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DOCUMENT NO.

HW-71408 Vol 2 RD

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DATE

August 15, 1962

TITLE

NPR Hazards Review - (Phase I - Production Only
Appendixes)

AUTHOR

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NPR HAZARDS REVIEW

(PHASE I - PRODUCTION ONLY - APPENDIXES)

by

N. R. Miller and R. E. Trumble

August 15, 1962

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INTRODUCTION

The NPR Hazards Review is being issued in a series of volumes. Volume 1, which has already been published, was of the nature of an expanded summary. It included the results of hazards analyses with some explanatory material to put the results in context.

Volume 2 presents results of reviews made after the preparation of Volume 1. It also contains supporting material and details not included in Volume 1.

Volumes 1 and 2 together provide a nearly complete "Design Hazards Review of the NPR". However, certain remaining problems still exist and are to be the subject of a continuing R & D program. These problems and programs are discussed in Appendix H.

Neither Volume 1 nor Volume 2 treat operational aspects of reactor hazards in detail. This area of concern will be the primary subject of a third volume of the NPR Hazards Review. This third volume, to be prepared and issued at a later date, may also contain information supplementing Volumes 1 and 2.

ACKNOWLEDGMENTS

This volume is the result of the efforts of many people, most of whom are recognized in the bibliographies. The editors, in addition, take pleasure in acknowledging the help of the following specific individuals.

R. L. Junkins, J. J. Fuquay, and others of the staff of Hanford Laboratories, who prepared Appendix A, "The Reactor Site."

R. W. Benoliel, who prepared part of the description of the reactor in Appendix D, "The Reactor Complex," and who wrote a portion of Appendix H, "Remaining Problems and Research and Development Programs."

R. Nilson and the staff of Reactor Physics, who prepared the section, "Physics," in Appendix C, "The Reactor Process."

R. E. Hall, who wrote the section on control of the heat dissipation system in Appendix C, and who also aided in the preparation of other phases of the volume.

W. A. Burns, who wrote the section on reactor cooling in Appendix C, and who also was an author of a document which was essentially abstracted for Appendix D, "Reactor Control and Safety Systems."

R. B. Hall and P. C. Jerman, who supplied information on waste disposal.

W. S. Nechodom, who also contributed to the document Appendix D is based on.

D. D. Stepnewski, who performed most of the calculations which are discussed in Appendix E, "The Confinement System," and who made many very helpful suggestions in the preparation of that appendix.

E. C. Waters, who wrote Appendix F, "Consequences of Fission Product Release."

D. H. Curtiss, who wrote a portion of Appendix H, "Remaining Problems and Research and Development Programs."

APPENDIX ATHE REACTOR SITE1. GEOGRAPHY

The Hanford Site covers a 602 square mile area in the south-central part of the State of Washington. In addition to the NPR, there are in the Hanford Reservation five other reactor areas for eight production reactors, two chemical processing areas, and an area containing fuel fabrication facilities, R & D facilities, the Plutonium Recycle Test Reactor and various supporting facilities. These manufacturing areas (shown in Figure 3.2 of Volume 1) cover an area of about 194 square miles.

The NPR is located about midway between two existing reactor areas, D Area and K Area. The finished grade elevation at the site ranges from a low of 421 feet MSL at the river pumphouse to about 455 feet MSL at the reactor and heat dissipation buildings.

2. POPULATION DISTRIBUTION

A graphic representation of the results of the 1960 Census for the State of Washington is given in Figure 3.1 of Volume 1. A more detailed map of the immediate vicinity of the plant is given in Figure A-1. The small towns nearest the plant are all indicated while at greater distances only the larger cities are indicated.

The population of the surrounding area is further illustrated by Table A-1 which gives some characteristics of the counties in the immediate vicinity of the HAPO plant.

* Population Information Not Available For These Unincorporated Towns.

FIGURE A-1

Population Distribution - Vicinity of Hanford Site - 1960

TABLE A-1POPULATION CHARACTERISTICS - 1960

<u>County</u>	<u>Area Sq. Mi.</u>	<u>POPULATION</u>			<u>Rural Population Density People/Sq. Mi.</u>
		<u>Urban</u>	<u>Rural</u>	<u>Total</u>	
Benton	1,712	40,555	21,515	62,070	12.6
Franklin	1,260	17,416	5,926	23,342	4.7
Yakima	4,273	74,250	70,862	145,112	16.6
Grant	2,690	21,116	25,361	46,477	9.4
Adams	1,895	2,669	7,260	9,929	3.8
Kittitas	2,315	8,625	11,842	20,467	5.1
Walla Walla	1,272	28,567	13,628	42,195	10.7
Spokane	1,763	234,876	43,457	278,333	24.6
Lincoln	2,300	---	10,919	10,919	4.7
Umatilla, Oregon	3,231	28,486	15,683	44,169	4.8
Morrow, Oregon	2,059	2,655	2,195	4,850	1.1

3. GEOLOGY AND HYDROLOGY

The geology at the N Reactor site is that common to the other manufacturing areas at HAPO, differing only in detail. Two formations of interest overlie the greater than 10,000 foot thick basaltic bedrock. These are the Pleistocene Ringold formation and the late-Pleistocene to recent fluviatile and glaciofluviatile sediments. Both formations are of concern in the disposal of contaminated materials to the ground and in the damage potential of earthquake shocks.

The Ringold Formation lies on basalt at a depth of about 500 feet, or about 50 feet below sea level. The Ringold Formation beneath the NPR site consists of silt, sand, clay and gravel in variable proportions. Sands and sandy gravels, locally cemented, predominate near the top of the formation. Permeabilities and transmissibilities accordingly vary widely in the different materials. Available data⁽²⁾ indicated that with "normal" gradients, the average velocity of water in the uppermost aquifer of the Ringold Formation is about 60 feet per day northwestward toward the Columbia River. However, the average rates incorporate some velocities at least several times the average so that reliance upon a maximum rate or minimum time of travel is hazardous.

The late Pleistocene to recent fluviatile and glaciofluviatile sediments overlie the irregularly eroded surface of the Ringold Formation at a depth of about 55 to 70 feet, and extend upward to the ground surface. There they are lightly veneered with wind-blow fine sand and silt. These fluviatile sediments are in part reworked from the older Ringold Formation sediments so that in places they are distinguishable with difficulty from those older sediments. Elsewhere they consist of poorly sorted, but

generally coarse-grained sand to pebble and cobble gravel and boulders, largely derived from the basalt plateau to the north and northeast. These fluviatile sediments are generally far more permeable than the underlying Ringold Formation sediments.⁽³⁾ In some instances differences in permeability are as great as 100-fold.

The ground water table normally lies about 5 feet below the Ringold Formation surface. During periods of high river level, or locally at the site of discharge of large volumes of water to ground, the water table will be raised, at least locally, into the overlying highly permeable fluviatile sediments. Flow rates and travel times to the Columbia River, indeed the path itself, are materially altered, depending on where and how much of the water table in the flow path lies in those fluviatile sediments.

The vertical rate of movement through the fluviatile sediments down to the water table was estimated⁽²⁾ to be about 15 feet per day with an infiltration rate of about 10 to 20 gallons per square foot per day.

4. SEISMOLOGY

The Hanford area lies in a region susceptible to earthquake damage from the active seismic zones of Western Washington and from the seismic zone that includes the Walla Walla area.⁽⁴⁾ For building code purposes the entire area east of the Cascade Mountains is in Zone 2 of the Seismic Probability Map.⁽⁵⁾ However, this generalization encompasses Zones 0 to 3, depending upon specific and local geologic conditions, the natural vibration periods of materials upon which structures are founded, and the vibration periods of the structures themselves. The zone range implies considerable structural damage to some structures even in a mild shock.

Although some regional maps published as recently as 1944 show that the Hanford area marks the eastern end of a network of faults having a general east-west trend, the Geologic Map of Washington (1961) discloses few faults of more than local significance there. Extensive field mapping between 1944 and 1961 in many key areas failed to disclose positive evidence of faults in the Hanford vicinity. Hence, it must be assumed that the fault pattern earlier indicated was based upon assumptions not later substantiated. However, earthquakes often occur along faults that are unrecognized or that do not have surface expression. An earthquake at the town of Corfu in 1918 (see Figure A-1), a swarm of minor shocks at Ellensburg in 1934, and some at other times, and a swarm at Othello in 1957 imply an active seismic zone along the north side of the Saddle Mountains to the north of the reservation. If so, though weak, it is a possible source of shocks.

Geologic studies indicate that the region is still tectonically active,⁽⁶⁾ in contrast to the conclusions of earlier workers. Substantiation of tectonic deformation recently is provided by abundant, relatively unmodified landslides in the Saddle Mountains, in the Snively Ranch area of the Rattlesnake Hills, and along the face of the Horse Heaven Hills between Kiona and Toppenish, Washington. Landslides for several miles along the north face of the Saddle Mountains were caused by the Corfu earthquake of 1918, which achieved a local intensity not greater than about MM 6.

Subsequent to 1949, increased concern about earthquakes and increased attention to seismic events has disclosed the fact that many earth-tremors are being detected at Hanford.⁽⁷⁾ This however, is the

result of that better awareness and is not necessarily the result of greater seismic activity in the area.

5. METEOROLOGY

A meteorological unit is maintained at HAP0 for the dual purpose of recording conditions and making forecasts and of performing research and development pertinent to plan operations. A 410 foot tower fitted with temperature and wind measuring devices is located on the plateau between the two Separations Areas and is the principal source of local information. In addition, a network of stations measuring ground winds is maintained.

Atmospheric diffusion studies are carried out using a network of 20 air sampling masts which permit a three dimensional characterization of the diffusional process.

5.1 Temperature Stratification

The difference in temperature between the 3-foot and 200-foot level at the meteorological tower is used as a measure of atmospheric stability. If the air at the 200-foot level is warmer than at ground level by 2.5 F or more, the atmosphere is considered very stable. If the air at the 200-foot level is less than 2.5 F warmer, but no more than 0.5 F cooler than at ground level, the atmosphere is considered stable. If the ground level temperature exceeds the 200-foot temperature by 0.5 F to 1.5 F the atmosphere is considered neutral. If the ground level temperature exceeds the 200-foot temperature by more than 1.5 F, the atmosphere is considered unstable. These definitions are shown algebraically in Table A-2. The table also presents the frequency of occurrence of each atmospheric condition for each season and for the entire year for the period January 1955 through June 1961.

Only very limited information on temperature stratification above 400 feet is available at HAPO. The observations indicate that temperature inversions up to 10,000 feet MSL are very rare during late spring, summer, and early fall but are quite frequent during the colder season.

The frequency of higher level inversions in various height intervals for the period November 1955 through March 1956 is shown in Table A-3.

5.2 Wind Speed Versus Lapse Rate

The joint frequency distribution, in per cent of total time, of various class intervals of wind speed at 200 feet and the temperature lapse rate from 3 feet to 200 feet is shown in Table A-4.

5.3 Wind Roses at Two Hundred Feet

The wind roses for the 200-foot level at the Meteorology Tower are summarized in Tables A-5 through A-9. These tables show the per cent of time the 200-foot wind blew from each of 16 directions and with an hourly mean speed in each of 7 class intervals of speed during each season and during the entire year for four atmospheric stability classifications.

5.4 Wind Direction During Periods of Precipitation

The frequency of wind direction at the 50-foot level at the Meteorology Tower during periods of precipitation is presented in Table A-10.

5.5 Wind Distributions in the Vicinity of the Reactor Areas

The surface wind distributions in the vicinity of the reactor areas are summarized in Figures A-2 through A-4. Measurements of wind direction and speed at remote wind stations are summarized for the period 1952 to 1956. The wind roses are contrary to the conventional notation, and indicate the direction toward which the wind was blowing to facilitate visualizing trajectories.

Figure A-2 summarizes the wind patterns observed during the nighttime hours, 1900 to 0700 PST. The nocturnal drainage wind observed at Hanford is evident from the high frequency of downslope or northwest winds along the major axis of the valley and general convergence of flow along the river.

TABLE A-2
PERCENTAGE FREQUENCY OF OCCURRENCE OF VERY STABLE
MODERATELY STABLE, NEUTRAL, AND UNSTABLE LAPSE RATES
AT THE HANFORD METEOROLOGY TOWER

(Based on hourly observations for the period
 January 1955 through July 1961)
 $\Delta T = (T_{200'} - T_3')$, °F

<u>Lapse Rate Classification</u>	<u>SEASON</u>				
	<u>Winter</u>	<u>Spring</u>	<u>Summer</u>	<u>Fall</u>	<u>Annual</u>
Very Stable (VS) ($\Delta T > 3.5$)	24	22	17	22	24
Moderately Stable (MS) ($3.5 > \Delta T > -0.5$)	43	32	29	31	34
Neutral (N) ($-0.5 > \Delta T > -1.5$)	24	10	9	13	14
Unstable (U) ($\Delta T < -1.5$)	9	36	45	23	28

TABLE A-3
PER CENT OF UPPER AIR OBSERVATIONS WHICH SHOWED
A TEMPERATURE INVERSION WITHIN THE INDICATED HEIGHT INTERVAL

<u>Height Interval (Ft Above MSL)</u>	<u>Frequency of Temperature Inversion (% of Observations)</u>
1000-3000	23
3000-5000	10
5000-10000	12

TABLE A-4

PERCENTAGE FREQUENCY DISTRIBUTION OF 200-FOOT WIND SPEED
AND ATMOSPHERIC STABILITY AT THE HANFORD METEOROLOGY TOWER

(Based on hourly observations for the period
January 1955 through July 1961)

<u>Wind Speed Group</u>	<u>Stability</u>	<u>Winter</u>	<u>Spring</u>	<u>Summer</u>	<u>Fall</u>	<u>Annual</u>
0 - 4	VS	9.66	5.52	2.50	11.44	7.13
	MS	13.70	3.38	2.16	7.32	6.54
	N	13.66	3.19	2.90	6.62	6.51
	U	2.43	9.54	12.62	8.94	8.41
5 - 9	VS	7.07	7.27	5.89	10.63	7.63
	MS	8.61	5.34	3.18	5.76	5.70
	N	4.95	2.04	1.99	2.55	2.87
	U	2.73	11.01	16.42	7.06	9.42
10-14	VS	5.22	6.45	6.22	8.44	6.53
	MS	9.14	8.37	6.97	7.38	7.98
	N	3.04	1.94	1.35	1.70	2.01
	U	1.75	6.55	8.24	3.03	4.98
15-19	VS	1.91	2.11	2.17	2.21	2.10
	MS	5.99	7.90	7.62	5.38	6.78
	N	1.31	1.30	0.97	1.09	1.17
	U	0.90	3.97	4.04	1.74	2.71
Greater than 19	VS	0.40	0.18	0.18	0.35	0.27
	MS	5.90	6.97	8.71	5.15	6.73
	N	0.80	1.69	2.08	1.26	1.47
	U	0.83	5.29	3.77	1.93	3.02

**PERCENTAGE FREQUENCY DISTRIBUTION OF WIND SPEED AND
WIND DIRECTION AT 200-FOOT LEVEL VS. ATMOSPHERIC STABILITY**

(January 1955 through July 1961)

SPRING

	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	N	Var.	Calm	Total
0 - 3	VS 0.11	0.12	0.08	0.14	0.21	0.28	0.18	0.17	0.20	0.27	0.19	0.42	0.30	0.37	0.28	0.23	0.12	0.26	3.93
	MS 0.11	0.16	0.08	0.08	0.26	0.23	0.10	0.11	0.10	0.09	0.08	0.13	0.10	0.10	0.17	0.17	0.10	0.13	2.33
	N 0.10	0.25	0.13	0.18	0.25	0.22	0.06	0.10	0.06	0.08	0.03	0.14	0.10	0.14	0.20	0.25	0.16	0.09	2.53
	U 0.50	0.69	0.48	0.50	0.36	0.28	0.08	0.28	0.19	0.17	0.11	0.17	0.19	0.37	0.43	0.84	0.58	0.	6.23
4 - 7	VS 0.12	0.12	0.09	0.11	0.16	0.29	0.21	0.23	0.15	0.40	0.52	0.89	0.98	0.98	0.43	0.32	0.01	0.	6.01
	MS 0.09	0.08	0.07	0.19	0.23	0.42	0.23	0.25	0.21	0.21	0.33	0.47	0.42	0.46	0.25	0.23	0.01	0.	4.16
	N 0.05	0.09	0.06	0.10	0.08	0.27	0.13	0.11	0.06	0.06	0.12	0.14	0.20	0.25	0.08	0.10	0.03	0.	1.93
	U 0.76	0.86	0.55	0.56	0.38	0.58	0.37	0.61	0.57	0.60	0.41	0.39	0.56	1.26	1.12	1.25	0.19	0.	11.33
8 - 12	VS 0.08	0.11	0.08	0.12	0.06	0.09	0.19	0.13	0.16	0.25	0.60	1.17	1.91	1.61	0.44	0.20	0.	0.	7.21
	MS 0.11	0.13	0.03	0.08	0.05	0.19	0.23	0.28	0.31	0.40	0.65	1.10	1.65	1.37	0.27	0.16	0.	0.	6.99
	N 0.07	0.04	0.01	0.05	0.05	0.06	0.10	0.09	0.10	0.14	0.14	0.15	0.32	0.51	0.10	0.10	0.	0.	2.04
	U 0.49	0.43	0.21	0.10	0.11	0.12	0.21	0.21	0.47	0.82	0.71	0.46	0.77	1.44	0.49	0.55	0.	0.	7.55
13-18	VS 0.05	0.03	0.03	0.03	0.	0.05	0.09	0.04	0.05	0.10	0.36	0.47	0.96	1.42	0.16	0.17	0.	0.	4.01
	MS 0.09	0.04	0.04	0.01	0.03	0.12	0.22	0.14	0.30	0.71	1.21	1.53	3.21	2.44	0.14	0.18	0.	0.	10.40
	N 0.12	0.01	0.01	0.01	0.01	0.03	0.03	0.10	0.08	0.17	0.25	0.20	0.33	0.34	0.03	0.03	0.	0.	1.73
	U 0.32	0.21	0.06	0.	0.	0.01	0.05	0.09	0.39	0.79	0.99	0.40	0.86	1.21	0.10	0.12	0.	0.	5.60
19-24	VS 0.	0.	0.01	0.02	0.	0.01	0.01	0.01	0.	0.01	0.06	0.01	0.04	0.16	0.01	0.	0.	0.	0.34
	MS 0.05	0.01	0.01	0.	0.	0.05	0.11	0.06	0.23	0.52	0.85	0.49	1.76	1.42	0.01	0.02	0.	0.	5.59
	N 0.01	0.03	0.	0.	0.	0.01	0.01	0.04	0.08	0.18	0.19	0.08	0.23	0.30	0.01	0.01	0.	0.	1.17
	U 0.14	0.08	0.01	0.01	0.	0.	0.01	0.05	0.29	0.60	0.80	0.18	0.36	0.64	0.01	0.06	0.	0.	3.25
Over 24	VS 0.	0.	0.	0.	0.	0.	0.	0.	0.	0.01	0.01	0.	0.01	0.	0.	0.	0.	0.	0.02
	MS 0.01	0.	0.	0.	0.	0.	0.01	0.02	0.20	0.63	0.38	0.05	0.51	0.69	0.	0.	0.	0.	2.49
	N 0.01	0.	0.	0.	0.	0.	0.	0.01	0.07	0.25	0.13	0.03	0.08	0.17	0.	0.	0.	0.	0.75
	U 0.02	0.03	0.	0.	0.	0.	0.	0.01	0.16	0.86	0.66	0.19	0.19	0.52	0.01	0.01	0.	0.	2.66
Totals	VS 0.36	0.38	0.29	0.42	0.42	0.73	0.68	0.58	0.55	1.03	1.74	2.97	4.20	4.54	1.32	0.93	0.13	0.26	21.53
	MS 0.46	0.41	0.23	0.36	0.57	1.00	0.89	0.87	1.35	2.57	3.50	3.77	7.55	6.47	0.85	0.76	0.10	0.13	31.96
	N 0.36	0.41	0.21	0.35	0.39	0.59	0.33	0.44	0.45	0.87	0.86	0.73	1.26	1.71	0.41	0.49	0.19	0.09	10.16
	U 2.22	2.31	1.30	1.17	0.85	0.99	0.73	1.25	2.07	3.85	3.67	1.30	2.94	5.44	2.16	2.83	0.76	0.	36.35

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TABLE A-6
PERCENTAGE FREQUENCY DISTRIBUTION OF WIND SPEED AND
WIND DIRECTION AT 200-FOOT LEVEL VS. ATMOSPHERIC STABILITY

(January 1955 through July 1961)

SUMMER

		NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	N	Var.	Calm	Total
0 - 3	VS	0.07	0.06	0.07	0.07	0.07	0.14	0.05	0.05	0.03	0.10	0.13	0.17	0.16	0.14	0.10	0.10	0.03	0.11	1.67
	MS	0.04	0.05	0.06	0.03	0.13	0.16	0.03	0.10	0.04	0.07	0.07	0.14	0.14	0.16	0.03	0.12	0.09	0.09	1.54
	N	0.09	0.10	0.10	0.17	0.15	0.18	0.10	0.07	0.10	0.06	0.10	0.11	0.08	0.16	0.17	0.26	0.14	0.03	2.14
	U	0.31	0.62	0.29	0.47	0.44	0.41	0.18	0.41	0.18	0.44	0.23	0.24	0.24	0.36	0.40	0.75	1.03	0.	7.00
4 - 7	VS	0.16	0.16	0.10	0.14	0.13	0.23	0.12	0.07	0.12	0.20	0.22	0.34	0.75	0.56	0.29	0.15	0.01	0.	4.25
	MS	0.05	0.03	0.05	0.08	0.14	0.24	0.08	0.10	0.10	0.17	0.23	0.49	0.31	0.28	0.10	0.06	0.01	0.	2.52
	N	0.07	0.10	0.05	0.11	0.18	0.25	0.10	0.11	0.07	0.12	0.08	0.17	0.20	0.26	0.18	0.11	0.03	0.	2.20
	U	1.07	1.28	0.69	0.94	0.80	1.10	0.76	1.01	0.99	1.12	0.90	1.03	0.71	1.54	1.34	1.77	0.35	0.	17.91
8 -12	VS	0.16	0.16	0.10	0.15	0.08	0.07	0.18	0.08	0.05	0.14	0.54	1.10	1.93	1.49	0.41	0.12	0.	0.	6.77
	MS	0.10	0.12	0.03	0.07	0.08	0.10	0.06	0.03	0.05	0.16	0.48	1.19	1.68	0.64	0.10	0.12	0.	0.	5.03
	N	0.05	0.04	0.05	0.02	0.02	0.05	0.03	0.03	0.04	0.06	0.14	0.18	0.26	0.34	0.05	0.04	0.	0.	1.41
	U	0.82	0.58	0.18	0.13	0.11	0.16	0.14	0.16	0.44	1.26	1.04	0.59	0.78	2.03	0.61	0.75	0.01	0.	9.72
13-18	VS	0.05	0.04	0.03	0.04	0.	0.03	0.10	0.01	0.01	0.03	0.11	0.28	1.33	1.66	0.24	0.03	0.	0.	3.97
	MS	0.06	0.03	0.03	0.01	0.02	0.05	0.01	0.01	0.06	0.14	0.62	1.27	4.73	2.41	0.15	0.05	0.	0.	9.65
	N	0.01	0.01	0.01	0.	0.	0.03	0.03	0.	0.03	0.12	0.20	0.18	0.29	0.35	0.02	0.	0.	0.	1.29
	U	0.24	0.24	0.08	0.01	0.01	0.05	0.05	0.05	0.16	0.85	0.90	0.34	1.00	1.85	0.12	0.15	0.	0.	6.12
19-24	VS	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.02	0.	0.	0.02	0.26	0.01	0.	0.	0.	0.31
	MS	0.03	0.05	0.01	0.01	0.01	0.	0.01	0.01	0.01	0.08	0.18	0.29	2.67	3.15	0.05	0.01	0.	0.	6.58
	N	0.01	0.03	0.	0.01	0.	0.	0.	0.	0.02	0.02	0.13	0.05	0.35	0.52	0.	0.	0.	0.	1.14
	U	0.02	0.07	0.01	0.	0.	0.	0.01	0.	0.06	0.21	0.50	0.15	0.37	1.22	0.03	0.01	0.	0.	2.69
Over 24	VS	0.	0.01	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.01
	MS	0.	0.01	0.	0.	0.	0.01	0.01	0.	0.01	0.04	0.04	0.01	1.12	2.07	0.01	0.	0.	0.	3.33
	N	0.	0.01	0.	0.	0.	0.	0.	0.	0.	0.03	0.07	0.01	0.26	0.74	0.01	0.	0.	0.	1.13
	U	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.11	0.20	0.05	0.16	1.07	0.01	0.	0.	0.	1.59
Totals	VS	0.44	0.43	0.31	0.40	0.29	0.47	0.44	0.21	0.21	0.46	1.02	2.39	4.20	4.11	1.05	0.40	0.04	0.11	16.96
	MS	0.29	0.28	0.19	0.20	0.38	0.55	0.20	0.24	0.27	0.67	1.63	3.39	10.65	8.71	0.45	0.37	0.10	0.09	28.65
	N	0.23	0.29	0.22	0.31	0.35	0.50	0.25	0.22	0.27	0.40	0.72	0.71	1.44	2.37	0.43	0.41	0.16	0.03	9.30
	U	2.48	2.80	1.26	1.55	1.35	1.72	1.14	1.63	1.83	3.99	3.77	2.40	3.26	8.07	2.52	3.42	1.89	0.	45.09

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TABLE A-7
PERCENTAGE FREQUENCY DISTRIBUTION OF WIND SPEED AND
WIND DIRECTION AT 200-FOOT LEVEL VS. ATMOSPHERIC STABILITY

(January 1955 through July 1961)

FALL

		NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	N	Var.	Calm	Total
0 - 3	VS	0.28	0.31	0.23	0.40	0.41	0.67	0.30	0.34	0.31	0.31	0.34	0.67	0.60	0.96	0.73	0.66	0.22	1.05	8.77
	MS	0.22	0.31	0.22	0.32	0.50	0.56	0.21	0.24	0.15	0.16	0.26	0.15	0.28	0.46	0.39	0.55	0.14	0.83	5.93
	N	0.34	0.47	0.42	0.38	0.40	0.56	0.16	0.14	0.11	0.11	0.05	0.18	0.16	0.46	0.57	0.45	0.18	0.41	5.56
	U	0.50	1.16	0.60	0.66	0.60	0.30	0.15	0.12	0.05	0.08	0.06	0.11	0.12	0.34	0.54	0.78	0.30	0.03	6.50
4 - 7	VS	0.24	0.24	0.14	0.17	0.23	0.47	0.36	0.35	0.46	0.43	0.60	1.18	1.52	1.71	1.06	0.51	0.02	0.	9.69
	MS	0.21	0.11	0.14	0.15	0.21	0.47	0.28	0.19	0.18	0.17	0.19	0.29	0.62	0.79	0.54	0.45	0.01	0.	5.01
	N	0.09	0.11	0.06	0.08	0.24	0.35	0.13	0.10	0.08	0.07	0.05	0.12	0.18	0.64	0.34	0.11	0.01	0.	2.79
	U	0.73	0.76	0.36	0.39	0.49	0.45	0.21	0.16	0.21	0.18	0.18	0.09	0.28	0.91	1.01	1.33	0.04	0.	7.77
8 -12	VS	0.18	0.09	0.10	0.05	0.02	0.24	0.36	0.12	0.15	0.28	0.54	1.31	2.13	2.58	0.82	0.40	0.	0.	9.36
	MS	0.11	0.05	0.02	0.04	0.10	0.25	0.30	0.17	0.24	0.47	0.47	0.67	1.38	1.92	0.39	0.18	0.	0.	6.75
	N	0.04	0.08	0.01	0.03	0.02	0.06	0.08	0.07	0.05	0.07	0.11	0.05	0.18	0.70	0.24	0.08	0.	0.	1.87
	U	0.37	0.33	0.03	0.01	0.05	0.08	0.03	0.09	0.11	0.18	0.29	0.19	0.20	0.73	0.53	0.47	0.	0.	3.69
13-18	VS	0.03	0.02	0.01	0.03	0.	0.08	0.10	0.02	0.02	0.08	0.24	0.41	1.31	2.06	0.31	0.04	0.	0.	4.77
	MS	0.11	0.04	0.	0.01	0.01	0.10	0.15	0.15	0.33	0.75	0.79	0.82	1.72	2.13	0.19	0.15	0.	0.	7.44
	N	0.05	0.02	0.	0.01	0.	0.05	0.06	0.09	0.09	0.15	0.19	0.10	0.15	0.32	0.17	0.09	0.	0.	1.54
	U	0.28	0.10	0.02	0.	0.	0.07	0.03	0.02	0.13	0.29	0.41	0.15	0.35	0.56	0.08	0.11	0.	0.	2.60
19-24	VS	0.	0.	0.	0.	0.	0.	0.01	0.	0.01	0.01	0.03	0.03	0.06	0.28	0.	0.01	0.	0.	0.43
	MS	0.02	0.01	0.	0.	0.	0.02	0.05	0.20	0.29	0.63	0.51	0.24	0.66	1.24	0.03	0.01	0.	0.	3.89
	N	0.	0.01	0.	0.	0.	0.02	0.02	0.04	0.08	0.17	0.16	0.04	0.13	0.25	0.01	0.02	0.	0.	0.93
	U	0.04	0.05	0.01	0.	0.	0.	0.02	0.01	0.02	0.23	0.23	0.07	0.27	0.43	0.	0.02	0.	0.	1.38
Over 24	VS	0.	0.	0.	0.	0.	0.	0.02	0.02	0.	0.01	0.	0.	0.	0.	0.	0.	0.	0.	0.05
	MS	0.	0.	0.	0.	0.	0.	0.01	0.09	0.42	0.49	0.18	0.15	0.20	0.43	0.01	0.	0.	0.	1.97
	N	0.	0.	0.	0.	0.	0.	0.01	0.01	0.06	0.14	0.05	0.03	0.06	0.15	0.01	0.01	0.	0.	0.52
	U	0.	0.02	0.	0.	0.	0.	0.	0.02	0.25	0.09	0.07	0.06	0.24	0.02	0.	0.	0.	0.	0.77
Totals	VS	0.73	0.67	0.47	0.64	0.66	1.45	1.12	0.85	0.96	1.11	1.77	3.59	5.62	7.59	2.93	1.61	0.24	1.05	33.08
	MS	0.67	0.51	0.37	0.52	0.83	1.40	0.99	1.03	1.61	2.66	2.40	2.32	4.85	6.96	1.54	1.34	0.15	0.83	30.99
	N	0.52	0.70	0.49	0.50	0.67	1.03	0.45	0.44	0.48	0.71	0.61	0.53	0.87	2.51	1.35	0.76	0.19	0.41	13.22
	U	1.93	2.41	1.02	1.05	1.13	0.90	0.44	0.40	0.54	1.22	1.25	0.68	1.28	3.20	2.18	2.71	0.34	0.03	22.71

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TABLE A-8
PERCENTAGE FREQUENCY DISTRIBUTION OF WIND SPEED AND
WIND DIRECTION AT 200-FOOT LEVEL VS. ATMOSPHERIC STABILITY

(January 1955 through July 1961)

WINTER

		NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	N	Var.	Calm	Total
0 - 3	VS	0.20	0.33	0.19	0.28	0.30	0.59	0.34	0.42	0.27	0.33	0.31	0.62	0.48	0.69	0.55	0.54	0.24	0.90	7.59
	MS	0.38	0.51	0.41	0.47	0.89	0.99	0.57	0.43	0.24	0.38	0.25	0.52	0.67	0.93	0.91	0.81	0.18	1.69	11.23
	N	0.57	0.72	0.49	0.71	0.81	0.96	0.59	0.40	0.25	0.29	0.30	0.50	0.53	1.05	1.10	1.05	0.13	1.48	11.92
	U	0.21	0.16	0.23	0.17	0.08	0.07	0.01	0.03	0.06	0.03	0.01	0.04	0.08	0.15	0.22	0.18	0.03	0.05	1.32
4 - 7	VS	0.22	0.23	0.13	0.17	0.13	0.28	0.21	0.25	0.16	0.37	0.44	0.83	0.93	1.00	0.87	0.42	0.03	0.	6.68
	MS	0.29	0.26	0.24	0.22	0.27	0.49	0.28	0.19	0.24	0.31	0.29	0.58	1.01	1.73	1.10	0.60	0.03	0.	8.12
	N	0.19	0.21	0.21	0.11	0.12	0.15	0.17	0.11	0.06	0.15	0.22	0.28	0.63	1.53	0.67	0.30	0.01	0.	5.09
	U	0.22	0.15	0.11	0.10	0.04	0.07	0.05	0.03	0.03	0.01	0.01	0.02	0.11	0.60	0.40	0.43	0.01	0.	2.39
8 -12	VS	0.06	0.03	0.03	0.02	0.04	0.15	0.11	0.06	0.10	0.27	0.53	0.74	1.25	1.91	0.58	0.10	0.	0.	5.97
	MS	0.10	0.07	0.02	0.08	0.08	0.24	0.17	0.12	0.24	0.28	0.32	0.60	1.73	3.71	0.67	0.17	0.	0.	8.60
	N	0.08	0.06	0.03	0.02	0.03	0.08	0.05	0.02	0.05	0.05	0.08	0.10	0.66	1.94	0.30	0.14	0.	0.	3.72
	U	0.17	0.04	0.03	0.	0.01	0.01	0.02	0.02	0.07	0.04	0.09	0.07	0.15	1.03	0.27	0.17	0.	0.	2.18
13-18	VS	0.01	0.04	0.	0.	0.	0.03	0.05	0.03	0.05	0.23	0.26	0.46	0.62	1.50	0.18	0.03	0.	0.	3.50
	MS	0.07	0.01	0.	0.01	0.02	0.08	0.14	0.28	0.35	0.83	0.72	0.61	1.40	3.82	0.24	0.09	0.	0.	8.67
	N	0.06	0.	0.	0.	0.01	0.	0.01	0.01	0.08	0.15	0.15	0.08	0.34	1.02	0.07	0.06	0.	0.	2.05
	U	0.14	0.01	0.	0.	0.	0.	0.01	0.01	0.06	0.12	0.20	0.12	0.10	0.31	0.08	0.10	0.	0.	1.28
19-24	VS	0.	0.	0.	0.	0.	0.02	0.01	0.	0.05	0.07	0.03	0.06	0.06	0.13	0.	0.	0.	0.	0.43
	MS	0.02	0.04	0.	0.	0.	0.01	0.10	0.12	0.41	1.03	0.44	0.34	0.26	0.89	0.08	0.01	0.	0.	3.75
	N	0.02	0.	0.	0.	0.01	0.	0.01	0.02	0.10	0.10	0.08	0.03	0.01	0.13	0.02	0.02	0.	0.	0.57
	U	0.02	0.01	0.	0.	0.	0.01	0.	0.	0.02	0.13	0.17	0.03	0.02	0.08	0.	0.03	0.	0.	0.52
Over 24	VS	0.	0.	0.	0.	0.	0.01	0.01	0.	0.03	0.04	0.01	0.	0.	0.	0.	0.	0.	0.	0.10
	MS	0.01	0.	0.	0.	0.	0.03	0.06	0.22	0.72	1.23	0.37	0.13	0.06	0.10	0.02	0.01	0.	0.	2.96
	N	0.	0.	0.	0.	0.	0.	0.	0.05	0.10	0.19	0.03	0.01	0.01	0.01	0.	0.01	0.	0.	0.41
	U	0.01	0.	0.	0.	0.	0.	0.	0.01	0.08	0.23	0.07	0.01	0.01	0.05	0.	0.	0.	0.	0.45
Totals	VS	0.49	0.64	0.36	0.48	0.47	1.08	0.74	0.76	0.65	1.30	1.57	2.71	3.33	5.22	2.18	1.10	0.28	0.90	24.26
	MS	0.87	0.89	0.67	0.78	1.26	1.85	1.32	1.35	2.20	4.06	2.39	2.77	5.12	11.20	3.02	1.69	0.21	1.69	43.33
	N	0.92	0.98	0.74	0.84	0.98	1.19	0.83	0.61	0.65	0.93	0.85	1.01	2.18	5.69	2.17	1.58	0.14	1.48	23.76
	U	0.76	0.38	0.37	0.27	0.13	0.15	0.09	0.10	0.32	0.58	0.54	0.29	0.46	2.21	0.98	0.92	0.04	0.05	8.64

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TABLE A-9
PERCENTAGE FREQUENCY DISTRIBUTION OF WIND SPEED AND
WIND DIRECTION AT 200-FOOT-LEVEL VS. ATMOSPHERIC STABILITY

(January 1955 through July 1961)

ANNUAL

		NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	N	Var.	Calm	Total
0 - 3	VS	0.16	0.20	0.14	0.22	0.24	0.41	0.21	0.24	0.20	0.25	0.24	0.46	0.38	0.53	0.41	0.37	0.15	0.56	5.37
	MS	0.19	0.25	0.19	0.22	0.44	0.48	0.22	0.22	0.13	0.17	0.16	0.23	0.29	0.40	0.37	0.41	0.12	0.67	5.17
	N	0.27	0.38	0.28	0.36	0.40	0.47	0.22	0.18	0.13	0.14	0.12	0.23	0.22	0.44	0.50	0.50	0.15	0.50	5.47
	U	0.38	0.65	0.40	0.45	0.36	0.26	0.11	0.22	0.12	0.18	0.10	0.14	0.16	0.30	0.40	0.64	0.49	0.02	5.38
4 - 7	VS	0.18	0.19	0.11	0.15	0.16	0.31	0.22	0.22	0.21	0.35	0.44	0.93	1.03	1.04	0.65	0.35	0.02	0.	6.57
	MS	0.16	0.12	0.12	0.16	0.22	0.40	0.22	0.18	0.18	0.22	0.26	0.46	0.58	0.81	0.49	0.33	0.01	0.	4.92
	N	0.10	0.13	0.10	0.10	0.15	0.25	0.13	0.11	0.07	0.10	0.12	0.18	0.30	0.66	0.31	0.16	0.02	0.	2.98
	U	0.70	0.77	0.43	0.50	0.43	0.56	0.35	0.47	0.46	0.49	0.38	0.39	0.42	1.09	0.97	1.20	0.28	0.	9.88
8 -12	VS	0.12	0.10	0.08	0.09	0.05	0.14	0.20	0.10	0.11	0.23	0.55	1.07	1.80	1.88	0.55	0.20	0.	0.	7.28
	MS	0.11	0.09	0.02	0.07	0.07	0.19	0.19	0.15	0.21	0.33	0.48	0.90	1.62	1.89	0.35	0.16	0.	0.	6.84
	N	0.06	0.05	0.03	0.03	0.03	0.06	0.06	0.05	0.06	0.08	0.12	0.12	0.36	0.87	0.17	0.09	0.	0.	2.26
	U	0.47	0.35	0.11	0.06	0.07	0.09	0.10	0.12	0.28	0.59	0.54	0.33	0.49	1.33	0.47	0.49	0.00	0.	5.91
13-18	VS	0.04	0.03	0.02	0.02	0.	0.05	0.08	0.02	0.03	0.11	0.25	0.41	1.05	1.64	0.22	0.07	0.	0.	4.04
	MS	0.08	0.03	0.02	0.01	0.02	0.09	0.13	0.14	0.26	0.60	0.84	1.07	2.81	2.71	0.18	0.12	0.	0.	9.10
	N	0.06	0.01	0.01	0.01	0.00	0.03	0.03	0.05	0.07	0.15	0.20	0.14	0.28	0.51	0.07	0.04	0.	0.	1.66
	U	0.25	0.15	0.04	0.00	0.00	0.03	0.03	0.04	0.19	0.53	0.64	0.26	0.59	1.00	0.10	0.12	0.	0.	3.97
19-24	VS	0.	0.	0.00	0.01	0.	0.01	0.01	0.00	0.01	0.02	0.03	0.03	0.04	0.20	0.00	0.00	0.	0.	0.37
	MS	0.03	0.03	0.01	0.00	0.00	0.02	0.07	0.09	0.23	0.56	0.50	0.35	1.37	1.69	0.04	0.01	0.	0.	5.00
	N	0.01	0.02	0.	0.00	0.00	0.01	0.01	0.02	0.07	0.12	0.14	0.05	0.18	0.30	0.01	0.01	0.	0.	0.96
	U	0.06	0.05	0.01	0.00	0.	0.00	0.01	0.01	0.10	0.30	0.44	0.11	0.26	0.60	0.01	0.03	0.	0.	2.00
Over 24	VS	0.	0.00	0.	0.	0.	0.00	0.00	0.01	0.01	0.01	0.01	0.	0.00	0.	0.	0.	0.	0.	0.04
	MS	0.00	0.00	0.	0.	0.	0.01	0.02	0.08	0.33	0.60	0.24	0.08	0.48	0.84	0.01	0.00	0.	0.	2.70
	N	0.00	0.00	0.	0.	0.	0.	0.00	0.02	0.06	0.15	0.07	0.02	0.10	0.27	0.00	0.01	0.	0.	0.71
	U	0.01	0.01	0.	0.	0.	0.	0.	0.01	0.07	0.37	0.27	0.08	0.11	0.48	0.01	0.00	0.	0.	1.41
Totals	VS	0.50	0.52	0.35	0.48	0.45	0.91	0.73	0.59	0.58	0.97	1.52	2.90	4.30	5.29	1.83	0.99	0.17	0.56	23.07
	MS	0.57	0.52	0.36	0.46	0.75	1.19	0.85	0.87	1.35	2.48	2.49	3.09	7.15	8.34	1.45	1.03	0.14	0.67	33.74
	N	0.50	0.59	0.41	0.49	0.59	0.82	0.46	0.43	0.46	0.73	0.77	0.75	1.45	3.06	1.07	0.61	0.17	0.50	14.04
	U	1.85	1.97	0.99	1.02	0.86	0.95	0.61	0.87	1.22	2.47	2.37	1.32	2.02	4.80	1.96	2.48	0.77	0.02	28.55

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FIGURE A-2

FIGURE A-3

FIGURE A-4

6. ECOLOGY

Both migratory fowl and marine life in the river represent possible routes by which contamination released in the HAPO Reservation could be transported to the environs with subsequent human ingestion.

Of these, the marine life represent the most significant practical problem. In particular, the uptake of P^{32} by river algae and later concentration by fish in the river is the subject of careful and continued surveillance. The waste from the NPR will be managed so that the problem is not aggravated.

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APPENDIX B

THE REACTOR COMPLEX

1. LAYOUT OF THE REACTOR AREA

Figure B-1 is a plot plan of the reactor area. The function of the various facilities and building are described below:

The 105 and 109-N Buildings are the "heart" of the reactor complex. The 105-N Building houses the reactor, a central control room for the plant and supporting facilities such as offices, a maintenance shop, unirradiated and irradiated fuel storage areas, ventilation systems and the reactor gas atmosphere circulation and purification system.

The 109-N Building houses the steam generators, primary pumps, dump condensers, the primary system pressurizer, the graphite moderator heat exchangers and pumps, the primary system decontamination facilities, condensate return pumps, and ventilation systems for the 109-N Building.

The 181-N Building at the river shoreline is the river pumphouse. It houses the four normal river pumps which supply river water to the dump condensers, pump drive condensers, turbine generator condenser, the water treatment plant and various miscellaneous services. It also houses the two emergency river pumps which will supply water for reactor cooling and for fog sprays in case of emergency.

1908-N is the seal well and outfall structure which returns cooling water to the Columbia River.

The 183-N Building houses the reactor filter plant. Raw river water is treated and filtered using essentially the same proven methods employed at the eight single-pass cooled production reactors. Filtered water flows

to a clearwell and is then pumped to the filtered water storage tanks. Filtered water is used for some normal miscellaneous cooling requirements, may be used during reactor outages for single-pass reactor cooling and is used as the influent to the demineralization plant.

The 163-N Building is adjacent to the 183-N Building and houses the demineralisation plant. Here filtered water is demineralized, degassified and pumped to the demineralized storage tanks. The building also houses resin regeneration equipment, a control center for the 183-N and 163-N Buildings, plus a water treatment control laboratory.

The 182-N Building is called the High-Lift Pump House. It houses pumps which inject water into the primary and secondary coolant loop, pumps which supply cooling water to the rupture monitor and the moderator cooling system heat exchangers. It also houses diesel-driven emergency pumps which draw water from the emergency storage tank for reactor cooling and other diesel-driven pumps which supply fog sprays and backup cooling for the moderator cooling system, control rods, and shields. Finally, the 182-N Building houses the potable water supply tank, the fire supply systems, and air compressors and receivers.

The 184-N Building is called the Standby Powerhouse. The name in a sense is a misnomer, because the building houses the local 15 MW turbine generator which normally produces half the required electrical supply as well as an oil-fired boiler which can supply steam to drive the primary pumps for circulation of coolant at up to half flow, plus the entire steam requirements of the turbine generator. The 184-N Building also houses chemical mixing and injection equipment.

166-N is the Fuel Oil Storage Tank for the oil fired boiler. The adjacent diesel oil storage tank farm for the emergency diesel driven pumps is termed 1715-N.

Radioactive dummy fuel elements are sent by an underground conveyer system to the 1303-N Radioactive Dummy Burial Facility.

Spent decontamination solutions are sent to a storage facility, 1310-N. While Figure B-1 shows two tanks, at the time of this writing it is not firm whether one or two tanks will be provided.

The 1614-N Radiation Monitoring Building will be used to monitor the activity of the decontamination wastes. Other radioactive wastes are routed to 1301-N, a crib, and sanitary wastes are vented to the tile field east of the 1310-N facility.

The 151-N Building is the 230 KV electrical substation receiving power from the BPA. Voltage is reduced to 13.8 KV and power is routed to 153-N Building which also obtains power at 13.8 KV from the turbine generator. Power at 13.8 KV is supplied to river pumps and to 13.8/4.16 KV transformers.

The 117-N Building houses air filters used normally for filtering air from potentially contaminated areas in the 105-N Building. These filters will also be used after a severe accident involving significant fission product release. Effluent from the filters and some of the other parts of the 105-N Building is exhausted to the 116-N Air Stack.

In the event that the coolant in the primary coolant system is dumped, it will be quenched in a dump tank. Figure B-1 shows the tentative location of the dump tank to be a short distance north of the seal wells.

Other miscellaneous facilities and buildings include 108-N, a Chemical Unloading Facility; 1716-N, a Garage; 1734-N, a Storage Building for com-

pressed gases and 1704-N an Administration Building.

2. THE 105-N AND 109-N BUILDINGS

The 105-N and 109-N Buildings at the 0'-0" level are shown in Figure 3.10 of Volume 1. The 105-N Building is shown in greater detail in Figures B-2, B-3, B-4, and B-5. The 109-N Building is shown in greater detail in Figures B-6, B-7, and B-8.

Figure B-2 shows plan views at 0'-0", plus 28'-3", and plus 60'-0" levels. Similarly, Figure B-3 shows plan views at the minus 8'-0" level, the plus 14'-6" level, at the plus 40'-0" level and at other levels below 0'-0". These figures are ventilation schematic diagrams and show the various ventilation or confinement zones as described in Section 7 of Volume 1. Figures B-4 and B-5, also drawings made of the 105-N Building are elevation views. Figure B-4 is a building section taken at a point to the right of the reactor and facing towards the right side. Figure B-5 is a section parallel to Figure B-4, facing the same direction but starting at the reactor mid plane.

Figures B-6 and B-7 are plan views of the 109-N Building. Figure B-6 shows the southeast half of the building at the 0'-0" level. Figure B-7 shows the entire building at the minus 16'-0" level. Figure B-8 is a section of the 109-N Building looking toward the common wall between the 105-N and 109-N Buildings.

The reactor occupies a central location in the 105-N Building. Immediately to the left* and right of the reactor are the inner and outer rod rooms which house the horizontal control and safety rods, rod drives and associated equipment.

*All references to left and right are made from the orientation of one facing the reactor "front" or charging face.

The space immediately below the reactor is utilized for Ball 3X "draining", cleaning and transfer equipment and for reactor gas distribution and collection piping to the inlet and from the outlet reactor gas plenum space.

Ball 3X filling equipment is housed in the floor immediately above the reactor. A flexible hose connects the fill hopper to individual Ball 3X hoppers located within the top shield.

Reactor gas treatment and circulation equipment is located in a wing of the building to the front and right of the reactor.

Elevators giving access to the charge and discharge reactor face are located immediately to the front and rear of the reactor. A third very large elevator is located to the front of the charge or "C" elevator and will carry the fuel charging machine.

The main control room for the entire reactor site is located in a room to the left and rear of the reactor. This center will be used to control the nuclear process and the heat dissipation equipment in the 109-N Building plus other equipment throughout the complex vital to reactor safety.

All ventilation air to the 105-N Building except for the control room is supplied by blowers in a fan room to the left and front of the reactor. The air ventilation system is zoned as described in Section 7 of Volume 1. Ventilation air for the control room is supplied by equipment in a special room to the left of the control room.

Three exhaust fan rooms are provided for discharging air from the building. Zone II air is exhausted from a fan room below and to the right of the reactor. The Zone I exhaust fan room is above and to the right of the reactor and the Zone III exhaust fan room is above and to the left of the reactor. Zone III air exhaust can be routed to the Zone II exhaust fans if desired as explained in Section 7 of Volume 1.

The remaining space in the 105-N Building houses shops, storage places, instrument and electrical rooms, plus various other supporting facilities.

The 109-N Building is basically less complex than the 105-N Building. The Pipe Gallery is located in the 109-N Building immediately adjacent to the common wall separating the two buildings. The primary coolant piping connecting the five steam generator cells to the reactor is housed in this pipe gallery. The pressurizer for the primary coolant loop, the surge tank for the moderator coolant and an emergency storage water tank (for filling confinement water seals) are located in a "penthouse" above the main 109-N Building roof.

The five steam generator cells each house two steam generators, a primary circulating pump and connecting primary coolant system, condensate return and steam piping. An auxiliary system cell is provided which houses pumps and heat exchangers for the moderator coolant system.

The Cell Access Corridor connects the entrances of the six cells, on the ends away from the pipe gallery, at the -22'-0" level.

Pump drive turbines are located on pedestals in the turbine drive bay. This bay is adjacent to the cells and above the Cell Access Corridor and the adjoining Turbine Drive Bay basement. The system dump condensers and the secondary system surge tanks are located along the outside wall of the Turbine Drive Bay. The basement houses condensate return pumps, the condenser for the turbines drive plus auxiliary equipment.

A "radioactive chemical" laboratory and radioactive sampling and monitoring room is located between the turbine drive bay and the auxiliary system cell.

A wing on the east side of the building, the Service Bay, houses facilities for primary coolant system decontamination, heating and ventilation equipment, shop areas, office space and miscellaneous supporting facilities.

3. THE REACTOR

The reactor consists of a graphite stack pierced longitudinally (front to rear) by process tubes, vertically by the Ball 3X Safety System Channels, and transversely by the control and safety rod channels and by the graphite coolant channels. The moderator is enclosed by both thermal and biological shields.

Figure B-9 is an isometric drawing showing most of the details of the reactor block. Figures B-10, B-11, and B-12 are a plan view, a longitudinal section and a transverse section of the reactor block, respectively.

3.1 The Moderator

The moderator is shown in some detail in Figure B-9 through B-12. In addition, Figure B-13 is an expanded isometric drawing of a portion of the moderator.

The figures show the open array of the core graphite which permits the venting of steam should a process tube fail. Figure B-13 shows how the tube blocks and filler blocks are notched to provide structural stability and precise positioning. The tube blocks are notched 1/4 inch deep on the top and bottom and the filler blocks 1-1/4 inches on the top and bottom. These notches position tube and filler blocks two inches apart. To prevent irradiation induced contraction of the graphite bars in the active core being totally cumulative either side to side or front to rear the interlocking is not complete in two vertical planes. One plane is

midway between the front and rear of the stack. The other is midway between the two sides.

Shown, but not identified, in the Figure B-3 are the trunnion blocks provided at each junction between tube blocks. These trunnion blocks provide a smaller I. D. bore than the tube blocks. Hence, the tube will be supported by the trunnion blocks. Moderator cooling tubes will be positioned over and under each except the front and rear most trunnion blocks, reducing the process tubes outer surface temperatures at the point of contact. Moderator cooling tubes are also positioned midway between adjacent trunnion blocks. Hence, if distortion ultimately causes tube to contact the tube block, it will again contact cooled graphite.

Open channels have been provided for the horizontal control and safety rods. Moderator cooling tubes have been provided in the filler blocks on both sides of the horizontal rod channels except for the front most and rear most columns of rods. For these columns, cooling tubes are supplied only on the side nearer the reactor center; heat generation rates will be lower at the front and rear of the reactor, reducing the need for cooling.

Nine open channels through the moderator stack are provided for graphite displacement measurements to monitor the distortion of the stack. Six of these are through the top reflector just above the active core; three run from front to rear and three from side to side. The other three open channels pass through the active core from side to side in a vertical plane half way between the front and rear. The channels are at elevation of $1/4$, $1/2$ and $3/4$ the height of the active core.

These last three side to side channels can also be used as positions for irradiating test samples of graphite.

There are seven replaceable thermocouple stringers which run from side to side through the core. Each stringer has five thermocouples along its length. There are also 98 non-replaceable thermocouples installed in the active core. Eighty-four of these are distributed in cooled filler blocks at five elevations in three vertical support planes parallel to the front face of the reactor. These planes are at the quarter points from front to rear. The other 14 non-replaceable thermocouples are in cooled and uncooled "cells" typical of the packing in filler block zones.

There will be 25 flux monitors positioned in a three dimensional array in moderator cooling tubes through the active zone. Finally, traveling wire flux monitors are provided which run from the rear to the front through the active core.

Ball 3X channels have been provided with graphite liners. These liners will prevent spillage of the balls into the open graphite structure.

3.2 Shielding

The reactor shield is composed of two parts. The inner thermal shield is water cooled and protects the outer biological shield from excessive temperature resulting from thermal neutron radiation. The arrangement of the shield is shown in Figures B-11 and B-12. Dimension of the shields are included in Table 3.1, Page 3.11 of Volume 1 to this Hazards Review.

In addition to the reactor shielding, the walls enclosing Zone I in both the 105-N and 109-N Buildings are designed to be thick enough to reduce radiation levels to less than one mrem/hour in areas normally occupied. These walls are typically on the order of five feet in thickness.

3.3 The Horizontal Control and Safety Rods And The Ball 3X System

There are a total of 87 horizontal control and safety rods. Forty-one of these will enter the reactor from the left side; forty-six enter from the right side.

The arrangement of the left side rods is shown in Figure B-14. Two "rod rooms" are provided to house the rods and their associated equipment. The inner rod rooms are located immediately adjacent to the reactor. When fully inserted, the rods will extend 28'-10" beyond the shielding into the process unit. When fully withdrawn the rods will extend into the reflector four inches past the shielding.

Even when fully withdrawn, the irradiated sections of the rods will be totally within the inner rod room. The rods pass through the inner rod room enclosed in sleeves. These sleeves minimize contamination spread. Reactor gas is prevented from leaking into the room by dynamic gas seals between rod and sleeve at the outside wall of the inner rod room and by static seals between the sleeve and reactor wall.

The Ball 3X hoppers are located in the top primary shield. Balls dropped in the reactor are drained out through valves at the bottom of the reactor, cleaned, returned to a space above the reactor with a skip hoist and relocated onto the individual ball hoppers.

Further details on the equipment and control of the rods and ball systems are presented in Appendix D of this volume, "Reactor Control and Safety Systems".

3.4 The Process Tube Assembly

The inlet process tube assembly is shown in Figure 3.12 of Volume 1 and in more detail in Figure B-15 of this volume. The process

tube is sealed to the nozzle with a rolled joint. Three grooves are machined in the nozzle body, the tube inserted and rolled joint. Three grooves are machined in the nozzle body, the tube inserted and rolled into the grooves obtaining a 12 per cent tube-wall reduction after clearance elimination. This mechanical joint has been the object of intensive testing program and has demonstrated the requisite mechanical strength plus positive sealing as fabricated even after extensive thermal cycling. The front nozzle is connected to a tube sleeve by a threaded connector. This tube sleeve extends into the graphite reflector protecting the outer surface of the tubes.

A thimble enclosing the inlet nozzle tube sleeve joint is bolted to front shield plate. Seals are provided between thimble and the shield plate and between the thimble and the nozzle to prevent escape of the reactor gas. The seal between the thimble and nozzle is a dynamic seal, i.e., the nozzle is free to move in the seal, accommodating thermal expansion of the process tubes.

The outlet process tube shown in Figure B-16 assembly is quite similar to the inlet. The main difference between the two is the absence of a thimble on the rear face and the accompanying absence of a dynamic gas seal. The nozzle itself is bolted to the rear shield plate preventing movement of the rear assembly. All thermal expansion will be accommodated by the inlet assembly.

3.5 Process Tube Flow Distribution

Coolant is supplied to the process tubes via the inlet nozzle from the inlet coolant supply header through the individual inlet connectors. Coolant discharged from the tubes is returned to outlet

coolant headers through a similar system. Figure B-17 shows the outlet arrangement.

Each inlet connector is fitted with a venturi meter and a butterfly valve both located near the inlet coolant supply header. The flow can be reduced, but not stopped by the butterfly valve. Instrument lines connecting to a point upstream and at the throat of the venturi extend to instrument rooms (one room for the venturis on the right side and one room for the venturis on the left side of the reactor). There the pressure signal is connected to an electronic signal and transmitted to the control room. In addition to indicating flow, the signal from each venturi is included in the safety system to scram the reactor in case of either low flow (indicating a failure of coolant pumps, a line failure upstream of the venturi or a flow blockage downstream of the venturi) or high flow (indicating a line failure or process tube failure downstream of the venturi). Details are supplied in Appendix D.

The butterfly valves are provided for two different purposes. First, they provide for adjustable flow trim to match tube flow to tube power, minimizes the variation of tube outlet temperatures, and secondly, provide throttling for fuel discharge operations during reactor outages. No provision is made for adjustment of the fine, flow-trim except during outages.

The coolant outlet temperature of each tube is monitored by a strap-on RTD (resistance temperature detector) affixed to each outlet connector. A zone temperature monitor consisting of the outlet temperature from 109 tubes is also provided. This system utilizes the more rapidly responding immersion type RTD's for obtaining a temperature

signal. The zone temperature monitor is included in the setback and safety circuits. It will cause a reactor power setback should the outlet temperature of any two of the 109 tubes exceed limit or a reactor scram should the outlet temperature of any three of the 109 tubes exceed limits. A single tube exceeding limits will be enunciated in the control room. Details of the circuits are presented in Appendix D.

A sample of coolant is withdrawn from each tube, transmitted to the inner rod room, cooled and thence transmitted to a rupture monitor room. Here the relative gross gamma activity is monitored to indicate the occurrence of a fuel failure in any process tube. Twenty-four gamma spectrometers are also provided to monitor individual tube samples for specific gamma energies (associated with characteristic fission product emission).

Should a fuel failure signal occur the coolant flow from a connector may be routed to a diversion system to minimize the spread of contamination to the main primary coolant system as the reactor is shutdown and the coolant temperature reduced.

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100-N	REACTOR
100-N	CHEMICAL UNLOADING
100-N	HEAT EXCHANGER
100-N	AIR SINK
107-N	AIR FILTERS
101-N	230 KV ELECTRICAL SUBSTATION
103-N	4 MW ELECTRICAL SUBSTATION
103-N	REHYDRATION PLANT
104-N	FUEL OIL STORAGE TANK
101-N	WATER WHEEL PUMP HOUSE
102-N	WATER LIFT PUMP HOUSE
103-N	WATER FILTER PLANT
104-N	STEAM-67 POWERHOUSE
1303-N	ARMACHINE BUNKER
1310-N	ARMACHINE CHEMICAL WASTE TREATMENT
1014-N	ARMACHINE MOUNTING BUILDING
1700-N	ADMINISTRATION BLDG.
1710-N	GARAGE FACILITY
1734-N	STORAGE BUILDING
1000-N	CENTRAL STRUCTURE

Figure B-1
100-N Plot Plan

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ZONE IV

RM NO.	ROOM OR CORRIDOR	SUP. CFM	EXHAUST CFM	RM NO.	ROOM OR CORRIDOR	SUP. CFM	EXHAUST CFM	RM NO.	ROOM OR CORRIDOR	SUP. CFM	EXHAUST CFM
101	CARPENTER SHOP	2,300	ROOF EXH.	134	CONFERENCE RM.	1,600	CORR. 8	140	INT. SHOP	2,700	CORR. 8
102	PAINT SHOP	1,600	ROOF EXH.	135	OFFICE	350	CORR. 8	141	OFFICE	350	CORR. 9
103	RIGGING LOFT	1,700	ROOF EXH.	136	"	350	CORR. 8	142	"	350	CORR. 9
104	MACH. & MILL WRIGHT	1,500	"	137	ELECT. SHOP	2,700	CORR. 13	143	AIR COND. EQUIP.	4,500	RM. 144
105	FABRICATION AREA	500	CARPENTER SHOP								
106	WELDING SHOP	1,000	ROOF EXH.								
107	PIPE SHOP	1,500	"								
108	TOOL CRIB	500	"								
109	WAREHOUSE	20,000	RE-CIRCULATED								
110	SUPPLY FAN RM.	10,000	"								
117	SOILED CLOTHES	350	ROOF EXH.								
118	MISC. STG.	200	CORR. 13								
120	JAN. CLOSET	150	RM. 122								
122	W. LOUNGE	350	ROOF EXH.								
127	7 OFFS @ 350 CFM	2,450	CORR. 13								
130	OFFICE	450	"								
131	CHANGE RM.	3,900	ROOF EXH.								
132	LUNCH RM.	3,700	"								
133	B.R.F.	550	CORR. 13								

PLAN AT ELEV. +20'-3"

RM NO.	ROOM OR CORRIDOR	SUP. CFM	EXHAUST CFM	RM NO.	ROOM OR CORRIDOR	SUP. CFM	EXHAUST CFM
166	CHANGE RM.	2,700	ROOF EXH.	1600	C.A.A. C-12	1,600	C.A.A. C-12
167	READY RM.	850	CORR. 13	900	CORR. 13	900	CORR. 13
168	PER. DECON.	500	ROOF EXH.	1	COMPRESSOR RM.	1,500	RE-CIRC.
169	RMU	300	TO PER. DECON.	1,200	CORR. 13	1,200	CORR. 13
C-7	"	1,300	CORR. 8	3,000	WHSE. RET. DUCT	3,000	WHSE. RET. DUCT
C-8	"	2,000	CORR. 13	650	CORR. 13	650	CORR. 13

TOTAL ZONE IV 15,000 CFM

PLAN AT ELEV. +20'-3"

RM NO.	ROOM OR CORRIDOR	SUPPLY CFM	EXHAUST CFM
403	INST. RM.	1,500	2,500
170	OUTER ROD RM.	5,000	TO INNER ROD RM.
27-A	CORR.	3,000	TO 'W' ELEV. & RM. 403
402	'R' ELEV. ENTRY	1,500	'H' ELEV. & RM. 401
401	ELECT. RM.	1,750	2,500
24	CORR.	1,500	1,500
404	INST. RM.	1,500	2,000
27-B	CORR.	2,000	TO 'W' CL. & RM. 404
176	OUTER ROD RM.	5,000	TO INNER ROD RM.
TOTAL		22,750	

TOTAL 2,000

ROOM OR CORRIDOR	SUPPLY C.F.M.	EXHAUST C.F.M.	ROOM OR CORRIDOR	SUPPLY C.F.M.	EXHAUST C.F.M.	ROOM OR CORRIDOR	SUPPLY C.F.M.	EXHAUST C.F.M.
WITCHGEAR RM.	1200	ROOF EXH.	157 OFFICE	350	CORR. 10	161 CHANGE RM.	1200	RM 166
REFRIG. RM.	400	CORR. 9	158 "	350	CORR. 10	162 ELECTRONICS	1100	CORR. 10
FILES 350 EA	700	CORR. 9	159 "	400	CORR. 10	163 TOILET & SHR.	900	ROOF EXH.
UNCL. RM.	1950	RM 163	160 "	350	CORR. 10	165 OFFICE	400	CORR. 10

SUMMARY ZONES II AND III

LEVEL	TOTAL C.F.M. SUPPLY
BELOW 0'-0"	28,800
0'-0" TO +14'-0"	53,150
14'-0" TO 28'-3"	14,500
28'-3" TO 40'-0"	22,750
40'-0" TO 60'-0"	19,000
60'-0" ABV.	2,000
TOTAL	140,200

DEFINITION OF ZONES

- ZONE I** - PRIMARY RADIATION AREAS
5 P.S.I. CONFINEMENT
OPER. PRESS. -0.45 IN. W.G.
- ZONE II** - SECONDARY RADIATION AREAS
0.5 P.S.I. CONFINEMENT
OPER. PRESS. -0.30 IN. W.G.
- ZONE III** - NORMAL ACCESS AREAS
0.2 P.S.I. BUFFER
OPER. PRESS. -0.15 IN. W.G.
- ZONE IV** - UNLIMITED ACCESS AREAS
ATMOS. PRESS.
- ZONE V** - WARRANTED ACCESS AREAS
ATMOS. PRESS. (CONTROL RM.)

LEGEND

- SUPPLY
- EXHAUST
- ⊗ RADIAL INLET DAMPER
MOTOR OPERATED
- ⊗ SHUT-OFF DAMPER
MOTOR OPERATED
- ⊗ CONFINEMENT VALVE
- ⊗ ROOF EXHAUSTER
- ⊗ MANUAL DAMPER
- ⊗ UNIT HEATER
- ⊗ EVAP. COOLER
- ⊗ INDICATE ZONE I AREAS (5 P.S.I.)
- ⊗ BOOSTER HEATER
- ⊗ INDICATES ZONE II AREAS (0.5 P.S.I.)

NOTE: -
UNLESS OTHERWISE NOTED SPACES VENTILATED ARE ZONE III.

ROOM OR CORR.	SUPPLY C.F.M.	EXHAUST (C.F.M.)
C-114 CORR.	1,850	TO "W" ELEV. SHAFT & RMS.
C-114 CORR.	1,250	111, 112, 114, 115 & 116
112 ST'G.	250	350
111 SOILED CLO. ST'G.	300	500
114 HOT SHOP	750	1,300
115 MAT'L. DECON.	750	1,300
116 BLUE TOOL RM.	750	1,300
C-146 CORR.	1,250	TO RM. 172
172 R' ELEV. ENTRY	750	2,750
174 BLUE TOOL RM.	500	TO "R" ELEV. ENTRY
173 READY RM.	250	" " " "
C-7 CORR. RIGHT	3,000	TO EMERG. ACCESS
C-7 CORR. LEFT	3,000	" " " "
192 STORAGE BASIN	5,000	TO TRANSFER AREA
193 TRANSFER AREA	3,000	TO ATMOS. (NORMAL TO STORAGE BASIN & WAG)
C-22 CORR.	3,000	TO STORAGE BASIN & CHANGE RM.
C-18 CORR.	2,500	TO "W" ELEV. SHAFT & "N" ELEV. SHAFT
187 BLUE TOOL RM.	1,000	1,500
191 DUMMY DECON.	2,500	3,500 TO ZONE II
190 CASK ST'G.	2,000	TO DUMMY DECON. & PERM. HOT ST'G.
189 PERM. HOT ST'G.	1,500	2,500
188 TOILET & SHR. & CHANGE RM.	1,500	2,000
186 GAS INST. RM.	2,500	3,750
185 REFRIG. EQUIP.	2,500	3,750
184 DRYER RM. NO. I	2,500	TO GAS TUNNEL, ZONE II
183 DRYER RM. NO. II	2,500	" " " "
181 METAL PREP.	4,000	TO "W" ELEV. SHAFT
182 DUMMY ST'G.	3,000	" " " "
C-20 GAS CORR.	2,500	TO GAS INST. & REFRIG. EQUIP. RMS.
TOTAL	53,150

FIGURE B-1
PLAN VIEW 100-N BUILDING
VENTILATION SCHEMATIC ZONES II, III, IV

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[illegible]

PLAN BELOW ELEV. 0'-0"

BATTERY RM. 301
OUTLET PIPING RM. 17
ELEVATOR
STORAGE BASIN
ELECT ROOM RM. 302
FAN RM. 303

ZONE III SUP UP TO VALVE OR GAS
ZONE III ENH UP
ZONE III ENH UP
ZONE III SUP
ZONE II ENH UP
ZONE II ENH UP
ZONE I ENH.
ZONE II ENH.
ZONE II SUP.

PLAN AT ELEV. +14'-6"

Rm No.	ROOM OR CORR.	SUPPLY	EXHAUST (C.F.M.)
303	INST. R.M.	1 500	1500
304	CORR.	2000	TO "W"ELEV & ST'G. RMS.
302	"W"ELEV ENTRY	1 500	TO "W"ELEV & BATT. RM.301
301	BATTERY RM.	1 000	1750
308	INST. R.M.	1 500	2 000
309	CORR.	2000	TO "W"ELEV & INST RM.308
307	HOT ST'G.	1 500	1800
306	ACID ST'G.	1 500	1800
305	MISC. ST'G.	2 000	2 350
TOTAL		14,500	

2-ZONE HEEL UP			
HR NO.	ROOM OR CORR.	SUPPLY	EXHAUST (C.F.M.)
303	INST. RM.	1500	1500
302A	CORR.	2000	TO "W" ELEV. & ST'G. RMS.
302	"W" ELEV ENTRY	1500	TO "W" ELEV. & BATT. RM 301
301	BATTERY RM.	1000	1750
304	INST. RM.	1500	2000
303A	CORR.	2000	TO "W" ELEV. & INST RM 303
307	HOT ST'G.	1500	1800
304	ACID ST'G.	1500	1850
305	MISC. ST'G.	2000	2350
	TOTAL	14,500

PLAN AT ELEV. -8'-9"

ROOM OR CORR.	SUPPLY	EXHAUST C.F.M.	NO.	AREA	SUPPLY	EXHAUST C.F.M.
TRANSDUCER RM.	1,250	3,350	30		1,000	
CORR LEFT	2,100	TO RM. 3	31	GAS OPER. GALL.	2,000	2,000
" RIGHT	1,500	TO RM. 29	36	TRANSDUCER RM.	1,750	2,750
RUPTURE MON. RM.	3,000	TO RM. 22	C-14	CORR.	750	TO RM 36
BALL SAFETY RM.	700	" " 22	37	RUPTURE MON. RM.	2,500	2,500
CORR	1,250	TO RM. 17 B	19	ZONE II EXH. FAN RM	2,500	2,500
R' ELEV. ENTRY	750	3,000	2	CONFINEMENT VALVE	750	TO RM. 29
	500	TO RM. 7	28		500	1,250 TO RM. 29
	250	" " 7	38		750	1300
CORR.	1,500	" " 7	39		1,000	TO RM. 36
R' ELEV. EQ. RM.	2,500	EXH. @ 40'-0" LEVEL				
				TOTAL	28,800	

[illegible]

Rm. No.	ROOM OR CORR.	SUPPLY C.F.M.	EXHAUST (C.F.M.)
505	ZONE I EXH. FAN RM	7,500	7,500
C29B	CORR.	750	TO RM. 29
C29A	COR R.	1,500	TO RMS. 503 & 504
503	ZONE II EXH. FAN RM	4,500	6,000
502	"R" ELEV. ENTRY	1,500	TO RMS. 501 & 503
501	ELECT. RM.	1,000	1,750
504	INLET VALVE OP G.	750	2,250
C26	CORR.	750	750
507	OUTLET VALVE OP G.	1,500	1,500
	TOTAL	19,000	

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FOR GENERAL NOTES SEE SK-1-26749

LEGEND

- THERMO-BARRIER
- HIGH TEMPERATURE AREA
- STEEL LINER & 2" INSULATION - EMERGENCY PRIMARY COOLANT CONTAINER

FIGURE B-4
ELEVATION VIEW TO RIGHT
OF REACTOR 100-N BUILDING

UNCLASSIFIED

UNCLASSIFIED

SECTION 'D-D' SK.

FOR GENERAL NOTES SEE SK-1-26749

LEGEND

- THERMO-BARRIER
- HIGH-TEMPERATURE PIPING SPACE
- STEEL & LINER & 2" INSULATION - EMERGENCY PRIMARY COOLANT CONTAINER

26749

FIGURE B-4
ELEVATION VIEW THROUGH REACTOR
106-N BUILDING

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[illegible]

FIGURE B-4
PLAN VIEW ZERO ELEVATION
100-N BUILDING

UNCLASSIFIED

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GENERAL NOTES

1. FOR FLOOR FINISHES, FLOOR DRAINS, PUMPS, CONC. CURBS, CONC. PADS, OPENINGS IN CONC. FLOORS & CONC. WALLS, SEE STRUCT. DWGS. & MECH. DWGS.
2. DESIGN OF BLDG. IS BASED ON ANTICIPATED APPROXIMATE SIZES & WEIGHTS OF EQUIPMENT, DRAWINGS NOT AVAILABLE.
3. FOR SIZES & LOCATIONS OF PIPES, DUCTS & CONDUITS SEE MECHANICAL, STRUCTURAL & ELECTRICAL DWGS.
4. DATUM 616" IS ACTUAL RL. 454.50'

LEGEND:

- P.M. --- PROTECTED MESH
- C.J. --- CONTROL JOINT
- (S) --- DETAIL NUMBER
- (D) --- DRAWING NUMBER
- [] --- SPACE NUMBER
- (DO) --- DOOR NUMBER
- E.W.C. --- ELECTRIC WATER COOLER
- F.D. --- FLOOR DRAIN
- R.D. --- ROOF DRAIN
- [] --- POURED CONC.
- [] --- MASONRY BLOCK
- [] --- INSULATION

FIN. FLOOR EL. (-) 22'-0"

MUSCH-OUT PANEL

PLAN AT ELEV. (-) 22'-0"

FIGURE B-1

PLAN VIEW MINUS 16 FEET
ELEVATION
100-N BUILDING

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SECTION 'A-A'
DWG. H-1-29116 (H-1-29116)

SECTION
DWG. H-1-2

117, W-429120 4-14-63
A-A (CONT)

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1 ARRANG

NO.	DESCRIPTION
1	ARRANGEMENT.
2	TOP SHIELD.
3	REAR FACE 'B' SHIELD
4	FRONT FACE 'B' SHIELD
5	RIGHT SIDE 'B' SHIELD.
6	TOP GRAPHITE REFLECTOR
7	BOTTOM GRAPHITE REFLECTOR
8	RIGHT SIDE GRAPHITE REFLECTOR
9	FRONT GRAPHITE REFLECTOR H-1-27109
10	GRAPHITE MODERATOR H-1-27008 THRU H-1-27098
11	BALL SAFETY SYSTEM H-1-28571-1
12	SLEEVE ASSY. FLUX MONITOR. H-1-27915-60
13	SLEEVE ASSY. PUSH ROD H-1-27915-60
14	'B' SHIELD COOLING TUBES
15	PLATE GIRDER H-1-27921-1
16	GAS PURGE SYSTEM.
17	STEAM VENT
18	STEAM VENT- 3' WIDE
19	CONTROL RODS RIGHT SIDE H-1-27991
20	COOLING TUBE SHIELD- H-1-27905-12
21	CONTROL ROD GUY BARREL H-1-28012-1
22	MODERATOR COOLING TUBES
23	THERMAL SHIELD COOLING TUBES
24	THERMAL SHIELD (BORON STEEL)
25	BOTTOM GIRDERS H-1-28261
26	PROCESS TUBE SLEEVE (NOT TUBED) H-1-27926
27	CAST IRON BLOCKS- H-1-28250
28	PROCESS TUBE INLET H-1-28227
29	PERMANENT THERMOCOUPLE STRINGERS SEACH IN LAYERS- 21, 22, 45, 57 & 69 H-1-27368
30	TRAVERSE OPENING - 3 PLACES FRONT TO REAR 6 PLACES SIDE TO SIDE
31	REMOVABLE THERMOCOUPLES 7 PLACES SIDE TO SIDE
32	FRONT TO REAR TRAVELLING WIRE FLUX MONITORS - 3 PLACES
33	CONTROL RODS LEFT SIDE H-1-27991
34	INSULATING SHIELD COOLING TUBES.
35	FOUNDATION COOLING TUBES

FIGURE B-1
W REACTOR ISOMETRIC

MENT

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QTY.	NO.	DESCRIPTION	DRAWING NUMBER
1		ARRANGEMENT	
2		PROCESS TUBES	H-1-28194
3		MODERATOR COOLING	H-1-28533
4		BALL SAFETY SYSTEM	H-1-28571
5		HORIZONTAL CONTROL RODS	H-1-27992
6		REFLECTOR BLOCK DONUTS	SK-1-21379
7		CAST IRON THERMAL SHIELD	H-1-28345
8		MODERATOR	H-1-27000 THRU H-1-27009
9		BALL CHANNEL BLOCKS	H-1-27008

PARTS LIST ALSO PERTAINS TO DWGS.
H-1-27001 AND H-1-27002

OUTLET

NOTE:
ALL SHIELD PENETRATIONS ARE
TYPICAL REPRESENTATIONS. ACTUAL
DETAILED ARRANGEMENTS WILL BE
SHOWN ON COMPONENT SCOPE DWGS.

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Figure B-10

105-N Basic Process Unit Plan

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DECLASSIFIED

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QTY.	PT. NO.	DESCRIPTION
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FOR PARTS LIST SEE DRAWING H-1-27000

NOTE:
ALL SHIELD PENETRATIONS ARE TYPICAL
REPRESENTATIONS. ACTUAL DETAILED
ARRANGEMENTS WILL BE SHOWN ON
COMPONENT SCOPE DRAWINGS.

c Process Unit Elevation Transverse Section

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Figure B-13
Graphite Moderator Isometric

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① ARRANGEMENT FOR INSTALLATION SEE H-1-27991

NOTE: ELEVATION OF ROD END ARE 1' BELOW NOMINAL E OF BARRICADE HOLES. & ROD ELEVATION 4'

TOP OF GRTG 1
 TOP OF GRTG 2
 TOP OF GRTG 3
 TOP OF GRTG 4
 TOP OF GRTG 5
 TOP OF GRTG 6
 TOP OF GRTG 7
 TOP OF GRTG 8
 TOP OF GRTG 9
 TOP OF GRTG 10
 TOP OF GRTG 11
 TOP OF GRTG 12
 TOP OF GRTG 13
 TOP OF GRTG 14
 TOP OF GRTG 15
 TOP OF GRTG 16
 TOP OF GRTG 17
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 TOP OF GRTG 19
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 TOP OF GRTG 27
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ELEVATION-LEFT SIDE

X	1	ARRANGEMENT	
21	2	HORIZONTAL ROD, CONTROL	H-1-27982 P. 1
1	3	VALVE RACK ARRANGEMENT	H-1-27984 P. 1

ENT
IN NOTES

32'-0" INNER ROD ROOM
TOP OF GRG ELN 7'-6"
EL(1)16'-6 3/8"
-11-25'-6 3/8"

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FIGURE B-14
ARRANGEMENT OF LEFT SIDE
CONTROL RODS

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Figure B-15
105-N Inlet Process Tube Assembly

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Figure B-16
105-N Outlet Process Tube Assembly

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Figure B-17

Process Tube Connector Array

PLAN - LEFT SIDE OUTLET

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APPENDIX C

THE REACTOR PROCESS

1. PHYSICS

This section includes information which is new or has been revised since the issuance of Volume 1 as well as an expanded treatment of those physics details of the reactor which would have been awkward to include in the earlier volume because of space limitations. The principal new or revised information deals with the control systems. Corrections of an error discovered in the original layup of the exponential pile has resulted in a reduction of about 15 per cent in the control rod strengths quoted in Volume 1. Also reactivity ramps are considered that are larger than those considered to be the maximum in Volume 1. These ramps would result from certain failures or errors in the use of the control systems. Neither of these revisions alter any of the previous conclusions with regard to nuclear safety of the reactor.

1.1 Basic Physics Processes

Like the other Hanford reactors, N Reactor is a converter - the principal nuclear processes being the transmutation of U^{238} to Pu^{239} and the fissioning of U^{235} to provide the neutron carriers of the chain reaction and the source neutrons for plutonium production. The heat released in the fissioning process may ultimately be used to produce usable electrical energy and the reactor and fuel were designed to permit this.

Fission neutrons are born in the reactor fuel with an average energy of two Mev. To be useful as chain carriers, they must be slowed down to thermal energies. The principal slowing down process for neutrons in N Reactor is elastic scattering with the carbon nuclei of the reactor's moderator. The cooling water also contributes to the moderating process.

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N Reactor is a heterogeneous reactor; that is, its fuel and moderator are separated into discrete volumes. The lumping of the uranium is necessary to overcome excessive losses of neutrons to resonance capture. The resonance neutron capture probability in N Reactor, in spite of the fuel lumping, is approximately 0.18 which is larger than in the older Hanford reactors. The reason for the higher resonance neutron capture probability is the undermoderated nature of the N lattice. The C-U atom ratio is 33 compared to a ratio of 75 for the K Reactors.

Since the predominant absorber of resonance neutrons in a production reactor is U^{238} , the importance of resonance absorption becomes apparent. Unlike the multiplication factor, which becomes a maximum at the particular C-U ratio, the absorption of neutrons in U^{238} increases as the C-U ratio is decreased. Beyond the point of optimum reactivity, a chain reaction eventually becomes impossible in a graphite-moderated reactor fueled with natural uranium. In N Reactor, an expected uranium enrichment of 0.947 w/o U^{235} is required to sustain the chain reaction with reasonable excess reactivity to accommodate various transients. The increased amount of U^{235} will decrease the plutonium production from thermal neutron absorption in U^{238} , but the decrease will not be enough to counterbalance the increased resonance neutron absorption, yielding an improved neutron economy compared to the existing reactors.

An important feature of the undermoderated reactor, quite apart from its better over-all neutron economy, is the fact that the lattice is fail-safe upon loss of coolant. Water, which acts as both a moderator and a poison, is worth more as a moderator in N Reactor and hence its removal reduces the moderating efficiency of the reactor and increases the resonance capture probability. (The resonance capture pro-

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bability in the dry N lattice is about 0.28).

Related to the fail-safeness of the lattice on coolant loss is the reverse effect arising from an increase in the amount of water in the reactor. Both a cold water injection at equilibrium reactor power level (which would increase the density and hence the amount of water in the process tubes), and a flooding of the graphite structure can add reactivity.

The relative reaction rates which take place in a nuclear reactor are functions of the concentrations of each material present, the temperatures of these materials, and the average energy (temperature) of the neutrons present in the material. The dynamic equilibrium among the various competitive nuclear reactions will therefore change as new materials are formed (fission products and transuranic elements, for example) and as the transient proceeds from a cold, shutdown condition to the equilibrium condition. These temperature and exposure-dependent effects are very important and their consideration must be taken into account in the reactor and fuel design. Sufficient reactivity must be available to permit control over all of the useful reactor operating cycle. Too much reactivity, however, must be avoided from a safety standpoint. The principal operating reactivity effects are those associated with the buildup of Pu^{239} and burnout of U^{235} , buildup of fission products Xe^{135} and Sm^{149} , and heating of moderator, coolant and uranium.

1.2 Lattice Characteristics*

1.2.1 Description of Lattice

*Several special terms are commonly used to describe lattice characteristics. Among them are:

Cold	-	Graphite and fuel at or near room temperature.
Hot	-	Graphite and fuel at or near equilibrium operating temperature.
Clean	-	No short-lived, fission product poison present.
Green	-	Fuel at low or zero total exposure.
Ripe	-	Fuel at or near goal fuel exposure.

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The N-Reactor lattice is rectangular; the vertical spacing is nine inches and the horizontal spacing is eight inches. The process tubes are nominal quarter-inch wall, Zircaloy-2, 2.7-inch I. D. tubes. The fuel elements are concentric tubes of metallic uranium enriched to 0.947 w/o U^{235} . The fuel cladding is Zircaloy-2. Additional lattice dimensions are given in Table C-1 and a cross-sectional view of the lattice is shown in Figure C-1.

TABLE C-1
LATTICE DIMENSIONS

Spacing	8" horizontal x 9" vertical	
Equivalent Cell Diameter (Inches)	9.575	
Fuel Element Diameter (Inches)	<u>Outer</u>	<u>Inner</u>
O. D. Clad	2.406	1.249
O. D. Fuel	2.354	1.166
I. D. Clad	1.819	0.492
I. D. Fuel	1.764	0.438
Process Tube Diameter (Inches)		
O. D.	3.258	
I. D.	2.708	

The process tubes pierce the reactor core front-to-rear. A graphite cooling system is provided composed of 640 Zircaloy-2 side-to-side cooling tubes in every filler layer vertically and on 16-inch centers horizontally. These tubes have an 0.75 inch O. D.

1.2.2 Cold, Clean, Green Lattice Parameters

The cold, clean, green lattice parameters for the N Reactor have been determined experimentally for a prototypical lattice. The de-

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FIGURE C-1

CROSS SECTION OF N REACTOR LATTICE CELL

LEGEND

1. Water
2. Zircaloy-2 Cladding
3. Zirconium
4. Vent
5. Graphite

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scriptions of the experimental arrangements are given in Reference 1.

1.2.2.1 Measurements Made In The Physical Constant Test Reactor (PCTR)

Two lattice arrangements were studied. One was as exact a mockup of the actual lattice as possible; the other was a "condensed" lattice in which the large steam vent and other voids in the graphite stack were removed, and the lattice spacing compressed to maintain the same graphite-to-uranium ratio.

1.2.2.2 Exponential Pile Measurements

The material buckling for the mockup and condensed lattices were measured in an exponential pile. The measurement of the material buckling and k_{∞} (in PCTR) enabled a qualitative determination of the migration area. The migration area deduced for the condensed lattice is 300 cm²; for the mockup lattice, 646 cm². The ratio of the two values is 2.15 which is in good agreement with the ratio predicted theoretically to account for the effect of the voids on neutron diffusion.

1.2.2.3 Best-Value Lattice Parameters

The availability of experimental lattice parameters for a lattice almost identical to the N lattice makes it possible to normalize theoretical lattice parameters and determine best values for the actual N lattice. The best-value cold, clean, green lattice parameters for N Reactor are given in Table C-2.

1.3 Core Characteristics

1.3.1 Description of Core and Loading

The N-Reactor core, at initial startup, will be comprised of 1004 lattice units as described in Section 1.2.1: provisions have been made to increase this number to 1072 if future operating trends justify

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TABLE C-2

BEST-VALUE, COLD, CLEAN, GREEN LATTICE PARAMETERS*

<u>Parameter</u>	<u>Value</u>
k_{∞}	1.0778
η	1.4184
ϵ	1.0411
P	.8291
f	.8803
B_m^2	115.6 Mb

*No cross-cooling tube included. Effect would reduce k_{∞} to 1.0752.

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additional irradiation volume. This core is pierced by three systems of openings, in addition to the process channels and steam vents - horizontal control rod channels; the Ball 3X Safety System channels and the graphite moderator cooling tubes all described in Appendix B.

These three void systems, along with the 2" x 3" rectangular steam vents running both parallel and perpendicular to the process channels, reduce the core graphite density from a nominal 1.7 gms/cm^3 to an effective 1.3 gms/cm^3 .

The physical dimensions of the core are set by the lattice geometry and the charge length. There are 34 vertical rows of tubes, resulting in a core width of 34×0.667 feet or 22.667 feet; there are 32 horizontal rows of tubes, or a $32 \times 0.75 = 24$ -foot core height. The corners are truncated as shown in Figure C-2 resulting in a core equivalent radius of 12.64 feet; the core length is defined by the portion which contains fuel material, or 35 feet. This roughly cylindrical arrangement contains 693.0 tons of graphite (density 1.70 gms/cm^3) tons of uranium.

The core is surrounded radially by a reflector of high-purity graphite which is infinite in effect; the front and rear "reflector" is composed of graphite plus other material (e.g., process tube, water, dummies). (The reflector neutron effect will be discussed in Section 3.4).

1.3.2 Slowing Down and Diffusion of Neutrons

As a consequence of the undermoderated lattice of the N-Reactor, the fast- to thermal-flux ratio is quite large relative to that of existing, overmoderated Hanford reactors. As an illustration of this difference, the ratio of fast- to thermal-flux is estimated to

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FIGURE C-2

NFR CORE GEOMETRY

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be 2.50⁽²⁾ at N and 1.08 at a K reactor; this of course enhances the production of plutonium in the reactor and also makes it fail-safe (from a reactivity standpoint) upon loss of water.

The lattice migration area deduced from an exponential experiment is 646 cm² (see Section 1.2.2.2); this is separable into a Fermi Age (τ) of ~ 480 cm² and a Diffusion Area (L^2) of ~ 160 cm² at room temperature. At operating temperature M^2 increases by $\sim 10\%$.

1.3.3 Neutron Streaming

The large void fraction in the reactor core allows preferential neutron passage along the void channels; this effect, called "streaming", manifests itself in various lattice parameters, in particular τ and L^2 and thus in reactor leakage and multiplication. As an illustration of the magnitude of this effect, if the graphite were "smeared" over the core (retaining the same number of graphite atoms but reducing the density and filling all void space) the over-all pile reactivity would be increased by 0.8 per cent due to decreased leakage.

1.3.4 Neutron Leakage - k_{eff}

The neutron leakage in a reactor is a function of the reactor size and the neutron migration area; it is strongly influenced by the presence of a reflector of pure moderator material surrounding the active core. For a uniform, homogeneous loading, the effective and infinite lattice multiplication factors are related by the equation:

$$k_{eff} = \frac{k_{\infty}}{1 + Bg^2L^2} e^{-Bg^2\tau} \frac{k_{\infty}}{1 + Bg^2(L^2 + \tau)}$$

where:

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- k_{eff} = effective pile multiplication,
 k_{oo} = infinite lattice multiplication,
 B_g^2 = pile geometric buckling,
 L^2 = lattice diffusion length squared, and
 γ = Fermi Age.

This equation may be used to calculate an effective geometric buckling for a heterogeneous reflected reactor if the various parameters are known.

Multi-group calculations by Bowers indicate an effective multiplication for the fully-loaded "green" reactor of ~ 1.061 at room temperature; independent calculations by Gumprecht indicate $k_{eff} \approx 1.059$. The effective geometrical buckling may be determined from a rearrangement of the Fermi Age Equation:

$$B_g^2 = \frac{k_{oo} - k_{eff}}{k_{eff} M^2}$$

Assuming "best values" of the parameters are given in Section 1.2.2, and a k_{eff} for the fully loaded reactor of 1.060, the total geometrical buckling is 22.1 micro-bucks ($1 \mu b = 10^{-6} \text{ cm}^{-2}$).

The front and rear reflector augmentation at N Reactor is estimated from a cosine flux distribution and the longitudinal buckling component is $\sim 7.3 \text{ ub}$; this indicates a radial buckling component of $14.8 \mu b$ and a radial reflector augmentation of ~ 240 centimeters. (All older Hanford production reactors have radial augmentations of the order of 60 centimeters).

The rather large reflector savings is due to the pre-dominantly "fast" neutron spectrum in the core; epithermal neutrons emerging from the core into the reflector are quickly thermalized. In

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the thermal state, there is a good probability that the neutrons will be transferred back to the core and utilized in the fission process; in fact, there is a net gradient of thermal neutrons from the reflector inward to the core.

It should be noted that the reflector augmentation as defined by the geometrical buckling is extremely sensitive to the total pile reactivity; an error of only 0.2 per cent $\Delta k/k$ in the determination of k_{eff} can result in an error of over 60 centimeters in the estimate of δ , the reflector savings. Best estimates for the effective reactor size (in terms of neutron boundaries) are:

Effective radius = 625 centimeters

Effective length = 1160 centimeters

These dimensions are based upon $k_{eff} = 1.060$ at room temperature.

The total leakage of NPR compares to leakage at a BDF type Hanford reactor as follows:

NPR, cold, clean 0.0143% $\Delta k/k$

NPR, hot, clean 0.0151

BDF, hot, clean 0.0192

1.3.5 Core Power Distribution

The significant reflector augmentation results in an exceptionally high flattening efficiency in N Reactor, even without the smoothing effect of control rods or other flattening agents. Theoretical flattening efficiencies of 80 - 98 per cent are calculated for routine operation, with actual efficiencies (including microscopic fine structure) of 70 - 90 per cent. The best estimate for the flattening efficiency is 90 per cent. Radial power distribution for various flattening efficiencies

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are plotted in Figure C-3.⁽³⁾ The axial power distribution will follow an approximate cosine shape with a peaking upstream of pile center due to the higher coolant temperature towards the inlet end of the reactor.

1.4 Reactivity Transients

The sources of the reactivity variations over a normal operating cycle, including shutdown and startup, in N Reactor are similar in kind to those in the other Hanford reactors. One additional source, that resulting from coolant density variations, occurs in N Reactor.

1.4.1 Temperature Coefficients of Reactivity

Temperature changes in a reactor affect the neutron population in two separate ways. The first is by changing the nuclear properties of the materials. For example, the nuclear cross sections are a function of neutron temperature (speed). The second is by changing the physical properties of the materials. For example, the coolant density is a function of the coolant temperature.

The temperature-dependent sources of reactivity variations in N Reactor include, in addition to the usual fuel and graphite heating effects (which are nuclear), one associated with coolant heating (which is both nuclear and physical). The fuel temperature and coolant temperature coefficients of reactivity are negative; the graphite temperature coefficient is positive. Other physical effects, such as fuel expansion, are too small to produce any significant temperature coefficient.

The integral graphite temperature reactivity effect is shown in Figure C-4 as a function of local graphite temperature. The slope of the curve at any given temperature is the differential coefficient, $1/k \, dk/dT$. To obtain a reactor-averaged coefficient, the flux-

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squared weighted graphite temperature distribution over the active volume of the reactor is volume-averaged. It is noted that the graphite coefficient is positive for all exposures up to 1400 MWD/T; however the differential coefficient increases with increasing exposure. This exposure dependence is not as pronounced as that observed in the natural uranium-fueled, Hanford reactors. The coefficient is also largest for cold graphite; the "hot" differential coefficient is less than half the "cold" value.

The integral coolant temperature reactivity effect is shown in Figures C-5 and C-6, as a function of the inlet coolant temperature. The magnitude of the differential coefficients (slopes) are constant with temperature for zero power (Figure C-5), but increase with temperature for the full power case (Figure C-6). The sign of the coolant coefficient is negative.

The curves in Figure C-5 depict the reactivity behavior with coolant temperature which would be characteristic during a reactor startup. Under zero power conditions the coolant inlet temperature is equal to the average coolant temperature. The curves in Figure C-6 are representative of the reactivity effect of varying the coolant inlet at full reactor power, and in this case the average coolant temperature would be approximately the mean of the inlet and outlet temperatures, the outlet being 105 C hotter than the inlet. (The graphite and coolant temperature coefficients are theoretical).

The reactivity effect associated with increasing the fuel temperature (Doppler effect) is shown in Figure C-7. The Doppler effect is not a function of fuel exposure, but is dependent on the fuel temperatures, the dependence being $T^{-1/2}$ where T is the absolute fuel

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temperature. The curve in Figure C-7 is plotted against the average fuel temperature, and is based on an experimental determination of the fuel temperature in the Physical Constants Test Reactor using dry N-Reactor fuel in the N lattice. The coefficient can be expressed as:

$$\frac{1}{k} \frac{dk}{dT} = -5 \times 10^{-4} (T^{\circ}K)^{-1/2}/^{\circ}C.$$

1.4.2 Exposure-Dependent Effects

The reactivity effects associated with exposure include the effects of U²³⁵-depletion, Pu²³⁹, Pu²⁴⁰ and Pu²⁴¹-formation, and fission product formation. The net reactivity effect of U²³⁵-burnout and plutonium formation is positive in spite of a less-than-one-for-one replacement of fissionable atoms by the conversion process. This increase results from the higher fission cross section and neutron yield of Pu²³⁹ compared to U²³⁵. The net exposure effect, including fission products, is negative, however. The following sections deal with the various exposure-dependent effects on reactivity.

1.4.2.1 Long Term Gain (Cold)

The effects of plutonium formation, U²³⁵-burnout and fission product formation (other than xenon and samarium) are most conveniently combined into one effect, called the long term gain. The long term gain is given most simply for the cold reactor, since it is usually convenient also to include exposure effects in the temperature coefficients.

Based on observation of long-term reactivity effects in the present Hanford reactors, the cold long term effect is positive in natural uranium fuel up to an exposure of 550 MWD/T. In

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N Reactor, the higher fuel enrichment will tend to decrease the long term gain, but the harder neutron spectrum should be a compensating factor. Calculations based on U^{235} -depletion and plutonium formation and assuming an average cross section of 50 barns for fission products (other than xenon and samarium), indicate a cold, long term reactivity gain to an average exposure of 700 MWD/T of $\sim 2 \times 10^3$.

1.4.2.2 Xenon Poisoning

After a reactor startup, xenon poisoning reaches its equilibrium value at 30 MWD/T (2.5 days) and has a value of 2.8 per cent $\Delta k/k$ for an average pile exposure of 700 MWD/T. After reactor shutdown, the xenon poisoning initially increases, due to the decay of the precursor iodine-135, then decreases and eventually decays to zero. A typical startup and shutdown xenon transient is shown in Figure C-8 for an average fuel exposure of 700 MWD/T. The exposure dependence of the operating xenon poisoning is shown in Figure C-9.

1.4.2.3 Samarium Poisoning

Samarium-149 poisoning reaches an equilibrium value of about $6.5 \times 10^{-3} \Delta k/k$ at 100 MWD/T exposure. After each reactor shutdown, the samarium poisoning increases due to the decay of its precursor promethium-149. The equilibrium value is regained by the time an additional exposure of 100 MWD/T is reached after each startup. The equilibrium samarium is shown in Figure C-9 as a function of fuel exposure.

1.4.2.4 Neptunium Holdup

Calculations of the reactivity gains in a reactor due to plutonium buildup usually assume instantaneous formation of plutonium. The fact that the formation of plutonium-239 is delayed through the 2.33 day, beta-decay of neptunium-239 makes it necessary to correct the

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FIGURE C-8

XENON TRANSIENT - N REACTOR
(AVERAGE EXPOSURE 700 HR./T)

STARTUP
OPERATING
TRANSIENT

REACTOR
SHUTDOWN
AT "60 HRS"

TRANSIENT
FOLLOWING
SHUTDOWN

REACTOR STARTUP
AT TIME ZERO

REACTIVITY (ρ , %)

TIME (HOURS)

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calculated reactivity gains. This is most conveniently done by assigning a reactivity "loss" for the neptunium holdup effect. The neptunium holdup reactivity defect at equilibrium is about 4×10^{-3} . The equilibrium neptunium holdup effect is shown in Figure C-9 as a function of exposure.

1.4.2.5 Over-All Long Term Reactivity Effects

For additional clarity, two sets of curves are given of the total exposure-dependent reactivity variation. Neptunium holdup, samarium poisoning, other fission product poisoning, U^{235} burnout and plutonium buildup are all included. The exposure dependence of xenon poisoning is included, but not the initial, green 3.1 per cent $\Delta k/k$ loss. The curve give essentially the reactivity transient the reactor would experience after a startup and equilibrium xenon and temperature had been reached. Figure C-10 is for a uniform exposure. Figure C-11 is for a "mixed" exposure - that is exposures running uniformly from zero to goal. The abscissas in the figures are to be interpreted as follows: In the uniform exposure case, the goal exposure is equal to the pile-average exposure. In the mixed exposure case, the pile-average exposure is assumed to be about half of the goal exposure. The latter probably most closely represents the most likely charging and discharging scheme. The hot-pile, over-all, long term effects are also given in Figures C-10 and C-11, thus indicating the exposure-dependence of the temperature coefficients.

1.4.3 Operational Reactivity Transients

The various temperature and exposure reactivity contributions have been combined to determine typical reactor startup and shut down reactivity transients.

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1.4.3.1 Startup Reactivity Transient

A reactivity transient for a startup from an infinite outage (zero xenon) was given in Figure 5.1 of Volume 1. As presented, the curves show a higher cold, clean reactivity for the 700 MWD/T exposure case than for the green case. This will actually not be the case; the cold, clean, ripe reactivity in Figure 5.1 of Volume 1 (as well as the cold, clean ripe k_{ex} listed in Table 5.2 of Volume 1) did not include the samarium and neptunium effects. Taking these long term effects into account will reduce the 700 MWD/T, cold clean reactivity to 5.3%.

1.4.3.2 Normal Shutdown Transient

Typical reactivity shutdown transients are given in Figure C-12 for the reactor loaded with green fuel and fuel at an average exposure of 600 MWD/T. The initial equilibrium reactivities are those expected with 0.947 per cent enriched fuel - actually the reactivity carried in control rods at equilibrium will be nearer one per cent, so the entire curves can be moved down about one per cent. To be consistent with the startup transient (Figure 5.1 in Volume 1) the reactivities are shown as they are expected to be with the full 0.947 per cent enrichment.

The assumed power reduction in Figure C-12 is 2-1/2 per cent per minute. The initial, small reactivity peak is due to the initial cooling of the fuel and coolant. Following the initial rise in reactivity is a decrease for about an hour as the graphite cools and xenon builds up. During this period the primary coolant system pressure is reduced and the average coolant temperature is held constant at about 338 F. Further cooling of the coolant and graphite to shutdown temperature (100 F) and further xenon buildup produces the transient shown be-

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FIGURE C-12
TYPICAL SHUTDOWN REACTIVITY TRANSIENTS (LATTICE)

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tween 1-1/2 hours and ten hours after starting down. After ten hours, the xenon decay becomes predominant and the reactivity increases to the cold pile values of 5.9 and 5.3 per cent respectively for the green and ripe cases.

1.4.3.3 Scram Transient

If the reactor is shutdown from equilibrium by the scrambling of the horizontal safety rods, the short-time behavior of the reactivity transient depends on the strength of the rods being scrambled. Several scram transient curves are given in Figure C-13. The rise following the initial reactivity decrease in each case is due to cooling of the fuel and coolant. The curves in Figure C-13 (unlike the other transient curves) are plotted in terms of the reactivity of the reactor - that is the sum of the lattice reactivity and the reactivity effect of the rods. It is seen that nearly the full worth of the rods is obtained at about 1.3 seconds after the time the master safety circuit breaker operates.

1.5 Reactor Control Systems

The reactor control systems were discussed in limited detail in Volume 1. Additional information about the systems, sensors and control circuits appear in Appendix D of this volume. This section will deal mainly with the physics aspects of the control and safety systems, in particular expanding and updating information presented previously.

FIGURE C-13

SCRAM REACTIVITY TRANSIENTS (PILE)

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1.5.1 Control Philosophy

An essential part of any discussion of the control systems is the philosophy which governed their design. Although, the philosophy of reactor control and safety, as it is applied to the Hanford reactors, originated in the days of the Manhattan Project and has successfully stood the test of time without many modifications, it was judged prudent to alter in certain respects the philosophy as applied to the design of N Reactor. For example, the N-Reactor lattice is fail-safe upon coolant loss - a departure from the earlier Hanford lattices. The final philosophy as it is applied to the design of the N-Reactor control and safety systems is presented here. Additional information can be obtained in a reference treatment of the N-Reactor control systems.⁽⁵⁾

1.5.1.1 Operational Control Criteria

The operational control system must meet the following criteria:

- (1) Adequate strength to maintain the reactor subcritical under all normal operating and shutdown conditions.
- (2) Adjustment at rates fast enough to override all normal, operational reactivity transients and to minimize delays at startup. However, the system must be limited in its withdrawal rate to that rate which, in coincidence with simultaneous failure of the Primary Safety System, would not result in fuel melting.

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- (3) Precision positioning and movement to permit establishing and maintaining the reactor power level within desired limits.
- (4) Adequacy of geometrical coverage and flexibility to permit attaining adequate power-flattening efficiencies and satisfactory control over spatial power cycling.

1.5.1.2 Safety Control Criteria

There must be two safety control systems. One of the systems, depending on the nature of the accident, is to provide the primary protection; the other, then, will function as a backup. The safety control systems must meet the following requirements:

- (1) Both primary and backup system must have adequate strength (if inserted) to maintain the reactor subcritical for all credible* accidents at least until the reactor is damaged beyond repair as a result of the causative accident. Beyond this point the inserted safety control system or systems are required to maintain control only to the extent that the consequences of the incident are no worse than those which would result from the causative accident alone.

- (2) The primary (backup)** system must have speed of control insertion adequate to (a) limit a power excursion from credible accidents which would in themselves result in damage (loss of coolant, for example) to one of a magnitude such that the ultimate consequences of the

* Credible accidents exclude acts-of-war, sabotage, and earthquakes severe enough to prohibit entry of poison from either safety system.

** Read words in parentheses for backup system.

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excursion are no worse than those which would have resulted from the causative accident along, or (b) limit a power excursion from other credible accidents (such as fast rod withdrawal) to one of a magnitude such that no damage beyond mild fuel stressing (fuel melting) will result.

1.5.1.3 Backup Philosophy

The purpose of a backup safety system is to provide safety protection in the event of a failure in the primary system. There is to be no intent in design to permit the removal of one or part of one of the safety systems from service on the premise that the backup system is in service and will act if necessary.

Each safety system must back up the other system for as many failure and accident conditions as possible. There may be certain accidents, such as a severe earthquake, in which one system is disabled, rendering it useless in either a primary or backup capacity. Therefore, the two systems should be different in design, if possible, to reduce the probability of both systems being disabled by one accident.

Both safety systems shall be activated by automatic trips as well as by manual trips. The two safety systems shall have independent drives. It is desirable that they also take their activating signals from independent sensors through independent circuits. If common sensors or common circuits are used in the two safety systems, the reliability of these common elements must be high enough to yield an operational fault rate which is commensurate with that obtainable under a wholly-independent arrangement.

The degree to which the control system meet these criteria is discussed in later portions of this section. The next

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subsections discuss the latest information on the control system strengths.

1.5.2 Horizontal Control Rods

Revised values of the control strengths of the horizontal control rods have been obtained since the issuance of Volume 1 of this hazards report. The reason for the revision, which is in the downward direction, was the discovery of an error in the original stacking of the exponential pile in which the control strengths of the rods were measured. Restacking of the exponential pile and remeasurement of the rod strengths have been completed and the values reported in this appendix are based on the corrected measurements and supersede those in Volume 1.

As will be seen later, there is no impairment of reactor safety resulting from the lowered control rod strengths. The reason for this is that the initial reactor fuel loading was designed so that the transients were within the theoretical control strengths. The increase in control strength indicated by the first measurements was considered as an unexpected bonus and the fuel element was not redesigned (made larger) to take credit for the apparent increased control capacity.

1.5.2.1 Control Strengths

The experimental determinations for the N Reactor control strengths consisted of determining the cold-reactor, control effectiveness of a single rod and of six rods arranged in the proper spacing in an exponential pile. Using the same theoretical methods which were used to calculate the control strengths in the N Reactor, itself, the exponential pile results were correlated with the theory. The observed relationship between calculated and experimental values in the exponential pile was then used to establish most probable values for the control strengths in

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N Reactor under various operating conditions. The results of this normalization are listed in Table C-3. The calculations are described in Reference 6.

The local control strengths of the control rods included in the table are defined as the change in buckling of a region with a cross sectional area of 1150 in.² (32 in. x 36 in.). The migration areas listed are those used to derive the reactivity effect by the relationship $\Delta k/k = \Delta B^2 M^2$. The full-pile strengths are lower than the local strengths due to incomplete coverage of the reactor core with control rods. The lower values in each case represent an extrapolation from the six-rod exponential measurement, whereas the higher values are derived from the single-rod exponential measurement. No explanation has been found for the difference. Also included in the table are the purely-theoretical values (in the last column). The agreement between theory and experiment is quite good.

TABLE C-3
MOST PROBABLE CONTROL STRENGTHS
(HORIZONTAL RODS)

<u>Reactor Condition</u>	<u>Local Control Strength (a)</u> <u>$\Delta B^2(\mu b)$</u>	<u>FULL REACTOR STRENGTH</u>			
		<u>Most Probable (a)</u>			<u>Theoretical</u>
		<u>$\Delta B^2(\mu b)$</u>	<u>$M^2(\text{cm}^2)$</u>	<u>$\Delta k/k(\%)$</u>	<u>$\Delta k/k(\%)$</u>
Cold Reactor	170-213	120-133	640	7.7-8.5	7.0-8.3 ^(d)
Hot Reactor	165-205	118-131	710	8.4-9.3	7.7-9.1
Cold Reactor Flooded ^(b)	205-245	131-140	409	5.4-5.7	5.1-5.6
Hot Reactor Flooded ^(b)	200-238	130-138	479	6.2-6.6	5.9-6.5
Hot Reactor-Cold Water	170-213	120-133	647	7.8-8.6	7.1-8.4
Maximum Accident ^(c)	205-245	131-140	416	5.4-5.8	5.2-5.7

(a) Lower value represents extrapolation from six-rod exponential pile measurements; higher value from single-rod exponential pile measurement.

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- (b) Flooded to extent that maximum reactivity is achieved. Rod strengths will continue to decrease beyond "optimum" flooding but poisoning due to water exceeds the loss in control strength beyond point of optimum flooding.
- (c) Optimum flooding combined with cold water injection in hot reactor.
- (d) Lower value calculated with steaming correction on neutron free path; higher value with no steaming correction.

1.5.2.2 Maximum Worth of Single Rods or Groups of Rods

One-dimensional, three-group diffusion theory calculations have been made to determine the worst flux distortion under which rods may be withdrawn. Two separate cases were studied. In each case, all rods were assumed inserted except for a single bank of rods (in N Reactor there are 11 rod banks). Withdrawing an entire center bank (nine rods) would result in an increase of about 3 per cent $\Delta k/k$. Withdrawal of an entire outermost rod bank (six rods) would result in an increase of 2.5% $\Delta k/k$. Withdrawing two adjacent banks could result in reactivity increases of 6 per cent and 4.3 per cent for central and outer rods, respectively. With normal hydraulic system action, the maximum reactivity withdrawal rate is 4 per cent per minute which corresponds to pulling two adjacent outer rod banks (12 rods) at the maximum rate.

1.5.2.3 Stuck Rod Limit

In order to determine a failed (or stuck, or OFF) rod limit, it is necessary to first define the maximum reactivity condition which the rod system alone is expected to control. This is defined as the condition attained either under hot graphite, xenon-free, optimum-flooded conditions, or under hot graphite, xenon-free, cold-water-injector conditions, but not both.

The calculation of a failed-rod limit was simplified by conservatively assuming that the failed rods are centrally located, adjacent to each other, and in the region of maximum flux. A one-energy

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group, multi-regional buckling calculation was used to determine the lattice material buckling at which the reactor would be just-critical with various numbers of rods out of service. It was found that three adjacent rods could be removed from service while maintaining the reactor subcritical under either accident assumption. The failed-rod limit has, therefore, been set at three. The safety circuits are designed so that when four or more rods fail to reach their 75 per cent limit switch within 1.5 seconds after a safety circuit trip, these failed rods are backed up by an automatic drop of the balls in the rows in front of and behind each failed rod.

1.5.3 Ball 3X System

1.5.3.1 Control Strengths

Exponential pile measurements and calculations have been performed to determine the strength of the Ball 3X system using techniques similar to those used for determining rod strengths. The results are given in Table C-4.

1.5.3.2 Inoperable Ball Column Limit

An "out-of-service Ball 3X hopper limit" has been determined in the same way as the failed-rod limit described previously. A total of four ball hoppers may be locked out while still maintaining assured capacity to drive the reactor subcritical under either the cold water or flooding accident conditions.

1.6 Total Control Considerations

The operational control criteria (Section 1.5.1.1) and part of the safety control criteria (Section 1.5.1.2) are restatements of the Hanford Total Control Criterion. As far as the operational control

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TABLE C-4
MOST PROBABLE CONTROL STRENGTHS
(BALL 3X)

<u>Reactor Condition</u>	Local Control Strength (a) <u>$\Delta B^2(\mu b)$</u>	<u>FULL REACTOR STRENGTH (a)</u>		
		<u>$\Delta B^2(\mu b)$</u>	<u>$M^2(\text{cm}^2)$</u>	<u>$\Delta k/k(\%)$</u>
Cold Reactor	117-138	104-120	640	6.7-7.7
Hot Reactor	108-132	96-116	710	6.8-8.2
Cold Reactor Flooded(b)	144-165	125-140	409	5.1-5.7
Hot Reactor Flooded(b)	135-159	118-135	479	5.6-6.5
Hot Reactor Cold Water	117-138	104-120	647	6.7-7.8
Maximum Accident(c)	144-165	125-140	416	5.2-5.8

(a) Lower value represents extrapolation from six-channel exponential pile measurement; higher value from single-channel exponential pile measurement.

(b) Flooded to extent that maximum reactivity is achieved.

(c) Optimum flooding combined with cold water injection in hot reactor.

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criteria are concerned, the horizontal control rods have adequate strength in the operating reactor (minimum of 7.7 per cent $\Delta k/k$ cold) to control any foreseeable transient. The measured cold, clean, green reactivity with 0.94 per cent enriched fuel is 5.9 per cent. For ripe fuel, the cold, clean reactivity is 5.3 per cent. Thus, the cold reactor shutdown margin with the control rods is at a minimum 1.8 per cent $\Delta k/k$. Actually, the reactor fully enriched with 0.947 per cent U^{235} fuel would probably have about an excess reactivity at equilibrium about one per cent higher than desirable, and an effective enrichment reduction by the use of natural uranium columns will probably be employed to obtain a maximum cold excess reactivity of 4.9 per cent. This would serve to increase further the shutdown margin.

The more restrictive criterion is the one dealing with adequacy of control over various accident conditions. The first safety control criterion states that either control system must be able to maintain the reactor subcritical for all credible accidents.*

*The qualifying clause "at least until the reactor is damaged beyond repair, etc." is included to make it unnecessary to maintain subcriticality beyond the point when the consequences of the accident would not be increased significantly even if the reactor did go supercritical. Such is the case in the present Hanford reactors for the water-loss accident. When the reactor has heated sufficiently from fission product decay heat to render it no longer possible to re-establish cooling, there is no real point in being concerned about a delayed-critical state since a meltdown is inevitable. Going supercritical would only hasten the meltdown with no significant increase in fission product inventory or fission product release probability. The situation at the N Reactor is slightly more complicated. Even if flow cannot be re-established in the process tubes, accident severity is minimized by operation of the moderator cooling system. Supercriticality during operation of this system would increase severity of consequences.

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The shutdown margins for both the horizontal rod and Ball 3X systems for a number of reactivity states are given in Tables C-5 and C-6.

It is seen that either safety system is able to maintain the reactor subcritical under all credible conditions. A simultaneous cold water accident and optimum flooding incident resulting in a five per cent increase in reactivity (which is considered incredible) would result in a reactivity state outside of the strength of the horizontal rods. However, assuming a just-critical state prior to the postulated accident, the shutdown margin of the Ball 3X system would be positive even for this accident. It is stated in Volume 1 (Page 4.29) that either the scrammable horizontal rods or the Ball 3X system could control this combination accident. Though this not now considered correct, the total control criterion is still satisfied for any credible accident.

1.7 Speed-of-Control Considerations

The second safety control criterion (Section 1.5.1.2) is a restatement of the Hanford Speed-of-Control Criterion. Since the water-loss accident in N Reactor does not result in a rapid increase in reactivity as is the case in the present reactors, the accident governing the response of the safety systems is one associated with rod-withdrawal. Since the horizontal rods combine both operational and safety functions, the limitation on withdrawal rate specified in the operational control criteria can also be included in this discussion.

There are two degrees of accidents involved and the allowable consequences differ for each. Firstly, there is the case where the control rods are withdrawn so as to result in the maximum possible

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TABLE C-5

SHUTDOWN MARGIN - HORIZONTAL ROD SYSTEM

<u>Reactor Condition</u>	<u>Excess Lattice Reactivity %^(a)</u>	<u>Minimum Probable Rod Strength %</u>	<u>Shutdown Margin %^(b)</u>
Cold, Clean, Green	4.9	7.7	2.8
Hot (Turnaround) ^(c)	3.3	8.4	5.1
Hot (Turnaround) ^(c) Plus Cold Water	6.9	7.8	0.9
Hot (Turnaround) ^(c) Plus Flooding	5.5	6.2	0.7
Hot (Equilibrium)	1.0	8.4	7.4
Hot (Equilibrium) Plus Cold Water	4.6	7.8	3.2
Hot (Equilibrium) Plus Flooding	3.2	6.2	3.0

(a) Adjusted to give one per cent excess reactivity at equilibrium.

(b) Assuming no other control poison in the reactor.

(c) Ripe - 700 MWD/T.

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TABLE C-6

SHUTDOWN MARGINS - BALL 3X SYSTEM

<u>Reactor Condition</u>	<u>Excess Reactivity %^(a)</u>	<u>Minimum Probable Ball 3X Strength (%)</u>	<u>Shutdown Margin^(b)</u>
Cold, Clean, Green	4.9(c)	6.7	1.8
Hot (Turnaround)	0	6.8	6.8
Hot (Turnaround) Plus Cold Water	3.6(d)	6.7	3.1
Hot (Turnaround) Plus Flooding	2.2(d)	5.6	3.4
Hot (Equilibrium)	0(d)	6.8	6.8
Hot (Equilibrium) Plus Cold Water	3.6(d)	6.7	3.1
Hot (Equilibrium) Plus Flooding	2.2(d)	5.6	3.4
Hot (Scram Recovery)	5.1(e)	6.8	1.7

(a) Adjusted to give ~ one per cent excess reactivity at equilibrium.

(b) Assuming no shadowing due to control rods already inserted. Assuming no rods already inserted are ejected.

(c) Lattice excess reactivity. No control rods or other poisons inserted.

(d) Pile excess reactivity. Sufficient control rods in reactor to maintain reactor just critical at equilibrium or turnaround.

(e) Assumes all horizontal rods are withdrawn to make a scram recovery, reactor is scammed with Ball 3X and cools to xenon-free state.

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reactivity ramp and the reactor is automatically shut down by a safety rod scram (primary system). In this case, the speed of shutdown should be fast enough to limit any damage to mild fuel stressing only - definitely no boiling or fuel melting should occur.* This may at first appear to be a much more demanding requirement on the primary safety system than is now required in the older Hanford reactors. However, the possible reactivity insertion rates (poison withdrawal rates) in N Reactor are so much smaller than that possible from coolant loss in the older reactors that a "no-boiling" criterion actually is less restrictive than the no-melting criterion in effect in the speed-of-control interpretation in the present reactors. (As will be seen later, a major factor in this result is the strongly-negative prompt temperature coefficient in the NPR).

Secondly, there is the case where the control rods are withdrawn at their maximum rate (or reactivity is added by some other means - cold water accident, for example) and the primary safety system (safety rods) fail to enter the reactor. This is a more improbable accident (two failures or accidents are required), and the requirement that the backup system response must meet is set at no fuel melting. No fuel melting is taken to mean no uranium melting.

In order to assess the degree to which the safety systems meet both these speed-of-control requirements, it is first necessary to consider the sources and magnitudes of the reactor excursions. Then, particular excursions are examined to determine whether the criteria are indeed satisfied.

*From a reactor efficiency standpoint, any power transient should not stress the fuel elements to the point where a rupture rash would occur. Procedural rod withdrawal rates will be less than those maximally possible and will undoubtedly be governed by fuel rupture considerations.

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1.7.1 Reactivity Sources

The principal accidental sources of positive reactivity are (1) core flooding, (2) cold water insertion during equilibrium reactor operation and (3) excessive control rod withdrawal rates. Although these were considered in some detail in Volume 1, certain aspects were not covered in the earlier treatment. Principal among these omitted aspects is the possibility of larger reactivity ramps than those discussed in the companion volume due to one or both of two types of failures or errors occurring in the withdrawal of control rods. The two failures or errors are (1) increased rod withdrawal speeds due to inadvertent admission of air or gas into the hydraulic systems, and (2) steeper ramps resulting from maximum strength rods being involved in the rod withdrawal.

1.7.1.1 Control Rod Withdrawal

As stated in the companion volume, the maximum rate at which horizontal control rods can be withdrawn depends on the pumping capacity of the oil pumps serving the rod hydraulic systems. The possibility of air or some other gas leaking into the hydraulic lines has been considered since rod speeds could be increased by the presence of gas in the hydraulic system. The maximum reactivity ramp due to such a failure in the hydraulic system is believed to be 0.04 per cent per second. This compares to the 0.025 per cent per second maximum ramp given in Volume 1 for a normally-working hydraulic system.

The possibility that the particular rods chosen to be withdrawn together are the maximum-worth rods will also result in a reactivity ramp larger than 0.025 per cent per second. If the twelve rods in one end of the reactor (rear or front) were ganged together and

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withdrawn at the maximum speed (assuming normal operation of the hydraulic system) the reactivity ramp would be 0.062 per cent per second. Such a ganging will be prohibited by mandatory Process Standards and the occurrence is very unlikely. After operating experience is gained, the control rods will probably be grouped in permanent gangs and thus the flexibility of any ganging combination will only exist during the initial operating period of the reactor. The ganging of control rods before startup is considered unwise since the best control rod patterns can only be determined after some reactor operation has been achieved.

Assuming a combination of two circumstances, both hydraulic system failure (gas in the lines) and a withdrawal of the strongest rods at one time at full speed, would result in an average ramp of 0.10 per cent $\Delta k/k$ per second. This ramp is only half of that currently possible in the present Hanford reactors from the simultaneous withdrawal of all vertical safety rods at their maximum speed.

1.7.1.2 Cold Water Addition

The only time when cold water addition would produce a reactivity increase when the reactor is at equilibrium. The average coolant density at operating conditions is about 0.82 gm/cm^3 , and a sudden increase to density of about one by the substitution of cold emergency cooling water will produce an expected transient of about 0.036 per cent $\Delta k/k$ per second, assuming 100 seconds for the full temperature transient. This transient is on the same order of that associated with the maximum rod withdrawal coincidental with gas in the hydraulic systems.

1.7.1.3 Core Flooding

A bursting of a process tube during reactor operation will admit high temperature water into the graphite stack. If it is

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assumed that the water does not flash to steam but remains in a saturated state at low pressure in the hot stack and collects in the graphite in such a manner that the maximum reactivity effect is achieved (around 10 - 15 per cent of the available voids filled - those closest to the process tubes), an increase in reactivity of 2.2 per cent is calculated. The rate at which water could fill the stack to this amount from a single process tube leak (assuming a critical flow rate of 200 pounds/second*) is about 150 seconds. Thus, the average reactivity ramp from such a break would be about 0.015 per cent per second. A faster addition of water to the stack could only come from a multiple tube break which is considered very highly improbable.

1.7.1.4 Summary of Reactivity Ramps

For convenience a graphic summary of the possible positive reactivity ramps is shown in Figure C-14. Shown also are the potential positive reactivity increases in a present Hanford reactor (C Reactor).

1.7.2 Consequences of Excursions

1.7.2.1 Excursion With Scram

The safety philosophy requires that the primary safety control system has a response which is fast enough to limit any credible power excursion to one of a magnitude that no damage beyond mild fuel stressing will result. Since the magnitudes of the credible reactivity ramps in N Reactor are considerably smaller than in the older Hanford reactors, speed of control in N Reactor is not so severe a problem as it is in the present reactors. Nevertheless, the scram

*Only 130 pounds is liquid water.

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FIGURE C-14

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[REDACTED]

speed of the horizontal rods is very high; the rods reach 90 per cent effectiveness from their full-out position in about 1.2 seconds. A plot of the reactivity characteristic curve for the horizontal safety rods is given in Figure C-15. The initial delay includes instrument and circuit delays as well as the time required for the rods to enter the active zone of the reactor from their full-out position.

To test the meeting of the criterion on primary safety system response, reactor kinetic calculations have been made under a variety of assumptions. The scram response of the horizontal rods is shown to be more than adequate to shutdown the reactor from any credible excursion before any boiling occurs. A plot of a startup excursion due to a reactivity ramp of 0.05 per cent per second (which is equivalent to twice the normally-expected maximum reactivity insertion rate due to rod withdrawal) is shown in Figure C-16. As shown, the excursion is characterized by a sharp initial peak which, in this case, is turned around at about 11,000 MW (reactor power) by the combined action of the scram rods and the negative temperature coefficients. The scram signal is assumed to be given by the high-level neutron flux monitor (set to initiate a trip at 4400 MW - 110 per cent of equilibrium level*).

The scram rods continue to drive the power and temperatures down as shown in the figure. The peak fuel temperature is 240 F and the outlet coolant temperature peaks well below 212 F. Half-coolant flow is assumed in the startup excursion calculations; the loop pressure is 300 psia and

*This would be the last nuclear trip in the safety circuit. In other words, the calculation has assumed that all lower periods, power-rate and level trips failed.

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FIGURE C-15

CHARACTERISTIC CURVE FOR N REACTOR HORIZONTAL SCRAM RODS

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FIGURE C-16

STARTUP EXCURSION - 0.05%/SEC. RAMP NORMAL HSR CORE

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the reactor is initially 4.3 per cent $\Delta k/k$ subcritical.

Figure C-17 illustrates the case in which the excursion is assumed to occur at equilibrium level. The assumed reactivity ramp rate in this case is 0.025 per cent per second, which corresponds to the simultaneous movement of the maximum number of control rods limited by the pumping capacity of the hydraulic system and a balanced-flux rod withdrawal. The power variables in Figure C-17 is that for a nominal central tube operating initially at 4400 kw. The temperature increases are mild even though there is nearly a ten per cent increase in tube power. Boiling would occur at an outlet temperature of about 585 F.

If the safety rods fail to act, the Ball 3X system is required to shutdown the reactor from any credible excursion before fuel melting. The Ball 3X insertion rate is somewhat slower than originally expected and reported in Reference 3 and Volume 1 of this review, because of the mechanical construction of the ball hoppers and entry valves, and, also, because of the time required to fill the exit piping and ball columns. The full-pile reactivity effect on Ball 3X insertion is, therefore, characterized by an initial 2.4 $\Delta k/k$ drop due to the partially-black stream of balls falling during the period of filling the exit piping which is followed, by a gradually-increasing poisoning effect as the ball columns fill up. A full 38 seconds is required from Ball 3X trip to achieve the full effectiveness of the Ball 3X poisoning. A plot of the expected scram characteristic curve of the Ball 3X system is given in Figure C-18.

The kinetic calculations show that in spite of the slow-acting nature of the Ball 3X system, the initial 2.4 per cent $\Delta k/k$ effect is strong enough to adequately shut down the reactor from

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FIGURE C-17

EQUILIBRIUM EXCURSION - 0.025%/SEC RAMP
NORMAL MSR SCRAM

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FIGURE C-19

CHARACTERISTIC CURVE FOR BALL 3X SYSTEM:
(N REACTOR)

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any credible excursion if the primary system fails. A 1.5 second time from a 1X trip has been assumed for the circuitry of the Ball 3X system to sense failure of the rods to scram properly. The results of the calculations are shown in Figures C-19 and C-20. It is seen that peak fuel temperature achieved in the equilibrium excursion case is about 660 F. In fact, the outlet coolant temperature is also held about 25 F below the boiling point (585 F). This is an important point since once boiling occurs, a reduction in heat transfer rates could accelerate fuel heating. A much more complex analysis would be required to answer the question of whether the Ball 3X system satisfactorily meets the speed-of-control criterion if it became necessary to consider boiling and two-phase flow relationships.

A major reason the slower-acting Ball 3X system has time to adequately shut down the reactor is that the reactor, itself, has a powerful shutdown mechanism, the prompt negative fuel and coolant temperature coefficients. Thus, as soon as the reactor enters the significant power-producing range in the excursion, the reactivity-damping effects due to fuel and coolant heating are sufficient in themselves to prevent a runaway power excursion. Time is, therefore, provided during a subsequent, slower-acting, potential excursion resulting from a continuation of the ramp (combined with reactor heating) to permit the entry of Ball 3X balls and the termination of the power rise before boiling occurs. Figure C-19 clearly illustrates this behavior. The initial excursion is turned around solely by the prompt temperature coefficients.

1.7.2.2 Ultimate Shutdown Mechanism

Figures C-21 and C-22 illustrate the consequences of a nuclear excursion with failure of both automatic scram systems. The

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FIGURE C-19

STARTUP EXCURSION - 0.05%/SEC RAMP
RODS FAIL TO SCRAM NECESSITATING BALL 3X SCRAM

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FIGURE C-20

EQUILIBRIUM EXCURSION - 0.025%/SEC RAMP
RODS FAIL TO SCRAM NECESSITATING BALL BN SCRAM

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FIGURE C-21

STARTUP EXCURSION - 0.05%/SEC RAMP WITH FAILURE TO SCRAM

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FIGURE C-22

EQUILIBRIUM EXCURSION - 0.025%/SEC RAMP WITH FAILURE TO SCRAM

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1.8 Errata

Subsequent to the reproduction of the preceding sections, results of a more detailed study on cold water injection were received. These results indicate that in case of a primary coolant system dump, cold raw water may start to reach the core in 20 seconds and may completely fill the core in an additional 30 seconds. The ramp occurring between 30 and 50 seconds would amount to 0.12 per cent $\Delta k/k$ per second. This ramp of 0.12 per cent $\Delta k/k$ per second is larger than the maximum rate considered possible from rod withdrawal. However, the ramp covered by the cold water injection is not considered as severe as the "rod withdrawal" ramp. When the incident starts with the emergency river water flowing toward the reactor, there will be a trip of the Primary Safety Circuit (the horizontal rods) and the Ball (3X) Safety System, and there will be an elapse of 20 seconds before the reactivity ramp is initiated. Hence, failure of both safety systems would be required before a reactivity excursion could occur.

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2. DESIGN HEAT GENERATOR AND REMOVAL PARAMETERS

2.1 Design Heat Generation Parameters

The design split of the heat generation among the various reactor components is shown below in Table C-7.

TABLE C-7

SPLIT OF HEAT GENERATION

<u>Component</u>	<u>MW</u>	<u>Percent</u>
Inner Fuel Element	1160	25
Outer Fuel Element	<u>2600</u>	<u>69</u>
Total Fuel	3760	94
Process Tubes	40	1
Graphite	<u>200</u>	<u>5</u>
TOTAL REACTOR	4000	100

2.2 Design Process Tube Parameters

The design process tube parameters are listed in Table C-8.

TABLE C-8

DESIGN PROCESS TUBE PARAMETERS

(1) Number of Tubes	1004
(2) Tube Powers	
Average	3925 Kw
Average Control Zone	4427 Kw
Design "Hot Channel"	5000 Kw
(3) Tube Coolant Flow Rates	
Average	75,697 lbs/hr
Nominal Central Tube	79,903 lbs/hr

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(4) Heat Splits

Outer Element	73.5 Per cent
Inner Element	<u>26.5</u>
	100.0 Per cent
Outer Coolant Channel	44.5 Per cent
Middle Coolant Channel	46.2
Inner Coolant Channel	<u>9.3</u>
	100.0 Per cent
Outer Surface, Outer Element	44.5 Per cent
Inner Surface, Outer Element	29.0
Outer Surface, Inner Element	17.2
Inner Surface, Inner Element	<u>9.3</u>
	100.0 Per cent

(5) Flow Splits

Outer Channel	48.1 Per cent
Middle Channel	43.8
Inner Channel	<u>8.1</u>
	100.0 Per cent

(6) Coolant Temperatures

Reactor Inlet	369° F
Reactor Bulk Outlet	527
Average Central Zone Tubes	
Tube Bulk Outlet	535° F
Outer Coolant Channel Outlet	523
Middle Coolant Channel Outlet	544° F
Inner Coolant Channel Outlet	558° F

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Design "Hot Channel"

Tube Bulk Outlet	560° F
Outer Channel	547° F
Middle Channel	570° F
Inner Channel	586° F

(7) Specific Fuel Power

Average Tube

Total Fuel, Avg.	109.5 Kw/ft, 11.17 Kw/Kg
Inner Element, Avg.	33.8 Kw/ft, 10.37 Kw/Kg
Outer Element, Avg.	75.7 Kw/ft, 11.55 Kw/Kg

Average Central Zone Tube

Total Fuel, Avg.	122.9 Kw/ft, 12.54 Kw/Kg
Inner Element, Avg.	37.9 Kw/ft, 11.65 Kw/Kg
Outer Element, Avg.	85.0 Kw/ft, 12.96 Kw/Kg

Design "Hot Channel" Tube

Total Fuel, Avg.	138.7 Kw/ft, 14.17 Kw/Kg
Inner Element, Avg.	42.8 Kw/ft, 13.17 Kw/Kg
Outer Element, Avg.	95.9 Kw/ft, 14.65 Kw/Kg

Total Fuel, Peak	206.5 Kw/Ft, 21.11 Kw/Kg
Inner Element, Peak	63.5 Kw/ft, 19.60 Kw/Kg
Outer Element, Peak	142.7 Kw/ft, 21.82 Kw/Kg

(8) Uranium Temperature

Average Central Zone Tube

Inner Element, Maximum	811° F
Outer Element, Maximum	749° F

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Hot Channel Tube

Inner Element Maximum	879° F
Outer Element Maximum	803° F

(9) Heat Fluxes

Average Central Zone Tube, Peak Values

Process Tube	53,000 BTU/hr/sq ft
Outer Surface, Outer Element	441,000
Inner Surface, Outer Element	518,000
Outer Surface, Inner Element	393,000
Inner Surface, Inner Element	505,000

Design "Hot Channel" Tube, Peak Values

Process Tube	60,000 BTU/hr/sq ft
Outer Surface, Outer Element	498,000
Inner Surface, Outer Element	585,000
Outer Surface, Inner Tube	444,000
Inner Surface, Inner Tube	570,000

2.3 Design Heat Balances

Heat balances are shown below in Tables C-9, C-10, and C-11 to cover the reactor, the primary coolant system, and the secondary coolant system.

TABLE C-9
REACTOR HEAT BALANCE

Input

Fuel	3760 MW
Tubes	40
Graphite	<u>200</u>
TOTAL	4000 MW

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Removal

Primary Coolant System	3935 MW
Moderator Coolant System	60
Thermal Shield	4
Control Rods	1
Gas System	.2
Heat Losses	<u>.3</u>
TOTAL	4000 MW

TABLE C-10

PRIMARY COOLANT SYSTEM HEAT BALANCE

Input

Reactor	3935 MW
Pumps	<u>30</u>
TOTAL	3965 MW

Removal

Steam Generators	3930 MW
Coolant Feed and Bleed	31
Heat Loss From Piping	<u>4</u>
TOTAL	3965 MW

TABLE C-11

SECONDARY COOLANT SYSTEM HEAT BALANCE

Input

Steam Generators	<u>3930 MW</u>
TOTAL	3930 MW

Removal

Dump Condensers	3604 MW
Heat Exchanger Blowdown	3
Drive Turbines, Heating & Misc.	<u>323</u>
TOTAL	3930 MW

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3. REACTOR COOLING

3.1 General

The NPR primary piping system is subjected to a wide variety of stresses under normal steady-state operating conditions and during the various transients which accompany a reactor scram and emergency cooling water injection. Selection of materials and the design of all piping components has been guided by the various applicable codes as discussed in Section 6.3 of the first volume, and by engineering judgement based on actual tests and mathematical analyses where specific applications are not covered by experience and codes. Adequacy of final design of the primary system has been analysed by mathematical techniques, including use of computers, and by laboratory testing of individual components. Prior to reactor startup, considerable testing will be performed on the as-built system by use of Acceptance Test Procedures and Operational Tests which are designed to confirm system operability in accordance with the Criteria (8) and design intent.

Analyses and tests having a bearing on reactor safety have involved specific consideration of the material of construction for process tubes, rupture and plugging of process tubes and primary piping, thermal shock from scram transients and emergency water injection, primary piping mechanical coupling, failures, primary coolant thermal contraction effects, coolant flow transients, emergency cooling system reliability, and the general response of the secondary system to transients occurring during normal operation and following reactor scram. Pertinent findings from these tests are summarized in this section.

3.2 Process Tubes

The NPR process tube will be subjected to stresses from internal pressure and thermal gradients across the tube wall resulting from heat

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transfer from the graphite to the primary coolant. The stresses resulting from these conditions must not be allowed to exceed the allowable values which are based on a maximum of one per cent creep in 20 years of tube operating life. This allowable stress is shown in Figure C-23.

The temperature distribution and stresses in the process tubes at various locations in the active zone have been calculated for various assumed neutron flux distribution and positions of the tube within its graphite channel. Comparisons of calculated with allowable stresses are shown in Figures C-24, C-25, C-26, C-27, C-28 and C-29. In all cases, the calculated process tube thermal and internal pressure stresses are below maximum allowable values.⁽⁹⁾

Stress analyses of the zircaloy-2 process tubes under conditions of temperature transients more severe than those expected during dump and emergency cooling water injection have shown that neither a step decrease of inside tube-wall temperature of 100 degrees F or a ramp decrease of 150 degrees F in three seconds will produce stresses in excess of about 80 per cent of the 0.2-per-cent yield strength of vacuum - annealed Zircaloy-2.⁽¹⁰⁾

Additional factors relating to the operational safety of process tubes were examined to demonstrate the adequacy of design. Both experiment work (see Section 6 of Volume 1) and mathematical techniques have been used to evaluate potential for flow instability and fuel-element burn-out. The mathematical treatment has shown that the 5000-Kw maximum tube power for 4000-MW design operation is well within accepted safe operating limits. These calculations have indicated that conditions leading to burn-out will not exist until either tube power is increased 13 per cent or tube flow is decreased 11.5 per cent from design conditions.

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FIGURE C-23

ALLOWABLE STRESSES FOR ZIRCALLOY-2

(Based on 1% Allowable Creep During 20 Years Service)

FIGURE C-24

PROCESS TUBE AT FIRST (DELET) TRUNNION (COSINE FLUX DISTRIBUTION)

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FIGURE C-25

PROCESS TUBE AT FIRST (INLET) TURNION (FLUX PEAKED UPSTREAM)

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FIGURE C-26

PROCESS TUBE AT LAST (OUTLET) TRUNNION (FLUX PEAKED UPSTREAM)

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FIGURE C-27
PROCESS TUBE AT LAST (OUTLET) TRUNNION (FLUX PEAKED UPSTREAM)

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FIGURE C-28
PROCESS TUBE TOUCHING AT COOLED TRUNION (CENTER OF ACTIVE ZONE AT 4,000 IN, 226 Kw/ft)

FIGURE C-29

PROCESS TUBE TOUCHING MIDWAY BETWEEN TRIMMIONS (CENTER OF ACTIVE ZONE AT 4,000 IN, 226 Kw/Ft)

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Critical process-tube ruptures and plugging incidents were postulated to permit estimation of the magnitude of flow variations to be expected. For tube-rupture calculations, a worst-case assumption of zero flow in the downstream outer fuel element coolant channel was assumed. The resulting curves shown in Figure C-30 based on these assumptions show that tube flow will increase immediately following the failure for ruptures at all locations with a minimum flow increase of about seven per cent for a tube rupture at the outlet of the active zone. Since flow beyond the rupture is assumed to be zero and the outer fuel cylinder would, therefore, be steam blanketed, a limiting case was examined on the basis of allowable uranium and Zircaloy-2 temperatures. The results indicate that high-flow trip settings of five per cent above normal flow would be required based upon allowable Zircaloy-2 temperatures and eight per cent above normal based upon the allowable uranium (α -B transformation) temperature. Both settings are undesirably close to normal flow, especially when allowance is made for instrument error. A tube-rupture flow-trip setting at ten per cent above normal flow, including a two-per-cent instrument-error allowance, is reasonable and gives a maximum calculated uranium temperature not above 1300 F. A manual scram of the reactor could be initiated at any time based upon an alarm annunciated by the moisture detector in the helium gas system. Transient temperature calculations for the worst case of tube rupture followed by a reactor scram indicate that the uranium will approach the beta-gamma transformation temperature (1420 F) as shown in Figure C-31.⁽¹¹⁾

Flow reductions caused by obstructions in the flow channels of the process tubes were investigated analytically to determine the effect on uranium fuel temperature and adequacy of protective measures. For certain plugs developing in the outer or middle annuli, flow instability

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FIGURE C-30
HYDRAULIC DEMAND CURVES FOR VARIOUS RUPTURES WITHIN ACTIVE ZONE

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FIGURE C-31

OUTER FUEL CYLINDER TEMPERATURE RISE AFTER PROCESS TUBE RUPTURE

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should result with only a five-to-six per cent flow reduction, but drastic flow reductions are expected to follow any cases which might lead to high uranium temperature. However, burnout conditions in the center channel can follow with only a four-per-cent reduction in tube flow, and a total plug would result in only a seven-per-cent reduction. Thus, a low-flow trip is not feasible for protection against burnout following partial plugging of flow channels. Without a scram, maximum uranium temperature in the center section could approach 1715 F as shown in Figure C-32. Continued operation may result in fuel element rupture with subsequent release of activity to the coolant loop and annunciation of the rupture monitor alarm. For the other annuli, use of trip after instability should result in maximum uranium temperatures not in excess of the alpha-beta transformation temperature (1242 F) as shown in Figure C-33. Hydraulic demand curves for the various degrees of plugging in the coolant annuli of the fuel elements are shown in Figures C-34 and C-35.⁽⁴⁾

Further experimental thermal-hydraulic studies are planned to determine the flow behavior and transient temperatures to be expected as a consequence of partial plugging of flow channels. These studies will be performed by use of full scale mockup of the NPR process tube.

3.3 Primary Piping

Flow and pressure and temperature transients occurring in the ex-reactor primary coolant system have been analyzed from the viewpoint of stress and fatigue and effect on process. Many conditions have been studied ranging from the normal scram transients to significant pipe ruptures and loss of the heat removal system.

The principal operating cycle of the NPR is the startup - equilibrium - shutdown sequence. Analyses of reactor inlet and outlet piping

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FIGURE C-32

INNER FUEL CYLINDER TEMPERATURE RISE AFTER MIDNITE CHANNEL PLUG

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FIGURE C-33
FUEL ELEMENT TEMPERATURE AFTER TRIP-AFTER-INSTABILITY

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HYDRAULIC DEMAND CURVE WITH CENTER FUEL ELEMENT FLOW CHANNEL PLUGGED

FIGURE C-34

FIGURE C-35

HYDRAULIC DEMAND CURVES WITH PLUG AT OUTLET OUTER FLOW CHANNEL

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stresses (thick-walled vertical headers and long connectors) have conservatively assumed that 1000 shutdowns would take place at scram speed, resulting in coolant outlet temperature changes on the order of 80 F in 20 seconds. Another operating transient is that occasioned by use of the emergency cooling system which injects low temperature (150 F) water in the event of loss of pumping power, loss of heat removal capacity, or major piping rupture. Use of the last ditch cooling system is expected to occur only a few times, if at all, during the reactor life. However, six rapid depressurizations (dumps) have been assumed for design and fatigue analyses.

Fatigue analyses have taken into account the fluctuating stresses resulting from changes in fluid temperature and pressure as well as the effect of stress concentrations. The fraction of the useful life of inlet and outlet piping which would be expected to be used up during a design life composed of cycles described above is one per cent.⁽¹²⁾

The analytical technique used for fatigue analyses on the primary piping embodies most of the features set forth in Reference (13). The principal technique involves the use of a modified Goodman diagram on which alternating and steady stress intensities are plotted instead of principal stresses. (The stress intensities are the differences between the principal stresses). For a triaxial stress condition, this corresponds to application of the "maximum shear" theory of failure instead of the "maximum stress" theory. These diagrams for the outlet connectors and outlet connector-vertical header joint are shown in Figures C-36 and C-37.

More extensive studies of thermal shock and fatigue effects on primary piping are currently being made by the Electric Boat Division of General Dynamics Corporation under Contract AT(45-1)-1507. Reactor scram

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FIGURE C-36

FATIGUE DIAGRAM FOR OUTLET CONNECTOR

KEY

- - SCRAM CYCLE
- - EMERGENCY COOLING CYCLE

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FIGURE C-37

FATIGUE DIAGRAM FOR OUTLET RISER-CONNECTOR JOINT

KEY

- - SCRAM CYCLE
- - EMERGENCY COOLING CYCLE

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transient data (based on full rod scram from Phase I equilibrium operation) which form the bases for these studies are shown on Figures C-38, C-39, C-40, C-41 and C-42. Emergency cooling transient data (based upon a complete circumferential rupture of the inlet mixing header under worst case (Phase III) conditions) are shown on Figures C-43 and C-44 for the emergency cooling transient. It is to be noted that Figure C-44 does not give the effects of cold emergency water, but is indicative of average thermal conditions in the hot and cold legs of the primary cooling system and is based on maintaining saturated coolant temperatures corresponding to the system pressure transient of Figure C-43. The critical temperature transients will exist at the base of the inlet vertical headers, the emergency cooling nozzles, and the lower inlet connector nozzles which are the first to receive the cold coolant. These temperatures and flow transients are shown in Figure C-45. The emergency coolant flow transient shown in Figure C-45 is based on bringing two 13,600-gpm emergency cooling pumps into full operation against the falling-system-pressure curve of Figure C-43 and the hydraulic characteristics of the emergency cooling system.⁽¹⁴⁾

The NPR design includes the use of more than 7000 two-bolt mechanical joints in inlet and outlet connectors from the individual process tubes, and about 26 six-inch couplings on the outlet vertical headers. With so many of these joints, each subject to failure under pressure by the loss of one holding bolt, it is important to be certain that the consequences of such a failure are known and can be accommodated.

Mathematical analyses⁽¹⁵⁾ of the consequences of mechanical coupling failures in process tube connectors have shown that:

1. For a failure upstream of a process tube, reverse flow is adequate for cooling fuel elements in a maximum power tube except for the

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FIGURE C-38
OPERATING PRESSURE AT REAR RISERS AFTER SCRAM

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FIGURE C-39

OUTLET CONNECTOR COOLANT TEMPERATURE AND FLOW DURING SCRAM

* TO OBTAIN FLOW/outlet conn, $\div 1004$
" " " " vert hor, $\div 16$

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FIGURE C-40

OUTLET CONNECTOR AVERAGE METAL TEMPERATURE DURING SCRAM

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FIGURE C-41

OUTLET VERTICAL RIVER AVERAGE COOLANT AND METAL TEMPERATURE DURING SCRAM

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FIGURE C-42

PRIMARY COOLANT TEMPERATURE AND FLOW TRANSIENT IN DIVERSION SYSTEM FOLLOWING SCRAM

- * MAX RATES OF TEMP CHANGE DOWNSTREAM OF DIVERSION HDR NOZZLES WILL NOT EXCEED TEMPERATURE CHANGES SHOWN.
- ** MAX FLOW/DIVERSION HDR = 5 TIMES ORDINATE ON CHART.

FIGURE C-43

PRIMARY LOOP PRESSURE DURING COOLING, INLET HEADER RUPTURE PHASE I & III CONDITIONS

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FIGURE C-44

PRIMARY COOLANT TEMPERATURES INLET HEADER RUPTURE PHASE III CONDITIONS

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FIGURE C-45

EMERGENCY COOLANT FLOW AND TEMPERATURE AT NOZZLE AT BASE OF INLET VERTICAL HEADER
INLET HEADER RUPTURE PHASE III CONDITIONS

* FLOW/INLET VERT HDR = ABOVE FLOW ÷ 16

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first ten seconds during which time the fuel element temperature will rise somewhat but no uranium melting will occur. Coolant exit quality will be in the neighborhood of nine per cent and bulk boiling will occur over about 15 feet of the active zone but without film boiling.

2. The makeup injection system capacity is sufficient to supply leakage due to a failure after loop pressure falls to about 900 psia, and use of the emergency cooling system will not be required.

3. Less-than-complete failure of a coupling (partial failure which provides leakage equivalent to that from a 0.25 to 0.75 inch circular hole) can result in flow decay to zero in about 30 seconds with stagnant conditions maintained indefinitely thereafter.

Analyses of failures of six-inch couplings on the outlet vertical headers indicate that the dump valves will open about 100 seconds after the failure. During the transition it is expected that bulk boiling will occur in the maximum power tube over about 20 feet of the active zone with the possibility of film boiling beginning as early as 38 seconds following failure. In this event fuel elements will approach the Q-B transformation temperature but no fuel element melting should occur. A plot of system pressure decay and leakage for this case is shown in Figure C-46. ⁽¹⁵⁾

Because of the facts that (1) emergency cooling water is introduced at the base of the inlet vertical headers, and (2) each of these headers serves as a coolant distributing manifold for about one-sixteenth of the process tubes, the consequences of a rupture of such a header was analyzed mathematically to determine what changes in process condition might be anticipated and what effect such a failure might have on emergency coolant injection.

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FIGURE C-46

SYSTEM PRESSURE DECAY FOLLOWING A 6 INCH RISER COUPLING FAILURE

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The results of this study indicated that an immediate scram would occur as a consequence of a low-flow signal from affected process tubes. Outlet vertical header pressure would decrease to about 860 psia within about one second and opening of the dump valves would take place after 20 seconds. Emergency coolant would begin to enter the reactor after about 53 seconds when the pressure has decreased to about 350 psi. The entire loop would be filled with water below the boiling point at about 300 seconds. Two-phase flow should exist in the reactor core during the transient, but no damage to fuel elements was indicated by the study. Changes in the process during the rupture and dump transient are shown in Figures C-47 and C-48. (16)

3.4 Coolant Thermal Contraction

One consequence of the rapid shutdown of heat generation, which occurs at the time of a reactor scram, is a rapid contraction of primary coolant. Since the NPR primary loop is a closed system, contraction volume must be supplied by the injection system and the water inventory of the pressurizer under conditions which will prevent bulk boiling of coolant. The scram transient has been studied to ensure that the final design of the NPR will adequately meet the criterion of no bulk boiling.

The analyses are based upon scram from Phase I equilibrium operating conditions and include investigation of the effects of various departures from "normal" system operation, i.e., one or more injection pumps out of service, variations in rate of achieving steam generator bypass, variations in pump coast-down rate, failure of some steam-generator cells to effect bypass or pump coast-down, and simultaneous (with scram) decreases in secondary system steam pressure. Pertinent conclusions resulting from these investigations are given below. (17)

FIGURE C-47

BLOWDOWN TRANSIENT

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(1) The normal scram sequence based upon the flow and heat absorption transients of Figures C-49 and C-50 will prevent draining of the pressurizer even with only two injection pumps available for makeup as shown in Figure C-51. Even small delays in initiating flow coastdown (two seconds) and small decreases in secondary cooling system steam pressure (to 73 psia) can be tolerated as shown in Figures C-52 and C-53.

(2) Failure to achieve bypass in three cells will cause draining of the pressurizer if only two injection pumps are in service as shown in Figure C-54. To preclude the occurrence of this condition methods for issuing proper control are being studied.

(3) Failure to achieve flow coastdown in all five loops will cause draining of the pressurizer even with four injection pumps available for makeup as shown in Figure C-55. Of the cases studied, this is the only one that would cause flashing of primary coolant during the scram transient. For this reason, the scram flow coastdown signal has been made fail-safe.

3.5 Emergency Cooling

Because of the importance of the NPR emergency "last ditch" cooling system in satisfying the requirements of the Safety Criteria for Hanford Reactors, much thought and attention has been given both to the design of the system and to investigations into its adequacy. A general description of the system is presented in Section 6.3.4 of Volume 1 of this document, and only matters relating to system adequacy and reliability are discussed in this appendix.

The emergency cooling system is intended to function upon the occurrence of any of the following:

1. Medium or large size primary loop ruptures (low pressure, low flow).

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FIGURE C-49

REACTOR PRIMARY FLOW & HEAT GENERATION FOLLOWING SCRAM

FIGURE C-50

PRIMARY FLOW THROUGH STEAM GENERATORS FOLLOWING SCRAM USING 1/3 BYPASS IN 14 SECONDS

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FIGURE 1

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PRESSURE RESPONSE DURING SCRAM FLOW COASTDOWN DELAYED BY TWO SECONDS

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2. Failure of the secondary cooling system (high pressure, high temperature).
3. Loss of primary pumps.
4. Opening of dump valves from any cause.

Instrumentation provides for a reactor scram which would normally precede the introduction of once-through emergency water by several seconds upon the occurrence of any of the above-listed incidents. Conversely, instrumentation provides for starting the emergency cooling pumps simultaneously with each scram so that the system will be in a state of readiness.

The maximum pressure to which emergency system piping is subjected is the shutoff head of the pumps (355 psig) but system pressure subsequently decreases to less than 200 psig when the emergency water flow is established through the reactor piping. The piping is adequately designed to withstand these pressures as well as additional lateral seismic forces of 0.5 g. Special provisions are made at building wall penetrations to prevent the pipe from being subjected to shearing forces resulting from earth movement. All high-pressure portions of this piping system (360 and 400-psig design pressure) are fabricated of pipe made according to ASTM A155 with material conforming to ASTM A212 Grade B firebox quality. This high grade material is the same as that utilized for the primary coolant loop.

Parallel pumps and parallel and redundant instrumentation circuits are used to provide high system reliability. The emergency pump starting circuits are failsafe. Oil for the diesel engines is supplied from two sources, and the engine starting air system consists of parallel headers and compressed air tubes with check valves to guard against loss of tank air. Only a single pipeline has been provided to supply emergency water, but a review of the system has indicated that the piping is very probably

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more reliable than the highly reliable pump and instrumentation systems.

A review of the expected transient performance of the emergency cooling system has shown that once the master trip occurs, the emergency water inlet and dump valves start to open simultaneously with the dump valves full open in about six seconds and the inlet valves full open in about 15 seconds. Opening time is adjustable. Primary loop pressure is relieved through the dump valves and through the opening in the primary loop pipe, if ruptured. Check valves prevent blowing primary coolant through the open inlet valves back into the inlet emergency water piping. Because valves are normally set for routing coolant to the dump tank, the rupture disk at the inlet to the tank will break almost immediately and primary loop water will into the piping leading to the tank and then into the tank. The primary coolant will be flashing as it enters the dump tank, but the quench water in the tank will condense the steam almost instantaneously and cool the liquid phase and condensed steam, thus dissipating the energy of the primary coolant. The once-through water starts to flow when the loop pressure has fallen to about 330 psig at the vertical inlet headers. The once-through flow rate will increase as pressure is continuously relieved through the dump valves.

One low-lift pump operating at 650 RPM provides ample emergency cooling and fog spray water for most emergency conditions. If there is an unusual demand for emergency cooling water, the level of water in the raw water storage tank might fall. Level can be recovered in a number of ways. The quickest method is to decrease the speed of the high-lift diesel engines by remote adjustment of the governor or by reducing the number of high-lift pumps in service. A second, but less desirable method, is to close one or more dump valves. If practical, it is also possible to start a second low-lift diesel. All of these changes can be made from the 105-N control center.

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Speed regulation of the high-lift pumps is required to meet the system demand curves without pump cavitation or exceeding the driver power because the system curves continually change with time due to power decay of the reactor. The pumps are self-regulating for this purpose. The required shut-off head of 770 feet is achieved at a pump speed of about 1460 RPM. Speed will be held constant by the diesel engine governor as the flow rate increases from shut off until the brake horsepower equals the diesel engine rated power. At this time, the flow rate is about 7000 gpm for each pump, and the engine will start to slow down because of the high load. Speed will decay until the maximum engine power at that speed exactly balances the pump power required at the intersection of the system demand curve and supply curve at the new pump speed. This self-regulation continues until the system curve for an all-liquid system is established and steady-state conditions exist. The engine throttle will be set before reactor startup to give satisfactory fuel economy and operation over the entire range of operation.

High-lift pump operation is illustrated in Figure C-56.

Initial temperature of the once-through water is 150 degrees F to avoid excessive thermal shock to reactor piping. About 2×10^6 pounds of 150-degree F water contained in the storage tank and piping to the reactor will enter the reactor if there is no mixing of cold river water with the 150 degree F water in the storage tank. This water, when heated by reactor decay heat, will be well above the saturation temperature at atmospheric pressure. However, as the temperature of the water in the tank approaches river water temperature, the temperature of the once-through water leaving the reactor will be low enough to permit the coolant to be diverted safely to crib.

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FIGURE C-56

HIGH LIFT PUMP PERFORMANCE

Basis: Pump Data Given in P. O. 6463

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FLOW RATE, GPM FOR ONE PUMP

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Diversion to crib will be made automatically. Pressure and temperature switches just upstream of the two valves used to route flow to either tank or crib will trip a relay which in turn actuates both valves simultaneously. The coolant may be diverted to crib when the pressures and temperatures at the inlet to both valves are equal to or less than, say, 200 psig and 250 degrees F. respectively. The combination of pressure and temperature trips are selected to prevent exceeding the design conditions of the low-pressure piping downstream of the valves and to prevent excessive flashing in the piping and at the crib for any and all accidents. The trip settings are locally adjustable. Instrumentation is such that the valves remain in the "as is" position if there is an electrical supply failure.

The dump tank is sized to hold primary loop water, initial once-through water, and cold quench water for a maximum final mixed volume temperature of 190 degrees F, based on a maximum initial quench water temperature of 70 degrees F. If both selector valves fail to divert to the crib, the once through water continues to flow to the quench tank and then overflows from there to the river. The overflow is sized to handle the full flow of two high-lift pumps for an all liquid system; the overflow is located in the dump tank such that the tank holds at least 148,000 cubic feet of water before overflowing commences. Dump tank liquid-level instrumentation, which includes an indicator and level alarms, is provided in the 105-N Control Room.

Emergency cooling would normally continue until supervision has made provision for cooling the reactor by other means and has decided that once-through cooling is no longer needed. At this time, the inlet and dump valves may be closed (depending on new method of cooling) and taken off automatic control, and the low-lift and high-lift pumps may be shut

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off manually from the 105-N Control Room. The position of the path selector valves is optional at this time.

A study of the reliability of the inlet valve and dump valve electrical circuitry has shown that if the electrical parts are tested for non-fail-safe failures every six months, and if components are replaced before they reach wear-out age, odds in favor of continuous process protection, i.e., no period of failure during a month, will be at least 50:1. If the electrical parts are checked monthly these odds increase to 333:1. If the automatic system does not actuate in the event of the occurrence of one of the four listed incidents, there is about one chance in 5000 that an attempt to actuate the system manually will fail. There is about one chance in 100,000 that an individual valve will fail to actuate when required to do so. (19)

A mathematical analysis of the flow distribution of emergency coolant on a tube-to-tube basis was performed to ensure that all tubes would be adequately cooled in spite of the sharply reduced pressure and flow capacity available as compared with the normal primary system. The calculations indicate that under the worst condition, the minimum process tube flow rate during the transient is about 0.35 lb/sec. which occurs during cooling with 150 F water at 16 per cent reactor power (50 seconds following scram from 4000 MW equilibrium power). At this flow rate, a part of the fuel - element charge will be cooled by superheated steam. However, even if this "worst case" should have an opportunity to attain equilibrium, the calculations indicate that fuel-element melting will not occur. However, it is not until reactor power has decayed to 6.3 per cent (about 26 minutes following scram) that the bulk coolant flow will be saturated liquid in the outlet vertical headers. Finally, when the reactor power

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has decayed to 5.7 per cent (about 30 minutes after scram) the outlet coolant would be saturated liquid at atmospheric pressure. The hydraulic demand curves shown in Figure C-57 show that the entire reactor is adequately cooled even where inlet water temperature is about 367 F. A "burnout" condition is not indicated for the average tube, since maximum heat flux at 16 per cent power is less than 100,000 Btu/(hr)(ft²) and maximum fuel-element temperature in an average tube should not exceed 560 F. (20)

The enthalpy of the bulk and hottest-tube emergency coolant at the various shutdown reactor power levels are shown in Figure C-58 as are the saturated liquid and steam enthalpies at the conditions existing in the outlet vertical headers after equilibrium is established. These curves indicate that the hottest process tube is cooled by superheated steam for reactor power levels greater than 14 per cent. Some steam will exist in the hottest tubes until the reactor power has decayed to about 6.3 per cent. However, the bulk coolant is expected to remain subcooled liquid after reactor power has decayed to about 14 per cent. (20)

4. HEAT DISSIPATION SYSTEM CONTROL

4.1 General

Section 6 of the first volume of this Hazards Review contains a description of the heat dissipation system and a brief outline of the instrumentation for control of heat removal. Since the issue of the first volume, considerably more detail on the control system has been developed; hence, it is now possible to present a more complete account of the system control plus information on steps taken to promote system reliability.

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EMERGENCY COOLING SYSTEM HYDRAULIC DEMAND CURVES

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FIGURE C-58

COOLANT ENTHALPY CHANGE DURING EMERGENCY COOLING WITH 150° F INLET WATER

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4.2 Basic Components

The various control systems are composed of a selection of basic components arranged in a manner to fulfill the required function. These basic components are discussed below:

4.2.1 Adjustable Power Supply, Adjustable Bias Supply, and Fixed Power Supply

These units rectify a regulated 118 volt input to deliver a smooth d-c output for various services. The adjustable power supply and the adjustable bias supply furnish a fixed or variable voltage output, the fixed supply furnishes a fixed voltage output.

4.2.2 DC Operational Amplifiers

These units receive an input signal and supply an output signal. Depending on the particular feedback and input network, the amplifier may perform one of a variety of functions. Among these are:

Time Lag - The output has a potential opposite in sign to the input. On a step change in input, the output changes linearly with time until it reaches the same absolute magnitude.

Proportional Action - These units have an output of a polarity negative to the input and of a magnitude proportional to the input. The absolute magnitude is proportional to the "gain" which is variable.

Summing Units - These units can take multiple inputs and produce a single output. The output voltage will be the negative algebraic sum of the individual input voltages times individual "gains".

Rate Action - These units supply an output voltage proportional to the time rate of change of the input voltage.

Proportional Plus Rate Action - Output voltage has a magnitude proportional to the input voltage times the gain plus the rate

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of change of input voltage times a time constant. The output polarity is opposite to that of the input.

4.2.3 Demodulator

The demodulator is an instrument that produces a d-c output with a magnitude depending on the position of the core in a movable core transformer.

4.2.4 Servo Follower, Servo Multiplier

The servo follower is a device in which the output voltage is continuously compared with the input in order to reduce any difference to zero. When differences in voltage do occur, a circuit is unbalanced, the rotor of a motor is caused to move and a potentiometer on the motor shaft is turned to change the output to again equal the input. The follower is used to continually cause manual controls to track automatic operation. On shift to manual operation, the output is applied to an electric-pneumatic converter (see Section 4.2.8 for description of the converter) to provide control action without any system upset.

The multiplier is similar to the follower except two input voltages may be impressed and the output is proportional to the product of the two inputs.

4.2.5 Auto-Balance Selector Station

These devices are used to transfer a control system from manual to automatic or automatic to manual without requiring manual balancing of the controllers and manual signals before the transfer. These stations may utilize a servo follower (see Section 4.2.4 of this appendix) if the final control element is pneumatic.

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4.2.6 Transfer Relay Modules

Transfer relay modules are used in conjunction with auto-balance selector stations to transfer between manual and automatic control.

4.2.7 Manual Loaders

Manual loaders are devices which produce an output d-c current signal proportional to a manual dial setting or to an applied voltage signal.

4.2.8 Electric Pneumatic Converter

The electric pneumatic converter is used to convert a d-c voltage input into a proportional pneumatic control pressure.

4.2.9 High - Low Alarm Monitor

This is a device used for monitoring a signal to provide energization of an alarm should the signal deviate from a preset range. It may also be used to cause a transfer to manual from automatic control.

4.3 Control System Design

Schematic diagrams of the control system are shown in Figures C-59 through C-67. There are many design features common to all control systems which promote system reliability. The high-low alarm monitors, called summing point monitors, are used extensively in all control systems. These monitors are used to detect control component failures, alarm and identify the failed component, and, where desirable, to take a corrective action such as transfer of control to manual operation.

Servo followers and servo multipliers are used extensively in conjunction with the high-low monitors so that the manual control ordinarily "tracks" the automatic control. Therefore, transfer from automatic to manual will not cause a system disturbance.

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The more critical components in the system are duplicated so that a single component failure will not result in loss of the complete control system. Examples of components that are duplicated are sensors which supply a critical control signal and controllers which provide a function that cannot be practically duplicated by manual control.

The use of a single selection of basic components for all of the heat dissipation control system is another factor promoting reliability. With such a single set of components, a better knowledge of system characteristics by operating and servicing personnel would be expected.

The manner in which this basic design approach, using the various standard components, is employed for the various operating control schemes is discussed in the following section.

4.4 System Control During Reactor Operation

4.4.1 Primary Coolant System Operation

During reactor operation there will be direct control of the primary coolant temperature, pressure, flow, and inventory. In addition, values of other system parameters needed for operation of the plant will be indicated in the control room.

4.4.1.1 Reactor Coolant Inlet Temperature Control

During steady-state operation, the reactor coolant inlet temperature will ordinarily be maintained at a constant value to reduce power fluctuations occurring as the result of the effects of coolant and fuel temperatures on nuclear reactivity. The reactor inlet temperature is sensed downstream of the five primary pumps, averaged, and compared with a "set point". Deviations are used as control signals to regulate the steam pressure in the secondary sides of the steam generators, thus

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correcting the inlet temperature. For example, if the reactor inlet temperature rises above the set point, the deviation is used to signal a decrease in the steam pressure. This decreased steam pressure results in a decrease in secondary side temperature increasing the amount of heat removed from the primary coolant and reducing its temperature.

As an additional compensation for rapid temperature transients, the reactor outlet temperature is measured at the inlet to each of the five steam generator cells, averaged, and used to adjust the set point for the steam pressure controller. This will effectively anticipate the response required to maintain a constant inlet coolant temperature to the reactor.

During periods when large changes are made in the reactor power level, the constant reactor inlet coolant temperature mode of control breaks down. The steam pressure can be allowed to vary over only a limited range because of the design requirements of the drive turbines. If, for instance, a relatively large increase in reactor power is made during operation with constant inlet temperature control, the steam pressure will automatically be reduced in an attempt to prevent an increase in the inlet temperature. However, if the steam pressure drops to too low a value, the pump drive turbines will slow and reactor coolant flow rates will decrease.

Because of this factor, an alternate method of control is available during startup and shutdown transients. In this mode, the steam pressure is held constant and the reactor inlet coolant temperature will vary. The nuclear reactivity effects of this varying coolant temperature will be overridden by the movement of the control rods made to change the power level and at these times there will be no need

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for constant inlet temperature control.

Figure C-59 is a schematic diagram of the primary coolant temperature control system. The use of the various basic components and the application of the general design philosophy is well illustrated in the diagram. Summing point monitors are used to detect and alarm failure of all key amplifiers in the main control circuit (at the right of the figure).

4.4.1.2 Primary Coolant System Pressure Control

Pressure in the primary coolant system is maintained by the on-line, boiling-water pressurizer. Water temperature, and hence steam pressure, is regulated by electric immersion heater banks and by a spray of water from the cold leg of the primary coolant system. A control bank of heaters will ordinarily operate continuously and the sprays intermittently. Larger capacity heaters and spray units are available as backups for control during the more severe transients. A vent system is provided to prevent accumulation of non-condensable gases which would adversely affect the ability to control pressures. The pressurizer is hydraulically connected to the remainder of the primary coolant loop at the outlet manifold located in the 109-N Building Pipe Gallery.

The primary coolant system design pressure rating is 1825 psig except where the use of 900 psi ASA fitting reduces the pressure ratings. At temperatures above 525 F, the 900 psi fittings are rated for service below 1825 psig, e. g., at 600 F the maximum allowable pressure is 1600 psig. Actual system operating pressures are shown in Table C-12. This table shows that operating pressures will be considerably below allowable limits.

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TABLE C-12

EXPECTED PRIMARY COOLANT SYSTEM OPERATING PRESSURES

	<u>Design Pressure*</u>	<u>Startup (50% Flow)</u>	<u>Normal Operation</u>
Pump Discharge	1825 Psig	1455 Psig	1655 Psig
Reactor Inlet Riser	1825	1440	1590
Reactor Outlet Riser	1825	1400	1440
Pressurizer	1650	1350	1350
Steam Generator Inlet	1825	1377	1370
Pump Suction	1825	1364	1315

*As limited by 900 pound fittings assuming an inlet temperature of 369 degrees F and an outlet temperature of 527 degrees F.

During expected operational reactor transients the pressure control mode will not be changed; however, for rapid or prolonged transients, normal pressure control may be lost temporarily. For the pressure transient encountered during loop startup, a separate, special bank of manually controlled heaters are provided to increase pressurization rates.

Primary system pressure control is shown in Figure C-60. Some of the features included to increase system reliability are duplicate pressure transmitters; summing point monitors on all critical amplifiers; automatic transfer to manual control following component failure; use of servo followers and transfer units to permit a smooth transition from manual to automatic to manual control; and duplicate electric-pneumatic converters.

A series of special controls and devices are provided to prevent system overpressurization. Assuming that the system pressure is initially at the low end of the control range and slowly increases in spite of all previous corrective actions, the following would occur:

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1. The normal cycling spray system would turn on.
2. An alarm would sound.
3. Backup spray valve would open.
4. All heaters would de-energize.
5. A reactor setback trip would occur.
6. A reactor scram trip would occur.
7. Pressure relief valve would open.
8. Dump valves would open if high pressure were accompanied by high or low flow rates, high temperature, or slow pump speed (see Section 6.4 in the first volume).

4.4.1.3 Primary Coolant System Flow Control

The primary coolant system flow control scheme is shown in Figure C-61. Flow control will be accomplished through adjustments to the speed of the primary pump drive turbines. Discharge flow rates from each of the five primary pumps will be monitored and a constant pressure differential across each flow monitor will be maintained by adjustments to each individual turbine speed control. A total flow controller will be utilized to control the sum of the five cells flow, i.e., total flow, by re-adjusting the five individual turbine speed controllers. Primary coolant flow control is a vitally important system and steps were taken to enhance system reliability and minimize the importance of component failure. Liberal use was made of summing point monitors, servo followers, and selector stations to accomplish these objectives.

During the reactor startup transient, steam will be supplied by the oil-fired boiler until process generated steam is available. The boiler capacity will be sufficient to provide reactor flows of about half

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the normal operating flows.

Compensating actions must be taken after a reactor scram to prevent excessive thermal and pressure shock to the primary coolant system piping and equipment. One of these actions is to reduce primary coolant system flow after a scram to reduce the rate of change of the reactor effluent temperature. This flow coastdown will be automatically triggered after each scram by resetting the control point on each loop's flow controller. Coastdown rates will be controlled by the system inertia.

4.4.1.4 Primary Coolant Inventory Control

The control of the primary coolant inventory is shown in Figures C-62 and C-63. The coolant liquid level in the pressurizer must be maintained essentially constant under various phases of reactor operation to assure surge capacity and pressure control capability. Liquid level changes in the pressurizer will operate the primary loop injection flow control valves, indirectly controlling the liquid level in the pressurizer. Surges in level above the normal control ranges will result in operation of the spill system and will initiate a reactor power setback. Drops in the pressurizer level below the normal control range may lead to serious consequences if not arrested. Efforts have been made to avoid excessively low levels or to minimize their consequences. If the level drops below the normal control point an alarm sounds. If, in spite of corrective actions taken after the alarm, the level continues to drop, a power setback trip will occur. The purpose of the setback is to reduce power before the pressurizer is depleted and loss of pressure control is threatened. In case of a still further drop in level, a reactor scram trip will be incurred automatically. While the scram will cause a net reduction of coolant temperature, an

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effective density increase and a further reduction in pressurizer level, it is desirable to take the scram while the liquid level is still adequate to handle the ensuing transient.

4.4.2 Secondary Coolant System Operation

During reactor operation, the secondary system is operated as a steam-condensate system to transfer heat removed from the primary loop to the circulating raw water system. Heat transfer from the primary loop results in boiling on the secondary side of the ten main heat exchangers. This steam then flows to the main 46" steam header atop the 109 Building. Sufficient steam to drive the six drive turbines and to supply auxiliary areas uses flows to the turbine supply header; the remainder is condensed in the dump condensers, completing the heat transfer to the circulating raw water system. Secondary coolant pressures, flow rates and inventory are directly controlled; coolant temperatures are dependent variables.

4.4.2.1 Secondary Coolant System Pressure Control

Secondary coolant system pressure control methods are shown in Figures C-64 and C-67. Steam pressures and reactor inlet temperature controls are closely interrelated as discussed in Section 4.4.1.1 of this appendix. The mode of steam pressure control is based upon controlling the main steam header pressure. Deviations from the control point are used to adjust the quantity of steam being condensed in the dump condensers, either by throttling the dump condenser inlet butterfly valves, or by adjusting the liquid level in the dump condensers through surge tank pressure changes. When primary loop inlet temperatures are being controlled, error signals are used to adjust the set point on the main steam header pressure controller. Reactor outlet temperature rate-change sig-

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nals will be used to anticipate rapid transients of secondary loop pressure by reset action in the main steam header pressure controller.

Condensate system pressure will be adjusted to follow transients in the steam pressure to reduce the condensate return pump head requirements. To accomplish this, the secondary coolant system surge tank will be pressurized from the main steam header through adjustable differential-pressure control valves. Pressure in the surge tank is transmitted to the dump condensers as a factor in maintaining the desired liquid level and to the dump condenser condensate pump suctions to maintain net positive suction head through the dump condenser condensate return header. Two modes of operating control are under consideration for the surge tank pressure control: (1) a constant pressure differential may be maintained between the main steam header and the surge tank, and (2) a controlled variable differential may be maintained to control the flooding level in the dump condensers. Choice of the mode will depend upon studies now underway.

Listed below are design pressure for the various portions of the steam condensate loop:

TABLE C-13

SECONDARY LOOP PRESSURE

<u>Location</u>	<u>Pressure, Psig</u>
Steam Generator	72
46" Main Steam Header	65
Turbine Header	61
Turbine Throttle	55
Surge Tank	50
Condensate Suction Header	60
Area Auxiliary Steam	55

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4.4.2.2 Secondary Coolant Inventory Control

The control of the secondary coolant inventory is shown in Figure C-65. The inventory will be indirectly controlled by maintenance of level in the secondary loop surge vessel. Surge tank level increases occurring as a result either of system load increases or pressure changes result in opening of system spill valves, permitting excess fluid to drain to the afterheat removal water storage tank. An eight-inch overflow line connecting to a 36 inch line is provided in case the afterheat removal storage tank should overflow. Surge tank level decreases which might occur as a result of system load decreases, leakage, or pressure fluctuations result in opening of fill valves from the deaerator through the boiler feed water pumps. Constant losses of secondary loop fluid to the turbine and area auxiliary steam systems will call for a continuous makeup through this system. A reactor scram, or extremely low water level in the surge vessel will result in starting of the afterheat removal pumps and opening of the emergency fill valves to maintain the desired liquid level in the steam generators and dump condensers. Low liquid level in the steam generator is undesirable but does not represent a nuclear safety condition. It is feared that low level leading to the bareing of tubes could promote stress corrosion cracking and impaired operating efficiency. Low liquid level in the dump condensers would result in excessive condensation rates, reduced secondary system pressures and, indirectly, more rapid than desired reductions in primary coolant temperatures. This represents again an undesirable (because of thermal stresses) condition but not an unsafe condition in the nuclear sense.

4.4.2.3 Secondary Coolant System Flow Control

Control of the secondary system flow is shown in Figure C-66. The system flow is controlled to maintain the liquid level

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in the main steam generators. As the coolant on the secondary side is boiled, liquid-level controls open feed water control valves from the condensate pump discharge header to maintain this level. Since liquid level will be extremely difficult to determine due to the large steam fraction, the apparent level will be compensated by steam flow and pressure signals and manually biased as necessary for stable level control.

4.4.3 Circulating Raw Water System Operation

Reactor generated heat is transferred to the circulating raw water system in the dump condensers for ultimate dissipation in the Columbia River. About 400,000 gpm of untreated Columbia River water will be pumped from the river, through the tube side of the dump condensers and turbine condensers, and back to the river. Flow control is exercised primarily through the number of circulating raw water pumps online; however, trimming adjustments are possible at each of the condensers. Since the drives for these pumps do not incorporate speed control, fluctuations in river level will also effect the flow rates in this loop.

Since discharge of 180 F water from the dump condensers could have a detrimental effect on marine organisms in the Columbia River, it is necessary to discharge the return raw water into the center of the river to mix it as rapidly as possible. The effluent from the service condensers will be at a temperature of about 90 F and will be segregated and discharged at the river shoreline.

4.4.4 Backup Systems

Descriptions of normal and backup cooling operation are included in Section 6 of the first volume of this review. In particular, Section 6.6 covers "Assurance of Continuity of Cooling". These descriptions

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and discussions will not be repeated here. Rather, this section covers the operating control method of the backup systems during reactor operation and Section 4.5.4 covers control during reactor shutdown.

4.4.4.1 Standby Boiler

The oil-fired boiler will be used to produce steam when process steam generated from reactor heat is not available and to back-up process steam in the event of scrams. Even after a scram some process heat will still be available to generate steam to drive the pumps and the turbine generator. The length of time that sufficient steam will be available will depend on the power level, the fuel exposure, and mode of operation of the heat dissipation system at the time of the scram. When the reactor is operating at the design equilibrium power level and with the fuel at average fuel exposure, the reactor afterheat will produce adequate steam for essential services for at least eight minutes after a scram. During such periods the boiler will be maintained in a warm banked condition producing steam in the range of 2500 to 25,000 pounds of steam per hour, as necessary. During such periods the boilers will be capable of load pickup to full load in about eight minutes.

Under conditions where reactor afterheat would be insufficient to provide adequate backup to the reactor plant, the boiler must be operated within its normal control range, available to provide steam to the drive turbines and deaerator if required. Regardless of the standby status of the boiler, a reactor scram shutdown will automatically cause the boiler to be brought into normal control range as rapidly as possible. An equivalent amount of steam generated by the boiler will be indirectly dissipated to the dump condensers when the reactor is operating at normal levels.

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4.4.4.2 Electric Pony Motors

The pony motors are direct-shaft linked to the primary drive turbines for reduced circulation of the primary loop coolant should steam be lost to the drive turbines. The BPA power system is used to power the electric drive units. These units are normally de-energized during steam turbine operation but will be automatically energized if pump drive speeds falls below 900 RPM.

4.4.4.3 Emergency Raw Water System

Should it be evident at any time that the normal recirculation systems are inadequate to remove the reactor heat, provisions have been made to dump the primary coolant and convert to once-through, raw water, reactor cooling. Diesel driven pumps are provided to sequentially pump river water to an intermediate storage tank, and from this tank to the reactor. Automatic starting of these units is initiated whenever the reactor safety circuits are broken, unless bypassed. A manual start is also provided. Entry of the raw water into the primary loop will normally be prevented by closed valves at the reactor building, however, if a dump system trip has also been triggered, raw water will enter the loop when pressure decays below the shutoff head of the emergency raw water pumps (approximately 370 psig). Dump system actuation requires either a manual trip or any two coincident trips of the following:

1. Extremely high reactor outlet temperature
2. High or low primary loop pressure
3. Low primary loop flow rates
4. Low tachometer indications on three of the drive turbine units

The trip actuates the effluent dump valves and the inlet raw water valves, but raw water entry will not occur until pri-

mary loop depressurization permit check valves to open in the front riser system.

4.5 Shutdown Plant Operation

4.5.1 Primary Coolant System Operation During Reactor Outages

The primary coolant system will normally remain on recirculation during a reactor shutdown; however, provision has been made for conversion to once-through cooling in order that maintenance of the primary loop can be performed. Normal charge-discharge and routine maintenance do not require single pass cooling and, under controlled conditions, the loop can be depressurized and opened. Normal modes of flow and pressure control will be used during these conditions except that control equipment of appropriate range must be substituted. This equipment has as yet not been designed, therefore, the following discussion is based upon currently proposed control techniques.

4.5.1.1 Fuel Charge-Discharge

Fuel charging will involve establishment of a low pressure recirculation at a low system temperature with as large a coolant makeup as is required to permit single pass operation of those tubes set up for charge-discharge. The coolant flow will vary between 15 and 20 gpm per tube and pressure conditions required are outlined below:

TABLE C-14

CHARGE-DISCHARGE-SYSTEM PRESSURE

<u>Location</u>	<u>Pressure - Psig</u>
1. Pump Discharge	57.0
2. Inlet Manifold	56.0
3. Top-of-Front Riser	23.0
4. Top-of-Rear Risers	7.5
5. Pressurizer	-0-
6. Outlet Manifold	31.0
7. Pump Suction	27.0

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Additional pressure control to individual process tubes will be introduced through throttling of the inlet connector valves using butterfly valves which cannot reduce the flow below about 15 gpm/tube when the tube is uncapped. Fine pressure control will be obtained by liquid level adjustments in the pressurizer vessel.

Losses of primary coolant are reduced by valving all tubes which are to be uncapped to the drained diversion riser system on the rear whence the coolant will flow through a cooler to the crib. Flow out the rear of the uncapped tubes will be restricted to 15 - 20 gpm under these conditions. Flow rates below approximately 15 gpm can cause the tube to operate only partially full of water with subsequent loss of radiation shielding. Available loop makeup capacity restricts the number of tubes which can be uncapped at one time to approximately 50. Dissipation of the afterheat removed from the fuel elements not involved in the discharge operation will occur in the steam generators which will be operating as water-to-water exchangers. At least two cells will be in service during the reactor shutdown, but up to four will be required during the early outage period to cool the primary coolant enough for contact work on the rear face. Conditions expected three hours after a nuclear shutdown are a 90 degree F maximum reactor inlet temperature, and a 120 degree F maximum reactor outlet temperature.

4.5.1.2 Maintenance Operation

When maintenance of the heat dissipation systems is needed it may be necessary to operate the primary loop on a once-through basis unless the affected part can be isolated. Up to 10,000 gpm of filtered water can be introduced either at the inlet manifold or through the emergency raw water system. These flows, adequate for fuel element cooling provided no

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nozzle closures are removed, will permit contact maintenance on normally unavailable equipment.

4.5.2 Secondary Coolant System Operation During Reactor Outages

The secondary coolant system will continue to operate as during reactor operation, until the reactor is shut down and low inlet water temperatures are desired. At this time, the steam generators and dump condensers can be filled with stored secondary loop water from the afterheat removal storage tank; and flow lines opened to bypass the steam system. Under these conditions, heat removed by water-to-water heat transfer in the steam generators is transferred to the tertiary loop in the dump condensers, also by water-to-water exchange. The condensate pumps downstream of the dump condensers provide normal flow through the condensate system up to approximately 30,000 gpm. Pressure is controlled by adjustments to the steam admission system to the surge tank. The main steam system under these conditions is controlled at or near normal pressure by steam from the stand-by boiler system.

4.5.3 Circulating Raw Water System Operation During Reactor Outages

Operation of the circulating raw water system will be required during reactor outages to provide coolant to the pump turbine drive condensers and process dump condensers and makeup to the filter plant. It can, however, be operated at reduced flows by reducing the number of operating pumps. Normally two will be in service, one driven from each power source, to satisfy backup requirements; however, one pump can provide sufficient flow to shut down the plant and maintain it shutdown.

4.5.4 Backup System

Backup requirements during reactor outages will not normally be as rigid as during reactor operation; however, whenever possible two

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levels of coolant backup will be available with at least one set for automatic actuation.

The standby boiler will be required during a reactor outage at steam generation rates up to its design capacity (575,000 pounds per hour). This steam will be required to drive the five pump drive turbines, the turbine generator, for deaeration of secondary loop makeup, for maintenance of main steam header pressure and for auxiliary system uses.

Primary pump drive pony motors will normally not be used during an outage; however, some units may be placed on electric drives to conserve steam. Automatic actuation will normally be effective if the loop is on recirculation since drive unit speeds will be in excess of 900 rpm.

Shortly after a nuclear shutdown residual heat generation rates will be low and the emergency raw water system may be bypassed and removed from service to minimize the chance of raw water introduction into the primary system. Manual actuation will provide backup if normal cooling is lost and fuel overheating is approached.

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FIGURE C-56

HIGH LIFT PUMP PERFORMANCE

Basis: Pump Data Given in P. O. 6463

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NOTE: ALL EQUIPMENT NOT LOCATED BY IS MOUNTED IN CONTROL CABINET BBB

3M REACTOR FLOW
VIBES FAILURE ALARM
ANEL "B" THRU "V" (SEE TABLE 1)

FIGURE C-61
PRIMARY COOLANT FLOW
CONTROL

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FIGURE C-42
PRIMARY COOLANT INVENTORY
CONTROL PART - A

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FIGURE C-13
PRIMARY COOLANT INVENTORY
CONTROL PART - B

•

FIGURE C-44
SECONDARY COOLANT PRESSURE
CONTROL

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CONDENSATE DISCHARGE
HEADER PRESSURE
TRANSMITTERS
TYPE KP220A

MAIN STEAM
HEADER PRESSURE
FROM 10.8.D
ES035559 (N-1-40003)

10.8.A
4 REQUIRED
STEAM TO
TURBINE
HEADER

CONDENSATE
RETURN FLOW
TEMPERATURE

FROM TRANSMITTER
MEASURING RETURN
FLOW FROM STANDBY
POWER PLANT BOILER
FEED PUMP DISCHARGE

WMSA
RECORDER

TAYLOR
RECORDER

SUMMING
UNIT

TRANSFER
CIRCUIT

SERVO
FOLLOWER
(10.8.N)

1/2" CONDENSATE
RECIRCULATION VALVE

6" x 7" x 6" SECONDARY COOLANT MAKE-UP FLOW
CONTROL VALVE

SEE S/R DWG. N-1-23849 SHY. 21 BOX 1 / N-1-23934 SHY. 10 BOX 4

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A

B

C

D

E

F

G

H

FOR

FIGURE C-47
SECONDARY SYSTEM PRESSURE
CONTROL

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5. WATER TREATMENT

Water treatment may be considered comprised of two elements: (1) removal from the water those impurities which are deleterious, and (2) addition to the water of materials which are desirable. A "progressive" water treatment scheme is used at the NPR where the degree of treatment is tailored to the intended use. In this manner water with a satisfactory quality is available without uneconomical over-treatment.

A schematic diagram of the treatment process is shown in Figure C-68.

5.2 Raw Water

5.2.1 Treatment

River water flowing to the suction of the river water pumps flows first through a bar screen and then through a traveling screen. The bar screen serves to exclude the larger pieces of debris and protects the pumps. The traveling wire screen which has openings of about 1/8 inch prevents fingerlings from entering the pumps and excludes debris that passes through the bar screen.

Provisions have been made for chlorine addition to the raw water automatically on a continuous basis, on a programmed intermittent basis (a few times per day) or manually when required. The ultimate mode of chlorination will depend on need as shown by experience. Chlorination would serve two purposes. First it would provide its usual role in controlling algae growth. Also, the addition of chlorine reduces the coolant pH which may aid in the dissolving of any calcium scales and thus preventing fouling of condensers. This second role will probably not be required.

5.2.2 Uses of Raw Water

Raw water without further treatment is used for a variety of cooling services, other miscellaneous services, and is used as the feed

FIG.-C-68 - WATER TREATMENT SCHEMATIC

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to the filter plant. The major normal uses other than feed to the filter plant are outlined below:

- a. Dump condenser cooling.
- b. Pump turbine drive condenser cooling.
- c. Turbine generator condenser cooling.
- d. Rupture monitor heat exchanger cooling.
- e. Graphite moderator heat exchanger cooling.
- f. Spill cooler cooling.
- g. Sample heat exchanger cooling.
- h. Cooling around penetration of primary coolant system lines through concrete walls*

5.3 Filtered Water

5.3.1 Treatment

A part of the raw river water will be pumped to a chemical mix tank, aluminum sulfate coagulant and chlorine added and the mixture sent to an up flow coagulator. In these units, the small coagulated particles are gently agitated to permit growth into larger flocs (flocculation). After flocculation, the water flows up from the bottom of the coagulator to overflow near the top. The units are arranged so that the velocity continuously decreases as the water flows from bottom to top. If the turbidity level and alum dosage are high enough, a sludge blanket will form with the larger particles finally settling to the bottom of the coagulator and clarified water overflowing to the filter. Ordinarily, however, Columbia River quality is such that it probably will not be necessary or economical to maintain a sludge blanket and the unit will serve as a more conventional settling basin.

*Sodium dichromate is added to this stream of raw water to inhibit corrosion of the aluminum cooling tubes.

A polyacrylamide filter aid will be added to the clarified water as it enters the filters to improve the quality of the treated water. The NPR filters are quite similar to the standard Hanford filters used to produce cooling water for the single-pass reactors. The filter media consists of six inches of sand topped by 24 inches of anthracite. The sand and anthracite are sized to provide a high quality of filtered water with reasonably long filter runs between filter backwash cycles.

The filtering process is expected to result in water containing no more than 0.01 ppm turbidity. This quality is suitable for demineralizer feed and for many plant services. The filter plant design rating is 9,000 gal/min; however, it will probably produce up to about 20,000 gal/min.

5.3.2 Uses of Filtered Water

The filtered water is used as the feed to the demineralization plant. Other major filtered water uses include:

1. Control and safety rod cooling. (Sodium dichromate will be added to the rod cooling water to inhibit corrosion of the aluminum components of the control rods.
2. Reactor shield cooling.*
3. Single-pass reactor cooling during reactor shutdowns.*
4. Decontamination makeup solutions and initial rinses after decontamination.*
5. Filter backwash.
6. Potable water (extra chlorine is added to this service).
7. Reactor gas loop condenser cooling.
8. Irradiated fuel storage basin supply.

*Sodium sulfite is added to the filtered water used in these services to scavenge dissolved oxygen and limit carbon steel corrosion.

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5.4 Demineralized Water

5.4.1 Treatment

A portion of the filtered water output will be used to feed the demineralizer plant. The plant consists of primary and secondary cation and anion units and a deaerator. The filtered water first passes through the primary cation unit where most of the cations are replaced with hydrogen ion. The (acidic) effluent from the cation unit is heated and routed to the vacuum deaerator. Here carbon dioxide and dissolved air are removed. The water then flows through the primary anion bed where most of the remaining anions are removed. The effluent from the primary anion unit is then routed through the secondary cation and anion units for a final polishing.

The plant is designed to produce water of the following quality:

TABLE C-15

DEMINERALIZED WATER QUALITY

Total Dissolved Solids	0.2 ppm maximum
Chlorides	0.02 ppm maximum
Silica	0.02 ppm maximum
Dissolved Oxygen	0.005 cc/l maximum

The cation exchangers will be regenerated using sulfuric acid and the anion beds with sodium hydroxide. Hydrochloric acid will specifically not be used to minimize the chloride ion content in the finished water.

5.4.2 Uses of Demineralized Water

The major normal uses of demineralized water will be:

- a. Primary coolant system. The primary coolant system will be supplied with the demineralized water. Lithium hydroxide will be

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added to control the pH at about ten. Hydrogen and hydrazine will be used during reactor operation and shutdown periods respectively to minimize dissolved oxygen content.

b. Secondary coolant system. The secondary system will be supplied with demineralized water. Ammonia will probably be added to control the pH at about 9, and hydrazine added to scavenge dissolved oxygen.

c. Graphite moderator cooling system. The graphite moderator cooling system will utilize demineralized water with the same additives as the primary system coolant.

6. DECONTAMINATION

6.1 General

During the scope design period, two factors emerged which gave special emphasis to the development of decontamination process for the NPR. First, the fuel was to be fabricated employing metallic uranium cores. The rapid reaction between high temperature water and metallic uranium was well known. It appeared necessary that provisions be made to cope with quite severe system contamination resulting from the spread of debris from ruptured fuel elements. Secondly, a decision was made to use carbon steel piping. While the use of carbon steel itself was not expected to cause the contamination problem to be extraordinarily severe, the use of carbon steel precluded the use of strong oxidizing acids as decontamination agents for rupture clean up and in general appeared to limit the range of suitable reagents.

A number of different decontamination processes have been developed which appear suitable for decontaminating all or parts of the primary coolant system.

6.2 Decontamination Processes

6.2.1 Fission Product Decontamination

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When a metallic uranium fuel element ruptures and debris is transported around the coolant system, it is found that the major amount of activity is associated with small particles of uranium dioxide. Some of these particles may be flushed out of the system. More complete decontamination depends upon the dissolution of these small UO_2 particles.

A number of different uranium dioxide dissolution reagents have been studied at HAPO for NPR application. The best characterized process is termed the "peroxide - bicarbonate" process and is described below:

TABLE C-16

PEROXIDE BICARBONATE PROCESS

<u>Reagent</u>	<u>Remarks</u>
Hydrogen Peroxide	Oxidizes the UO_2
Sodium Carbonate) Sodium Bicarbonate)	The carbonate and bicarbonate complexes the uranium and buffers the pH at about 9.5
8-Hydroxyquinoline	Stabilizes the H_2O_2
EDTA*	Complexes fission product
Sodium Laurylsulfate	Wetting agent

Highly successful results have been obtained using this process. However, other reagents appear to have potential for even better effectiveness particularly if metallic uranium is intermixed with the oxide. These may be adopted after further study. These agents all contain hydrogen peroxide, oxalic acid, sodium oxalate, and 8-hydroxyquinoline an organic acid buffering agent.

6.2.2 Removal of the Oxide Film

Other than fuel rupture debris, the remaining source of contamination will be "crud" irradiated in the reactor core, released, and in-

*Ethylenediaminetetraacetic Acid

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corporated into the oxide film layer formed from the oxidation of the underlying metallic surface. This activity is removed by the removal of the oxide film layer exposing a bare metal surface.

Oxide films formed on carbon steel surfaces can be removed with inhibited acids. Most experience at HAPO has been obtained with proprietary, inhibited, phosphoric acid formulations. These effectively remove the oxide layer and with it most of the included activity. However, the discharge of phosphorus into the river upstream of existing production reactors cannot be permitted. The uptake of phosphorus by these reactors and the conversion of P^{31} to P^{32} would aggravate an existing river problem. At this time it is not certain if waste disposal provisions will be made which permit the use of phosphoric acid. If such provisions are not made, other inhibited acids may perform the same role as phosphoric acid in a satisfactory fashion. Inhibited acids which show some promise include ammonium citrate, acid bisulfates, and sulfamic acid.

Oxide films formed on stainless steel surfaces cannot be removed by any known single step process without excessive corrosion. However, if the stainless is first exposed to a solution of sodium hydroxide (about 1.5 lbs/gal) and potassium permanganate (about 0.25 lbs/gal) the films can be removed by treatment with an inhibited acid with little corrosion. The most thoroughly studied process is based on use of an inhibited bisulfate. The process for removing film from stainless steel surfaces is also effective with carbon steel.

6.3 Expected Decontamination Procedures

6.3.1 Single Tube

In the event of a fuel failure, the affected tube will be put "on diversion", i. e., the effluent coolant will be routed through a special discharge line, cooled, and sent to a crib. This action should

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aid in avoiding gross system contamination. It will not prevent gross contamination of the process tube, the outlet connector and, possibly the diversion system. It may therefore be desirable to decontaminate the process tube connector and diversion system after each failure to reduce rear face radiation levels. The decontamination would also prevent spread of rupture debris throughout the balance of the primary system when reactor operation is resumed, thus minimizing system activities and the background to the rupture monitor.

This decontamination procedure will probably be carried out by injecting the UO₂ dissolving solution into the inlet nozzle and removing it through the diversion system. If this does not result in a satisfactory decontamination, it can be followed by a film removal process.

6.3.2 Reactor Front and Rear Face Piping

Because of the massive areas of the inlet and outlet piping, it is not improbable that reactor front and rear face activity levels will build up to nuisance values. Refueling will require front and rear face contact operations. Also, front and rear face components: butterfly valves, nozzle caps, RTD's, and diversion valves may require maintenance. Therefore, front and rear face decontamination may be required at times when activity levels at other parts of the system are not troublesome. Provisions have been made for such occasions. In this event, a concentrated acid can be injected into two inlet supply headers at a time and the acid removed via the rear flush line. During such operations, a carbon steel-film removal process will probably be used.

6.3.3 Steam Generator Cell Decontamination

Access into the steam generator cells will be required during outages. Maintenance of the circulating pumps and steam generators

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may be required occasionally. Therefore, it may be desirable to decontaminate the part of the primary system in a steam generator cell. This can be done by draining the piping and equipment in the cells, filling with a decontamination solution and recirculating the solution through a special connecting line between the lines entering and leaving the steam generator cell. The regular primary pumps will be used for recirculating the decontamination solution.

The process used for steam generator cells will probably be a stainless steel film removal process possibly preceded by a uranium dioxide dissolution process.

6.3.4 Entire Primary Coolant System

On rare occasions, for extensive plant modifications or after some incident involving gross primary coolant system contamination, it may be desirable to decontaminate the entire primary coolant system. This can be accomplished by routing a stream of primary coolant through a chemical mix tank, adding reagents to the tank and returning the solution to the system and continuing this until the proper concentration is achieved. The solution will be drained from the system by cooling the reactor on a single pass basis flushing out the reagents and by gravity drainage of the remaining piping.

Such a full plant decontamination will probably employ a stainless steel film removal process possibly preceded by a uranium dioxide dissolution step.

6.3.5 Moderator Coolant System Decontamination

The external portion of the graphite moderator coolant system may be contaminated from materials (e.g., corrosion products) being transported into the reactor core, being irradiated and finally being dis-

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charged from the core with subsequent redeposition on piping walls.

This activity can be removed by the use of a carbon steel film removal operation. The low temperature operation of the system (about 200 F maximum) should permit a rapid, efficient decontamination; experience has shown that film formed at low temperature are relatively easier to remove than high temperature films.

6.4 Special Decontamination Equipment

Equipment required for accomplishing the various decontamination processes and procedures has been included in the reactor design. Storage of dry and liquid chemicals has been provided. Design was on the basis of storing dry potassium permanganate, sodium bisulfate and sodium carbonate - bicarbonate. However, other dry chemicals can be substituted. Liquid storage provisions were made specifically for phosphoric acid, sodium hydroxide solution and hydrogen peroxide. Again, these facilities can also be used for other chemicals.

A chemical mix tank is provided where the dry chemicals can be dissolved and the liquid chemicals diluted if desired. A heat exchanger is provided for heating chemicals to the desired temperature. Finally, pumps and lines have been provided for transferring chemicals to the system or part of system being decontaminated.

7. WASTE DISPOSAL

7.1 Solid Wastes

If long-lived activity is present, solid wastes will be stored in pits or trenches with suitable backfilling. If exclusively short-lived radioisotopes are involved, the waste may be retained in a controlled area until the residual activity decays to the point where the material can be treated as non-radioactive.

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7.2 Liquid Wastes

A simplified flow diagram showing the management of liquid waste streams is shown in Figure 7.6 of the first volume of this review. A short discussion of liquid waste management is presented in Section 7.2 of the first volume. The following gives an estimate of the daily burden of the wastes disposed of by various means:

TABLE C-17

NPR LIQUID WASTE STREAMS

<u>WASTE STREAM</u>	<u>ACTIVITY BURDEN</u>	<u>DISPOSAL POINTS</u>
Shield Cooling	6.7 Curies/Day ⁽¹⁾	River Mid-Channel
Fuel-Spacer Decontamination	.3 Curies/Day ⁽²⁾	River Mid-Channel
Primary Coolant Bleed	123 Curies/Day	NPR Crib
Graphite Moderator Coolant Bleed	15.3 Curies/Day	NPR Crib
Diversion System	90 Curies/Fuel Failure ⁽³⁾	NPR Crib
Single Pass Reactor Cooling During Outage	100 Curies ⁽⁴⁾	NPR Crib
Control Rod Cooling Water	68 Curies/Day ⁽⁵⁾	River Mid-Channel or Crib
Decontamination Wastes	2000 Curies/Single Pass Decontamination ⁽⁶⁾	Storage Tanks
Primary Coolant System Dump	2000 Curies ⁽⁷⁾	Quench Tank and Crib ⁽⁸⁾

- (1) Based on experience at the K Reactors; corrected for lower thermal flux in cooling tubes at NPR.
- (2) Intermittent load.
- (3) This assumes release of 10 gm of fuel. If, as an extreme, an entire element reacts, the burden would approach 170,000 curies.
- (4) This activity results from the displacement of the regular primary coolant.
- (5) Based on experience at the K Reactors.
- (6) Based on estimate for front and rear face decontamination to remove "crud" activity. The load following decontamination of one steam generator cell would be about 3 - 4 curies. If decontamination is required because of spread of rupture debris, the load will depend on the nature of incident.
- (7) The burden is equal to that removal expected from reactor front and rear face piping and hence, is a limiting value. The estimate assumes no fission products are released as a result of incident leading to "dump".
- (8) The quench tank will receive the first water released from the system, and, probably, most of the radioactive burden. Eventual disposition of the quench tank water will depend on the particular situation.

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Estimated isotopic breakdowns for the more significant waste streams are shown in the tables below:

TABLE C-18

RADIOACTIVE LIQUID WASTE CONTRIBUTION FROM THE
THERMAL SHIELD COOLING SYSTEM

<u>Radionuclide</u>	<u>Release Rate at 1800 GPM</u> <u>(Curies/Day)</u>
Na ²⁴	0.655
p ³²	0.0131
Sc ⁴⁶	0.0006
Cr ⁵¹	0.164
Mn ⁵⁶	5.24
Cu ⁶⁴	0.458
Zn ⁶⁵	0.0164
As ⁷⁶	0.0983
Np ²³⁹	<u>0.164</u>
Total	6.72

TABLE C-19

RADIOACTIVE LIQUID WASTE CONTRIBUTIONS FROM THE PURITY
CONTROL BLEED SYSTEM OF THE PRIMARY COOLANT LOOP

<u>Radionuclide</u>	<u>Release Rate at 350 GPM</u> <u>(Curies/Day)</u>
Na ²⁴	33.0
p ³²	1.12
Cr ⁵¹	0.023
Mn ⁵⁶	84.6
Co ⁵⁸	0.002
Fe ⁵⁹	0.008
Co ⁶⁰	0.019
Cu ⁶⁴	3.87
Ni ⁶⁵	0.579
Zr ⁹⁷	<u>0.002</u>

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TABLE C-20

RADIOACTIVE LIQUID WASTE CONTRIBUTIONS FROM THE BLEED
STREAM OF THE MODERATOR COOLANT

<u>Radionuclide</u>	<u>Release Rate at 26 GPM</u> <u>(Curies/Day)</u>
Na ²⁴	3.36
P ³²	0.06
Cr ⁵¹	0.02
Mn ⁵⁶	11.3
Co ⁵⁸	0.0002
Fe ⁵⁹	0.0008
Co ⁶⁰	0.0015
Cu ⁶⁴	0.39
Ni ⁶⁵	<u>0.145</u>
Total	15.3

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TABLE C-21

FISSION PRODUCTS RELEASED TO EFFLUENT STREAM BASED ON GRAMS OF
URANIUM FROM RUPTURED FUEL ELEMENT (ZERO DECAY TIME)

	10 g (Curies)	1.914×10^4 (Complete Fuel Element) (Curies)
Sr ⁸⁹	3.17	6,070
Sr ⁹⁰	0.04	70
Y ⁹⁰	0.04	70
Y ⁹³	5.76	11,000
Zr ⁹⁵	3.94	7,540
Ru ¹⁰⁶	0.08	158
I ¹³¹	3.17	6,070
Te ¹³¹	3.17	6,070
Cs ¹³⁷	0.03	57
Ba ¹⁴⁰	5.76	11,000
Ce ¹⁴¹	5.18	9,910
Ce ¹⁴³	5.18	9,910
Ce ¹⁴⁴	1.06	2,030
Sm ¹⁵³	0.144	276

Np ²³⁹	<u>53.0</u>	<u>101,400</u>
Total	90	170,000

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TABLE C-22

RADIOACTIVE LIQUID WASTE CONTRIBUTIONS FROM THE
CONTROL ROD COOLANT SYSTEM

<u>Radionuclide</u>	<u>Release Rate at 1200 GPM</u> <u>(Curies/Day)</u>
Na ²⁴	6.55
P ³²	0.131
Sc ⁴⁶	0.006
Cr ⁵¹	1.64
Mn ⁵⁶	52.3
Cu ⁶⁴	4.59
Zn ⁶⁵	0.164
As ⁷⁶	0.982
Np ²³⁹	<u>1.64</u>
Total	68.0

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TABLE C-23

ONE STEP SINGLE PASS IN-PILE DECONTAMINATION

<u>Radionuclide</u>	<u>Activity Removal Per Tube (Curies)</u>
P ³²	0.001
Cr ⁵¹	0.762
Fe ⁵⁹	0.003
Co ⁶⁰	0.148
Cu ⁶⁴	0.788
Zr ⁹⁵ -Nb ⁹⁵	0.016
Ru ¹⁰⁶⁻¹⁰³	0.004
Ce ¹⁴⁴⁻¹⁴¹	0.197
Np ²³⁹	<u>0.00002</u>
Total	2.0

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TABLE C-24

STEAM GENERATOR CELL DECONTAMINATION

<u>Radionuclide</u>	<u>(Curies)</u>
Na ²²	0.216
Cr ⁵¹	8.19
Co ⁵⁸	0.149
Fe ⁵⁹	2.24
Co ⁶⁰	0.112
Cu ⁶⁴	14.9
Se ⁷⁵	0.372
As ⁷⁶	7.45
Zr ⁹⁵ -Nb ⁹⁵	0.670
Ag ^{110m}	0.238
Sb ¹²⁴	3.358
Np ²³⁹	<u>0.447</u>
Total	35.3

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APPENDIX D

REACTOR CONTROL AND SAFETY SYSTEMS

1. GENERAL

1.1 Background

The reactor nuclear control and safety systems were briefly covered in Sections 3 and 5 of the first volume of this review. Further, the "physics aspects" of these systems, i. e., control philosophy, system nuclear strengths and total control and speed of control considerations are covered in Sections 1.5, 1.6, and 1.7 of Appendix C preceding. Also, Section 4 of Appendix C covers control of the heat dissipation system.

This appendix presents a physical description of the nuclear control and safety system, details of the control circuits, a description of the actuating sensors and a general discussion of steps taken to make the systems highly reliable and adequate.

1.2 Safety and Control Systems - A Brief Summary

1.2.1 Horizontal Rods

The 87 horizontal rods are individually driven by separate mechanical systems which are actuated by hydraulic fluid pressurized by pumps and accumulators. The horizontal rods have both control and safety functions. Separate hydraulic power supplies are provided for each function. Rods selected for manual control (or automatic power "setback") are operated at either slow (6 feet per minute) speed or intermediate (30 feet per minute) speed by a low-pressure (1500 psig) hydraulic system. The rate of reactivity increase by control rod withdrawal is limited by hydraulic pump capacity. Certain neutron absorbing rods must be reserved for the scram (safety) function. Upon receipt of a scram signal, the safety rods, plus all operational control rods, are driven full-in at high speed (about

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29 feet per second) by high pressure (2700 psig) oil stored in gas-pressurized accumulators (one for each rod). Up to three rods may be removed from service. However, if four or more rods are deactivated by switching to OFF or WITHDRAWAL, these rods are automatically restored to the safety (scram) function. Four separate hydraulic systems are provided: a high pressure and a low pressure system for the rods entering from the left side of the reactor; and a high pressure and a low pressure system for the right side. The speed and capacity of any 84 of the 87 horizontal rods are adequate to provide protection for all credible accidents as discussed in Appendix C.

Multiple sensors, redundant fail-safe circuits and electrical components and multiple, carefully designed mechanical components make the Horizontal Rod Control and Safety System high reliable. The sensors and the logic used for power setback and safety (scram) trips are outlined in Table 5.9 of the first volume.

1.2.2 Ball Safety System

An independent poison-ball safety system is provided to assure that the reactor is driven and maintained subcritical in the event of failure of part or all of the Horizontal Rod System due, for example, to an earthquake, or in the event of actuation of the emergency raw-water cooling system which would result from primary loop depressurization. Upon receipt of a trip signal, the balls are released to fall by gravity into vertical channels in the active zone of the reactor. Excess hopper capacity is provided to assure that the balls fill the channels to a level above the top of the active zone.

The Ball Safety System will not be used during normal operation, but will be reserved for emergency situations. The balls will be

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dropped by:

1. Seismoscope signals (MM 5 or greater) on 2-out-of-3 logic.
2. Actuation of the emergency once-through cooling system.
3. Failure of the neutron flux to decay to 25 per cent of design level within ten seconds after a rod scram signal.
4. Failure of the neutron flux to decay to 0.1 MW within 150 seconds after a rod scram signal. This protection continues in effect until the primary safety circuit is made up again.
5. Manual trip.

The signal which trips the balls will also trip the rod scram circuit. The first of the stream of falling balls will reach the bottom of the active zone in about two seconds after the trip signal activates the hopper release mechanism and will provide a significant fraction of the full-channel control capacity. Allowing for the normal distribution of control rods in the reactor, the added poison provided by the Ball Safety System (less any four hoppers) will hold the reactor subcritical for either the "optimum" flooding or cold-water-injection accidents, or the combination of both.

If a limited number of rods (greater than three) fail to enter the reactor on a rod scram signal, the rows of ball hoppers on each side of each failed rod will drop balls into the reactor, assuring adequate poison to scram the reactor. The remaining hoppers retain the capacity to

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drop balls on demand. On all other trip signals all ball hoppers trip simultaneously.

A high degree of reliability is provided for the Ball Safety System using essentially the same methods as for the Horizontal Rod System.

1.3 Potential Mishaps and Consequences

Without exception, the potential mishaps which can occur to affect the functioning of the safety systems follow the general rule that the higher the probability is that a mishap will occur, the lower the severity of the consequences in terms of its effect on reactor safety. A general classification of incidents according to probability is given in Section 4.1 of the first volume. At one end of the spectrum are expected incidents and probable accidents which are expected to occur, such as the infrequent failure of a rod to scram due to binding in some part of the mechanism, or failure of a mechanical or hydraulic component. Since it is unlikely that failures of this type would affect more than one rod or one ball channel at a time, the effect of such a failure on reactor safety is negligible. Into this category will fall the majority of system failures, including most personnel errors.

In a much lower probability category are errors or malfunctions considered improbable accidents or credible accidents such as, for example, failure on the part of the operator to unbypass several major instrumentation systems prior to startup, or an unsuspected system defect which, undetected, would disable a major portion of one of the safety systems. Such a mishap, while it should not cause reactor damage nor violate the reactor safety philosophy, would probably result in actuation of the backup

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[REDACTED]
safety system.

In an extremely low (vanishingly small) probability category (for the present purposes, incredible) are occurrences such as complete failure of the Primary (1X) Safety Circuit, simultaneously with an abnormal operating condition such as continuous fast rod withdrawal (implying failure of another independent circuit) or "optimum" stack flooding (in itself, "incredible"). In these circumstances, mild to moderate reactor component damage would be expected, such as over-stressing of fuel elements and primary loop piping with some small probability that a limited degree of uranium melting would occur. The excursion would be terminated by a scram of the Ball (3X) Safety System, and the emergency once-through coolant system would be activated.

In the event of failure of both the Primary (1X) Safety System and the Ball (3X) Safety System simultaneously with uninterrupted fast rod withdrawal, graphite flooding and emergency cold water injection, there is no doubt that catastrophic fuel melting would occur.

1.4 Compliance With Requirements

In design concept the NPR nuclear control and safety systems comply with the system requirements. Completion of the detailed design, fabrication, installation and use appear to offer no unexpected problems.

It is recognized that rigorous and comprehensive acceptance tests will be required to ensure initial compliance. Assurance of continued compliance with system requirements will demand continued rigorous and systematic inspection, preventive maintenance and testing.

2. REACTOR CONTROL AND SAFETY PHILOSOPHY

The philosophy controlling the design of the safety and control systems discussed in detail in Appendix C includes the following elements:
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- a. An operational control system meeting rigorous requirements on strength and speed must be provided.
- b. Two safety systems required. Each required to meet strength and speed of control requirements.
- c. Backup philosophy must be satisfied; i. e., the removal of part of one system on the basis of existence of other system is not allowed.

3. DESCRIPTION OF SYSTEMS

3.1 General

The safety systems and circuits are sequentially described by beginning with mechanical features, drive or trip mechanism features and electrical circuits. Electrical equipment terminology is presented in Section 6 of this Appendix.

3.2 Horizontal Rod System

3.2.1 General

The system is composed of the 87 rods with their associated drive, contacts, position indicator, and auxiliaries. Rods enter the reactor in a pattern of 11 rods wide and nine rods high with three rods omitted at each corner. The general arrangement is shown in Figure B-14 of Appendix B.

3.2.2 The Control Rods

Figure D-1 is a drawing of a horizontal control and safety rod. The nuclear poison, boron carbide grit and powder is packed into the annular cavities of an aluminum extrusion. The boron carbide is held in place by a magnesia cement followed by fiberglass also packed into the extrusion. The boron carbide section is about 29'7" long, the magnesia cement section 9'-8" long and the fiberglass section is about 2-1/4" long. A vented aluminum cap is welded over the fiberglass to form the end clos-

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ure. This vent, which will release helium generated in the reaction:

terminates at the outboard end of the control rod inside the steel sleeve in the inner rod room.

The aluminum extrusion is contained in a titanium sheath which will ride on the reactor graphite channel blocks. The sheath is 40'7" long, 2-3/4" in diameter and has an 0.065" wall thickness. The cooled tip of the sheath extends about five inches beyond the boron carbide packing.

Special tip design features permit movement of the rod through a distorted reactor core corresponding to a center stack contraction of eight inches.

3.2.3 Rod Drive System

3.2.3.1 General

The rod drive system consists of three parts:

(1) the basic mechanical-hydraulic drive system which imparts motion to the rods, (2) a normal hydraulic control system to drive the rods during normal operation, and (3) a scram hydraulic drive system. The over-all drive system is shown in Figure D-2.

3.2.3.2 The Piston-Cylinder-Chain Drive

The mechanical drive system is schematically shown in Figure D-3 and pictorially in Figure D-4. Each rod is driven by an endless roller chain affixed to the outer sections of the rods in the outer rod room. The chains pass around sprocket wheels and are attached to the drive cylinder. The arrangement is such that the motion of the cylinder is doubled in transmission to the rod.

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The drive cylinder fits over a stationary piston. Hollow piston rods affixed to both ends of the piston extend out the ends of the cylinders through seals. Hydraulic fluid enters into and is drained from the cylinder on both sides of the piston through holes penetrating into the hollow piston rods. The holes are located near the piston. A special choke or buffer tube on the outboard end of the cylinder fits closely over the piston rod for a distance of about 15 inches. By progressively covering oil outlet holes as the rod is driven full-in during a scram, inward motion is effectively decelerated to avoid mechanical damage to the rod. Each end of the cylinder is vented for removing any entrapped gas.

3.2.3.3 The Normal Drive System

The normal rod drive system is shown in Figure D-3, D-5 and D-6. The system is duplicated. Independent systems are provided for the rods entering each side of the reactor. The systems function by pumping hydraulic fluid from a reservoir to one side of the piston. The hydraulic fluid is drained from the cylinder on the other side of the piston through one or two parallel, constant flow orifices, and routed back to the reservoir.

Two pumps in parallel are used in each system. These pumps have different capacities; the smaller pump has a capacity of 3.25 gpm, operates continuously and permits operation of a small number of rods at the "slow" speed. The larger pump has a capacity of 5.25 gpm, permits withdrawal of larger numbers of rods at a given time and operation of rods at "high" speed. The slow rod speed is about 6 feet per minute; the more rapid speed is about 30 feet per minute. Electrical switching permits either, but not both, pumps to be operated at a given time.

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When rods are stationary, hydraulic fluid is needed only for make-up of leakage and a bypass system is utilized recirculating the oil through coolers. A four gallon nitrogen-charged accumulator is provided on the discharge line to accommodate fluid leakage, say past the bypass valves into the reservoir, for a reasonable length of time to prevent short period cycling of the bypass system.

Rod movement speed is ordinarily controlled by the two constant flow orifices on each rod system. The smaller orifice receives flow during all normal rod movement and permits slow speed rod movement. When fast speed rod movement is required, the larger parallel orifice is also opened. Flow regulation rates are variable. Rod speeds of about four to eight feet per minute and 25 to 50 feet per minute with the flow flow and with both orifices are possible; however, settings will be for six and 30 feet per second speeds.

Other features in the rod drive system are the "forward - reverse" valve which determines the direction of rod drive; check valves which provide an hydraulic lock and prevent depressurization of the cylinder during rod immobility; pressure equalizing valves to equalize pressure in both ends of the cylinder after a rod movement to permit rapid reversal of rod direction if desired. Further details on these features is supplied in the detailed analysis and description of the system. (1)

3.2.3.4 Scram Drive System

Two separate hydraulic systems are provided for the scram function; one for the rods entering each side of the reactor. The systems are shown on Figure D-3, D-5 and D-6. The scram system is

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designed to drive the rods 75 per cent of the way into the reactor in 1.1 seconds.

Two parallel pumps are provided for each scram system. The smaller pump has a capacity of 3.25 gal/min and is operated continuously to make up leakage and maintain pressure in the individual rod scram accumulators. The larger pumps have a capacity of 7 gal/min and is used with the smaller pump to recharge the accumulator rapidly after a scram.

The pumps on one side of the reactor are usually supplied with BPA power and the pumps on the other side with local power. However, in case of need both sets can be operated from either power source by manual switching. When a scram signal occurs valves on the exit of the scram accumulator open and the hydraulic fluid flows from the accumulator through a check valve into the cylinder. At the same time, a valve on the line at the end of the other piston rod opens giving a low-resistance flow path for the oil and bypassing the normal constant-flow orifices. The scram accumulator capacity is sufficient to drive a fully withdrawn control rod to the full in position.

3.2.4 Control Rod Position Indicator System

Switches have been provided to signal when the rods are full-in, full-out, 75 per cent in and if the rods are inserted 75 per cent of the way in within 1.5 seconds following a scram. Continuous rod position information is also generated. This information will be displayed on a cathode-ray tube in front of the reactor operator and is also supplied to a digital readout system. By dialing a rod number, the reactor operator obtains the length of that rod inserted in the reactor.

3.2.5 Electrical Control System

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3.2.5.1 General

Each horizontal rod circuit is equipped with a five-position "Service Selector Switch" by means of which the primary function of each rod is established. These functions are:

- a. Manual
- b. Power setback
- c. Safety
- d. Withdrawal
- e. Off

The switches are key-locked and located in a locked cabinet. The function of the rods in each circuit is described briefly in Table 5.8 of Volume 1. The Service Selector Switch action for one rod is shown in Figure D-7.

3.2.5.2 Manual

It will be a practical necessity to operate several control rods in the NPR simultaneously because of the large number of rods provided. However, it was not possible to group together the rods which would be simultaneously moved at the time of design. Therefore, a rod gang-ing system was provided. The rods in each gang will be operated together under the control of one actuator switch. Six actuator switches have been provided. By pushbutton, a rod may be assigned to any of the six gangs or to none.

Rods will be selected so that movement of a "gang" will not excessively perturb the flux and hence the balanced power distribution.

It will be possible to assign all rods to the same gang although there is no operational reason for doing this. This would

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permit withdrawal of all rods by the actuation of one switch. However, the rate of reactivity insertion (poison withdrawal) will be limited by the capacity of the hydraulic pumps as discussed in Section 1.7 of Appendix C.

The manual operating circuit is shown in Figure D-8.

3.2.5.3 Setback

The setback function is described in Section 5 of the first volume. Rods are selected for use in the setback function in two ways: (1) Every rod that is assigned to any of the six gangs is automatically assigned to the setback function unless the setback function is bypassed; (2) The rod may be assigned to set back by the selector switch. The setback circuit is shown in Figure D-9.

When a power setback signal occurs, timed relays intermittently apply power to the "In-Slow" solenoids of the rods assigned to setback. An interlock has been provided to prevent withdrawal of any rods during the time a setback signal is in effect.

3.2.5.4 Safety Control

The Primary Safety Circuit is shown in Figure D-10. Any rod may be assigned to safety control by the selector switch. Once this function is selected, the rod is removed from the manual control and setback function with two exceptions:

1. The manual "Out" solenoid may be energized until the rod is "full-out", at which time its manual functions are disconnected.
2. An all-rod insertion circuit is provided which will insert all rods simultaneously after a planned shutdown.

The number and pattern of rods designated "safety rods" will be controlled by mandatory Process Standards. All horizontal

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rods which are withdrawn act as safety rods when the reactor is scrammed. Only enough rods are designated specifically as safety rods to ensure that the reactor will be shutdown on demand.

3.2.5.5 Withdrawal

At times it will be highly desirable to withdraw rods when the Primary Safety Circuit is tripped. For example, if cooling water to a rod is lost, and if the rod cannot be conveniently withdrawn from the operating reactor, it may be necessary to scram the reactor and withdraw the rod to prevent aluminum melting. (The titanium sheath would minimize the severity of the problem since it would contain the molten aluminum).

Since rods on "Withdrawal" cannot be scrammed, the number assigned to this function and to the "Off" function (see Section 3.2.5.6 below) is limited to three. If more than three rods were assigned to these services, they will all be scrammed into the reactor if the Primary Safety Circuit is tripped. The Withdrawal circuit is shown in Figure D-11.

3.2.5.6 Off

Circumstances may arise when a rod must be totally disabled for repair services. In such a case the rod may be assigned to the "Off" function. Rods on "Off" cannot be controlled manually and are not affected by scram on setback trips. However, the counting function described above is still in effect to cause a scram of all rods should three or more be assigned to "Withdrawal" or "Off". The "Off" circuit is shown in Figure D-12.

3.2.5.7 Interlocks

Interlocks are provided to prevent withdrawal of control rods before rods identified as safety rods are first withdrawn

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and to prevent withdrawal of rods at the fast speed unless the reactor is sub-critical. Both of these interlocks are backed up by the automatic safety provisions and their failure would not be considered an unsafe condition.

A series of interlocks is also placed in the reset circuit to prevent make-up of the Primary (1X) Safety Circuit unless a number of critical valves, in the loop are properly positioned. Other interlocks in the 1X reset circuit are:

1. Scram hydraulic accumulator pressurized.
2. High flux trip on intermediate level nuclear instrumentation not bypassed.
3. Power rate-of-rise trip not bypassed.
4. Ball channel exit valve closed.
5. Less than five ball hoppers locked on.
6. Graphite cooling water trips not bypassed.
7. Emergency cooling water diesel system not bypassed.

3.2.6 Sensors

Some of the major sensing units which initiate Primary Safety Circuit (1X) trips are described below:

3.2.6.1 Process Channel Flow Monitors

An engineering diagram of the flow monitor system is shown in Figure D-13. Each inlet connector is fitted with a venturi meter serving as a flow rate sensor. The differential pressure signal is converted by a transducer into an electrical signal which is monitored electronically with high and low trip settings in a controller. The high and low trip settings of each flow monitor are individually adjustable, and in addition, all high and all low trips may be adjusted "in masse". The output of each controller is fed to a master logic circuit which incorporates a short (about 0.4 second) time delay to override electrical noise impulses which might cause spurious trips. The output of the master logic circuit consists of trip contacts in all Primary Safety Circuits.

3.2.6.2 Coolant Outlet Temperature Monitor

An engineering diagram of the temperature monitoring system is shown in Figure D-14. The zone temperature monitor consists of 109 primary sensors monitoring process tube outlet temperature. The output of the detectors is electronically monitored and a trip signal is generated when a preset high-temperature limit is exceeded. The number of trip signals is counted; two cause a power setback, three or more, a scram.

3.2.6.3 High Level Neutron Monitor

An engineering diagram of the high level neutron monitor system is shown in Figure D-15. The primary sensors are 24 ion chambers. The sensors are sensitive to power levels from 25 MW to 5000 MW. The monitors are arranged in pairs symmetrically located above and below the reactor core in the biological shield. One set of 12 is used mainly for reactor control; the other set performs the primary high-level monitoring function of interest here. The signals are utilized in the following manner:

a. The signals from each of 12 chambers from one set are monitored. High level on 2 of 12 will cause a setback; very high level on 2 of 12 will cause a scram.

b. The signals from the same set of 12 chambers noted above are divided into two sets of six. The signals from each set of six and from the other set of 12 are average and used as a signal for the power rate sensing circuit. High rate of increase will cause a setback; very high rates will cause a scram.

c. The signal from the set of 12 chambers primarily used for high level safety monitoring are arranged by averaging circuits into three signals each of which is the average of all twelve signals. A high level trip from two of the three signals will cause a reactor scram. A duplicate set of signals will cause a safety ball drop if the reactor power has not decayed a set amount in a short amount of time after a reactor scram. Finally, the average signal is also included in the setback circuit.

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3.2.6.4 Intermediate and Low-Level Range Flux Monitor

An engineering diagram of the intermediate range and low level instrumentation system is shown in Figure D-16. The primary sensors for the intermediate range instrumentation are three gamma-compensated ion chambers located in the top biological shield. The signal from each chamber is amplified and differentiated with respect to time. Two sets of rate signals are generated. One set will cause a setback (2 of 3 logic); the other set will cause a scram (also 2 of 3 logic).

The signal from each amplifier is also compared to a high-level trip. A reactor scram is caused by a trip from 2 of 3 of these signals unless the trips are bypassed (as will be required for increasing the reactor power from intermediate to equilibrium levels). A second signal will be used to trip the ball safety system if the reactor should again become supercritical after shutdown and prior to planned startup. A third signal prevents withdrawal of the rods at the "fast" speed when the power level exceeds some limit (adjustable between 0.001 MW and 4MW).

The sensors for the low subcritical level consists of two fission chambers located in the top shield near the reactor center-line. The signal in counts per second is supplied to an amplifier and a log count rate meter. The signal from the log rate meter is compared to two trip values. One causes a power setback; the other actuates the fast-rod-withdrawal interlock. A signal from the primary count rate meter is also included in the fast-rod-withdrawal interlock.

3.3 Ball (3X) Safety System

3.3.1 System Description

The NPR Ball 3X Safety System is comprised of 108 vertical channels passing through the active zone of the reactor with associated

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poison balls and ball storage, release, recovery and indicating systems. The ceramic balls are made of mainly from alumina (at least 2.7 gram/cc) with at least 0.38 gram/cc of samarium oxide an epithermal neutron absorber. The general arrangement of the ball system is shown in Figures D-17 and D-18.

Each channel is serviced by an individual ball hopper located in the top primary shield. These hoppers contain 2.25 cubic feet of poison balls, enough to fill the active-zone section of the channel with four to fourteen per cent excess, depending upon the length of exit pipe associated with a particular channel. Special shielding plugs filled with high density concrete are used to prevent significant activation or burnout of the balls while in storage and to provide shielding against radiation from activated balls in the channels. Each hopper is equipped with a special spring-operated-ball-release gate, which is held in place by a d. c.-solenoid-actuated release device. The solenoid has two holding coils which are powered by separate but identical circuits. When both solenoid holding coils are de-energized by a Ball 3X safety circuit trip, the release gate is lifted by action of the three compressed springs, permitting the balls to fall by gravity into the channel. The time required for balls to drop to the bottom of the active zone is about two seconds following de-energization of the solenoid. The individual ball hoppers are equipped with signal systems to transmit continuously information relative to ball content, position of the gate, and position of the cocking mechanism.

When released, the balls are retained in their channels by closed, air operated diaphragm valves of special design located at the end of each channel exit pipe in the Ball Handling Room below the reactor. These valves are provided with signal devices to transmit valve position

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information continuously. When the exit valves are opened, the balls flow by gravity through the collection system and enter a manual/automatic washer-dryer system, and may then be returned to the ball channels. An interlock prevents making up the reactor safety circuit while the exit valves remain open.

Control panels for operating the ball handling system are located in the Ball Control Room adjacent to the Ball Handling Room beneath the reactor. An important feature of the ball removal system is the necessity for energizing the control panel by means of a master key-lock switch before any ball removal or handling operations can be initiated. When this key-lock switch is operated, a light on the panel in the Reactor Control Room is turned on.

3.3.2 Electrical Control System

A simplified circuit diagram of the Ball (3X) Safety Circuit is presented in Figure D-19.

The Ball (3X) Safety System is identical in concept to the Primary (1X) Safety System, with one exception: each hopper latching mechanism contains two independently-powered sets of pull-in and holding coils in a latching solenoid. Both holding coils must be de-energized by a Ball (3X) Safety System trip to drop the balls. Thus, a failure of one signal coil or power circuit will not result in an accidental ball drop.

Parameters which are monitored for Ball (3X) circuit trip-actuation are: earthquake (magnitude 5 or greater on the modified Mercalli scale); emergency cooling water flow to reactor; failure of neutron flux to decay following a 1X safety circuit trip, and "reactor-supercritical-after-a-safety rod-scrum".

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The Ball (3X) Safety System is provided with lockout features to permit maintenance on individual channels during reactor outages without the risk of inadvertent ball drops. Sufficient control to assure sub-criticality must be inserted in the reactor from other sources (Horizontal Rod System, xenon, or supplementary poisons) when ball hoppers are locked out. Interlocks are provided to prevent makeup of the 1X safety circuit, and hence reactor startup, if more than four ball hoppers are locked out. These interlocks appear in Figure D-10.

3.3.3 Sensors

3.3.3.1 Neutron Flux

Only four automatic trips are included in the Ball (3X) circuit. The sensors for two of these, the flux monitors, were included in the Primary (1X) Safety Circuit sensors description. In addition to these, Ball (3X) insertion is called for by seismoscope trips or by emergency cooling system activation.

3.3.3.2 Seismoscopes

The seismoscope system consists of four pendulum-and-ring-type seismoscopes. One pendulum will be set to make contact and sound an annunciator in the control room when an earthquake of intensity two or greater on the modified Mercalli scale occurs. Each of the other three pendulums will be set to make contact when an earthquake of intensity five or greater occurs. Two of the three contacts must close to initiate a Ball (3X) scram. A single pendulum trip, or a component failure in the circuit, will activate an annunciator in the control room.

3.3.3.3 Emergency Coolant Injection

When either of the two circuits which actuate the emergency reactor coolant supply system is energized the Ball (3X) safety system will be tripped.

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4. RELIABILITY

4.1 Electrical

In normal safety circuit usage, an operational (non-fail-safe) failure of a relay will be the failure of a relay or its contacts to open when de-energized, since safety circuits are normally operated in the energized state. Unfortunately, relay failure data of this type are very sketchy, due to the rarity of this type of failure. In the analysis of the Ball (3X) Safety Circuit reliability, the probability of a relay failure of this type was considered to be zero.

With the assumption that the probability of a relay failing to open when de-energized is zero, the Primary (1X) Safety Circuit should be essentially completely fail-safe, since all trips are initiated by de-energizing relays and opening their contacts. In the Ball (3X) System, on the other hand, two of the four auxiliary trip circuits are activated by the closing of relay contacts, thus leading to some probability of a non-fail-safe failure and a reliability study of the system has been made.⁽²⁾ These two circuits are:

4.1.1 Emergency Cooling Injection

The emergency cooling system electrical circuit is operated normally de-energized in order to reduce the probability that a spurious failure will actuate the system. To minimize the probability of an operational failure, two identical circuits are provided, one of which is redundant. Energizing either circuit opens contacts on the Ball (3X) Circuit. A probability does exist that both of the relays with contacts in the Ball (3X) Circuit will fail to energize on demand. The resulting probability of an operational failure in the circuit has

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been calculated to be 0.006 per year for a mean time between failures of this sort of about 190 years.⁽²⁾

4.1.2 Reactor Supercritical After Shutdown

This circuit contains two relay contacts which must close when the Primary (1X) Safety Circuit trips to energize two timers, which then time out and de-energize relays with contacts in the Ball (3X) Circuit. A probability exists that both contacts will not close; a probability exists that both timers will not energize if the contacts do close. The resulting probability of an operational failure in this circuit has been calculated to be 0.00590 per year, for a mean time between failures of about 190 years.⁽²⁾

The assumption of a zero probability of failure of a relay to open when de-energized is not theoretically correct; this probability does exist, even though it is very small. While no calculations have been performed to estimate this probability, calculations have been completed for other, somewhat similar, situations. For example, it has been calculated that the probability of an operational failure of two-out-of-two Beckman flux monitors in the older Hanford reactors is about 1×10^{-9} per year; the probability of an operational failure of one Beckman flux monitor is about 3×10^{-5} per year.⁽³⁾ In the relatively simple case of the two redundant breakers in the Primary Safety Circuit, or of the similar breakers in the Ball (3X) Safety Circuit, the operational failure probability is expected to be even lower, or less than 1×10^{-9} per year. Similarly, other circuits (two-out-of-three logic circuits, etc.) will have a non-zero operational failure probability, but it will be very small.

4.2 Mechanical

The foregoing preliminary analyses indicate that the operational failure probability of the safety systems will be largely governed by the

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mechanical features of the system, not by the electrical circuitry. However, the failure probability of most mechanical components, while low, is largely unknown, due to the magnitude of the testing program which would be required in order to establish realistic failure rates.

The unknown failure probability of mechanical components, and thus their effect on the over-all integrity of the safety systems, is largely mitigated by the fact that each mechanical component is associated with only one horizontal rod or only one ball hopper. This fact, coupled with a routine testing and inspection program reduces the probability of an unexpected operational failure of a significant number of rods, or ball hoppers, to an extremely low level, which is accepted. The working pressures of the hydraulic system are below the burst or proof pressure by an adequate, code-determined margin and pressure relief is provided. Likewise, mechanical stresses in other critical parts are held to a fraction of the ultimate strength as dictated by good design practice. This design, plus periodic inspection, will serve to maintain a high degree of over-all mechanical system reliability.

5. POTENTIAL MISHAPS AND CONSEQUENCES

5.1 Horizontal Rods

5.1.1 Mechanical

5.1.1.1 Normal Control System Failure

With the rod system as presently design, only three modes have been identified by which rods can be driven out of the reactor at higher than normal control speeds. The conditions required for such maloperation are (1) reactor-end cylinder vent open, (2) gas in normal-control hydraulic system, or (3) a force within the reactor tending to push the rod out.

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Case (1) represents an effective bypass of the normal effluent oil metering orifice. In such an event, the rods could move out at a rate of about 50 feet per minute (as limited by flow resistance in the inlet hydraulic lines) and would continue until the operator released the rod actuator switch or until the oil capacity in the accumulator was exhausted. The rate of positive reactivity addition would depend on the number of rods affected, and such physics effects are discussed in Section 1.7 of Appendix C.

For Case (2) to be effective, gas from some source must first enter the reactor side of the drive cylinder(s) during a previous "In-Drive" operation. The only reasonable source of gas would be the nitrogen contained in the low-pressure system accumulator(s) (about 1.3 gallons per side at 1500 psig). When the rod(s) is subsequently switched to OUT-drive the gas would be compressed and forced out through the effluent metering orifices. Gas, instead of hydraulic fluid, passing through these orifices would permit the rod to move at a higher-than-normal speed, which would, however, again be limited to about 50 feet per minute by the flow resistance of the normal inlet piping, fittings, and valves through which hydraulic fluid, not gas, is passing. This higher-than-normal speed could continue only until the oil capacity of the system accumulators is exhausted. Thus, the rod withdrawal rates in this case are about the same as those described in Case (1) above. Reactivity effects are discussed in Section 1.7 of Appendix C.

Case (3) involves a force within the reactor tending to expel the rods. As mentioned in Section 3.2.3.2 of this Appendix, the rod drive cylinders are hydraulically locked when the rods are immobile. In order for Case (3) to be effective, the hydraulic locking system

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must fail or the force must be exerted during the 0.08 second elapsed time after rod movement ceases and the hydraulic lock goes into effect.

In the absence of a hydraulic lock (from any cause) when the rods are not in motion, a sufficient pressure buildup within the graphite stack in the vicinity of the tip of any rod(s) could force the rod(s) out. By measurement, approximately 100 pounds force on the rod tip is required to overcome drive-system resistance, giving a required stack pressure of about 17 psi. The limiting case would be the concurrent failure of more than ten process tubes pressurizing the entire stock to 20 psig or more.⁽³⁾ In this eventuality, all rods not hydraulically locked could be driven all the way out of the active zone - but not at very high speed. However, the probability is vanishingly small that the assumed process tube ruptures would not cause a trip of the process tube flow monitoring system and a scram of the Horizontal (1X) Safety System. The minimum inward force exerted by the scram hydraulic system is roughly a factor of five greater than the outward force generated by stack pressurization to 20 psig. Thus, even rods not hydraulically locked and being driven out of the reactor would be scrambled in. (It is possible that the scram speed might be reduced enough to actuate the ball 3X backup system).

In the event of a dangerously rapid rod withdrawal from any cause, nuclear instrumentation trips (pile period, rate-of-rise, and high-level flux monitors) would almost immediately scram the Horizontal Rod (1X) Safety System.

5.1.1.2 Scram System Failure

With the scram drive system as presently designed, five modes have been identified by which one or more rods could fail to

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scram properly. The off-standard conditions required involve (1) mal-operation of one or more of three automatic valve units; scram pilot valve, scram valve, or diversion valve; (2) incorrect positioning of three manual valves; rod lockout valve, maintenance valve downstream of diversion valve, or scram system vent valve; (3) little or no oil in scram accumulator; (4) little or no gas in scram accumulator; and (5) accumulator piston jammed. These failure could result in improper scram operation of the affected rod or rods. The rod or rods would either not scram or would scram at a reduced speed.

In general, a failure of the type listed above would affect only one horizontal rod. Gross human error would be required before a large number of rods could be affected. This will be guarded against by procedures including valve position indicator, and locks.

5.1.2 Electrical

5.1.2.1 Rod Control Circuits

The rod circuits can fail in only two ways: they can fail open, and then cause the failure of one or more rods to move when required, or they might short across, causing one or more rods to move in or out at slow or fast speed. The two limiting cases are the failure of all rods to respond to manual control, and the accidental withdrawal of all rods at the fastest effective speed.

If all rods fail to respond to manual control during equilibrium operation, the power level will drift either higher or lower depending on the reactivity status and various transients. This will continue until the reactor is manually scrammed (almost certain under the circumstances) or an automatic scram occurs. In either case there is

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no discernable nuclear hazard.

The most serious consequences of accidental fast rod withdrawal at the fastest possible rate might or might not result in minor fuel element over-stressing, but could not cause fuel melting or reactor damage if the rods are scrammed manually, or by automatic flux, flow or temperature trips.

5.1.2.2 Primary (IX) Safety Circuits

The following types of failure can be postulated:

a. Failure of Individual Trip Systems

If an open circuit failure occurs, the reactor will scram (relays in the primary safety circuit are operated energized) and no hazard will exist. If a system fails to trip on demand, the reactor will continue to operate until another parameter measuring trip occurs or until the off-standard conditions were noted and the reactor manually scrammed. Most abnormal process conditions will be detected by several monitors and the automatic reactor scram would be expected. Even in the case of the individual channel flow failure where only one automatic trip device is monitoring the channel, the probability of non-safe failure is so small that the addition of another trip system specifically for backup has not been justified. Manual trip action would be relied on; the abnormality would be indicated on both the flow and temperature data presentation systems and the 105-N Control Room annunciator.

b. Failure of Safety Circuit Breakers

To minimize the probability of this type failure, two independent breaker circuits have been provided to give redundancy. The probability of simultaneous failure of both breakers

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is vanishingly small. However, should such occur a manual trip of the Ball (3X) Safety System would provide backup. Moreover, manual rod insertion at "fast speed" may also be possible.

c. Failure of Individual Rod Scram Solenoid

This type failure is improbable. The chance of simultaneous failure of significant numbers of rods is highly improbable. However, should four or more rods fail to scram, the Ball (3X) Safety System would be automatically actuated.

Rather than consider the many possible combinations of safety-circuit failures and reactor operating conditions, a limiting case is considered. This accident would consist of withdrawal of the horizontal rods at the maximum possible rate during startup, in conjunction with complete failure of the Primary (1X) Safety System. Reactor kinetics studies following such an accident are discussed in Section 1 of Appendix C.

5.2 Ball (3X) Safety System

5.2.1 General

Conceivable ball safety system failures may be classified into three groups: (1) failure to fill hoppers, (2) failure of channels to fill when trip occurs, and (3) inadvertent draining of channel.

5.2.2 Failure to Fill Hoppers

The ball hoppers are fitted with probes to indicate whether or not the hoppers are filled with balls. Empty hoppers will be indicated in the 105-N Control Room. Failure of probe devices in the direction of indicating the hoppers are full when actually empty is improbable and in any case should affect only one or a few isolated hoppers.

Even if the probe device did fail, the failure would

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probably be obvious to the reactor operation who will likely know the status of the hoppers.

5.2.3 Failure of Channels to Fill

The most likely causes of failure of a channel to fill would be those causes which would affect only individual channels. Examples of this would be failure of the mechanism on a hopper, blockage of a channel, an individual hopper being locked out or an individual drain valve being opened.

Events affecting the entire system are high improbable. Circuitry has been designed to be highly reliable. Also, testing and inspection programs will be carried out to prevent the loss of a major part of the system by attrition resulting from an accumulation of single channel failure.

However, failure of the system in conjunction with failure of the horizontal rod system at a time of the worst conceivable accident (graphite flooding and emergency coolant injection) would certainly result in severe fuel melting.

5.2.3 Inadvertent Draining of Channels

An interlock is provided to prevent the opening of Ball System drain valves unless the horizontal rods are in. However, it is conceivable that the interlock would fail. If all exit valve were opened, no balls would be removed from the active core because of the capacity of the drain piping is less than the excess of balls in the top reflector. The greatest rate of increase or reactivity would result from opening two exit valves in each quadrant of the reactor simultaneously. The eight channels would be completely drained. The maximum positive reactivity

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ramp is given in the first volume of this report. (Table 5.7) as 0.003 per cent $\Delta k/k$ per second for a total reactivity insertion of 0.6 per cent.

6. TERMINOLOGY

The electrical equipment and equipment contact numbering system is designed to aid in the quick recognition and location of the many components used in the electrical safety circuits.

The electrical circuitry has been divided into functional sections for numbering purposes. Each section has been assigned an "Index" number. For example, the Primary (1X) Safety Circuit is Index 1, the power setback circuit Index 4, the Emergency Cooling System electrical circuit Index 33, etc.

When relay coils appear on an Index, they are numbered consecutively (with some exceptions: for example, when one relay is associated with the circuit for only one horizontal rod or only one Ball (3X) hopper it is numbered, when possible, with the rod or hopper number).

Relay contacts are numbered. All contacts which are closed when the relay is energized possess odd numbers (1, 3, 5, etc). All contacts which are open when a relay is energized possess even numbers. An exception is normally made when only one contact of a particular relay is used; in this case, the contact is identified by the relay number only. However, in this document, such contacts are numbered 1 to clearly distinguish them from the relay coil.

Different types of electrical equipment are assigned letters for identification; all relays, breakers, and timers are lettered K; push-buttons are lettered P, or N if they latch in; bypass switches are

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lettered H; instruments are lettered T, etc.

When a relay is identified, the first number is the number of the relay on the Index, the letter K identifies it as a relay, and the number following the letter is the Index number on which the coil appears. For example, relay 1K 4 is the first relay coil on Index 4.

Relay contacts many times do not appear on the same Index as the relay coil; tables (not included here) have been prepared which show the Index number where each contact of each relay appears. The relay contact number follows the Index number of the relay, separated by a dash. For example, 1K4-2 is a normally open contact of relay 1K4.

7. BIBLIOGRAPHY

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(Confidential)
2. C. L. Goss, "NPR Ball 3X Safety Circuitry Reliability", HW-68309, January 30, 1961.
3. W. J. Love, "Reactor Steam Release Within The Core", HW-66840, September 23, 1960.

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ASSEMBLY & TEST PROCEDURE

1. WELD SLOTTED RING (3) TO EXTRUSION (11).
2. AIR PRESSURE TEST WELDS AT 25 PSI. SOAP TEST NEEDS FOR LEAKS. NO LEAKS ALLOWABLE. SOAP USED SHALL NOT BE HARMFUL TO ALUM.
3. PACK EXTRUSION (11) WITH B4 C-VIBRATE TO MIN. DENSITY OF 1.0 GR/CC ACCORDING TO SPEC. HWS 6595. PACK 85% MAGNESIA CEMENT TO DENSITY .18 GR/CC. NO VOIDS ALLOWABLE. FILL IN WITH FIBER GLASS. (PACK TIGHT)
4. INSERT RING (4) & WELD TO EXTRUSION (11).
5. AIR PRESSURE TEST WELDS AT 25 PSI AS IN ITEM 2.
6. WELD SLEEVE (2) TO RING (4) OF ASSEMBLY, ALIGNING .033 DIA DRILLED HOLES, 1.001.

7. REDELL .033 HOLE THROUGH SLEEVE (2) & RING (4) TO OPEN HOLE THROUGH WELD.
8. AIR PRESSURE TEST ASSEMBLY FOR LEAKS WHERE HOLE IS DRILLED THROUGH WELD AS PER TEST PROCEDURE 2
9. HYDROSTATIC PRESSURE TEST COMPLETE ASSEMBLY (1) AT 250 PSI FOR ONE HOUR NO LEAKAGE ALLOWABLE.
10. INTERIOR OF ROD MUST BE KEPT CLEAN & DRY. IN EVENT INTERIOR GETS WET WITH WATER, IT MUST BE DRIED THOROUGHLY IMMEDIATELY. SUGGEST REMOVAL OF WATER WITH ALCOHOL & FORCED AIR DRYING.

GENERAL

1. WELDING PER FABRICATOR
2. UNLESS OTHERWISE SPECIFIED, ALL DIMENSIONS ARE TO BE FRACTIONAL PARTS OF AN INCH.
3. PARTS (4) EACH OTHER
4. APPLY WELD THREADS.
5. KEYS, PAR. GRIND IF NEEDED
6. CHANGE T

MBLY

* DIMETERS CONCENTRIC
WITHIN .001 P.I.R.

! NOTES:

03E MERT GAS SHIELDED ARC
WELDING HW-4900-S & HW-4903-S.

OTHERWISE SPECIFIED

INCHES
ON
DECIMALS ANGLES

INNESS IN ACCORDANCE
ASA 846.1-1955

5) ϕ TO BE CONCENTRIC TO R WITHIN .005 F.I.R

LYKOTE TYPE 'Z' TO ALL MALE

3 1/2", MUST BE FLUSH WITH TUBE O.D.

SUIT EXTRUDED DIA. OF PT.10 IF NECESSARY.

E.P. 105-N-1900-C200

FIGURE D-1
HORIZONTAL ROD

UNCLASSIFIED

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LEGEND

ANN ANNUNCIATOR
PI PRESSURE INDICATOR
PS PRESSURE SWITCH
MS MOTOR CONTROL SYSTEM
MOT MOTOR

EV SOLENOID VALVE
ACC ACCUMULATOR
FR FLOW RESTRICTOR
MS MOTOR STARTER

TO OIL TANK

DESCRIPTION OF NORMAL ROD OPERATION

EACH ROD IS ASSIGNED A PARTICULAR OPERATING FUNCTION BY POSITION OF THE ROD SERVICE SELECTOR SWITCH LOCATED ON THE ROD SERVICE SELECTOR SWITCH PANEL. WITH THIS SWITCH IN THE "NORMAL" POSITION, THE ROD CAN BE CONTROLLED ENTIRELY FROM THE CONTROL CONSOLE EXCEPT FOR CERTAIN SAFETY ELECTRICAL INTERLOCKS RESTRICTING SOME TYPES OF ROD MOVEMENTS.

AT THE ROD CONTROL CONSOLE, RODS ASSIGNED TO NORMAL OPERATION ARE CONTROLLED BY THE SELECTED ROD CONTROL SWITCH. THIS SELECTION IS MADE FROM THE SELECTOR SWITCHES LABELED 1 TO 6. ROD CONTROL SWITCHES OPERATE THE SOLENOID VALVES IN THE LOW PRESSURE OIL SYSTEM.

WHEN A POWER SETBACK OCCURS ALL RODS ASSIGNED TO THE ROD CONTROL CONSOLE WILL OPERATE, OVERRIDING THE ROD CONTROL SWITCHES REGARDLESS OF THE POSITION OF THE ROD CONTROL SWITCH. THE POWER SETBACK FEATURE CAN BE BYPASSED FOR ANY ROD BY OPERATING THE "P. S. BYPASS" SWITCH. THE "CANCEL" SWITCH RETURNS ALL ROD SELECTOR SWITCHES INCLUDING "P. S. BYPASS" TO THE NORMAL POSITION.

THE LABELED SECTION ON THE OUTSIDE OF EACH SWITCH IN THE ROD SELECTOR SWITCH GROUP IS FOR INDICATION ONLY I.E. "CONT'D SETBACK" INDICATES THE ROD IS IN THE CONTROLLED SETBACK CIRCUIT.

HIGH PRESSURE OIL SYSTEM IS USED ON ROD SCRAM AND ROD SCRAM TEST ONLY.

FIGURE D-1
HORIZONTAL ROD DRIVE INSTRUMENTATION

AIM

Association for Information and Image Management

1100 Wayne Avenue, Suite 1100
Silver Spring, Maryland 20910
301/587-8202

Centimeter

Inches

MANUFACTURED TO AIM STANDARDS
BY APPLIED IMAGE, INC.

4

of

5

G-132-673
REC-GE RICHLAND, WASH.

CO
M
O:

FROM SUPPLY 1500 PSIG
FORWARD - REVERSE VAL

CHECK VALV
CONTROL LINE

PRESSURE EQUALIZIN
CONT BY 1

UNCLASSIFIED

Figure D-3

Horizontal Rod Drive Cylinder Circuit

UNCLASSIFIED

Figure D-4

Horizontal Rod Drive Test Assembly

UNCLASSIFIED

TO ROD SCRAM
ACCUMULATOR MANIFOLD

LEFT SID
RIGHT SI
POWERED

G-132-674

AEC-GE RICHLAND, WASH

E UNIT SHOWN.
DE UNIT IDENTICAL EXCEPT PUMPS
BY STATION GENERATOR.

Figure D-5

Horizontal Rod Drive Power Unit

UNCLASSIFIED

Figure D-6
Valve Rack Test Assembly

DECLASSIFIED

FIGURE D-7

SERVICE SELECTOR SWITCH ACTION (ROD NO. 3 SHOWN)

Assigned Safety Rod

When the Service Selector Switch is turned to "Assigned Safety Rod" its 3H8-15 contact closes, energizing the 3K8 relay. Contact 3K8-7 on the 3K8 relay turns on the "Assigned Safety Rod" indicating light (2N6-L1) behind the pushbutton selector switches. The functions of other 3K8 relay contacts will be described later.

Assigned Power Setback

When the Service Selector Switch is turned to "Assigned Power Setback" its contact 3H8-5 closes, turning on the "Assigned Power Setback" indicating light (6N6-L2) behind the pushbutton selector switches. Contact 3H8-17 closes; its function is explained on Figure D-8.

Manual Control

When the Service Selector Switch is turned to "Manual", its 3H8-6 contact closes, turning on the light 4N6-L2 behind the pushbutton selector switches. This light indicates that the rod has been placed in the manual rod selection setback circuit; if the bypass pushbutton 523N6 is depressed, this setback function is cancelled and the light turned off.

WITHDRAWAL

When the Service Selector Switch is turned to WITHDRAWAL, its 3H8-18 contact closes, turning on the indicating light 523N6-L2 behind the pushbutton selector switches. Contacts 3H8-1, -2, -7, -9, -13, -17, -20, and -22 also close. The more significant functions of these contacts in other circuits will be described later. Functions not described are those associated with annunciation, or which have been discontinued since the drawing was completed.

OFF

When the Service Selector Switch is turned to OFF, its 3H8-1 contact closes. The function of this circuit is described on Figure D-12. Contacts 3H8-2, -4, -7, -9, -11, -13, -16, and -22 also close. The more significant functions of these contacts in other circuits will be described later. Functions not described are those associated with annunciation, on which have been discontinued since this drawing was completed.

INDEX 8

SERVICE SELECTOR SWITCHES

ROD NO.	1	2	3	4	5	6	7	8	9	10	11	12 *	13	14	15 *	16	17	18	19
3	CA9(+)	CA9(-)	CG11(+)	CGC	8-697(+)	3H8-15	8-1	3K8	108	8-88	3K8-7	2NG-L1	8-175	3H8-5	2NG-L2	3H8-8	8-2G2	1NG-7	8-345
4	CA9(+)	CA9(-)	CG11(+)	CGC	8-697(+)	4H8-15	8-2	4K8	208	8-89	4K8-7	3NG-L1	8-176	4H8-5	2NG-L2	4H8-8	8-2G3	7NG-7	8-350
5	CA9(+)	CA9(-)	CG11(+)	CGC	8-697(+)	5H8-15	8-3	5K8	308	8-90	5K8-7	4NG-L1	8-177	5H8-5	2NG-L2	5H8-8	8-2G4	13NG-7	8-355
6						6	4	6	4	91	6	20	178	6	24	6	265	19	352
7						7	5	7	5	92	7	26	179	7	30	7	266	25	353
12						12	6	12	6	93	12	32	180	12	34	12	267	31	354
13						13	7	13	7	94	13	38	181	13	42	13	268	37	355
14						14	8	14	8	95	14	44	182	14	48	14	269	43	356
15						15	9	15	9	96	15	50	183	15	54	15	270	49	357
16						16	10	16	10	97	16	56	184	16	60	16	271	55	358
17			CG11(+)			17	11	17	11	98	17	62	185	17	66	17	272	61	359
18			CG12(+)			18	12	18	12	99	18	68	186	18	72	18	273	67	360
21						21	13	21	13	100	21	74	187	21	78	21	274	73	361
22						22	14	22	14	101	22	80	188	22	84	22	275	79	362
23						23	15	23	15	102	23	86	189	23	90	23	276	85	363
24						24	16	24	16	103	24	92	190	24	96	24	277	91	364
25						25	17	25	17	104	25	98	191	25	102	25	278	97	365
26						26	18	26	18	105	26	104	192	26	108	26	279	103	366
27						27	19	27	19	106	27	110	193	27	114	27	280	109	367
28						28	20	28	20	107	28	116	194	28	120	28	281	115	368
29						29	21	29	21	108	29	122	195	29	126	29	282	121	369
31			CG12(+)			31	22	31	22	109	31	128	196	31	132	31	283	127	370
32			CG13(+)			32	23	32	23	110	32	134	197	32	138	32	284	133	371
33						33	24	33	24	111	33	140	198	33	144	33	285	139	372
34						34	25	34	25	112	34	146	199	34	150	34	286	145	373
35						35	26	35	26	113	35	152	200	35	156	35	287	151	374
36						36	27	36	27	114	36	158	201	36	162	36	288	157	375
37						37	28	37	28	115	37	164	202	37	168	37	289	163	376
38						38	29	38	29	116	38	170	203	38	174	38	290	169	377
39						39	30	39	30	117	39	176	204	39	180	39	291	175	378
41						41	31	41	31	118	41	182	205	41	186	41	292	181	379
42						42	32	42	32	119	42	188	206	42	192	42	293	187	380
43			CG13(+)			43	33	43	33	120	43	194	207	43	198	43	294	193	381
44			CG14(+)			44	34	44	34	121	44	200	208	44	204	44	295	199	382
45						45	35	45	35	122	45	206	209	45	210	45	296	205	383
46						46	36	46	36	123	46	212	210	46	216	46	297	211	384
47	CA9(+)	CA9(-)	CG14(+)	CGC	8-697(+)	47H8-15	8-37	47K8	3708	8-124	47K8-7	218NG-L1	8-211	47H8-5	222NG-L1	47H8-8	8-298	217NG-7	8-38

FOR CONTINUUM

NOTES:

- NOTES:
1. CONTACTS 56K1-1 AND 56K1-5 ON INDEX 6, AND 56K1-5 ON INDEX 12, OPEN AFTER A SCRAM TO PREVENT WITHDRAWAL OF ANY RODS NOT SELECTED FOR WITHDRAWAL ON INDEX 5. CONTACTS 108K35-1 AND 108K35-3 CLOSE 5 SECONDS AFTER A
 2. CONTACTS #1 & #2 ARE CLOSED ONLY WHEN HANDLE IS PULLED OUT AND ONLY IN POSITIONS 4 & 5.
 - 3 *OTHER HALF OF LAMP IS LOCATED ON DWG REF*7

INDEX 8

H.W.N.U. 708
W.N.O.

COINCIDE WITH
3 SB-10 DEVELOPMENT

"WITHDRAWAL" AND "OFF" ENERGIZING CIRCUIT

(SEE TABULATION FOR CONTINUATION)

SBIO SERVICE SELECTOR SWITCH DEVELOPMENT

CONTACT	INDEX NO.	POSITIONS				
		1 (SAFETY ROD)	2 (POWER SETBACK)	3 (MANUAL)	4 (WITHDRAWAL)	5 (OFF)
1-1	8				X SEE NOTE 2	
1-2	52				X	X
1-3	SPARE	X				
1-4	35					X
1-5	8		X			
1-6	8			X		
1-7	18				X	X
1-8	SPARE	X	X	X	X	X
1-9	52				X	X
1-10	SPARE	X	X	X	X	
1-11	35					X
1-12	SPARE	X	X	X		
1-13	17				X	X
1-14	SPARE	X			X	
1-15	8	X				
1-16	9				X	X

FOR CONTINUATION SWITCHES SEE DRAWING

20	21	22 *	23	24	25	26	27	28	29	30 *	31	32	33 *	34
B-438	523NG-6	4NG-L2	2NG-7	3NG-7	4NG-7	5NG-7	6NG-7	3HB-18	B-523	523NG-L2	B-610	3K35-7	5NG-L2	3HB-1
B-437	524NG-6	10NG-L2	8NG-7	9NG-7	10NG-7	11NG-7	12NG-7	4HB-18	B-524	524NG-L2	B-611	4K35-7	11NG-L2	4HB-1
B-436	525NG-6	16NG-L2	14NG-7	15NG-7	16NG-7	17NG-7	18NG-7	5HB-18	B-525	525NG-L2	B-612	5K35-7	17NG-L2	5HB-1
439	526	22	20	21	22	23	24	6	526	526	613	6	23	6
440	527	28	26	27	28	29	30	7	527	527	614	7	29	7
441	528	34	32	33	34	35	36	12	528	528	615	12	35	12
442	529	40	38	39	40	41	42	13	529	529	616	13	41	13
443	530	46	44	45	46	47	48	14	530	530	617	14	47	14
444	531	52	50	51	52	53	54	15	531	531	618	15	53	15
445	532	58	56	57	58	59	60	16	532	532	619	16	59	16
446	533	64	62	63	64	65	66	17	533	533	620	17	65	17
447	534	70	68	69	70	71	72	18	534	534	621	18	71	18
448	535	76	74	75	76	77	78	21	535	535	622	21	77	21
449	536	82	80	81	82	83	84	22	536	536	623	22	83	22
450	537	88	86	87	88	89	90	23	537	537	624	23	89	23
451	538	94	92	93	94	95	96	24	538	538	625	24	95	24
452	539	100	98	99	100	101	102	25	539	539	626	25	101	25
453	540	106	104	105	106	107	108	26	540	540	627	26	107	26
454	541	112	110	111	112	113	114	27	541	541	628	27	113	27
455	542	118	116	117	118	119	120	28	542	542	629	28	119	28
456	543	124	122	123	124	125	126	29	543	543	630	29	125	29
457	544	130	128	129	130	131	132	31	544	544	631	31	131	31
458	545	136	134	135	136	137	138	32	545	545	632	32	137	32
459	546	142	140	141	142	143	144	33	546	546	633	33	143	33
460	547	148	146	147	148	149	150	34	547	547	634	34	149	34
461	548	154	152	153	154	155	156	35	548	548	635	35	155	35
462	549	160	158	159	160	161	162	36	549	549	636	36	161	36
463	550	166	164	165	166	167	168	37	550	550	637	37	167	37
464	551	172	170	171	172	173	174	38	551	551	638	38	173	38
465	552	178	176	177	178	179	180	39	552	552	639	39	179	39
466	553	184	182	183	184	185	186	41	553	553	640	41	185	41
467	554	190	188	189	190	191	192	42	554	554	641	42	191	42
468	555	196	194	195	196	197	198	43	555	555	642	43	197	43
469	556	202	200	201	202	203	204	44	556	556	643	44	203	44
470	557	208	206	207	208	209	210	45	557	557	644	45	209	45
471	558	214	212	213	214	215	216	46	558	558	645	46	215	46
B-472	559NG-6	220NG-L2	218NG-7	219NG-7	220NG-7	221NG-7	222NG-7	4THB-18	B-559	559NG-L2	B-646	4TK35-7	222NG-L2	4THB-1

ION SEE DWG. REF. 3.

GRAM

CONTACT	INDEX NO.	1 (SAFETY ROD)	2 (POWER SETBACK)	3 (MANUAL)	4 (WITHDRAWAL)	5 (OFF)
1-17	9		X			
1-18	8				X	
1-19	SPARE	X				
1-20	50				X	X
1-21	SPARE	X	X			
1-22	SPARE					X

FIGURE D-1
SERVICE SELECTOR SWITCHES
ELEMENTARY DIAGRAM

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FIGURE D-8

MANUAL ROD OPERATION AND THE EFFECT OF OTHER ROD FUNCTIONS ON MANUAL CONTROL

NORMAL MANUAL OPERATION

The circuits are shown with relay contacts in the normal, "manual control" position, with the rod assigned to gang No. 1. To operate No. 3 rod manually, the following sequence is followed: The 1N6 selection pushbutton is depressed, assigning rod No. 3 to gang No. 1. The rod may now be moved in either direction at either fast or slow speed by turning the gang No. 1 operating circuit. Positions on the gang No. 1 switch are off, out-slow, out-fast, in-slow, and in-fast.

EFFECT OF OTHER FUNCTIONS ON MANUAL CONTROL

Fast Rod Withdrawal Interlock

If nuclear instrumentation indicates the reactor is supercritical, the fast rod withdrawal interlock circuit opens the 19K41-1 contact in the out-fast circuit to prevent actuation of the "travel-fast" solenoid in the out direction.

Power Setback

Where the power setback circuit is tripped (see Figure D-9, the 18K4 contact intermittently closes and opens. If the power setback bypass button (523N6-1) has not been depressed, and the rod had been selected for manual control (1K7-7 contacts closed), the power setback circuit directly energizes the in-slow solenoid. At the same time, 1K4-3 contacts open in the out-slow and out-fast circuits to prevent rod withdrawal. These contacts are not intermittent, but stay open until the power setback trip is cleared. The rod may be selected directly for power setback without being selected for manual control using the service selector switch (Figure D-7) which closes the 3H8-17 contacts.

Assigned Safety Rod

If the rod is not an assigned safety rod, the 3K8-5 contact is open, preventing withdrawal of the rod until all the assigned safety rods are out, thus closing the 1K12-1 contact.

If the rod is assigned as a safety rod, the 3K8-5 contact is closed and the 3K8-2 contact is open. The 3K8-10 contact opens but the 3K13-8 contact remains closed. The rod may now be withdrawn, but not inserted, until the out-limit switch is closed. When the out-limit switch is closed, the 3K13-8 contact opens, de-energizing all the manual control functions. The 3K8-2 contact remains open, preventing accidental insertion of the rod by the setback circuit. The rod may now be actuated only by a safety system trip, or by the all-rod insertion circuit, which actuates the in solenoid of all rods not assigned to withdrawal or off.

Withdrawal

When the service selector switch is turned to withdrawal, relay 3K52 (Figure D-11) is energized, closing its 3K52-3 contact, completing the out-slow circuit, and opening its 3K52-2 contact, preventing rod insertion. The rod may now be withdrawn whether or not all safety rods are out, and whether or not the safety circuit is made up.

Off

When a rod's service selector switch is turned to OFF, the entire manual control circuit for that rod is de-energized by opening the 3K35-4 contact. The 3K35-even contact is also opened, preventing rod insertion by the power setback circuit or the all-rod insertion circuit.

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Figure D-8

Manual Operating Circuit

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FIGURE D-9
POWER SETBACK (2C) CIRCUIT

PART A

Normal Energized Condition

All trip contacts are closed and the 1K4 breaker is energized. The 1K4-2 contact is open, thus relays 16K4, 17K4, and 18K4 are all de-energized. The 17K4-1 contact is open, and the 16K4-2 contact is closed.

Tripped Condition

When any trip contact opens the 1K4 breaker is de-energized and the 1K4-2 contact closes. The 17K4 and 18K4 relays are immediately energized through the closed 16K4-2 contact. As soon as the 17K4 relay is energized, the 17K4-1 contact closes, and the 16K4 timer relay starts timing. At the end of 0.67 seconds (adjustable) the 16K4 relay is energized, opening the 16K4-2 contact, de-energizing the 18K4 relay and starting the 17K4 timer. After 0.33 seconds (adjustable), the 17K4 relay de-energizes, opening the 17K4-1 contact and immediately de-energizing the 16K4 relay, which closes the 16K4-2 contact and starts the cycle over again. The result is to alternately energize the 18K4 relay for 0.67 seconds and de-energize it for 0.33 seconds; this results in driving all rods which are assigned to power setback in at six feet per minute for two-thirds of each second or an effective insertion rate of four feet per second by energizing the in-slow solenoid through the 18K4 relay contacts (see rod power circuit, Figure D-8).

PART B

Part B illustrates two trip logic circuits in the power setback circuit. In the two subcritical flux monitor channels, one-out-of-two logic is used. If either contact 1T250-S or 2T250-S opens, then the corresponding relay 19K4 or 20K4 is de-energized. De-energizing the 19K4 or 20K4 relays opens the respective contacts 19K4-1 and 20K4-1. Opening either of these contacts de-energizes the 2K4 relay (unless the bypass is closed). De-energizing the 2K4 relay opens the 2K4-1 contact in Part A and trips the setback circuit.

Two-out-of-three trip logic is illustrated in the intermediate range period trip circuitry. Again, each channel has trip contacts which de-energize corresponding relays. Tracing through the matrix composed of 21K4, 22K4, and 23K4 contacts, it is clear that de-energizing any two of the relays associated with these trip contacts results in de-energizing the 3K4 relay, opening the 3K4-1 contact in Figure 13A and tripping the setback circuit.

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A. POWER SETBACK CIRCUIT

B. FAST PERIOD TRIP LOGIC, SUB-CRITICAL MONITORING INSTRUMENTATION

Figure D-9

Power Set back Circuit

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FIGURE D-10

Normal Operating Conditions

Part A

All the trip contacts in both strings are closed. The 1K1 and 2K1 breakers are energized.

Part B

Contacts 1 and 3 of the 1K1 and 2K1 relays are closed, energizing the 3V9 scram solenoid. Thus the scram hydraulic system is closed off, permitting manual operation of the rod (in this case, again Rod 3).

Scrammed Condition

Any of the trip contacts in either string in Part A will call for a scram by de-energizing either the 1K1 or the 2K1 breaker. The contacts of the 1K1 and 2K1 breakers then remove power from the scram solenoid power circuit (Part B).

Reset Procedure

Part C

The 1K1 and 2K1 breakers are not self-resetting; to reset them, the auxiliary coils 1K1R and 2K1R must be energized. To energize the 1K1R and 2K1R coils, all the series contacts in the reset circuit must either be closed or bypassed. If the circuit is being made up in preparation for a reactor startup, all of the contacts in both the reset primary circuit strings will be closed. If the circuit is being made up during shutdown for testing purposes, most of the contacts in the reset circuit will have to be bypassed, plus those contacts in the primary strings which are tripped open, for example, by low flow or low pressure indications.

Certain contacts in both circuits cannot be bypassed, thus affording scram protection in the reactor in the unlikely event that it becomes supercritical during the testing period. The trips which cannot be bypassed are: the intermediate range excessive flux contacts, and the excessive power rate contacts. Furthermore, the ball-3X system must be unlocked and made up, and the ball drain valves must be closed. Contacts from intermediate range instrumentation indicating that the reactor power is less than a value between .001 and 4 MW (to be determined) must also be closed before the reset circuit can be actuated.

Auxiliary Functions

Part D

Breaker contacts 1K1-10 and 2K1-10 close when the safety circuit is tripped. Timer relay 66K1 energizes and closes five seconds later. When the 66K1-1 contact closes, relays 1K35 and 2K35 energize. The function of the 1K35 relay contacts are described on Figures D-11 and D-12.

Breaker contacts 1K1-7 and 2K1-7 open when the safety circuit is tripped. All of the relays in series with these contacts are thus de-energized. Relays 47K1 and 48K1 are timer relays whose contacts open after a timed interval after de-energization. The functions of the 52K1 relay contacts are described on Figures D-11 and D-12.

B. HORIZONTAL ROD SCRAM CIRCUIT

Figure D-10
Primary (1X) Safety Circuit

PRIMARY (IX) RESET CIRCUIT

D. AUXILIARY FUNCTIONS

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A. PRIMARY (IX) SAFETY CIRCUIT

2K9
AUXILIARY POWER CIRCUIT

B. HORIZONTAL ROD SCRAM CIRCUIT

C. PRIMARY (1X) RESET

Figure D-10
Primary (1X) Safety Circuit

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FIGURE D-11

PARTS A AND B (Rod No. 3 Illustrated)

When the service selector switch (3H8) for the rod is switched to the WITHDRAWAL position, the 3H8-9 contact is closed. The service selector switch handle is then pulled out closing the 3H8-2 contact. At the same time, the circuit in Figure 16B functions as follows: when the service selector switch is turned to WITHDRAWAL and pulled out, the 3H8-1 contact closes. This energizes 110K8 through the closed 108K8-2 contact. After a five second time delay, the 109K8 relay energizes. After a further 0.5 second time delay the 108K8 relay energizes, opening the 108K8-2 contact and de-energizes the 110K8 and 109K8 relay. Thus the effect of the circuit is to energize the 109K8 relay for 0.5 seconds after a 5 second time delay. When the 109K8 relay energizes, the 109K8-3 in Part A closes and remains closed for 0.5 seconds. The 3K52 relay is thus energized through the path of contacts 109K8-3, 3H8-2, and 3H8-9. When the 3K52 relay is energized, the 3K52-5 contact closes, also energizing the 3K52 relay through the 3K52-5 and 3H8-9 contacts. After the five seconds, the service selector switch handle may be allowed to snap back in, opening the 3H8-1 and 3H8-2 contacts, but leaving the 3K52 relay energized. The purpose of the five second time delay is to allow time for a rod to scram even though it is being switched to withdrawal when a scram trip occurs.

When a safety circuit trip occurs, the 52K1-3 contact (see Figure D-10) opens, thus de-energizing the entire withdrawal circuit, including all energized 1K52 relays (one for each rod). Five seconds after 1X trip occurs, the 1K35-3 contact closes, (see Figure 14) thus allowing the circuit to again be used to switch rods to WITHDRAWAL. Thus, all rods (including those switched to WITHDRAWAL) scram when a safety circuit trip occurs.

Energizing the 3K52 relay in the withdrawal circuit closes the 3K52-7 contacts in the 1X safety circuit (Figure D-10). Switching the service selector switch to WITHDRAWAL closes the 3H8-16 contacts on Figure D-10). Thus, the 3V9 scram solenoid is energized from an auxiliary power circuit which is not de-energized after the safety circuit is tripped. This allows rods to be switched to WITHDRAWAL and withdrawn with the safety circuit tripped during an outage.

PART C

The proposed circuit of Part C counts the number of rods switched to WITHDRAWAL or OFF. Each rod, when switched to these positions, opens its H8-22 contacts and adds 2300 ohms resistance to that leg of a bridge circuit. When the resistance exceeds 8000 ohms (4 rods), the current through the DC 1K11 relay reverses and the relay is de-energized. This opens the 1K11-1 contact and de-energizes the 2K11 relay, which in turn de-energizes the 2K9 power relay. Contacts of the 2K9 relay, (2K9-1 and 2K9-3) are inserted in series with each leg of the auxiliary power circuit in Figure D-10. Thus, when four or more rods are switched to WITHDRAWAL or OFF, the auxiliary power circuit in the 1X safety circuit (Figure D-10) is de-energized, and the scram solenoids for those rods must be energized by the normal power circuit. If the safety circuit is tripped, these rods will scram.

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A. WITHDRAWAL CIRCUIT

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B. WITHDRAWAL & OFF TIME DELAY ENERGIZING CIRCUIT

C. COUNTING CIRCUIT FOR WITHDRAWAL AND OFF (PROPOSED)

Figure D-11

Withdrawal Circuit

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FIGURE D-12

When the service selector switch (3H8) (Figure D-7) is switched to the OFF position, its 3H8-11 and the 3H8-4 contacts close. When the service selector switch handle is pulled out, the 3H8-1 contact in Figure 16B closes starting the timer circuit. At the end of five seconds, the 109K8-1 contact closes for .5 seconds (same timing circuit as for withdrawal function - Figure 16) energizing the 3K35 relay through the contacts 3K35-2, 3H8-4, 109K8-1, 52K1-1, and 2K52-3. As the 3K35 relay energizes, the 3K35-1 contact closes and the 3K35-2 contact opens, and the 3K35 relay remains energized through the 3K35-1 and 3H8-11, contacts. Note that the circuit is not de-energized when the safety circuit is tripped, as is the withdrawal circuit. (There is no safety circuit trip contact in the OFF circuit power line as in the WITHDRAWAL circuit power line).

When the service selector switch is switched to OFF, the 3H8-22 contacts on Figure D-11 open, and the rod is counted as being OFF or WITHDRAWN.

When the 3K35 relay is energized, it energizes the 3V9 scram solenoid in Figure D-10 through the 3H8-16 contacts. If four or more rods are switched to WITHDRAWAL or OFF, the counting circuit acts to de-energize the auxiliary power circuit and return the 3V9 solenoid to the normal power circuit; these rods would then scram with the rest of the horizontal rod system. If, however, only three rods were switched to WITHDRAWAL or OFF, those rods on withdrawal would scram because of de-energization of their K52 relay, but those on OFF would not scram, since their K35 relays would remain energized and their scram solenoids would remain energized by the auxiliary power circuit.

Figure D-12
Rod Off Circuit

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DESCRIPTION OF WORK	COST ACCOUNT

LEGEND

DPI	DIFFERENTIAL PRESSURE INDICATOR
FE	FLOW ELEMENT
HP	HIGH PRESSURE
LP	LOW PRESSURE
PI	PRESSURE INDICATOR
PRV	PRESSURE REDUCING VALVE
CAS	COMPRESSED AIR SUPPLY
E	ELECTRICAL
FI	FLOW INDICATOR
FT	FLOW TRANSMITTER
CAS	AIR SUPPLY
FIT	FLOW INDICATOR TRANSMITTER
ATM	ATMOSPHERE
ANN	ANNUNCIATOR
(Circled Number)	EQUIPMENT PIECE NUMBER
RTR	RIGHT TRANSDUCER ROOM OR VALVE ROOM
LTR	LEFT TRANSDUCER ROOM OR VALVE ROOM

NOTES

1. TUBES ACTUATING A TRIP SIGNAL SHALL BE IDENTIFIED PER REF N83
2. CONTACT REPRESENTS AN ACCUMULATION OF TRIPS; ACTUATION OF ANY TRIP SHALL OPERATE THE SAFETY CIRCUIT.
3. HIGH FLOW TRIP ADJUSTMENT WILL BE SEMI-PERMANENT TYPE

FIGURE D-13
FLOW MONITOR SYSTEM

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DESCRIPTION OF WORK	COST ACCOUNT

QTY.	SY.

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LEGEND

TE TEMPERATURE DETECTOR
 JB JUNCTION BOX
 ACCUMULATION OF TWO TRIPS)

VT

LET

LE
 0928

FIGURE D-14
 ZONE TEMPERATURE MONITOR

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OPERATION

POWER RATE INSTRUMENTATION

* INDICATES CHAMBER USED WITH GALVANOMETER

HIGH LEVEL INSTRUMENTATION

~LEGEND~

RAR - RADIATION AMPLIFIER
RR - RADIATION RECORDER
S - SCRAM ANNUNCIATOR
SB - SETBACK
PW-R1 - POWER RATE INDICATOR

NOTES:
1. RATE INDICATOR SCALE CAN BE CALIBRATED IN MORE CONVENTIONAL UNITS AFTER CALIBRATION OF HIGH LEVEL CHANNELS IS ESTABLISHED
2. SB-114 THRU SB-115 & S-11 THRU S-12 ARE HIGH FLUX SETBACK & NO TRIP ANNUNCIATORS

HIGH FLUX LABEL
SB-114, THRU SB-115
S-11 THRU S-12
HIGH FLUX S.C. FLUX LABEL
SB-130, S-28, S-29, S-30
SLOW FLUX OSCRY
S-24, S-27, S-28
EXCESS POWER
INTERRUPTIBLE TIME
SB-18-17, S-26
EXCESS POWER
INTERRUPTIBLE TIME
SB-18-17, S-26
EXCESS POWER
INTERRUPTIBLE TIME
SB-18-17, S-26

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FIGURE D-18
HIGH LEVEL NUCLEAR INSTRUMENTATION

DESCRIPTION OF WORK	COST ACCOUNT
Reactive Nuclear Monitoring	0001-601

IONIZING TRANSDUCER

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LEGEND

- RI - RADIATION INDICATOR
- RR - RADIATION RECORDER
- RAA - RADIATION AMPLIFIER
- RAE - EQUIP. PIECE NUMBER ENCLOSURE
- CPS - COUNTS PER SECOND
- J - JETBACK ANNUNCIATOR
- JB - JETBACK ANNUNCIATOR
- A - ALARM ANNUNCIATOR
- FRWI - FAST ROD WITHDRAWAL INTERLOCK
- DENOTES 2 OUT OF 3 COINCIDENT ROD INTERLOCK

NOTES:
1. CALIBRATION OF INTERMEDIATE LEVEL "LOG LEVEL" INDICATORS IS IDENTICAL TO RECORDER CALIBRATION.

FAST PERIOD	S-3	S-4	S-5
HIGH FLUX LEVEL	S-7	S-8	S-9
UNDECAVED FLUX	S-100	S-101	S-102
FAST PERIOD SUB-CRITICAL	S-107	S-108	A-175
INSUFFICIENT COUNT RATE	A-172	A-174	
ROD INTERLOCK	A-176

DESCRIPTION OF WORK	COST ACCOUNT
Nuclear Nuclear Monitoring - 0018-001	

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FIGURE D-14
LOW AND INTERMEDIATE LEVEL
NUCLEAR INSTRUMENTATION

Figure D-17

Ball System Schematic

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Figure D-18
Ball Hopper Assembly

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FIGURE D-19

PARTS A AND B

The functioning of the Ball 3X safety circuit is identical to that of the Primary (1X) safety circuit. The Ball 3X safety circuit is shown; a reset circuit similar to that for the 1X safety circuit but containing no interlocks, is also provided but not shown.

When one of the safety circuit strings is opened by any of the four trip contacts, the 1K3 or 2K3 breakers are de-energized. The 1K3-1 and 1K3-3 or 2K3-1 or 2K3-3 contacts then de-energize the power circuit to the holding solenoids of all ball hoppers on the Battery A system in Part B. Similarly, 1K3 or 2K3 contacts de-energize the solenoids in a battery B circuit (not shown). Thus, when either safety circuit string is de-energized, both battery systems are de-energized, and the balls are dropped. The dual battery system provides protection against the possibility that failure of a single holding solenoid could drop a channel of balls into the reactor. Other contacts on the 1K3 and 2K3 breakers also trip the 1X safety circuit (see Figure D-10). This feature initiates a 1X scram coincident with a ball 3X drop which is necessary to initiate the programmed shut down of the cooling system.

PART C

The seismoscope trip circuit is a common application of two-out-of-three redundant trip philosophy. When two of the seismoscope trip contacts open, the 3K3 and 4K3 relays are de-energized, thus opening (unless bypassed) both safety circuit strings.

The emergency cooling system circuit consists of two, identical, completely-independent electrical circuits. The 156K33 contact opens when one of the circuits is energized, and the 157K33 contact opens when the other circuit is energized. Energizing either of the emergency cooling water circuits thus causes a Ball 3X trip.

The high-flux-level-after-scrum trip circuit operates as follows: When the reactor power drops to a pre-set level (25% shown), the 13T252-L, 14S252-L and 15T252-L contacts close, energizing the 14K3, 15K3 and 16K3 relays, and closing their 1 and 3 contacts. If at least two of these relays are energized, the circuit is completed to energize the 7K3 and 8K3 relays. The 7K3 and 8K3 relays are hold energized for an adjustable time period after a 1X safety circuit trip by the 47K1 timed drop-out relay. Thus, the power level must drop to 25 per cent within the adjustable time period in order to provide a parallel path to energize the 7K3 and 8K3 relays before the timer drops out. If the power level is not properly reduced the 7K3-1 and 8K3-1 contacts (Part A) open and initiate the Ball 3X drop.

The reactor-supercritical-after-shutdown circuit operates in essentially the same manner as the high-flux-level-after-scrum circuit, except that the level trip setting is adjustable between .001 and 4 MW, and the time period is adjustable from 3 to 1200 seconds. Rather than providing a parallel path to energize the 9K3 and 10K3 relays, the timer relays operate to provide contacts in parallel with the 9K3-1 and 10K3-1 contacts in the safety circuits. The timer contacts (22K3 and 23K3) open at the end of the pre-set time interval, at which time the power level must have dropped below the trip point, thus energizing the 9K3 and 10K3 relays and closing their contacts, otherwise the safety circuit will be tripped.

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CIRCUIT

B. ONE OF TWO HARPER SOLENOID TRIP CIRCUITS (OTHER CIRCUIT IDENTICAL BUT POWERED BY BATTERY B)

Figure D-19

Ball 3-X Safety Circuit

C. BALL - 3X AUXILIARY CIRCUITS

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A. BALL - 3X SAFETY CIRCUIT

B. ONE OF TWO HARPER SO. CIRCUITS (OTHER CIRCUIT BUT POWERED BY BATTERY A)

SOLENOID TRIP
(IDENTICAL
TO CIRCUIT B)

C. BALL - 3X A

Figure D-19

Ball 3-X Safety Circuit

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APPENDIX E

THE CONFINEMENT SYSTEM

1. INTRODUCTION

The confinement system was described briefly in Section 7.1 "Airborne Fission Product Confinement System" in the first volume of this review. Parts of Section 4, "Reactor Hazards" also in the first volume described the operation of the system during various accidents. The purpose of this document is to supplement that information by discussing:

1. General Problems in Fission Product Confinement,
2. The Confinement Concept,
3. analytical Studies of the System Performance, and
4. The More Vital System Components.

2. GENERAL PROBLEMS IN FISSION PRODUCT CONFINEMENT

The best confinement is the cooled, clad fuel element. Major emphasis has been placed upon maintaining adequate cooling to the fuel elements and thus preventing release of fission products at the source.

The probability of releasing significant quantities of fission products from the fuel elements is believed to be extremely low. However, it is not possible to reduce this probability to zero. Thus, it is prudent to provide for limiting the potential release of fission products to the environs in the event they escape from the fuel elements in significant quantities.

Considerable reactor operating experience at Hanford combined with an analysis of the limited number of non-standard operations, which have occurred suggests that accidents which could result in significant fission product release from fuel elements in the reactor would most likely result

from a highly improbable sequence of events involving some combination of (1) uncorrected system or component deterioration, (2) unappreciated system weakness, (3) specific equipment failure, and (4) unexpected sequences of operational error. Indeed, before the fact, the causitive sequence of events might well be classed as "incredible".

Thus, it is not feasible to start with a broad spectrum of causitive sequences and design a confinement system to accommodate these sequences. Instead, a pattern of severe failures is postulated, including a "maximum design accident", and the system is designed to accommodate these failures. During the early stages of design relatively little attention is focused on the potential causitive sequences. Quantitative analyses will evaluate the performance of the system and indicate its basic capabilities and flexibility.

Mature judgement is required as it always is required in significant engineering problems - to determine if the system as designed, satisfactorily meets the requirements imposed upon it.

Finally, the quality of the completed system at the time of actual need is the ultimate criteria for evaluation. Regardless of the conservatism and excellence of design, thoroughness of evaluation, efforts to expose and rectify weaknesses and quality of construction, a devotion to maintenance, testing and continual system surveillance combined with proper modes of operation is essential if the system is to be expected to perform satisfactorily when called upon.

3. SUMMARY AND CONCLUSIONS

3.1 System Requirements

The NPR confinement concept has the goal of minimizing the release of fission products to the environs after all credible accidents. In

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particular: "The confinement system shall be designed so that the quantity and character of fission products which may be released to the surroundings will be such as not to require long-term evacuation of land outside the boundaries of the present Primary Control Zone of the HAPO reservation in the event of any credible reactor incident."⁽¹⁾

It is not possible to confine all fission products, even with a full pressure dome. And it has not been found feasible to provide any significant constraint on the release of noble gases while the building is exhausting through the filters. The design criteria call for:

1. Passing at least 99 per cent of the effluent gases from Zone I through the filters (except when the Zone I area is sealed).
2. Removing at least 99.9 per cent of the particulates from the filtered stream, and
3. Removing at least 50 per cent of the halogens from the filtered stream. (In practice about 95 per cent removal is expected).

3.2 Categories of Accidents Involving Release of Fission Products From Fuel

Three categories of accidents are identified and discussed as follows:

3.2.1 Fuel Melting Not Associated With Primary Loop Rupture

Fuel could be melted during a power excursion which caused the heat generation rate to exceed the maximum heat removal capacity of the coolant system, or by a coolant flow reduction below that required to remove heat from the fuel at the rate of generation during operation with the heat generation rates at, or below, designed limits. If the primary loop, which includes the process tubes, remains intact the fission products released from the fuel will remain in the primary

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loop coolant except for the normal system "bleed" which would be quickly reduced, or diverted to storage tanks. The fission products would thus not endanger the environs, nor would the confinement system be required.

3.2.2 Primary Loop Rupture

The rupture of a major line in the primary loop could result in melting of fuel due to inadequate cooling even though the reactor is immediately shutdown by insertion of control rods. The confinement system is designed primarily to handle this class of accidents. The approach is to vent the steam and reseal Zone I of the confinement system before any significant quantity of fission products can be released from the fuel. The performance of the confinement system is extensively discussed in following sections.

3.2.3 Fuel Melting Followed by Primary Loop Rupture

The primary loop consists of two essentially different parts: (1) The 1004 process tubes penetrating the reactor, and (2) The exterior piping, pumps and heat exchangers. These are discussed separately.

3.2.3.1 Process Tube Rupture

Steam vented from a ruptured process tube would be prefiltered to at least a limited extent by the graphite stack and conducted to the Zone 1 exhaust filters via a low pressure rupture disc and normally open butterfly valve in the reactor gas recirculation system. An alternate path, guarded by a rupture disc and a normally closed butterfly valve leads to the right rear pipe space (Zone I) in the reactor building. The alternate route is available in the event of excessive pressure buildup, or excessive release of fission products from the exhaust stack, and makes the entire 105-N (and 109-N) confinement Zone I available for storage of contaminated vapor.

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There has been some concern expressed on the possibility of an explosion occurring after a process tube rupture. The steam released from the tube contracting the hot graphite moderator will produce some hydrogen and carbon monoxide from the water gas reaction. While the mixture is in the reactor, where graphite temperatures are above the automatic ignition of the gases the remaining atmosphere will consist of helium, unreacted steam and only minute traces of other gases. Hence, there is no danger of combustion or explosion of the gases in the reactor stack.

During the initial venting when the mixture is exhausted out the stack and filters, a fire or explosion is likewise highly improbable. The gases will be at temperatures below the auto ignition temperature and it is believed that there are no "sparking" devices along the route.

Later in the transient it is possible, though not likely, that the gases will be vented into the right rear pipe space. While not considered likely the possibility must be acknowledged that an explosive mixture might come into contact with a "spark producing" instrument or device producing a fire or explosion. This event might damage the structural integrity of the Zone I; its ultimate effect on safety of environs is concluded to be small.

3.2.3.2 Primary Loop Major Line Rupture

If general fuel melting precedes a line rupture by sufficient time that the initial steam release contains fission products, the confinement system cannot accommodate the accident by venting steam and resealing the building before significant release of fission products could occur.

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The major protection against such an event would be provided by control of the nuclear process and by automatic (or manual) rapid depressurization of the primary loop in the event of overpressure. Thus, this accident would probably require a causative agent which was in itself catastrophic or an incredible sequence of events.

3.3 Qualitative Evaluation of the System

The confinement system is still under construction and detailed designs of some features is not yet complete. However, no problem areas have been identified which are likely to prevent the system from achieving the intent and essential requirements of such a system.

The system is expected to be capable of handling operational incidents of the more probable sort, and most of the more severe, but very improbable incidents. However, the system cannot accommodate multiple simultaneous primary loop line ruptures, or gross fuel melting followed by a rapid system depressurization through primary loop line ruptures which would require venting steam from the building. The latter class of incidents would probably only be caused by events, such as severe earthquake, or missile blast, which would in themselves be catastrophic.

The system has generally the capacity to:

1. Accommodate the steam pressure surge resulting from unrestricted flow from both sides of a complete circumferential rupture of any single primary coolant line without overpressurizing the building provided that the enthalpy of the primary coolant does not greatly exceed design operating values, and

2. Cleanly separate the non-contaminated* and fission product bearing effluent, from any single primary coolant line break, even with

*The term non-contaminated is a misnomer to some extent. Actually, the primary coolant may carry a radioactive burden of about 100 curies in addition to the short-lived nitrogen activity and an additional amount of activity (estimated to be as high as 2000 curies) which would be dislodged from fuel and piping surfaces as the primary coolant flowed to the coolant line break. Some, though not all, this activity will be entrained with the steam and released to atmosphere.

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total failure of the emergency once-through coolant system, provided that the specific fuel powers at the time of the accident does not appreciably exceed design "Hot Spot" values.

3.4 Confinement Concepts

The NPR confinement system is based on two concepts which are defined as follows:

3.4.1 Separation of Energy Concept

The volume of the primary coolant system is about 22,300 cubic feet. At design equilibrium operating levels the enthalpy of the primary coolant as compared to the enthalpy of saturated water at 212 degrees F is about 265,000,000 BTU or about 280,000 MW - sec.

A fundamental part of the confinement concept is the release of this energy to the atmosphere. Further, the release must be accommodated without damage to the building, for all steam release rates which are to be accommodated by sizing venting capacity so that pressure buildup will be within the design capacity of the buildings. Finally, the release must be completed and the building resealed before significant fission product release commences, following all accidents which are to be accommodated, by skillful primary system design, provisions for after-heat removal and other features.

3.4.2 The "Controlled Release" Concept

It is assumed in the confinement system design concept that after steam is vented, the Zone I area will be resealed and further emission from the building will be "controlled releases".

By controlled release is meant that after an accident, if passages to the air filters are opened, that at least 99 per cent of the

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steam or air escaping from the building will pass through the filters and not past seals, out cracks, or out other openings directly to the atmosphere. Hence "controlled release" might be termed "filtered release".

The following points should be made in regards to the "controlled release concept":

1. While it is assumed in the above criteria that the vents to the filters will be open this is merely to guide design and is not necessarily a prediction of actual system operation. It may be desirable and possible to keep the building sealed for a period of time after certain classes of accidents.

2. There is, on the other hand, no guarantee that total, or near total, retention of building gases will be possible for significant periods (say, at least a few hours) after all accidents. Thus, whether or not the building is sealed or whether there will be a release of filtered gas will be strongly influenced by the specifics of the incident.

4. ANALYTICAL STUDIES OF SYSTEM PERFORMANCE

4.1 Steam Venting Transients

By studies of steam venting transients is meant those calculations made to determine if the Zone I area can be vented with pressure transients within the range of building design pressures. The most severe accident cases from this standpoint are those which would release steam the most rapidly. As noted before one can postulate accidents giving release rates which would overpressure the building. However, many multiple line failures would not give such a result.

The calculations are roughly divided into two parts: (1) the transient accompanying release of the flashing steam water mixture from the break in the primary coolant line, and (2) the transient accompanying release of air

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on air and steam out the building vents. The building space pressure has essentially no effect on the release from the coolant system.

Release rates from the primary coolant system have been calculated by Locke⁽²⁾ and these results used by Stepnewski⁽³⁾ to calculate building pressure transient. Other calculations of steam release rates have been made by Stepnewski and Goss⁽⁴⁾ which check Locke's results.

Details on the assumptions made in the cases where both steam flow rates from the coolant line break and building pressure transients are tabulated in Table E-1.

The actual dynamics of the building pressure transient depend on the flow transient from the line break, the flow of steam into the various parts of Zone I and the venting capacity of the steam vents. In a particular case, transients are characterized by a set of differential equations. However, before exploring the particular cases, it is helpful to consider a much cruder model, that is the steady state venting capacity of the building vents.

The appropriate equation is:

$$W^2 = \frac{P}{K_t}$$

W = Steam flow rates, lbs/second.

P = Gage pressure referred to actual atmospheric in psig.

K_t = Resistance coefficient, $\frac{\text{lbs force sec.}}{\text{sq. inch lbs mass}}$

Using values of K_t employed by Stepnewski and the total vent area (13 vents six feet in diameter) one obtains the results shown in Figure E-2 where steam flow is shown as a function of building pressure. Note that at a pressure of five psig the vent capacity is 15,300 lbs/second.

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TABLE E-1
CALCULATION CASES FOR STEAM VENTING TRANSIENT

<u>Case</u>	<u>Break Location</u> ⁽¹⁾	<u>Break Size</u>	Reactor ⁽²⁾	Steam ⁽³⁾	Primary ⁽⁴⁾	Injection ⁽⁵⁾
			Heat Considered	Generators Considered	Pumps Considered	System Considered
A	Outlet Downcomer Upstream of Check Valve	Complete Failure at Point Where Pipe Diameter is Six Inches	Yes	Yes	No	No
B	Outlet Downcomer Downstream of Check Valves	Complete Failure 18-Inch Line	Yes	Yes	No	No
C	Inlet Header	Complete Failure 24-Inch Line	No	Yes	No	No
D	Inlet Riser	Complete Failure 18-Inch Line	Yes	Yes	Yes	Yes
E	Steam Generator Bypass	Complete Failure 18-Inch Line	No	No	No	No

(1) Break locations are shown in Figure E-1

(2) By reactor heat is meant the afterheat produced in fuel elements. Heat transfer from graphite as considered to be zero in all cases.

(3) "Yes" means that it is assumed that the steam generators continue to function normally removing important amounts of thermal energy in some cases. "No" means no heat transfer from the reactor coolant via the heat exchangers.

(4) "Yes" means that it is assumed that the pumps continue to function normally, i.e., will coast down and continue to operate until cavitation commences. "No" means that the effect of pumps is neglected.

(5) "Yes" means that makeup water at about 120 degrees F is continually injected at a rate of about 1000 lbs/second. "No" means that no addition of makeup water is assumed.

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TABLE E-2

PEAK STEAM RELEASE RANGES FROM COOLANT LINE BREAKS

<u>Case (1)</u>	<u>Peak Steam Flow, Lbs/Second</u>	<u>Corresponding Fictitious Steady State Pressure, Psig</u>
A	1,340	0.2
B	2,800	0.4
C	8,800	1.7
D	2,100	0.3
E	2,200	0.3

(1) Cases defined in Table E-1.

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While this number does not show the whole story, it is obvious with the large flow areas throughout the Zone I space that unless peak steam flows are somewhat in excess of this value overpressurization will not occur. Actually, peak steam flow in all cases shown in Table E-1 are lower than 15,300 lbs/second as is shown in Table E-2 so the "inertia" represented by the building volume is not required to handle any of the breaks considered. Of course other cases could be assumed that would give a closer approach to design pressures.

These conclusions are confirmed in general by other calculations made by Stepnewski. In a series of calculations aimed generally at specifically characterizing events after smaller breaks, he also covered complete circumferential failures of the inlet manifold and outlet manifold - obtaining release rates of 4,800 and 6,400 lbs/second, respectively.

The dynamics of transients are also of interest to show "inertial" effect of building volume decay rates, etc. These transients are shown in Figure F-2 through F-6, one figure for each case. The transient calculations were made with the use of the following assumptions:

1. Coolant temperatures were 384 degrees F inlet and 537 degrees F outlet.
2. The dump valves were considered not to be open.
3. Only ten of 13 available roof vents opened.
4. The gas flowing into building spaces is pure steam, that flowing out pure air.
5. Fog spray water available at 70 degrees F with condensing capacity of 170 lbs/second.

The importance of these assumptions and shown in Table C-1 are covered in the closing paragraphs of this Section 4.1.

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In summary, it appears that calculations made using conservative assumptions demonstrate an ability of the venting system to prevent overpressurization following a rather broad spectrum of line failures.

4.2 Calculations Pertaining to Separation of Non-Contaminated From Contaminated Vapors

4.2.1 General

Calculations have been made to assess the confinement system in terms of its adequacy in providing a clean separation between the relatively non-contaminated steam released early in a transient from the highly contaminated vapors expected after fuel melting.

In order to achieve a clean separation it is desirable to be able to reseal the steam vents at or before the time fuel melting starts without a subsequent building overpressurization. This is a somewhat conservative goal since there will be an elapse of time after fuel melting first starts and when fission products appear at the steam vents. Also, the vents to the filters can be opened, if necessary, to prevent overpressurization if steam flows are not excessive.

There are three key elements in a study of the capacity of the confinement system to produce clean separations. First, it is necessary to know steam release rates as a function of time. These rates will depend heavily on size of failure and location of failure, the operation of the dump system, as well as upon other parameters. A second important element is the time between failure initiation and the start of fuel melting. This time depends primarily on fuel power and exposure at the time of the incident plus the effectiveness of reactor cooling during the depressurization transient. A final factor is the condensing capacity of the fog spray system, and the storage capacity of the building air spaces at the design

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pressure limit. This last factor with steam release rates determines the building pressure transients after the steam vents are resealed, assuming the filters are left valved off.

4.2.2 Interpretation of Calculations

The results of the calculations discussed in some detail can be generalized as follows:

(1) With the largest breaks, the complete system will be rapidly depressurized and steam venting should be nearly complete long before fuel melting starts unless fuel powers are abnormally high. Fog spray operation will be required to prevent overpressurization if filters are left valved off; however, the fog spray design capacity greatly exceeds requirements under the most adverse conditions.

(2) With small breaks (breaks large enough to cause system depressurization but not much larger) the situation also appears favorable. Depressurization will be extended in time because of the restriction to flow out the break, if the dump system fails to work on demand. However, fuel cooling would continue during the transient and the vapor release rates should drop within the capacity of the fog spray system long before fuel melting starts if fuel powers were not quite abnormally high - say higher than the design hot channel value. If the moderator cooling system functions, fuel melting may never occur even if there is a complete failure of the reactor last-ditch cooling system.

(3) With intermediate size breaks, the separation cannot be assumed so highly probable. Much will depend on the fog spray system operating at at least its minimum capacity on operation of dumps, or on the cooling afforded by two phase flow during the depressurization transient. Actual fuel power will also be highly critical. If powers are at, near, or

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below normal peak powers the ability to obtain a clean separation will be enhanced.

If resealing without subsequent overpressurization cannot be effected, the effluent could probably be routed to the filters and not discharged directly to atmosphere.

(4) A number of equipment failures would have important effects on the condensing. The importance of the fog spray system and moderator cooling system is apparent from the preceding discussions. Functioning of primary closures is also important. Backup closures on steam vents and ventilation ducts have been provided; however, the backup closures cannot be operated as early in the transient as can the primary closures and they will provide effective sealing only at lower pressures than the primary closures. Use of backup closures would require, therefore, a delay in vent resealing and, probably, that the vent to the filters be opened as soon as the backup closure is closed.

Failure of both a primary and its backup, a highly unlikely situation, would represent a serious break in confinement. If operable, an exhaust fan might be turned on a low speed to ensure flow of air in from through the open vent with effluent air vented through the filters.

In practically all cases failure of the dump valves has been assumed as a part of the failure of the reactor last ditch cooling system in making analyses. However, the assumption is strictly arbitrary in nature and operability of the dump valves would greatly enhance the probability of obtaining a clean separation.

4.2.3 Assumptions and General Calculations Used

Certain assumptions and basic calculations were made

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pertinent to all cases. These are covered below:

4.2.3.1 Fog Spray Capacity

Fog spray capacity has been calculated at two limiting conditions. In the first condition, giving a minimum fog spray efficiency it is assumed that the building is filled with pure air at the time the steam vents are resealed. When the air space pressure increases to 5 psi, the building design value, condensing capacity is calculated to be 90 lbs/second.

In the second case, it was assumed that the building space contained pure steam when the steam vents resealed. In this case when the pressure increases to 5 psi, the fog capacity will be 170 lbs/second or more.

In both cases, water supplied to the fog nozzles was assumed to be 9200 gal/minute, the design value. Changes in this supply would affect condensing rates.

4.2.3.2 Building Steam Storage Capacities

If the steam vents reseal before the steam release rate has decayed to the fog spray capacity, the issue of whether or not building overpressurization occurs depends on part on the storage capacity of the building. If the capacity is high enough the excess steam may be stored at less than a 5 psi pressure until the steam release rate does decay to within the fog spray capacity. The building storage capacity after vent resealing is calculated to be at least 30,000 pounds of steam.

4.2.3.3 Primary Pump Operation

In the following analyses, continued operation of the primary pumps circulating primary coolant is assumed. It is assumed that at the time of the incident the pumps start a coastdown from full speed to one-quarter speed (corresponding to pony motor operation) and continues until

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cavitation begins.

Failure of the pumps would increase the severity of most accidents. An exception would be the case of the largest breaks where no reactor cooling is assumed during the very rapid depressurization.

Continued pump operation is not assured. However, reliability of the pumps has been maximized through system design by providing for both steam and electrical drive. Pump operation past about five to ten minutes is, furthermore, not critical to the results in many cases.

4.2.3.4 Operation of Steam Generators

Operation of steam generators is also assumed. This assumption seems realistic. The inventory of secondary coolant in the steam generator would provide some cooling during the transient. Operation of either BPA or local powered condensate feed pumps would permit a continued full capacity functioning of the system.

If the steam generators do not continue to provide cooling, the consequences of some incident would be worsened. These cases have not been examined in detail.

4.2.3.5 Reactor Scram

It is assumed that the reactor scrams at the time the incident starts. This assumption is in accord with what is by far the most probable course. Should the reactor not scram the incident severity would increase in some cases. However, boil out of coolant would cause the reactor to be driven to sub-criticality and in some cases (large failures) the course of events might not be importantly different.

4.2.3.6 Heat Transfer During Transient

All calculations made have assumed that the heat transferred from the fuel to the coolant flowing through the reactor is

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exactly equal to the fission product decay heat generated in the fuel. This assumption may overstate the heat load from this source and, hence, tend to overstate effluent enthalpies and flow resistance out breaks. Hence, this assumption is in the direction of overestimating the severity of consequences.

Two assumptions have been made regarding heat transfer from the graphite moderator to the coolant. The first assumption made was that there would be no heat transfer from the graphite. This plainly underestimates the heat load to the coolant from that source, and, probably late in the transient when fuel powers have decayed to quite low values, leads to or under estimation of the total heat load. It was, therefore, decided to redo the analysis assuming heat transfer from the graphite continued during the transient at about normal rates. As expected this latter assumption leads to longer release times. It is believed that actual cases would probably lie somewhere in between the limits defined by the two assumptions, i.e., the second assumption probably overstates the heat load in most cases. In the following discussions, the case of no heat transfer from the graphite is first discussed and then the affects of heat transfer are discussed.

4.2.3.7 Operation of Injection Pumps

It is assumed that the primary coolant injection pumps continue to function to supply 1000 lbs/second of 150 F water. The main effects of this assumptions are: (1) this water represents a heat sink, and (2) it prolongs the period until cavitation starts (the water is injected just upstream of the pumps). These effects are important in those cases where reactor cooling is important during the depressurization transient.

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4.2.4 Detailed Calculations

The detailed calculations are covered below. The assumptions covered before are not repeated in all these cases.

4.2.4.1 Breaks In The Inlet Header

Five different sized failures in the inlet coolant header (position "e" in Figure E-1) were studied. The failures assumed ranged from 20 square inches in cross sectional area to a complete circumferential break.

Steam flows were calculated using the assumptions discussed in Section 4.2.3 and the cumulative steam releases are shown in Figure E-8.

It appears that with the assumptions made the whole range of failure considered may be accommodated with a clean separation of non-contaminated and contaminated vapors. In the three largest breaks considered, the venting process is complete in less than 300 seconds, the minimum time for melting to start for fuel at normal peak powers, and with no cooling after start of incident. Actually, some cooling will occur in all these cases as the water trapped downstream of the check valves on the inlet supply lines flow through the reactor to the break. In the case of the smaller breaks, the steam release rate drops rapidly enough to permit resealing without subsequent overpressurization before melting would start even with no cooling.

The assumption of heat transfer from the graphite extends the time for total depressurization and also (predictably) increases the total amount of steam released. However, with the three largest breaks, resealing can still be accomplished before fuel melting starts even though

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the system may not be completely vented. With the two smaller breaks the amount of steam released is increased but, while the margin is reduced, the same conclusions still hold as stated for the case of no heat transfer from graphite.

Hence, it appears that the entire spectrum of failure sizes in the inlet coolant header can be accommodated with a clean separation within the assumptions already stated. In particular, no credit is taken for dump valve operation

4.2.4.2 Failures In An Inlet Riser - Upstream of Check Valves

Six failure sizes were considered in the study of the consequences of failure in an inlet supply riser upstream of the check valve. The failures at this point are of particular interest because flow to the two columns of tubes supplied by the stricken riser is particularly affected. With failures larger than a critical size, flow past the break is not obtained even with continued functioning of the primary pumps. Further, back flow cooling is not obtained because of the check valve. Typically, then, there is no - or very little - flow through the tube until the system depressurizes below the saturation pressure corresponding to the bulk inlet temperature. At this point, flow may be resumed by coolant in the "inlet leg" boiling and flowing through the reactor.

Cumulative steam release is shown in Figure E-9. The figure shows that the two largest breaks, complete circumferential and 119 square inches, give complete depressurization quite rapidly. With the smaller breaks covered, 42 square inches or smaller, the depressurization is extended in time but resealing is possible before melting would be expected in even the stricken two columns of tubes. However, the picture is not so clear cut when the heat transfer from the graphite is assumed. For

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instance, in the case of the 49 square inch break, the steam release rate at 460 seconds is still about 110 lbs/second; resealing at 300 seconds might cause overpressurization. Hence, prevention of melting until after resealing would be dependent on cooling during the transient. Also, the time of transport of fission products from tube outlets to the break in the inlet piping assumes importance. This transport time should be about one minute and this may represent a critical margin. However, flow will initially be lost from the two columns of "stricken" tubes and whether or not cooling will occur later on will depend on factors (steam sweep-out parameters) not yet established. Hence, in certain size failures in the order of 50 square inches occurring in the riser, a clean separation may or may not occur depending on heat transfer rates from the graphite and on cooling characteristics (provided multiple failures of dump valves occur).

On the other hand, only a small fraction of the core will be affected, flow could be routed through the filters when release of fission products is started and, after perhaps two or three minutes of filtered release, the building could probably be sealed off. But instead the primary goal, ability to reseal the building without an overpressurization cannot be assumed.

Even with the assumption of heat transfer from the graphite the largest and smallest breaks can still be accommodated. In the former case depressurization is still completed rapidly enough; in the later case, the release rate decays in soon enough time, to fog spray capacity.

4.2.4.3 Failure In An Inlet Riser Downstream of Check Valves

This particular case was not specifically covered by calculations. However, with the failure downstream of the check valves,

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back flow cooling through tubes supplied by the stricken line might provide important reductions in accident severity.

4.2.4.4 Breaks In An Outlet Downcomer - Upstream From Check Valves

Three rather small sized breaks were studied at a point on an outlet downcomer upstream of the check valves. The failure size ranged from about 7 square inches to 25 square inches. In these failures, the coolant flow to the break has not previously passed through the heat exchanger. Hence, the enthalpy of water passing to and through the breaks is higher than in the two cases previously considered.

Cumulative steam release is shown in Figure E-10. Steam release rates decay to the minimum fog spray capacity after 325 seconds, 275 seconds and 75 seconds for the three failures in reverse order of size. In taking into account steam storage capacity, vent resealing can be accomplished after about 150 seconds for the largest failure and immediately for the smallest failure without overpressurization. When heat transfer at the assumed rate is taken into account, the transient is prolonged and in the case of the largest break vent resealing without subsequent overpressurization would not be possible prior to about 500 seconds. However, it is calculated that single phase reactor cooling will continue during the entire transient so the delay in minimum resealing time - if it did occur - would not be of practical importance.

In summary, failure size up to 25 square inches in the outlet downcomer upstream of the check valve can be accommodated with a clean separation with the assumptions made which include complete malfunction of the dump system.

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4.2.4.5 Break In An Outlet Downcomer - Downstream of Check Valves

Five sizes of failures in the outlet downcomer downstream of the check valves were studied. The failures ranged from 14 square inches up to 175 square inches on a complete circumferential failure.

Cumulative steam releases are shown in Figure E-11. As shown on the figure, the entire range of failure can be accommodated with a clean separation even with malfunction of dump valve. Steam vents can be resealed before 180 seconds, the minimum time for melting at the design hot channel specific power.

When graphite heat transfer at the assumed rate is considered, the calculated transient is prolonged. However, it still appears that resealing by 180 seconds will be possible if required. The reactor cooling during the transient will be two phased. This is evaluated as meaning that the certainty of cooling at all tubes is reduced below that of the previous case where single phased cooling was predicted. If vapor boiling should occur there might or might not be sufficient pressure available to sweep steam and re-establish flow in the highest powered tubes. Operation of dump valves should alleviate the situation.

4.2.4.6 Failure In Steam Generator Bypass Line

A complete failure in the line bypassing the two steam generators was studied. Cumulative steam release is shown in Figure E-12. With the complete break, vent resealing can be accomplished before 180 seconds if necessary and the assumed rate of graphite heating has no effect on the results.

4.2.4.7 Failure in The Outlet Header

Four failure sizes in the 26 inch diameter outlet header were studied. Sizes ranged from 418 square inches (complete circum-

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ferential failure) down to 14 square inches.

Cumulative steam releases are shown in Figure E-13.

With the assumption made, resealing in all cases can be effected in 180 seconds or less. If the graphite heat transfer assumption is factored in, the transient is prolonged, in some cases resealing is prolonged to about 300 seconds; however, it is also calculated that single phase reactor cooling will continue throughout these transients probably assuming a significant delay in melting.

5. CONFINEMENT SYSTEM INSTRUMENTATION AND CONTROL

5.1 General

The confinement system is composed of a number of different components which may be called upon to serve important functions during a reactor incident. The control systems for these components and particularly their reliability and adequacy are, therefore, key factors in over-all system quality.

5.2 Mode of Control

Certain control functions associated with the confinement system will be automatic; others may be automatic or manual depending on final decisions regarding system operation; some will certainly be manual.

The initiation of any incident involving a coolant line rupture will probably occur unexpectedly and events will occur rapidly. Parts of the confinement system dealing with these initial events, e. g., ventilation duct closures, fog spray actuation, and steam vent opening will occur automatically. Events occurring later in the transient may be more complex. For example, later in the transient it will be desirable to reseal the steam vents and, perhaps, simultaneously to reopen ventilation passages to the filters. It is not clear at this time whether this should be done auto-

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matically or if manual operation would be preferable.

Finally, the last stage of an incident involves re-entry, decontamination, and return of the system to normal. These steps will be planned in detail according to facts at hand. These steps will be by "manual" control.

5.3 Building Pressure Instrumentation

A primary coolant line failure will be detected by the monitoring of air pressure in Zone I. This building pressure will be used to control much of the confinement components.

Normally, the ventilation system of the 105-N and 109-N Buildings are separate and the air spaces are physically isolated, one from the other. This isolation will continue during an incident unless the pressure in one or the other building increases to about 1.8 psig when interconnecting vents will blow open (See Section 7 of the first volume). Therefore, two different sets of building pressure instrumentation have been provided, one for guiding the control of each building.

5.3.1 109-N Building Zone Pressure Instrumentation

Zone I air pressure in the 109-N Building is sensed by 32 pressure switches. The switches are divided into four sets of eight each. Each set of the four sets performs a different function. One switch in each of the four sets is located in each of the five steam generator cells, one in the auxiliary system cell, one in the pipe gallery (the corridor adjacent to the 105-N and 109-N common wall) and one in the pressurizer penthouse.

Set No. 1 - The first set of eight switches will all ordinarily be held closed but will open should the Zone I air pressure become higher than a negative 0.1 inches of water as compared to the pressure in Zone III of the 109-N Building. When the circuit is broken, an alarm

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will sound in both the 105-N and 109-N Buildings. The switches will be set to reclose should pressure decrease again to about a negative 0.3 inches of water gage.

Set No. 2 - The second set of switches are normally held closed but will open should the pressure increase to about a positive two inches of water. When any two of the eight switches open, a relay will trip closing one set of contacts and opening a second set. When the one set of contacts is opened, a second relay is de-energized, its contacts are opened and duplicate solenoid valves on the fog spray valve are de-energized causing the valve to open. The duplicate solenoids obtain power from separate d.c. power supplies so a single power failure will not actuate the fog spray.

A third "latching relay" is energized when the set of contacts on the reversing relay close. The latching relay stays in the energized position unless a special reset coil is energized. The purpose of the latching relay is to prevent the solenoid valves from being re-energized should the pressure decay and the pressure switches reclose (as they will at some pressure still above atmospheric).

The latching relay's reset coil can be energized by a manual reset button. This action will cause the fog sprays to turn off should 7 or 8 of the pressure switches be closed.

Set No. 3 - The third set of pressure switches will ordinarily be held closed but will be set to open should the pressure decay to about a negative two inches of water gage. Should two or more of these switches open, a reversing relay will trip, and the end results will be that the reset coil of the latching relay (discussed above) will be energized shutting off the fog sprays. Should the pressure increase then to about a negative 1.4 inches of water gage, the pressure switches will reclose and the entire system will

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be returned to normal - ready to be re-actuated should the pressure again increase to two inches of water gage.

There are six independent fog spray control valves. Any one individual valve or all valves may be bypassed so that they will not respond to a pressure trip. This will permit functional testing of the circuitry without incurring an actual spray. The bypass switches will be controlled in the 105-N Control Room and lights in the control room will indicate when any valve is bypassed.

Set No. 4 - The fourth set of switches will be used to control the remaining automatic pressure actuated confinement functions.

The eight switches in this set are usually open. When two or more close (at a positive two inches of water pressure gage) a relay is reversed causing:

- (1) A second relay to be de-energized sounding an annunciation and actuating interlocks on four doors into Zone I (preventing accidental entry).
- (2) A third relay to be de-energized in turn causing:
 - a. Actuation of the door centerline for all other doors into 109-N Zone I.
 - b. Closure of ventilation system valves for 109-N Zone I and II. (By Zone II is meant the Burns and Roe definition, see Section 7 of Volume 1).
 - c. Stopping of exhaust fans from Zones I and II and closing the Zone III inlet damper. When the exhaust fans are turned off the supply fans also automatically stop. When the Zone III damper closes, the Zone III system operates on total recirculation of air.

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The circuitry is so arranged that a total d.c. power failure (two independent sources) will have no effect on the ventilation closure positions. Details on bypassing features have not been firmly established at this time.

5.3.2 105-N Building Zone Pressure Instrumentation

There are two sets of four switches each in the 105-N Building Zone I air space. One set of switches will be set to open should the pressure reach about 2 inches of water gage and reclose when the pressure drops to about 1.7 inches of water gage. The other set of switches will be set to close should the pressure decay to a negative 2 inches of water gage, and reopen should the pressure increase to a negative 1.7 inches of water gage. The functions and logic are discussed below:

Set No. 1 - Each switch is wired in series with two parallel coils. When the switch opens because of high pressure the coils are de-energized. Hence, here power failure has the same effect as a high pressure trip. The switches use different power supplies, however, so a single power failure will not cause a confinement trip.

The relay contacts are arranged in several different circuits. Trips of two or more of any four will cause to occur:

- (1) The fog spray system in the 105-N Building will turn on. Further, the fog spray will stay on after the pressure switches reset unless the fog spray circuit itself is reset. Total d.c. power failure will cause actuation of the fog spray system.
- (2) The confinement valves on supply and exhaust ducts will close. Total power failure will cause closure. However, two separate power supplies are used and if

either power supply is in service a confinement trip would cause valve closure.

- (3) The Zone V ventilation system (the 105-N Control Room) is transferred to total air recirculation.
- (4) Inlet and exhaust fans for Zone I turn off.
- (5) Zone I doors interlock tripped preventing entry into Zone I.
- (6) An pneumatic instrument line transmitting Zone I air pressure to two differential pressure transmitters is valves off (preventing instrument damage and, perhaps, a break in confinement).

Set No. 2 - The second set of gages will automatically turn the fog spray valve off should the pressure decay to about 3.2 inches of water gage.

5.4 Primary Coolant Pressure

One major indicator of the approaching end of significant steam release is the primary coolant system pressure. Therefore, a coincidence system of four pressure switches sensing primary coolant system pressure has been provided.

The ultimate use of this instrumentation has not been finally decided - it likely will be a factor in the control of steam vent resealing.

5.5 Radiation Detectors

Contamination levels in air or air steam mixtures may provide an important part of the basis for operation of the confinement system. Thus, various radiation detectors have been provided.

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5.5.1 109-N Building

Nine radiation monitors covering the range 0.01 mr/hr to ten rems/hour are provided in the 109-N Building with the signal transmitted to the 105-N Control Room. In addition, air samples in the Zone I and II air exhaust in 109-N are taken, and filtered to show contamination levels in 109-N air exhaust.

5.5.2 105-N Building

Twenty-two radiation monitors similar to those described for the 109-N Building are provided in the 105-N Building. In addition, the exhaust air leaving Zone I and II filters is sampled, and passed through a particulate filter and a charcoal halogen filter in series. Count rates are transmitted to the 105-N Control Room.

6. DISPOSAL OF CONTAMINATED WATER

In case of an accident involving line failure, parts of Zone I will be flooded with water from the primary loop, from the emergency water system if it operates, and from the fog spray system. If the emergency coolant system does not function and melting occurs, the water may be very severely contaminated with fission products and its disposal may become an important problem.

The two buildings, 105-N and 109-N, are joined for steam release by cross vents. However, these vents are near the top of the buildings and the buildings are not joined in such a way that water flow from one to the other is possible. The 105-N Building Zone I is smaller and presents more of a problem in this contaminated water storage and disposal standpoint - hence it is discussed in brief detail here.

The 105-N Building Zone I space will hold 33,000 cubic feet of 230 F water plus an additional 89,000 cubic feet of cooled water.

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The primary coolant system volume is 32,300 cubic feet. If the entire contents were dumped into the 105-N Zone I, there would be a net remaining storage of about 100,000 cubic feet.

The design flow of the 105-N fog spray system is 6,700 gal/min. Hence, this storage represents about 110 minutes of full capacity operation. Actually, the fog spray is expected to operate at full capacity for no longer than about ten minutes and this will probably cycle on and off to control pressure between plus two and minus two inches of water gage and to rain out fission products. The 105-N storage, therefore, apparently will accommodate the water resulting from most accidents of the type where the water contamination is serious.

Water from the 105-N Zone I can be drained from the building to the 105-N sump and routed to one of two places. It can be returned directly to the river by a gravity drain if contamination levels are consistent with this disposal mode or it can be pumped to a crib. The pumps in question are electrically powered and one of the two power sources would be required for this to be done in a normal way.

It is possible that total power loss might occur as a cause or result of the incident. In this case storage of the water in the building will be required until an emergency source of pumping capacity, e.g., portable diesel-driven units, are provided. In this case it might be necessary eventually to regulate fog spray operation on the basis of building storage requirements instead of on airborne fission product management. This course would probably not be highly significant.

Water pumped to the crib will filter back to the river. The hold-up time for fission products will vary with the chemical specie, e.g., the concentration of iodine in the river, following a cribbing, will reach a

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maximum in about 30 days. This represents a significant delay.

Ignoring probabilities, one can conceive of a situation where there is a significant fission product release, the building is flooded, drainage cannot be provided, and it is imprudent to stop supply water to the break. In such a case, the gravity to drain to the river would have to be used or the building would overflow and eventually the water would reach a depth so great that the building would be structurally damaged permitting local drainage.

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FIGURE E-1

SIMPLIFIED DIAGRAM PRIMARY COOLANT SYSTEM

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FIGURE E-2

STEADY STATE BUILDING VENTING CAPACITY

FIGURE E-3

TRANSIENT FOLLOWING 6" DIAMETER BREAK OF OUTLET DOWNCOMER

FIGURE E-4

TRANSIENT FOLLOWING BREAK OF DOWNCOMER DOWNSTREAM OF CHECK VALVE

FIGURE E-5

TRANSIENT FOLLOWING FAILURE OF INLET HEADER

FIGURE E-6

TRANSIENT FOLLOWING FAILURE OF INLET RISER

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FIGURE E-7

TRANSIENT FOLLOWING FAILURE OF STEAM GENERATOR BYPASS LINE

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FIGURE E-8
STUDIES OF INLET HEADER FAILURES

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FIGURE E-9

STUDIES OF FAILURES OF INLET RISER - UPSTREAM OF CHECK VALVE

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FIGURE E-10

STUDIES OF FAILURE IN OUTLET DOWNCOMER - UPSTREAM OF CHECK VALVE

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FIGURE E-11

STUDIES OF FAILURES IN OUTLET DOWNCOMER DOWNSTREAM OF CHECK VALVE

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STUDY OF FAILURE OF STEAM GENERATOR BYPASS LINE

FIGURE E-1

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FIGURE E-13

STUDIES OF FAILURES IN OUTLET HEADER

APPENDIX FCONSEQUENCES OF FISSION PRODUCT RELEASE1. GENERAL

Section 9 of Volume 1 of the Hazards Review contains a discussion of the consequences of a partial and total meltdown of the N Reactor fuel core. Estimates of areas were presented within which various levels of exposure and contamination could be expected. The purpose of this Appendix is to present material supporting the calculations and to present results of some additional calculations performed since Volume 1 was published.

2. DISPERSION, DEPLETION AND DISPOSITION OF AIR-BORNE CONTAMINANTS

Volume 1 presented results obtained from calculations made assuming different atmospheric stabilities, wind velocities and release heights. Two mathematical models were used in determining the dispersion of fission products. Sutton's equations (1) were used for neutral and unstable atmospheric conditions (no temperature inversion), and modified diffusion equations (2) were used for stable atmospheric conditions (temperature inversions present).

The most severe case from the standpoint of contamination of the environs is the assumption of a stable atmosphere, i.e., the maximum contamination outside the plant would be expected if there was a temperature inversion.

Table F-1 compares the time integrated concentrations at ground level under several atmospheric conditions.

TABLE F-1

COMPARISON OF TIME-INTEGRATED CONCENTRATION
WITH ATMOSPHERIC STABILITYAverage Wind Speed 5 m/s
Height of Release 70 m

	Unstable	Neutral	Stable	
			Moderately	Highly
Distance to Maximum Concentration at Ground Level (70 m Stack)	500 m	1,600 m	28,000 m	360,000 m
Concentration* at:				
Point of Maxima	9.5×10^{-4}	9.5×10^{-4}	9.0×10^{-7}	1.1×10^{-7}
10,000 meters	1.2×10^{-7}	8.6×10^{-7}	4.5×10^{-7}	$< 10^{-11}$
20,000 meters	3.5×10^{-8}	2.7×10^{-7}	8.4×10^{-7}	$< 10^{-11}$
50,000 meters	7.0×10^{-9}	4.0×10^{-8}	8.0×10^{-8}	8.8×10^{-10}

* In units of $\mu\text{c/cc}$ per c/sec emitted.2.1 Atmospheric Dispersion

In stable atmospheres (inversion conditions), the time integrated concentration (E) at ground level can be expressed by Equation (1):

$$E = \frac{Q'}{\pi \sigma_y \sigma_z \bar{u}_h} \exp \left[-\frac{1}{2} \left(\frac{y^2}{\sigma_y^2} + \frac{h^2}{\sigma_z^2} \right) \right] \quad (1)$$

For neutral and unstable atmospheres, the time integrated concentration (E) at ground level can be expressed by Sutton's equation, as follows:

$$E = \frac{2Q'}{c_y c_z \bar{u}_h x^{2-n}} \exp \left[-\frac{1}{x^{2-n}} \left(\frac{y^2}{c_y^2} + \frac{h^2}{c_z^2} \right) \right] \quad (2)$$

For both equations, the symbols are defined as:

E = time integrated concentration at ground level in $\frac{\mu\text{c}}{\text{cc} \cdot \text{sec}}$.

Q' = apparent quantity emitted in curies as viewed by an observer
at a distance x .

$\sigma_y^2 = a x^m$ units of square meters.

$a = 0.016$

$m = 1.6$

x = downwind distance in meters.

$\sigma_z^2 = a [1 - \exp(-k^2 t^2)] + bt.$

a , b , and c are functions of the degree of stability. Their values for two conditions are shown in Table F-2.

C_y , C_z and n are Sutton's parameters, and the values used in this analysis are shown in Table F-3.

t = time of cloud travel to point x .

U_h = average wind speed at height of emission in meters per second.

y = cross-wind distance as measured from the centerline in meters.

h = height of release above ground level in meters.

TABLE F-2

NUMERICAL VALUES OF STABLE
ATMOSPHERIC DISPERSION PARAMETERS

	Atmospheric Condition	
	Degree of Stability	
	<u>Moderate</u>	<u>High</u>
a	97 m^2	34 m^2
b	0.33 $m^2 \text{ sec}^{-1}$	0.025 $m^2 \text{ sec}^{-1}$
k^2	$2.5 \times 10^{-4} \text{ sec}^{-2}$	$8.8 \times 10^{-4} \text{ sec}^{-2}$

TABLE F-3

NUMERICAL VALUES OF ATMOSPHERIC DISPERSION PARAMETERS
FOR NEUTRAL AND UNSTABLE ATMOSPHERES

	<u>Release Level</u>	<u>Wind Speed</u>	<u>Unstable</u>	<u>Neutral</u>
C_y	Ground	1 m/s	0.35	0.21
		5 m/s	0.30	0.15
		10 m/s	0.28	0.14
	Elevated	1 m/s	0.30	0.15
		5 m/s	0.26	0.12
		10 m/s	0.24	0.11
C_z	Ground	1 m/s	0.35	0.17
		5 m/s	0.30	0.14
		10 m/s	0.28	0.13
	Elevated	1 m/s	0.30	0.15
		5 m/s	0.26	0.12
		10 m/s	0.24	0.11
n	---	---	0.20	0.25

2.2 Cloud Depletion

Allowance was made for depletion by ground deposition of radionuclides from the cloud using Equation (3) for stable atmospheres and Equation (4) for neutral and unstable atmospheres.

$$\frac{Q'}{Q} = \exp \left\{ -\sqrt{\frac{2}{\pi}} \left(\frac{V_A}{u_0} \right) \left(\frac{\bar{u}_0}{\bar{u}_n} \right) \bar{u}_n \int_0^x \frac{\exp \left(-\frac{h^2}{2\sigma_z^2} \right)}{\sigma_z} dx \right\} \quad (3)$$

$$\frac{Q'}{Q} = \exp \left\{ - \frac{2}{n \bar{u} C_z \sqrt{\pi}} \left(\frac{V_{\Delta}}{\bar{u}_0} \right) \left(\frac{\bar{u}_0}{\bar{u}_h} \right) \int_0^x x^{\frac{n-2}{2}} \exp \left[-\frac{h^2}{2x^2} \right] dx \right\} \quad (4)$$

Where the additional symbols mean:

Q = the quantity emitted (curies).

V_{Δ} = the deposition velocity in meters per second.

$\left(\frac{V_{\Delta}}{\bar{u}_0} \right)$ = the deposition coefficient - dimensionless. Dependent upon particle size and chemical nature of radionuclides.

$\left(\frac{\bar{u}_0}{\bar{u}_h} \right)$ = the ratio of average wind speed at ground level ($\bar{u}_0$) to the average wind speed at level of emission ($\bar{u}_h$).

Both the deposition coefficient and wind speed ratio depend upon the degree of atmospheric stability⁽²⁾ as shown in Table F-4. The integrals have been evaluated numerically, and the wind speed ratio has been evaluated assuming the height of the elevated release to be 200 feet.

TABLE F-4

GROUND DEPOSITION PARAMETERS

<u>Atmospheric Condition</u>	<u>Deposition Coefficient</u>		<u>Wind Speed Ratio</u> $\frac{\bar{u}_0}{\bar{u}_h}$
	$\left(\frac{V_{\Delta}}{\bar{u}_0} \right)$ <u>Particulates</u>	<u>Halogens</u>	
Highly Stable	1.5×10^{-4}	2.4×10^{-3}	0.17
Moderately Stable	2.2×10^{-4}	3.4×10^{-3}	0.35
Neutral	3.0×10^{-4}	4.6×10^{-3}	0.50
Unstable	6.0×10^{-4}	8.0×10^{-3}	0.70

From Equations (1) and (3), it can be shown that the air concentration at ground level per curie per second emitted, corrected for cloud depletion, for stable atmospheres is:

$$\left(\frac{E}{Q}\right) = \frac{A}{\pi \sigma_y \sigma_z \bar{u}_h} \exp \left\{ -\frac{1}{2} \left[\frac{y^2}{\sigma_y^2} + \frac{z^2}{\sigma_z^2} \right] \right\} \quad (5)$$

Where $A = \frac{Q'}{Q}$ as defined by Equation (3).

Similarly, the air concentration for neutral and unstable atmospheres can be expressed by:

$$\left(\frac{E}{Q}\right) = \frac{2B}{\pi C_y C_z \bar{u}_h x^{2-n}} \exp \left[-\frac{1}{x^{2-n}} \left(\frac{y^2}{C_y^2} + \frac{h^2}{C_z^2} \right) \right] \quad (6)$$

Where $B = \frac{Q'}{Q}$ as defined by Equation (4).

2.3 Ground Deposition

Deposition of radionuclides on the ground is estimated by:

$$d = Q \left(\frac{E}{Q} \right) \left(\frac{V}{\bar{u}_0} \right) \left(\frac{\bar{u}_0}{\bar{u}_h} \right) \bar{u}_h \times 10^6 \frac{\mu c}{m^2} \quad (7)$$

Where:

d = ground deposition in $\mu c/m^2$.

All other symbols are as previously defined.

The factor 10^6 results from conversion of units from curies to μc .

Assuming vegetation coverage (ρ) for pastures and vegetable gardens of 1,000 g/m², the ground deposition in units of $\mu c/g$ of vegetation is:

$$\frac{d}{\rho} = Q \left(\frac{E}{Q} \right) \left(\frac{V}{\bar{u}_0} \right) \left(\frac{\bar{u}_0}{\bar{u}_h} \right) \times 10^3 \frac{\mu c}{g/m} \quad (7a)$$

Equations (7) and (7a) were used to estimate ground and food contamination, respectively.

2.4 Areas of Contamination

The distances from the source to various time-integrated concentrations or ground deposition predicted by Equations (5) through (7) depend upon a constant wind velocity. This condition seldom prevails for any length of time, and it certainly varies with distance as the cloud moves downwind. On the other hand, areas enclosed by contours of equal concentration or contamination are far less dependent upon the constant velocity assumption. As a matter of fact, the areas enclosed by these isopleths have proven to be much more predictable than distances in both Project Prairie Grass⁽³⁾ and Hanford's Green Glow experiments.⁽⁴⁾

Isopleth areas cannot be easily calculated rigorously. However, they can be very closely approximated. Equation (5) can be written as:

$$\left(\frac{E}{Q}\right) = \left(\frac{E^*}{Q}\right) \exp \left[-\frac{y^2}{2\sigma_y^2} \right] \quad (8)$$

Where $\frac{E^*}{Q}$ is the centerline ($y = 0$) value of Equation (5).

Solving Equation (8) for y yields:

$$y = \pm 2 \sigma_y \left[\ln \left\{ \frac{\left(\frac{E}{Q}\right)^*}{\left(\frac{E}{Q}\right)} \right\} \right]^{\frac{1}{2}} \quad (9)$$

Now let: Area = $\Delta x (2 y^*)$ approximate (10)

Where:

$x_2 - x_1 = \Delta x$ = the length of the area (in the direction of wind).

x_1 = minimum distance from the source to the isopleth.

x_2 = maximum distance from the source to the isopleth.

y^* = the normal distance from the centerline at the point $\frac{\Delta x}{2}$ to the edge of the zone.

Evaluating Equation (9) at $\frac{\Delta x}{2} = x^*$ and substituting the result in Equation (10) yields:

$$\text{Area} = 0.145 (x^*)^{1.8} \left[2.0 \ln \left\{ \frac{\left(\frac{E}{Q} \right)^*}{\left(\frac{E}{Q} \right)^*} \right\} \right]^{1/2} \quad (11a)$$

Where:

$$\frac{E}{Q}^* = \frac{E}{Q}^* \text{ at } x^*$$

Equation (11a) has been compared with numerically integrated areas enclosed by Equation (9). These comparisons were made for several values of exposure and for five different values for atmospheric stability, three different release heights, three wind speeds, and three values of deposition coefficients. As a result, it was found that Equation (11a) overestimates the enclosed area consistently by a factor of $\sim 1/3$. Therefore, the estimated area is:

$$\text{Area} = 0.11 (x^*)^{1.8} \left[2.0 \ln \left\{ \frac{\left(\frac{E}{Q} \right)^*}{\left(\frac{E}{Q} \right)^*} \right\} \right]^{1/2} \quad (11)$$

As an example, curves of Equation (11) for a moderately stable atmosphere are plotted in Figures F-1 through F-6 for 1, 5, and 10 m/s, respectively. Each figure shows areas for three different deposition coefficients.

3. RELATIONSHIP BETWEEN DOSE AND CONTAMINATION

Once the integrated exposure is determined, the next step is to convert the exposure to a radiation dose. The results shown in Section 9 of Volume 1 are given for both direct exposure to the cloud, i.e., dose received from cloud passage, and indirect exposure, i.e., dose from contaminants deposited on the ground in the cloud's wake.

The equations defining the appropriate meteorological parameters in terms of dose for each zone are summarized in Table F-5. A brief development of each equation is given in the following paragraphs.

TABLE F-5

RELATIONSHIP OF DOSE AND METEOROLOGICAL EXPOSURE BY ZONES

<u>Zone</u>	<u>Mode of Exposure</u>	<u>Organ of Interest</u>	<u>Derived From Equation: (Number in Text)</u>	<u>$\frac{E}{Q}$</u>
A, B, C	External and inhalation during cloud passage	Total Body	21	$0.195 + 230 (f_{ak} TB)$
D, E	Inhalation during cloud passage	Thyroid	22	$4.35 \times 10^{-3} \frac{D_{THY}}{Q (f_{ak} THY)}$
F, G	External from residual ground contamination	Total Body	23	$2.25 \times 10^{-4} \frac{D_{TB}}{H Q} *$
H, I	Internal from ingestion of milk - Cs-137	Total Body	25	$9.62 \times 10^{-7} \frac{D_{TB}}{H Q (Cs-137)} *$
J, K	Internal from-ingestion of milk - Sr-90	Bone	26	$2.99 \times 10^{-10} \frac{D_B}{H Q (Sr-90)} *$
L	Internal from ingestion of milk - I-131	2 g Thyroid	27a	$1.12 \times 10^{-8} \frac{D_{thy}}{H Q (I-131)} *$
M	Internal from ingestion of milk - I-131	20 g Thyroid	27b	$1.12 \times 10^{-7} \frac{D_{THY}}{H Q (I-131)} *$

$$* H = \frac{V}{u_o} \frac{\bar{u}_o}{\bar{u}_h} \bar{u}_h$$

AIM

Association for Information and Image Management

1100 Wayne Avenue, Suite 1100
Silver Spring, Maryland 20910
301/587-8202

Centimeter

Inches

MANUFACTURED TO AIM STANDARDS
BY APPLIED IMAGE, INC.

5

of

5

Figure F-1

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Figure F-2

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Figure F-3

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Figure F-4

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Figure F-5

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Figure F-6

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3.1 Dose to Organ of Interest as a Function of Intake of a Radioactively Decaying Source

The dose rate (D_r) to an organ as a result of continued breathing of contaminated air can be shown to be:

$$D_r = k' q \frac{\text{REM}}{\text{SEC}}$$

$$= k' \left\{ \frac{f_a P \exp(-\lambda t)}{\lambda_e - \lambda} \left[1.0 - \exp\left\{-(\lambda_e - \lambda) t\right\} \right] \right\} \quad (12)$$

Where:

$$k' = 5.92 \times 10^{-4} \frac{\epsilon}{m} \text{ rem}/\mu\text{c-sec.}$$

$$\epsilon = \sum_i E_i F_i (\text{RBE})_i n_i \text{ in } \frac{\text{Mev-rem}}{\text{dis-rad}} \text{ as defined in reference (5).}$$

m = mass organ of interest in grams.

q = quantity of radionuclide in organ of interest - μc .

f_a = fraction of radionuclide inhaled which reaches organ.

P = initial rate of intake $\mu\text{c/sec}$.

λ = radioactive decay constant in sec^{-1} .

λ_e = effective decay constant in organ of interest in sec^{-1} .

t = time since initial intake in seconds.

If short-lived radionuclides can be ignored, then:

$$k' = 51.2 \frac{\epsilon}{m} \frac{\text{rem}}{\mu\text{c-day}}$$

and all other units expressed as functions of days instead of seconds.

The dose (D) can be estimated from Equation (12) as follows:

$$D = \int_0^t D_r d\tau = f_a k' P \left\{ \frac{\lambda_e [1 - \exp(-\lambda t)] - \lambda [1 - \exp(-\lambda_e t)]}{\lambda (\lambda_e - \lambda)} \right\} \quad (13)$$

IN REMS

Where:

$$k = \frac{k'}{\lambda_e} = 51.2 \frac{\epsilon}{\lambda_e n} \quad \frac{\text{rem}}{\mu\text{c}}$$

When $\lambda_e = \lambda$, Equation (3) becomes indeterminate. However, it can be evaluated by conventional mathematical methods. Such an evaluation results in:

$$D = f_a A P \frac{\exp(-\lambda t)}{\lambda} \left[\exp(\lambda t) - (\lambda t + 1) \right] \text{ (REMS)} \quad (13a)$$

Equations (13) and (13a) can be applied to the ingestion of contamination, if the time interval of integration is at least 10 days; in which case, substitution of f_w (the fractional uptake from ingestion of water) for f_a in either equation, will result in an appropriate equation for estimating organ dose from ingestion.

So far, it has been assumed tacitly that the intake continues indefinitely. If the intake is discontinued at some time, say t_1 , then:

$$D = \int_0^{t_1} D_T d\tau + \int_{t_1}^T D_T t_1 \exp[-\lambda(T - t_1)] d\tau$$

or,

$$D = f_a A P \left\{ \frac{\lambda_e A - \lambda [B + CF]}{\lambda (\lambda_e - \lambda)} \right\} \text{ (REMS)} \quad (14)$$

Where:

$$A = 1 - \exp(-\lambda t_1).$$

$$B = 1 - \exp(-\lambda_e t_1).$$

$$C = 1 - \exp[-\lambda_e(t - t_1)].$$

$$F = \exp(-\lambda t_1) [1 - \exp(\lambda_e - \lambda)t_1].$$

and,

$f = f_a$ for inhalation.

$F = f_w$ for ingestion.

When $\lambda_g = \lambda$,

$$D = \frac{FAP}{\lambda} \left[1 - \exp(-\lambda t_1) - \lambda t_1 \exp(-\lambda t) \right] \text{ (REMS)} \quad (14a)$$

3.2 External Dose to the Whole Body from Gamma-Emitters in Cloud

The dose rate (D_r) to the whole body from external radiation assumes the person is surrounded by a semi-infinite cloud of concentration c .

The dose rate is then approximated by:

$$D_r = 0.279 \ c \ E_\gamma \quad (\text{rems/sec}) \quad (15)$$

Where:

c = concentration of released fission products in μc per cc of air.

E_γ = average gamma energy in Mev/disintegration.

Neglecting radioactive decay during cloud passage and assuming the time of emission is sufficiently long, so that edge effects can be ignored, the dose becomes:

$$D = 0.279 \ c \ t \ E_\gamma \quad (\text{rems}) \quad (16)$$

Where:

t = time of cloud passage in seconds.

3.3 Internal Dose from Inhalation of Radionuclides

Inhalation of fission product mixtures is assumed to take place in a sufficiently short time so that the dose received during buildup can be ignored. The dose delivered to the organ of interest can then be approximated by:

$$D = bct \langle f_{ak} \rangle \quad (\text{rems}) \quad (17)$$

Where:

b = breathing rate in cc/sec.

c = air concentration of fission products in $\mu\text{c/cc}$.

t = time of breathing contaminated air in sec.

$\langle f_{ak} \rangle$ = the effective dose per unit of radioactive material inhaled and is calculated by $\sum_i f_i (f_{ak})_i$. The units are rems per μc inhaled.

f_i = fractional amount of the i^{th} radionuclide.

$(f_{ak})_i$ = amount of i^{th} radionuclide reaching the organ of interest.

Calculated values of the effective dose for inhaled fission products are shown in Table F-6. Values are given for whole body, bone, and thyroid as the organ of interest, for release based on both the maximum and the worst conceivable accident (e.g. 100 percent meltdown of core).

Equation (17) is assumed to estimate the dose to both children and adults, since the breathing rate is roughly inversely proportional to the organ weight. This is particularly true for the case where the thyroid is the organ of interest. (6)

3.4 External Dose from Ground Deposited Radionuclides

The initial dose rate from fission products deposited on the ground at three feet above the ground is approximately:

$$D_F^0 = 1.05 \times 10^{-5} \text{ d} \quad (\text{rems/hr}) \quad (18)$$

TABLE F-6

CALCULATED VALUES OF f_{ak} PER CURIE RELEASED
(rems per uc inhaled)

<u>Release Basis</u>	<u>Organ of Interest</u>	<u>f_{ak} At Selected Decay Times In Days</u>			
		<u>0</u>	<u>1</u>	<u>10</u>	<u>30</u>
Worst Conceivable Accident	Whole Body	1.5×10^{-5}	5.1×10^{-5}	5.9×10^{-5}	6.1×10^{-5}
	Bone	6.0×10^{-5}	2.6×10^{-4}	4.7×10^{-4}	7.0×10^{-4}
	Thyroid	5.8×10^{-3}	1.7×10^{-2}	1.2×10^{-2}	3.8×10^{-3}
Maximum Credible Accident	Whole Body	7.4×10^{-8}	2.1×10^{-7}	1.4×10^{-7}	4.8×10^{-8}
	Bone	6.6×10^{-9}	2.6×10^{-8}	4.6×10^{-8}	7.0×10^{-8}
	Thyroid	4.3×10^{-5}	1.2×10^{-4}	7.7×10^{-5}	2.4×10^{-5}

Where:

1.05×10^{-5} = conversion factor (7) and has units of rads/hr per $\mu\text{c}/\text{m}^2$.

d = ground deposition in $\mu\text{c}/\text{m}^2$.

Assuming that the activity of the released fission products is proportional to $t^{-1.2}$ at any decay time, t , between one day and 365 days, the first year dose can be approximated by:

$$D = \int_0^T R(t) dt \quad (19a)$$

Where:

$R(t) = a t^{-1.2} d_0$ r/hr per $\mu\text{c}/\text{m}^2$.

$$a = \left(1.1 \times 10^{-5} \frac{\text{r/hr}}{\mu\text{c}/\text{m}^2} \right) \left(1.13 \frac{\text{rems}}{\text{r}} \right)$$

$$= 1.2 \times 10^{-5} \frac{\text{rem/hour}}{\mu\text{c}/\text{m}^2}$$

$T = 365 \text{ days} = 8.76 \times 10^3 \text{ hours}$

d_0 = ground deposition corrected to one day's decay (in $\mu\text{c}/\text{m}^2$).

Substitution of the above values in and integration of Equation 19(a) yields

$$D = 2.1 \times 10^{-5} d_0 \quad \text{rems} \quad (19)$$

3.5 Internal Dose from Ground Deposited Radionuclides

The mixture of released fission products deposited on vegetation and subsequently ingested is a much more complex problem than the inhalation problem, due to:

Radioactive decay over the time of ingestion.

Selective absorption of radionuclides on vegetation.

Wide variation in effective half-lives in the organs of interest.

Discrimination throughout man's food chains.

Because of these complexities, the dose from ingestion of mixed fission products was not evaluated. However, the dose from ingestion of Sr^{90} , I^{131} , and Cs^{137} were evaluated by use of equations (14) and (14a).

a. Consumption of Contaminated Leafy Vegetables

The whole body, bone, and thyroid were selected as the organs of interest. It was determined that the dose from consumption of leafy vegetables could be neglected when compared to consumption of one liter of milk per day, if the dairy cattle were grazing on pasture grass contaminated to similar levels. Therefore, only the milk path was considered.

b. Consumption of Contaminated Milk

Assumptions:

- 1) 0.1 of cow's intake of iodine, cesium, or strontium appears in its daily milk output.
- 2) Cow's daily intake is 4×10^4 g of pasture grass.
- 3) Cow's daily milk output is 10^4 g.
- 4) Quantity of vegetation above ground - $1,000 \text{ g/m}^2$.
- 5) Per cent of contaminate absorbed or adsorbed to edible part of vegetables.
 - a) Iodine - 100%
 - b) Strontium - 40%
 - c) Cesium - 20%

- 6) Growing season is 10 months, during which no weathering or decontamination takes place.
- 7) The total intake occurs during the first year. The intake which occurs as a result of plant uptake through the root system in succeeding years is ignored.

From these assumptions, the milk concentration can be estimated:

$$c_m = \frac{0.1 (4 \times 10^4)}{10^4} c_p \frac{\mu\text{c}}{\text{g}}$$

Where:

c_m = milk concentration in $\mu\text{c/g}$.

c_p = pasture grass concentration in $\mu\text{c/g}$.

$c_p = 1.0 \frac{d}{10^3} = 10^{-3} d$ for iodine.

$c_p = 0.4 \frac{d}{10^3} = 4 \times 10^{-4} d$ for strontium.

$c_p = 0.2 \frac{d}{10^3} = 2 \times 10^{-4} d$ for cesium.

d = ground deposition per unit area in $\mu\text{c/m}^2$.

The daily intake (P) was estimated on the basis of the following additional assumption:

- 8) Daily milk intake for both child and adult is assumed as 1,000 g/day.

Because of its long effective half-life in the bone, Sr^{90} will not deliver its total dose, as calculated by Equation (3), during man's lifetime. Therefore, additional assumptions for Sr^{90} estimates are:

- 9) The average age of the exposed populace is 33 years.

- 10) The life expectancy at age 33 is 35 years, or t in Equation (14) is $35 \times 365 = 1.28 \times 10^4$ days.

3.6 Dose Equations in Terms of Meteorological Exposure

a. External and Inhalation Dose

The concentration of air-borne contaminants following an accidental release is:

$$C = \left(\frac{Q}{t}\right) \left(\frac{E}{Q}\right) \quad (\mu\text{c}/\text{cc}) \quad (20)$$

From Equations (16), (17), and (20), the whole body dose is:

$$D_{TB} = [0.195 Q + 230 \sum_j Q_j \langle \kappa_{Fa} \rangle] \left(\frac{E}{Q}\right) \quad (\text{rems}) \quad (21)$$

Where:

Q = total quantity of fission products released.

Q_j = total quantity of j^{th} group of fission products released,
i.e., noble gases, halogens, etc.

$\left(\frac{E}{Q}\right)$ is as previously defined in Section 2.

The dose to a 2g thyroid can be estimated from Equation (17) and Equation (20) as:

$$\begin{aligned} D_{THY} &= 23(0.23)(64.5) Q \left(\frac{E}{Q}\right) \\ &= 341 Q \left(\frac{E}{Q}\right) \quad (\text{REMS}) \end{aligned} \quad (22)$$

(NOTE: The Q 's now refer to the quantity of I^{131} emitted.)

Equation (22) holds for both a 2g and 20g thyroid, since the breathing rate for a child with a 2g thyroid is approximately 0.1 that of the standard man. (6)

b. External Whole Body Dose from Residual Contamination

Substituting Equation (7) into Equation (19) from a preceding paragraph results in an estimate of the external whole-body, first-year dose from residual ground contamination.

$$D_{TB} = 4.44 \times 10^4 \text{ HQ} \left(\frac{E}{Q} \right) \quad (23)$$

Where:

$$H = \left(\frac{V}{U} \right) \left(\frac{\bar{U}_0}{\bar{U}_h} \right) \bar{U}_h$$

c. Dose from Ingestion of Cow's Milk

As previously noted, the milk concentration (c_m) is:

$$c_m = 0.4 c_p \text{ (}\mu\text{c/g)}$$

Where:

$$c_p = 10^{-3}d \text{ for iodine.}$$

$$c_p = 4 \times 10^{-4}d \text{ for strontium.}$$

$$c_p = 2 \times 10^{-4}d \text{ for cesium.}$$

and, the ground deposition (d) is:

$$d = \text{HQ} \left(\frac{E}{Q} \right) \times 10^6$$

Since $c_m = 0.4 c_p$,

then,

$$c_m = 400 \text{ HQ} \left(\frac{E}{Q} \right) \text{ for iodine } \frac{\mu\text{c}}{\text{g}} \quad (24a)$$

$$c_m = 160 \text{ HQ} \left(\frac{E}{Q} \right) \text{ for strontium } \frac{\mu\text{c}}{\text{g}} \quad (24b)$$

$$\text{and } c_m = 80 \text{ HQ} \left(\frac{E}{Q} \right) \text{ for cesium } \frac{\mu\text{c}}{\text{g}} \quad (24c)$$

Substitution of 10^3 c_m for P and appropriate values for the other parameters results in the following equations:

- 1) The total body dose from ingesting milk contaminated with Cs^{137} is:

$$D_{TB} = 1.04 \times 10^6 \text{ HQ} \left(\frac{E}{Q} \right) \quad (\text{rems}) \quad (25)$$

- 2) The bone dose from Sr^{90} in milk:

$$D_B = 3.34 \times 10^9 \text{ HQ} \left(\frac{E}{Q} \right) \quad (\text{rems}) \quad (26)$$

- 3) The dose to a 2g thyroid for I^{131} in milk:

$$D_{thy} = 8.94 \times 10^7 \text{ HQ} \left(\frac{E}{Q} \right) (I^{131}) \quad (\text{rems}) \quad (27a)$$

- 4) The dose to a 20g thyroid:

$$D_{thy} = 8.94 \times 10^6 \text{ HQ} \left(\frac{E}{Q} \right) (I^{131}) \quad (\text{rems}) \quad (27b)$$

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ERRATA

In Table F-1, Page F-1 there are typographical errors in the values listed for the concentrations at the maximum for the cases of the unstable and neutral atmospheres. In both cases, the values should be 9.5×10^{-6} instead of 9.5×10^{-4} .

APPENDIX G
MATERIALS1. INTRODUCTION

The purpose of this appendix is to discuss the use of new materials or the new use of "old" materials in the N Reactor to the extent that these usages have a significant bearing on reactor safety. The major new materials are Zircaloy-2 and titanium. Carbon steel is being used for primary loop piping, a critical and essentially new application.

2. ZIRCALOY-2

Zircaloy-2 was used as the material of construction for the process tubes, graphite coolant tubes, and fuel cladding. The nuclear safety aspects of these applications are discussed below.

2.1 Zircaloy-2 Process Tubes

The Zircaloy process tubes will contain the heat generating fuel elements and the pressurized water used for removing the heat energy.

As designed and fabricated, the process tubes will have ample strength for this service. However, the originally ductile process tubes will be subject to some degree of radiation hardening which will make the material less ductile. Further, the material is capable, under rather specific conditions, of absorbing hydrogen with consequent severe embrittlement and degradation of the material. Trace amounts of oxygen or water vapor appear to be capable of inhibiting the absorption of hydrogen by Zircaloy-2 in the range of operating conditions expected for the NPR.

The major potential hazard associated with use of Zircaloy-2 process tubes would be the propagation of a single-tube rupture

to adjacent tubes. Analysis of the maximum strains in adjacent process tubes, based upon data obtained from an elaborate mockup designed for the purpose, indicate that no rupture propagation will result from a maximum tube rupture incident any place in the reactor, even if the adjacent tube has been embrittled far beyond any reduction in ductility expected to occur.

An extensive monitoring and evaluation program is planned to provide assurance that 1) the condition of the process tubes in the reactor is known, and 2) that the condition of the process tubes can be properly related to the potential for brittle fracture. This program is discussed in Appendix H.

2.2 Fuel Cladding

The use of Zircaloy-2 for fuel cladding does not constitute a nuclear hazard beyond the probable occurrence of occasional fuel element failures. Such failures may involve release of a fraction of the uranium and fission product of the fuel into the primary coolant loop. No release to the environs would be expected, except via the bleed stream, which will be adequately controlled.

2.3 Graphite Cooling Tubes

The use of Zircaloy-2 coolant tubes has little impact on nuclear safety. Leaky tubes would introduce an operational problem requiring their replacement.

No method for inducing gross multiple failures of the graphite cooling tubes has been identified which would not involve an accident which of itself would be catastrophic. However, such failures could contribute to a flooding accident, or could remove

the heat sink provided by the graphite cooling system. This heat sink would contribute significantly to reducing the melt-down associated with complete loss of coolant to the process tubes.

3. 17-4 PH STEEL

17-4 PH Steel has been used in several places in the NPR. However, heat treatment at or above 1025°F has been generally required. The only potentially critical applications for which - due to the advanced state of procurement - it has not been feasible to use 1025°F minimum temperature heat treatment is in wear rings, shaft sleeves, or deflector discs in the graphite coolant recirculation pumps and dump condenser condensate pumps.

The low water temperature, the water quality control, and the removal of 10 mils of surface after the final heat treating will each reduce the probability of stress corrosion cracking. The location of parts made of 17-4 PH, H900 have been identified so that any failures can lead to planned replacement of like parts.

4. TITANIUM SHEATHING FOR CONTROL RODS

Titanium sheathing is used for the control and safety rods of the NPR. Extensive literature review has indicated that water-cooled titanium will perform satisfactorily in the reactor environment. The material was chosen primarily because of its high melting temperature, which will permit the control rod to continue to perform its function even after tube coolant loss had permitted the internal aluminum parts to melt.

The major uncertainties lie in the factors determining the life of the titanium sheath. However, progressive damage due to possible

unanticipated chemical reactions with the reactor gas would be indicated by surveillance programs and could, at worst, lead to eventual replacement.

5. CARBON STEEL

Much of the primary coolant system is fabricated from carbon steel. In making this selection, both the structural and chemical suitability of the material were considered. These items are discussed below.

5.1 Structural Considerations(1,2)

Two grades of carbon steel piping were specified as suitable for primary system piping. ASA B31.1, Section 1, Code-approved A-212, Grade B, Firebox quality plates fabricated into piping by electric-fusion welding according to an augmented ASTM A-155 Standard was specified for the larger piping. For smaller piping, e.g., the 2008 tube connectors, augmented ASTM A-106, Grade B seamless tubing was specified. The material required by the A-106 Specification is similar to A-212, Grade B, Firebox quality identified above. The specifications for both materials require special care in melting, forming, heat treatment, identification, and control. These certified pipes represent the highest quality carbon steel piping available from industry. The quality is further enhanced by NPR specifications.

The A-106 Grade B seamless piping has a minimum yield strength of 35,000 psi and a minimum tensile strength of 60,000 psi, with a maximum carbon content of 0.70 per cent. Pipe fabricated from A-212 Grade B, Firebox quality carbon steel, according to Specification A-155, has a minimum yield strength of 38,000 psi,

tensile strength of 70,000 psi, and a carbon content between 0.31 and 0.35 per cent.

The metallurgy of both the ASTM 100, Grade B seamless piping and the ASTM A-155 piping, fabricated from A-212 Grade B, Firebox quality steel, involves straightforward and well understood mill and fabricating practices, including:

- a. Fine grain melting practice, which involves "silicon killing" for deoxidation, and some small amount of treating with aluminum.
- b. Proper heating and forming.
- c. Normalizing the steel in either the hot forming or final stages of fabrication.

As a general rule, the above practices also produce a material with satisfactory notch ductility properties.

It should be noted that the term, "Firebox Quality," as applied in ASTM A-212 Grade B formulation, is a grade terminology that imposes rigid controls and close supervision of mill practices. The term denotes a material intended for application in pressure vessels required to withstand repeated thermal and mechanical shock. A-212, Grade B, Firebox Quality is superior to either flange grade material or Grade A material for this service.⁽³⁾

When compared to stainless steel, carbon steel exhibits significantly improved resistance to both transient thermal stresses and steady-state, thermally induced expansion stresses.⁽⁴⁾

Thus, it is concluded that carbon steel is a suitable choice for reactor primary coolant system material from a structural

standpoint. While some difficulties were experienced in the fabrication of piping to the exacting specifications, it appears that the quality of the installed system will conform to specification requirements, and the primary coolant system integrity is commensurate with its importance in the achievement of nuclear safety.

5.2 Water Chemistry Considerations

In the selection of carbon steel for primary coolant system material, attention was also focused on its chemical behavior.⁽⁶⁾ While carbon steel is not so resistant to uniform corrosion attack or some types of pitting attack as stainless steel, its resistance is acceptable. Further, carbon steel is superior to stainless steel in some aspects, e.g., it is not subject to chloride stress corrosion cracking.

The use of carbon steel, therefore, does not appear to have an adverse effect on nuclear safety by reason of its chemical behavior.

6. MATERIALS REVIEW

The exact material of construction of all components was not specified in purchase orders. Rather, service conditions were described and material specifications were included. The vendor could exercise considerable option in the choice of materials for applications not identified as critical. The materials and corresponding heat treatment of components in critical service in the plant are now being reviewed and cataloged. This review should uncover gross misapplications of materials, if any, which can then be rectified.

This cataloging of material applications should have the added advantage of permitting rapid identification of specific applications of any particular material which may, in the future, be found to be a source of potential component failure.

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APPENDIX H
REMAINING PROBLEMS AND R&D PROGRAMS

1. INTRODUCTION

The principal remaining technical problems, and research and development programs designed to provide solutions, are outlined below.

The remaining problems are associated with the evaluation of factors determining the service life of components such as graphite and Zircaloy, tubing, extending the lives of these components, refining and extending our knowledge of pertinent process parameters, and refining and extending our quantitative understanding of the integrated system.

Further, making hazards evaluations is a continuing process. Improvement modifications to equipment, revisions to operating procedures, and changes in process variables to effect optimization of operations require periodic hazards evaluations during the life of the plant.

2. ZIRCALOY-2 PROCESS TUBES

2.1 Irradiation Damage

The hazard associated with the use of Zircaloy-2 as a material for the high-pressure process tube stems primarily from the potential rate and degree of degradation of the integrity of the tube as a result of service conditions. Thus, development programs currently in progress have a three-fold objective: 1) to establish material property characteristic changes resulting from exposure to reactor service conditions; 2) to provide engineering limits

and criteria for the replacement of process tubes, either individually or collectively; and 3) to establish a monitoring program which will routinely determine the physical condition of the tubes, preferably non-destructively and in situ.

The materials testing program, initiated in part prior to the design of the reactor, includes measurements before, during and after irradiation of those material properties that may impose operating and service limits upon the pressure tubes, and the manner in which these properties may be affected by the operating environment. Pre-irradiation testing was undertaken to distinguish those material properties defining design and application limits; such as ultimate, impact, creep, and stress-rupture strength, ductility and total deformation to rupture, and corrosion resistance. In-reactor testing and post-irradiation measurements are being undertaken to determine the direction and magnitude of changes in these material properties as a result of such reactor operating environmental conditions as temperature and pressure cycles and gradients, neutron irradiation, and hydride formation and concentration.

An extensive testing program has been initiated to determine the burst strength, stress-rupture strength, and susceptibility to crack propagation of the process tube under conditions of elevated temperature and variable hydride concentration. Initially, these tests are made using samples of actual process tube under biaxial stress. These tests will provide the bases for comparison with similar tests made on samples of irradiated process

tubes. It is to be emphasized that a major portion of this work will be the determination of the effect of exposure to reactor environmental conditions on the crack propagation characteristics. The crack propagation tests will establish the range of temperature and stress under which crack propagation does or does not occur.

It is expected that, by periodically subjecting samples cut from one or two process tubes removed from the reactor to crack propagation tests, and comparing the data obtained to existing reactor operating conditions of stress and temperature, the rate of approach to limits of stress and/or total neutron dose will be monitored; and in this way multiple tube failures should be prevented.

The monitoring program will provide routine measurement of the condition of process tubes. This program will include in-situ dimensional measurements and visual examination and, as noted above, the periodic removal and examination of one or two process tubes per year.

It is expected that the materials testing program and the pre-irradiation portion of the engineering limits program will be sufficiently complete by reactor startup to provide a sound basis for initial operation of N Reactor. Experience in reactor operation gained at the Plutonium Recycle Test Reactor and that data obtained from post-irradiation examination of PRTR process tubes and KER process tubes (including the N-Reactor tubes in KER-3 and KER-4) will provide the basis for the initial criteria for NPR process tube removal for examination.

Samples of Zircaloy-2 prepared for tensile, charpy-impact, and other tests are currently being irradiated in the ETR. It is planned to keep samples irradiated to higher exposure values than are current in the NPR to provide pilot information on irradiation damage.

2.2 Chemical Damage

It must be recognized that there is the prospect of leakage of water into the graphite moderator and into the reactor gas system. Most of the water which reaches the silica gel driers in the gas circulation system will be removed. Water vapor that contacts the hot surface of the Zircaloy process tubes at any point where bare metal is exposed will react only to the extent of forming a thin protective zirconium oxide coating on the tube. The relatively small amount of hydrogen formed will react with and diffuse throughout the metal of the tube. Water vapor that contacts hot graphite reacts to produce primarily CO and H₂, but also lesser proportions of CO₂ and even CH₄. The H₂ from the reaction of water vapor and graphite that contacts the surface of the Zircaloy process tubes where bare metal is exposed will react with the metal relatively very rapidly. There will be essentially no reaction where the H₂ contacts the surface of the tubes where there is an unbroken zirconium oxide coating.

The reactions mentioned above present two primary problems: 1) excessive reaction of water vapor with the graphite could impair the structural integrity of the moderator stack, and 2) excessive hydriding of the Zircaloy process tubes could induce severe embrittlement of the tubes.

Inasmuch as it does not seem warranted to assume that there will never be a water leak into the reactor, it will be necessary to assure that the Zircaloy process tubes will always have an unbroken, protective oxide coating. It appears that the most appropriate way of assuring this is to assure that there is always a sufficient concentration of water vapor at the surface of the tube that any break in the oxide coating will promptly be repaired through reaction of the exposed metal surface with the water vapor before there can be a damaging amount of reaction with hydrogen. If there is a substantial water vapor concentration at the surface of the tubes, however, there may be a potential for an excessive rate of oxidation of the graphite. Experimental work indicates that the rate of reaction between graphite and water vapor at water vapor concentrations which are low but sufficient to protect the Zircaloy tubes is significantly reduced in relatively low concentrations of either CO or H₂ or both. Carbon monoxide at the concentration that will sufficiently reduce the rate of reaction between water vapor and the hottest graphite will decompose to deposit soot at other locations at lower temperature. Hydrogen is not as effective as CO in reducing the rate of reaction between water vapor and graphite. There would also be the undesirable aspect of using H₂ alone that the rate of hydriding zirconium is a function of the H₂ concentration. An optimum practicable condition appears to be the use of CO and H₂ together at approximately the same concentration. For a given effect on the rate of reaction between water vapor and graphite, the concentrations of CO and H₂

when present together are less than the concentration of either one alone. Also, in the presence of an equal concentration of H_2 , the CO does not deposit soot.

It appears, then, that a practicable scheme for protecting the Zircaloy tubes from excessive hydriding and the graphite moderator from excessive oxidation will require 1) being able to limit the water vapor, H_2 or CO concentration through use of silica gel driers, gas purging and/or a gas purification unit; and 2) being able to add water vapor, H_2 and CO to maintain an adequate concentration of each. Gas analysis instrumentation to permit control of this type of operation is being provided. It seems appropriate not to install a gas purification system for initial plant operation, but to operate on a feed and bleed basis. This can be done satisfactorily. The use of a gas purification system is primarily an economic matter. Both the engineering requirements and the economic justification for a purification system can be best determined from actual plant operation.

While investigations of the graphite-Zircaloy-atmosphere system will be continued and additional knowledge obtained before initial operation of the reactor, it is recognized that there is a limit to which actual reactor conditions can be simulated in experimental work. The performance of the reactor with respect to the considerations described above will be continuously monitored. Specifically, this will include the removal, examination, and destructive testing of process tubes, as mentioned earlier.

3. GRAPHITE

3.1 Irradiation Damage

It is known that reactor grade graphite is subject to distortion under neutron irradiation. At the temperatures of NPR operation, this distortion will be a contraction in physical dimensions, which will result in a settling of the top central region of the stack. The stack has been designed to accommodate a subsidence of 8 to 14 inches. Continuing through 1966, irradiation of graphite samples in test reactors (the GETR) will provide a basis for predicting graphite contraction rates in the NPR and the total graphite contraction to be experienced during the life of the reactor.

3.2 Chemical Damage

The primary chemical reactions of concern are oxidation and reaction with water vapor. These reactions are intimately connected with the service life of Zircaloy-2 process tubes and were discussed in section 2.2 of this appendix.

4. PHYSICS

Analyses have been performed to assist in the design and to assure that the reactor can be operated safely. These analyses will be refined and extended to more closely define optimum operating conditions and safe limits for operating parameters.

This will include buckling measurements using exponential piles, neutron spectrum studies for heterogenous cells, development of computer codes, and kinetic studies, including analog simulation studies. These studies will be performed for candidate fuel elements, various

control rod configurations, ball channel use, and for various situations, including the flooding accident.

5. THERMOHYDRAULICS

Heat transfer and hydraulic studies of the NPR process tube, fuel element systems will be continued and extended to a full-scale, electrically heated mockup of the downstream half of an NPR process tube. This mockup will generate heat with the cosine heat generation distribution that corresponds to a cosine flux distribution along the process tube.

Using actual and candidate fuel element geometries, the conditions leading to boiling burnout will be defined for a wide range of operating and emergency conditions, and will be used to establish safe operating limits. It is anticipated that these studies will be extended to include measurement of single-phase pressure drop and two-phase (water and steam) pressure drop in process tube fittings and connectors, and perhaps to include the effects of two-phase flow in the process tube.

6. SYSTEMS ANALYSIS

As component and subsystem studies yield additional results, studies of the integrated reactor and heat dissipation system complex will be extended and refined, with special emphasis on optimizing control of the heat dissipation system.

These system studies must be made. However, the breadth and completeness of the integrated analyses depend strongly upon the availability of an adequately sized analog simulation facility.

A project proposal, requesting authorization for such a facility, has been submitted to the Atomic Energy Commission. The requested facility would also include an activated control room mockup, which would be used for testing operating procedures and training personnel.

7. HAZARDS ANALYSES

Making hazards evaluations is a continuing process. As the details of operating procedures become defined, and as additional data become available from acceptance tests, startup and operational tests and from research and development programs, previous hazards evaluations must be reviewed and updated.

Further, changes in process parameters, operating procedures, or equipment will require specific hazards evaluations and general reviews of hazards periodically during the life of the plant.

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