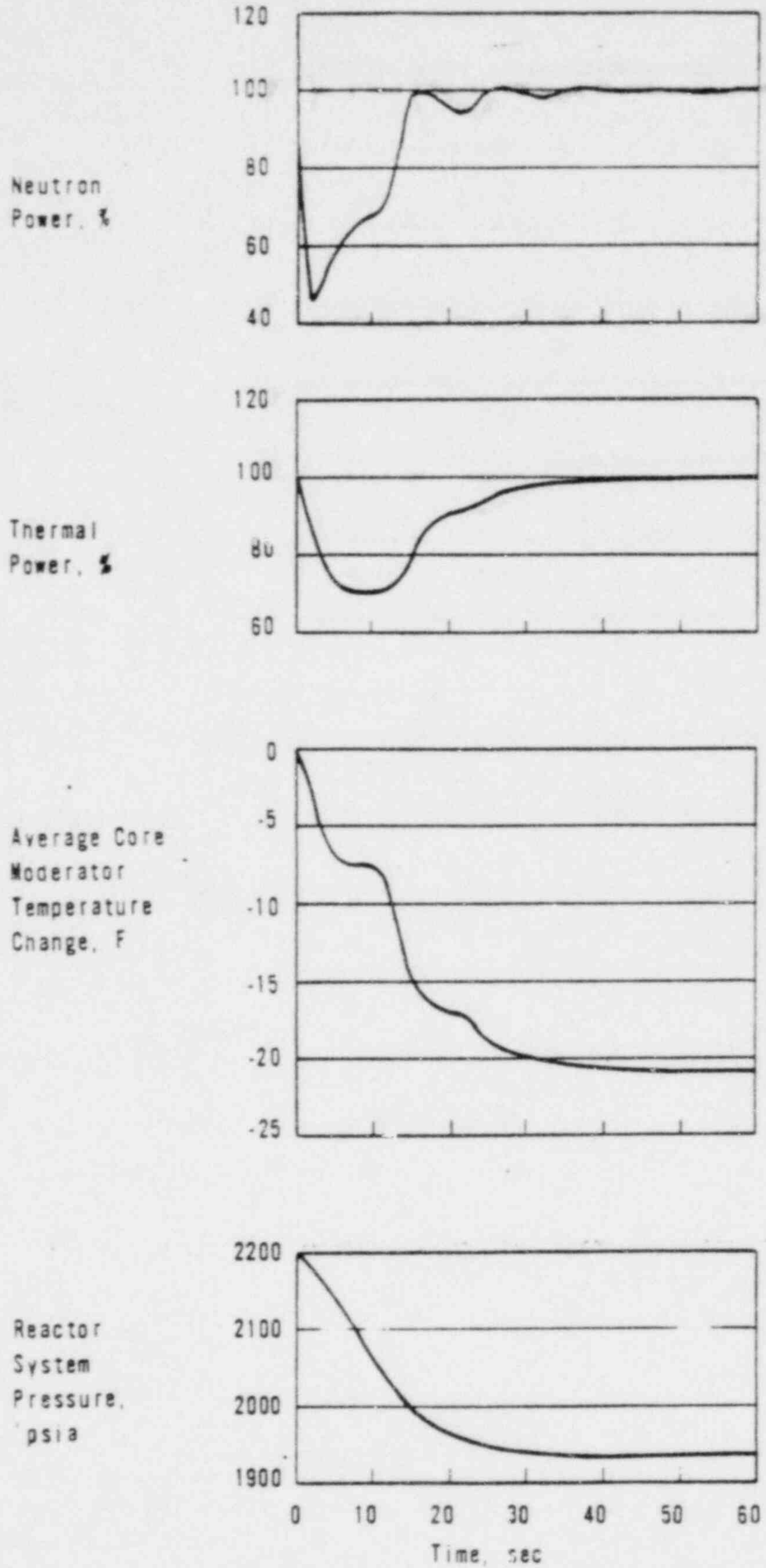
The results are presented in Figure 15.1.3-1. The neutron power decreases causing a rapid decrease in both the core moderator temperature and the fuel temperature. These temperature decreases overcompensate for the worth of the control rod, and the neutron power rises slightly above the initial neutron power level. The neutron power then decreases to below the initial power level and eventually levels out at the initial power level. The thermal power response is similar to the neutron power; however, the thermal power level never exceeds the initial rated power value. Both the core moderator temperature and pressurizer pressure decrease during the transient and level out at a value lower than the initial value. Since the thermal power never exceeds the initial value and the pressure decreases during the transient, the safety evaluation criteria are met.

Cases have been run for rod drops at beginning-of-life conditions and lower rod worths. These transients are not included in this discussion because they represent less severe conditions than the end-of-life conditions and the maximum calculated rod worth.

TABLE 15.1.3-1

DROPPED ROD ACCIDENT PARAMETERS

Moderator Coefficient, $(\Delta k/k)/F$	-3.00×10^{-4}
Doppler Coefficient, $(\Delta k/k)/F$	-1.37×10^{-5}
Control Rod Worth at Rated Power, % $\Delta k/k$	0.65



0.65% $\Delta k/k$ ROD DROP FROM RATED POWER AT EOL CONDITIONS

THREE MILE ISLAND NUCLEAR STATION UNIT 2

FIGURE 15.1.3-1

3.1.12 Electromatic Relief Valve

Applicability

Applies to the settings, and conditions for isolation of the electromatic relief valve.

Objective

To prevent the possibility of inadvertently overpressurizing the primary loop.

Specification

3.1.12.1 The electromatic relief valve shall not be taken out of service, nor shall it be isolated from the system (except that the electromatic relief valve may be isolated to limit leakage to within the limits of specification 3.1.6.), unless one of the following is in effect:

a. High Pressure Injection Pump breakers are racked out or MU-V16A/B/C/D and MU-V217 are closed.
b. Head of the Reactor Vessel is removed.
c. T avg. is above 320°F.

3.1.12.2 The electromatic relief valve settings shall be as follows, within the tolerances of ± 25 psi and ± 12°F:

Above 275°F - 2450 psig
Below 275°F - 485 psig

3.1.12.3 If the reactor vessel head is installed and T avg. is ≤275°F, High Pressure Injection Pump breakers shall not be racked in unless:

a. MU-V16 A/B/C/D and MU-V217 are closed, and
b. Pressurizer level is ≤ 220 inches.

Bases

If the electromatic relief valve is removed from service, sufficient measures are incorporated to prevent overpressurization by either eliminating the high pressure sources or flowpaths or assuring that the RCS is open to atmosphere. In order to prevent exceeding leakage rates specified in T.S. 3.1.6. the electromatic relief valve may be isolated.

The electromatic relief valve setpoints are specified with tolerances assumed in the bases for Technical Specification 3.1.2.

With RCS temperatures less than 275°F and the makeup pumps running, the high pressure injection valves are closed and the pressurizer level is maintained less than 220 inches to prevent overpressurization in the event of any single failure.

M.1 T.S. 3.1.6
10 GPM Identified

b. T.S. 3...12.1 Does not permit isolating the PORV unless it is necessary to meet the leakage spec. OR > 320 °F RCS °F

M.2 OARP - RB Ventilation System pg. 23
1. Limited by waste gas release permit authorization
2. Air temperature due to relatively high NDTT of containment isolation valves($60^{\circ}F$)

M.3 a. T.S. 1.5.6
A - weighting factor, f
B - Power
C - 15%
D - 100%

b. $CTP = \alpha Q_{sec} + (1-\alpha) Q_{PRIM}$

c. $\alpha = \left(\frac{POWER - 15}{55}\right)$

M.4 a. i. LER 78-015
Order operator to immediately borate rods out of "not Allowed" region
- Contact Duty Officer/ Ops Super - decision must be made whether plant shutdown will be necessary
-Initiate procedural change - obtain proper copy of index curve and incorporate into procedures

ii. T.S.3.5.2.5 - Corrective measures taken immediately to achieve an acceptable rod position.

b.i. LER 78-020
EP 1202-08 CRD Equipment Failures
ICS runs back to 55% - have operators carry out the three immediate action steps; Check OP 1102-4 for power reductions
- Subsequent action - step 5 - says if more than one rod is inoperable then shutdown reactor and declare an alert.

ii. Inop rod T.S. 4.7.1.2, also 3.5.2.2.a, Rod Index Limits, 3.5.2.5.e, QPT - 3.5.2.4.a

c.i. LER 78-025
"On discovery, NRWP correctly selected for ES actuation". Others?

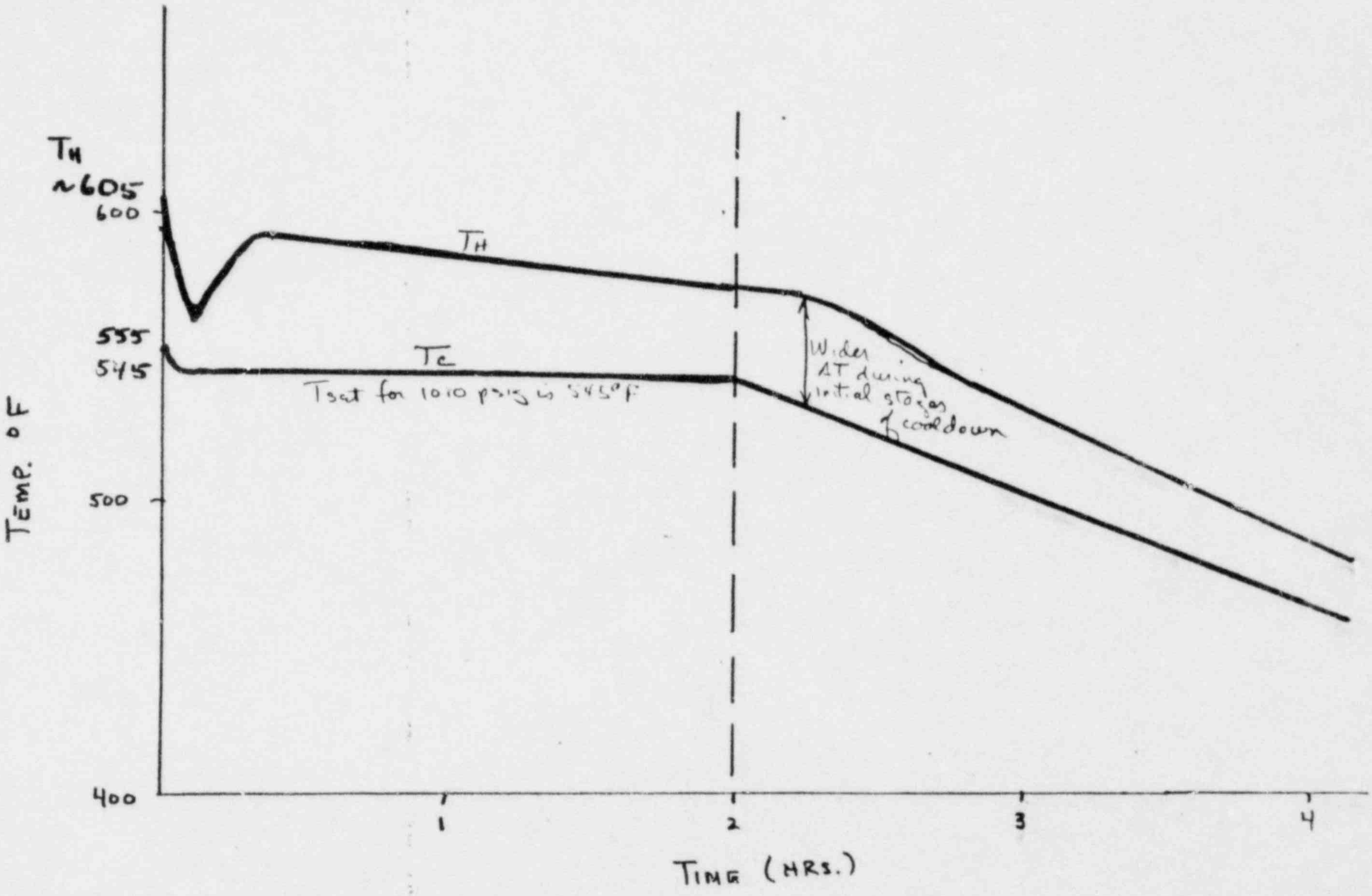
ii. T.S.3.3 ?? (We are missing Spec Page)
3.3.1.4 # of NSR water pumps operable (need 2)

M.4 d. i. LER 78-029
Check Tech Specs - if ? ··rument was required by T.S. Table 3.1?-1 (which it certainly must be since it's in a D-G room) immediately (since 1 hour is already up, establish a fire watch to inspect the zone

ii. T.S. 3.18 Specif' .e required operability of fire detectors
NO FIRE WATCH

M.5 AP1001
1. when time or plant conditions do not permit the use of PCR's
2. when the change is of a temporary nature and should not result in a a permanent revision
3. to direct operations during testing, refueling, maintainance and modifications
4. to provide, in unusual situations, not within scope of normal procedure
5. to ensure orderly and uniform operations for short periods when the unit, system or component is not performing in a manner not covered by existing detailed procedures.

M.6 1. Permission to operate or energize ES equipment must come from the Shift Supervisor/Foreman
2. Permission to operate or energize non-ES equipment may be granted by any duty operation's control room personnel
3. All permissions must include specific instructions as to the position the equipment must be left in when requested operations have been complete. The person in charge of the work shall work with due regard for the actual condition existing.
4. Permission to energize or operate a blue-tagged component shall be granted only in situations where energizing or operating the equipment is required to complete the work item for which the tags were placed. In all other situations requiring energizing or operating tagged equipment, cleared tags must be removed in accordance with applicable sections of the procedures.
5. When any ES related equipment or tech spec required equipment are returned to service after being tagged for maintainance the swithhing order must include the position the valves must be placed in and the normal operating condition for the device. — DOUBLE VALVE VERIFICATION



TH
~605
600
555
545
500
400
TEMP. °F
TH
TC
Tsat for 1010 psig is 545°F
Wider
ΔT during
initial stages
of cooldown
1
2
3
4
TIME (HRS.)

FIG N.B2

6.3 (Cont.)

N.6 ~~B~~a.

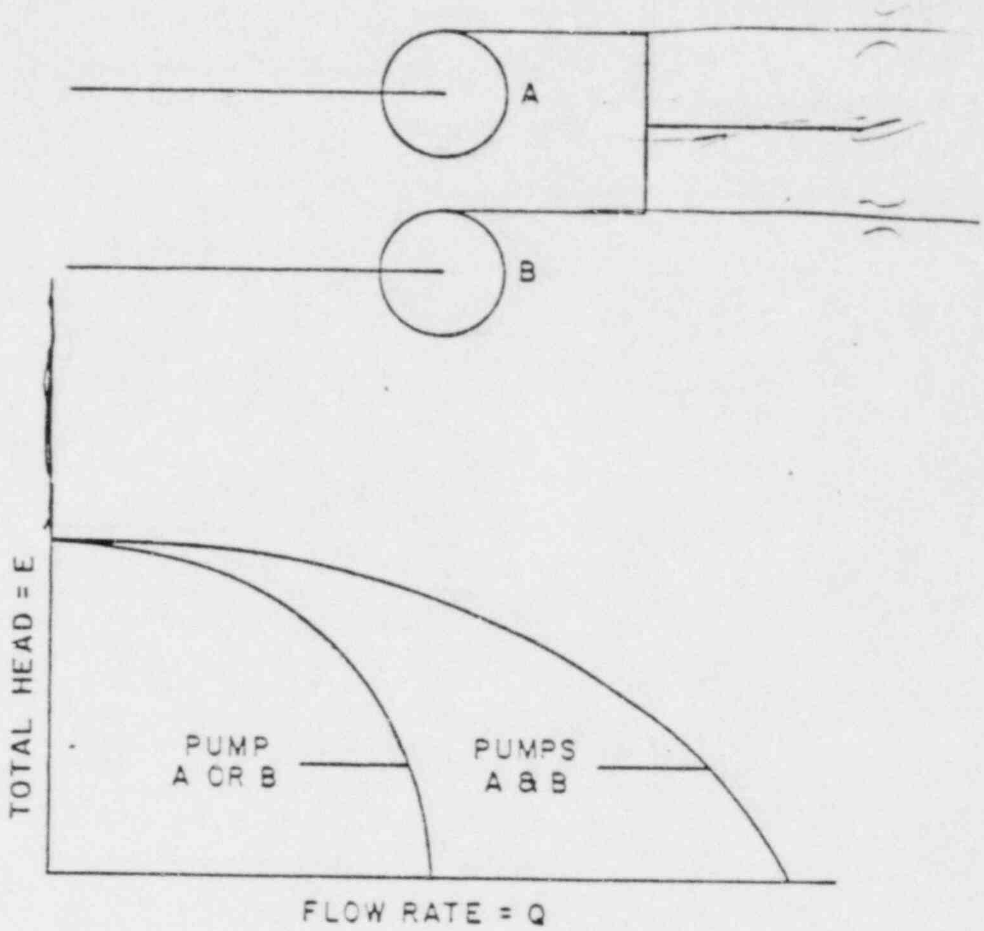


FIGURE 6.3-1 TWO CENTRIFUGAL PUMPS IN PARALLEL

<u>Centrifugal Pumps in Series</u>

By placing two identical centrifugal pumps in series, we find that the flow rate remains the same but the total pump head doubles. A resultant set of pump curves would look like Fig. 6.3-2 (i.e., condensate, condensate booster pumps).

N.1 Cavitating venturi - combination of flow rate & pressure - approx. 550 gpm & 600 psi fluid reaches saturated conditions in venturi and cavitated such that continued flow of liquid is blocked & prevents pump runout.

N.2 a. On figure.

b. On figure.

N.3 OARP - heat transfer fundamental - lesson notes pg 43-48.

a. a-b Adrabatic Compression - fluid is compressed without heat loss as piston strokes closed.

b-c Is thermal expansion - heat is reversibly added at constant temperature with conseq. increase in S.

c-d Adrabatic expansion - fluid adrabatically and reversibly expanded. Piston strokes open.

d-a Isothermal compression - fluid cooled at const. temp to initial state - piston strokes closed.

b. Expansion will be less - c-d or line d-a will be at higher temperature - thus more heat rejected to heat sink - less efficiency.

N.4 a. 1/3/80 GPU letter Module 6
@1800 psi - 175 gpm one pump
205 gpm two pumps

b. 275gpm/1 pump
450gpm/2 pumps

c. Following a unit trip it was customary to isolate letdown, start another makeup pump and if necessary to maintain pressurizer level, open an additional injection valve (16 series). This resulted in unnecessary thermal shocks on the valves. MUV 217 is intended so that opening injection valves & therefore thermal shocks of the HPI lines are no longer necessary.

N.5 a. Ref: thermodynamics by Van Wylen pg. 205
Gbbs - Dalton law for a mixture of ideal gases:
$P_t = pstm + P_{H2} + P_{N2}$

Therefore for a constant P_T, and increasing P_{H2} & P_{N2}, P_{stm} must decrease.

b. Since P_{stm} is the only compressible gas and P_{H2} and P_{N2} are non-compressible the resultant increase in pressure will be greater than if it had been a pure steam bubble.

N.6 a. On figure - 2 same capacity pumps in parallel - head remains the same - flow rate doubles.

b. On figure - 2 pumps in parallel - HPI has much larger head.

U. S. NUCLEAR REGULATORY COMMISSION

REACTOR OPERATOR LICENSE EXAMINATION

Facility: TMI-1

Reactor Type: PWR-B&W

Date Administered: 4/23/81 WED

Examiner: BWilson/BBoger/JMcMillen

Applicant: MASTER- B

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 parentheses after the question.

Category Value	% of Total	Applicant's Score	% of Cat. Value	Category
13	13			A. Principles of Reactor Operation
12	12			B. Features of Facility Design
13	13			C. General Operating Characteristics
13	13			D. Instruments and Controls
13	13			E. Safety and Emergency Systems
14	14			F. Standard and Emergency Operating Procedures
10	10			G. Radiation Control and Safety
12	12			H. Principles of Heat Transfer and Fluid Mechanics
100	100			

Final Grade ________%

A. PRINCIPLES OF REACTOR OPERATION (13)

A.1 a. What two isotopes are primarily responsible for the Doppler Coefficient of reactivity? (1.0)

b. What is the origin of each of these isotopes? (1.0)

A.2 The following data is given for TMI-1 Cycle 5 percentage of core power contribution.

BOL	EOL	β
U-235 - 68%	U-235 - 53%	.0064
U-238 - 7%	U.238 - 7%	.0148
Pu-239 - 25%	Pu-239 - 40%	.0021

a. Calculate the value of β_{eff} for BOL and EOL for cycle 5. (1.0)

b. What effect on reactor control does this difference in β_{eff} value have? (1.0)

c. The percentages from U-235 + Pu-239 change somewhat over core life but U-238 doesn't. Why? (1.0)

A.3 For the following two statements answer which one you believe is true and why. (1.0)

a. When reactor power is leveled off at 10^{-8} amps 5 hours after a trip from equilibrium xenon, since the reactor is critical, xenon will burn out and not peak as it usually does following a trip.

b. When reactor power is leveled off at 10^{-8} amps 5 hours after a trip from equilibrium xenon, the reactor power will have virtually no effect on the xenon peak.

A.4 Which of the following is true for Samarium: (1.0)

a. The production rate is only proportional to power and the only means of removal is by natural decay.

b. The production rate is only from the decay of Promethium but the removal of Samarium is only proportional to flux and therefore power level.

A.5 For the following situation which of the two statements below is true?

From a just critical condition (e.g., 5 X 10^{-8} amps) regulating rods are withdrawn to place the reactor on a positive startup rate: (1.0)

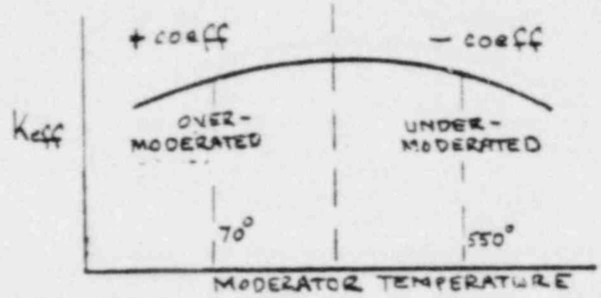
a. Since delayed neutrons come into effect later than prompt neutrons, they begin to multiply more and increase the Startup Rate.

b. Since delayed neutrons come into effect later than prompt neutrons, they lengthen the generation time and decrease the Startup Rate.

A.6 At Beginning of Life with a high boron concentration it is possible to have a slightly positive moderation coefficient. The standard explanation is that as the moderator heats up, the boron is displaced out of the core area such that the positive effect of the reduced boron overcomes the negative effect of the natural MTC.

a. Does this mean that the boron concentration in the core is effectively reduced from say 1000 ppm to something less? Explain. (1.0)

b.



The above curve is contained in the Nuclear Theory Lesson Plan (OARP). Which way does the curve shift with increasing boron concentration? (1.0)

c. How does the Startup Rate <u>and</u> reactor power respond as you enter the point of adding sensible heat with a positive moderator coefficient? (1.0)

A.7 Assume you are watching the response of an intermediate range channel during a trip from full power.

a. Approximately how far (in terms of percent power or decades) will the indicator drop <u>initially</u>? Why? (1.0)

b. The rate of decrease in fission rate is controlled at about -80 sec or -1/3 DPM by the decay of B_r^{87}. The half-life of B_r^{87} is about 55 seconds. Why is the rate of decay on neutron power -80 seconds? (1.0)

B. FEATURES OF FACILITY DESIGN (12)

B.1 The attached Figure B.1 is a simplified flow diagram of the HPI portion of the Makeup and Purification System. Also shown in dotted lines are the modifications that have been made to the original system. For each of the numbered modificatins (3), give the reasons why the modification was made. (3.0)

B.2 List the ___ provisions that will ensure Emergency Feedwater Valves 30A and 30B can be controlled by the operator if necessary due to:

a. Loss of instrument air with off-site power, CHECK (1.0)

b. Loss of instrument air without off-site power. (1.0)

B.3 Explain the following limitations on the NSRW system:

a. Do not attempt to throttle the river water inlet or discharge valves or control temperatures by opening and closing those valves. (1.0)

b. The NSRW system will not be used to back up the Secondary Service River Water System when the reactor is critical. (1.0)

B.4 With respect to the pressurizer:

a. What are the two major purposes of the spray bypass line? Include how opening this valve achieves each purpose. (2.0)

b. Under what two conditions is spray valve (RC-VI) use prohibited? (1.0)

B.5 After taking the shift you notice the following lineup in the nuclear services closed cooling water system. Explain whether or not the system is properly aligned. (2.0)

Pump	Status	Power Supply	43SS TO
NS-P-IA	out of service	IP	IP — 1A
NS-P-IB	normal-after stop	IS	IS — TO 1B
NS-P-IC	running	IS	

REF: GILBERT ASSOC. DWG C-302-661 REV 18

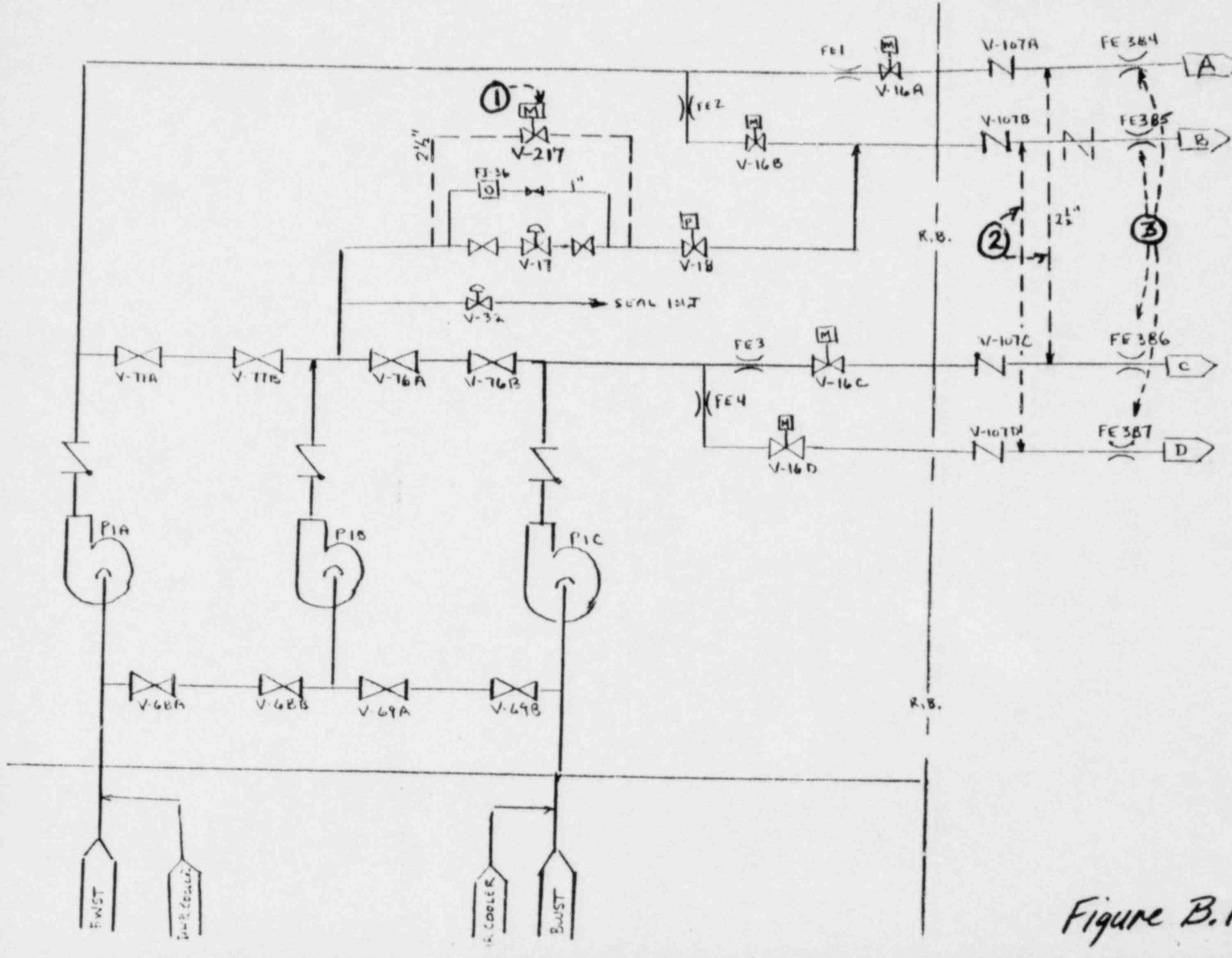


Figure B.1

C. GENERAL OPERATING CHARACTERISTICS (13)

C.1 Attached Figure C.1 is found in the Technical Specification.

a. Give two reasons why control rod overlap is desirable? (1.0)

b. What are the three reasons for rod position limits? (1.0)

c. What are the limitations for operating in the "Restricted" region? (1.0)

C.2 Describe how the turbine bypass valves and decay heat closed cooling water system valves are manipulated to achieve a smooth transition from steam generator to decay heat removal system cooling. (2.0)

C.3 Mixed bed demineralizers are used in several systems at TMI-1.

a. Explain how the principle of ~~selectivity applies to~~ ion exchange in a demineralizer. (1.0)

b. Explain how pure water is obtained from a mixed bed demineralizer. (1.0)

c. Explain why a mixed bed demineralizer can be regenerated. (1.0)

C.4 In June 1980, during a natural circulation cooldown, a steam void was produced in the reactor vessel head at St. Lucie.

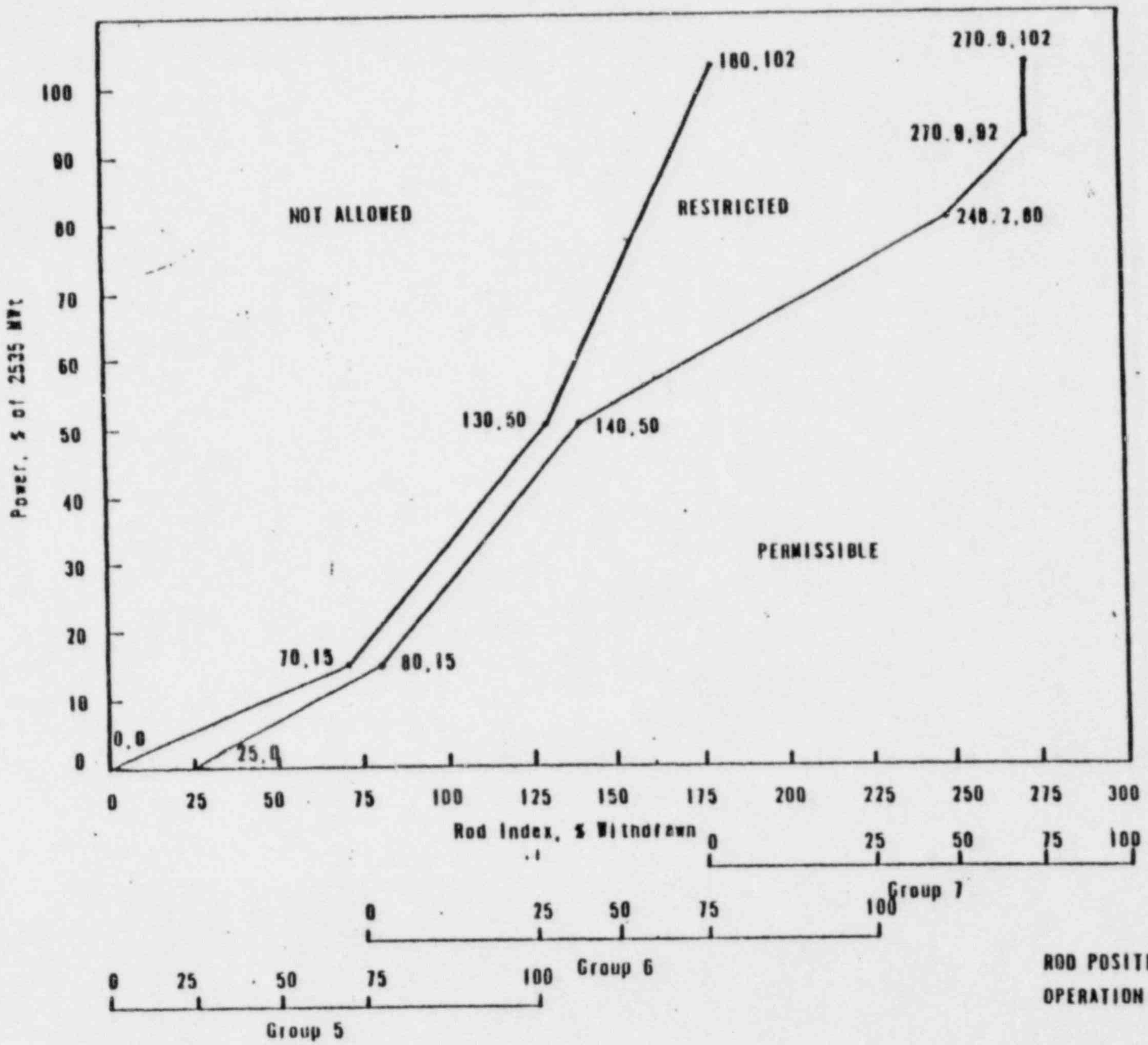
a. Is it possible to form such a void at TMI-1 when the saturation meter indicates a subcooled RCS? Explain. (1.0)

b. Explain pressurizer level response to increasing pressurizer heater output ~~when~~ while a void exists in the vessel head. (1.0)

C.5 While operating the unit at 70% power with both loop feedwater controllers in manual a loop B RCP trips.

a. List five indications that would identify the presence of this tripped RCP. Do not include indications that would result from any automatic actions due to this trip. (1.0)

b. Describe how and why individual loop feedwater flow and total feedwater flow must be adjusted by the operator in response to this RCP trip. (2.0)



ROD POSITION LIMITS FOR 4 PUMP
OPERATION FROM 0 TO 125 ± 5 EFPD
TMI-1

Figure 3.5-2A

Figure C.1

D. INSTRUMENTS AND CONTROLS (13)

D.1 Sensors to start or initiate emergency, safeguard, or control system actions come from a variety of different sources. List the sensors that will initiate an automatic action for the following abnormal situations (for example, high flux as sensed by the linear power range detectors causes the RPS to trip the control rods).

a. Turbine trip that causes the reactor to trip. (0.5)

b. Auto initiation of EFW due to loss of all four reactor coolant pumps. (0.5)

c. Isolation of Decay Heat Removal System due to high pressure. (0.5)

d. Kidney (RB Atmos. Cleanup) System fire protection spray pumps. (0.5)

D.2 The source range and intermediate range channels overlap by 2 decades according to the OARP Manual.

a. At what count rate on the source channels would you expect to see the Intermediate channels come on scale? (0.5)

b. Has the reactor achieved criticality by this time? (0.5)

c. From your answer in (a) to when the source range channels are de-energized, prove that there is at least a two decade overlap. (1.0)

D.3 a. Each section of the nuclear power range channel has one single high voltage input connection and two separate output signal connections. What two indications do you obtain from these output signal connections. Use a simple sketch to explain your answer. (2.0)

b. The source range channels use a discriminator to eliminate gamma signals while the Intermediate use compensating voltage. Why is it not necessary to eliminate the gamma signal in the power range channels? (1.0)

D.4 Figure D.4 is a partial sketch of the ICS control scheme of the EFW Control valves. Complete the sketch showing how the operator can take manual control of the EFW valves from the control room and totally separate from the ICS. (2.0)

D.5 Should the ~~new~~ flow indicators on the HPI injection lines be used to isolate a break? Explain. (1.0)

D.6 When synchronizing a diesel-generator to a bus or the main generator to the grid you must adjust the machine to get proper rotation of the synchroscope.

a. What two parameters does a synchroscope measure? (1.0)

b. What is the proper position of the synchroscope to close the breaker and what is this telling you? (1.0)

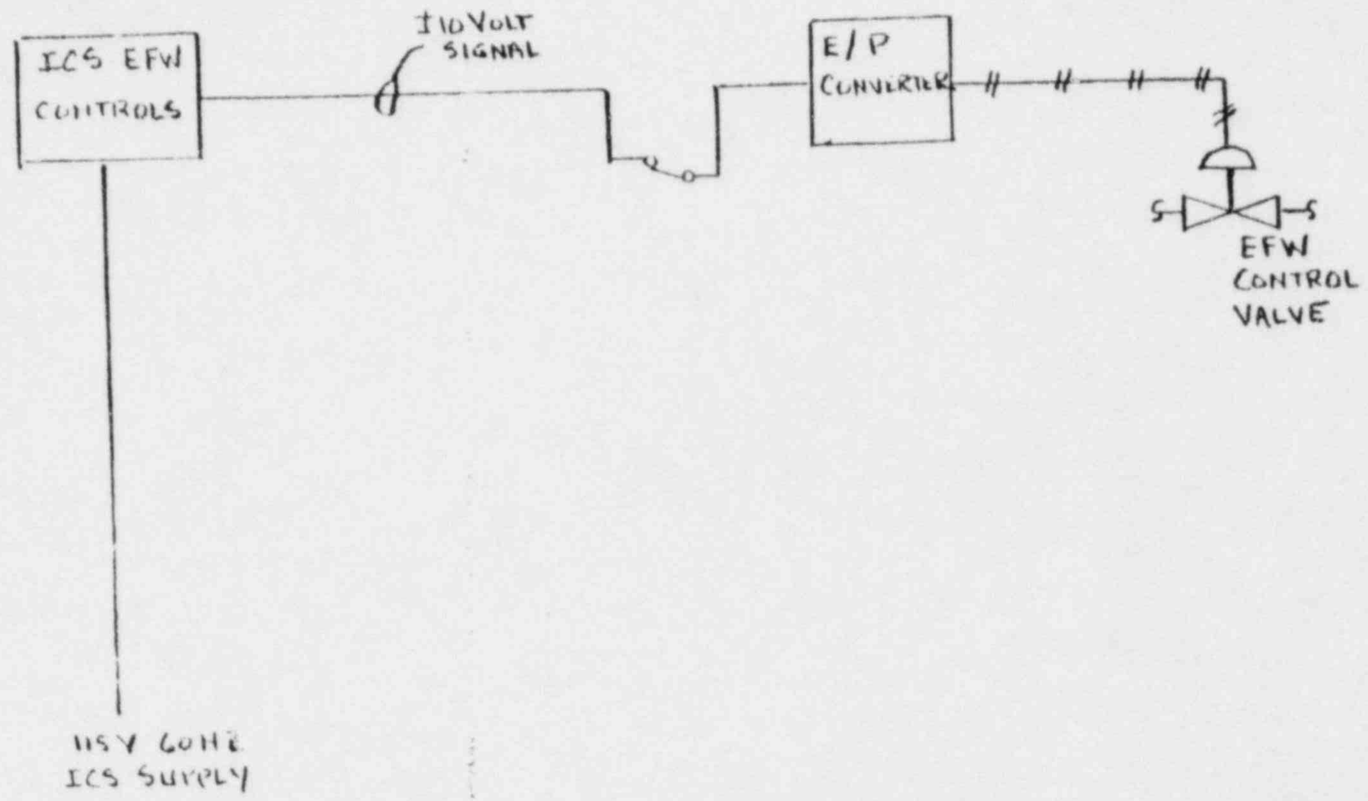

ICS EFW
CONTROLS
±10 VOLT
SIGNAL
E/P
CONVERTER
EFW
CONTROL
VALVE
115 V 60 Hz
ICS SUPPLY

FIG D-4

D.7 When the ICS, portion of the Feedwater System is in full automatic (also full power) by what signal(s) are the speeds of both feedwater pumps controlled? (1.0)

E. SAFETY AND EMERGENCY SYSTEMS (13)

E.1 How can you identify whether one of the steam safeties on the emergency feedwater pump turbine is open after a loss of feedwater transient? (1.0)

E.2 After automatic actuation on a 4 psig RB pressure signal, most ES component status lights on PCR will turn from white to blue, however; a few remain white. For each component listed below indicate whether the status light should be white or blue. If not indicated on the status panel so indicate. (2.0)

a.	IC-V-6	f.	EF-V30A
b.	MU-V-20	g.	MU-P-1B
c.	BKR-G11-02	h.	CF Tank A CF-V-1A
d.	WDL-V-534	i.	MU-V-18
e.	RBS Pump A	j.	NS Deicing NR-V4A

E.3 a. Four reactor trips may be represented graphically on a plot of RCS Pressure vs. Reactor Outlet Temperature. Sketch this figure, label each of the trip functions (boundaries) and give the value or equation of each. Use Figure E-3. (2.0)

b. For each of these trips, give the basis for the trip. (1.0)

E.4 After automatic actuation, of the RB emergency cooling system after a LOCA, how would the following situations affect system performance? Include potential consequences and backup components.

a. RR-V6 fails to open. (1.0)

b. Fan AH-E-IA fails to shift to slow speed. (1.0)

E.5 With respect to the emergency diesel generators:

a. How do you know from the controlroom if the local emergency stop pushbutton has not been reset? (.75)

b. What is meant by speed droop? CHECK (.75)

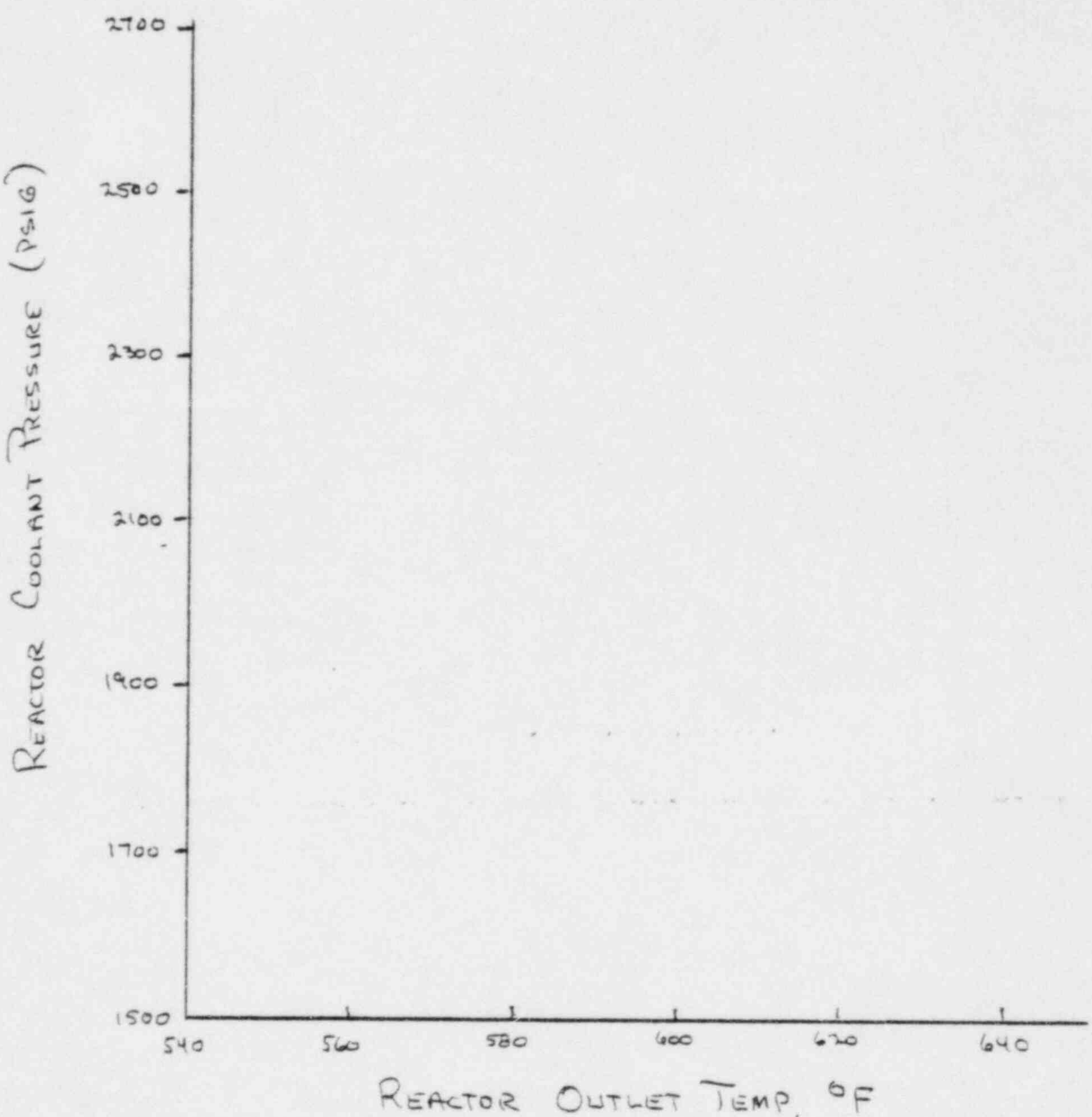
c. List the 3 conditions required to auto start the DC fuel oil pump. (.75)

d. List the ~~three~~ 2 control room alarms that could be initiated by problems with the starting air system. (.75)

E.6 a. List the main and backup sources available to the penetration pressurization system. (.75)

b. How is penetration overpressurization prevented? (.75)

c. Describe how the electrical penetration pressurization subsystem responds to a containment isolation signal. Include status before and after the signal. (.75)



Reactor Coolant Pressure (psig)
2700
2500
2300
2100
1900
1700
1500
540
560
580
600
620
640
Reactor Outlet Temp, °F

Fig E-3

F. Standard and Emergency Operating Procedures (14)

F.1 You are the responsible operator at the controls when the unit experiences a complete loop A main steam line rupture.

a. In this instance, what signal(s) will start the emergency feedwater pumps? Explain your answer. (1)

b. Describe the interlocks on the emergency feedwater pump turbine steam supply valves that ensure a steam supply from the unaffected OTSG. (1)

c. If you are unable to supply feedwater to either OTSG, what are your actions? Include in your answer any components that must actuate as a result of your actions. (1)

F.2 a. Under what three conditions may the operator "at the controls" leave the area immediately in front of the console? (1.5)

b. While at cold shutdown, an instrument technician wants you to grant him permission to withdraw and insert control rods for a RPI check. May you allow him to perform this task? Explain. (1.5)

F.3 a. During the TMI-2 accident, explain why PORV tail pipe temperature did not reach pressurizer temperatures. (1)

b. According to EP1202-29, "Pressurizer Systems Failure", what are operator's directions on determining whether the PORV is or was opened. (1)

c. What design features were installed to aid in the identification of an open PORV? (1)

F.4 The following precautions and/or limitations are found in the TMI-1 procedures. Fill in the blanks.

a. The pressurizer must not be filled with water to solid (440") water conditions at any time except __________. (.5)

b. The boron concentration in the RCS shall not be reduced unless _______ or _______. (.5)

c. When reactor coolant temperature is less than ______ no more than 3 RCP's shall be run at one time. (.5)

d. In case of ______ or ________, borate immediately to 70 degrees F, 1 percent $\Delta K/K$ shutdown concentration. (.5)

570 S.G 532 Feed nozzles

F.5 a. Explain why feedwater is desirable during small break LOCAs. (1)

b. How can you determine whether an accident is a LOCA or a non-LOCA overcooling event? (1)

c. How can heat be removed from the core if both hot legs become steam bound during a small break LOCA? (1)

ROUTING AND TRANSMITTAL SLIP

Date

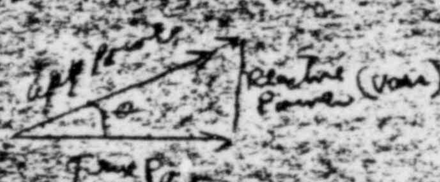
TO: (Name, office symbol, room number, building, Agency/Post)	Initials	Date
1.		
2.		
3.		
4.		
5.		

	Action		File		Note and Return
	Approval		For Clearance		Per Conversation
	As Requested		For Correction		Prepare Reply
	Circulate		For Your Information		See Me
	Comment		Investigate		Signature
	Coordination		Justify		

REMARKS

Apparent Pow = Volt x amp
voltamperes

True Power = Volts x amps x cos θ
watts



v x a x Sin θ = vars

DO NOT use this form as a RECORD of approvals, concurrences, disposals, clearances, and similar actions

FROM: (Name, org. symbol, Agency/Post)	Room No.—Bldg.
	Phone No.

5041-102

OPTIONAL FORM 41 (Rev. 7-76)
Prescribed by GSA
FPMR (41 CFR) 101-11.206

G. Rod Control & Safety (10)

G.1 Define the following terms (do not use formulas):

a. Radioactivity (0.5)
b. Gamma radiation (0.5)
c. Effective half-life (0.5)

G.2 How are RAD and REM related? (0.5)

G.3 List five different limitations or criteria that you would expect to ~~find in~~ 10 CFR 20? (1.0)

G.4 I-131 is a radioactive material that is often encountered at a reactor. Give the nuclear process for the formation of this isotope and why it is considered to be a hazard. (2.0)

G.5 In the use of radiation survey instruments: (1.0)

a. an ion chamber instrument should be used to determine dose rate.

b. a GM tube instrument should be used to determine dose rate.

c. an ion chamber instrument should be used to survey for low level contamination.

d. a GM tube instrument should be used to survey low level contamination.

e. a and d are correct.

f. b and c are correct.

g. all of the above are correct.

h. none of the above are correct.

G.6 Tritium is produced in large quantities in a PWR. Will the containment area monitors alarm if the tritium concentration in the containment increases? Exlain your answer. (1.0)

G.7 Fuel assemblies located in a reactor facility can be generally classified in three categories; (a) new fuel, (b) recently removed irradiated elements and (c) irradiated fuel stored for some time. Which of these classifications of assemblies would presenta direct radiation hazard? (1.0)

G.8 A 28-year old operator has accumulated a life time radiation dose of 49.4 Rem up to the beginning of this most recent quarter. He has been told to go and open a valve which is located 8 feet from a source that reads 32r/hr at 2 feet. If it takes him 15 minutes to open the valves will he exceed the dose permitted by 10 CFR Part 20? (Show all work). (2)

H. Principles of Heat Transfer and Fluid Mechanics (12)

H.1 Evaporation and boiling are both thermodynamic processes in which a liquid such as water changes phase to a vapor such as steam. What is the difference between evaporation and boiling? (1)

H.2 Define the following terms:

a. BTU (.5)
b. enthalpy (.5)
c. latent heat of vaporization (.5)
d. adiabatic (.5)

H.3 Figure H.3 is a longitudinal section of an OTSG. At full power, feedwater enters the OTSGs in a slightly subcooled condition and exits as superheated steam.

a. Briefly explain the design feature of the OTSG that brings the feedwater from subcoooled to approximately a saturated condition. (1)

CHECK b. Show on the Figure the three regions of heat transfer in the OTSG as the feedwater/steam mixture rises up the tube bundle. Indicate the relative location or area of each. (2)

c. Which of the three regions has the best heat transfer characteristics? Also, second best and the worst? (1)

H.4 You are told to determine whether a heat exchanger is experiencing tube fouling.

a. How and why does tube fouling affect the heat transfer capabilities of a heat exchanger? (1.5)

b. Figure H.4 represents an elevation view of the heat exchanger. Using the pressure indications on this figure, is the heat exchanger exceeding the design limits for differential (inlet to outlet) pressure of 15 psi? Show your work and explain your answer. (1.5)

H.5 Figure H.5 is a cross sectional view of a fuel rod and coolant channel.

a. On Figure H.5 sketch the temperature profile from point 1 to point 5 for operation at 100% power. (.5)

b. Where does the best heat transfer occur? Explain. (.75)

c. Where does the worst heat transfer occur? Explain. (.75)

CHECK

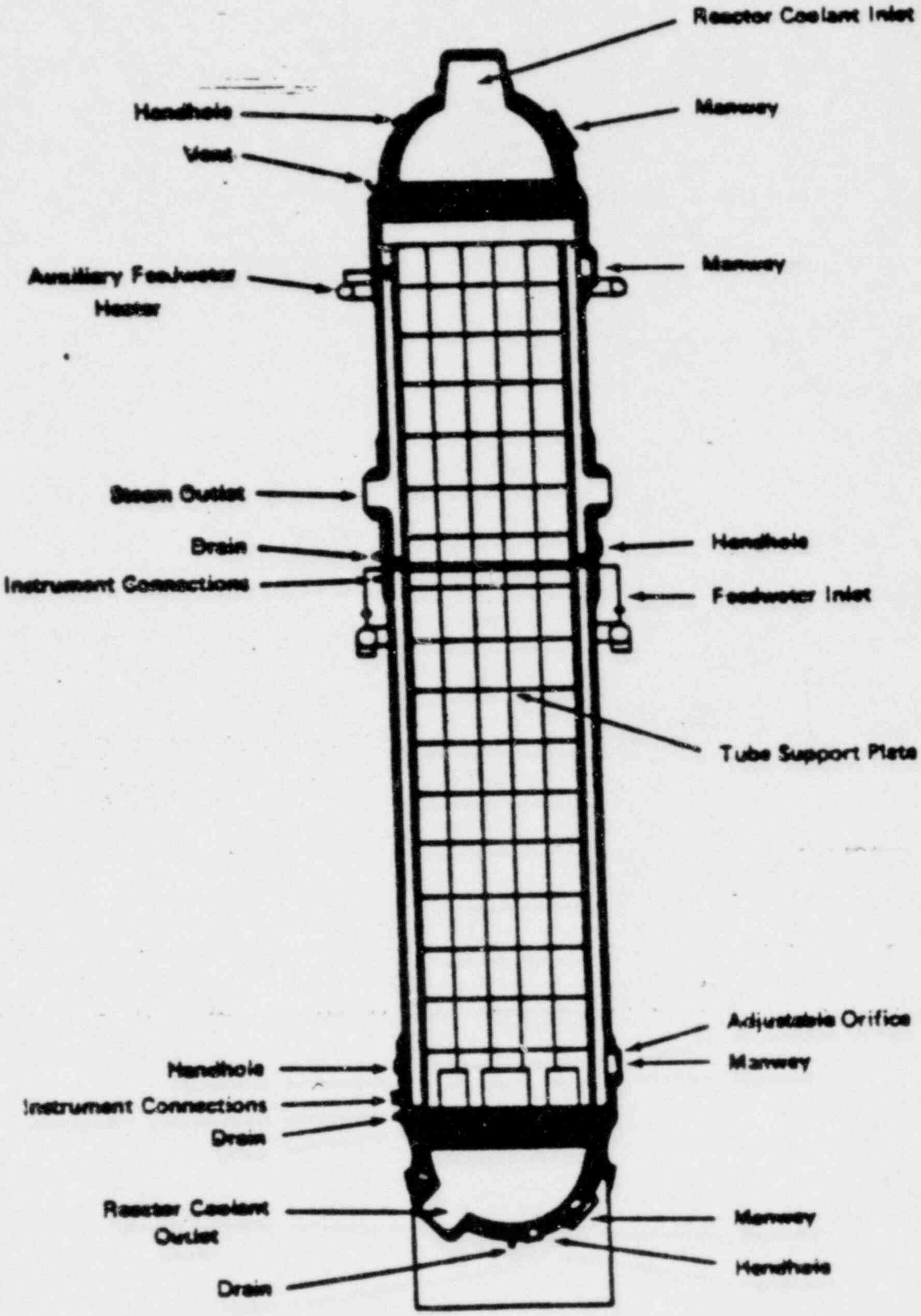


Figure H-3 Once-Through Steam Generator

377

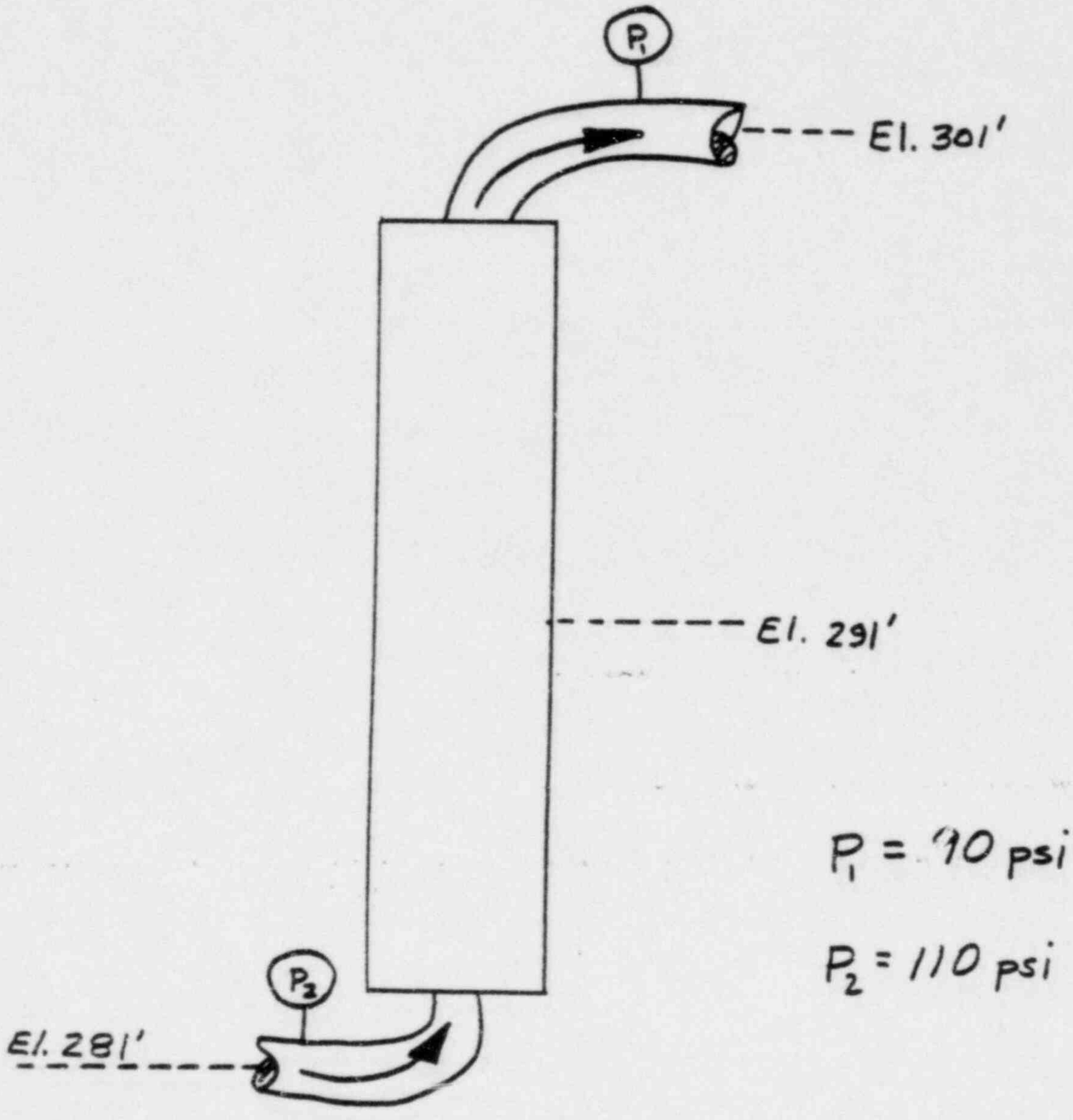


Figure H.4

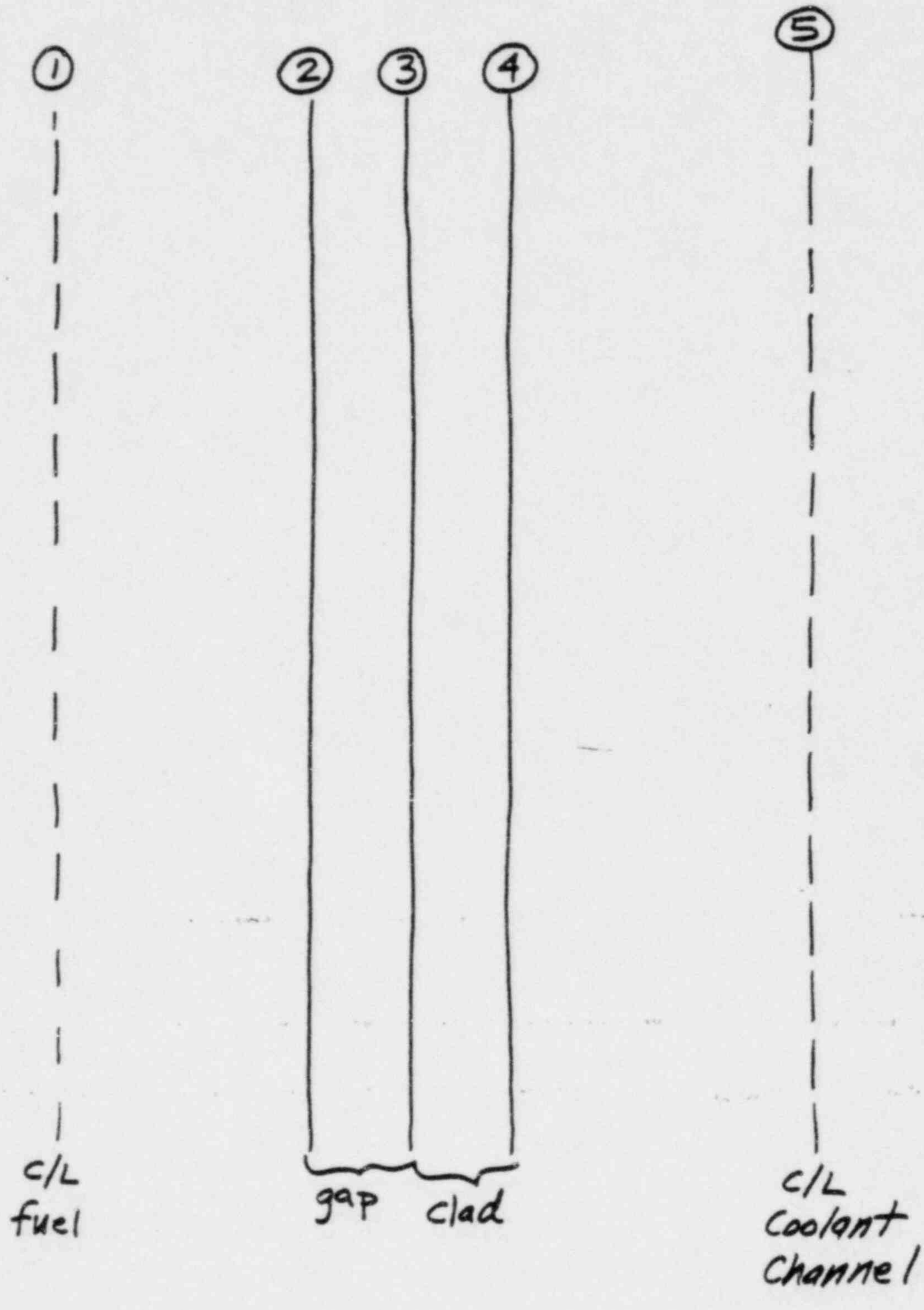


Figure H.5

EXAM B ANSWERS

(OARP pg 29)

A.1 a. U-238, Pu-240

b. U-238 Naturally occurring ~ 98.5% of fissionable material.
Pu-240; $U\text{-}238 + {}_0n^1 \rightarrow U^{239} \xrightarrow{-\beta} {}_{93}Np^{239} \xrightarrow{-\beta} Pu^{239} + {}_0n^1 \rightarrow Pu^{240}$

A.2 a. OARP pg 37
BOL β = .0058
EOL β = .00523

b. Since $T \propto \frac{l}{\beta - \rho}$ the period will be shorter for lower value of β.

c. Relatively, U-238 is a hugh mass, U-235 is only 2-3% enriched, Pu-239 comes from U-238. Mass of U-238 can essentially be considered ∞. and fissions by fast on' only.

A.3 b. 10^{-8} amps is ~ 6 decades below full power, Xe is ϕ dependent - virtually no effect on xenon.

A.4 (b.) OARP pg 28

A.5 (b.)

A.6 a. No. Boron is in solution in the water in Terms of parts per million.

$\frac{\text{Weight of Solute X } 10^6}{\text{Weight of Solution (water)}}$ = ppm which is a very dilute solution and concentration will not vary unless solubilitie limits (e.g., freezing temps) are reached.

Nuclear Theory Lesson Plan

b. pg 29 Curve shift to the right and down with increasing boron concentration.

c. As moderator heats up, $+\rho$ will be added however, Tave is controlled by steam dumps at 532° while fuel temp must increase to create Δ T to increase mod temp. therefore Doppler should overcome + MTC.
SUR ↓ due to Doppler

A.7 OARP pg 47

a. Prompt drop goes to ~ .5% reactor power. or 2 decades KB Absorption & prompt $_0n^1$ $(1-\beta) \approx .95$

b. Decay constant $\lambda = \frac{.693}{T_{1/2}} = .0126$

or Mean Life $\bar{\tau} = \frac{1}{\lambda} \approx \frac{55.6}{\ln 2} = 80$ sec.

ADDED TO PROVIDE ADDITIONAL MAKEUP CAPACITY. ANSWER MAY ALSO SAY NEEDED TO PRECLUDE NECESSITY TO OPEN MUV 16B THUS GIVING A THERMAL SHOCK SHOULD GIVE CREDIT BUT NOT REQUIRED

B.1 1. RM-14 pg. 3 - Following a trip, due to pressurizer shrink MUV-17 would open full but was only able to pass ~75 GPM. This was usually not sufficient to maintain pressurizer level and it was necessary for the operator to open MUV-16B. In order to limit the thermal shocks on the injection nozzles, the 2½" bypass line including MUV-217 was added to augment makeup flow.

2. RM-14 pg. 2 - A postulated break between the RCS cold leg and the first isolation check valve would result in both a SBLOCA and a HPI line break. Certain failures, coupled with this type of break would prevent the HPI system from performing as required. The cross-connections ensure HPI to all legs.

MAY ALSO GET SOMETHING SAYING PRECLUDE LOSS - FI THRU A FAILURE.

3. RM-14 pg. 2 - Manual throttling of HPI valves may be necessary requiring an operator to enter a high radiation area (potentially). In order to eliminate the need for operator action, cavitating venturis have been designed to limit the injection flow to less than that which would cause HPI pump runout without restricting the flow to unacceptable levels in other cases.

B.2 SDS 424A pg. 3

CHECK! 2HR BOTTLE SUPPLY?

a. Station Air Valve 1A-V1 opens @ 75 to 80 psia.

b. *Reservoir @ valve good for several cycles*

c. 80 gal receiver tank supplied by 5 Hp, 18 SCFM air compressor = supply line opens @ 70 psig. (*all I.A. compressors*)

d. *ecm for air bottles & emergy air supply header*

MAY GET ANSWER [illegible] AND [illegible]

B.3 a. Assures maximum flow during accident situations.

PROC

b. SSRW is not safety related and a failure in this system would prevent proper operation of the NSRW System during an accident.

Ref: OP 1103-5

B.4 a.1. Maintain temperature in spray line - with constant flow line will not stagnate at lower temperature.

(3)

2. Equalize boron concentration between RCS and pressurizer with constant flow dump RCS to prz, causes heaters to stay energized which causes outsurge back to RCS.

Tech Specs

b. Per 1101-1, Limits & Precautions - (1) if ΔT between prz & cold leg is 250°F (1103-5 says 430°F). (2) if N_2 bubble exists unless prz temp. <210°F & H_2 pressure <3 psig.

(1)

B.5 Not properly aligned since B pump is aligned to same power supply as 'C'. On ES and failure of IS power supply no NS pump would be available. 'B' should be lined up to 1P.

CAR°

(9½)

C.1 a. Constant reactivity addition minimizes peaking factors

b. Shutdown Margin
Ejected CEA worth
Local peaking factors

Accept 2 Reasons also may get 3 or 2.

c. Inadvertant operation for up to 4 hours is allowable without exceeding LCO

C.2 Initially have cooling via TBV's & DHR system in service. Open DHCCS bypass around DHR Cooler (DC-V65 A & B) and slowly open DHR cooler inlet (DC-V2A & B). As the DHR cooler becomes effective and starts to cool the RCS, the TBV's may be throttled. Continue process of closing bypass opening inlet and closing TBV's.

OARP

C.3 a. Resin beds hold H^+ or OH^- at collection sites. Due to the order of selectivity, the impurities are more attracted to the collector sites - they are selected to remain on the resin bed while the H^+ or OH^- ion is released. (1)

OARP

b. In a mixed bed demin, the cation resin will collect + impurity and release H^+ while the anion resin will collect - impurity and release OH^-. H^+ and OH^- combine as pure water. (1)

c. Highly concentrated acid or base will reverse the process and cause the resin beads to release impurities and collect H^+ and OH^- ions. (1)

Anion gets caustic
Cation gets acid

C.4 a. Yes, since the saturation meter looks at the highest T_H which may not be indicative of RV head temperature if insufficient head flow exists. B&W SAYS WE DON'T HAVE A PROBLEM + MIXING IS ADEQUATE — MAY GET EITHER

St. Lucie

b. Heaters output increase would heat water in prz., which would increase pressure in prz (it is at saturation) which would tend to collapse void in head, which would cause prz level to drop.

C.5 Expected Transient Resp. Pg. 10
RCP Motor Trip Alarm
RCP Breaker Lights
Total RCS Flow ↓
RC Total Low Flow Alarm
RC Loop B Flow ↓
RC Loop B Flow low alarm

(1)

MANY ANSWERS INDICATE TEMP VARIATIONS

PLEASE CHECK ME

b. Why? To bring ΔTc into acceptable values and to keep RCS pressure within acceptable bands.
How? Increase FW to loop A while RCS pressure >2155 psig and ΔTc is positive, decrease FW to loop B while RCS pressure is <2155 and ΔTc is positive, this way ΔTc is being controlled by individual flows and RCS pressure via total FW flow. (2)

D.1 a. Low EHC oil pressure

b. Sys. Desc. 424A=pg. 9 52 X 1 relays from RCP monitors 1 + 2

c. GAI DWG 302-640 RC3A-PS 2 for DH-V1 RC3A-PS5 for DH-V-2 control bistables
All press transmitters are on hot legs - cannot identify from C-302-650.

d. OARP RB Vent Sys. pg. 39 - FSP 5A/B auto start whenever two temp switches close.

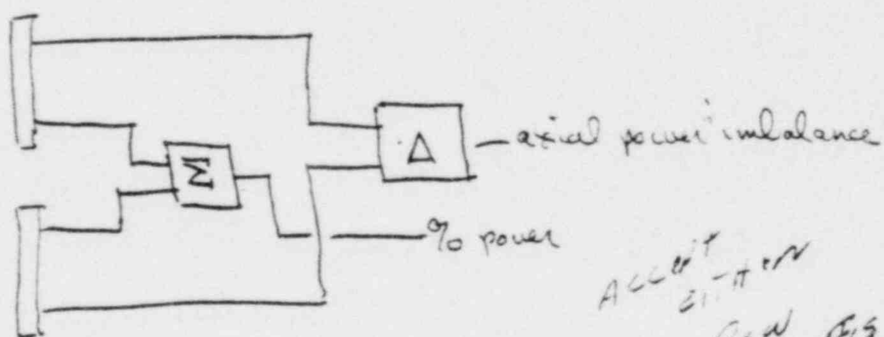
D.2 a. OARP pg. 12 - Fig N-2 do not agree IRM's do not indicate until SRM 10^4 CPS - go off scale @ 10^6 cps.

b. Yes, reactor should be critical somewhere around 10^3 cps - 10^4 cps.

POOR QUESTION TO TEACH TO CRITICALITY.

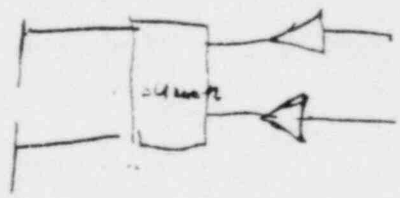
c. NI-3 & 4 deenerg. 1 & 2 @ 10^{-9} amps. - lower range of 3 & 4 is 10^{-4} amps - therefore must be indicating prior to 10^4 cps on SRM.

D.3 a.



ACCEPT EITHER

SEE C+W NSS Fig NI-1 SECTION 15



b. OARP pg. 13 & 14 1% of detectors full output is due to gamma field. The major of the gamma field is directly proportional to reactor power and the gamma response does not introduce any significant error provided it is held to a small percentage of the total signal.

D.4 On figure.

D.5 Quiz in Volume 3 of OARP
No - Cavitating ventures make it impossible to determine which leg contains a break by reading flow instr.

UARP VOL 3 SDD 424A Pg 7 + Sketch

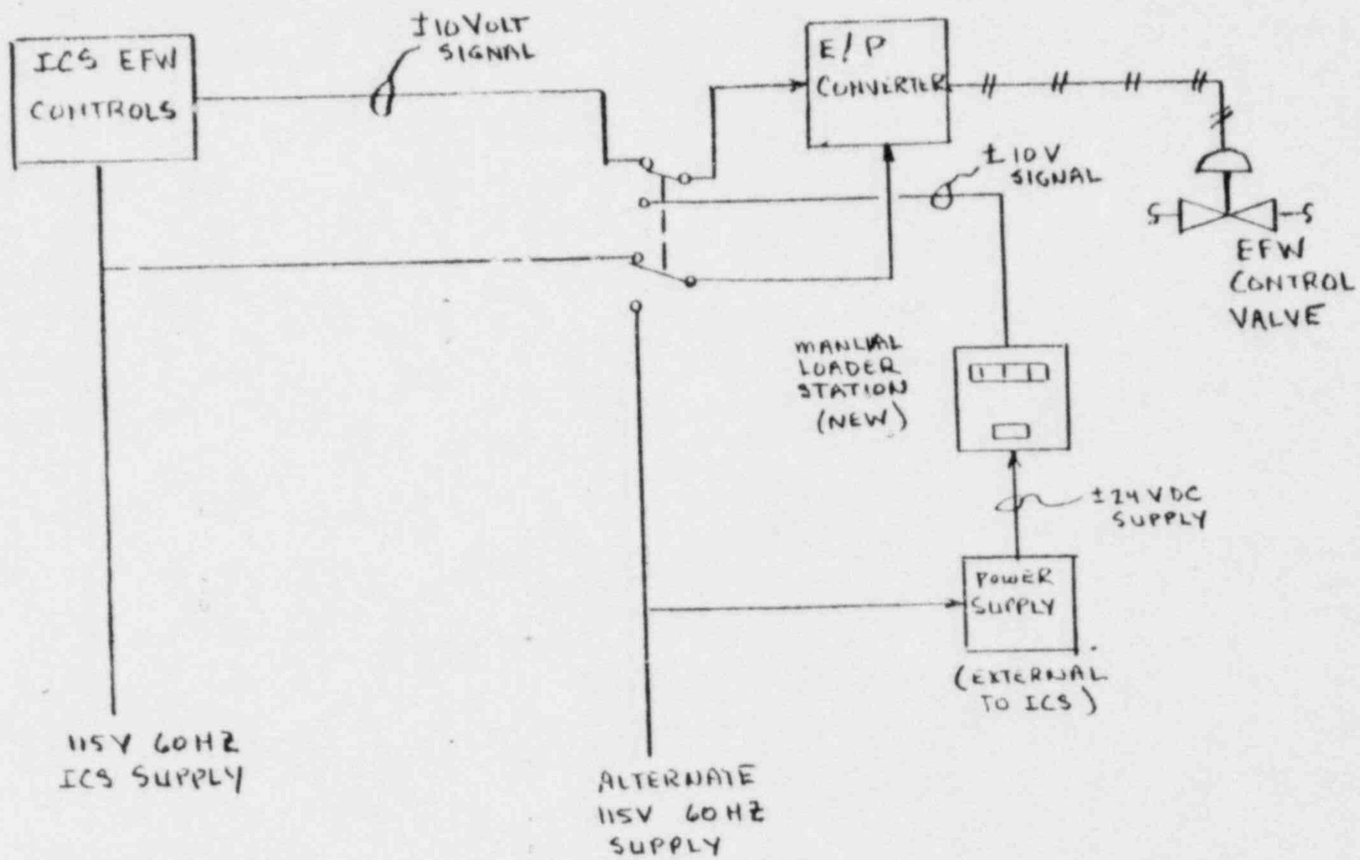
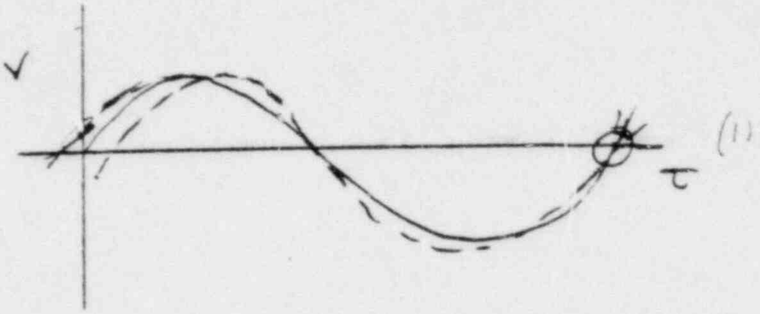


FIG D-4

D.6 a. Phase angle & different in frequency. (1)



e.g. system @ 60 Hz above represents cycle in 1/60" of sec.

b. Dotted line indicated slightly > 60 Hz and systems in phase @ circle.
Close breaker @ 5 to 12 to give closing time so phases meet. OR SLIGHTLY BEHIND GRID SO THAT GRID DRAGS TG UP TO IT AT 12

D.7 OARP Module 3
Fw System lecture
Two signals
1. Total Fw ~~flow~~ demand from ICS
2. Lowest Fw Valve ΔP error selected by ICS

MANY SAY
(1) NOISE TRANSDUCER
IN TAIL PIPE
CAUSES ALARM.
GAVE CREDIT
(2) INDICATING LIGHTS
?(RED) FOR OPEN
VALVE POSITION

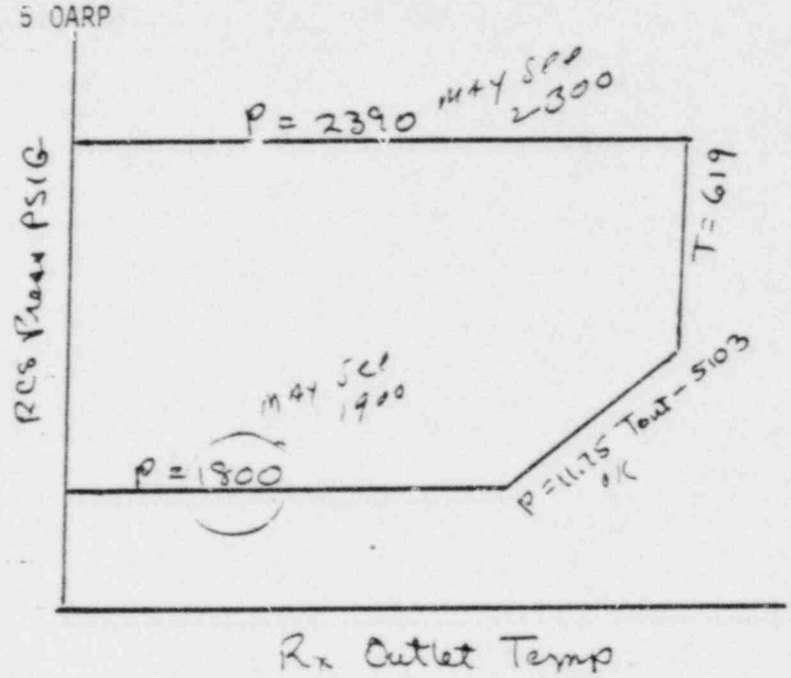
OARP - SYS. MODS. from noise Transducer

E.1 ~~PORV~~/SV indication panel will be in alarm - one of the channels will be lit (channel 5), EFPT discharge pressure will be low.

E.2 OARP
Containment Isol Lecture pg. 5 — Change White to Amber
a. White (~~Pressurizer~~ CFD inlet)
b. White (Seal Inj.)
c. White (~~Seal Inj.~~) D.GEN
d. Blue (Cont. sump)
e. White (RB spray)
f. not indicated (EF feed value)
g. Blue (Makeup pump)
h. White (isol. valve) ? — Being changed may get ans Blue
i. Blue (~~Seal Ret. Isol~~) make up inlet
j. Blue (NS. deicing)

E.3 RPS Lecture - Fig. 5 OARP

a.



b. RPS lecture
2300 P = 2390 hi Press
Limit integrity of RCS piping & components prevent lifting safeties, trip setting established by ASME Code Section III Article 9, Winter 1968. ?

1900 P = 1800 - Maintain > DNBR 1.3 for Transients & accidents
T 619° - Prevent excessive core coolant temps in operrange.

Variable temp/Press - Maintain DNBR 1.3

E.4 OARP a. This is common discharge value, system will not remove sufficient heat. Have MOV-RR5 which can be used by bypass RR-6.

? b. Fan will probably overload when trying to pump denser atmosphere (Trip?) the two other fans and containment spray act as a backup.

E.5 a. Will have overspeed alarm (pg. 8). OARP — LOCAL TROUBLE ALARM IN CR

b. % drop in speed (% change in load) pg8
Allows engine speed to vary w/ load
ALLOW DIESELS TO FLOAT WITH GRID FREQ.
GAVE CREDIT

c. EDG start signal, SW in auto, oil pressure < 10 psi sensed downstream of duplex filter.

d. Low pressure, manual valve closed, SW not in auto. ASKED FOR TWO
ALARM RESPONSE — DIESEL TROUBLE, FUEL BLOCKED
OARP pg17

MECHANICAL

E.6 a. ESAS Lecture pg. 9 & 10
Main: Instr. air @ 60 psig
Backup: N_2 @ 59 psig
- service air @ 57 psig

ELECTRICAL
MAIN: N_2 @ 30 PSIG
BACKUP: INST AIR @ 60 PSIG

b. Overpressure regulator, relief valve

c. Normally @ 30 psi w/N_2, after CI signal N_2 supply & relieving devices are isolated and the main subsystem supplies the penetration pressurization.

GAVE FULL CREDIT.
ANSWERS INDICATE < 50# ΔP ACROSS MFPs START EFPs.

F.1 a. AP 1203-24
1600
Complete rupture will initiate ESAS since less than 1500 psig all 4 RCP's must be tripped - this signal starts the EFWPs. Loss of both MFWPs would also start EFP but mechanism to cause MFP trip is unknown.

b. MSV-13A opens first of OTSG 'A' pressure is greater than 100 psig
MSV-13B opens if OTSG 'A' is <100 psig and OTSG B is >100 psig.
Mod — interlock deleted

c. Initiate full HPI
HPI pumps start, 16A → D valves open
Open PORV & block valve,
Suction from BWST open

@600psig - on OTSG A Stm line - V-16, 17, 5, 92, 30 shut

F.2 AP 1028

a. 1. Verify receipt of an alarm, investigate parameters relating to suspected abnormal conditions.

2. Initiate corrective action in the event of an emergency.

3. Operate aux. equipment control which are located in the area.

b. No, 10 CFR 55 does not allow the manipulation of the "controls" unless they are in training for a license (establish cooling water to CRD's). ①. Admin reason ②. Equip prot. MIN PRESS FOR OPERATION ✓

F.3 a. Isenthalpic Expansion out of PORV - Enthalpy remains constant - press & temp decrease.

b. EP 1202-29. Temp 200^{0}F if open for 10 sec. 10 min. Required to cooldown to 200^{0}F.

c. ΔP indicator and alarm
Acoustic monitor alarm

ANSWERS INDICATE UP TO 10 WAYS GAVE CREDIT DIDN'T HOLD THEM TO EP

F.4 a. 1102-1 pg. 3....

as required for system hydrostatic tests ME. ALSO got Required by EP?

b. 1102=1 pg. 6....

At least one RCP or DHRP is circulating reactor coolant ←

OTHER ANS SAFETY RODS OUT - SD MARGIN VERIFIED GAVE CREDIT

c. 1102-1 pg. 3 500^{0}F

d. 1102-10 pg. 6 Stuck Rod, if cooldown is certain

F.5 a. Small Break Operator Guidelines
Energy heat removal is insufficient due to size of break - must have heat sink mechanism - in this case, OTSG's are only heat sink so must have FW.

b. EP 1202-6B

7. Evaluate for a Non-LOCA overcooling event by observing the following:

NOTE: During a loss of coolant accident, RCS conditions will approach saturation. Temperature will remain fairly constant and pressure will decrease to saturation pressure. During a Non-LOCA overcooling event, pressure and temperature will both decrease rapidly. Some subcooling will be maintained unless pressurizer level is lost.

NOT GAVE MORE THAN CR TEMP + PRES

a. RCS pressure, temperature, and saturation margin.

b. OTSG pressure: Decrease for steam line break.

c. OTSG level: >50 percent for overfeeding event, low level for steam line break.

d. Main and startup feedwater flow: High flow for overfeeding event, isolated for steam line break.

If non-LOCA overcooling is indicated proceed per Attachment-2.

F.5 c. Attachment 3- ICC
Open RC-RV-2 ← SAME AS NO OTSG AVAILAB -.
& subsequent steps

HPI ON
PORV & BLOCK OPEN
BWST SUCTION

or Reflux Boiling Mode

G.1 a. The property of some atoms of emilting alpha, beta, gamma or some other particle or wave to establish a more stable nucleus.

b. Short wave elect-omagnetic radiation very similar to X rays.

c. A weighted half life of a radioactive material which takes into account the decay characteristics of the material and the ~~eon~~ reten tion of the material within the body.

G.2 RAD X QF = REM

G.3 MPC - liq + gas, restr & non restr. Quarterly, yearly dose limits, whole, skin, extremity, repating requirements, etc.

G.4 I-131 is fission product that can easily escape through a leak in the primary system. If it gets into a room it may become a gas and can easily travel to the environment. If breathed or taken in with milk it concentrates in the thyroid gland and can cause a large dose in this area.

G.5 (e)

G.6 No. Tritium decays by emission of a very low energy Beta which cannot be detected by area monitors.

G.7 Irradiated elements both recent and stored for some time.

G.8 (5)(n-18) = 5(28-18) = 50 Rem permitted
has accumulated 49.4
.6 rem still in book

work will add $\frac{D_1}{R^2} = \frac{D_2}{R^2} = \frac{32}{8^2} = \frac{P_2}{2^2}$

? still 1/4 R/8

$D_2 = \frac{32}{64} \times 4 = 2.0 R/hr$

only work for 1/4 hr. so D = 0.5 rem

will not exceed 10 CFR 20 still has 0.1 rem in bank

H.1 OARP pg. 5

Evaporation - phase change occurs below boiling point - boiling - liquid at saturated conditions - temp. of total pressure above free surface.

H.2 a. OARP pg. 14; Heat req to raise 1lbm of water from 59.5° to 60.5°F

b. OARP pg. 21; H=U + pV (Internal Energy + Press. X volume)

c. OARP pg. 5; heat supplied to a liquid to change it to a vapor without changing its Temp.

d. OARP pg. 14; Q=0 process which there is no heat transfer.

H.3 a. Aspirating ports in SG shroud allow ~10% steam flow to preheat FW at the sparger.

b. Nucleate boiling - largest area @ full power; film boiling constant area over power range; superheat - area decreases as power.

c. Best to worst are same as above - film boiling tends to form insulation of tubes.

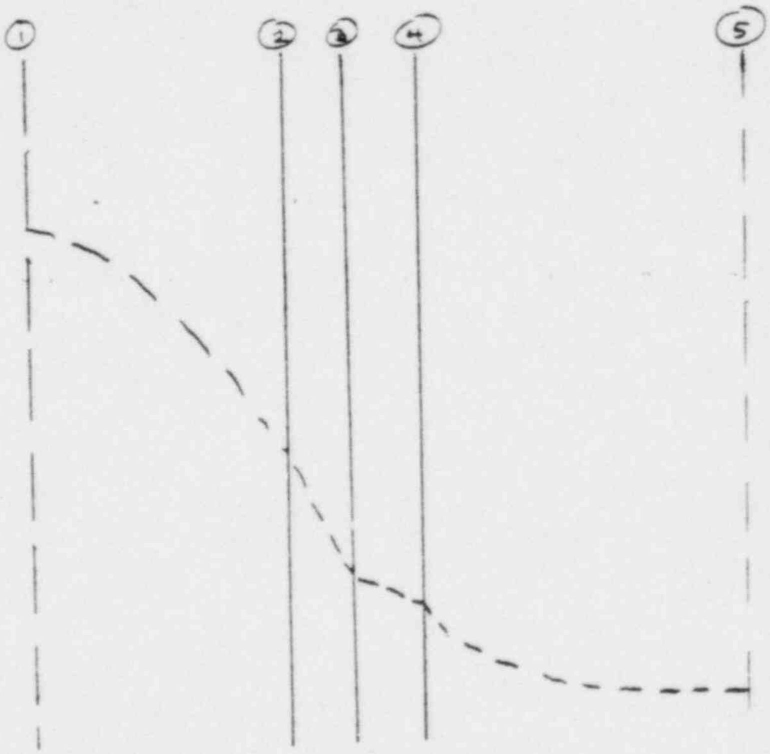
H.4 a. Decrease h - heat trans coeff. - creates. Essentially creates film barrier across tube.

b. Indicated Δp = 20 psi; however, there is an evaluation difference of 20 feet. The elevation difference accounts for 20 X .433 psi/ft = 8.66 psi. Therefore, heat transfer ΔP = 20 psi - 8.66 psi = 11.34 psi. Heat transfer is not exceeding Δp.

H.5

H.5

a.



b. Best heat transfer occurs across Zircalloy cladding. Conductive heat transfer through metallic body.

c. Worst heat transfer is across helium gap - element is pressurized with helium.

3.6 Shift Relief

3.6.1 a. All shift operations personnel shall be responsible for maintaining their duty station until properly relieved. The Shift Supervisor, Shift Foreman, Control Room Operators and Auxiliary Operators shall be relieved by qualified personnel only e.g. those personnel who are properly licensed and properly informed of the plant status, operations in progress. and any special instructions which may be applicable.

b. Prior to relieving the shift the relieving individual will discuss plant status, operations in progress, shift turnover checklist.

c. The oncoming CRO, SF or SS will each initial the ES Checklist to indicate his understanding.

3.6.2 Prior to assuming the shift, the relieving individual shall review station logs, records, special instructions, etc., which have been generated since his last shift. The logs and records to be reviewed should include:

1. Shift Foreman Log (Shift Foreman)
2. Control Room Log (Control Room Operator)
3. TCN and SOP Books (Shift Foreman and Control Room Operator)
4. Operations Memo Book (Shift Foreman and Control Room Operator)
5. Revision Review Book (Shift Foreman and Control Room Operator)

3.6.3 The Control Room Operator will acknowledge his understanding and awareness of the changes in the plant status since his own last entry by signing the Control Room Log prior to assuming the shift duty.

LESSON NO.: ______
REVISION NO.: 0
DATE: 8-12-80

REACTOR COOLANT LOOP FLOW

INSTRUMENT FAILURE

INDEX

PAGE

LIST OF FIGURES

LESSON NO.: ______
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REACTOR COOLANT LOOP FLOW

FAILURE

Goal

The learner will understand and be able to explain how a B & W Nuclear Steam System will react to the transient: reactor coolant loop flow instrument (pressure transmitter) failure.

Instructional Objectives

Given specific initial plant conditions and an overview of the transient:

A. The learner will be able to explain (write) how, when the electrical transmitter output fails, the RC flow indication will fail.

B. The learner will be able to explain (write), how and why, when the RC Flow indication fails to 50% (45×10^6 lbm/hr.), the ICS will respond.

C. The learner will be able to explain (write the plant overall consequences of the RC Flow failure.

D. The learner will be able to draw (sketch) the trend of plant related parameters during this transient, and explain (write) the reasons for such parametric trends.

E. The learner will be able to draw (sketch) the trend of plant related parameters for a synthesized transient, similar in nature to the transient discussed (RC Flow Failure); and explain (write) the reasons for such parametric trends.

LESSON NO.: ________
REVISION NO.: 0
DATE: 8-12-80

I. INITIAL PLANT CONDITIONS

A. Reactor at 100% power, all ICS stations are in automatic (integrated mode). The unit is at 870 MW_e with $T_{average}$ at 579°F.

II. OVERVIEW OF TRANSIENT

A. A failure of the differential pressure transmitter for the Reactor Coolant Loop Flow modifies Unit Load Demand and ΔT_c Control, causing the plant to runback and feedwater flow to be ratioed. The plant finally stabilizes out at approximately 78%.

III. SPECIFIC STEPS OF TRANSIENT

A. When the pressure transmitter for the Reactor Coolant Loop Flow fails, RC Flow in the affected loop goes to midscale (45×10^6 lbm/hr.). Therefore, total flow drops to approximately 81% of 100% demand, and one RPS Channel trips on "power to flow" but two RPS Channels are required for a Reactor Trip.

NOTE: The $T_{average}$ Auto/Manual Switch will transfer automatically to the loop with greatest flow; assuming one loop flow is greater than 95% and the other loop flow is less than 95%

B. The decrease in total flow creates a new High Load Limit for ICS Control. The ULD will runback to approximately 85% at 20% per minute, based on the RC flow change.

C. ΔT_c Ratio and BTU Limits are also affected with the immediate change in Loop "A" RC Flow.

ΔT_c Ratio Circuit

The ΔT_c circuitry will detect an inbalance between Loop RC Flows and ratio loop feedwater demand flows accordingly. Therefore, feedwater flow demand is decreased to "A" Loop and increased to "B" Loop.

BTU Limits

Assuming normal values for the BTU Limit variables (T_{hot}, $OTSG_{pressure}$, $T_{feedwater}$), a decrease in Loop "A" RC Flow will decrease BTU Limit Feedwater Demand Flow. Therefore, "A" Loop Feedwater Demand becomes limited due to the calculated BTU Limit of 4.6×10^6 lbm/hr.

(BTU LIMIT = ($T_{hot} + T_{fdw} + P_{OTSG}$ -200) % RC Flow)

D. Since Total Feedwater Flow is reduced immediately by BTU Limit, a feedwater mismatch develops between Feedwater Demand and Feedwater Flow. This mismatch results in a Feedwater Cross-Limit to Reactor Demand. The Feedwater Cross-Limit drives control rods into the core at 30 inches per minute reducing Reactor Power and $T_{average}$.

E. With ICS response by the ΔT_c Control Circuit and a plant runback in progress, feedwater demand is decreased and reratioed, clearing the Feedwater Cross-Limit. The decrease in feedwater demand to "A" Loop gradually decreases below the BTU Limit for "A" Loop and the BTU Limit condition is cleared. However, the re-ratio of feedwater demand to "B"

Loop overcomes the demand decrease such that "B" OTSG will become Hi Level Limited. Approximately 39% of feedwater demand is directed to "A" Loop and 61% to "B" Loop.

F. The decrease in feedwater flow to "A" OTSG and the increase in feedwater flow to "B" OTSG results in a ΔT_c error. However, the ΔT_c Control Circuit is limited for approximately 200 seconds to strictly RC Flow Differential Response. Once this time delay expires, the runback has been completed and feedwater flow is fine tuned on $T_{cold.}$ Ultimately, the ΔT_c Control balances the feedwater flow and the plant stabilizes at a reduced power level of approximately 78%.

G While the Feedwater Cross-Limit was in affect the Unit was in Track. The reduction in total feedwater flow resulted in a Turbine Header Pressure decrease. This decrease closed down on the Governor Valves and generated megawatts decreased slightly below the High Load Limit, initially calculated due to the reduction in Total RC Flow.

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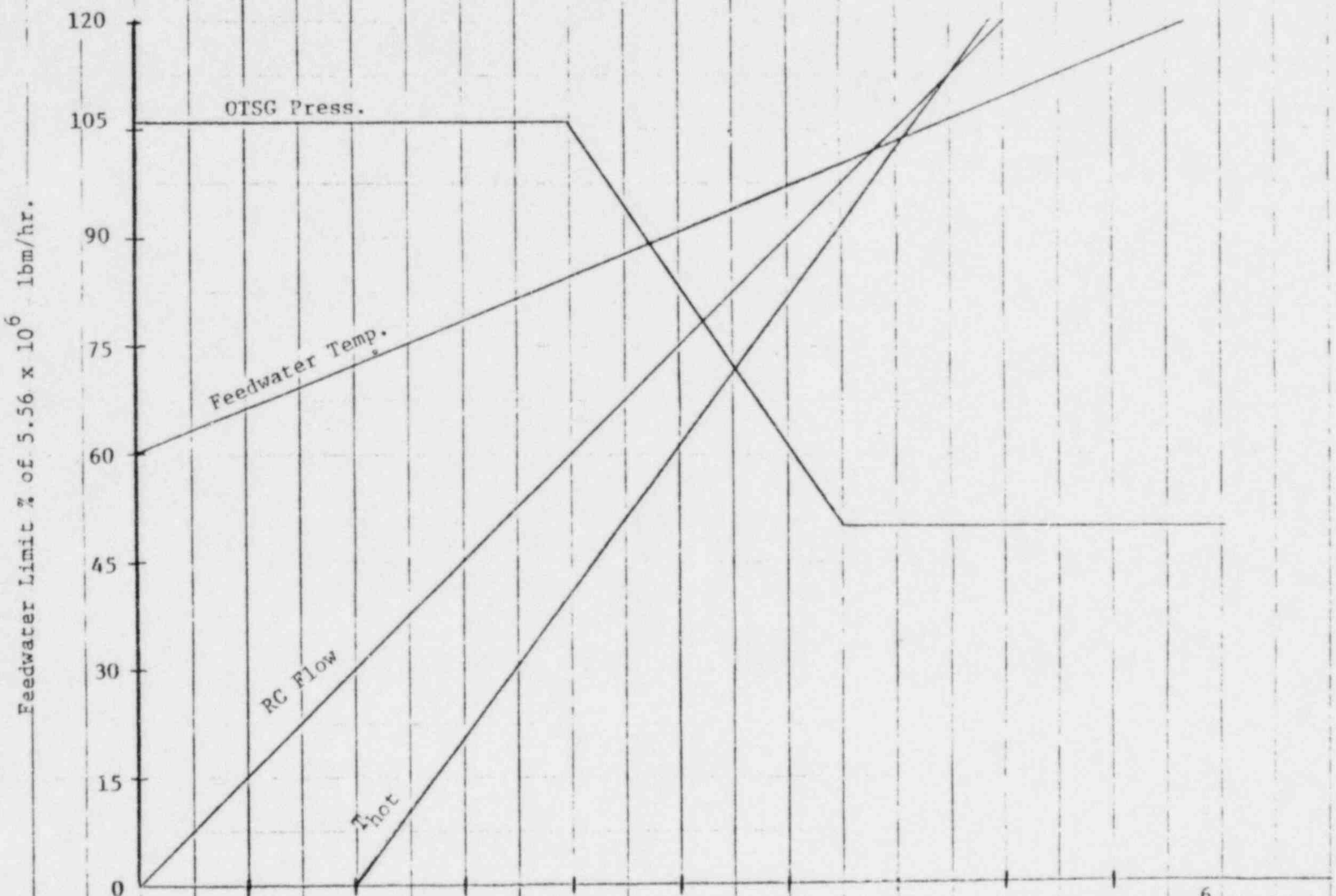
BTU LIMITS

FIGURE XII-1



RC Flow	0	10	20	30	40	50	60	70	80	90	x 10^6 lbm/hr.
T_{hot}	565	570	575	580	585	590	595	600	605	610	
OTSG Press.	800	850	900	950	1000	1050	1100	1150	1200	1250	
FWD. Temp.	100	150	200	250	300	350	400	450	500	550	

RC FLOW FAILURE

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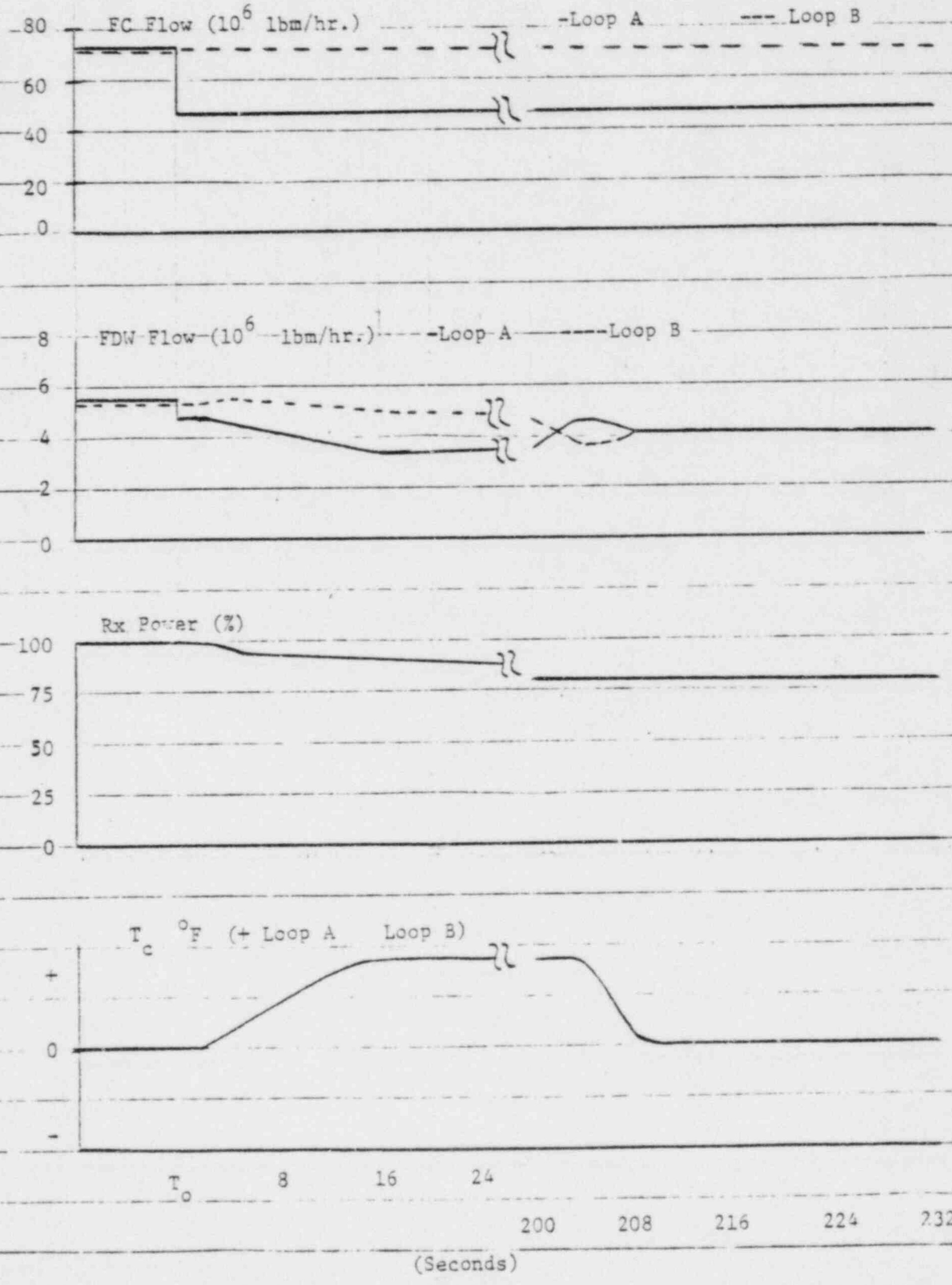
REVISION NO.:
DATE: 8-12-8

FIGURE XII -2



Main Annunciator Panel H
1-7

ALARM

Steam Gen. 1B Steam Line Rupture Detection System Actuated

SETPOINTS

≤600 psig pressure switches - PS-604, PS-605, PS-606 and PS-607

PS-604 and PS-606 are on steam line C
PS-605 and PS-607 are on steam line D } See Logic Diagram on Back

CAUSES

Steam line rupture.
Steam line rupture "Defeat" switches (normal and backup) left in the "Auto" position during plant cooldown.

AUTOMATIC ACTION

Automatic closure of startup, main and emergency feedwater valves, FW-V16B, FW-V5B, FW-V17B and EF-V30B if pressure switches PS-604 and PS-605 and/or PS-606 and PS-607 are actuated.
Automatic closure of FW-V92B is PS-606 and PS-607 are actuated.

OBSERVATIONS (CONTROL ROOM)

Console CC - One or more steam line rupture actuation lights.
FW-V16B, FW-V17B and EF-V30B indication that valves are closed.
Turbine throttle pressure recorder (PR-10) indicates ≤600 psig on one or more lights.
Low throttle pressure alarm.

MANUAL ACTION REQUIRED

Verify turbine and reactor trip and feed isolated to affected steam generator.
Initiate HPI if pressurizer level goes below 85" due to cooldown contraction from the steam break.
Refer to EP 1202-24, Steam Supply System Rupture (abn.)
If actuation occurs during normal cooldown reduce demand to FW-V16B and FW-V17B using the H/A Station, defeat the Steam Line Break Detection System and reopen FW-V92B. Reestablish normal feedwater flow.

CONTROLLED COPY
TRAINING DEPT.

Main Annunciator Panel H
1-6

ALARM

Steam Generator 1A Steam Line Rupture Detection System Actuated.

SETPOINTS

≤600 psig pressures switches PS-600, PS-601, PS-602 and PS-603

PS-600 and PS-602 are on steam line A
PS-601 and PS-603 are on steam line B } See Logic Diagram on Back

CAUSES

Steam line rupture
Steam line rupture "Defeat" switches (normal and backup) left in the "Auto" position during plant cooldown.

AUTOMATIC ACTION

Automatic closure of startup, main and emegency feedwater valves FW-V16A, FW-V5A, FW-V17A and EF-V30A, if pressure switches PS-600 and PS-601 and/or PS-602 and PS-603 are actuated.

Automatic closure of FW-V92A if PS-600 and PS-601 are actuated.

OBSERVATIONS (CONTROL ROOM)

Console CC - One or more steam line rupture actuation lights.
FW-V16A, FW-V17A and EF-V30A indication that valves are closed.
Turbine throttle pressure recorder (PR-10) indicates ≤600 psig on one or more inputs.
Low throttle pressure alarm.

MANUAL ACTION REQUIRED

Verify turbine and reactor trip and feed isolated to affected steam generator.
Initiate HPI if pressurizer level goes below 85" due to cooldown contraction from the steam break.
Refer to EP 1202-24, Steam Supply System Rupture (abn.).
If actuation occurs during normal cooldown, reduce demand to FW-V16A and FW-V17A using the H/A station. Defeat the steam line break actuation system and reopen FW-V92A. Reestablish normal feedwater flow.

CONTROLLED COPY
TRAINING DEPT.

ERT ASSOCIATES, INC. GINEERS AND CONSULTANTS READING, PA.	CLIENT PROJECT RM-13 (C)	FILING CODE W.O. PAGE OF

SYSTEM EMERGENCY FEEDWATER PUMPS AUTO-START

CALCULATION FOR SKETCH EFW-3

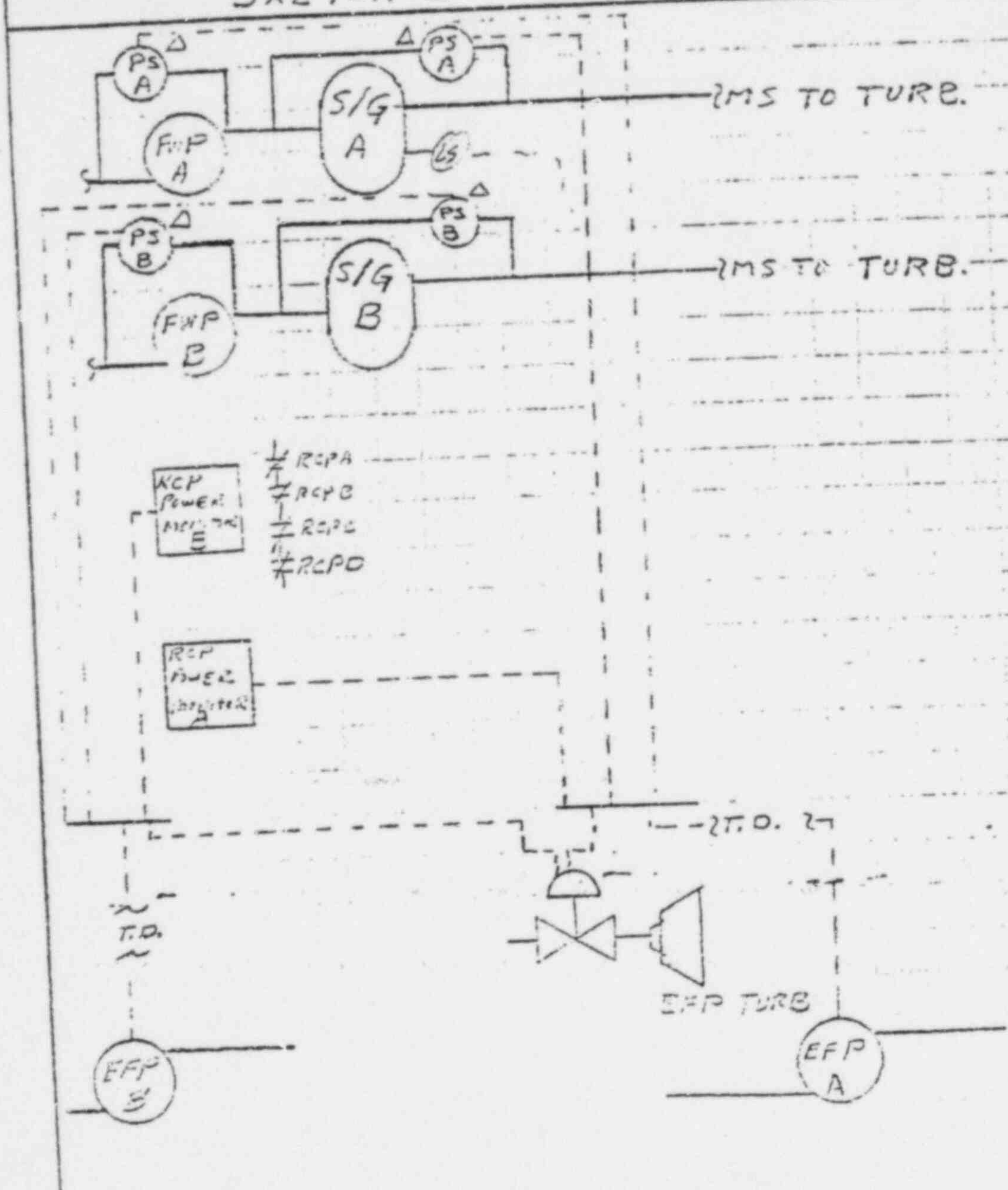
ORIGINATOR

DATE

REVIEWER

DATE

RESULTS



GPU Service **NUCLEAR SAFETY RELATED**

ENGINEERING CHANGE MEMO

TMI UNIT NO. 1

DESIGN ORGANIZATION
X GAI
GPUSC
OTHER

SECM NO. 076 REV. 2
DISCIPLINE I&C
SUBJECT: EMERG. FEEDWATER AUTO. - START
TASK NO. RM-13e

CHANGE ADDED PAGE 1 SH. 2, P.R. 86117, P.R. 86723, SKETCH CRG-0, AND DWGS. E-210-108 ID-0, S-212-009 RF-99A. REVISED PAGES 5 (SHEETS 1, 2, & 3) AND 6, PURCHASE REQUISTIONS 86196, 86197, 86198, AND 86199, AND DRAWINGS SS-202-012 SH. RV-1 IA-1, SS-202-012 SH. RV-2 IA-1, SS-208-203 IC-1, SS-208-205 IB-1, SS-209-101 IA-1, SS-209-108 IB-1, SS-~~209-755~~ IC-2, SS-209-756 IC-2, SS-209-860 IA-1, C-302-011 IA-1, E-308-604 IA-1, E-210-006 IE-2, E-210-009 IC-2, E-210-108 ID-0, E-210-112 ID-2, E-210-113 ID-2, E-215-196 IL-1, MERCURY CO. PC-6691-101 IC-1, PC-6691-102 ID-1, PC-6691-302 ID-2, PC-6691-401 ID-2, B&W 21-00-418 IA-1, AND ADDED P.R. 86740. PAGES 2 OF 6 AND 4 OF 6 NOT AFFECTED BY REVISION. CONTINUED ON SHEET 2 OF 4 (SEE ITEM 1)

REASON FOR CHANGE
THIS SECM HAS BEEN REVISED PER GPU COMMENTS IN LETTER TMI-1/E714 AND GAI COMMENTS IN LETTER TMI-ICS/3287 - CONTINUED ON SHEET 3 OF 4 (SEE ITEM 2).

REMARKS
ALL EQUIPMENT AND APPURTENANCES ADDED PER THIS MODIFICATION ARE NUCLEAR SAFETY RELATED. ALL DWGS. ARE NSR EXCEPT SS-209-101 IA-1, SS-209-108 ~~IB-1~~, SS-209-860 IA-1, E-210-108 IB, E-210-112 ID-2, E-210-113 ID-2, E-215-196 IL-1, RF99A, AND B&W 21-00-418 IA-1 WHICH ARE IMPORTANT TO SAFETY.

C.R. Goodwin 6/24/80 C.C. Strumpfler
ORIG. ENGINEER DATE SUPERVISOR

PORC APPROVAL 9/19
SRD REQUIRED 9/18
9/03/80

FOR INSTALLATION

REVISE SPEC. NO REVISE DWG. YES

SAFETY CLASSIFICATION
- [x] NUCLEAR SAFETY RELATED
- [ ] IMPORTANT TO SAFETY
- [ ] NOT IMPORTANT TO SAFETY

QC CLASSIFICATION
- [x] QC YES
- [ ] QC NO

G.W. Grammes for TAP 6/3/80
GAI QA APPROVAL (AS REQUIRED)

T. E. Manning 8/28/80
GPUSC QA APPROVAL (AS REQUIRED)

REV. ______ APPROVED FOR ISSUE AND CONSTRUCTION:

GPUSC CONST. SITE MANAGER/DATE

ACKNOWLEDGEMENT
MCM
James Kondias
DDCC/DATE

ITEM 1: "CHANGE" PARAGRAPH CONTINUED FROM SHEET 1 OF 4

TRANSMITTED WITH THIS ARE S-ECM-076, REV. 0, 1 & 2 PACKAGES. THE DETAILS ON EACH REVISION FOLLOWS:

- S-ECM-076, REV. 0 - IT WAS RETURNED TO GAI IN ORDER TO BE REWORKED PER GPUSC LTR.#TMI-1/E714, FROM D.G. SLEAR TO R.M. ROGERS, DATED APRIL 23, 1980 (A COPY OF THIS LETTER WITH ATTACHMENTS IS ENCLOSED AS ENCLOSURE 1 W/SECM-076, REV.2) AT THE TIME, THE ECM PACKAGE HAD ALL SIGNATURES EXCEPT THAT OF THE (GPUSC) QA GROUP ON THE COVER SHEET. AT THIS TIME, THE QA GROUP HAS SIGNED THE COVER SHEET.

- S-ECM-076, REV. 1 - THE COMMENTS ADDRESSED IN REV. 0 (PER ABOVE MENTIONED GPUSC LETTER # TMI-1/E714) WERE ALSO APPLICABLE TO REV. 1. AS A RESULT, REV. 1 WAS ALSO RETURNED TO GAI. AT THIS TIME GPUSC QA HAS SIGNED THE ECM COVER SHEET.

- S-ECM-076, REV. 2 - THE COVER SHEET IS STAMPPED "NUCLEAR SAFETY RELATED". THE "NUCLEAR SAFETY/ENVIRONMENTAL IMPACT EVALUATION" AND "FIRE HAZARDS ANALYSIS" FORMS ARE INCLUDED & SIGNED-OFF AS A PART OF REV. 0 PACKAGE. THESE FORMS ARE ALSO APPLICABLE TO REV. 2. THE GPUSC COMMENTS PER LETTER TMI-1/E714 ARE RESOLVED BY GAI PER LETTER # GAI/TMI-1CS/3287, FROM R.M. ROGERS TO D.G. SLEAR, DATED MAY 23, 1980. A COPY OF THIS LETTER IS ENCLOSED AS ENCLOSURE 2 WITH SECM-076, REV. 2. THE GAI RESPONSE PER LETTER GAI/TMI-1CS/3287 IS ACCEPTABLE WITH THE FOLLOWING EXCEPTIONS
  1. ITEM 7 - DID NOT VERIFY WHETHER OR NOT THE CIRCUITS WILL MEET THE SEPARATION CRITERIA PER IEEE-384-1974 FOR ASSOCIATED CIRCUITS, MAINLY THE ALARM CIRCUITS.
  2. ITEM 10 - INDICATES THAT "NOTE 1 (ON DWG. E-308-604, REV. IA-1) HAS BEEN REVISED TO INDICATE SEISMIC CLASS I (FOR TUBING)". HOWEVER THE DWG. E-308-604 THAT CAME WITH S-ECM-076, REV. 2 DID NOT INCORPORATE SUCH NOTE. THIS PROBLEM HAS BEEN RESOLVED PER RESOLUTION (1) IN GAI LETTER # GAI/TMI-1CS/3550, FROM R.M. ROGERS TO D.G. SLEAR, DATED AUG. 15, 1980. A COPY OF THIS LETTER IS ENCLOSED AS ENCLOSURE 3 WITH S-ECM-076, REV-2.

INSTALLATION OF COMPONENTS, LOGIC & APPURTENANCES PERTAINING TO M.S./FW ΔP CONDITION THAT WAS EXPECTED TO INITIATE EFW SYSTEM ON FW LINE BREAK HAS BEEN PLACED ON HOLD. THIS HOLD IS INDICATED ON DWG. SS-209-755, REV. IC-2 & SS-209-756, REV. IC-2. IN

ADDITION, IT IS ALSO APPLICABLE TO ALL OTHER DRAWINGS PERTAINING TO MS/FW ΔP SWITCHES, THEIR INSTALLATION, ROUTING OF INSTRUMENT TUBING, CABLE ROUTING, TERMINATIONS, ETC.

ITEM 2: "REASON FOR CHANGE" PARAGRAPH CONTINUED FROM SHEET 1 OF 4

THE REASON FOR PLACING HOLD ON MS/FW ΔP LOGIC FROM INITIATING EFW SYSTEM IS POOR RELIABILITY. WITH THE PRESENT LOGIC, MALFUNCTION OF ANY ONE MS/FW ΔP SWITCH CAN INITIATE THE ASSOCIATED TRAIN OF PUMPS PLUS IT WOULD ESTABLISH A ~~HIGHER~~ 30" LEVEL SETPOINT (~~30"~~ DURING FORCED RC CIRCULATION) FOR EFW CONTROL VALVES (EF-V30A & B). UNDER CERTAIN PLANT CONDITIONS, LOW LOADS IN PARTICULAR, THE STEAM GERATOR LEVEL IS EXPECTED TO BE BELOW 30". THUS THE MALFUNCTION OF A MS/FW ΔP SWITCH WOULD RESULT IN EFW INJECTION INTO THE STEAM GENERATORS. THE EFW INJECTION NOZZLES ARE DESIGNED FOR ≈10 CYCLES (INJECTIONS). THEREFORE THE EXISTING LOGIC IS NOT ACCEPTABLE. A MORE RELIABLE LOGIC WILL BE RECOMMENDED AT A LATER DATE WHICH WILL BE IMPLEMENTED DURING NEXT REFUELING OUTAGE.

GPU Service		SECM -076 Rev. 2 TASK RM-13(e)	
TITLE	EMERGENCY FEEDWATER AUTO.-START	PAGE 1 OF 6 SHEET 4 OF 4	
REV	SUMMARY OF CHANGE	APPROVAL	DATE
2	THE IMPACT EVALUATION AND FHA HAVE NOT BEEN AFFECTED BY THIS REVISION. HOWEVER, THE SAFETY EVALUATION, LIST OF ATTACHMENTS, AND THE LIST OF REFERENCES HAVE BEEN REVISED AND ATTACHED. ENCLOSURE 1, 2 & 3 ARE ADDED BY GPUSC - S.R.D. 9/03/80	C. C. Stumph	6/26/80

GPU Service

SAFETY EVALUATION
TMI-1 NUCLEAR STATION

Safety Evaluation

THE IMPLEMENTATION OF SAFETY EVALUATION SECT. 2.1.1.7.6 OF THE NRC RESTART REQUIREMENTS PRECLUDES THE PROBABILITIES OR CONSEQUENCES OF ACCIDENTS OR MALFUNCTION OF EQUIPMENT IMPORTANT TO SAFETY. ALL EQUIPMENT AND APPURTENANCES ADDED PER THIS MODIFICATION MEET ENVIRONMENTAL, SEISMIC, SEPARATION, AND REDUNDANCY REQUIREMENTS FOR A SAFETY GRADE SYSTEM. NOTE THAT THE DIFFERENTIAL PRESSURE SWITCHES ACROSS THE MAIN FW PUMPS ARE LOCATED IN THE TURBINE BUILDING WHICH IS A NON-SEISMIC BUILDING. HOWEVER THESE SWITCHES ARE TIED INTO SAFETY GRADE CIRCUITS THROUGH ISOLATION DEVICES. THE ISOLATION DEVICES ARE CLARK RELAYS WHICH ARE IDENTICAL TO THOSE USED IN THE ESFAS CABINETS.

Included ECM(s)	NSR	Non-NSR	Verified	Verified Per ~~EP-009~~ DCP 2:05
SECM-076-2	YES		yes	C.C.S. 6/23/80

	Yes	No	TR #
FSAR deviation must be documented	☐	☒	
Station Technical Specification amendment is required	☐	☒	
Quality Classification List amendment is required	☐	☒	
Other	☐	☐	

C.R. Goodwin
RESPONSIBLE ENGINEER

NRC APPROVAL HAS BEEN OBTAINED

U. S. NUCLEAR REGULATORY COMMISSION

REACTOR OPERATOR LICENSE EXAMINATION

Facility: TMI-1

Reactor Type: PWR-B&W

Date Administered: 4/ /81

Examiner: BWilson/BBoger/JMcMillen

Applicant: MASTER- A

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 parentheses after the question.

Category Value	% of Total	Applicant's Score	% of Cat. Value	Category
13	13	____	____	A. Principles of Reactor Operation
12	12	____	____	B. Features of Facility Design
13	13	____	____	C. General Operating Characteristics
13	13	____	____	D. Instruments and Controls
13	13	____	____	E. Safety and Emergency Systems
14	14	____	____	F. Standard and Emergency Operating Procedures
10	10	____	____	G. Radiation Control and Safety
12	12	____	____	H. Principles of Heat Transfer and Fluid Mechanics
100		____		

Final Grade ________%

A. Principles of Reactor Operation (13)

A.1 The average lifetime of prompt neutrons is in the order of about 10^{-5} seconds while that of delayed neutrons is about 10 seconds.

a. Why, therefore, is the total average neutron lifetime about 0.1 second? (1)

b. Since reactor control is dependent upon the average neutron lifetime, how would you expect the reactor to behave differently if the neutron lifetime were only due to prompt or only due to delayed neutrons? Answer for both cases. (1)

A.2 Startup of a reactor with an artificial source of neutrons is not required, however it is quicker and safer to rely on an artificial source.

a. Give two examples (reactions) of intrinsic or natural neutron sources. (1)

b. Give two advantages that an artificial neutron source will provide during a reactor startup. (1)

A.3 a. List four examples of <u>fissionable</u> materials used in a nuclear reactor. (1)

b. Of these four fissionable materials, identify which ones are fissile and which are fertile. (1)

A.4 For the following two statements answer <u>one</u> of the choices, <u>a</u> through <u>d</u>. (1)

a. Equilibrium xenon at full power is about twice as much as equilibrium xenon at 50% power.

b. It takes approximately 40 hours to reach a new equilibrium xenon concentration from a previous equilibrium condition for most power level changes (does not include trip).

c. a and b are both true.

d. Neither are true.

A.5 Samarium is classified as a neutron poison as is xenon. List two reasons why samarium is not regarded as much of a problem as is xenon. (1)

A.6 a. Define reactivity. (.5)

b. How much reactivity must be inserted into the core to go prompt critical? (.5)

c. Does the amount of reactivity required to go prompt critical remain the same over the life of the core? Explain. (1)

A.7 Is the point of adding "sensible" (nuclear) heat the same for any given startup during a particular operating cycle? Explain why or why not. (2)

A.8 Explain what is meant by a "1% shutdown margin". (1)

B. Features of Facility Design (12)

B.1 The attached Figure B-1 is a simplified flow diagram of the HPI portion of the Makeup and Purification System. Show on this diagram by different colored pencil, pen (not red) dotted line, etc. the modifications, according to Lesson Plan RM-14 that were made to this system. An explanation of the reasons for the modifications is not required. (2)

B.2 In the use of pressurizer heaters and control rod drive system, frequent use is made of the acronyn "SCR". What does "SCR" mean and what general purpose does it serve? (1)

B.3 a. Why must the nuclear services closed cooling water system surge tank be pressurized during power operations? (1)

b. How and why is the Kirk Key interlock used on NS-P-1B? (1)

B.4 Considering MSIV design, explain the difference in MSIV closure times in response to a main steam line break (1) upstream and (2) downstream of the valve. ~~Assume actuation of the steam line rupture detection system~~. (2)

B.5 With respect to RCP operations:

a. What is the purpose of the #1 seal bypass line? Include how opening this line affects the #1 seal. (1)

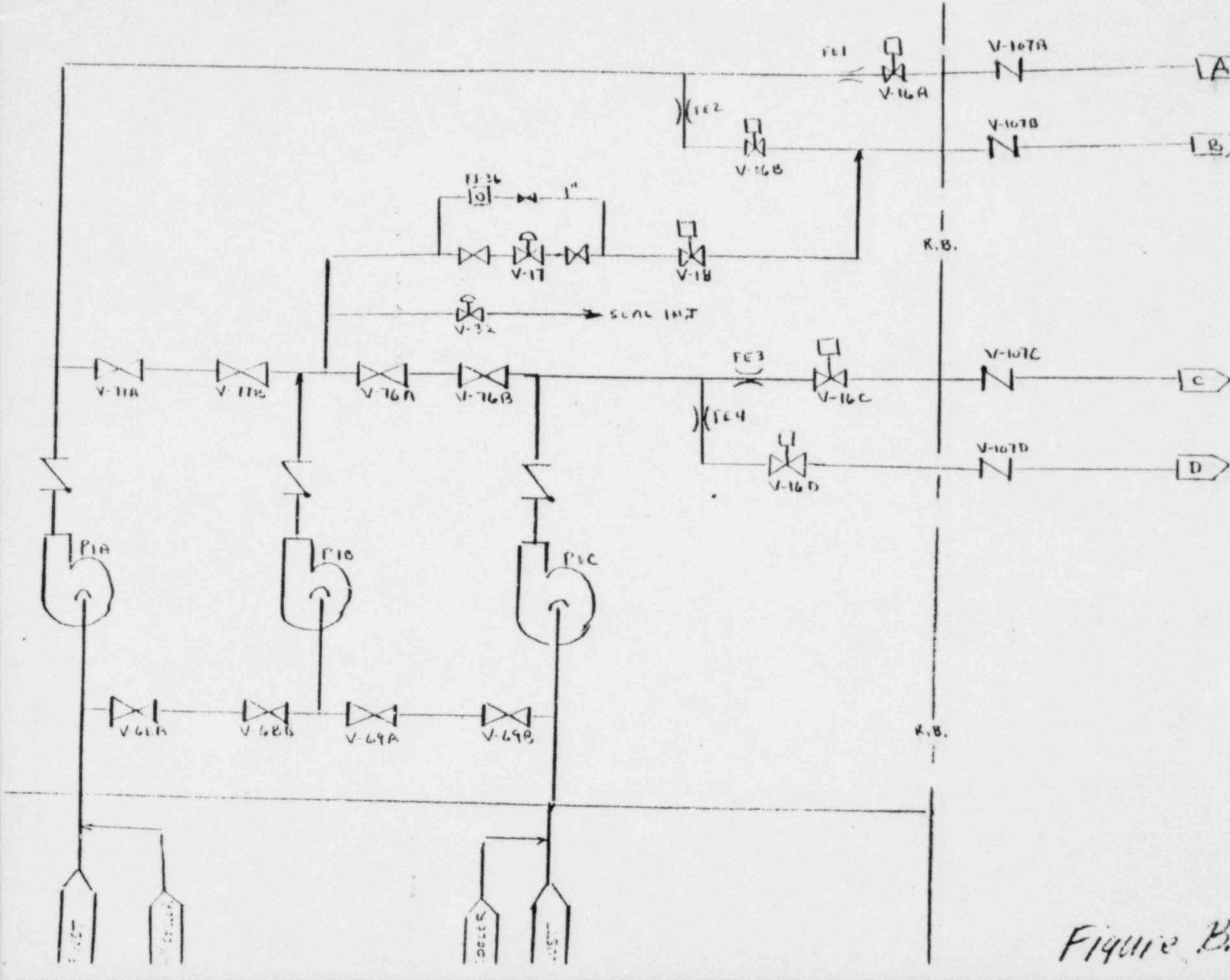
b. When may the #1 seal bypass valve (Mu-V38) be opened? (1)

c. When must a RCP be tripped due to high vibration? (Assume 4 pump operation). (.5)

d. What is the most probable cause of a standpipe high level? (.5)

B.6 a. How does the response of the NSRW system differ between a loss of offsite power with and without a LOCA? (1)

b. How does the operator know when a NSRW heat exchanger must be backwashed? (1)



FE1
V-16A
V-107A
A
FE2
V-16B
V-107B
B
FI 36
1"
V-17
V-18
K.B.
V-32
SEAL INJ
V-11A
V-11B
V-76A
V-76B
FE3
V-16C
V-107C
C
FE4
V-16D
V-107D
D
P1A
P1B
P1C
V-68A
V-68B
V-69A
V-69B
K.B.

Figure B.1

C. General Operating Characteristics(13)

C.1 Due to the design characteristics of the OTSG's, secondary side fluctuations can produce immediate reactions in the primary system. For instance, in June of 1973, Oconee Unit 1 tripped on high flux due to feedwater flow oscillations. Explain how a feedwater flow oscillation could cause a reactor trip on high flux. (2.0)

C.2 Control of pH is important to minimize corrosion of primary and secondary components.

a. Explain how the addition of hydrazine in the secondary system can affect pH. (1)

b. Primary pH can vary from 4.6 to 8.5. Describe the competing effects that determine primary pH and cause it to vary in this manner. (2)

C.3 Attached Figure C-1 is found in the Technical Specifications,

a. Define Axial Power Imbalance. (1)

b. Why does a positive imbalance pose more restrictive limits? (1)

c. If you are operating at full power in full ICS automatic control and an automatic runback to 60% occurs, how <u>and</u> why will imbalance change? (1)

C.4 a. Compare natural circulation obtained with a 50% OTSG level using (1) emergency feedwater pumps and (2) main feedwater pumps. Discuss heat removal capabilities and RCS flow rates. (1)

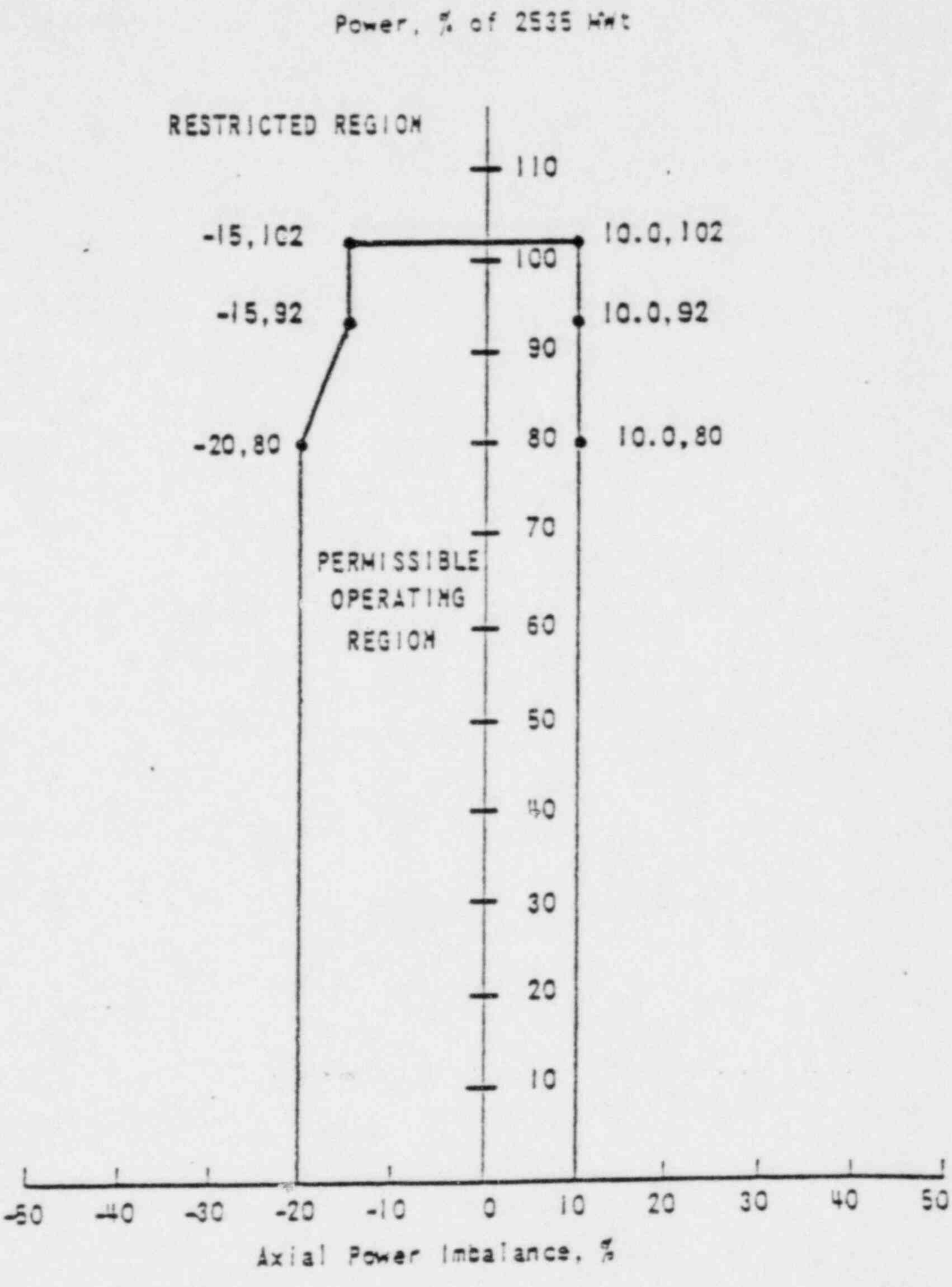
b. Compare natural circulation obtained with manual operation of the (1) turbine bypass valves and (2) atmospheric dump valves. Discuss heat removal capabilities and RCS flow rates. (1)

C.5 While operating the unit at full power with the Diamond Panel in manual, a group 2 control rod drops.

a. List five indications that would identify the presence of this dropped rod. (1)

b. Compare the ~~automatic~~ response to this dropped rod with a dropped rod with the Diamond Panel in automatic. Include the response of reactor power, feedwater, and turbine. (2)

✓



POWER IMBALANCE ENVELOPE FOR OPERATION FROM 0 EFPD TO EOC

Figure 3.5-2E

Figure C.1

D. Instruments and Controls (13)

D.1 There are four calibrating integrals used in the ICS.

a. Identify the location, i.e. portion of the ICS that each is located. (1)

b. What is the general function of a calibrating integral? (1)

c. When is a calibrating integral blocked and how long will it remain this way? (1)

D.2 Source range channels NI-1 and 2 are designed to provide neutron flux indication over a counting range of 0.1 to 10^6 counts per second.

a. What are the design features of the source range channels that are intended to insure you are reading a neutron flux signal rather than any other induced signal? Your answer should include the principle by which the neutrons are detected in the chamber. (2)

b. Describe the interlock and logic provided by the source range instruments. (1)

D.3 The latest heat balance for the plant indicates 98% power while the power range channels indicate 100% power. Is this in the conservative or non-conservative direction? Explain. (1)

D.4 In the four RPS cabinets are contained Auxiliary Power Supply Modules (see Figure D-4). Two of the cabinets have one of these modules in each while the other two cabinets have tw Auxiliary Power Supply Modules in each. Explain why (2)

D.5 Sensors to start or initiate emergency, safeguard, or control system action come from a variety of different sources. List the sensors that will initiate an automatic action for the following abnormal situations (for example, high flux as sensed by the linear power range detectors causes the RPS to trip the control rods).

a. Auto initiation of EFW due to Loss of main feedwater ~~line break~~. (.5)

b. Main steam line isolation. (.5)

c. Main transformer fire deluge. (.5)

d. ICS tracking signal. (.5)

D.6 a. Describe how RCS flow is measured. (1)

b. How would a failure, low signal, from an RCS flow transmitter affect the ICS? (1)

Auxiliary Power Supply Module

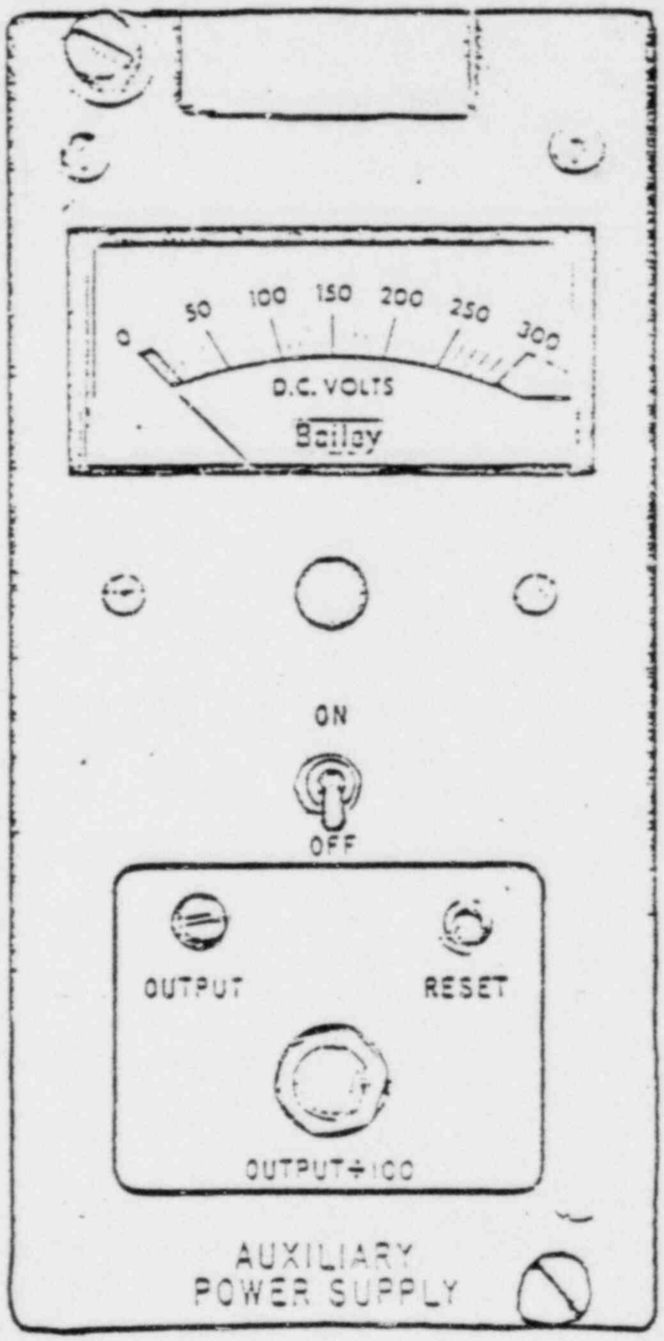


FIG D-4

Babcock & Wilcox

E. Safety and Emergency Systems (13)

E.1 A. "trip confirm" signal on the ICS Diamond panel indicates which of the following breakers are open? (1)

- a. 'A' and 'B'.
- b. 'A', 'B', 'C', 'D', 'E', & 'F'.
- c. C, D, E & F.
- d. A, C, D.
- e. B, E, F.
- f. Several combinations of the above 5. List them.

E.2 List all of the automatic signals which are designed to isolate letdown to prevent the inadvertent transfer of radioactive liquids. For each valve that isolates, include the actuation signal(s) (or instrument) and how (if at all) the operator could bypass each signal after actuation. (2)

E.3 With respect to a major steam line break inside the reactor building.

- a. Identify the main and backup signals that could cause the reactor to trip. Include setpoints and coincidences. (1.5)
- b. One of the concerns with this incident is the restart of the reactor. Explain how a restart could occur and how automatic actions would prevent a restart. (1.5)

E.4 Describe the two methods which are used to detect a leak in the RB emergency cooling system. (2)

E.5 With respect to the emergency diesel generators:

- a. Why is scavenging air important? (.75)
- b. What is meant by isochronous? (.75)
- c. List the power supplies for the two EDG fuel oil pumps. (.75)
- d. List the two conditions that will cause a start failure alarm. (.75)

E.6 a. How does the fluid block system operate to perform its intended function? (1)

- b. Describe the system design that prevents non-emergency actuation. (1)

F. Standard and Emergency Operating Procedures (14)

F.1 You are the responsible operator at the controls when the unit experiences a steam generator tube rupture.

a. Under what two conditions must the operator immediately reduce load? (1)

b. Assume the leak is large enough to cause an automatic reactor trip. List three methods to determine the affected OTSG. (1)

c. When and why may you steam the affected OTSG? (1)

F.2 a. List the six logs and/or records that must be reviewed by the oncoming CRO. (1.8)

b. When may the responsible (on watch) CRO enter the shift supervisors office for consultation? (1.2)

F.3 a. During the TMI-2 accident, explain why pressurizer level increased while RCS pressure decreased. (1.5)

b. TMI-1 procedures state that "...pressurizer level may not be a reliable indication of inventory conditions in the RCS". Give four situations when you would not consider pressurizer level indication to be reliable. (1)

c. Without pressurizer level, how could you tell if the core is covered? (.5)

F.4 The following precautions and/or limitations are found in the TMI-1 procedures. Fill in the blanks.

a. RC drain tank rupture disc is rated for ___ psig and the relief valve is set for ___ psig. (.5)

b. The maximum allowable heatup and cooldown rate for the pressurizer is ___ per hour. (.5)

c. Prior to submerging the main feedwater nozzles, the RCS must be borated to ______ percent $\Delta K/K$ subcritical at 532°F. (.5)

d. Axial power shaping rods should not be used for any purpose except ____. (.5)

F.5 a. Why are the RCP's tripped during a small break LOCA? (1)

b. Why is feedwater unnecessary during a large break LOCA? (1)

c. Under what conditions may HPI be throttled after ESAS initiation during a LOCA? (1)

G. Rod Control & Safety (10)

G.1 Define the following terms (do not use formulas):

a. Radiation (0.5)
b. Beta particle (0.5)
c. Half life (0.5)

G.2 What is the difference between RAD and REM? (0.5)

G.3 Briefly explain what you would expect to find in 10 CFR Part 19. (1.0)

G.4 CO-60 is a radioactive material that is often encountered at a reactor. Give the nuclear process for the formation of this isotope and why it is considered to be a hazard. (2.0)

G.5 The plastic shield that surrounds the detector for a neutron survey instrument: Which of the following statements is correct? (1.0)

a. is to moderate neutrons so the detector can count them.
b. is to absorb the thermal neutrons so that only the fast neutrons will be counted.
c. is used for counting geometry.
d. is so the detector will float if dropped in water.

G.6 Tritium is produced in large quantities in a PWR. What are two significant sources of this isotope? (1.0)

G.7 Fuel assemblies located in a reactor facility can be generally classified in three categories; a) new fuel, b) recently removed irradiated fuel elements, and c) irradiated fuel stored for some time. Which of these three classifications would present a contamination hazard? (1.0)

G.8 Portable radiation monitors record a dose rate of 45mr/hr at a distance 10 feet from radioactive point source. An operator is required to work on a valve located 3 feet from the source. How long can the operator work without exceeding his weekly exposure limit? (Show all work). (2.0)

H. Principles of Heat Transfer and Fluid Mechanics (12)

H.1 Heat added to a saturated vapor such as steam will increase its temperature whereas heat added to a saturated liquid will not change its temperature. Why? (1)

H.2 Quality and void fraction are both terms used to express the relative amounts of a liquid and a vapor in a closed space. What is the difference between quality and void fraction? (1)

H.3 Assume feedwater as a saturated liquid is flowing at 10^6 lbm/hr at 400°F. If water is an incompressible fluid then when this flow rate is converted to gal/min it should be the same for saturated feedwater flowing at 10^6 lbm/hr at 150°F. Use the conversion factor 7.48 gal/ft^3 and the steam tables to show that water is or is not a compressible fluid. Show all calculations. (2)

H.4 a. Why does nucleate boiling heat transfer remove more heat than non-boiling heat transfer? (1.5)

b. Why does film boiling remove less heat than nucleate boiling? (1.5)

H.5 a. What happens to NPSH requirements as pump speed increases? (.5)

b. What happens to the flow rate delivered by a pump as speed is doubled? (.5)

c. It is desired to increase the discharge head of a pump from 1200 psi to 1800 psi. How much does the speed of the pump have to increase? (.5)

d. In order to get the increased discharge head in (c) above, how much more power (Kw) is required by the pump? (.5)

H.6 Assume the RCS has just been filled. RCS temperature is 110°F, pressurizer level is 200", and a nitrogen bubble exists in the pressurizer.

a. Procedure 1103-2, "Fill and Vent of the RCS", states that a minimum of 30 psi on the wide range channel must be maintained to assure a full RCS. How can a pressure be used to assure a full RCS? State any assumptions that are necessary. (1.5)

b. After a RCP is started, pressurizer level drops. What is the most likely cause for level to drop? (.75)

c. After a RCP is started, pressurizer level rises. What is the most likely cause for level to rise? (.75)

EXAM A ANSWERS

A.1 a. Given β value of .007

$$\ell_{avg} = (1-\beta)\ell_p + \beta(\tau)$$
$$= (1-.993)10^{-5} + .007(10)$$
$$\simeq 0.1 \text{ sec}$$

b. Since $T \propto \frac{\ell}{\beta-\rho}$ for $\frac{10^{-5}}{.007-.002}$ = 200 X 10^{-5}

or $T \simeq$ .002 sec

For delayed neutrons

$$T \propto \frac{10}{.007-.002} = \frac{10}{.005} = 2000 \text{ sec.}$$

A.2 a. OARP Module 1 Rx theory Pg. 5

$$_2\alpha^4 + {_5B^{11}} \rightarrow {_7N^{14}} + {_0n^1}$$
$$\gamma + {_1H^2} \rightarrow {_1H^1} + {_0n^1}$$
$$_2\alpha^4 + {_8O^{18}} \rightarrow {_{10}Ne^{21}} + {_0n^1}$$

Spontaneous fission

b. Pg. 6

1. Increase background count rate so that source range instr. will indicate at least 2 Cps indicating operability of instru.

2. Provides a controlled approach to criticality - when reactivity is added the core will yield a much more noticeable count rate increase than it would be without the sources.

A.3 OARP pg. 8

a. Th-232, U-233, U-235, U-238, Pu-239, Pu-241, Pu-240
(Also others - Pg. 9)

b. Fissile U-233, U-235, Pu-239, Pu-241
Fertile Th-232, U-238, Pu-240

A.4 b.

A.5 a. σ_a much lower ~4.08 X 10^4 barns, Xe is 2.7 X 10^6 barns
Lower production rate.
Not ϕ dependent (only thru removal-varies from .6 to 1.0%)

A.6 a. Departure from criticality or $\rho = \frac{k-1}{k}$

b. $\rho = \beta$ or .007 (.0052EOL) or \#1

OARP c. No, β decreases due to Pu^{239} from ~ .0058 to .00523 for Cycle 5.

A.7 OARP pg. 90. Exact point of adding sensible heat production is variable since fission heat is added to the background heat production - decay heat. Because of this higher rates of heat production will raise the reactor power where sensible heat production begins.

A.8 The amount of reactivity that could make the reactor subcritical by 1% or a keff of $\simeq$ .99.

EXAM B ANSWERS

B.1 On Figure B.1

B.2 Power System Operation, R.H. Miller
pg. 131
Silicone Controlled Rectifier - fires on certain arc of A-C transmission to convert AC-to-DC.

B.3 a. To ensure minimum pressure in system & RB Fan is always > RB pressure during a LOCA. Ensures no out leakage from RB.

OARP Nuc. Ser.

b. It is used to prevent paralleling 1P & 1S 480 volt ES buses through NS-P-1B breaker. Must use key to allow removal - same key to permit insertion.

B.4 Upstream - almost instantaneous closure since this break causes low pressure upstream of valve & backpressure at turbine, MSIV will be forced down into its seat and seal - valve normally held out of flow path by upstream pressure, will fall by gravity or reverse Δp. Like Stop-Check

(Surry)

OARP Downstream - requires < two minutes to drive the valve shut ~~after~~ — does not shut MSIV's ~~actuation~~

B.5 a. Lowers the pressure in the #1 seal area, Lowers seal #1 ΔP offers lower head resistance to pump injection water, allows more injection flow to be diverted up shaft ~~through the seal and~~ past radial bearing for adequate cooling ~~This prevents binding and contact of seal faces~~.

OP1103-6 pg 4 step 9

b. If RCP seal radial bearing temperature and or #1 seal leak off temperatures are approaching alarm (225°F) and if #1 seal leakoff is <1 GPM and RCS pressure is between 100 and 1000 psig.

c. 20 mils - 4 pump ops } Bently
30 mils - single pump ops } Nevada sys
Motor Stand High Vibration 2 mils

d. Problem in #2 seal.

Blackout Proc. 1202-2 - Standby Pump Starts

B.6 a. NSRW pumps don't auto start unless there is a LOCA (ES) signal in which case they are block loaded.
ES — Two ES Selected pumps start - Standby (not sel. for ES) locked out

OARP Module 3

b. High outlet temperature (95°F) on NSCCW (or ICCW heat exchanger)

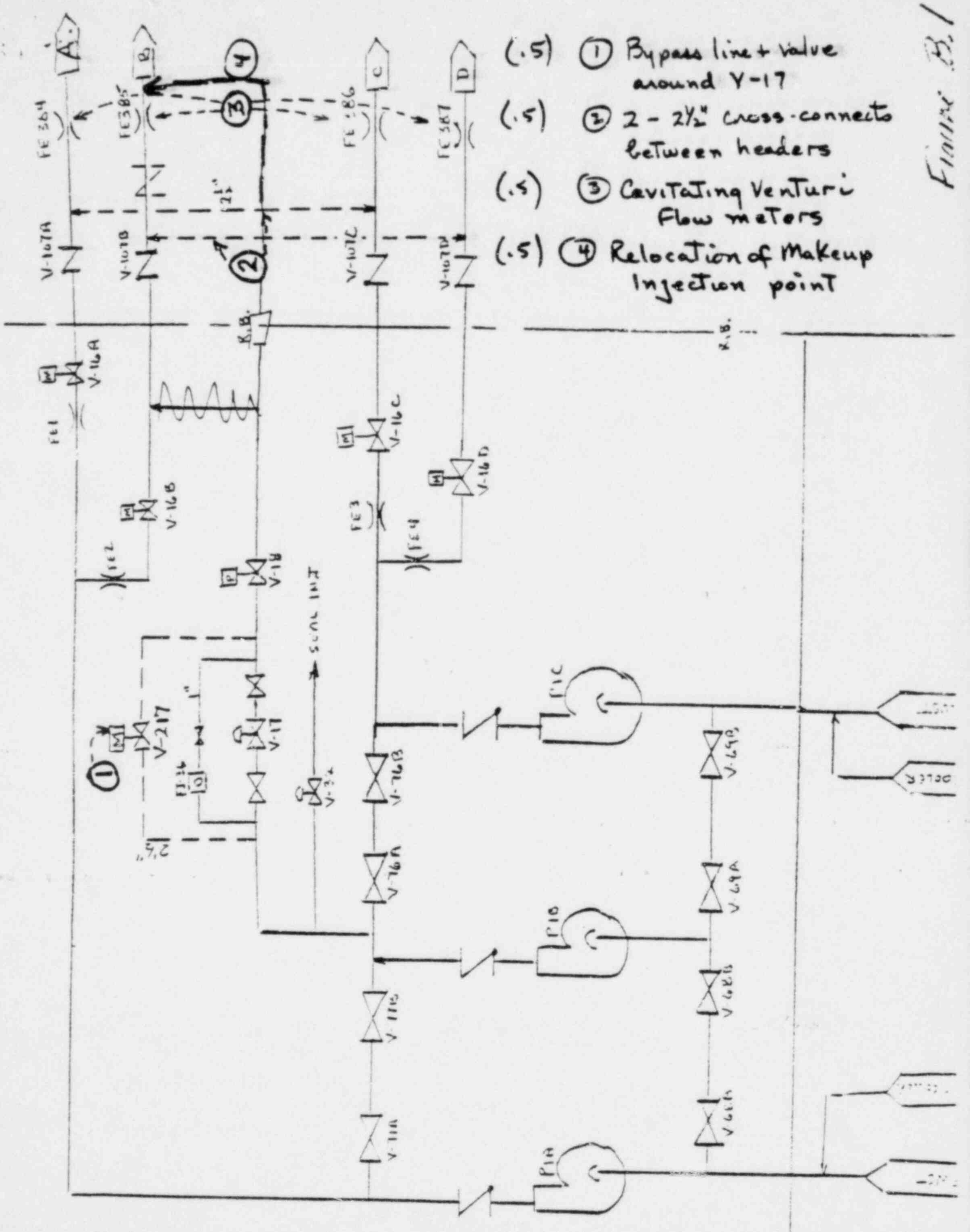
pg 15

?

River water side of NSCCW HX + ICCW HX requires backwashing @ 95°

(find reference
Fouling of NS - should not affect ICCW.

B.6 a. LOOP w/LOCA
2 ES Selected pumps start - (Standby not selected for ES locked out)
LOOP w/o LOCA
Standby pump starts



(.5) ① Bypass line + valve around V-17

(.5) ② 2 - 2½" cross-connects between headers

(.5) ③ Cavitating Venturi Flow meters

(.5) ④ Relocation of Makeup Injection point

Figure B.1

EXAM C ANSWERS

C.1 See attached Oconee trip report.

C.2 a. Chem Lecture pg. 4
Hydrazine, if added in excess, (beyond that needed for pH control) will react to form ammonia, which is alkaline and will cause pH to increase.

Disagree w/answer - will provide us with info.

b. (pg. 5,7) Boric acid and lithium hydroxide concentrations compete. BA conc. varies over core life for ρ control. BA causes pH to be lowered. LiOH is alkaline and causes pH to be increased. Decrease in BA over core life is dominant factor.

.2 - 2 ppm w/ Li Control
Lithiated demin bed

for good answers see Zawe + Boltz

C.3 a. T.S. 1.6.2
Power in top half minus power in bottom half of core expressed as % of rated power.

b. Upper half of core is closer to DNB - lower pressure and higher temp coolant, High relative power in the top decreases DNB margin.

c. Inbalance will become more negative due to insertion of control rods in the top half of the core which will decrease flux in the top half.

C.4 a. Natural circulation using EFW pumps is greater since the EFW admission point in the OTSG is higher and will cool the RCS higher in the OTSG and create a larger (higher) thermal driving head.

b. Natural circulation using either will be the same. If in manual the operator is controlling OTSG pressure which governs heat removal from OTSG. It doesn't matter whether TBV's or dump valves are used to determine the pressure.

Answers will vary depending on Bypass valve capability - condenser vacuum, etc.

C.5 a. Indic. Rod In Light
Abs. RPI @ 0%
Group In Limit Light
One or more 7" fault lights on PI panel
Group avg Indicator low
9" Asy. rod fault indicator light & alarm
(Also Nucl. Instr. - QPT, etc).

b. In automatic FW, Rx + Turbine would runback together ~ 30%/min to 55% power.) With rods in manual, unit will be in track, rather than using ICS runback, drop in Mwg will cause plant power reduction, rod ↓ Tavg ↓, Stm press ↓, Turbine ↓, Mwg ↓, FW will react as necessary to control RCS Tavg.

C.1

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

REACTOR TRIP #19

Time of Occurrence: June 2', 1973 (1500 hours)

Reactor Power Level: 75%

Ultimate Cause: High neutron flux trip

Sequence of Events:

The reactor was operating at 75% power level and the turbine-generator was producing 665 MWe. Flow oscillations were occurring in the condensate feedwater system. The alarm log indicated that abnormal fluctuations were occurring in various parts of the heater drain system. The first stage reheater drain pump recirculation and condenser dump valves were cycling about once a minute. (After this trip, the steam supply valves to the reheaters were adjusted). The "D" heater drain flow was also oscillating. The oscillations in feedwater flow to both steam generators extended all the way into the primary system as shown in Figure 1, a composite display of primary and secondary system parameters.

The sequence of events associated with this reactor trip is presented below: At approximately 1450 hours the "A" and "B" feedwater flowrates increased 20 to 30% of the initial value while the startup feedwater flowrate decreased a like amount. The increasing main feedwater flows are shown in Figures 2 and 3 and the decreasing startup flows are displayed in Figures 4 and 5. A logical explanation is that the main feedwater valves opened.

At approximately the same time, the alarm log indicated "ICS on Feedwater Limit" which is a demand for increased feedwater flowrate at a rate exceeding the 5% BTU limit for the system. Probable causes for the feedwater [illegible] signal to increase are discussed later.

REACTOR TRIP #19

Both feedwater pump speeds increased during the flow transient prior to the reactor trip and the two speeds are shown in Figure 6. Apparently there were two surges in feedwater flow demand - the first at about 1450, and the second about four minutes later.

For at least 30 minutes before the transient, the "A" loop feedwater flow and generator startup level had been slightly higher than the values for the "B" loop. When the feedwater flow increased, the "B" loop feedwater flow exceeded the "A" loop flow. The proportional part of the ΔTc controller probably biased the "B" flow higher than the "A" feedwater flow. This condition lasted about 1 minute and during this period, the integral portion of the ΔTc controller demanded that the "A" loop control valve open wider than "B".

After the feedwater system had increased flow to the steam generators, the larger flowrates into each generator started cooling the cold leg temperatures and reduced Tave. Reactor power increased which was a normal action for the ICS, and steam header pressure also increased. Thus, MWe started to increase but at a rate greater than 2% per minute. Refer to Figure 7.

With the reactor operating in a turbine following mode, the MWe generated signal becomes the power demand input to the ICS and, for a load change greater than 2% per minute, the integral portion of the ΔTc controller is grounded. This situation continued to cause more than normal flow demand to the "A" loop feedwater control valve and the "A" loop flow remained greater than "B" for approximately 5 minutes. In [illegible]

"A" to be cooled more than primary loop "B". Due to incomplete mixing of flows in the reactor core and the slightly positive moderator coefficient, power in the "A" loop portion of the core decreased below that in the "B" loop side..

Unfortuitously, the ICS was attached to a reactor power detector located on the "A" loop side. Its error signal demanded more reactor power and, as reactor power increased, the higher neutron flux of the "B" side caused the detector on that side to trip the reactor. The high flux trip for each of the four neutron flux detectors was set at 85% power. The time for the entire transient to occur was approximately 6 1/2 minutes.

The cause of the initial increase in feedwater demand (with a second pulse about four minutes later) is not obvious. The second pulse was larger than the first but the effect of an "external" input demand cannot be separated from the positive feedback effect of the first pulse as the system changed its operating condition. The positive feedback was enhanced by the turbine being operated in the manual mode with a non-optimized integrated control system.

Several possible sources for initiating the feedwater system upset are:

(1) The fluctuating conditions in the steam supply and drain flow of the reheaters could have upset the high pressure "A" heater such that feedwater temperature increased which in turn would cause a direct gain change in the feedwater demand signal.

(2) A spurious signal influenced the ICS feedwater demand. In the unusual environment during initial plant startup, someone may have [illegible] connected to non-buffered circuit.

REACTOR TRIP #19

(3) A loose connection or intermittent electronic fault existed in the ICS. The random and infrequent occurrence of this upset in the feedwater flow system will make the task of locating this loose connection very difficult.

Steps to Prevent Future Recurrence

(1) Minimize flowrate fluctuations in the heater drain system by verifying that all valves are working properly. Improve valve operation by readjusting liquid level set points and gains on valve. Verify that valves, pumps and controllers are suitable for the duty.

(2) Establish closer coordination of all activities of operating, maintenance, support and service personnel.

(3) Minimize the use of turbine manual mode.

(4) Optimize ICS adjustments.

(5) Evaluate the desirability of manual override of feedwater demand control - including the ratio control.

(6) Evaluate the feasibility of changing the modes and options of transmitting the reactor power level signal to the ICS.

In relating this trip to the standard categories for lifetime design cycle accounting, the assignment has been made to Transient No. 11

/mob

EXAM D ANSWERS

D.1 OARP ICS Less Plan pg. 8

a. Mw Error
ΔT_c
Tavg error demand to reactor
Tavg error demand to FW

b. Used for steady state calibrations

c. Blocked if (demand) ULD is changing > 2%/min. sustained for at least 10 sec. - blocked for 2 min after ULD is no longer changing at 2%/min.

D.2 a. BF_3 Proportional counters - $B(n, \alpha) Li^7$ reaction α + Li ionize gas, + and - charges attracted to electrodes. γ signal also ionizes gas however α + Li produced <u>pulses</u> are larger in magnitude and lower pulses removed by a pulse height descriminator.

OARP pg 7

b. OARP pg 7 - Control rod withdraw ~~and alarm~~ prohibit on high SUR from either channel @ 2 DPM - pg 11.

D.3 Conservative - NI's are always calib against a heat balance and should read ≥ heat balance for reactor protection.

? 0 D.4 Aux Power supply modules in two cabinets are for compensated voltage power supplies. Each cabinet has Aux. Power Supply module for RCP monitors.

D.5 a. ~~200 psid ΔP OARP~~ ~~[illegible]~~ Low 50# ΔP across feed pumps

b. 600 psi — press OARP switch's on Main Stm lines

c. Temp Sensors, ~~[illegible]~~, ~~Activation of Fire Prot Sys.~~ electrial protection manual

d. Mwgen - CT's + PT's from main Gen. phases — Mw Transmitter

D.6 a. Gentilty - Reverse acting Pilot Tubes

b. ICS FEMA - 1-18-Runback to 15% @ 20%/min. [illegible]

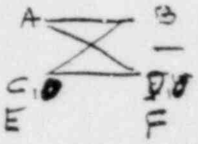
Could get:

1. ΔT_c reratio
2. BTU Limits
3. ICS Runback

d. Question was too vaguely worded. - sensor to <u>initiate</u> an <u>automatic</u> action
Will accept those signal that put ICS in Track

EXAM E ANSWERS

E.1 f - a, b + c



This diagram not complete answer

E.2 MU-V3- ?
OARP vs Rx Trip
1202-4

1) 4 psig RB pressure - can't bypass, must wait for actuation signal to reset, then reset B/S then reset channel. ?

2) Rx trip - Bypass with control switch. - No 4 psig Signal pres. ?
(10 sec Timer)

MU-V2A and MU-V2B

~~1) Rx trip - Bypass with control switch.~~ 1600 psi - HPI
2) 4 psig RB pressure - same as above.
3) High Rad RML 1 - Bypass with defeat switch at RML 1

E.3 a. (.3) 1. High Flux @ -104.75 5 105.5 on
(.6) 2. Low Press @ 1900 psig
~~3. Variable Low Press @~~ Must have these Two
(.6) 4 Hi RB Press 4 Psig
T.S. pg 2-7

Coincidence - any two of these signals on different channels will cause a Rx trip - including same signal on two channels or different signal on 2 channels.

b. Assumed Rx was Tripped
Cooldown will lead to $+\rho$ addition with a negative MTC - Automatic actions include Rx trip - rods insert $-\rho$, ES actuation - ρ from boron. gives at least 1% SDM, FW isolatation @ 600 psi

E.4 a. While system is shutdown, a rotometer located on the supply line is monitored locally.

OARP

b. While operating, a differential between inlet flow and outlet flow (temperature compensated) will alarm in the control room.
Drip Pan Alarm - 0.3 GPM

E.5 a. To sweep out remaining exhaust gases & refill cylinder with fresh air. (Pg 4 OARP)

b. Maintains same engine ~~speed~~ regardless of engine load. (pg 8)

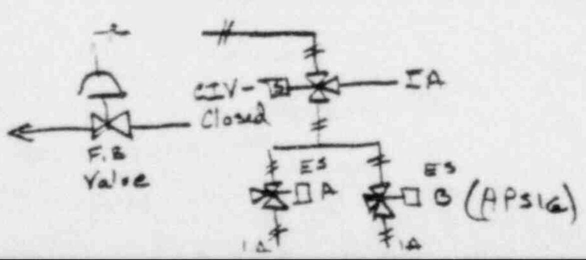
c. 1. Engine Driver
2. DC Distribution panel 1P (1Q) (pg. 11)

d. Speed < 250 RPM } within 7 seconds
Oil press < 6 psi } after start signal (pg 25) } Numbers have changed

E.6 (Ref: pg 8, 9 of ESAS lecture)

a. The fluid block system supplies pressurized water to all CIV bodies outside the reactor building. The pressure is greater than LOCA RB pressure. This assures leakage is into the RB, not out.

b. Two series valves hold fluid block valves closed. One is operated by 4 psig RB signal and the other is operated by CIV closed signal. Need to open both valves to actuate the system.



EXAM F ANSWERS

F.1 (OP-1202-5)

a. 1. Make up Tk reducing >10 gpm (1" per 3 min)
   2. RM-A-5 and/or steam line rad detectors have conferened alarms and PRZ level can not be maintained at 220" by MU-V-17.

b. 1. Sampling OTSG
   2. Surveying steamline
   3. Observation of OTSG levels & feed rates (see note in proc)

c. The effected OTSG must be steamed only as necessary to maintain level $\leq 95\%$ on operate range & OTSG pressure ≤ 1000 psig.
Water hammer, steam line stress, minimize release

F.2 a. 1. ~~Hourly log~~ (AP 1012)
   2. ~~CR log~~
   3. ~~Shift Foreman log~~
   4. ~~Check Lists~~
   5. ~~Recorder charts~~
   6. ~~Computer printouts~~

X See attached STEP 3.6.2

1. Control Room log
2. TCN + SOP
3. Ops Memo Book
4. Revision Review Book
5. Active Tagging Appl.
6. Locked Valve
7. Outst. Surv Sched

AP-1028 b. After obtaining a properly qualified operator at the controls.

F.3 a. Low pressure in the pressurizer steam space, expansion of coolant due to temperature increase, bubble forming in Reactor Vessel Head, HPI.
loss of heat sink

?

b. 1. PORV open
   2. Safety open
   3. RCS saturated pressure when recovered
   4. Bubble in loops or head
   Plus a number of others.

REF: ?

c. If the core exit Tcs are not superheated.
ICC Procedure should have more

F.4 a. 55, 45 (1103-5 p. 12)
b. 100°F (1103-5 p. 4)
c. ≥ 5 (1102-1 p.4)
d. axial power control (1102-4 p.5)

F.5 a. SBLOCA analysis - maintain PCT $< 2200°F$. Required by procedure - evaluate answer.

b. The heat sink is the energy release through the break and the refill by HPI and LPI.

c. See EP 1202-6B pg. 8.0 Rev. 7.0
Caution statement 3/5/81

1. LPI flow stable @ >1000 GPM for 20 min
2. 50° Subcooled + action is necessary to prevent Pzr from going off scale high
3. To prevent pump runout, throttle to 550 GPM/pump
4. To prevent violation of Rx Vessel Brittle fracture limit

F. 2a

STep 2.4

1. Control Room log
2. CRO Turnover ChecklisT
3. ES CheckliST (>250°F)

STEP 3.6.2

1. TCN + SOP Books
2. OperaTions Memo Book
3. Revision Review Book

STEP 3.6.4

1. AcTive Tagging ApplicaTion Book
2. Locked Valve Log
3. OutsTanding Surveillance Schedule

EXAM G ANSWERS

G.1 a. Energy Emission and propagation through space or a medium material in the form of waves.

b. A negatively charged particle having the mass and charge of an electron.

c. Time required for the activity of a radioactive isotope to be reduced by one half due to decay.

G.2 RAD is measure of energy absorbed/unit weight. REM gives the biological effectiveness of an absorbed dose.

G.3 Part 19 deals primarily with instructions to workers concerning radiation hazards and methods to limit same.

G.4 Co60 primarily from Co59 + n^1, where Co59 is found in the steel and other materials in the system. The hazard is that it has a 5 yr half life thus takes a long time to decay. and emits high energy γ's (2)

G.5 a.

G.6 Tertiary fission, and neutron reactions with Boron & Lithium

G.7 All three could but primarily irradiated, both recent and stored.

G.8 Dose at valve = $(45) \times \frac{(10)^2}{(3)^2} = 45 \times \frac{100}{9}$ = 500 mr/hr.

Weekly limit = 300 $\frac{300}{500}$ = .6 hr or 36 minutes

EXAM H ANSWERS

H.1 Saturated liquid must absorb latent heat of vaporization to change phase.

H.2 (Ref: OARP pg. 6)
Quality is expressed in terms of Mass $X = \frac{m_g}{m_g + m_f}$

Void fraction is expressed in terms of Volume $\alpha = \frac{V_g}{V_g + V_f}$

H.3 10^6 lbm/hr @ 400°F, $V_f = .01864\ ft^3/lbm$ $10^6(.01864)(7.48)/60 = 2320$ GPM
10^6 lbm/hr @ 150°F, $V_f = .01634\ ft^3/lbm$ $10^6(.01634)(7.48)/60 = 2037$ GPM

Therefore - more volume (gallons) is pumped at higher temperature since water is slightly compressible.

H.4 a. Nucleate boiling a essentially turbulent flow - promotes more mixing - coolant picks up latent heat of vaporization - carries it to cooler part of channel.

b. In film boiling, a film of steam coats the clad surface and essentially form an insulating layer, $\Delta T \uparrow$ substantially, because $\dot{q} \downarrow$.

H.5 a. (Ref: OARP pg 85) The faster a pump rotates, the greater its required NPSH.

b. Flow rate is doubled.

c. Discharge head $\propto n^2$ Speed increased by 1.22 or 22%
Need 4 times the speed to double the head $\therefore \sqrt[2]{(1.5)}$ = Speed of pump = ~~2.25~~

d. Power $\propto n^3$
Spee~~d required~~ to multiply the head X 1.5
~~$(1.5)^3 = 3.375$~~ $(1.22)^3 = 1.84$

H.6 a. Assuming the RCS has been properly filled and vented, a manometer effect exists in the RCS. The lower PRZ level tap is less than 50 ft. below the top of the hot leg $\therefore$ 50ft X .433 psi/ft = 22 psi is req'd to hold up the water in the hot leg. We have 30 psi & 200" in PRZ - This is sufficient to support the hot leg.

b. Voids existed, they were swept into the PRZ. Void volume was taken up by water from the PRZ.

c. This wold occur in a fully vented system where the S/G was warmer than the RCS. The RCS would heat up somewhat, density $\downarrow$, PRZ level $\uparrow$.
(Pump heat on long term - slower effect)
Credit = .25

PAUL F.COLLINS, CHIEF
OPERATOR LICENSING BRANCH
OFFICE OF NUCLEAR REACTOR REGULATION

1952	Graduated Rensselear Polytechnic Institute, BSME
1952-1953	U. S. Army
1953-1965	EI DuPont de Nemours and Company Reactor Department Supervisor at the Savannah River Plant Held the following jobs: Quality Control Division - Responsible for the receipt and inspection of all reactor components. Control Room Supervisor - Responsible for the day to day operation of a production reactor. Instructor - Reactor Department, Operator Training Div.
1965-1967	Examiner, Operator Licensing Branch, AEC
1968-1969	PWR Group Leader, Operator Licensing Branch, AEC
1969-Present	Chief, Operator Licensing Branch, NRC

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY, ET AL.	)	Docket No. 50-289
	)	(Restart)
(Three Mile Island, Unit 1)	)	

NRC STAFF RESPONSES TO AAMODT SUPPLEMENTAL DISCOVERY REQUESTS

Attached hereto and pursuant to the direction of the Special Master are the Staff's responses to the Aamodt's supplemental discovery requests related to the administration of the TMI-1 operators examinations in October 1981.

Affidavits and the professional qualifications of the person or persons who prepared these responses are also attached.

Respectfully submitted,

Lucinda Low Swartz

Lucinda Low Swartz
Counsel for NRC Staff

Dated at Bethesda, Maryland
this 16th day of November, 1981

STAFF REPONSES TO AAMODT SUPPLEMENTAL DISCOVERY REQUESTS

1. Provide copies of all forms of NRC RO and SRO licensing examinations used in October 1981 examination of TMI-1 operators.

   Answer

   Copies of the examinations have been provided.

2. Provide names and relevant experience of proctors used in above named examination.

   Answer

   Professor William Aungst
   Professor Joseph Douglas
   Professor Wesley Houser
   Professor William Welsh

   All of Pennsylvania State University Capital Campus, Middletown, Pa.

   We do not have information regarding the relevant experience of the proctors, however, they were selected from the faculty at Pennsylvania State University since we felt a college professor would have considerable experience in the proctoring of exams.

3. Provide written and verbal instructions given to proctors for above named examination.

   Answer

   The following written instructions were given to the proctors:

   > All written examinations shall be adequately proctored to assure the integrity of the examinations. Two individuals shall be available for proctoring. One proctor shall be in the examination room at all times giving his/her full attention to the candidates taking the examination. The proctor shall not read facility procedures or other material or grade examinations or engage in other activities that diverts his/her attention from the candidates. The sole exception is to answer a candidate's question on the examination.

   Verbal instructions were given to the proctors by P. F. Collins on September 24, 1981. The contents of these instructions were not recorded; however, they dealt primarily with the requirements to proctor the NRC examinations 100% of the time. Just prior to the examinations the proctors were present when the candidates were read the admonitions regarding cheating. In addition, the proctors were advised of the location of the NRC examiners during the examination and were instructed to contact one of these examiners if any of the candidates had questions regarding the examination.

4. Provide seating charts, including size of room, position of proctors, placement of operators and other candidates, indicating individuals by name or letters.

Answer

Seating charts and a reference drawing of the testing room are enclosed. Proctor position is not indicated since the proctors randomly moved throughout the testing room.

5. How many proctors were in the testing room at all times? Were there any times when no proctor was in the room. Provide these times, if any.

Answer

At least one proctor was in the testing room at all times.

6. Did the proctors keep a list of who left the room, the time an individual left and the time he returned? Provide such a list if it exists.

Answer

The times that individuals were absent from the testing room are indicated on the enclosed seating charts.

7. Did the proctors admonish the candidates regarding cheating? If so, provide what the proctors said.

Answer

Admonitions regard[illegible]eating were made by Bruce Boger (for the RO examinations, [illegible]ruce Wilson (for the SRO examinations). The candidates were [illegible]hat:

a. Cheating on th[illegible]minations means an automatic denial of their applicat[illegible] and could result in more severe penalties.

b. They should sig[illegible] the statement on the facing sheet that indicates that the work is their own, and that they have received or given no assistance in completing the examination.

c. Restroom trips are to be limited and only one candidate at a time may leave.

d. When an individual completes the examination, he shall:

   1. Turn in all scrap paper and the balance of the paper pad used for answering the questions, along with the examination.

2. Leave the general area, as defined by the examiner. If he is found in this area while the exam is still in progress, his application may be denied.

8. Were the candidates asked to sign a pledge on their examination to indicate that the responses on their examinations were solely their own work? If not, why not?

Answer

Yes, each candidate was asked to sign a statement to that effect on the front page of the examination.

9. Did the operators bring their own lunches into the testing room? If so, were the lunches inspected? By whom?

Answer

No. Box lunches were provided by the facility.

10. Was there a designated time for eating lunches? If so, when was that time? Was conversation allowed among the candidates at lunch time or at any other time?

Answer

Lunches were supplied by facility management. They were delivered to the examination room about 11:15 a.m. each day.

About noon each day the proctors served a lunch to each candidate at his table. Conversation between and among candidates was not allowed at any time during the examination, including lunch.

11. Were candidates allowed to leave the room more than one time? How many times? Was there any limitation on the amount of time an individual could remain out of the room?

Answer

Candidates were not limited in the number of times they were permitted to leave the room or in the amount of time they could be out of the room.

12. Did the proctors report any irregularities observed during the examination? If so, what were they? Did anyone interview the proctors following the examinations?

Answer

The proctors reported that one candidate requested that he be allowed to take a piece of paper with his anticipated scores from the testing room upon completion of his examination. This request was denied. The same individual initialed (rather than signed by signature) the pledge noted in question 8.

Although no formal interviews were conducted with the proctors, NRC examiners were present during and after the examinations. General discussions with the proctors after the completion of the examinations revealed only the irregularities noted above.

13. Where were the proctors' positions within the testing room? When an operator had a question, did the proctor attend to it at the operator's table? Did the proctors keep a list of questions? If so, provide this list and name of questioner, if known, and time of questioning.

Answer

The proctors circulated through the room during the examinations. When a candidate had a question, the proctor or an NRC examiner would attend to it at the candidate's table. No list of questions was maintained.

14. What Licensee personnel were present in the testing room at any time during the administration of the licensing examinations? Describe the purpose of each such person's presence, the times present and any conversations engaged in.

Answer

Licensee personnel, other than those taking the examination, were present in the testing room when the examinations were handed out and during the examiner's explanation of the examination rules. Their presence was permitted to allow observation of the discussion of the examination rules. It is estimated that the time these persons were present in the testing room was 15 minutes, during which no conversations took place. No list of extra licensee personnel present was maintained.

15. Provide the name of each proctor during the licensing examination on each day of the examination, indicating the times each individual was in the testing room and the reason and times of any absences from the testing room.

Answer

There were four individuals who proctored the examinations, as listed below. There was one proctor in the examination room at all times. The proctors left the room for personal comfort reasons or to phone one of the examiners regarding clarification of questions. The proctors did not keep a record of the times they were not in the room.

Professor William Aungst	October 22 & 24 - all day
Professor Joseph Douglas	October 21 & 28 - all day

Professor Wesley Houser	October 21 & 28 - all day October 22 & 29 - afternoon
Professor William Welsh	October 22 & 29 - morning

16. Provide the time each day of examination began and the time scheduled to complete the examination. Provide a list of times when each candidate turned in his examination responses for each day of the examination.

<u>Answer</u>

The examinations started between 8:00 a.m. and 8:30 a.m. on each day. Candidates were allowed 9 hours to complete the RO examination and 7 hours to complete the SRO examination. Actual completion times are noted on the seating charts. Completion times on October 21, 1981 were not recorded.

17. List any and all materials provided to the candidates.

<u>Answer</u>

The NRC provided examinations and answer pads to each candidate. The licensee provided each candidate with pencils, pens, erasers, and calculators. No other materials, apart from aspirins and the box lunches, which were provided by the Licensee, were given to the candidates.

18. List any and all materials carried into the testing room by the candidates.

<u>Answer</u>

Candidates carried some or all of the following into the testing room: pencils, pens, erasers, calculators, and coffee cups.

19. Were there any materials, other than those included in questions 17 and 18, that were available to the operators within or outside the testing room?

<u>Answer</u>

No other materials were available to the candidates within the testing room or outside the testing room in the examination area (between the testing room and the rest room).

20. Describe the testing room. What is Licensee's normal use of this room? Did Staff inspect it prior to the examination? If so, when?

<u>Answer</u>

The testing room was approximately 25 feet by 40 feet and is normally used as a classroom. The testing room was inspected by the Staff (P. F. Collins and proctors) on September 24, 1981.

21. What individual had possession of the examinations prior to the commencement of the testing? Were any Licensee personnel allowed to view any form of the examinations prior to its use? If so, indicate who and when and what forms.

Answer

After the examinations had been reproduced, Bruce Boger had possession of the RO examinations and Bruce Wilson had possession of the SRO examinations. No Licensee personnel were allowed to review any form of the examinations prior to their use.

22. What individual collected the examinations after each testing day? If more than one individual, indicate by name and day.

Answer

The examinations were collected either by the proctors present (see answer to question 15) or Bruce Boger on October 21 and 22 or Bruce Wilson on October 28 and 29.

23. Provide names of persons who will grade the examinations, indicating relevant experience and categories to be graded. When is the grading expected to be completed?

Answer

RO examination: Bryan Gore - categories A, B, and C; James Huenefeld - categories D, E, and F; Walter Apley - categories G and H.

SRO examination: Bruce Wilson - categories - all (SRO A); Joseph Buzy - categories L, M, and N (SRO B); Robert Campbell - categories I, J, and K (SRO B).

The graders' relevant experience is as follows:

Byran Gore - Technical training in B&W reactor systems, four years of class room experience as an Assistant Professor of Physics at the University of Idaho.

Walter Apley - Technical training in B&W reactor systems, officer in the U.S. Navy Nuclear Power Program.

James Huenefeld - Technical training in B&W reactor systems, officer in the U.S. Navy Nuclear Power Program.

Bruce Wilson - member Operator Licensing Branch since 1973.

Joseph Buzy - member Operator Licensing Branch since 1963.

Robert Campbell - member Operator Licensing Branch since 1967.

Grading is expected to be completed by November 30, 1981.

24. Provide the grading key for each form of the licensing examination. How was each grading key developed? Did Licensee personnel have any input regarding grading? Did EE? If yes to either, indicate what those inputs were.

Answer

A grading key for each examination will be provided after the grading is completed. The key was developed by the examiner writing the question. Licensee personnel have no input regarding grading; however, the licensee (including EE) reviewed the grading key after the start of the examination so that any inappropriate questions could be discussed and all answers in the key could be verified as currently valid.

25. What NRC licensing examinations, if any, were made available to Licensee since the April 1981 licensing examination? Indicate facility where used and date.

Answer

The only NRC examinations provided to the Licensee by the NRC were provided as a result of these proceedings. The April 1981 licensing examinations have been provided.

26. Indicate where Staff's administration of the October 1981 licensing examinations differed from the standards of ES-201 of 2/15/69. Provide the same comparison, including Revision 3 to ES-201.

Answer

Revision 3 to ES-201 was complied with during the October 1981 examinations. Since this revision meets or exceeds the previous requirements in ES-201, compliance with ES-201 of 2/15/69 was attained. For instance, revision 3 to ES-201 requires 100% proctoring while the previous requirement was to "make use of available facilities, in the manner he (the examiner) considers most satisfactory, to insure the integrity of the examination". Hence, the hiring of proctors to achieve 100% proctoring during the October 1981 examinations meets the requirements of ES-201, Revision 3, and exceeds the guidance provided in ES-201 of 2/15/69.

27. Were rest rooms proctored?

Answer

No. However, only one candidate was allowed out of the examination room at a time. The proctors performed a hall watch for the first few candidates until they had a time frame for absence from the room. After the time frame was established, absences were recorded and random hall watches were made.

28. Were measures taken to prevent use of phones as a means of eliciting information? If so, what were those measures?

Answers

Yes. There were no phones in the testing room. The proctors monitored restroom leaves and occasionally monitored the hallway between the restroom and the testing room. Candidates were notified prior to the examination that leaving the testing area would be grounds for denial of his application.

29. Were chance meeting places, like coffee stands, proctored?

Answer

Yes. The coffee pot was in the front of the testing room and was clearly visible to the proctors.

30. Were materials that examinees carried into the testing room, such as slide rules, calculators or steam tables, inspected?

Answer

No paper, including steam tables, was allowed into the testing room. Other materials carried by the examinees were not inspected.

31. Who wrote the licensing examination? Was that person assisted in any way by anyone else? If so, who? What was the nature of that help?

Answer

The RO examinations were written by Bryan Gore, James Huenefeld, and Walter Apley. Bruce Wilson reviewed these examinations for technical content, applicability and clarity. The SRO examinations were written by Bruce Wilson.

32. Will the grading process include any effort to screen examinations for cheating? If so, describe those efforts.

Answer

The examinations will be screened for indications of cheating by reviewing the answers for at least one question in 50% of the categories for 50% of the applicants.

33. Were there assigned seats for examinees?

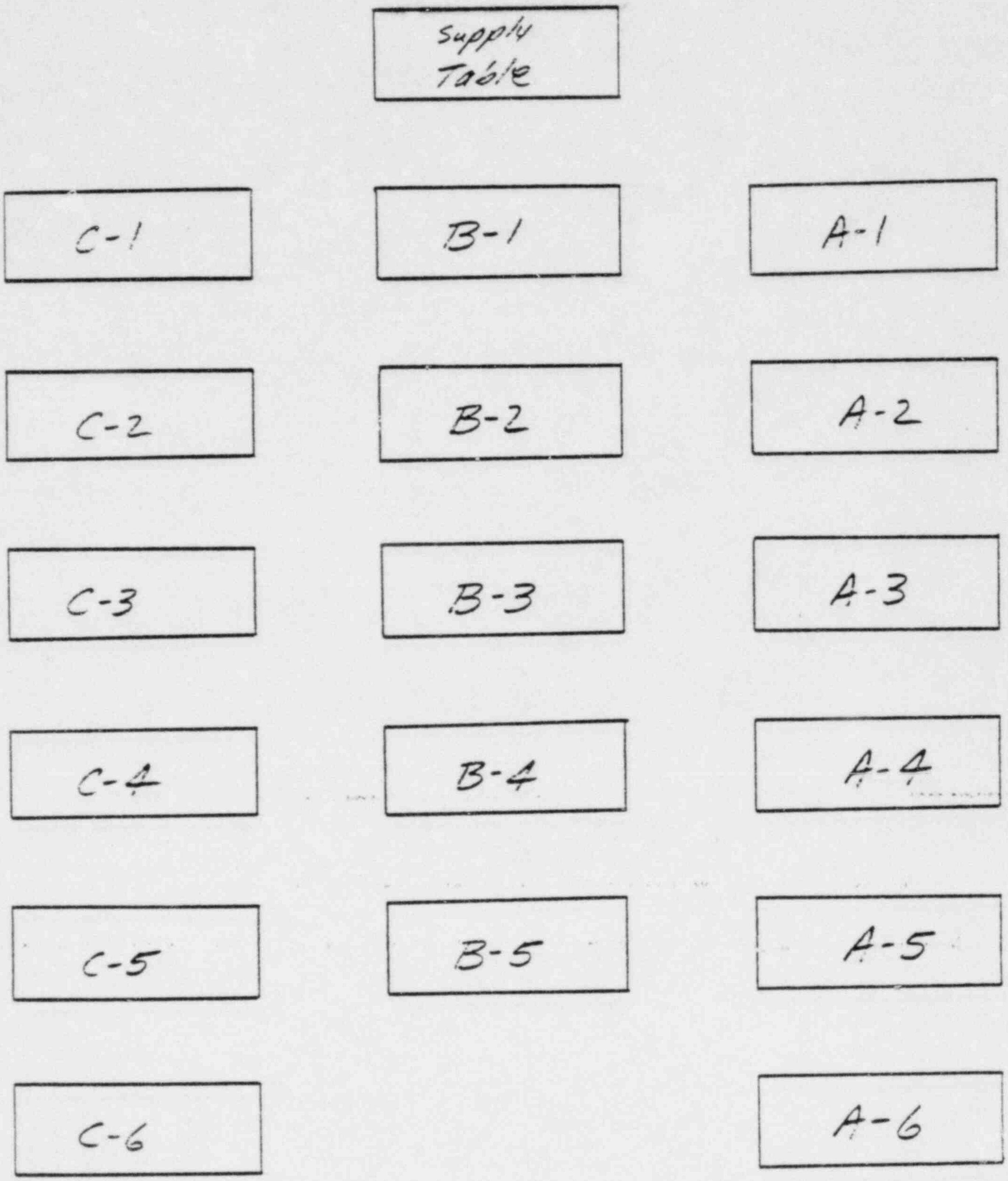
Answer

No, seats were not assigned.

34. Were examinees required to leave the building if they finished before the time limit of the test had run?

Answer

Yes. Candidates were notified prior to the examination that to be in the testing area while the exam was still in progress was grounds for denial of his application.



TESTING ROOM
APPROX. SIZE
25' x 40'

Seating Chart — October 21, 1981

(question 4) Seat No. / Candidate	(Question 6) Out of Room	(question 16) Completion Time
A-1 / A	10:35-10:37	*
A-2 / WW	10:02-10:04, 11:53-11:54	*
A-3 / AA	11:19-11:21, 16:13-16:15	*
A-4 / H	NONE	*
A-5 / G	10:15-10:17	*
A-6 / C	9:49-9:51, 13:55-13:58	*
B-1 / GG	9:42-9:46, 11:51-11:53	*
B-2 / FF	9:39-9:41, 11:33-11:35	*
B-3 / T	10:38-10:39, 15:49-15:50	*
B-4 / D	9:31-9:33, 11:03-11:05, 13:12-13:15, 15:02-15:03	*
B-5 / KK	12:47-12:49	*
C-1 / B	10:46-10:47, 13:07-13:09	*
C-2 / CC	9:46-9:48, 12:03-12:05	*
C-3 / E	11:54-11:57	*
C-4 / S	11:48-11:49, 14:20-14:21, 16:28-16:32	*
C-5 / BB	9:05-9:10, 12:39-12:42	*
C-6 / EE	11:59-12:01	*

* - Completion times were not recorded, but all examinations were completed before 17:30.

Seating Chart — October 22, 1981

(Question 4) Seat No. / Candidate	(Question 6) Out of Room	(Question 16) Completion Time
A-1 / QQ	NONE	16:15
A-2 / L	12:04-12:06, 13:46-13:48	16:07
A-3 / I	12:20-12:22	15:53
A-4 / Y	9:06-9:09, 11:15-11:17	14:32
A-5 / EMPTY TABLE		
A-6 / Z	10:21-10:23, 1:51-1:55	17:13
B-1 / OO	12:51-12:54	15:58
B-2 / V	9:54-9:56, 10:31-10:33, 11:54-11:56	14:29
B-3 / UU	10:23-10:25, 13:08-13:10, 14:56-14:58	15:46
B-4 / DD	9:57-9:59, 12:12-12:15, 14:15-14:18, 16:00-16:03	16:45
B-5 / Q	16:15-16:18	17:13
C-1 / U	9:19-9:21, 11:37-11:39, 13:27-13:29	15:47
C-2 / EMPTY TABLE		
C-3 / P	9:24-9:25, 10:40-10:41, 13:00-13:02	15:30
C-4 / R	12:27-12:29, 15:37-15:38	16:46
C-5 / RR	10:08-10:11, 13:04-13:06	17:13
C-6 / F	12:55-12:56	14:11

Seating Chart - October 28, 1981

(Question 4) Seat No. / Candidate	(Question 6) Out of Room	(Question 16) Completion Time
A-1 / EMPTY TABLE		
A-2 / WW	9:54-9:55, 11:34-11:36	14:38
A-3 / I	10:15-10:17	14:40
A-4 / EMPTY TABLE		
A-5 / Z	9:10-9:12, 11:05-11:07	14:58
A-6 / F	11:36-11:38	13:00
B-1 / FF	9:25-9:26, 11:02-11:04, 13:17-13:18	15:00
B-2 / EMPTY TABLE		
B-3 / EMPTY TABLE		
B-4 / EMPTY TABLE		
B-5 / EMPTY TABLE		
C-1 / EE	10:47-10:48	14:45
C-2 / CC	10:18-10:20, 13:15-13:17	14:33
C-3 / E	10:59-11:01	14:30
C-4 / S	1:29-1:36	15:00
C-5 / RR	9:22-9:24, 11:18-11:20, 14:04-14:05	14:50
C-6 / BB	9:14-9:17, 13:13-13:15	14:34

Seating Chart - October 29, 1981

(Question 4) Seat No / Candidate	(Question 6) Out of Room	(Question 16) Completion Time
A-1 / QQ	13:38 - 13:42	14:40
A-2 / A	10:45 - 10:47, 13:43 - 13:45	14:45
A-3 / KK	11:55 - 12:02	14:55
A-4 / EMPTY TABLE		
A-5 / EMPTY TABLE		
A-6 / EMPTY TABLE		
B-1 / U	9:51 - 9:53, 11:48 - 11:51	14:33
B-2 / GG	9:40 - 9:45, 11:29 - 11:31	14:57
B-3 / EMPTY TABLE		
B-4 / DD	8:51 - 8:53, 9:56 - 9:58, 12:59 - 13:01	14:52
B-5 / EMPTY TABLE		
C-1 / B	12:02 - 12:03	14:27
C-2 / P	9:02 - 9:03, 9:47 - 9:49, 11:04 - 11:06, 13:03 - 13:05	14:47
C-3 / EMPTY TABLE		
C-4 / EMPTY TABLE		
C-5 / EMPTY TABLE		
C-6 / EMPTY TABLE		

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	Docket No. 50-289
METROPOLITAN EDISON COMPANY, ET. AL.	)	(Restart)
(Three Mile Island, Unit 1)	)	

AFFIDAVIT OF Bruce A. Boger

I, Bruce A. Boger, being duly sworn, do depose and state:

1. I am employed by the Operator Licensing Branch, Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission. In th s position, I am responsible for administering licensing examinations. A copy of my professional qualifications statement is attached.

2. I prepared or helped to prepare responses to the following intterogatories which were filed in the TMI-1 restart reopened proceeding on management issues on October 2, 1981: Aamodt supplemental discovery requests: Questions 1-9, 11-14, 16-26, and 28-34.

3. I certify that the answers given to the interrogatories listed above are true and correct to the best of my knowledge.

Bruce A. Boger

Bruce A. Boger

Sworn before me this 16th day
of November 1981.

Marilyn Jolleston
Notary Public

My Commission expires: July 1, 1982

PROFESSIONAL QUALIFICATIONS LIST

BRUCE A. BOGER

Education

June 1971	Received BSNE - University of Virginia
June 1972	Received MENE - University of Virginia

Work Experience

June 1972 to June 1977	Virginia Electric and Power Company Surry Nuclear Power Station Assistant Engineer - Performed startup testing on Unit No. 2. Engineer - Assisted the Supervisor-Engineering Services; trained for and received a Senior Reactor Operator License. Supervisor - Engineering Services - Directed the activities of the onsite engineering staff.
June 1977 to September 1977	Virginia Electric and Power Company Richmond, Virginia Supervisor - Nuclear Engineering Services - Directed the activities of the offsite engineering staff in support of Surry Power Station.
October 1977 to Present	U. S. Nuclear Regulatory Commission Bethesda, Maryland Reactor Engineer in the Operator Licensing Branch - Administer licensing examinations to nuclear power plant and research reactor personnel.

Professional Affiliations

Registered Professional Engineer - State of Virginia

Member - American Nuclear Society

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of METROPOLITAN EDISON COMPANY, ET. AL. (Three Mile Island, Unit 1)	)	Docket No. 50-289 (Restart)

AFFIDAVIT OF Paul F. Collins

I, Paul F. Collins, being duly sworn, do depose and state:

1. I am employed by the Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission.

   In this position, I am responsible for the Operator Licensing Branch.

   A copy of my professional qualifications statement is attached.

2. I prepared or helped to prepare responses to the following intterogatories which were filed in the TMI-1 restart reopened proceeding on management issues on October 2, 1981: Aamodt supplemental discovery requests: Questions 10, 15, and 27.

3. I certify that the answers given to the interrogatories listed above are true and correct to the best of my knowledge.

Paul F. Collins

Sworn before me this ______ day
of November 1981.

Notary Public

My Commission expires: ____________

PAUL F. COLLINS, CHIEF
OPERATOR LICENSING BRANCH
OFFICE OF NUCLEAR REACTOR REGULATION

1952	Graduated Rensselear Polytechnic Institute, BSME
1952-1953	U. S. Army
1953-1965	EI DuPont de Nemours and Company Reactor Department Supervisor at the Savannah River Plant Held the following jobs: Quality Control Division - Responsible for the receipt and inspection of all reactor components. Control Room Supervisor - Responsible for the day to day operation of a production reactor. Instructor - Reactor Department, Operator Training Div.
1965-1967	Examiner, Operator Licensing Branch, AEC
1968-1969	PWR Group Leader, Operator Licensing Branch, AEC
1969-Present	Chief, Operator Licensing Branch, NRC

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

RECEIVED NOV 18 1981 U.S. NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY, ET AL.	)	Docket No. 50-289
	)	(Restart)
(Three Mile Island, Unit 1)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC STAFF TESTIMONY OF PAUL F. COLLINS REGARDING THE ADEQUACY OF THE STAFF'S ADMINISTRATION OF ITS EXAMINATIONS," "STAFF RESPONSES TO AAMODT SUPPLEMENTAL DISCOVERY REQUESTS," and "NRC STAFF RESPONSE TO MEMORANDUM AND ORDER ON RESULTS OF SRO AND RO WRITTEN AND OPERATING WALK-THROUGH EXAMINATIONS" have been served on the following by deposit in the United States mail, or in the Nuclear Regulatory Commission's internal mail system, or as indicated by an asterisk by hand-delivery this 17th day of November, 1981 in the above-captioned proceeding. Ivan W. Smith was served by hand-delivery on November 16, 1981 in the above-captioned proceeding.

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DS07 50/1

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Certified By [signature]

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